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DESIGN OF TEST BLANKET SYSTEM FOR ITER MODULE TESTING

October 1997

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Test blanket systems to be installed in ITER for developing demo blankets have been investigated. One of the main engineering goals of ITER is to test tritium breeding blankets relevant to a power reactor. The test foreseen on modules include the demonstration of a breeding capability that would lead to tritium self-sufficiency in a reactor and extraction of a high grade heat suitable for electricity generation. To accomplish these goals, several ITER equatorial ports are available to test the test blanket systems, both in the basic performance phase (BPP) and the enhanced performance phase (EPP).

Test blanket systems for water-cooled and helium-cooled type DEMO blankets with ceramic breeders, developed in Japan have been designed. The design activities include the neutronics, thermal and hydraulic analyses, and mechanical configuration considering port sharing, cooling systems and tritium recovery systems, and test blanket system compatible with the current ITER design has been developed.

Keywords : ITER, Test Module, DEMO Blanket, Tritium, Breeding, Water-Cooled Blanket, Helium-Cooled Blanket, Cooling System, Tritium Recovery System

This report is based on the ITER Design Description Document (DDD).

+ Department of ITER Project

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ITER装荷試験用原型炉ブランケット
テスト・モジュール・システムの設計

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（1997年9月16日受理）

原型炉用ブランケットを対象としたITERでの工学試験用ブランケットシステムについて検討した。原型炉用ブランケットの試験はITERの主な工学性能目標のうちの1つである。テストモジュールにより，燃料自給のためのトリチウム増殖能力と発電用の熱回収機能の試験および実証を行う。ITERの基本性能段階（BPP）および拡張性能段階（EPP）において，これらの試験のために，幾つかのITER水平ポートが利用可能であり，ITER参加極に割り当てられる。水平ポートのうちの1つは，ITERのEPP用トリチウム増殖ブランケット試験用に割り当てられる。

原型炉用ブランケットとしては従来より日本において検討されてきた水冷却およびヘリウム冷却のセラミック増殖材ブランケットを取り上げ，これらのテストモジュールの核・熱設計，他極のテストモジュールとの共有を考慮した試験ポートへの設置概念検討，冷却系およびトリチウム回収系の設計を実施した。その結果，現ITER設計と整合のとれたテストモジュール及び補機系の設計が提示された。

本報告はITER設計記述書（Design Description Document：DDD）の一部として作成したものに補筆を行ったものである。
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1. Introduction

One of the main engineering performance goals of ITER is to test and validate design concepts of tritium breeding blankets relevant to a power producing reactor. The tests foreseen on modules include the demonstration of a breeding capability that would lead to tritium self-sufficiency in a reactor and the extraction of high-grade heat suitable for electricity generation. To accomplish these goals, a number of the ITER horizontal ports are available to test the test blanket systems, both in the Basic Performance Phase (BPP) and the Enhanced Performance Phase (EPP). One of the ports will be dedicated to the testing of the tritium breeding blanket which will be used in the EPP of ITER.

The blanket test program will investigate various design concepts of tritium breeding blankets proposed by the Parties, namely Japan, the European Union, the Russian Federation and the USA. This document addresses the design description of the test blanket system(s) including interfaces with the ITER device.

Breeding blanket research and development is recognized as one of important areas for realizing an energy-producing fusion reactor, and its basic approach was determined in the Japanese Fusion Program in 1994 as "Testing breeding blanket module in a fusion experimental reactor is recognized to be essential to obtain engineering data of production and recovery of tritium as well as materials irradiation data. The research and development in this area needs to be promoted."

Breeding blanket research and development in Japan is being conducted at present placing ceramic breeder/water coolant/ferritic steel structure and ceramic breeder/helium coolant/ferritic steel structure as the main concepts of the blanket. A long-term program of breeding blanket research and development has been proposed by JAERI, where it is proposed to fully develop the two concepts above. This proposal is under the review process by the Fusion Council. The following document is described based on the proposal by JAERI, and from the above situation, it should be noted that it is a tentative proposal from the Japanese Fusion Program.

Two DEMO-relevant test blanket systems are proposed here: a water-cooled, ceramic breeder, ferritic steel structure blanket and a helium-cooled, ceramic breeder, advanced structural material blanket. Though the advanced structural materials including intermetallic compounds such as TiAl and SiC/SiC composites are considered for the helium-cooled blanket, the helium-cooled blanket may start with the ferritic steel structure because of immaturity of those materials development. Each of blanket concepts may occupy a half or all of a horizontal test port. The testing of the test articles may begin prior to the first D-T plasmas because valuable data may be obtained concerning high heat flux removal capability and durability against electromagnetic loads. The tests will continue to accumulate fluence through the BPP and EPP phases.

The purpose of the tests is to validate the design principles and the operational feasibility for the DEMO blanket system. This includes the basic subsystems of the test module (first wall / breeding blanket), shield, heat transport system and tritium removal system. In addition, the basic properties and operating characteristics of the system's materials will be validated. To assess those qualities and characteristics, the test blanket systems are to be exposed directly to the ITER plasma for relatively long, continuous operating periods.

These test blanket modules will be installed among ITER blanket modules, thus they must meet applicable ITER blanket requirements, including heat removal, shielding protection for the vacuum vessel welds and toroidal field magnets, and reduction of neutron streaming.

Breeding and recovery of tritium are important goals of the test program. Lithium ceramic compounds will be used as the breeder materials to be investigated. Subsystems to recover the bred tritium will be demonstrated to separate and remove the tritium from the purge stream.

Generation and extraction of high temperature coolant of water and helium will demonstrate the suitability of fusion for commercial power generation. These high temperature coolants will transfer the rejected heat to the ITER facility water coolant system.

The test blanket modules will be designed to: (1) be robust against the thermal and mechanical loads produced on them , (2) conform to the same safety requirements as other in-vessel components, and (3) have a minimum impact on reactor operation and availability due to any unscheduled test module removal.

The test blanket systems are to be installed and maintained through the horizontal test ports. Standard ITER remote handling equipment and procedures will be used to the maximum extent. All test blanket systems plumbing shall be contained within the vacuum chamber horizontal port extensions and pass through the horizontal port or shielding doors. Maintenance rails and other remote handling equipment are to be provided for use within the horizontal ports. Space shall be provided in the region outside the biological shield, near the ports, especially for tritium handling equipment and/or cooling equipment and for storage of test equipment during maintenance actions. Transport from the horizontal ports to the hot cells are to be provided as well as facilities in the hot cells for storage, maintenance, testing, refurbishing, and dismantling the test articles.

A general test port configuration is outlined in Chapter 2.
Design features of the water-cooled and helium-cooled test modules with their major characteristics are described in Chapter 3.
Designs of water and helium cooling systems and tritium recovery system are shown in Chapter 4 and Chapter 5, respectively, in consideration of their interfaces with ITER systems.
In Chapter 6, a consideration on the requirement and basic philosophy of test module remote handling system is briefly shown.
Finally, the summary of this design work is given in Chapter 7.

2. Test Port Configuration

A water-cooled attachment frame for test modules installation into a test port will be provided with coolant water at approximately 4 MPa and 140 °C with a maximum temperature rise of 50 °C under the maximum quasi steady state heat flux. It will be made with the same materials as the main blanket/shield structure. Each of the water-cooled and helium-cooled test module utilizes a blanket box structure attached to a shielding structure which might be then bolted to the attachment frame structure via a flange to account for the module / shielding weight and ITER disruption loads. The test module shielding structure is a separate component from the test module, which is water-cooled possibly by the same supply as the ITER vacuum vessel.

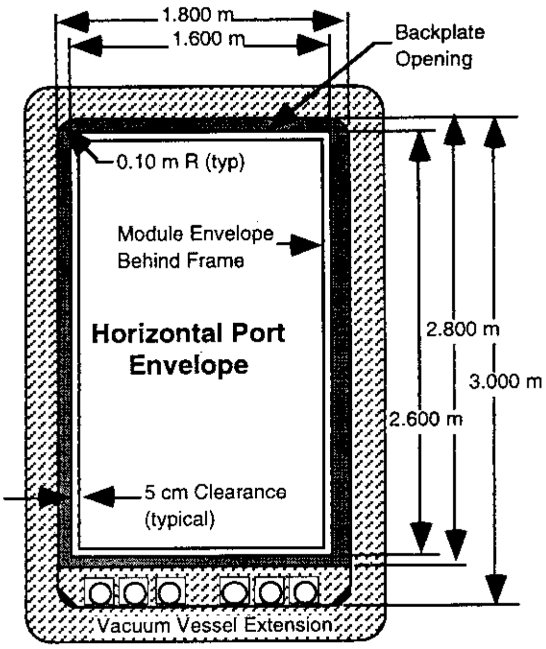


Fig. 2-1 Horizontal port dimensions

The physical size of the test module is determined by the constraints of the ITER horizontal port. The governing dimensions for the vacuum vessel extension and the backplate opening are given in Fig. 2-1. Provisions for ITER limiter and baffle cooling pipes on the floor reduce the available height by 20 cm within the vacuum vessel extension region (1.8 m x 2.8 m). Allowing some space for the attachment frame yields an opening size for the backplate of 1.6 m x 2.6 m. The thickness of the attachment frame is preliminary considered to be 200mm for satisfying the shielding performance together with the shielding structure at the back to the test module and to support the test module. Within the vacuum vessel extension, the shielding should fill the envelope as much as possible. The maximum size of the shield in this region will be the vacuum vessel extension envelope less a some, e.g. 5-cm, clearance all around for differential movement between the vacuum vessel and the backplate. The radial depth of the blanket is determined by the blanket design parameters. The first wall surface of the test blanket module (and the frame) may be recessed from the adjacent ITER blanket or limiter modules. The test blanket module would occupy the full width of the available test port, but perhaps only one-half of the poloidal height or vice versa (half width and full height). This would allow sharing the test port of two test modules with different configurations and geometries.

The static and the dynamic loads generated by any module located within the horizontal port will be transmitted through a mounting system, which may be a series of teeth and bolts surrounding the perimeter of the 2.6 m high by 1.6 m wide opening in the backplate or a fixing mechanism to the wall of the vacuum vessel extension. The test blanket subsystem would provide a mating flange mounting system around the perimeter of the test module to transmit the internal loads to the backplate or the vacuum vessel extension. The temperature of the flange shall be designed to be compatible with the nominal operating temperature of the backplate or the vacuum vessel of approximately 200 °C to reduce thermal stresses in the attachment mechanism. Access to the mounting bolts will be provided for the remote handling equipment. Shielding will be provided behind the test blanket module to meet the shielding requirements for the vacuum vessel and the magnets.

Figure 2-2 presents the overall scheme for the test blanket system installation. Test blanket modules are installed into the attachment frame. A complete test blanket attachment frame and shielding structure are mounted onto the backplate, or the vacuum vessel extension, which will carry all the static and dynamic loads and provide the proper dimensional control for alignment of the test module. Immediately behind the test blanket module will be shielding. Since the blanket is relatively neutron transparent, the shielding will be extended to provide the maximum shielding to the local vacuum vessel walls. The primary vacuum will be sealed with a standard vacuum vessel closure plate at the end of the vacuum vessel extension. The plumbing pipes associated with the test module, shielding, and attachment frame will either penetrate the vacuum vessel closure plate or will penetrate the side wall of the vacuum vessel extension wall. If the former approach is used, it will allow a complete assembly to be installed as a pretested unit without welding within the vacuum enclosure. If the latter approach is used, the vacuum vessel closure plate will have no penetrations which will allow easy access to the plumbing and the back of the test module.
The primary cooling system for the test blanket module will be separate and distinct from the ITER primary cooling system because of the different cooling media and the system temperatures and pressures. In case of a port shared by test modules of different Parties, independent coolant line for each test module would be used to obtain accurate test results avoiding the influence of their different test conditions. with each other.

The ITER secondary cooling system will be used as the secondary cooling system for test blanket modules. In case of the water-cooled test module, the coolant purification and detritiation system in the ITER plant could be used for the test module primary coolant through pressure and temperature adjusting equipment.

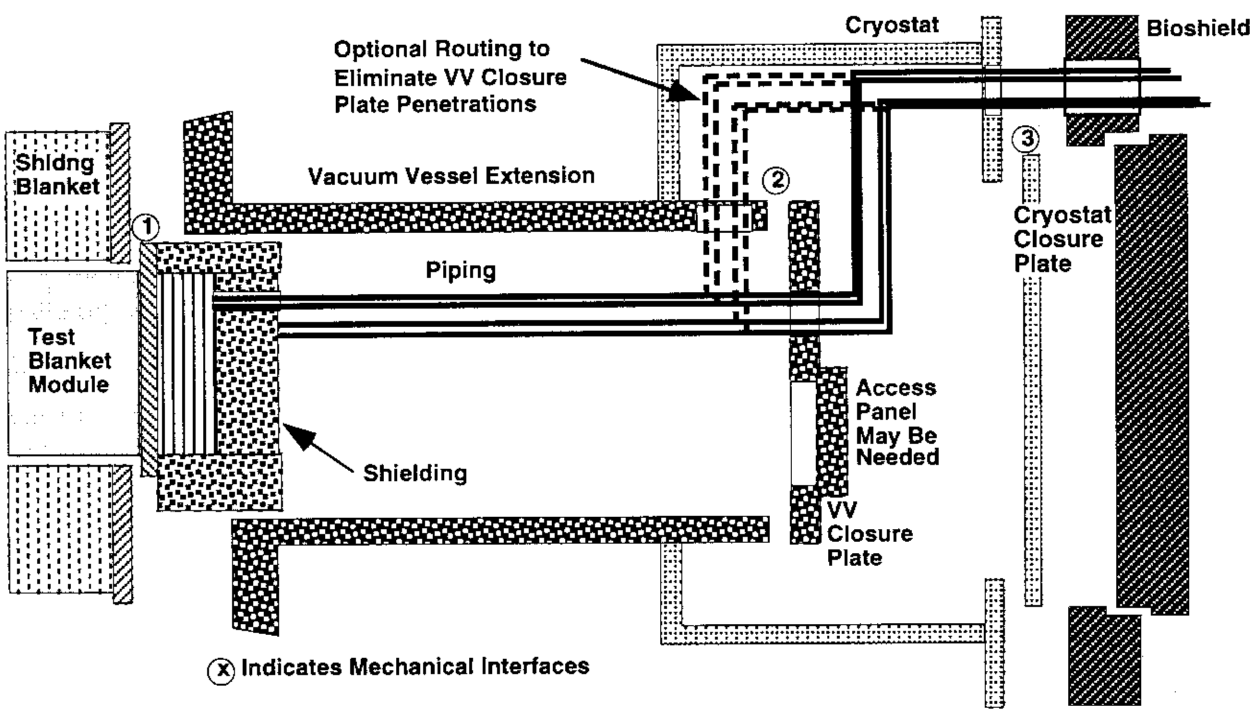


Fig. 2-2 Schematic of test blanket system elements in horizontal port

3. Test Module Design Description

3.1 Mechanical Configuration and Design Parameters of Test Module

3.1.1 Water-Cooled Test Module

A representative design of fusion DEMO reactor recently studied in Japan is SSTR (Steady State Tokamak Rector) [1]. A test module concept is based on the blanket designed for the SSTR[2]. A cross-sectional view of the test module is shown in Fig. 3.1.1-1. The concept is based on layered and BOT (Breeder Out-of-Tube) and represents front replaceable and back permanent regions proposed for SSTR. These regions are separated by structure wall. In the front region, ceramic breeder and neutron multiplier are packed in alternative layered fashion. On the other hand, only ceramic breeder is packed in the back region.

The first candidate for the breeder material is Li_2O with alternatives of other ternary ceramic materials including Li_2TiO_3. Beryllium is used as the neutron multiplier. Both of the ceramic breeder and Be are formed in small spherical pebbles. Rows of circular coolant tubes might be replaced by flat cooling panels with built-in cooling channels depending on future thermo-mechanical and structural evaluation.

Preliminary design parameters of the water-cooled test module are summarized in Table 3.1.1-1. A reduced-activation ferritic steel, F82H, is adopted as the structural material, which has been developed in Japan, and its material data including irradiation effects are being accumulated. Inlet/outlet temperatures of the water coolant are 280/320 °C corresponding to those of conventional PWR plant.

Table 3.1.1-1 Preliminary design parameters of water-cooled test module

Blanket type	Layered and BOT (Breeder Out-of-Tube)
First wall area	~0.93 m^2 (0.44 m^W x 2.12 m^H)
Thickness	0.6 m
Neutron wall load	1.2 MW/m^2
Average surface heat load	0.25 MW/m^2
Structural material	F82H
Coolant	Pressurized water
Inlet/Outlet temperature	280/320 °C
Pressure	15 MPa
Flow rate	~7.6 kg/s (27.3 t/h)
Breeder material	Li_2O sphere (<1 mm dia.)
^6Li enrichment	50 %
Operating temperature	450-750 °C
Multiplier material	Be sphere (<1 mm dia.)
Local tritium breeding ratio	1.2
Heat deposition	1.68 MW

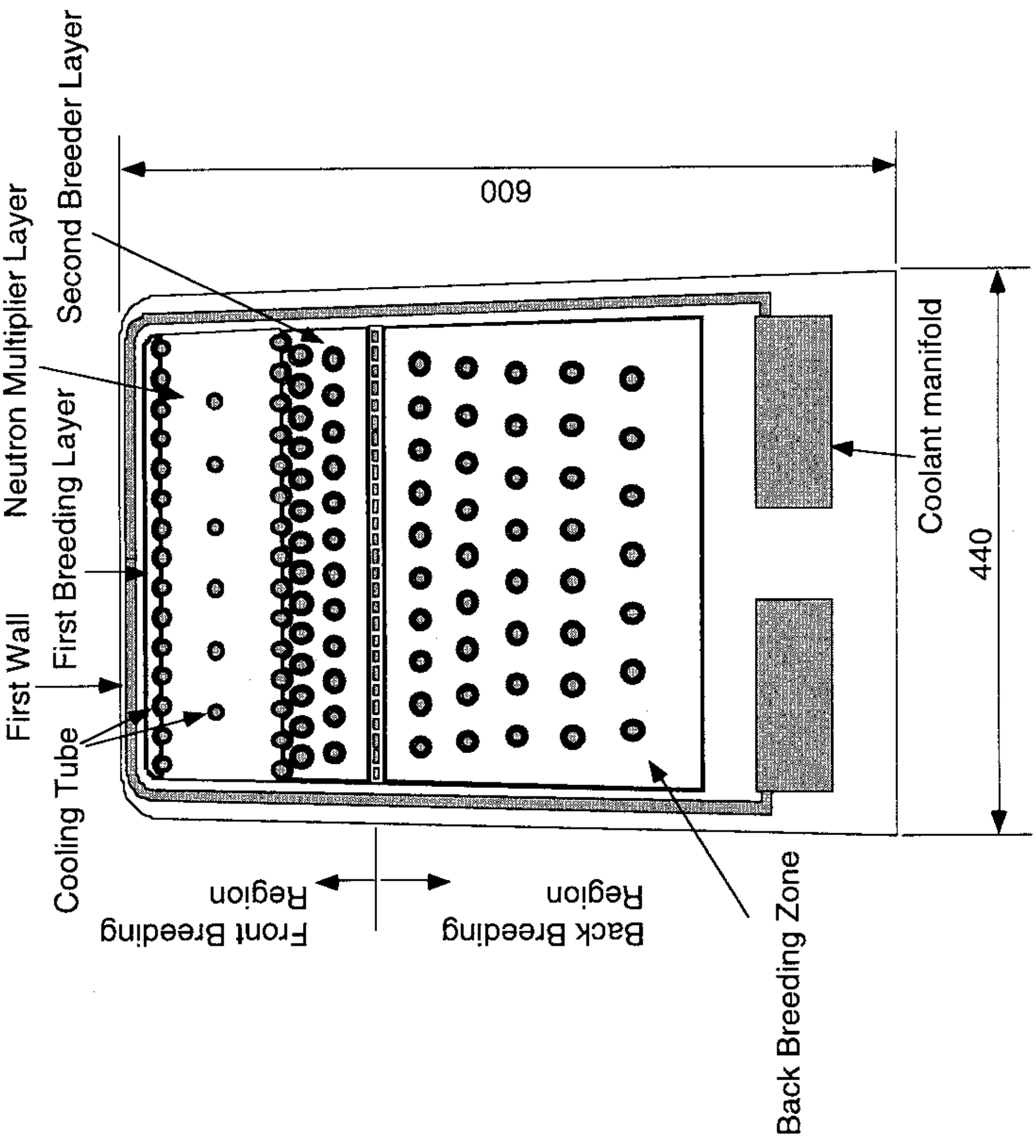


Fig. 3.1.1-1 Cross-sectional view of Water-cooled test module

The test module will be designed as so-called 'act-alike' module. Therefore, first wall dimensions and spacing of cooling tubes/panels will be modified for structures and breeder/multiplier materials to experience about the same level of stresses and prototypical temperature distributions as those in the DEMO blanket.

The test module is currently planned to be installed in a half space of the test port opening, i.e. a half width by full height for the water-cooled test module. Thus, two test modules, perhaps with one of other Party's, are installed in one test port with a common low temperature water-cooled frame structure. A schematic of a test module installation into the test port is illustrated in Fig. 3.1.1-2. Thermal power of this water-cooled test module roughly estimated is 1.68 MW including average surface heat load to the first wall of 0.25 MW/m^2.

Rough estimation on the weight of the water-cooled test module is summarized in Table 3.1.1-2. Total weight is about 1.9 ton.

Table 3.1.1-2 Weight of water-cooled test module

Material	Density (kg/m^3)	Volume (m^3)	Weight (kg)
F82H	7890	~0.18	1397
Li_2O	1966	~0.13	255
Be	1850	~0.05	99
Insulator*	3900	~0.01	59
Coolant	725	~0.08	60
Total	-	0.56	1870

* Assumed Al_2O_3

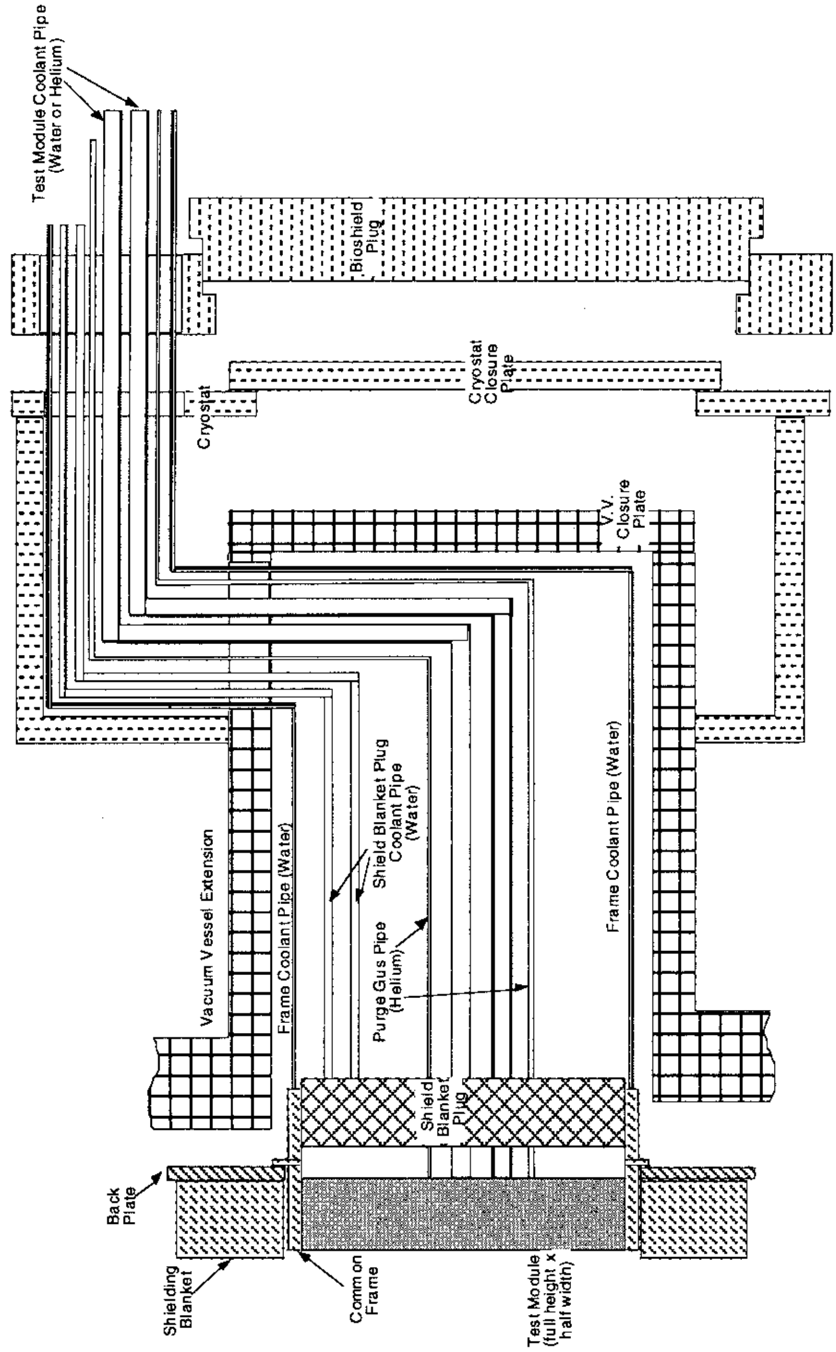


Fig. 3.1.1-2 Schematic of water-cooled test blanket installation into test port

3.1.2 Helium-Cooled Test Module

For a helium-cooled test module, a similar concept to the water-cooled test module has been pursued as shown in Fig. 3.1.2-1. As heat removal capacity of helium is less than water and also due to higher coolant temperature, more cooling paths are required than in the water-cooled test module.

Preliminary design parameters of helium-cooled test module is summarized in Table 3.1.2-1. Materials and their shapes of breeder and neutron multiplier are the same as those for the water-cooled test module. Also the same structural material, F82H, is used because the development of this material, among other advanced materials, is expected to meet the time scale for the fabrication of the test module and its testing in ITER. Taking high temperature durability of F82H into account, inlet/outlet temperatures of the helium coolant employed for the test module are 360/480 °C which are somewhat lower than those proposed for DEMO blankets to realize attractive helium coolant features. However, engineering data meaningful and possible to assess the DEMO blanket performance can be obtained even with these conditions. Advanced structural materials, e.g. refractory metal alloys, TiAl intermetallic compound, and SiC/SiC composite might be applied with elevated helium temperature depending on their materials development.

Table 3.1.2-1 Preliminary design parameters of helium-cooled test module

Blanket type	Layered and BOT(Breeder Out-of-Tube)
First wall area	~1.05 m^2 (1.12 m^W x 0.94 m^H)
Thickness	0.6 m
Neutron wall load	1.2 MW/m^2
Average surface heat load	0.25 MW/m^2
Structural material	F82H
Coolant	Helium
Inlet/Outlet temperature	360/480 °C
Pressure	8.5 MPa
Flow rate	~3.1 kg/s (11.0 t/h)
Breeder material	Li_2O sphere (<1 mm dia.)
^6Li enrichment	50 %
Operating temperature	500-750 °C
Multiplier material	Be sphere (<1 mm dia.)
Local tritium breeding ratio	1.3
Heat deposition	1.90 MW

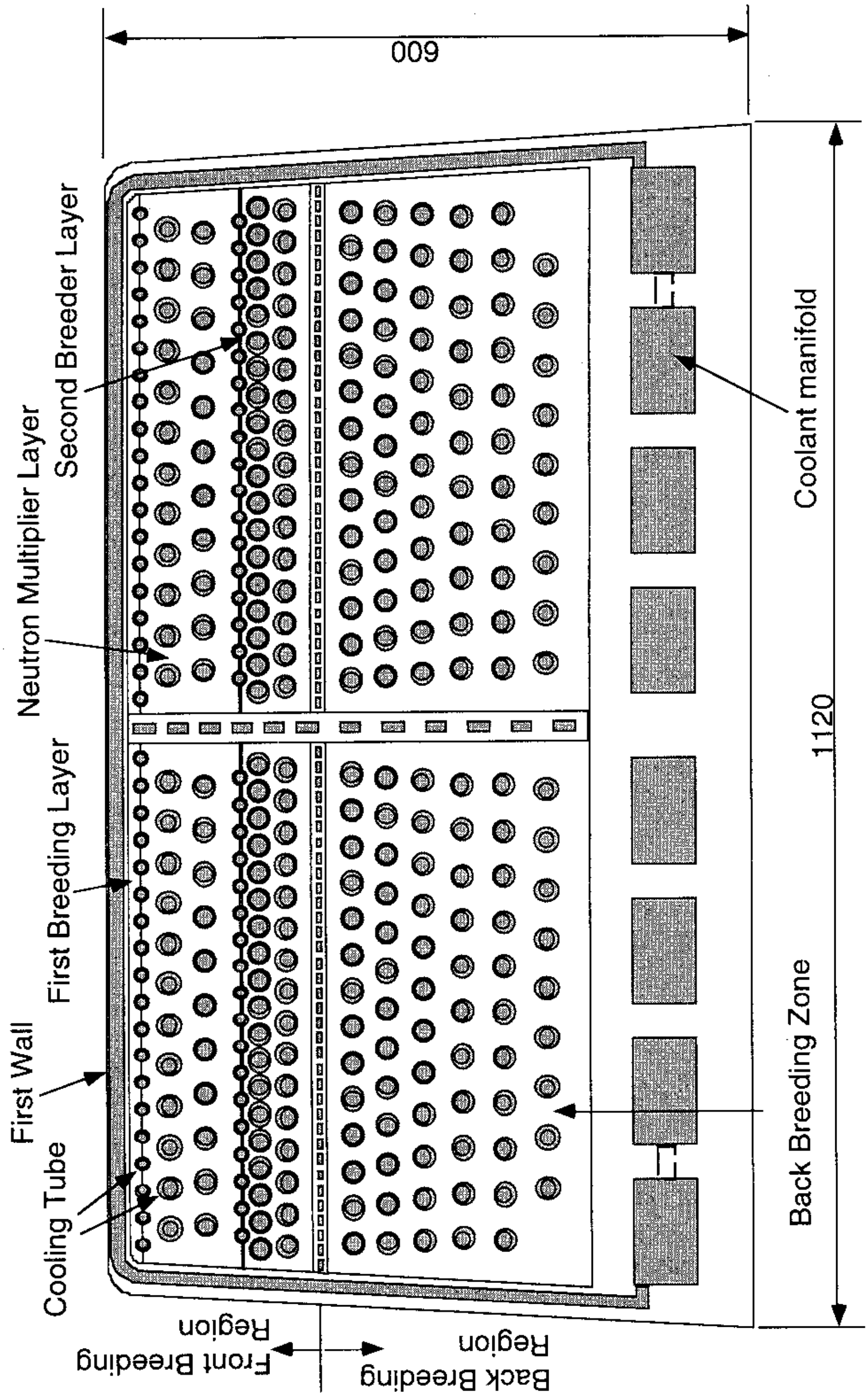


Fig. 3.1.2-1 Cross-sectional view of Helium-cooled test module

The 'act-alike' test module design and its installation in a half space of the port opening are the same as for the water-cooled test module though the space of a half height by full width of the port area will be occupied by this helium-cooled test module. A schematic of the helium-cooled test module installation is illustrated in Fig. 3.1.2-2. Thermal power of this helium-cooled test module roughly estimated is 1.90 MW including average surface heat load to the first wall of 0.25 MW/m^2.

Rough estimation on the weight of the helium-cooled test module is also summarized in Table 3.1.2-2. Total weight is about 2 ton.

Table 3.1.2-2 Weight of helium-cooled test module

Material	Density (kg/m^3)	Volume (m^3)	Weight (kg)
F82H	7890	~0.21	1674
Li_2O	1966	~0.14	271
Be	1850	~0.05	100
Coolant	5.765	~0.12	0.70
Total	-	0.63	2046

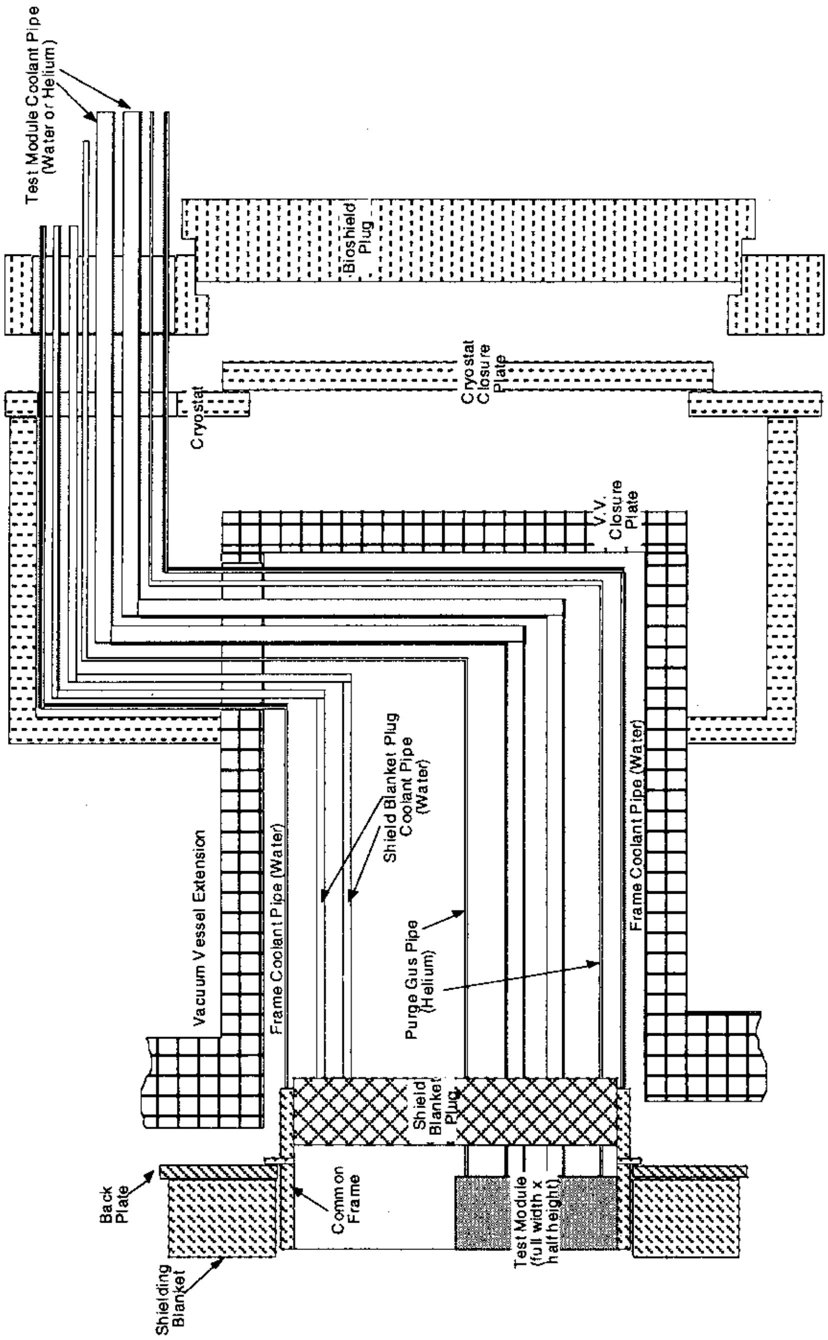


Fig. 3.1.2-2 Schematic of helium-cooled test blanket installation into test port

3.2 Nuclear Design

3.2.1 Calculation Method and Models

One dimensional neutronics analyses for the water-cooled and the helium-cooled test modules have been performed. Calculation models for evaluating TBR (tritium breeding ratio), nuclear heating rate distribution and shielding performance are shown in Figs. 3.2.1-1 and 3.2.1-2. Calculation code and data libraries used are as follows:

Calculation Code	ANISN/APPLE-3
Parameter	P_5 - S_8
Group Constant	FUSION-40 (based on JENDLE-3) Neutron : 42 groups, Gamma ray : 21 groups
Kerma Factor	F40KRMA

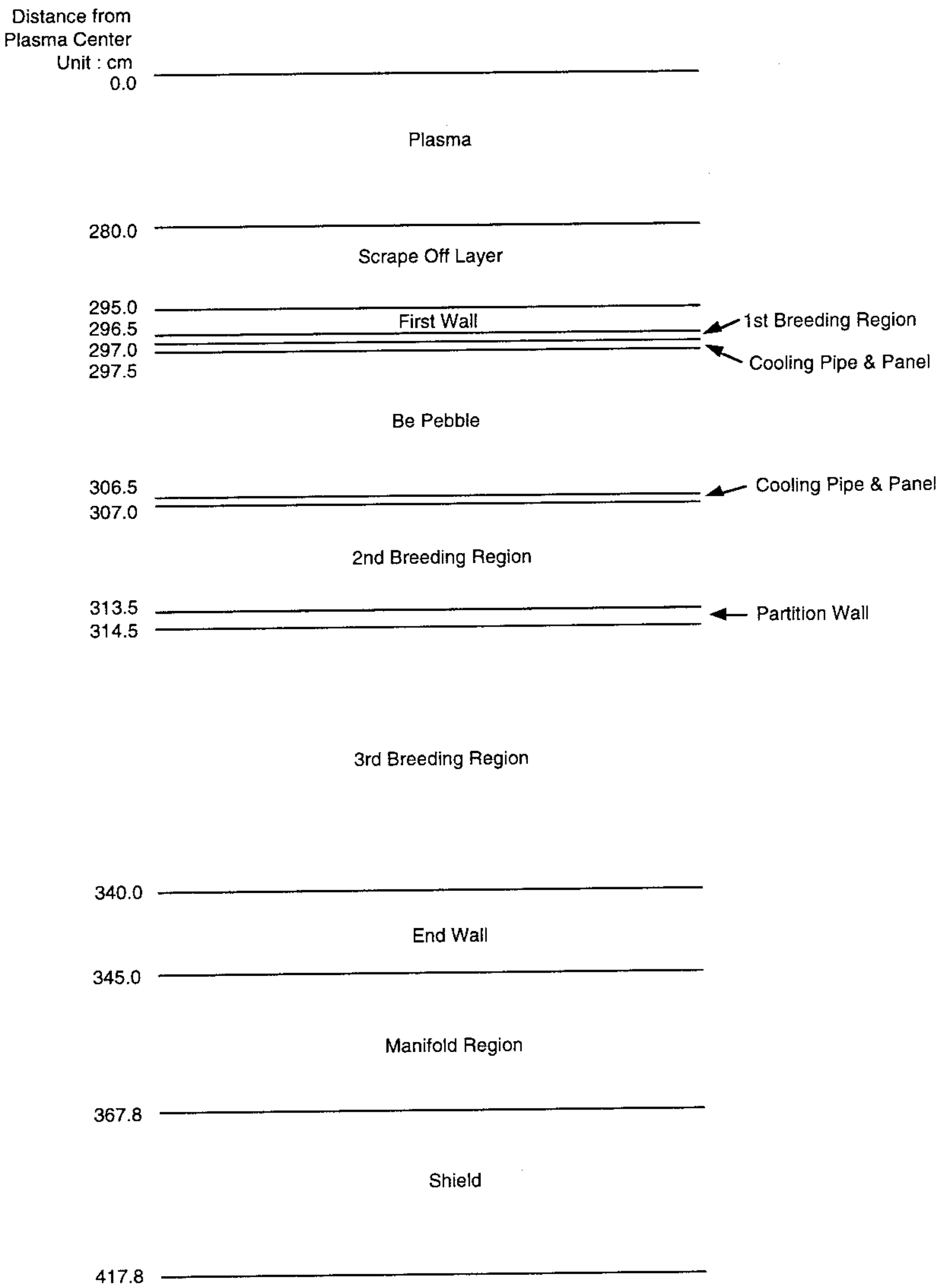


Fig. 3.2.1-1 Calculation model for TBR and nuclear heating rate distribution

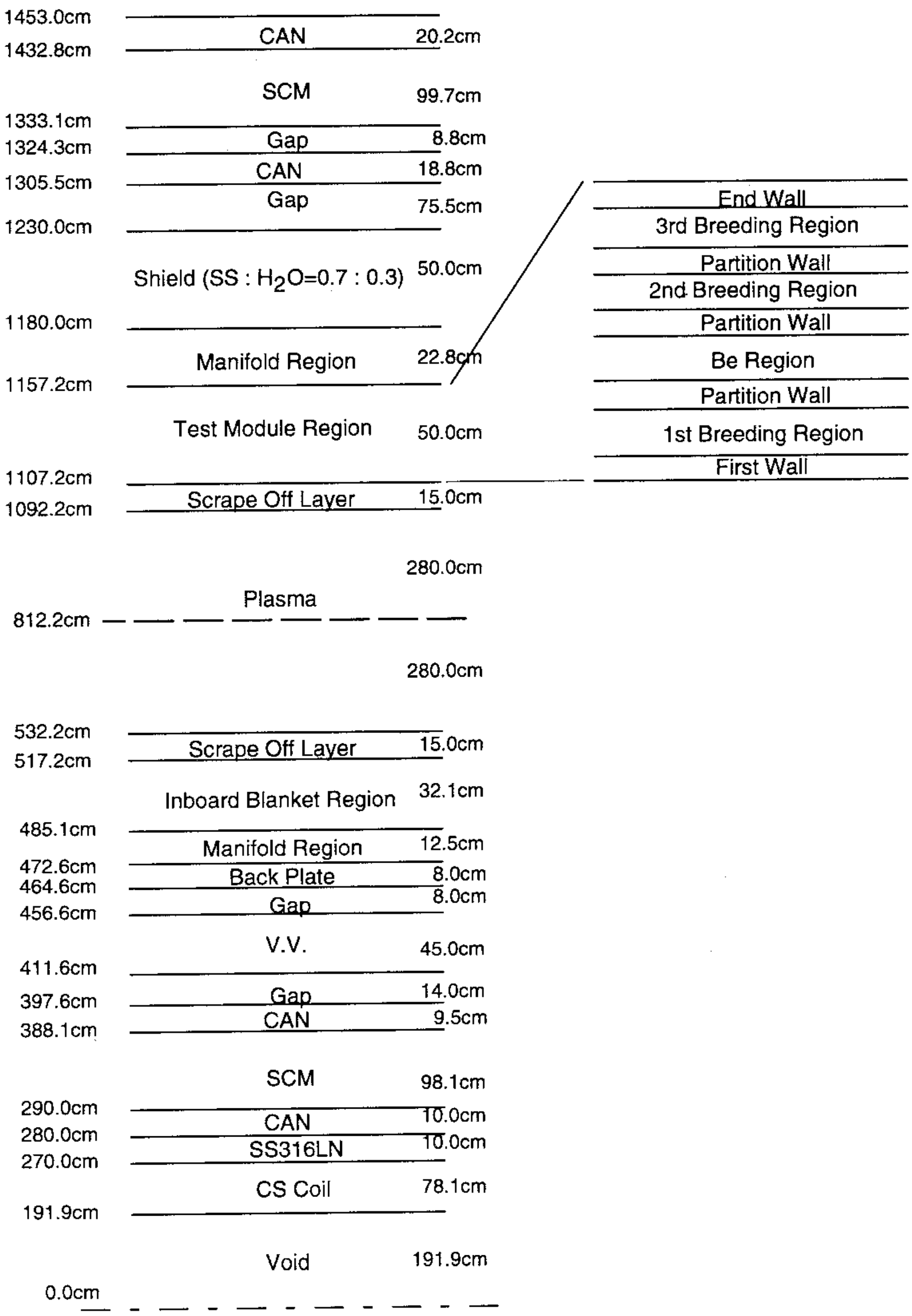


Fig. 3.2.1-2 Calculation model for shielding performance

3.2.2 Water-Cooled Test Module

Based on the calculation results, local TBR of 1.2 is expected for this test module taking for some reduction due to stiffening rib etc. into account. Figure 3.2.2-1 shows nuclear heating rate distribution in the water-cooled test module. With this distribution, thermal design for the internal configuration of the test module will be performed.

Shielding performance analysis is performed with parameter of thickness of the additional back shield provided at the back of the test module. This additional back shield is cooled with low temperature water coolant similar to that of the vacuum vessel. Table 3.2.2-1 and Fig. 3.2.2-2 indicate the shielding performance of the water-cooled test module and additional back shield comparing to the shielding performance of the ITER shielding blanket. Assuming the composition of the additional back shield as SS/water = 0.7/0.3, the back shield thickness of about 50 cm will be required to have the same shielding performance as the ITER shielding blanket. This calculation model assumes that the TF coil locates behind the test module, which is different from the real situation. The required thickness of the back shield could be reduced by more detail calculation taking the port configuration and neutron streaming through the port wall into account.

Table 3.2.2-1 Nuclear responses in TF coils with water-cooled test module

Neutron fluence: 1 MWa/m^2

Responses	GDRD requirement	ITER shielding blanket	Shielding behind test module			
			10 cm thick	30 cm thick	50 cm thick	100 cm thick
Nuclear heating in winding pack (W/cm^3)	**1×10^{-3}**	2.4×10^{-5}	5.6×10^{-3}	1.8×10^{-4}	1.0×10^{-5}	2.2×10^{-6}
End-of-life copper damage (dpa)	**6×10^{-3}**	1.2×10^{-5}	2.5×10^{-3}	9.5×10^{-5}	4.7×10^{-6}	1.1×10^{-6}
Peak organic absorbed dose (rad)	**1×10^{9}**	2.1×10^{7}	3.1×10^{9}	1.1×10^{8}	5.7×10^{6}	1.3×10^{6}
Fast neutron fluence (n/cm^2)	**1×10^{19}**	2.0×10^{16}	3.8×10^{18}	1.4×10^{17}	6.7×10^{15}	1.7×10^{15}

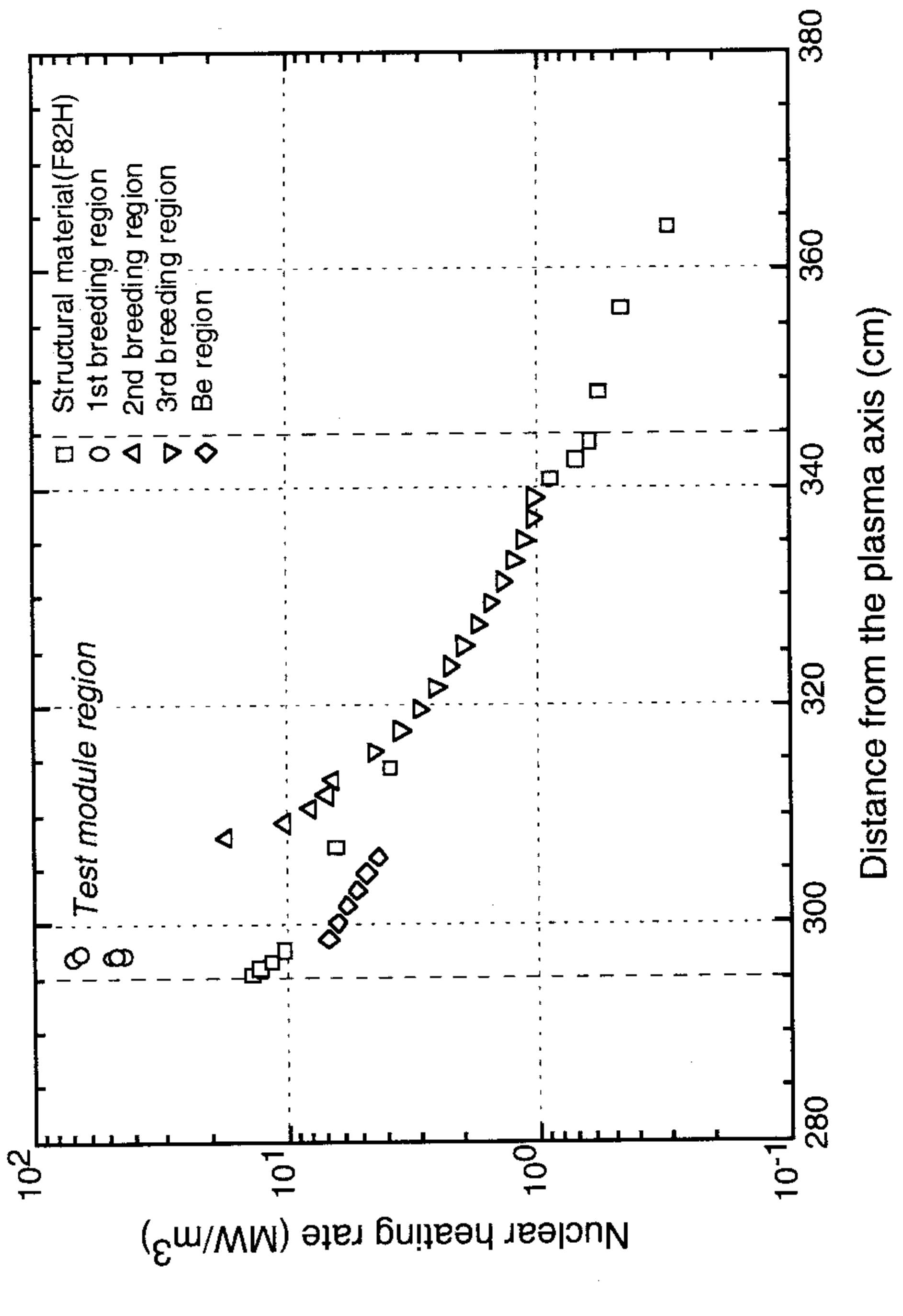


Fig. 3.2.2-1 Nuclear heating rate distribution in water-cooled test module

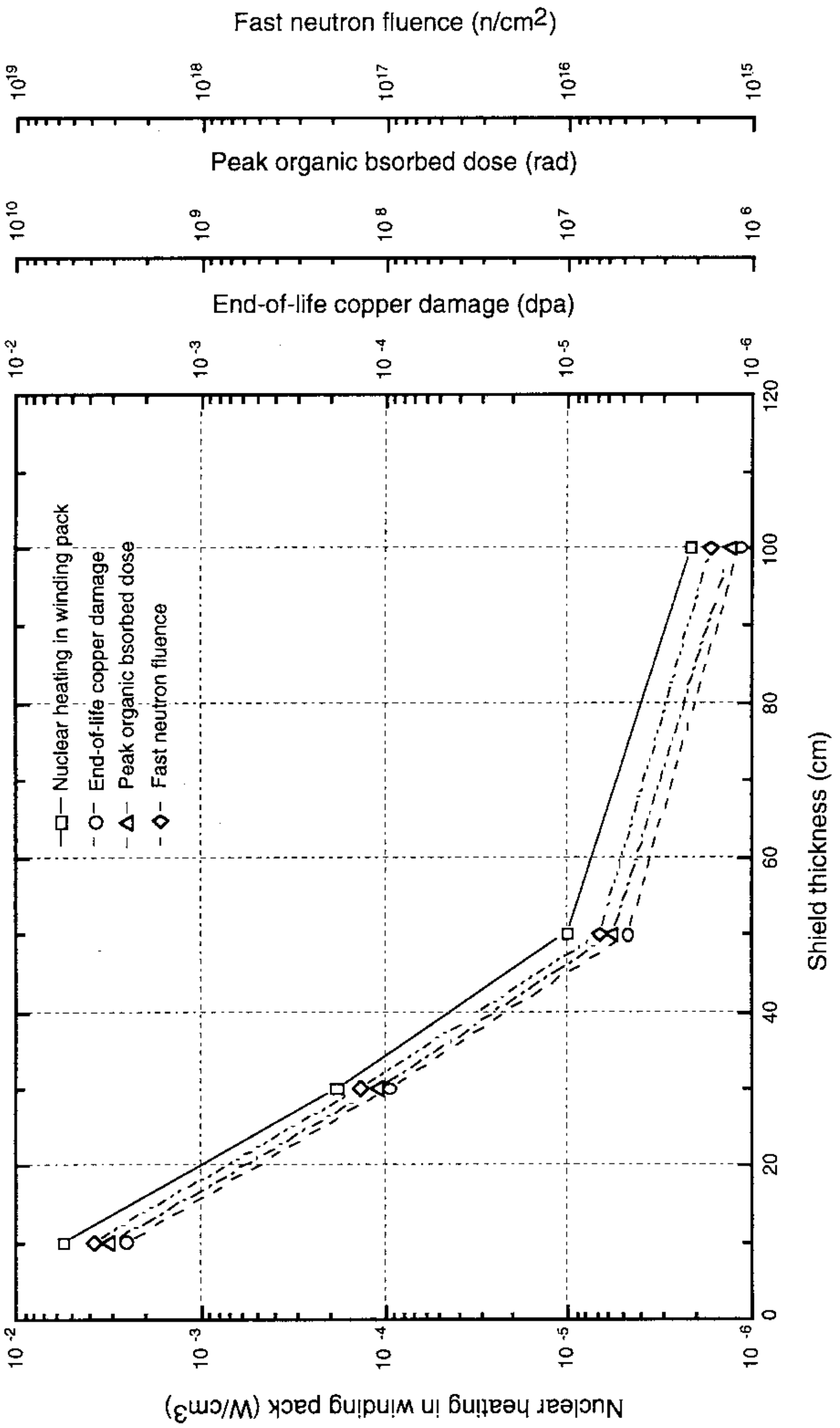


Fig. 3.2.2-2 Nuclear responce in the TF coils vs shield thickness
(Water-cooled test module, Fluence=1MWa/m^2)

3.2.3 Helium-Cooled Test Module

Local TBR of 1.3 is expected for this helium-cooled blanket based on the neutronics calculation taking into account the effect of stiffening rib on reducing breeder coverage and thus reducing breeding performance. The local TBR is slightly higher than that of the water-cooled test module because of lower attenuation and absorption of neutrons by He than by water. Nuclear heating rate distribution obtained as shown in Fig. 3.2.3-1 will be used as input for thermal design of this test module.

Similar shielding performance analysis to that of the water-cooled test module results in the required additional shield thickness of about 50 cm as shown in Fig. 3.2.3-2 and Table 3.2.3-1. Further detail calculation for this requirement is needed as mentioned above.

Table 3.2.3-1 Nuclear responses in TF coils with helium-cooled test module

Neutron fluence: 1 MWa/m^2

Responses	GDRD requirement	ITER shielding blanket	Shielding behind test module			
			10 cm thick	30 cm thick	50 cm thick	100 cm thick
Nuclear heating in winding pack (W/cm^3)	**1×10^{-3}**	2.4×10^{-5}	1.5×10^{-2}	6.3×10^{-4}	3.3×10^{-5}	7.7×10^{-6}
End-of-life copper damage (dpa)	**6×10^{-3}**	1.2×10^{-5}	7.6×10^{-3}	3.4×10^{-4}	1.7×10^{-5}	4.2×10^{-6}
Peak organic absorbed dose (rad)	**1×10^{9}**	2.1×10^{7}	8.9×10^{9}	3.8×10^{8}	2.0×10^{7}	4.7×10^{6}
Fast neutron fluence (n/cm^2)	**1×10^{19}**	2.0×10^{16}	1.1×10^{19}	5.0×10^{17}	2.5×10^{16}	6.1×10^{15}

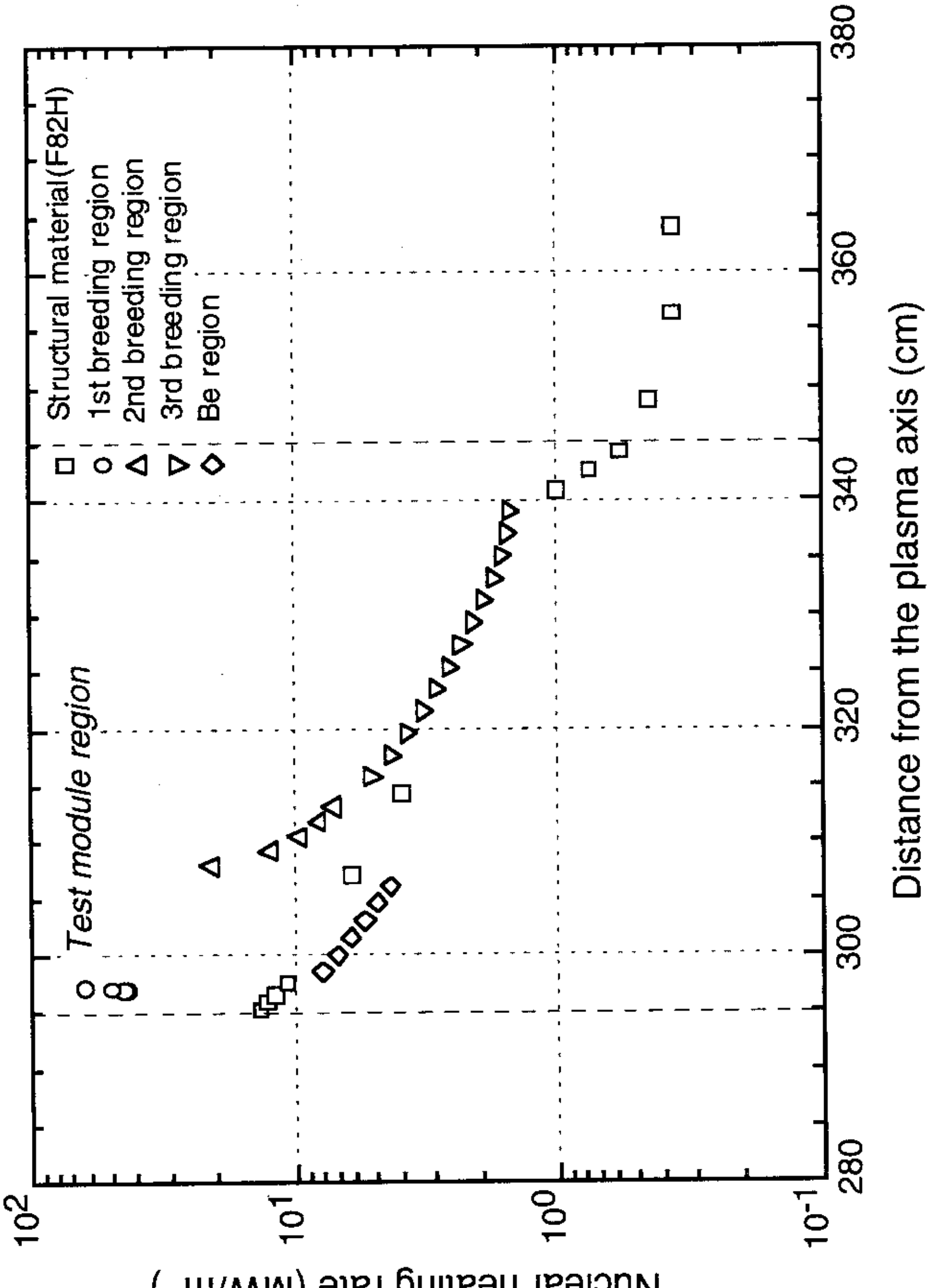


Fig. 3.2.3-1 Nuclear heating rate distribution in helium-cooled test module

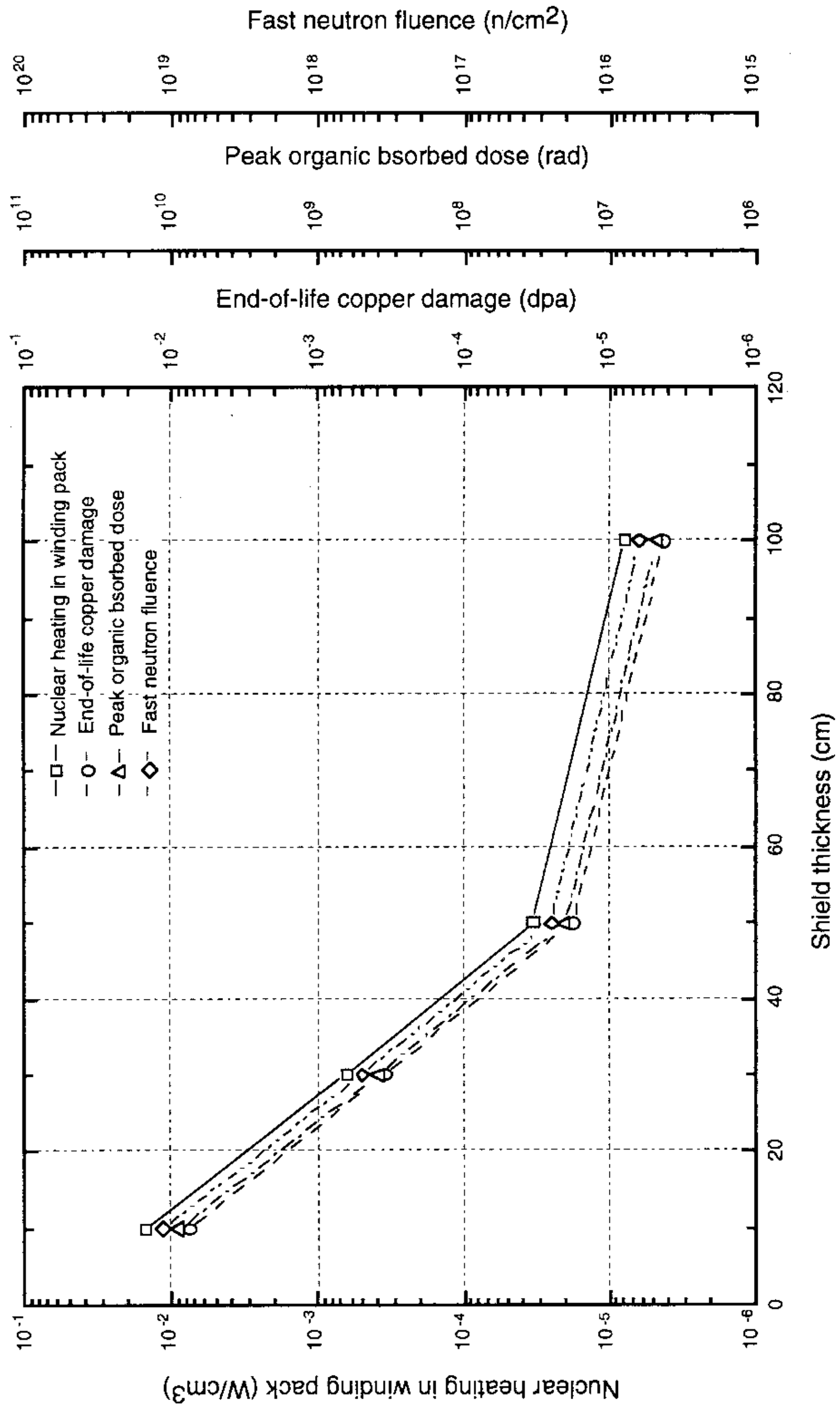


Fig. 3.2.3-2 Nuclear responce in the TF coils vs shield thickness (Helium-cooled test module, Fluence=1MWa/m^2)

3.3 Thermo-mechanical Design

The DEMO and also the test modules utilize an integrated box structure with the first wall with built-in rectangular coolant channels. Loading conditions of DEMO blankets, e.g. thermal loading, are higher than those of the test module installed in ITER. Thermo-mechanical analysis has been performed considering a method to simulate the performance of the DEMO blanket first wall by the test module as well as to clarify the effects of thermal and coolant pressure loads onto the test module first wall. Though the water-cooled DEMO blanket and test module are taken into consideration here, a similar approach can be taken to the helium-cooled blanket.

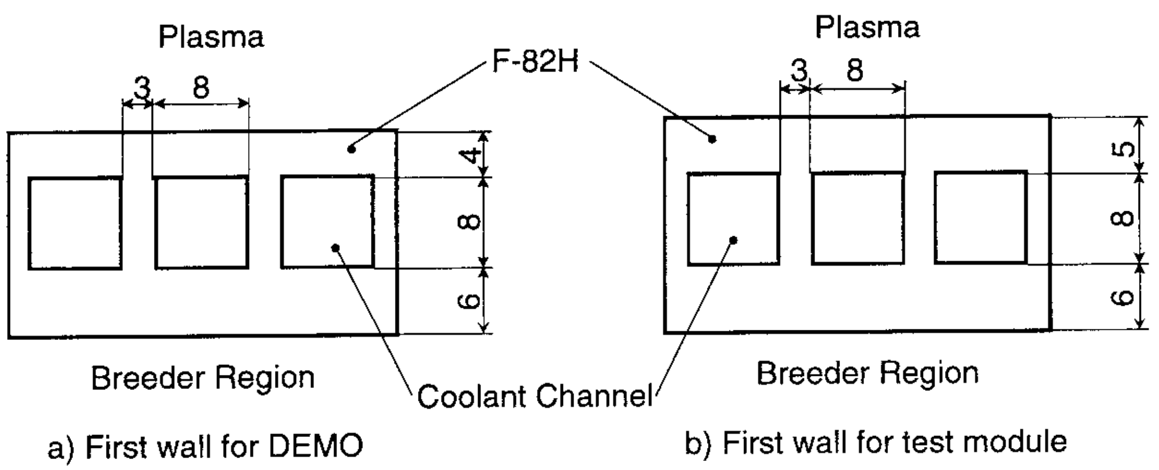


Fig. 3.3-1 First wall dimensions for DEMO blanket and test module

Dimensions of a DEMO blanket first wall, which are currently considered, are shown in Fig. 3.3-1. With operating conditions of the DEMO blanket summarized in Table 3.3-1, calculated temperature and stress distributions are shown in Figs. 3.3-2 and 3.3-3, respectively. Material properties of F-82H used in the analysis are shown in Table 3.3-2. The results are summarized in Table 3.3-3. Temperatures of the ferritic structural material, F-82H, are in the range of 338-420 °C. Maximum stress in F-82H is 290 MPa at the corner of the coolant channel.

Table 3.3-1 Operating conditions of water-cooled DEMO blanket and test module first walls

Operating condition	DEMO blanket	Test module
Heat flux from plasma (MW/m^2)	0.5	0.5
Heat flux from breeder region (MW/m^2)	0.3	0.2
Volumetric heating (MW/m^3)	45	15
Coolant temperature (°C)	320	320
Coolant pressure (MPa)	15	15
Coolant velocity (m/s)	5	4
Coolant heat transfer coefficient (W/m^2K)	38800	32500

Table 3.3-2 Material properties of ferritic steel, F-82H

Temperature (°C)	Thermal conductivity (W/mK)	Young's modulus (GPa)	Poisson's ratio	Thermal expansion coefficient (/K)
25	30.3	215	0.29	
100	34.1	215	0.29	1.00×10^{-5}
200	31.7	210	0.29	1.10×10^{-5}
300	32.2	205	0.29	1.14×10^{-5}
400	31.5	195	0.29	1.18×10^{-5}
500	32.9	190	0.29	1.21×10^{-5}
600	32.8	180	0.30	1.24×10^{-5}

Table 3.3-3 First wall thermo-mechanical analysis results

First wall characteristics	DEMO blanket	Test module
Temperature range (°C)	338-420	330-425
Maximum Tresca stress (MPa)	290	250

In Table 3.3-1, operating conditions of the test module are also indicated. Volumetric heating rate in the test module first wall, for instance, is almost three times lower than that in the DEMO blanket because of lower neutron wall loading in ITER than in DEMO. With these lower loads, one of ways to simulate the DEMO first wall performance in ITER is to increase the wall thickness of the test module thus increase the temperature and stress.

The case in which the test module first wall is 1 mm thicker than that of the DEMO blanket, as shown in Fig. 3.3-1, was analyzed. The results of temperatures and stresses are indicated in Figs. 3.3-4 and 3.3-5 and also summarized in Table 3.3-3. As seen from the figures and table, temperatures and stresses in the test module first wall are similar to those of the DEMO blanket, thus the test module first wall simulates well the thermo-mechanical behavior of the DEMO blanket first wall. As allowable limit of primary+secondary stress for F-82H, corresponding to 3Sm defined in ASME Code Section III, is in the level of 500 MPa, the stresses are well below the limit. Further analyses regarding to thermal and structural behaviors as the integrated first wall/blanket (test module) box structure are needed also considering the effects of this first wall modification on other features such as test module breeding performance and the effects of electromagnetic force on the primary stress in the first wall.

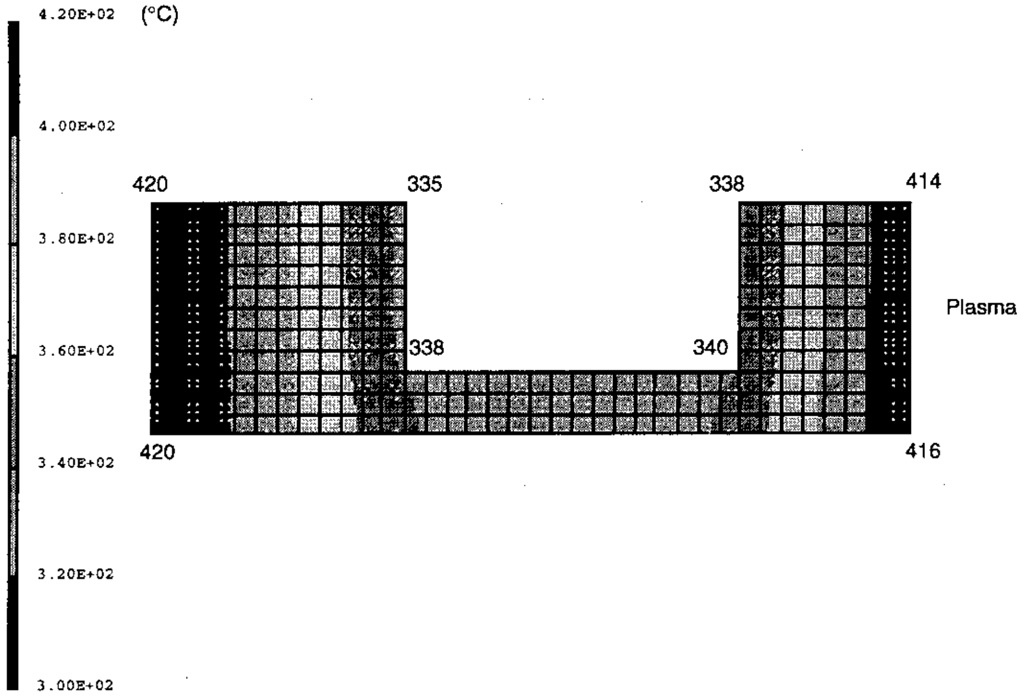


Fig. 3.3-2 Temperature distribution in DEMO blanket first wall

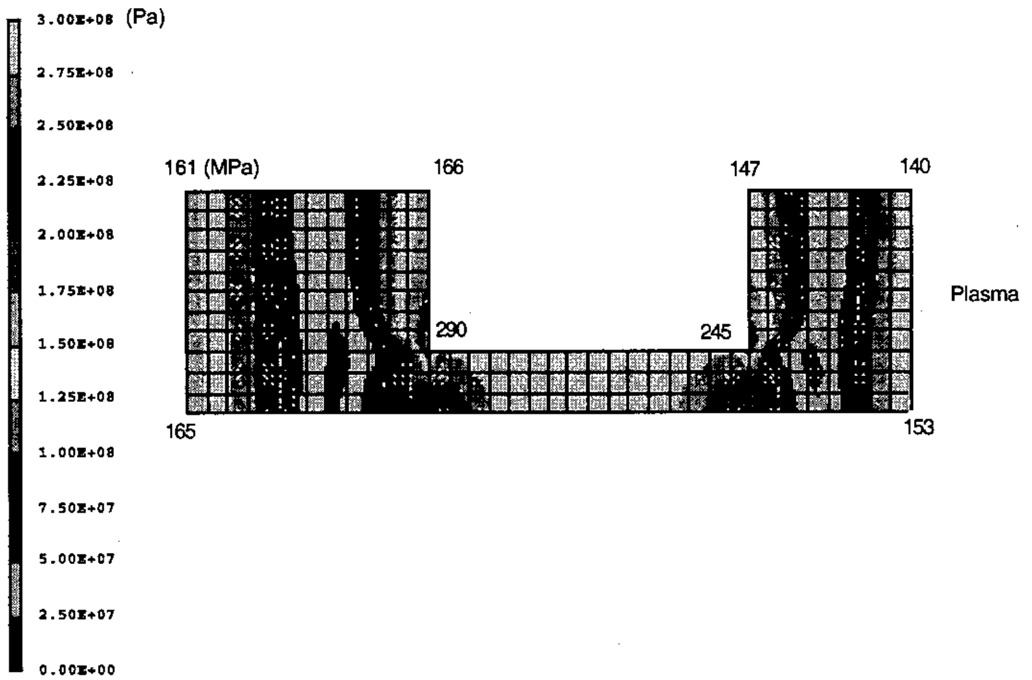


Fig. 3.3-3 Tresca stress distribution in DEMO blanket first wall

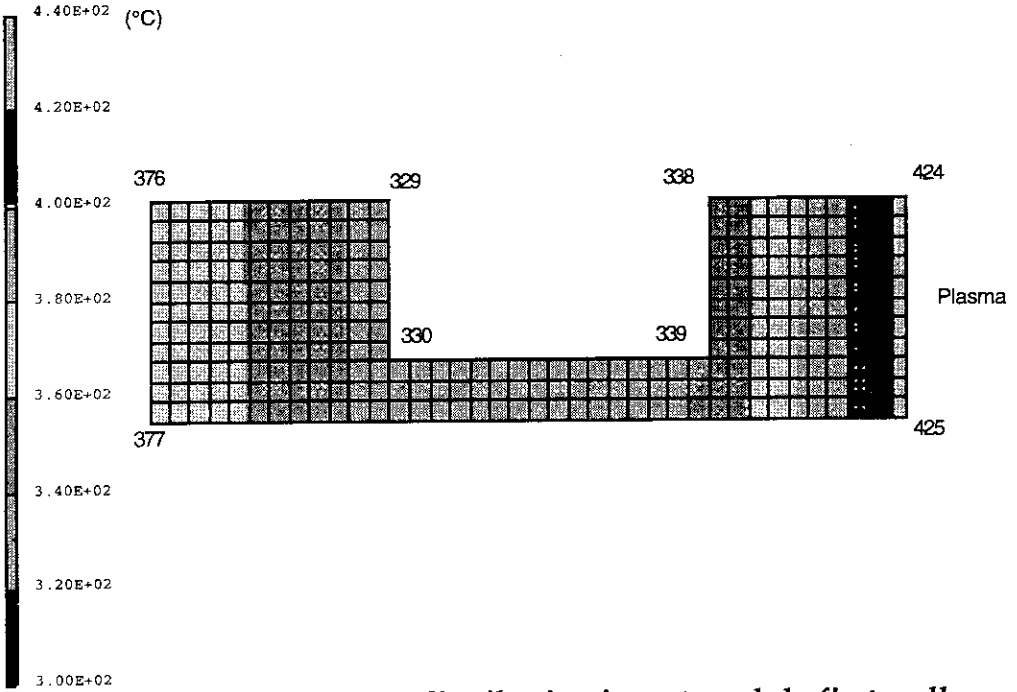


Fig. 3.3-4 Temperature distribution in test module first wall

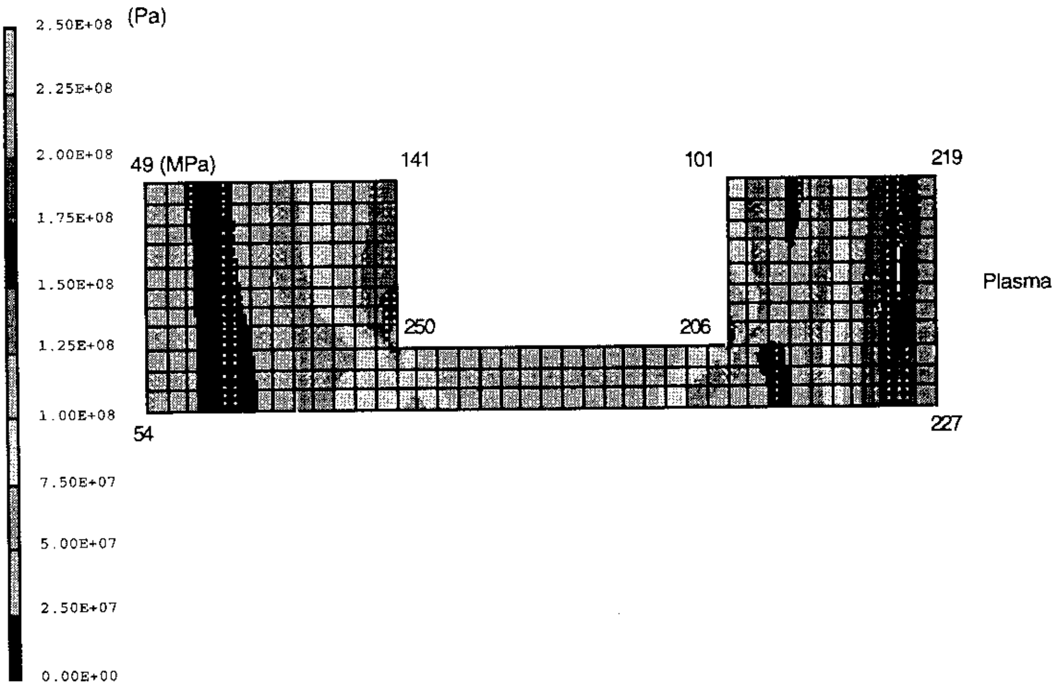


Fig. 3.3-5 Tresca stress distribution in test module first wall

3.4 Thermal-hydraulics Design

3.4.1 Water-Cooled Test Module

The thermal-hydraulics design is to handle the surface heat flux of 0.25 MW/m^2 (average) and the corresponding volumetric heat generation from the neutron wall loading of 1.2 MW/m^2. Based on the neutronics results, thermal-hydraulics analyses were performed to determine coolant temperature distributions and pressure drops. A static pressure of 15 MPa, an inlet temperature of 280 °C and an outlet temperature of 320 °C are selected for water coolant based on conventional PWR conditions. The proposed cooling concept utilizes coolant routed from the back poloidal manifold to cool the first wall and then re-routed to cool the breeding zone. The coolant flows in the radial-toroidal direction around the first wall. Small plenums are located at the first wall panel ends next to the module back wall for collecting inlet and outlet flows. The latter collector is connected to an inlet collector for the breeding region at the module bottom, then the coolant flows poloidally in the breeding region from a bottom collector to a top collector. This cooling scheme is selected in order to insure an thermo-mechanical operational margin for the first wall. The coolant flow rate for the proposed test module with a thermal power of 1.68 MW and coolant temperature rise of 40 °C is 7.6 kg/s. The calculated results are summarized in Table 3.4.1-1. Further optimization, e.g. to improve heat transfer performance at the first wall, is required.

Table 3.4.1-1 Thermal-hydraulics features of water-cooled test module

Plasma facing surface area	0.44 m^W x 2.12 m^H
Thermal power	1.68 MW
Surface heat flux from plasma	0.25 MW/m^2
Heat flux from breeding zone to first wall	0.2 MW/m^2
Volumetric heating rate	See nuclear analysis (Fig. 3.2.2-1).
First wall coolant channel size	8 mm x 8 mm
Breeding zone coolant channel diameter	10 mm
Coolant	water
Pressure	15 MPa
First wall inlet/outlet temperature	280/291 °C
Breeding zone inlet/outlet temperature	291/320 °C
Flow rate	7.6 kg/s
Velocity at first wall	0.86 m/s
Velocity at breeding zone	0.7 - 2.8 m/s
Heat transfer coefficient at first wall	9600 W/m^{2}K
Heat transfer coefficient at breeding zone	7577 - 23310 W/m^{2}K
Pressure drop at first wall	0.002 MPa
Pressure drop at breeding zone	0.018 MPa
Total pressure drop	0.020 MPa

3.4.2 Helium-Cooled Test Module

Thermal-hydraulics analyses are performed based on the heat loads of the surface heat flux of 0.25 MW/m^2 (average) and the volumetric heat generation from the neutron wall loading of 1.2 MW/m^2. A static pressure of 8.5 MPa, an inlet temperature of 360 °C and an outlet temperature of 480 °C are selected for helium coolant involving ferritic structural material. The proposed cooling concept utilizes coolant routed from the back poloidal manifold to cool the first wall and then re-routed to cool the breeding zone. The coolant flows in the radial-toroidal direction around the first wall. Small plenums are located at the first wall panel ends next to the module back wall for collecting inlet and outlet flows. The coolant then flows into a bottom collector and poloidally in the breeding region to a top collector. This cooling scheme is selected in order to insure an thermo-mechanical operational margin for the first wall and at the same time for the breeder temperature to be above the minimum allowable temperature for tritium release. The coolant flow rate for the proposed test module with a thermal power of 1.90 MW and coolant temperature rise of 120 °C is 3.1 kg/s. The results summarized in Table 3.4.2-1 are still preliminary. Further optimization including the dimension of breeding zone coolant channels so as to have higher heat transfer performance is required.

Table 3.4.2-1 Thermal-hydraulics features of helium-cooled test module

Plasma facing surface area	1.12^W m x 0.94^H m
Thermal power	1.90 MW
Surface heat flux from plasma	0.25 MW/m^2
Heat flux from breeding zone to first wall	0.2 MW/m^2
Volumetric heating rate	See nuclear analysis (Fig. 3.2.3-1).
First wall coolant channel size	12 mm x 12 mm
Breeding zone coolant channel diameter	16 mm
Coolant	helium
Pressure	8.5 MPa
First wall inlet/outlet temperature	360/394 °C
Breeding zone inlet/outlet temperature	394/480 °C
Flow rate	3.1 kg/s
Velocity at first wall	57 m/s
Velocity at breeding zone	6.2 - 18.2 m/s
Heat transfer coefficient at first wall	5140 W/m^{2}K
Heat transfer coefficient at breeding zone	800 - 1900 W/m^{2}K
Pressure drop at first wall	0.068 MPa
Pressure drop at breeding zone	0.0017 MPa
Total pressure drop	0.070 MPa

4 Cooling Systems for Test Blanket

4.1 Water-Cooled Test Module

Design conditions and assumptions are summarized in Table 4.1-1. Two test modules will be installed in one test port. As the dimension of the port opening is 1.6 m^W x 2.6 m^H, the first wall area facing plasma of one test module is approximately 0.44 m^W x 2.12 m^H assuming the installation of two test modules in 200 mm thick attachment frame. Thermal power (removal heat) of a test module is about 1.68 MW assuming 0.25 MW/m^2 of surface heat flux, nuclear heating due to neutron wall loading of 1.2 MW/m^2 and above first wall area. Primary coolant conditions are 280 °C and 320 °C at module inlet and outlet, respectively, and 15 MPa of pressure. The flow rate of the primary coolant is 27.3 t/h. It is also assumed that the thermal power of the test module is transferred to the ITER secondary coolant water of 35/75 °C at a heat exchanger inlet/outlet and 0.5 MPa. For the attachment frame and the additional shield, water coolants of the ITER first wall/blanket and vacuum vessel would be used, respectively.

Table 4.1-1 Cooling system design conditions for water-cooled test module

First wall area facing plasma	0.44 m^W x 2.12 m^H
Surface heat flux	0.25 MW/m^2
Neutron wall load	1.2 MW/m^2
Total removal heat	1.68 MW
Primary coolant	Water
Temperature (module in/out)	280/320 °C
Pressure	15 MPa
Flow rate	7.6 kg/s (27.3 t/h)
Secondary coolant	Water
Temperature (heat exchanger in/out)	35/75 °C
Pressure	0.5 MPa
Flow rate	10.0 kg/s (36.1 t/h)

A flow diagram of the cooling system for the water-cooled test module is shown in Fig. 4.1-1. Main components of this system are a heat exchanger, a circulation pump, a pressurizer and a heater. The heat exchanger is a type of shell-and-tube with counter flow. The high pressure primary coolant flows inside tube and low pressure secondary coolant outside tube. Main parameters of the heat exchanger is summarized in Table 4.1-2. Two circulation pumps will be equipped for redundancy in case of a pump trip accident. The pressurizer is designed to accommodate the volumetric change of water coolant due to its temperature rise from room temperature (20 °C) to 300 °C. The heater with a heating power of 200 kW is provided to warm-up the system by temperature rising rate at about 50 °C/h and also to adjust the test module inlet temperature during operation.

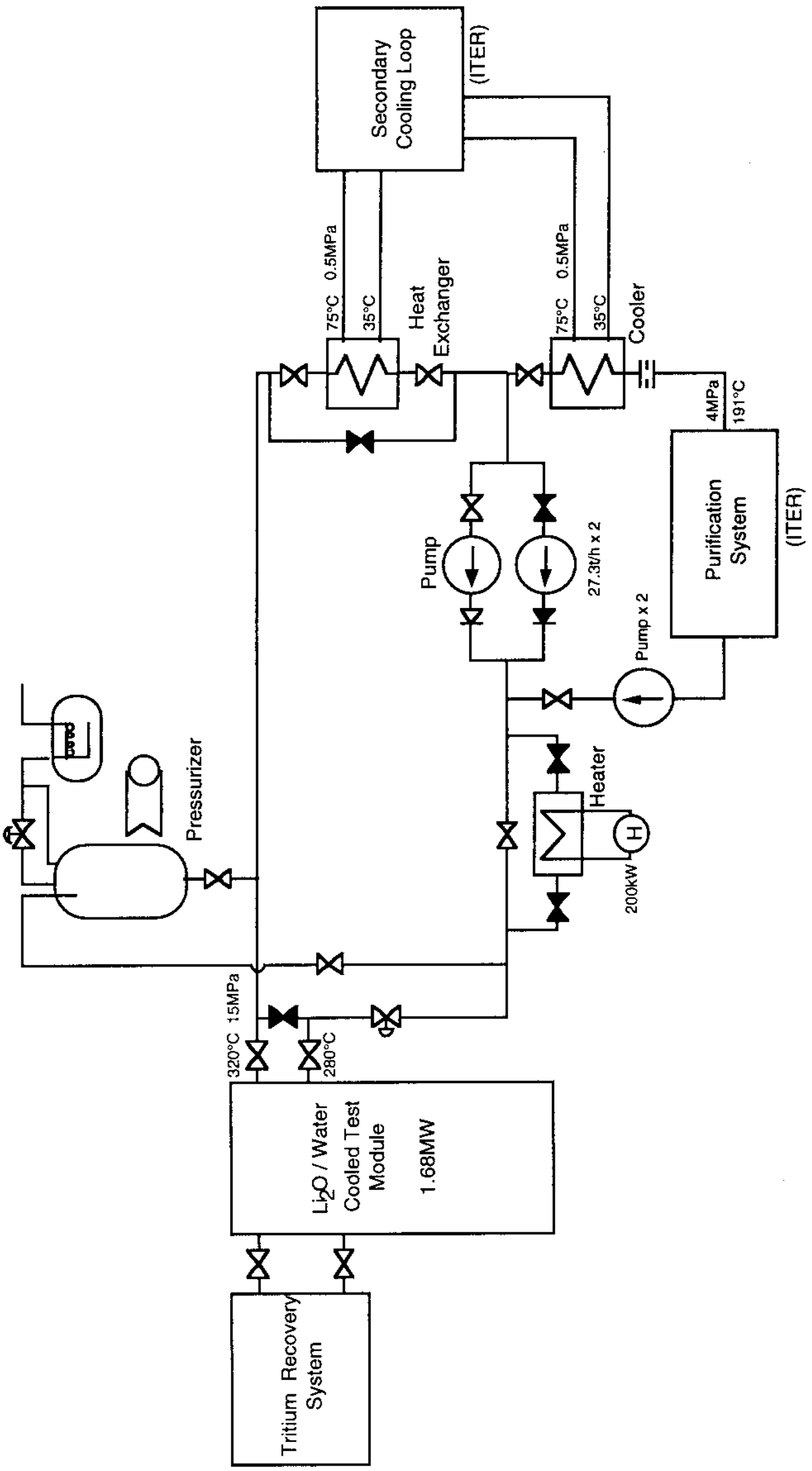


Fig. 4.1-1 Cooling system flow diagram for Water-cooled test module

Table 4.1-2 Heat exchanger specification for water-cooled test module

Type	Counter flow type shell and tube
Number of heat exchangers	1
Removal heat	1.68 MW
Heating tube size	25.4 mm^{OD}/ 2.6 mm^{t}
Number of heating tubes	85
Shell size	0.43 m^{OD}/ 1.7 m^{L}
Primary coolant	Water inside tube
In/out temperature	320/280 °C
Pressure	15 MPa
Flow rate	7.6 kg/s (27.3 t/h)
Velocity	0.4 m/s
Heat transfer coefficient	4200 $W/m^2/K$
Fauling factor	5800 $W/m^2/K$
Secondary coolant	Water outside tube
In/out temperature	35/75 °C
Pressure	0.5 MPa
Flow rate	10.0 kg/s (36.1 t/h)
Velocity	0.13 m/s
Heat transfer coefficient	3700 $W/m^2/K$
Fauling factor	5800 $W/m^2/K$

For the purification and detritiation of the test module primary coolant, it is proposed that those system in the ITER plant could be used. For this interface with the ITER system, a cooler and a pressure reducer to be consistent with the ITER coolant conditions and a pump for the return to the test module cooling system are to be provided.

Dimensions of above components are listed in Table 4.1-3. Major specifications of the coolant piping and their interfaces are summarized in Table 4.1-4. The water hold-up in the test module cooling system is about 1.5 m^3 assuming the length of the main coolant supply and return pipes (89.1 mm in outer diameter and 11.1 mm thickness) to be 60 m each.

Table 4.1-3 Dimensions of major cooling system components for water-cooled test module

Heat exchanger	0.43 m^{OD} x 1.7 m^{L}
Circulator	0.32 m^{OD} x 1.2 m^{L}
Pressurizer	0.8 m^{OD} x 1.2 m^{L}
Heater	0.4 m^{OD} x 1.1 m^{L}
Cooler*	0.27 m^{OD} x 2.0 m^{L}
Pump*	0.5 m^{OD} x 1.31 m^{L}

* for interface with ITER coolant purification system

Table 4.1-4 Water-cooled test module coolant piping interface

Pipe description	Size* and number	Pipe carrier	Operating conditions	Tritium contained	Connection from/to
TM water supply	89.1 mm^{OD} 1	Water	15 MPa 280 °C	< 1 g (TBD***)	TM <--TM/CS
TM water return	89.1 mm^{OD} 1	Water	15 MPa 320 °C	< 1 g (TBD***)	TM --> TM/CS
Ad. shield** water supply	48.6 mm^{OD} 1	Water	4 MPa 140 °C	0	TM <-- ITER/VV/CS (TBD****)
Ad. shield** water return	48.6 mm^{OD} 1	Water	4 MPa 150 °C	0	TM --> ITER/VV/CS (TBD****)
Frame water supply	TBD 1	Water	4 MPa 140 °C	< 1 g (TBD***)	TM <-- ITER/BL/CS (TBD****)
Frame water return	TBD**** 1	Water	4 MPa TBD****	< 1 g (TBD***)	TM --> ITER/BL/CS (TBD****)
Diagnostics conduit	TBD****				TM <--> TBD****

* thermal insulation around pipe not included
** assuming additional shield common to two test modules
*** preliminary estimated
**** to be determined after cooling conditions for the additional shield and common frame are agreed among the Parties and JCT
TM: Test Module, CS: Cooling System, VV: Vacuum Vessel, BL: Blanket
TBD: To Be Determined

A layout plan and an isometric view of the cooling system are shown in Figs. 4.1-2 and 4.1-3, respectively. The space required for the installation of this system is approximately 9.5 m x 5.8 m x 5 m^{H}. The available pit area, possibly 1.4-2.7 m x 8 m x 6.5 m^{H}, seems too tight for equipping this system, thus a space or a room in the ITER tritium plant building needs to be considered.

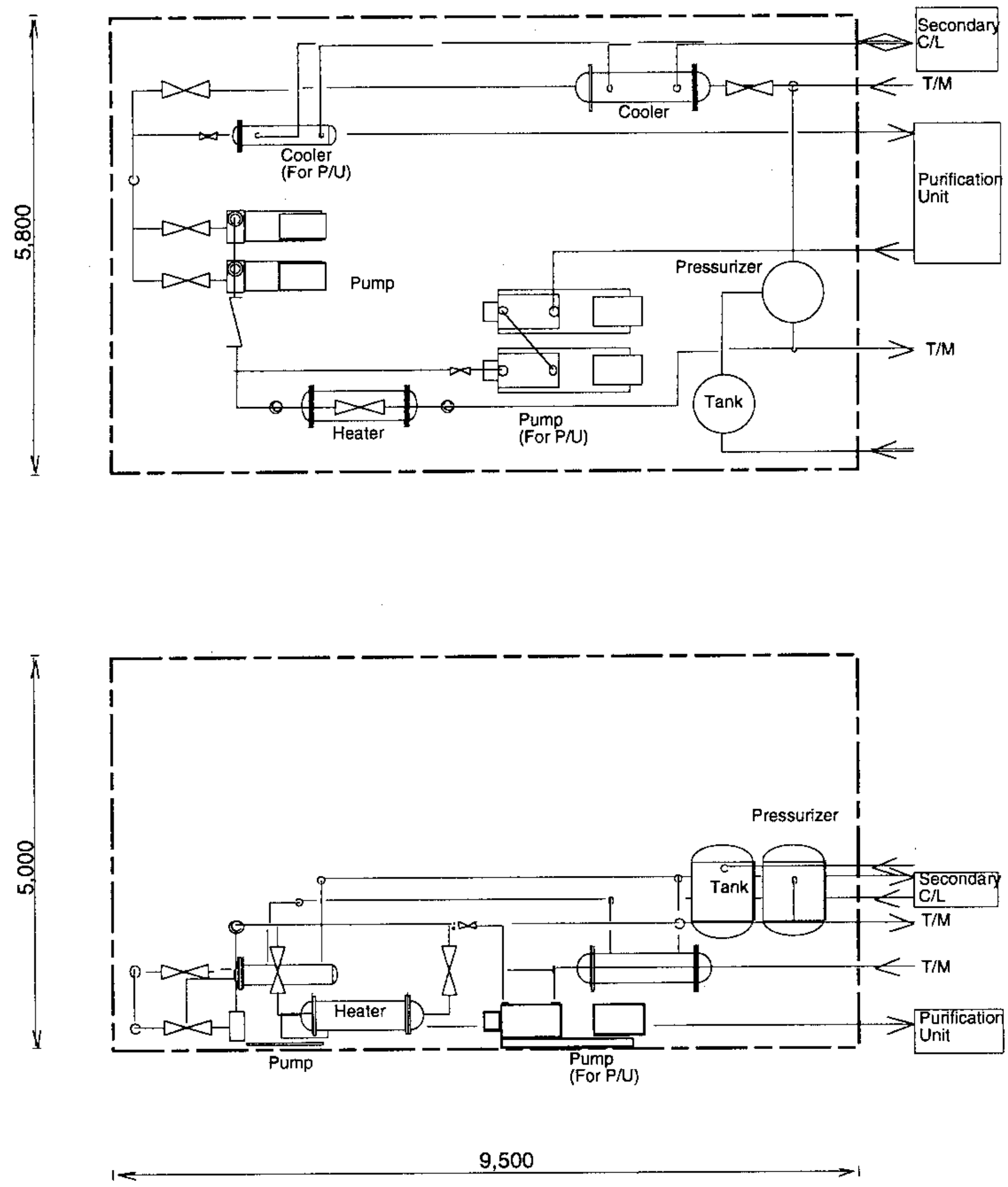


Fig. 4.1-2 Cooling system Layout plan for Water-cooled test module

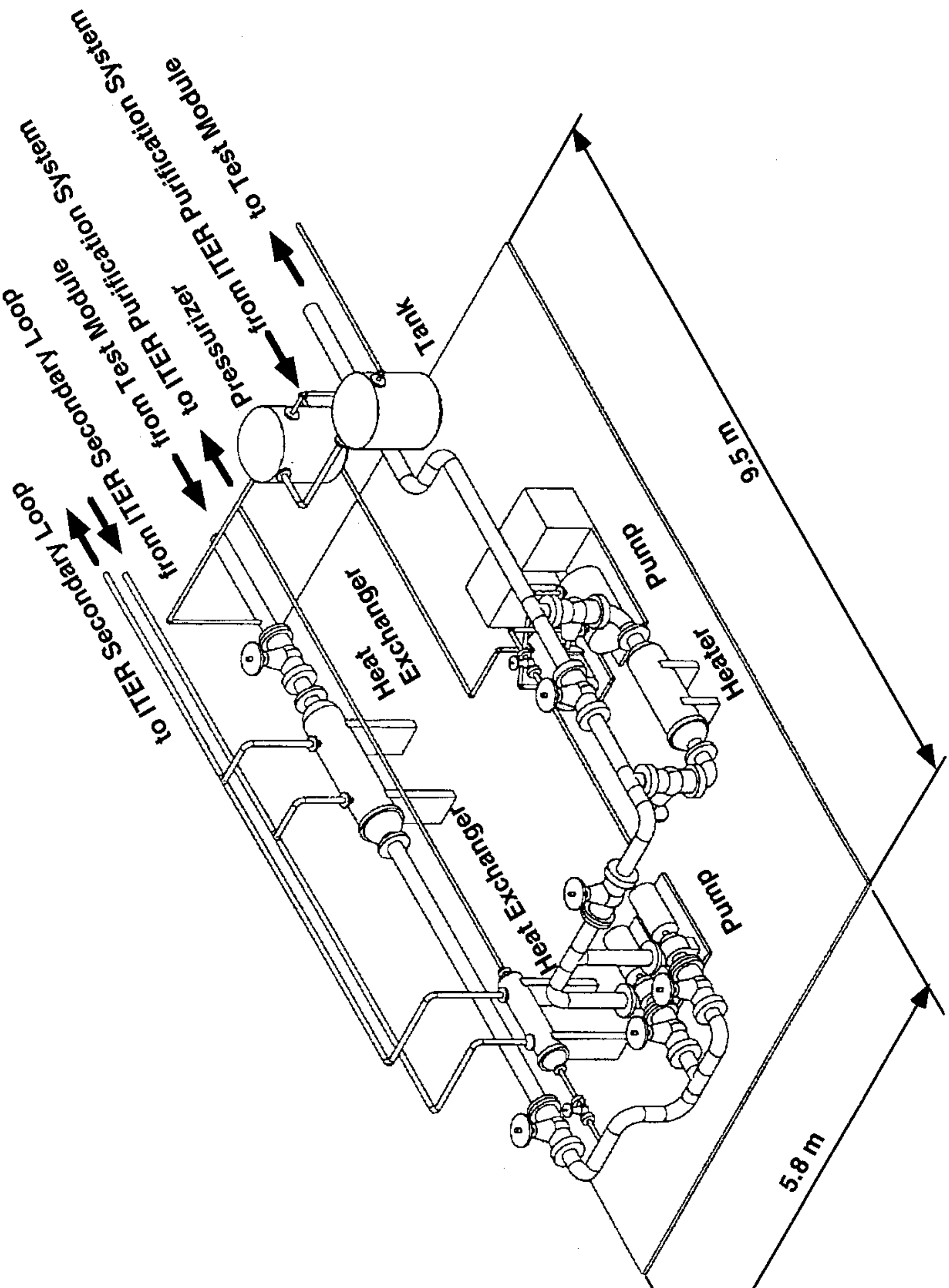


Fig. 4.1-3 Isotropic view of cooling system for Water-cooled test module

4.2 Helium-Cooled Test Module

Design conditions and assumptions are summarized in Table 4.2-1. Since two test modules will be installed in one test port, one test module is in the approximate size of 1.12 m^W x 0.94 m^H. Thermal power (removal heat) of a helium-cooled test module is almost the same as that of the water-cooled test module, i.e. 1.90 MW. Primary coolant conditions are 360 °C and 480 °C at module inlet and outlet, respectively, and 8.5 MPa of pressure. The flow rate of the primary coolant is 3.1 kg/s. The thermal power of the test module will be transferred to the ITER secondary coolant water of 35/75 °C at a heat exchanger inlet/outlet and 0.5 MPa. The use of ITER water coolants, possibly of first wall/blanket and vacuum vessel cooling systems, to cool the attachment frame and the additional shield, respectively, is the same as for the water-cooled test module.

Table 4.2-1 Cooling system design conditions for helium-cooled test module

First wall area facing plasma	1.12 m^W x 0.94 m^H
Surface heat flux	0.25 MW/m^2
Neutron wall load	1.2 MW/m^2
Total removal heat	1.90 MW
Primary coolant	Helium
Temperature (module in/out)	360/480 °C
Pressure	8.5 MPa
Flow rate	3.1 kg/s (11.0 t/h)
Secondary coolant	Water
Temperature (heat exchanger in/out)	35/75 °C
Pressure	0.5 MPa
Flow rate	11.4 kg/s (40.9 t/h)

A flow diagram of the cooling system for the helium-cooled test module is shown in Fig. 4.2-1. Main components of this system are a heat exchanger, a circulator, a heater and a purification unit. The heat exchanger is a type of shell-and-tube with counter flow. The high pressure primary coolant flows inside tube and low pressure secondary coolant outside tube. Main parameters of the heat exchanger is summarized in Table 4.2-2. Two circulators will be equipped for redundancy in case of a circulator trip accident. The heater with a heating power of 150 kW is provided for warming-up the system by temperature rising rate at 50 °C/h and to adjust the test module inlet temperature during operation. The purification unit is equipped aiming at extracting hydrogen isotopes as well as solid, liquid or gaseous impurities from the main coolant system and removing condensed water that may be entrained in the main coolant due to leakage or failures of the heat exchanger tubes.

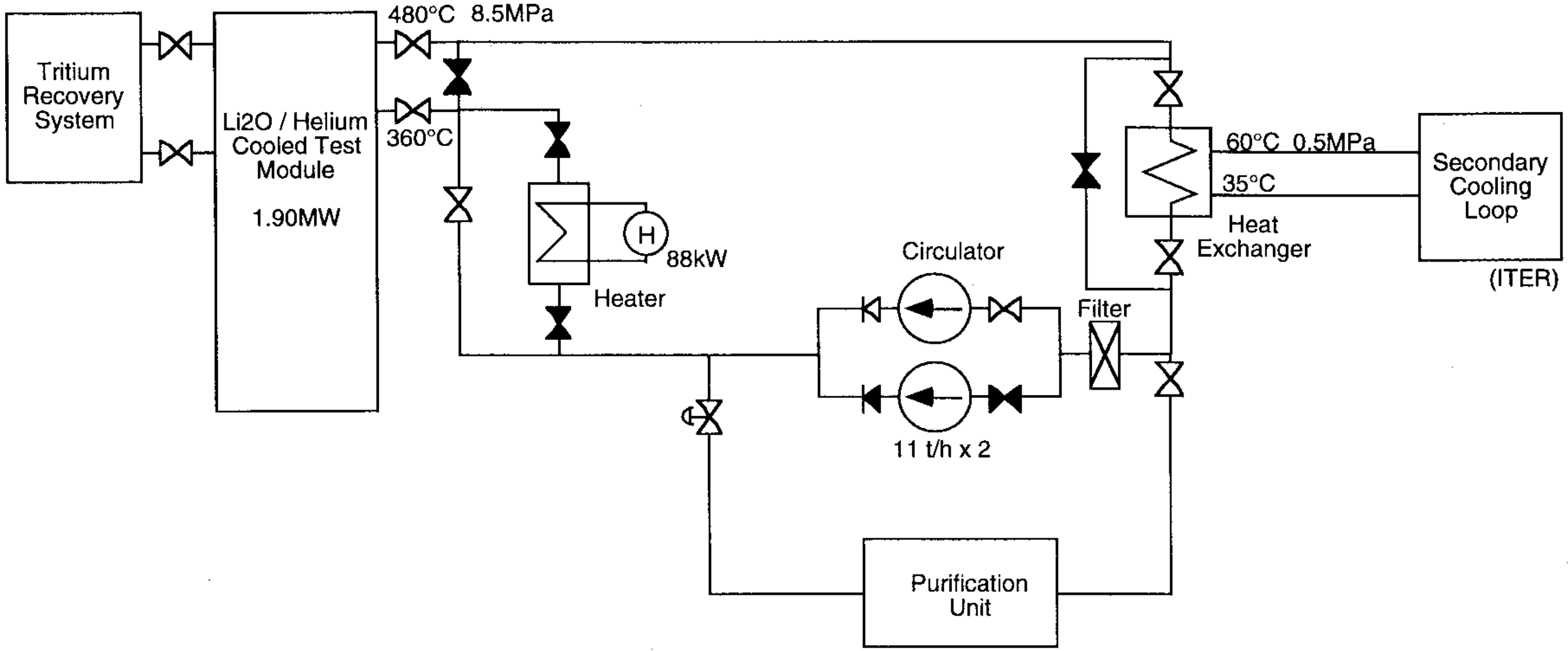


Fig. 4.2-1 Cooling system flow diagram for Helium-cooled test module

Table 4.2-2 Heat exchanger specification for helium-cooled test module

Type	Counter flow type shell and tube
Number of heat exchangers	1
Removal heat	1.90 MW
Heating tube size	31.8 mm^{OD}/2.0 mm^{t}
Number of heating tubes	91
Shell size	0.50 m^{OD}/1.8 m^{L}
Primary coolant	Helium inside tube
In/out temperature	480/360 °C
Pressure	8.5 MPa
Flow rate	3.1 kg/s (21.7 t/h)
Velocity	9.5 m/s
Heat transfer coefficient	940 W/m^2/K
Fauling factor	2900 W/m^2/K
Secondary coolant	Water outside tube
In/out temperature	35/75 °C
Pressure	0.5 MPa
Flow rate	11.4 kg/s (40.9 t/h)
Velocity	0.14 m/s
Heat transfer coefficient	3450 W/m^2/K
Fauling factor	5800 W/m^2/K

A fraction of 0.1 % of the main helium coolant stream is fed into the purification system which include catalytic oxidizer, water separator, cold trap and low temperature adsorber.

Dimensions of above components and unit are listed in Table 4.2-3. The design of the purification system will be revisited for further minimization in its size. Major specifications of the coolant piping and their interfaces are summarized in Table 4.2-4. The helium hold-up in the test module cooling system is about 2 m^3 assuming the length of main coolant supply and return pipes to be 60 m each.

A layout plan and an isometric view of the cooling system are shown in Figs. 4.2-2 and 4.2-3, respectively. The space required for the installation of the system is approximately 12.5 m x 8 m x 7.5 m^H. A space to place this cooling system would be found in the ITER tritium plant building.

Table 4.2-3 **Dimensions of major cooling system components for helium-cooled test module**

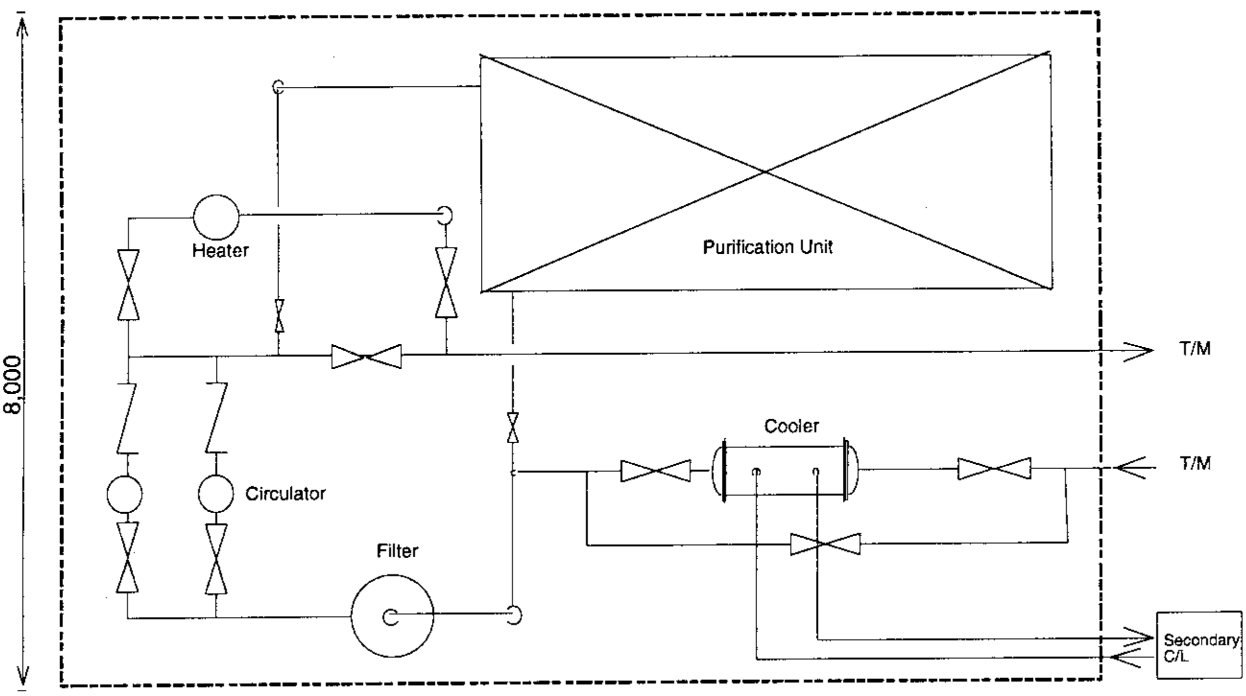
Heat exchanger	0.5 m^{OD} x 1.8 m^{L}
Circulator	0.4 m^{OD} x 1.5 m^{L}
Heater	0.55 m^{OD} x 1.3 m^{L}
Purification unit	6.85 m x 2.8 m x 4.6 m^{H}

Table 4.2-4 Helium-cooled test module coolant piping interface

Pipe description	Size* and number	Pipe carrier	Operating conditions	Tritium contained	Connection from/to
TM helium supply	139.8 mm^{OD} 1	Helium	8.5 MPa 360 °C	TBD***	TM <--TM/CS
TM helium return	139.8 mm^{OD} 1	Helium	8.5 MPa 480 °C	TBD***	TM --> TM/CS
Ad. shield** water supply	48.6 mm^{OD} 1	Water	4 MPa 140 °C	0	TM <-- ITER/VV/CS (TBD****)
Ad. shield** water return	48.6 mm^{OD} 1	Water	4 MPa 150 °C	0	TM --> ITER/VV/CS (TBD****)
Frame water supply	TBD**** 1	Water	4 MPa 140 °C	TBD***	TM <-- ITER/BL/CS (TBD****)
Frame water return	TBD**** 1	Water	4 MPa TBD****	TBD***	TM --> ITER/BL/CS (TBD****)
Diagnostics conduit	TBD****				TM <--> TBD****

* thermal insulation around pipe not included
** assuming additional shield common to two test modules
*** to be evaluated
**** to be determined after cooling conditions for the additional shield and common frame are agreed among the Parties and JCT

TM: Test Module, CS: Cooling System, VV: Vacuum Vessel, BL: Blanket
TBD: To Be Determined



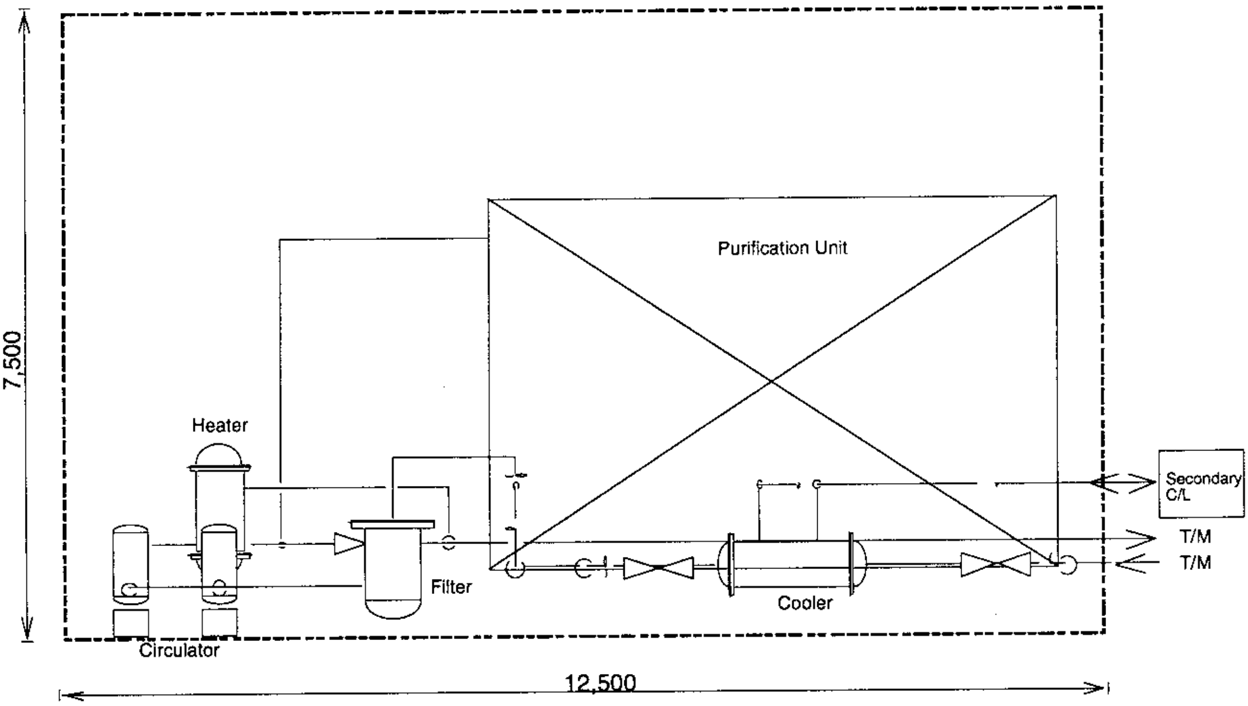


Fig. 4.2-2 Cooling system Layout plan for He-cooled test module

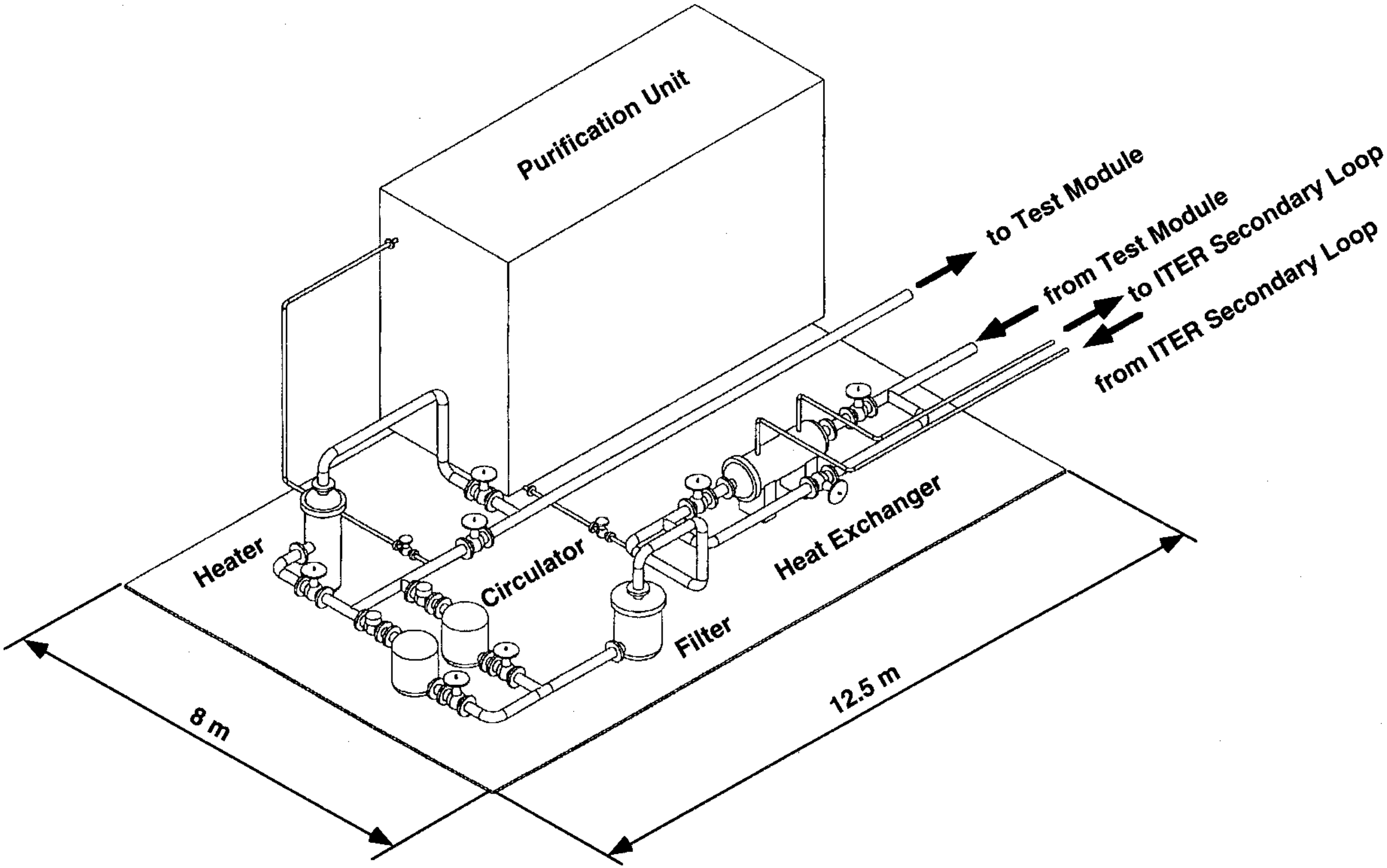


Fig. 4.2-3 Isometric view of cooling system for Helium-cooled test module

5 Tritium Recovery System

5.1 Functions and Design Conditions

Tritium recovery system for the test module has very important roll to recover tritium from helium purge gas of the test module, clean up humidity or vapor from helium purge gas, and supply clean purge gas to the test module. At the same time, it is required to transfer recovered tritium in the most suitable form to ITER tritium plant with minimal impact to ITER tritium plant systems.

The major functions required to the tritium recovery system of the test module are,

1) to measure gas composition of helium purge gas for the purpose of the evaluation of test module function,
2) to recover H_2 and HT, H_2O and HTO from the helium purge gas,
3) to cleanup purge gas (humidity and vapor) and condition

The basic concept and the general configuration of the system, the preliminary estimate of the dimensions of major components and the space requirement will be described in the following sections.

Table 5-1 summarizes the design conditions for the tritium recovery system of test modules. Basically, the helium purge gas will contain about 0.5 % of H_2 swamping gas (H/T=100) to promote tritium release in the form of HT from the ceramic breeder material. The chemical form and mole fraction of tritium are assumed as HT/HTO = 95/5 by rough estimation. The atmosphere control of the breeder material that influences on the formation of LiOH and tritium release behavior needs to be examined in more detail in the test module design. In this design, the existence of LiOH/LiOT vapor is assumed. Analysis system is to be designed so that the tritium concentration in the purge gas is measured in each chemical form not only for the purpose of the test module performance evaluation, but also for the control and monitoring of tritium recovery system itself.

5.2 Concept and Configuration of the Tritium Recovery System

Schematic flow diagram of the test module tritium recovery system is shown in Fig. 5-1. The tritium recovery by helium purge gas separate from the main coolant is the same for the water-cooled and helium-cooled test modules. The tritium producing capability, thus the purge gas conditions, are also similar for both test modules. Therefore, almost the same tritium recovery system described below can be applied to these test modules. The tritium recovery system consists of LiOH/LiOT vapor trap, purge gas cooler, cryogenic molecular sieve bed, palladium diffuser, purge gas heater, transfer pump and gas analysis system. The schematic diagram of gas analysis system is shown in Fig. 5-2. The gas analysis system is equipped at the outlet of the test module and consists of moisture detector, ion chamber, gas chromatography and small dryer bed. These detectors will be set to identify H_2, HT, H_2O and HT concentration, separately. Also, appropriate detectors will be set to the important analysis points of tritium recovery system components for monitoring of test module tritium recovery system performance.

Table 5-1 Design conditions of test module tritium recovery system

Item	Unit	Water-cooled	Helium-cooled
Tritium Production			
Burn	sec	1000	1000
Dwell	sec	1200	1200
Duration of burn-dwell operation	days	7	7
Local TBR		1.2	1.3
Tritium production	g/FPD	0.23	0.29
	g/FPH	0.022	0.023
Purge gas		He	He
H/T ratio		100	100
HT/HTO ratio		95/5	95/5
At Test Module Outlet			
Temperature	K	723	723
Pressure	MPa	0.1	0.1
Purge gas flow rate	Nm^3/h	0.72	0.90
H_2 concentration	ppm	4905	4905
HT concentration	ppm	95	95
H_2O concentration	ppm	45	45
HTO concentration	ppm	5	5
Tritium concentration in He purge gas	Ci/Nm^3	131	131
At Test Module Inlet			
Temperature	K	298	298
Pressure	MPa	0.102	0.102
Purge gas flow rate	Nm^3/h	1.62	1.74
H_2 concentration	ppm	5000	5000
H_2O concentration	ppm	1	1

Chemical form of tritium recovered into the system is very important item, because it affects the requirements to the ITER tritium plant systems which will deal with tritium transferred from the tritium recovery system of test module. On the basis of the design conditions in Table 5-1, the final load of tritium and tritiated water to be processed is 0.72-0.90m^3/h in total (Q_2+Q_2O) and the HT ratio is 100, which is equivalent to 94-118 Ci/h tritium and tritiated water. Net volume of the recovered gas will not be a major impact to the current design of the ITER tritium plant although tritium concentration is relatively high. If "oxidation-adsorption" process is applied, such high level tritiated water (16300 Ci/liter, 2.2 liter/week, total) should not be stored because of unstable pressure increase by tritium radiolysis. Eventually, tritiated water should be treated to turn its chemical form to hydrogen isotopes, which become impact to the water detritiation system of the ITER tritium plant. This situation causes the necessity to recover H_2 and HT without oxidation and transfer to the cryogenic distillation column directly. Thus, the recovery of H_2 and HT from the helium purge gas should be done by non-oxidation process.

The most reliable and established process to recover H_2 and HT from helium purge gas is cryogenic molecular sieve bed (CMSB) coupled with back-end permeator. The back-end permeator is to purify recovered H_2 and HT. Also, it is necessary to remove tritiated humidity in the upstream of CMSB to avoid the water entrainment.

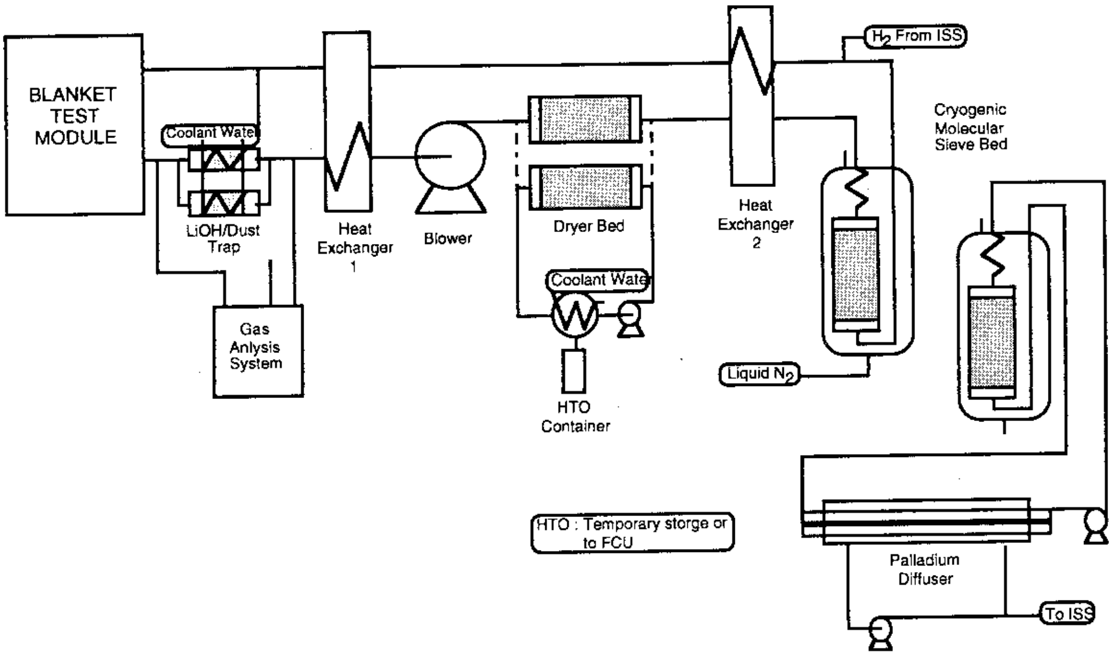


Fig. 5-1 Schematic flow sheet of test module tritium recovery system

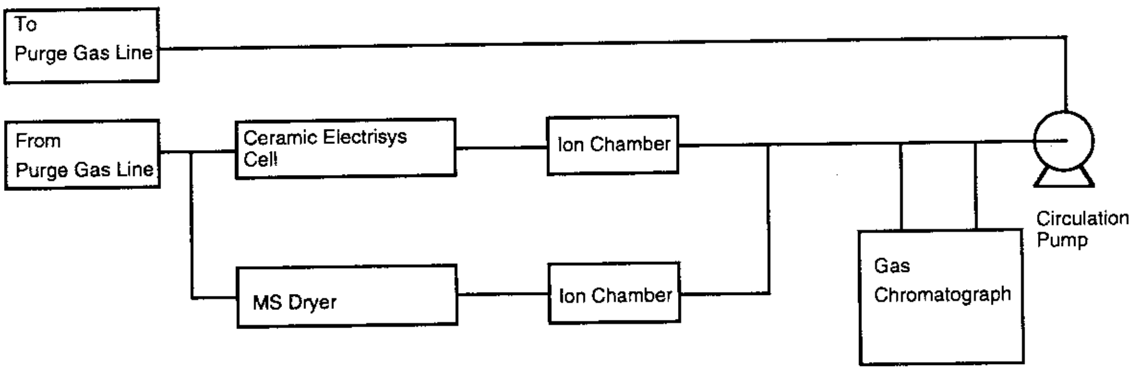


Fig. 5-2 Flow diagram of test module gas analysis system (GAS)

5.3 Components Description

Table 5-2 summarizes the specification of the major components. The objective of the test module tritium recovery system is not only to recover tritium or clean up purge gas, but also to enable testing test module operation in various condition. The design flow rate of purge gas is assumed as the nominal value and the design includes some margin to enable this objective.

Table 5-2 Specification of major components of tritium recovery system

Component	Specification	Dimension [cm]	Weight [kg]	Number
LiOH Trap	Vertical cylinder SS wool cooled by water	26 φ x 54 h	30	2
Dryer Bed	Vertical cylinder MS5A packed bed at ambient temperature	30 φ x 40 h	50	2
Cryo-MS Bed	Vertical cylinder MS5A beds in cold box of LN_2	50 φ x 150 h	200	3
Palladium Diffuser	Palladium diffuser for recovered Q_2 purification	30 φ x 120 h	150	1
Heat Exchanger 1	Shell & tube heat exchanger to cool outlet purge gas from TM and to heat inlet purge gas to TM	20 φ x 100 h	200	1
Heat Exchanger 2	Shell & tube heat exchanger as the pre-cooler and after heater of CMSB	30 φ x 100 h	450	1
Blower	Purge gas circulation pump	40 x 80 x 50	100	1
Vacuum Pump 1	Scroll pump for dryer regeneration	40 x 40 x 40	50	2
Vacuum Pump 2	Scroll pump for CMSB regeneration	40 x 40 x 40	50	2
Gas Analysis System for TM	Lined up analysis set in glove box	100 x 100 x 100	100	1
Gas Analysis System for Components	Gas chromatography	50 x 50 x 50	20	2
	Ion chamber	2 φ x 2 h	----	10-20
	Moisture detector	30 x 20 x 20	5	2

LiOH Trap: Though two types of LiOH trap, wet type and dry type, are available, the dry type makes maintenance easier. Thus, dry type trap with heat exchange function is selected. Cooled SS wool can trap LiOH vapor more effectively.

Dryer Bed: Dryer bed has the function to remove humidity from the helium purge gas. Net amount of humidity is estimated no more than 1 kg per a week continuous operation. However, tritium concentration is very high in this case. Thus, special care is necessary in regeneration of the dryers and tritiated water treatment. The operation cycle is about 48 hours a week depending on the humidity.

Cryogenic Molecular Sieve Bed: Cryogenic molecular sieve bed will remove all gas species (Q_2, residual Q_2O, CO_2, O_2, N_2) except for He. Thus, regeneration gas will contain other gas species as well as recovered hydrogen isotopes and residual helium gas. The operation cycle is 1 hour.

Palladium Diffuser: Purification process by palladium diffuser is necessary to transfer pure hydrogen isotopes to cryogenic distillation column of the ITER ISS and other exhaust gases to the tritiated waste gas treatment system. The average flow rate of hydrogen isotopes to the ITER ISS is estimated about 0.32-0.4 mol/hour with HT ratio of 100. The average flow rate of exhaust gas to the ITER ERS is estimated less than 0.8 mol/hour.

Heat Exchanger 1: The purge gas is estimated to flow into tritium recovery system at 450 K. Temperature of the purge gas should be cooled to about ambient temperature so that the process works effectively. For this purpose, heat exchanger 1 is necessary. The heat exchanger 1 receive hot purge gas and cool it with coolant water. The coolant water condition is assumed as 50 liter/min in 8 °C.

Heat Exchanger 2: The purge gas is processed by CMSB at LN_2 temperature. Thus, it is necessary to warm up at least to ambient temperature to avoid frosting around the tubing. The shell-and-tube type heat exchanger will be applied to warm up the processed purge gas, at the same time the high temperature outlet gas from the test module will be cooled.

Blower: The blower is used for circulation of the purge gas.

Vacuum Pump: Dryer beds and CMSBs are operated in batch-wise cyclic operation mode. The regeneration operation will be performed by using vacuum pumps compatible with tritium.

Glove Box: Components with high tritium concentration should be contained in the glove boxes. The components are LiOH trap, gas analysis system, dryer bed, regeneration lines (palladium diffuser) of CMSB.

5.4 Interface Condition with Other Systems

Tritium recovery system itself is an interface system between the test module and the ITER tritium plant systems. Thus, it is very important to clarify the interface condition with other systems. The followings are the interface condition with other systems.

Helium purge gas:	0.72-0.90 Nm^3/h (nominal), Q_2 5000 ppm, H/T=100, **from test module** 0.72-0.90 Nm^3/h(nominal), H_2 5000 ppm possibly tritium contaminated, **to test module**
Hydrogen isotopes:	0.32-0.4 mol/h (nominal) , HT ratio 100 **to ITER ISS (cryogenic distillation**)
Effluent gas:	0.8 mol/h(nominal), He with low concentration of Q_2, Q_2O, CQ_4, CO_2, O_2, and so on, **to ITER effluent gas detritiation system**
Tritiated water:	less than 600-760 cc/week, 1.31 Ci/cc tritiated water (The treatment of this should be considered from the view points of tritiated radio active waste water management,

technical possibility of having the ability of water detritiation system for high level tritiated water and so on because of small amount and high tritium concentration.)

Emergency room air , GB atmosphere : to ITER ADS

Liquid nitrogen coolant: 1000 liter(LN_2)/day,
from utility lines to stack after monitoring

Chilled coolant water: 50 liter/min, 8 °C
from utility lines

Service vacuum: for purging out the tubing and components (Necessary pumping speed is TBD. Pressure should be as low as about 0.1 Pa.)
to utility lines

Service inert gas: Helium, Ar, H_2, N_2, etc. for purging out and backfill the tubing and components (Flow rate is TBD.)

The important interface penetration tubings are as follows:

Helium purge gas: 1 inch diameter primary tubing **from test module**
1 inch diameter primary tubing **to test module**

Hydrogen isotopes: 0.5 inch diameter tubing **to ITER ISS (CD)**

Effluent gas: 1 inch diameter tubing **to ITER effluent gas detritiation system**

Tritiated water: 1 inch diameter liquid water tubing in case of on-line continuous treatment

ACS ducts: 40 cm x 60 cm

GB atmosphere: 10 cm diameter

Liquid nitrogen coolant: 2 inch diameter tubing with insulation **from ITER utility and to stack**

Chilled coolant water: 1 inch diameter tubing with thin insulation **from/to the utility coolant**

Service vacuum: 2 inch diameter tubing **to the ITER utility vacuum system**

Service inert gas: 0.5 inch diameter tubings **from the ITER service gas headers**

Pipe size of each interface line is listed up in Table 5-3 as well as the gas condition.

Table 5-3 Interface condition and pipe size

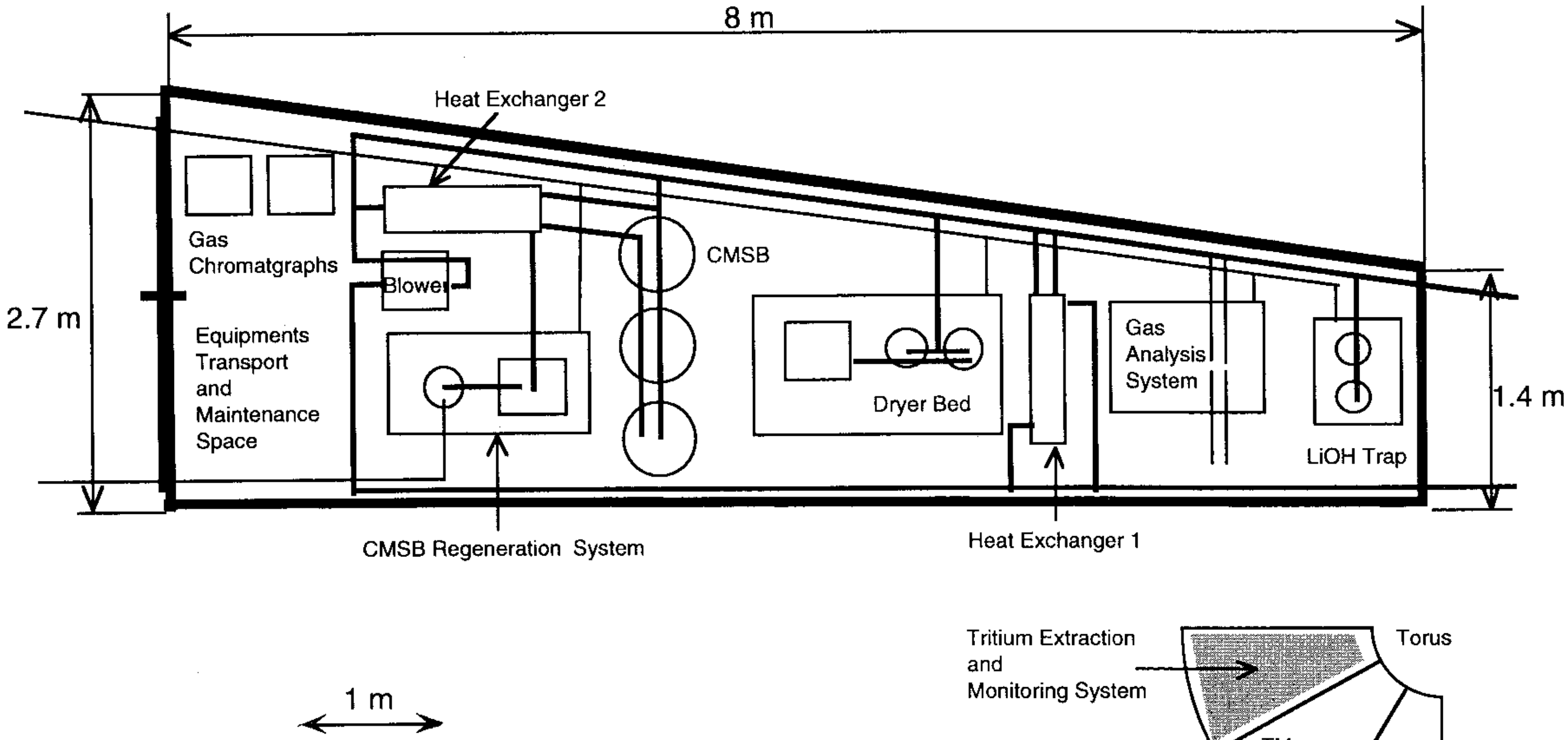
Pipe carrier	Destination	Flow rate and tritium level	Pipe diameter
Helium purge gas	from test module	0.72-0.9 Nm^3/h (nominal) 131 Ci/m^3	25.4 mm
Helium purge gas	to test module	0.72-0.9 Nm^3/h (nominal)	25.4 mm
Hydrogen isotopes	to ITER ISS	7.2-8.96 N liter/h H/T=100 (13100 Ci/m^3)	12.7 mm
Tritiated waste gas	to ITER waste gas treatment system	8 N liter/h 131 Ci/m^3	12.7 mm
Tritiated water	TBD	600-760 cc(liquid)/week 1.31 Ci/cc(liquid)	12.7 mm
Emergency room air	to ITER ADS	TBD*	40 cm x 60 cm
Glove box atmosphere	to ITER ADS	TBD*	50.8 mm
Liquid nitrogen	from utility	TBD*	50.8 mm (insulation tube)
Evaporated nitrogen	to stack	TBD*	50.8 mm
Service vacuum	from utility	TBD*	38.1 mm
Service inert gas	from utility	TBD*	38.1 mm
Coolant water	from and to utility	50 liter (liquid water)/min	50.8 mm (insulation tube)

TBD: To Be Determined

* to be determined after cooling conditions for the additional shield and common frame are agreed among the Parties and JCT

5.5 Layout and Space Requirement

Figures 5-3 and 5-4 show the layout of the tritium recovery system including gas analysis system. This tritium recovery system can be placed in one half of the pit area leaving the space for the test module transporter. The LiOH traps, gas analysis system and regeneration lines of dryer bed and CMSB should be contained in glove boxes, because they process pure LiOT, H_2+HT or HTO, which tend to increase the contamination of the outer surface of components. Some of the components are too heavy and large to be transported in the small room. Therefore, the crane and crane hall is necessary for maintenance and replacement work. The penetration of purge gas line is on the torus side. On the other hand, the other penetration such as the tubing to transfer recovered Q_2 gas or utility lines would be on the floor or the opposite side of the torus.



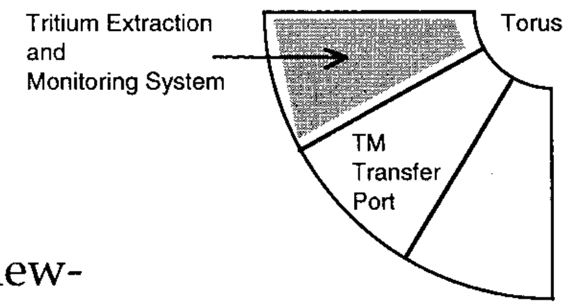


Fig. 5-3 Layout plan -top view-

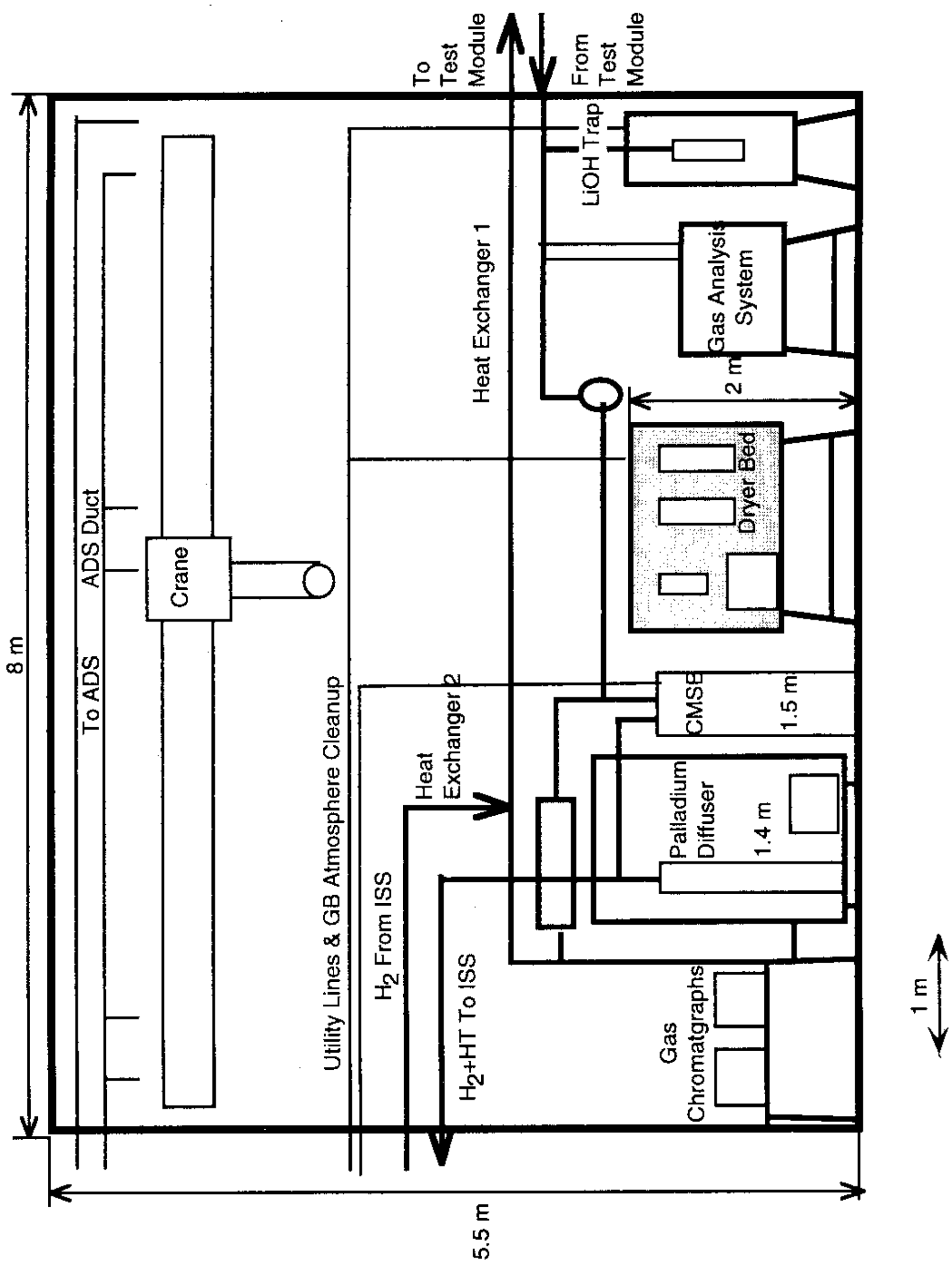


Fig. 5-4 Layout plan -side view-

6 Consideration on Test Module Remote Handling System

All equipment to be used in the horizontal ports should be designed for radial installation and removal of test module components through the port extensions. Since the components will be highly activated after reactor operation, it will be necessary to use remote handling systems for all operations. The design of the remote handling system of the test blanket modules is dependent upon the piping layout within the port extension. One of the requirement s is to minimize the amount of remote operations inside the port extension. A concept to realize this requirement is to combine the test blanket modules, the shielding structure, the coolant pipes and the vacuum vessel door as one assembly. This may allow full functional testing of the assembly prior to installation within the port. As a result, the remote handling system will be adapted to handle this assembly. The overall length of this assembly is estimated to be ~5.5m which is within the allowed limits of the transporter of 8 meters. The weight of the assembly will be in the range of 20-30 tonnes.

Coolant pipes will be routed through the vacuum vessel door, then through the cryostat wall and bioshield wall. The advantage of such layout is to minimize the amount of remote operations required to remove and install the blanket modules and to eliminate remote operations inside the port extension. This will also reduce the amount of time required to remove and install a test module assembly.

The remote handling system for the test module assemblies will take full advantage of the equipment designed for ITER components, such as a shield plug installed in the horizontal maintenance port, to minimize duplication of efforts and to standardize system operations. The transporter will be the standard ITER design with overall dimensions of 8 m long, 3.8 m wide and 5 m high. All operations that are identical to other ITER operations will use the same ITER system to perform, such as removing the shield plugs and the cryostat plugs. However, additional operations that are specific to the test module system will be required

Removing and installing a test module assembly will involve a number of steps. Prior to removing the test module modules, procedures will be established to prepare the modules for removal such as breaking the vacuum in the vessel, draining the test modules, flushing the system of contaminants and purging the system to reduce the amount of residual tritium inside the modules. The next step would be to disconnect the tritium processing system from the test modules and clear the way for the module removal operations to start.

Removing an individual test module within the port will complicates the operation and will increase the time required for removal and installation. On the other hand, removing two or more test modules together as described in above scenario will be harmful to well-kept testing conditions of the test module(s) that is unnecessary to be changed. Further investigation will be required on this point.

The remote handling system for the test blanket modules will consist of a number of components designed to perform certain tasks. A number of the tasks required for the removal and installation of the test modules are identical to ITER tasks, such as shield plug installation, cryostat plug interface and the pipe cutting and welding operation. For those tasks, ITER equipment will be used. Special equipment such as the test module assembly support vehicle, the remote bolting tool and test module assembly transporter will be designed.

The test module remote handling transporter will be based on the standard transporter design as it is developed for ITER basic machine. This transporter should be equipped with special tools to perform a number of specific operations. A manipulator is needed to plug and unplug the power and diagnostics cable bundles without damaging them. This manipulator will serve other tasks by exchanging the end effector tools to fit a specific task. Some of those tools include a fastening tool to handle the vacuum vessel plug bolts. Another tool is needed to cut and weld the lip seal weld of the vacuum vessel plug and the access hatch. Deployment of the temporary tracks between the cryostat door and the vacuum vessel is also handled with this manipulator. Other tools include inspection equipment and possibly viewing equipment such as a camera to perform remote visual inspection.
The transporter is also to be designed to contain the test module assembly. Internal tracks are installed to allow the module support vehicle to travel into and out of this transporter. Room should be provided within the transporter to store the temporary tracks when not in use. Monitoring equipment designed to monitor the status of the test modules during transport to/from the test port should be built into the transporter with capabilities to transmit important or emergency status data to the control room. Emergency recovery operations should be designed and built into this transporter to enable it to recover from certain emergency conditions without interrupting the operation of ITER. Other equipment stored inside the transporter include the remote bolting tool and the blanket support vehicle. Active cooling inside the transporter for the test blanket module may be required depending on the needs of the test module system.

7. Summary

Test modules to be tested in ITER for water-cooled and helium-cooled DEMO blankets with ceramic breeder, which have been developed in Japan, are designed. Their overall mechanical configurations to be consistent with the ITER horizontal ports including their support systems are developed. Major characteristics of the test modules are studied with neutronics, thermo-mechanical and thermal hydraulics analyses resulting in local tritium breeding ratios of 1.2 for the water-cooled and 1.3 for the helium-cooled test modules, sufficient shielding performance for superconducting magnets with additional shield provided behind the test module, sufficient heat removal performance, possible temperature control of ceramic breeder materials and so forth.

Ancillary systems for the test modules, namely a water cooling system, a helium cooling system and tritium recovery system, are also investigated including their system compositions and specifications of major components. The interface conditions of these systems with the ITER system are clarified. Especially, the space requirements for their installation are shown. The cooling systems require large spaces, i.e., 9.5m by 5.8m and 5m high for the water cooling system and 12.5m by 8m and 7.5m high for the helium cooling system. On the other hand, the tritium recovery system requires relatively small space so that it can be installed in the pit around the reactor hall.

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Design studies in more detail on the test modules characteristics including their supporting/fixing systems to the ITER horizontal port and safety aspects will be performed in the future. Also for the ancillary systems, minimization of their required spaces will be tried aiming at their installation, at least their primary loops, into the pit around the reactor hall.

Acknowledgment

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[1] Y. Seki and SSTR Design Team, 13th Int. Conf. on Plasma Phys. and Contr. Nucl. Fusion, Washington, DC, IAEA-CN-53/G-1-2(1990)
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