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JAPANESE CONTRIBUTIONS TO ITER TESTING PROGRAM OF SOLID BREEDER BLANKETS FOR DEMO

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Japanese Contributions to
ITER Testing Program of Solid Breeder Blankets for DEMO

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ITER Conceptual Design Activity (CDA), which has been conducted by four parties (Japan, EC, USA and USSR) since May 1988, has been finished on December 1990 with a great achievement of international design work of the integrated fusion experimental reactor. Numerous issues of physics and technology have been clarified for providing a framework of the next phase of ITER (Engineering Design Activity; EDA).

Establishment of an ITER testing program, which includes technical test issues of neutronics, solid breeder blankets, liquid breeder blankets, plasma facing components, and materials, has been one of the goals of the CDA.

This report describes Japanese proposal for the testing program of DEMO/power reactor blanket development. For two concepts of solid breeder blanket (helium-cooled and water-cooled), identification of technical issues, scheduling of test program, and conceptual design of test modules including required test facilty such as cooling and tritium

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recovery systems have been carried out as the Japanese contribution to the CDA.

Keywords: ITER, Testing, Blanket, Test Module, Solid Breeder, Helium Cooling, Water Cooling, Tritium Recovery System

ITERにおけるDEMO／動力炉用固体増殖材ブランケット工学試験計画

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（1991年3月7日受理）

日本および欧州連合，米国，ソ連の4極により1988年5月から開始された国際熱核融合実験炉（ITER）共同設計作業は，1990年12月概念設計段階（CDA）の諸目的を達成すると共に次の工学設計段階（EDA）の目標，課題を明らかにして終了した。

本報告書は，原型炉（DEMO）／動力炉用機器，材料等の開発・試験を目的としたITERテストポートにおける工学試験（中性子工学試験，固体増殖材および液体金属増殖材ブランケット試験，プラズマ対向機器試験，各種材料試験）のうち，わが国が検討・提案したヘリウム冷却および軽水冷却DEMO／動力炉ブランケットの試験計画およびテストモジュール構造概念，試験設備（冷却系，トリチウム回収系）仕様等についてまとめたものである。

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1. Introduction

ITER has been designed as the first integrated nuclear experimental device to approach the DEMO fusion reactor.

ITER has been designed to operate in two phases. The first phase, which lasts for 6 years, is devoted to machine checkout and physics testing. The second phase lasts for 8 years and is devoted to technology testing. The available test ports are three in the first phase and five in the second phase. The accumulated neutron fluence on the test ports in each phase are 0.08 and 1.53 MWa/m^2, respectively. The major test items during the two phases are neutronics tests, liquid metal blanket tests, solid breeder blanket tests, plasma facing component tests, and material tests.

Test issues of the neutronics are: 1) demonstration of tritium self-sufficiency of the various blanket concepts, 2) verification of neutron transport codes and nuclear data predicting tritium production rate, heating rate, and activation, 3) confirmation of shield design to account for streaming and penetration through slits/gaps such as NBI and main exhaust ducts.

Test issues of the liquid metal blankets (self-cooled and water-cooled types) are: 1) tritium self-sufficiency, 2) MHD effects, 3) heat transfer, 4) materials interactions (e.g., corrosion), 5) structural response in a fusion environment, 6) tritium recovery and control, and 7) components and system interactions.

For the solid breeder blankets (helium-cooled and water-cooled types), major issues are closely related to component materials such as ceramic breeders, beryllium multiplier, structure materials and coolant. Behavior of an integrated component of each blanket concept will be demonstrated in a fusion environment (high neutron fluence and temperatures, stationary and cycling thermal stresses and other stresses).

Test issues for the plasma facing components are categorized into two types. They are: 1) plasma related short-term tests to determine impurity buildup, plasma edge characteristics, plasma characteristics at the divertor, and vacuum system pumping capability, and 2) engineering related issues similar to those conducted on blankets, such as thermal-hydraulics performance, thermo-mechanical behavior etc.

The goals of ITER material testing are 1) to validate irradiation results obtained by fission reactor and other simulation device, and 2) to accumulate irradiation data on material properties. The goal fluence of ITER (1 - 3 MWa/m^2) is very limited to provide a complete assessment of materials (especially structural materials) behavior under irradiation in the DEMO reactor. However, numerous important fusion environmental data can be obtained for various fusion materials.

As the Japanese design contribution in the CDA, authors have proposed test program of helium-cooled and water-cooled concepts of solid breeder blanket.

The Japanese proposal includes conceptual design of test modules of each concept, identifications of test issues and test schedules and design of test facility such as cooling system and tritium recovery system as well as their configuration in the ITER building.

2. Test Program

2.1 Helium-Cooled Blankets

2.1.1 DEMO Blanket Concepts

The major design parameters and performance characteristics of candidate blankets for DEMO/power reactors (Fusion power: ~ 3 GW) studied at JAERI are summarized in Table 2.1.1[1,2]. Various kinds of blankets for a DEMO reactor including liquid metal breeder have been studied.

One of the advantages of a helium-cooled blanket is the possibility to obtain high thermal efficiency. From this point, the outlet temperature of helium coolant is selected to be 700°C in the above design studies. As for structural material, molybdenum alloy is selected to allow the use of high temperature helium coolant. For the Mo-alloy/Li_2O/He/Be blanket, the BOT (Breeder Out-of-Tube) concept is adopted as shown in Fig. 2.1.1 and Fig. 2.1.2. The former has a layer of beryllium multiplier which is separated from a breeder region. The latter is a beryllium/breeder mixture type.

Breeder (Li_2O) is fabricated in the form of small-spherical pebbles (< 1 mm^{ϕ}) to avoid thermal cracking of breeder. The breeder temperature is maintained between 450°C and 950°C to recover tritium efficiently in the low pressure helium purge stream. There is no thermal resistant layer around the coolant tubes because the inlet helium temperature is 400°C, thus the breeder temperature is naturally kept above 400°C.

The helium coolant pressure is selected to be 9 MPa in terms of sufficient heat removal capability. To keep the coolant pressure drop acceptably low, coolant channels and manifolds in the blanket are poloidally divided into two or three subunits.

The major parameters of the energy conversion system are also summarized in Table 2.1.1. A turbine system of the latest (oil-burning) power plant is employed for the helium coolant system, and gross thermal efficiency is estimated to be 47.2%. Though pumping power of the helium coolant is about five times as large as those of the water and liquid metal coolants, the helium coolant still gives a larger net electric power (i.e., subtracting the pumping power from the gross electric power) than the other two coolants.

2.1.2 Technical Issues for DEMO Blanket

Major technical issues for the high temperature helium-cooled blanket are listed below.

(1) Development of refractory alloys and material data base

Development of a new refractory alloy (e.g., Mo alloy) or substantial improvement of the existing alloys (e.g., ferritic steels) is necessary. Material data base including following items must be completed.

- Physical/chemical properties
- Mechanical properties
- Irradiation-induced property changes
- Compatibility
- Radioactivity
- Mass production and fabrication (welding, etc.)

(2) Arrangement of coolant tubes in blanket

- Fabrication and assembling accuracy of coolant tubes must be considered because temperature control of breeder is realized by the tube arrangement.

(3) Mass production of small size breeder and/or multiplier pebbles

- small size (< 1 mm^{ϕ})
- Low production cost

(4) Packing of breeder and/or multiplier pebbles

- Method for uniform and high density packing
- Non-destructive investigation of breeding zone

(5) Thermal stress on structures

- Careful design to absorb large thermal expansion is necessary because of large temperature difference of coolant.

(6) Tritium permeation through coolant pipe

- From the safety point of view, reduction method of permeation of tritium into/from the coolant through high temperature pipe wall must be considered (e.g., surface coating of coolant tubes in blanket and adoption of double-walled pipe for high temperature outlet pipe etc.)

(7) Radiation streaming through large-size coolant pipe

- Reduction method (e.g., shield design and pipe arrangement) must be considered.

(8) Thermo-mechanical design of high heat flux components

- Feasibility of helium-cooled divertor/limiter should be investigated to avoid multiple coolant materials (e.g., helium for blanket and water for divertor).

Among above issues, (2) to (4) are the same as those for the ITER driver blanket. Therefore, they might be solved entirely or partially in the driver blanket development and must be assured by the start of the DEMO design through the design, fabrication, and operation of the ITER blanket.

2.1.3 Strategy/Approaches

From the beginning of experiments with D-T burning in the Physics Phase, tests of blankets for a DEMO/power reactor will be performed. Though a neutron fluence and a plasma burn time are very limited in the Physics Phase, tests to characterize the neutronic environments, to check out blanket systems, and to check out instrumentation can be conducted. It is proposed that during this phase one horizontal port be allocated to solid breeder blankets (for both of helium-cooled and water-cooled concepts). Submodules described below would be used for these tests.

During the Technology Phase, two horizontal ports would be allocated to solid breeder blankets: one for helium-cooled and another for water-cooled concepts. The Technology Phase is planned to continue for eight years. During the first four years of the Technology Phase, it is envisaged that the test port for helium-cooled solid breeder blankets would be divided into 3-4 submodules which be tested in parallel. This would allow for testing of several design approaches and for a first screening of a number of concepts. To start with screening tests with relatively low neutron fluence, multiple effect tests including investigation of irradiation effects on breeder/multiplier properties would be followed.

For availability reasons, the tests of the submodules in the Physics Phase and the early period of the Technology Phase will be conducted without a direct exposure to the plasma by placing test modules behind a separate water-cooled first wall similar to the first wall of the driver blanket.

During the next three years of the Technology Phase, blanket performance tests using modules of full port size will be conducted. It is intended to reduce the number of concepts to be tested by eliminating and combining different designs in the former stage of the Technology Phase. It is desirable to achieve agreement on a single reference design, which will be tested for the next three years, otherwise three or four modules of different blanket concepts will be tested successively, such as for about one year each.

For one or two designs, it may be necessary to perform tests with complete segments or even sectors in the final period of the Technology Phase in order to provide confidence in the reliability of the blankets to be installed in a DEMO/power reactor.

The concept of the port division is described in Fig. 2.1.3. A scenario of neutron wall load and accumulated neutron fluence and a schedule for the testing are shown in Table 2.1.2 and Fig. 2.1.4, respectively.

Some of the small tests, such as tritium release measurements from solid breeder materials and others, will be performed in the port allocated for materials testing.

Because the test space and time are strictly limited in ITER, out-of-ITER tests (out-of-pile as well as in fission reactors) are essential to develop blankets for a DEMO/power reactor. These tests will reduce the number of concepts that need to be tested in ITER.

2.1.4 Test Description

According to the strategy in Section 2.1.3, tests required in ITER are as follows:

① Neutronics test and system check-out
② Screening test
③ Multiple effect test
④ Performance test
⑤ Segment test
⑥ Material irradiation test.

Major aspects of their tests are summarized in Table 2.1.3.

A neutronic test for blankets including system/instrumentation check-out will be performed with very low neutron fluence during the Physics Phase. Submodules for 5-10 candidate blanket concepts will be tested in series using one-fourth (or one-third) region of the test port allocated to solid breeder blankets. Distributions of nuclear heating rate and tritium generation rate, γ-ray and induced activity should be measured in order to characterize neutronic aspects of the candidates.

Tests during the Technology Phase start with a screening test. In this test, tritium recovery performance based on tritium release from the breeder and breeder temperature control will be mainly investigated.

The number of submodules to be tested will be 2-6 depending on the results of the neutronic test. Submodules will be installed also in one-fourth (or one-third) region of the port allocated to helium-cooled solid breeder blanket. Neutron fluence required for this test is relatively low ($\leq$ 0.01-0.02 MWa/m^2), therefore the test will be performed in one year.

Following the screening test, multiple effect test will be conducted. Main objective of this test is to investigate neutron irradiation effects on tritium recovery performance in terms of property changes of breeder and multiplier materials. One or two concepts chosen from the screening test will be tested with the same submodule used in the screening test. Indication of irradiation damage (sintering, cracking, grain growth etc.) would be also obtained. Neutron fluences of >0.2 MWa/m^2, therefore 1-2 years of reactor operation based on the scenario in Table 2.1.2 will be required for this test.

In the performance test, almost all of basic characteristics of blankets, such as heat generation and removal, tritium generation and recovery, and some of thermo-mechanical performances, will be demonstrated. The size of modules will be changed to full port size, and 1-3 blanket concepts will be selected from the results of above series of submodule tests. Test modules will directly face the plasma for the first time in the testing. (In above tests, test modules would not be directly exposed to the plasma.) More than 0.2 MWa/m^2 of neutron fluences will be required in order to investigate irradiation effects on breeder/multiplier properties as in the multiple effect test. According to the neutron fluence scenario in Table 2.1.2, this test will be sufficiently performed in the next three years of the Technology Phase when the number of the candidates is reduced to 1-3.

It is desirable to perform a segment test with one complete segment or even one sector in order to demonstrate accommodative performance of blankets to poloidal wall load distribution and thermo-mechanical integrity of large blanket structure.

Material irradiation test including the investigation of tritium release characteristics of breeder materials is performed in the port allocated for materials testing.

2.1.5 Specification and Characteristics of Test Articles

A test article considered here is a full port size module (first wall area; ~3.5 m^2) which will be tested at the performance test phase. The module will be directly exposed to plasma.

Major test items of the helium-cooled test module are as follows:

① Verification of thermal-hydraulic performance of cooling mechanism

② Verification of integrity of blanket structure under normal and off-normal condition

③ Verification of neutronics performances (tritium production rate and nuclear heating rate etc.)

④ Demonstration of coolant conditions expected in a DEMO/power reactor, i.e. coolant outlet temperature > 450°C, as high as possible

⑤ Demonstration of in-situ tritium recovery under the DEMO coolant conditions

(1) Consideration on structural material

In order to obtain high thermal efficiency, one of the most important design parameters of the helium-cooled blankets is outlet coolant temperature from the blanket. The outlet temperature depends on structural material of the blanket. Rough estimates of allowable outlet helium temperature for various structural materials are shown in Table 2.1.4.

Refractory metal alloys, such as molybdenum alloys, have the most attractive feature in terms of high coolant temperature, and they were selected as one of the candidate structural materials as described in Section 2.1.1. Data base of these materials, however, is not sufficient for the present design of test modules in ITER. Therefore, ferritic/martensitic steels or nickel-based alloys will be considered as structural material for the test modules.

Nickel-based alloys which are used in a high temperature gas-cooled fission reactor (HTGR) could not be applied to a fusion DEMO/power reactor because of their irradiation damages (i.e., helium embrittlement and radioactivity). They might be applied, however, to test modules in ITER because neutron fluence during the testing is low. Physical and mechanical properties are seemed to be good enough to show advantages of helium-cooled blankets.

Ferritic/martensitic steels will also supply the basic understanding of the performance of helium-cooled blankets though outlet temperature of the coolant may not be high enough to demonstrate attractive features of helium-cooled blankets.

During the early period of the testing in ITEr, ferritic/martensitic steels or nickel-based alloys will be used as structural material of test modules. In the later stage of the testing, they might be replaced by newly-developed refractory metal alloys which prospect for a DEMO/power reactor.

(2) Test module concept

A test module design with ferritic/martensitic steel as the structural material is presented here. Selection of structural material and blanket concept would be changed by progress of material development and out-of-ITER R&D.

The BOT (Breeder Out-of-Tube) concept with beryllium/breeder mixture zone has been adopted for the one of helium-cooled test modules as shown in Fig. 2.1.5 and Fig. 2.1.6. Breeder (Li_2O) is fabricated in the form of small spherical pebbles (< 1 mm^{ϕ}). The breeder temperature is maintained between 450°C and 950°C to recover tritium efficiently in the low pressure helium purge stream. In order to eliminate thermal resistant layer around the coolant tubes, the inlet helium temperature of the breeding zone is set to be above 400°C to keep the breeder temperature above 400°C. On the other hand, the inlet temperature of the first wall coolant is set to be below 400°C to keep the first wall temperature below the maximum allowable temperature for ferritic steel (around 500°C). Therefore, the following coolant flow scheme is selected; the coolant flows in series through the module box structure (including the first wall) and the breeding zone. The helium coolant pressure is selected to be 9 MPa similar to the DEMO blanket.

Major characteristics of the test module is summarized in Table 2.1.5.

References

(1) TONE, T., et al., Technical Evaluation of Major Candidate Blanket Systems for Fusion Power Reactor, JAERI-M 87-017 (in Japanese) (1987)

(2) KURODA, T., et al., Technical Considerations on Breeding Blanket Systems for a Fusion Power Reactor, Fusion Engineering and Design 8 (1989) 219-226

Table 2.1.1 Major design features of candidate blankets for DEMO[2)]

Blanket concept	PCA/Li_2O/H_2O/Be	Mo-alloy/Li_2O/He/Be	Mo-alloy/$LiAlO_2$/He/Be	V-alloy/Li/Li/none	Mo-alloy/Li/He/none
Thermal power/gross electric power (MW)	3820/1320	3720/1810	3510/1710	3670/1450	3500/1700
Neutron wall loading (MW/m^2)	3.3	3.3	3.3	3.3	3.3
Heat flux on the first wall (MW/m^2)	0.9	0.9	0.9	0.9	0.9
Nuclear heating in the first wall (MW/m^2)	43	27	27	23	20
Structural material	Ti-modified SS	Mo-alloy	Mo-alloy	V-alloy	Mo-alloy
Maximum temperature of structural material (°C)	423	852	852	730	857
Breeder/neutron multiplier	Li_2O/Be	Li_2O/Be	$LiAlO_2$/Be	Li/none	Li/none
^6Li enrichment (%)	30	30	95	no	no
Breeder configuration	outside tube	outside tube	inside tube	self-cooling	outside tube
Temperature range of breeder (°C)	450 ~ 950	400 ~ 950	400 ~ 725	350 ~ 530	400 ~ 860
Tritium recovery	He purge stream	He purge stream	He coolant	Li coolant	Li extraction
Outboard/inboard local breeding ratio	1.20/1.13	1.37/1.32	1.24/1.17	1.39/1.28	1.36/1.22
Net breeding ratio	1.03	1.19	1.06	1.15	1.14
Coolant	H_2O	He	He	Li	He
Pressure (MPa)	15.5	9	9	1.5	9
Inlet/outlet temperature (°C)	280/320	400/700	400/700	350/530	400/700
Flow direction: first wall/breeder region	toroidal/poloidal	toroidal/poloidal	toroidal/toroidal	toroidal/poloidal	toroidal/poloidal
Maximum velocity: first wall/breeder region (m/s)	7.9/7.1	63/52	63/8.1	0.3/0.25	56/60
Pressure drop (MPa)	0.37	0.11	0.11	1.0*	0.10
Number of primary cooling loops	4	7	7	4	7
Pumping power (MW)	26	109	103	30	102
Steam pressure/temperature (MPa/°C)	6.5/280	24.6/538	24.6/538		
Gross thermal efficiency (%)	34.4	47.2	47.2	39.2	47.2

* Inboard first wall channel is critical (0.3 m/s, B_T=8T).

Table 2.1.2 Neutron wall load and accumulated fluence scenario

Year	Plasma	Neutron Wall Load		Pulses/year	Pulse Length	Availability	Accumulated Fluence	
		Average (MW/m^2)	on Port (MW/m^2)		(sec)	× Duty Cycle	Average (MWa/m^2)	on Port (MWa/m^2)
Physics Phase								
1	H,D	0	0				0	0
2	H,D	0	0				0	0
3	D/He^3	0	0				0	0
4	D/T	1.0	1.6		≤400		0.01	0.02
5	D/T	1.0	1.6		≤400		0.03	0.05
6	D/T	1.0	1.6		≤400		0.05	0.08
Technology Phase								
7	D/T	0.8	1.2	1000	2250	0.07	0.11	0.17
8	D/T	0.8	1.2	1000	2250	0.07	0.17	0.25
9	D/T	0.8	1.2	2500	2250	0.18	0.31	0.47
10	D/T	0.8	1.2	2500	2250	0.18	0.45	0.68
11	D/T	0.8	1.2	2500	2250	0.18	0.59	0.89
12	D/T	0.8	1.2	2500	2250	0.18	0.73	1.10
13	D/T	0.8	1.2	2500	2250	0.18	0.88	1.32
14	D/T	0.8	1.2	2500	2250	0.18	1.02	1.53
Possible Extended Phase								
15-?	D/T	0.8	1.2				3.0	4.5

Table 2.1.3 Test items for helium-cooled solid breeder blanket

No.1

Test name	: Neutronics test and system check-out
Blanket type	: Helium-cooled solid breeder blanket
Description of measurements	: Nuclear heating rate distribution, tritium generation rate distribution, neutron and γ-ray flux, induced activity
Size	: 0.4 m^w × 0.5-1 m^h × 0.5 m^t (submodule)
Number of test elements in 1/4 port	: 1 - 2
Location	: 0 - 50 cm from first wall
Boundary condition	: No-plasma exposure
Support equipment	: Cooling system
Operation time	
Duration of individual test	: a few short pulses × ~10 iterations
Number of tests in series	: ~6
Total reactor time	: < 0.001 MWa/m^2, < 2-3 years
Special requirements	:
Remarks	: to be tested during the Physics Phase

No.2

Test name	: Screening test (mainly with regard to tritium recovery performance)
Blanket type	: Helium-cooled solid breeder blanket
Description of measurements	: Tritium breeding, tritium release rate, tritium inventory, breeder temperature control
Size	: 0.4 m^w × 0.5-1 m^h × 0.15-0.5 m^t (submodule)
Number of test elements in 1/4 port	: 1 - 2
Location	: 0 - 50 cm from first wall
Boundary condition	: No-plasma exposure
Support equipment	: Tritium recovery system, tritium measurement system, cooling system
Operation time	
Duration of individual test	: 1000-10000 sec × 5-10 iterations
Number of tests in series	: 2 - 3
Total reactor time	: ≤ 0.01-0.02 MWa/m^2, < 1-2 years
Special requirements	: Total burn time ≥ 4-5 days is required for saturation of tritium inventory. Continuous operation time ≥ 1000-10000 sec and dwell time ≤ 300-400 sec are required for breeder temperature control.

Table 2.1.3 (Continued)

No.3

Test name	: Multiple effect test (including irradiation effects on breeder/multiplier properties)
Blanket type	: Helium-cooled solid breeder blanket
Description of measurements	: Tritium release rate and inventory, effects of property changes due to irradiation on tritium release and temperature control (thermal conductivity, compatibility, breeding and tritium release properties), Indication of irradiation damage (sintering, cracking, grain growth etc.)
Size	: 0.4 m^w × 0.5-1 m^h × 0.15-0.5 m^t (submodule)
Number of test elements in 1/4 port	: 1 - 2
Location	: 0 - 50 cm from first wall
Boundary condition	: No-plasma exposure
Support equipment	: Tritium recovery system, tritium measurement system, Cooling system, post irradiation test system
Operation time	
Duration of individual test	:
Number of tests in series	: 1 - 2
Total reactor time	: ≥ 0.2 MWa/m^2, ~ two years
Special requirements	: The same operation mode as tritium recovery test (No.2) is required.
Remarks	: Sequential to the screening test using submodules chosen from the results of the screening test

No.4

Test name	: Performance test
Blanket type	: Helium-cooled solid breeder blanket
Description of measurements	: Heat generation and removal (outlet coolant temperature for electricity generation), Continuous tritium recovery, tritium breeding, tritium inventory, thermal-hydraulics and thermo-mechanical performances, irradiation effects on materials
Size	: 1 m^w × 3 m^h × 0.5 m^t (full port size module)
Number of test modules in one port	: 1
Location	: 0 - 50 cm from first wall
Boundary condition	: With plasma exposure
Support equipment	: Tritium recovery system, Cooling system, (electricity generation system), post irradiation test system
Operation time	
Duration of individual test	: ≥ 1000-10000 sec × operation cycles (TBD)
Number of tests in series	: 1 - 2
Total reactor time	: ≥ 0.2 MWa/m^2, ≥ 1-2 years
Special requirements	: Burn time ≥ 800 sec (for dwell time = 100 sec) or burn time ≥ 1000 sec (for dwell time = 200 sec) is required to obtain thermal steady state (Indication of electric power generation).

Table 2.1.3 (Continued)

No.5

Test name	: Segment test (demonstration of DEMO act-like blanket)
Blanket type	: Helium-cooled solid breeder blanket
Description of measurements	: Heat generation and removal (outlet coolant temperature for electricity generation) continuous tritium recovery, tritium breeding, tritium inventory, thermal-hydraulics and thermo-mechanical performances
Size	: 1 m^w × (full height of driver blanket) × 0.5 m^t
Number of test segments in one sector	: 1 (1/48 torus) or 3 (1/16 torus = 1 sector)
Location	: 0 - 50 cm from first wall
Boundary condition	: With plasma exposure
Support equipment	: Tritium recovery system, Cooling system, (electricity generation system), post irradiation test system
Operation time	
Duration of individual test	: ≥ 1000-10000 sec × operation cycles (TBD)
Number of tests in series	: 1 - 2
Total reactor time	: TBD
Special requirements	: Burn time ≥ 800 sec (for dwell time = 100 sec) or burn time ≥ 1000 sec (for dwell time = 200 sec) is required to obtain thermal steady state (Indication of electric power generation).
Remarks	: To be conducted at the final stage of the Technology Phase

No.6

Test name	: Material irradiation tests
Blanket type	: Not specified
Description of measurements	: Irradiation damage (swelling, compatibility etc.) and its effects on physical and mechanical properties of structural material, shell conductor, insulator, breeder and so forth
Size	: Total plasma facing surface area of 1.5 m^2 (small articles)
Number of test elements in one port	: TBD
Location	: 0 - 50 cm from first wall
Boundary condition	: No-plasma exposure and with plasma exposure
Support equipment	: Cooling system, post irradiation test system
Operation time	
Duration of individual test	: TBD
Number of tests in series	: TBD
Total reactor time	: TBD (≥ 3 MWa/m^2)
Special requirements	:
Remarks	: to be performed in the port allocated for material testing

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Table 2.1.4 Rough estimates of achievable helium outlet temperature under DEMO conditions (average neutron wall load: ∿3 MW/m^2, average surface heat load: ∿1 MW/m^2)

	Maximum allowable temperature of structure (°C)	Maximum outlet temperature of coolant (°C)
① Austenitic steels (PCA etc.)	∿450°C	∿430°C (∿200°C)
② Ferritic/martensitic steels (HT9 etc.)	∿500°C	∿480°C (∿250°C)
③ Nickel-based alloys (Inconel, Hastelloy)	∿700°C*	∿680°C* (∿450°C*)
④ Molybdenum alloys (TZM, Mo-Re alloy etc.)	∿950°C*	∿880°C* (∿700°C*)

* Tentative values

( ) Coolant temperature for first wall

Table 2.1.5 Major design parameters and characteristics of helium-cooled test module

Location	Outboard, midplane
Module type	BOT (Breeder Out-of-Tube)
Module size	
First wall area	3.5 m^2 (3.4 m^h × 1.1 m^w)
Thickness	0.6 m
Neutron wall load	1.2 MW/m^2
Surface heat load	0.15 MW/m^2
Coolant	Helium
First wall (module box)	
Temperature (in/out)	360/400°C
Pressure	9 MPa
Breeding zone	
Temperature (in/out)	400/480°C
Pressure	9 MPa
Structural material	Ferritic/martensitic steel
Allowable temperature	< 500°C
Breeder	Lithium oxide: 1mm ϕ pebble
^6Li enrichment	natural (mixture type)
	30% (separate type)
Temperature control	Cooling tube arrangement
Operating temperature	450 - 600°C
Neutron multiplier	Beryllium:
	1mm ϕ pebble (mixture type)
	block (separate type)
Operating temperature	450 - 600°C (mixture type)
	< 500°C (separate type)
Local tritium breeding ratio	~1.6 (mixture type)
(1-D poloidal model)	~1.3 (separate type)
Heat deposition	
Module box (first wall)	1.73 MW
Breeding zone	4.20 MW
Tritium recovery	In-situ and continuous with helium purge gas
Tritium inventory	TBD

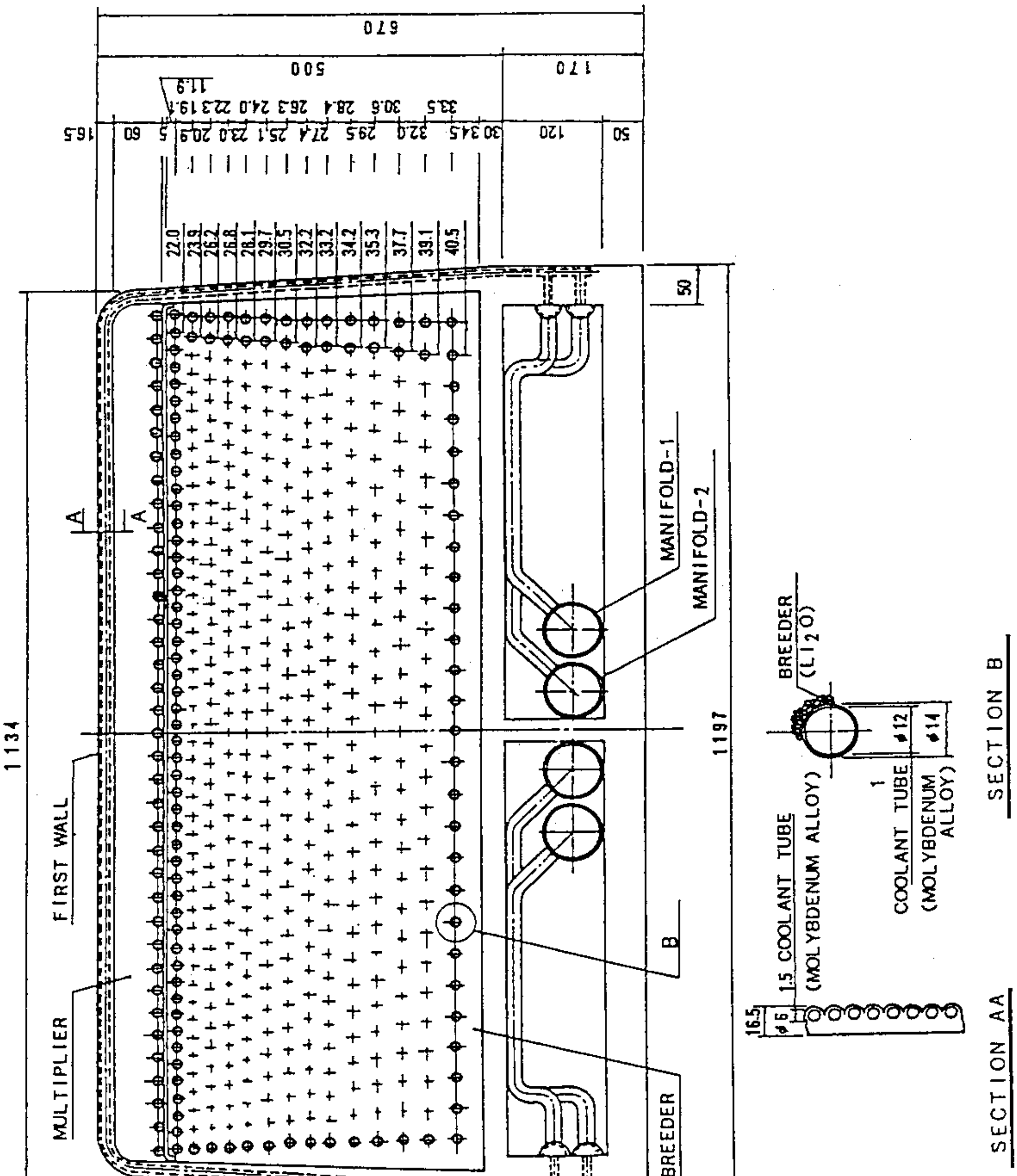


Fig. 2.1.1 Cross-sectional view of Mo-alloy/Li_2O/He/Be blanket for DEMO reactor (separate multiplier zone type)

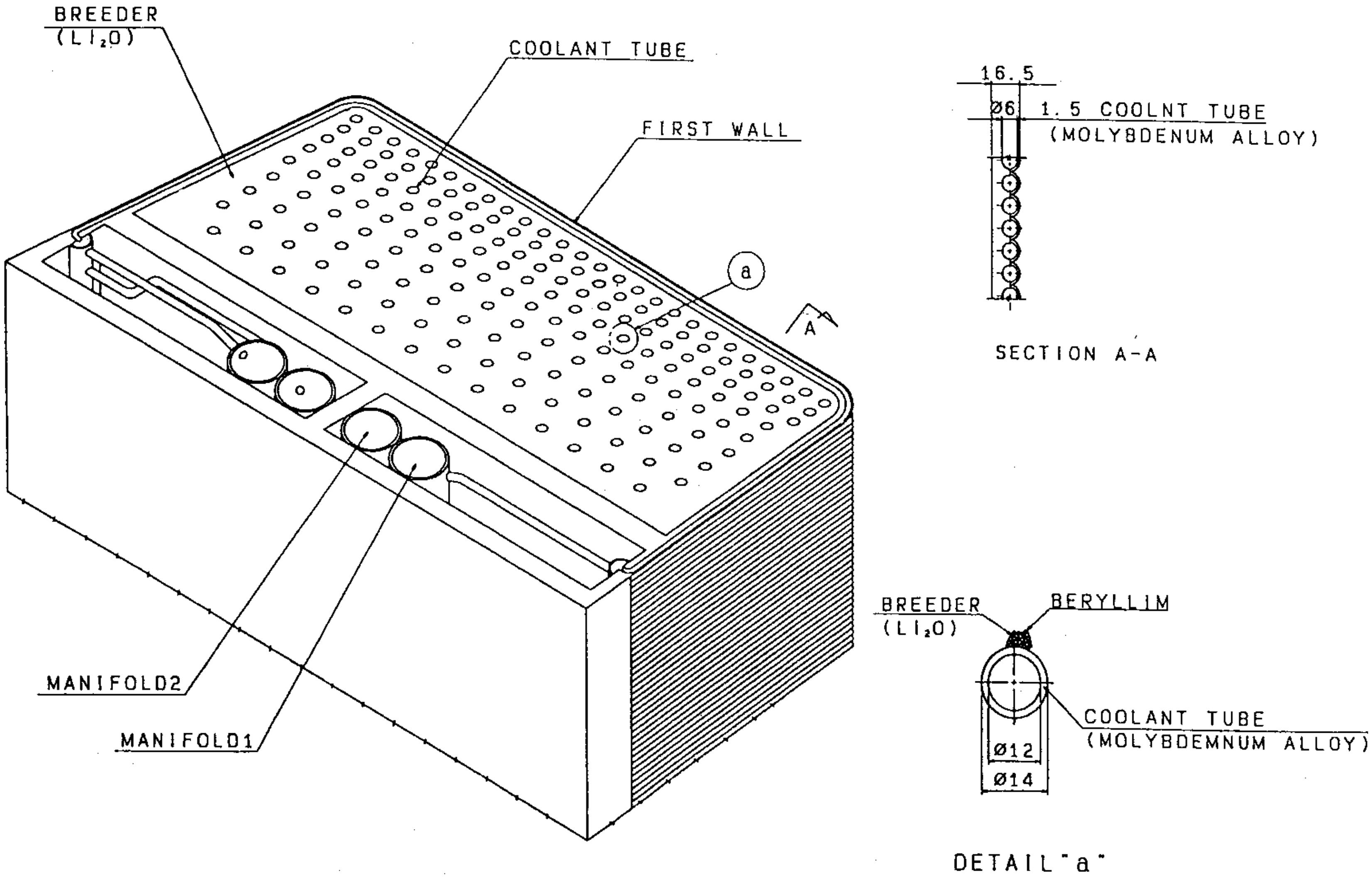


Fig. 2.1.2 Schematic view of Mo-alloy/Li_2O/He/Be blanket for DEMO reactor (breeder/multiplier mixture type)

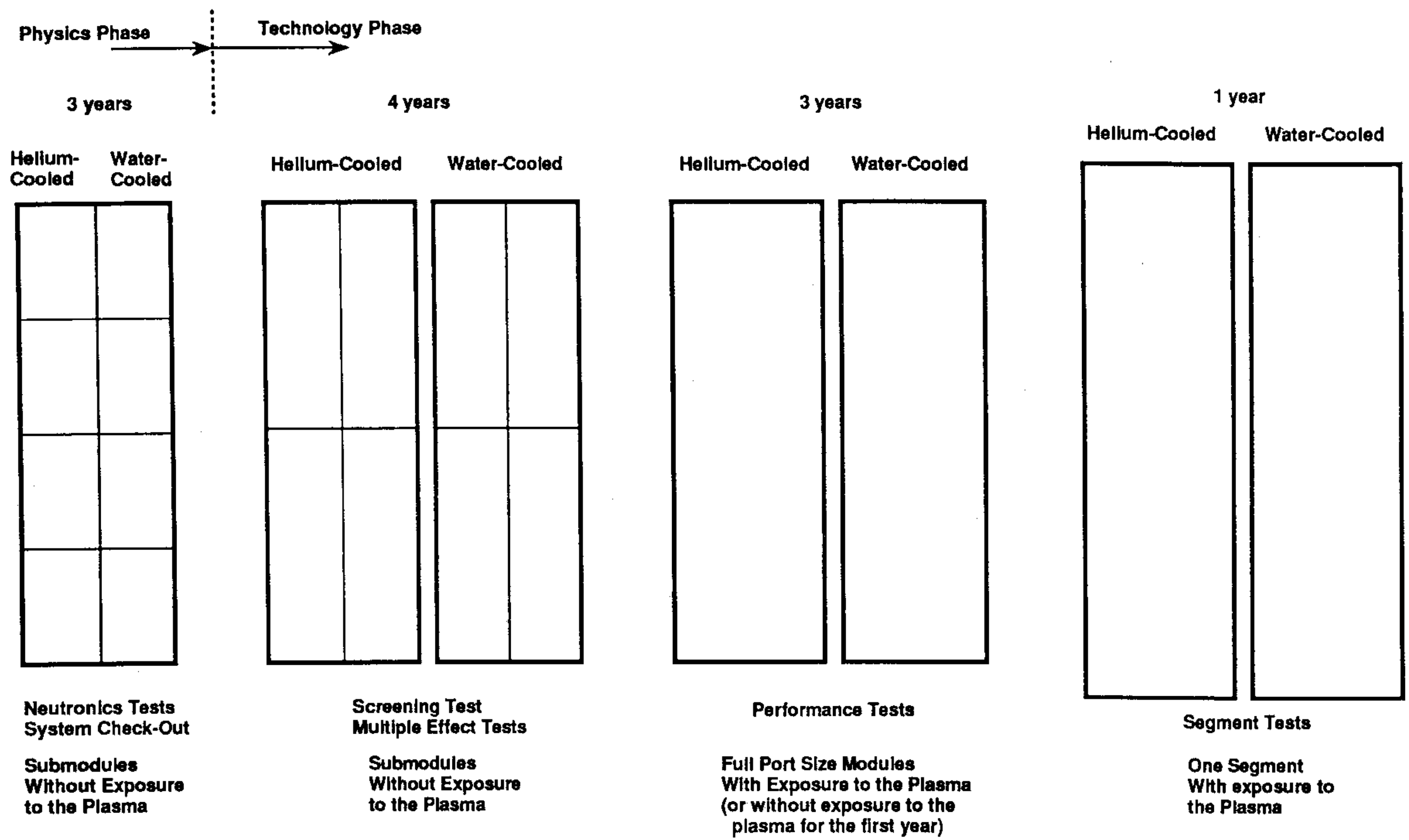


Fig. 2.1.3 Test port allocation to helium- and water-cooled solid breeder blankets

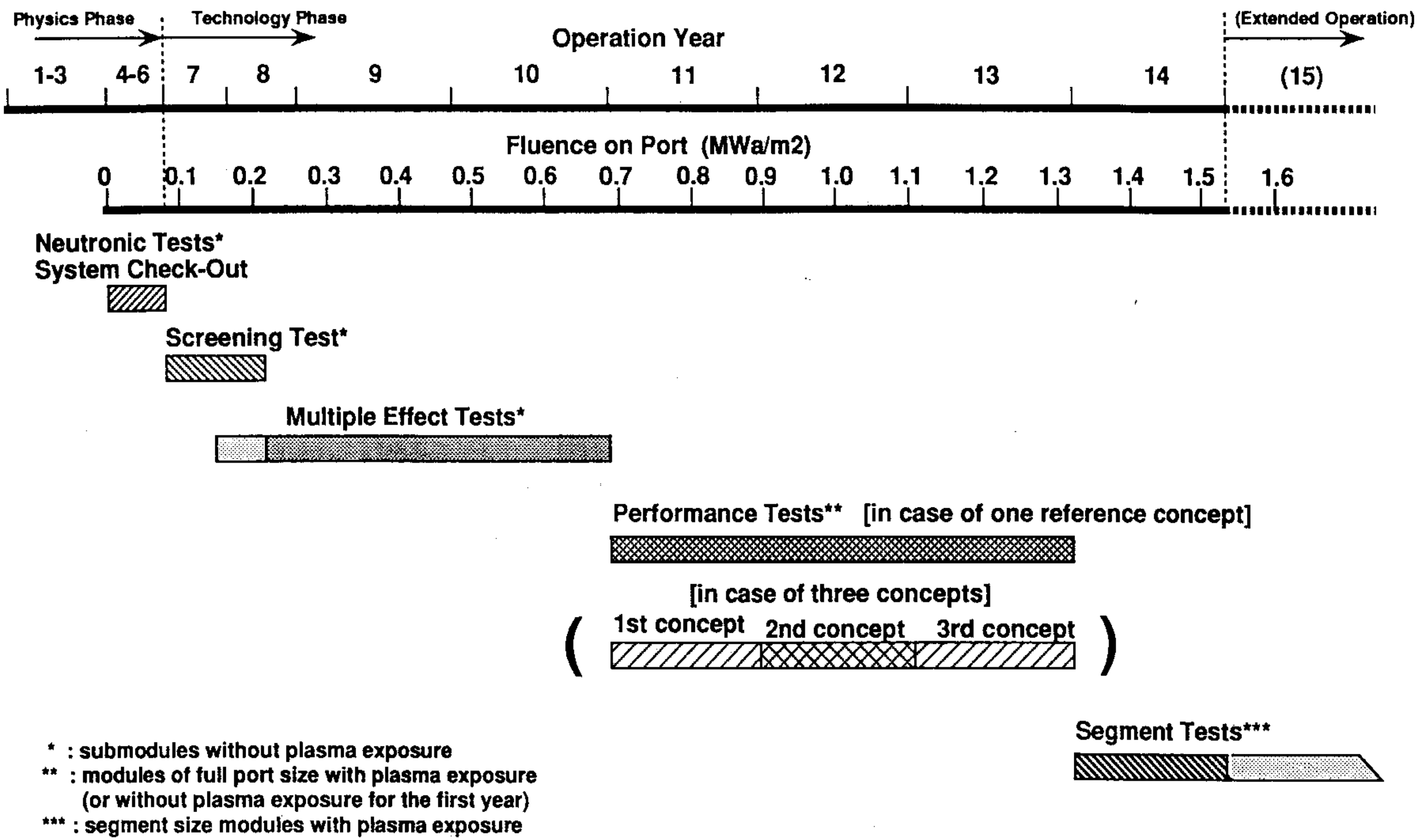


Fig. 2.1.4 Testing schedule for helium-cooled solid breeder blankets

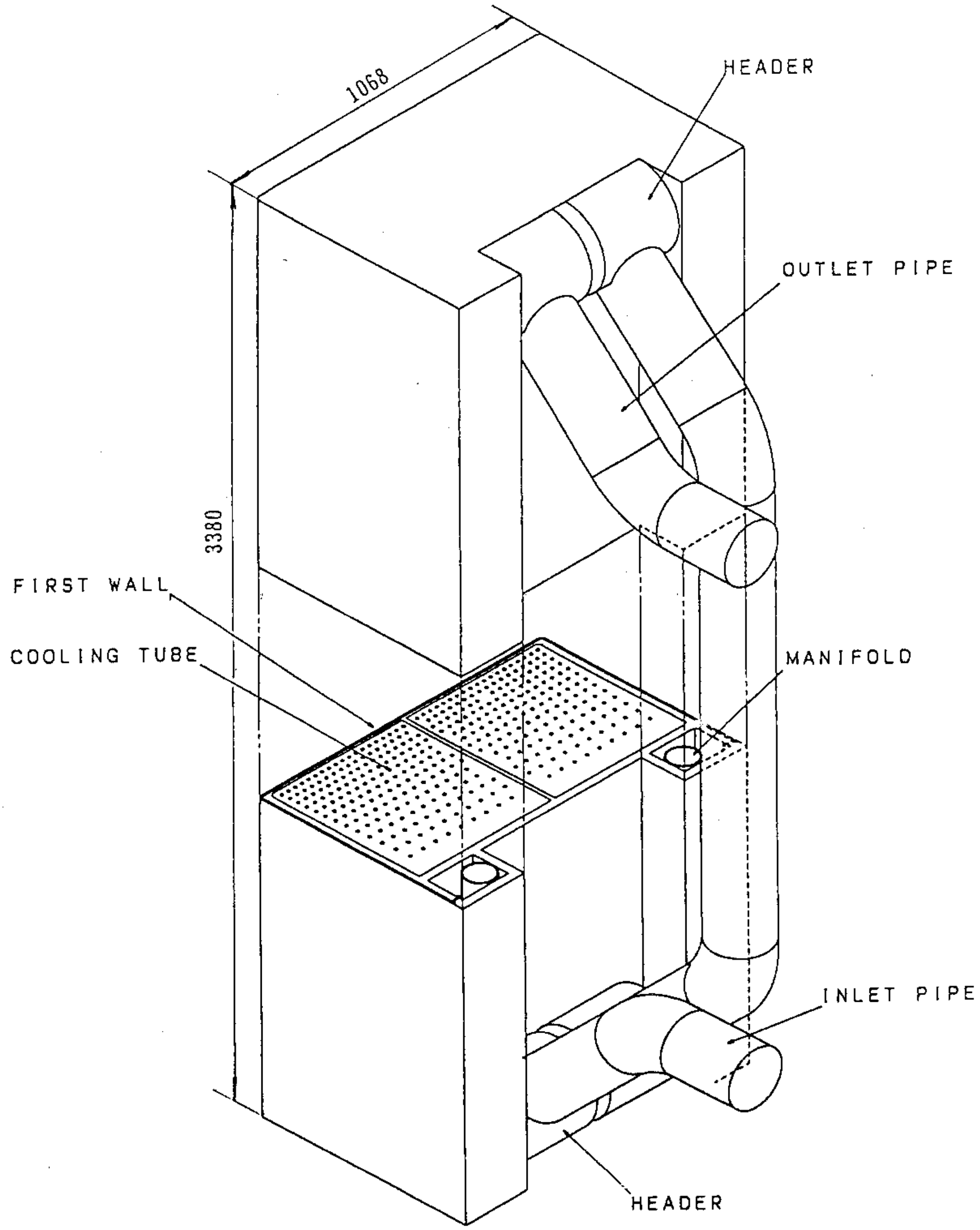


Fig. 2.1.5 Schematic view of helium-cooled test modulue for ITER testing (breeder/multiplier mixture type)

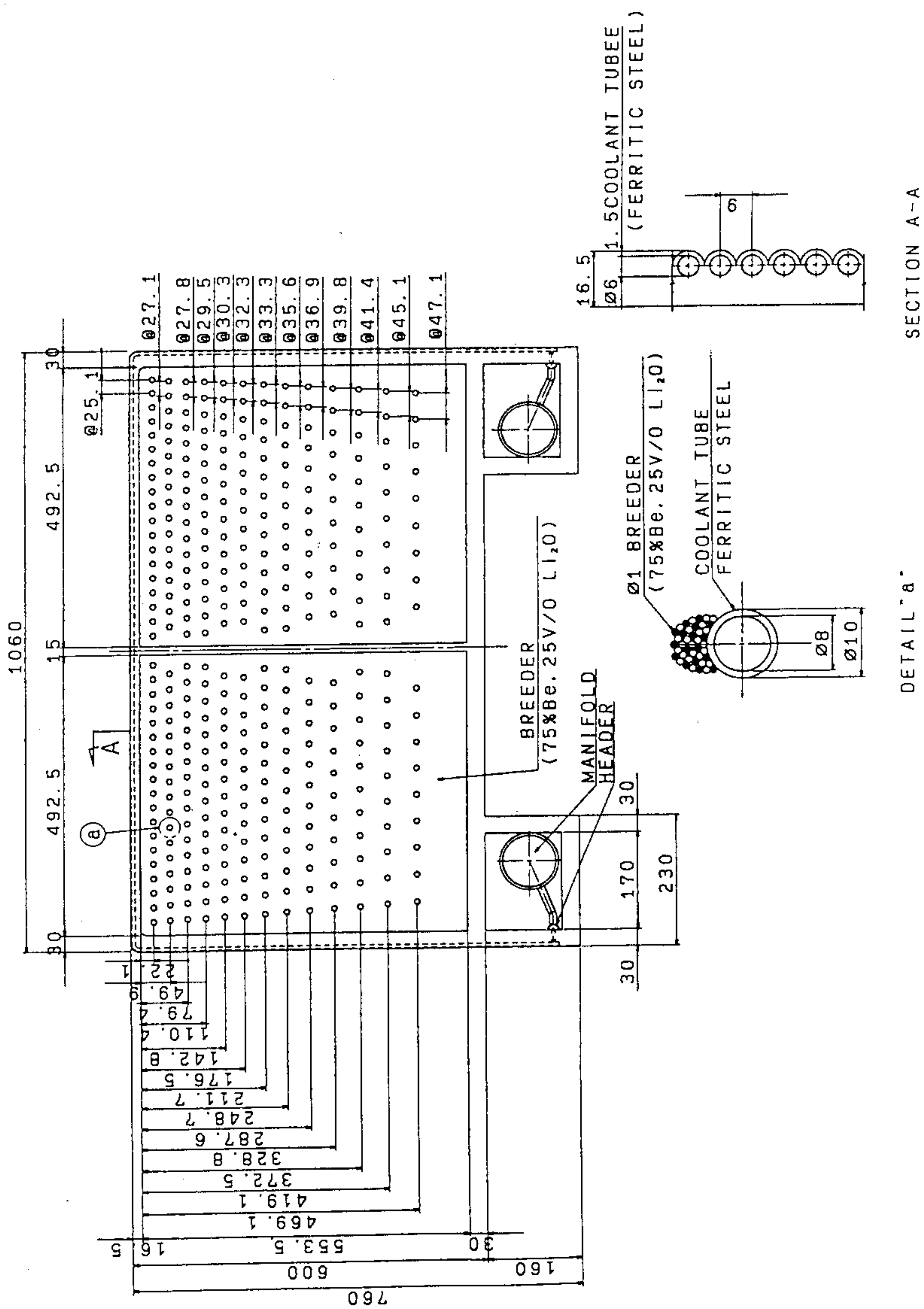


Fig. 2.1.6 Cross-sectional view of helium-cooled test module for ITER testing (breeder/multiplier mixture type)

2.2 Water-Cooled Blankets

2.2.1 DEMO Blanket Concepts

The major design parameters and performance characteristics of candidate blankets for DEMO and power reactors (Fusion power: ~ 3 GW) studied at JAERI are summarized in Table 2.2.1[1,2]. Various kinds of blankets for a DEMO reactor, including liquid metal breeder have been studied.

One of the advantages of a water-cooled blanket is the possibility of utilization of water cooling technology which has been developed for a pressurized water reactor (PWR). From this point, the outlet temperature of water coolant is selected to be about 320°C in this design studies. As for structural material, austenitic stainless steel will be selected, considering its abundant material data base. However, the existing austenitic steel, for example, 316SS, is not enough to be used in a DEMO/power reactor in terms of irradiation property change. The modified austenitic stainless steel (i.e. PCA) is selected in the design study. The BOT (Breeder Out-of-Tube) concept is adopted as shown in Fig. 2.2.1 and Fig. 2.2.2. It has a layer of beryllium multiplier which is separated from a breeder region. Another concept in which beryllium and breeder pebbles are mixed homogeneously is also proposed.

Breeder (Li_2O) is fabricated in the form of small-spherical pebbles (< 1 mm^{ϕ}) to avoid thermal cracking of breeder. The breeder temperature is maintained between 450°C and 950°C to recover tritium efficiently in the low pressure helium purge stream. A thermal resistant layer around the coolant tube is necessary to keep the breeder temperature above 400°C because the coolant temperature is between 280°C and 320°C. The resistant layer consists of the spacer tube (i.e., gap layer) as shown in Fig. 2.2.2 or the solid insulator (e.g., ceramics).

The water coolant pressure is selected to be 15 MPa to apply the PWR coolant condition. In order to accommodate the poloidal power variation, coolant channels and manifolds in the blanket are poloidally divided into two or three subunits.

The major parameters of the energy conversion system are also summarized in Table 2.2.1. A turbine system of the PWR power plant is employed for the water coolant system and gross thermal efficiency is estimated to be 34.4%.

2.2.2 Technical Issues for DEMO Blanket

Major technical issues for the water-cooled blanket are listed below.

(1) Development/Modification of structural materials
- Modification of austenitic and ferritic stainless steels is necessary. In particular, substantial improvement is required in terms of irradiation-induced property change.

(2) Safety consideration against coolant tube rupture
- Safety analysis for transient phenomena after coolant tube rupture in blanket is necessary (incl. vaporization of pressurized water, Li_2O/H_2O reaction, pressure rise of blanket vessel and temperature response).
- Mechanism for pressure relief under the accident condition

(3) Temperature control of breeder
- Fabrication and assembling accuracy of coolant tubes (including thermal resistant layer) must be considered.

(4) Recovery of tritium from water coolant
- Development of effective method and a compact apparatus to recover tritium leaking into water coolant

(5) Mass production of small-size breeder and/or multiplier pebbles
- Small size ($< 1\ mm^{\phi}$)
- Low production cost

(6) Packing of breeder and/or multiplier pebbles
- Method for uniform and high density packing
- Non-destructive investigation of breeding zone

(7) Thermo-mechanical design of high heat flux components
- Design consideration against boiling and burn-out of pressurized water for first wall and divertor.

2.2.3 Strategy/Approaches

Strategy/approaches for water-cooled solid breeder blankets are similar to those for helium-cooled solid breeder blankets.

In the Physics Phase, tests to characterize the neutronic environments, to check out blanket systems, and to check out instrumentation will be conducted sharing one horizontal port with helium-cooled solid breeder blankets.

In the Technology Phase, one test port would be allocated to water-cooled solid breeder blankets as mentioned in Section 2.1.3. During the first stage (for four years) in the Technology Phase, screening tests and multiple effect tests including irradiation effects on breeder/multiplier properties will be performed using submodules.

During the next stage (for three years) in the Technology Phase, blanket performance tests using full port size modules exposed to the plasma will be performed for concepts chosen from above tests.

At the final stage of the Technology Phase, segment test will be desirably conducted to demonstrate overall performance of blankets including accommodation to poloidal wall load distribution and thermo-mechanical integrity of large blanket structure.

Subdivision of a port space and a schedule of the testing for water-cooled solid breeder blankets are described in Fig. 2.2.3 and Fig. 2.2.4, respectively.

2.2.4 Test Description

Tests to be performed in ITER for water-cooled solid breeder blankets are the same as those for helium-cooled solid breeder blankets, which are as follows:

① Neutronics test and system check-out
② Screening test
③ Multiple effect test
④ Performance test
⑤ Segment test
⑥ Material irradiation test.

Major aspects of tests are summarized in Table 2.2.2.

A neutronic test for blankets including system/instrumentation check-out will be performed with very low neutron fluence during the Physics Phase. Submodules for 5-10 candidate blanket concepts will be tested in series using one-fourth (or one-third) region of the test port allocated to solid breeder blankets. Distributions of nuclear heating rate and tritium generation rate, γ-ray and induced activity should be measured in order to characterize neutronic aspects of the candidates.

Tests during the Technology Phase start with a screening test. In this test, tritium recovery performance based on tritium release from the breeder and breeder temperature control will be mainly investigated. The number of submodules to be tested will be 2-6 depending on the results of the neutronic test. Submodules will be installed also in one-fourth (or one-third) region of the port allocated to water-cooled solid breeder blanket. Neutron fluence required for this test is relatively low (≦ 0.01-0.02 MWa/m^2), therefore the test will be performed in one year.

Following the screening test, multiple effect test will be conducted. Main objective of this test is to investigate neutron irradiation effects on tritium recovery performance in terms of property changes of breeder and multiplier materials. One or two concepts chosen from the screening test will be tested with the same submodule used in the screening test. Indication of irradiation damage (sintering, cracking, grain growth etc.) would be also obtained. Neutron fluences of >0.2 MWa/m^2, therefore 1-2 years of reactor operation based on the scenario in Table 2.1.2 will be required for this test.

In the performance test, almost all of basic characteristics of blankets, such as heat generation and removal, tritium generation and recovery, and some of thermo-mechanical performances, will be demonstrated. The size of modules will be changed to full port size, and 1-3 blanket concepts will be selected from the results of above series of submodule tests. Test modules will directly face the plasma for the first time in the testing. (In above tests, test modules would not be directly exposed to the plasma.) More than 0.2 MWa/m^2 of neutron fluences will be required in order to investigate irradiation effects on breeder/multiplier properties as in the multiple effect test. According to the neutron fluence scenario in Table 2.1.2, this test will be sufficiently performed in the next three years of the Technology Phase when the number of the candidates is reduced to 1-3.

It is desirable to perform a segment test with one complete segment or even one sector in order to demonstrate accommodative performance of blankets to poloidal wall load distribution and thermo-mechanical integrity of large blanket structure.

Material irradiation test including the investigation of tritium release characteristics of breeder materials is performed in the port allocated for materials testing.

2.2.5 Specification and Characteristics of Test Articles

A test article considered here is a full port size module (first wall area; ~3.5 m^2) which will be tested at the performance test phase. The module will be directly exposed to plasma.

Major test items of the water-cooled test module are as follows:

① Verification of thermal-hydraulic performance of cooling mechanism

② Verification of integrity of blanket structure under normal and off-normal conditions

③ Verification of neutronics performances (tritium production rate and nuclear heating rate etc.)

④ Demonstration of coolant conditions expected in a DEMO/power reactor, i.e. coolant outlet temperature ~ 320°C

⑤ Demonstration of in-situ tritium recovery under the DEMO coolant conditions

(1) Consideration on structural material

Austenitic stainless steel, for example 316SS, is the first candidate material for the water-cooled test module because it has a great experience in fission reactors. The Ti-modified austenitic stainless steel (PCA), in which irradiation swelling property will be modified, has been developed and could be used for the ITER test modules.

Ferritic steel (e.g., HT9) is another candidate material because it will supply superior properties in terms of irradiation swelling and thermal conductivity.

During the early period of the testing in ITER, 316SS will be used as structural material of the test module. In the later stage of the testing, it might be replaced by newly-developed austenitic and/or ferritic steels which prospect for a DEMO/power reactor.

(2) Test module concept

A test module design with 316SS as the structural material is presented. Selection of structural material and blanket concept might be changed by progress of material development and out-of-ITER R&D.

The BOT (Breeder Out-of-Tube) concept with beryllium/breeder mixture zone is adopted for the one of water-cooled test modules as shown in Fig. 2.2.5 and Fig. 2.2.6. Breeder (Li_2O) is fabricated in the form of small spherical pebbles ($< 1\ mm^{\phi}$). The breeder temperature is maintained between 450°C and 950°C to recover tritium efficiently in the low pressure helium purge stream. In order to keep the breeder temperature abve 400°C, thermal resistant layer, which is made of ceramics insulator or a spacer tube, is provided around the coolant tube. The water coolant pressure is selected to be 15 MPa similar to the DEMO blanket.

Major characteristics of the test module is summarized in Table 2.2.3.

References

(1) TONE, T., et al., Technical Evaluation of Major Candidate Blanket Systems for Fusion Power Reactor, JAERI-M 87-017 (in Japanese) (1987)

(2) KURODA, T., et al., Technical Considerations on Breeding Blanket Systems for a Fusion Power Reactor, Fusion Engineering and Design 8 (1989) 219-226

Table 2.2.1 Major design features of candidate blankets for DEMO[2)]

Blanket concept	PCA/Li_2O/H_2O/Be	Mo-alloy/Li_2O/He/Be	Mo-alloy/$LiAlO_2$/He/Be	V-alloy/Li/Li/none	Mo-alloy/Li/He/none
Thermal power/gross electric power (MW)	3820/1320	3720/1810	3510/1710	3670/1450	3500/1700
Neutron wall loading (MW/m^2)	3.3	3.3	3.3	3.3	3.3
Heat flux on the first wall (MW/m^2)	0.9	0.9	0.9	0.9	0.9
Nuclear heating in the first wall (MW/m^2)	43	27	27	23	20
Structural material	Ti-modified SS	Mo-alloy	Mo-alloy	V-alloy	Mo-alloy
Maximum temperature of structural material (°C)	423	852	852	730	857
Breeder/neutron multiplier	Li_2O/Be	Li_2O/Be	$LiAlO_2$/Be	Li/none	Li/none
^{6}Li enrichment (%)	30	30	95	no	no
Breeder configuration	outside tube	outside tube	inside tube	self-cooling	outside tube
Temperature range of breeder (°C)	450 ~ 950	400 ~ 950	400 ~ 725	350 ~ 530	400 ~ 860
Tritium recovery	He purge stream	He purge stream	He coolant	Li coolant	Li extraction
Outboard/inboard local breeding ratio	1.20/1.13	1.37/1.32	1.24/1.17	1.39/1.28	1.36/1.22
Net breeding ratio	1.03	1.19	1.06	1.15	1.14
Coolant	H_2O	He	He	Li	He
Pressure (MPa)	15.5	9	9	1.5	9
Inlet/outlet temperature (°C)	280/320	400/700	400/700	350/530	400/700
Flow direction: first wall/breeder region	toroidal/poloidal	toroidal/poloidal	toroidal/toroidal	toroidal/poloidal	toroidal/poloidal
Maximum velocity: first wall/breeder region (m/s)	7.9/7.1	63/52	63/8.1	0.3/0.25	56/60
Pressure drop (MPa)	0.37	0.11	0.11	1.0*	0.10
Number of primary cooling loops	4	7	7	4	7
Pumping power (MW)	26	109	103	30	102
Steam pressure/temperature (MPa/°C)	6.5/280	24.6/538	24.6/538		
Gross thermal efficiency (%)	34.4	47.2	47.2	39.2	47.2

* Inboard first wall channel is critical (0.3 m/s, B_T=8T).

Table 2.2.2 Test items for water-cooled solid breeder blanket

No.1

Test name	: Neutronics test and system check-out
Blanket type	: Water-cooled solid breeder blanket
Description of measurements	: Nuclear heating rate distribution, tritium generation rate distribution, neutron and γ-ray flux, induced activity
Size	: $0.4\ m^w \times 0.5\text{-}1\ m^h \times 0.5\ m^t$ (submodule)
Number of test elements in 1/4 port	: 1 - 2
Location	: 0 - 50 cm from first wall
Boundary condition	: No-plasma exposure
Support equipment	: Cooling system
Operation time	
Duration of individual test	: a few short pulses × ~10 iterations
Number of tests in series	: ~6
Total reactor time	: < 0.001 MWa/m^2, < 2-3 years
Special requirements	:
Remarks	: to be tested during the Physics Phase

No.2

Test name	: Screening test (mainly with regard to tritium recovery performance)
Blanket type	: Water-cooled solid breeder blanket
Description of measurements	: Tritium breeding, tritium release rate, tritium inventory, breeder temperature control
Size	: $0.4\ m^w \times 0.5\text{-}1\ m^h \times 0.15\text{-}0.5\ m^t$ (submodule)
Number of test elements in 1/4 port	: 1 - 2
Location	: 0 - 50 cm from first wall
Boundary condition	: No-plasma exposure
Support equipment	: Tritium recovery system, tritium measurement system, cooling system
Operation time	
Duration of individual test	: 1000-10000 sec × 5-10 iterations
Number of tests in series	: 2 - 3
Total reactor time	: ≤ 0.01-0.02 MWa/m^2, < 1-2 years
Special requirements	: Total burn time ≥ 4-5 days is required for saturation of tritium inventory. Continuous operation time ≥ 1000-10000 sec and dwell time ≤ 300-400 sec are required for breeder temperature control.

Table 2.2.2 (Continued)

No.3

Test name	: Multiple effect test (including irradiation effects on breeder/multiplier properties)
Blanket type	: Water-cooled solid breeder blanket
Description of measurements	: Tritium release rate and inventory, effects of property changes due to irradiation on tritium release and temperature control (thermal conductivity, compatibility, breeding and tritium release properties), Indication of irradiation damage (sintering, cracking, grain growth etc.)
Size	: 0.4 m^w × 0.5-1 m^h × 0.15-0.5 m^t (submodule)
Number of test elements in 1/4 port	: 1 - 2
Location	: 0 - 50 cm from first wall
Boundary condition	: No-plasma exposure
Support equipment	: Tritium recovery system, tritium measurement system, Cooling system, post irradiation test system
Operation time	
Duration of individual test	:
Number of tests in series	: 1 - 2
Total reactor time	: ≥ 0.2 MWa/m^2, ~ two years
Special requirements	: The same operation mode as tritium recovery test (No.2) is required.
Remarks	: Sequential to the screening test using submodules chosen from the results of the screening test

No.4

Test name	: Performance test
Blanket type	: Water-cooled solid breeder blanket
Description of measurements	: Heat generation and removal (outlet coolant temperature for electricity generation), Continuous tritium recovery, tritium breeding, tritium inventory, thermal-hydraulics and thermo-mechanical performances, irradiation effects on materials
Size	: 1 m^w × 3 m^h × 0.5 m^t (full port size module)
Number of test modules in one port	: 1
Location	: 0 - 50 cm from first wall
Boundary condition	: With plasma exposure
Support equipment	: Tritium recovery system, Cooling system, (electricity generation system), post irradiation test system
Operation time	
Duration of individual test	: ≥ 1000-10000 sec × operation cycles (TBD)
Number of tests in series	: 1 - 2
Total reactor time	: ≥ 0.2 MWa/m^2, ≥ 1-2 years
Special requirements	: Burn time ≥ 800 sec (for dwell time = 100 sec) or burn time ≥ 1000 sec (for dwell time = 200 sec) is required to obtain thermal steady state (Indication of electric power generation).

Table 2.2.2 (Continued)

No.5

Test name	: Segment test (demonstration of DEMO act-like blanket)
Blanket type	: Water-cooled solid breeder blanket
Description of measurements	: Heat generation and removal (outlet coolant temperature for electricity generation) continuous tritium recovery, tritium breeding, tritium inventory, thermal-hydraulics and thermo-mechanical performances
Size	: 1 m^w × (full height of driver blanket) × 0.5 m^t
Number of test segments in one sector	: 1 (1/48 torus) or 3 (1/16 torus = 1 sector)
Location	: 0 - 50 cm from first wall
Boundary condition	: With plasma exposure
Support equipment	: Tritium recovery system, Cooling system, (electricity generation system), post irradiation test system
Operation time	
Duration of individual test	: ≥ 1000-10000 sec × operation cycles (TBD)
Number of tests in series	: 1 - 2
Total reactor time	: TBD
Special requirements	: Burn time ≥ 800 sec (for dwell time = 100 sec) or burn time ≥ 1000 sec (for dwell time = 200 sec) is required to obtain thermal steady state (Indication of electric power generation).
Remarks	: To be conducted at the final stage of the Technology Phase

No.6

Test name	: Material irradiation tests
Blanket type	: Not specified
Description of measurements	: Irradiation damage (swelling, compatibility etc.) and its effects on physical and mechanical properties of structural material, shell conductor, insulator, breeder and so forth
Size	: Total plasma facing surface area of 1.5 m^2 (small articles)
Number of test elements in one port	: TBD
Location	: 0 - 50 cm from first wall
Boundary condition	: No-plasma exposure and with plasma exposure
Support equipment	: Cooling system, post irradiation test system
Operation time	
Duration of individual test	: TBD
Number of tests in series	: TBD
Total reactor time	: TBD (≥ 3 MWa/m^2)
Special requirements	:
Remarks	: to be performed in the port allocated for material testing

Table 2.2.3 Major design parameters and characteristics of water-cooled test module

Location	Outboard, midplane
Module type	BOT (Breeder Out-of-Tube)
Module size	
First wall area	3.5 m^2 (3.4 m × 1.1 m)
Thickness	0.6 m
Neutron wall load	1.2 MW/m^2
Surface heat load	0.15 MW/m^2
Coolant	Water
First wall (module box)	
Temperature (in/out)	280/320°C
Pressure	15 MPa
Breeding zone	
Temperature (in/out)	280/320°C
Pressure	15 MPa
Structural material	Austenitic steel (316SS)
Allowable temperature	< 450°C
Breeder	Lithium oxide: 1mm ϕ pebble
^6Li enrichment	natural (mixture type) 30% (separate type)
Temperature control	Cooling tube arrangement and thermal resistant layer around tube
Operating temperature	450 - 600°C
Neutron multiplier	Beryllium: 1mm ϕ pebble (mixture type) block (separate type)
Operating temperature	450 - 600°C (mixture type) < 400°C (separate type)
Local tritium breeding ratio (1-D poloidal model)	∿1.5 (mixture type) ∿1.2 (separate type)
Heat deposition	
Module box (first wall)	1.70 MW
Breeding zone	4.11 MW
Tritium recovery	In-situ and continuous with helium purge gas
Tritium inventory	TBD

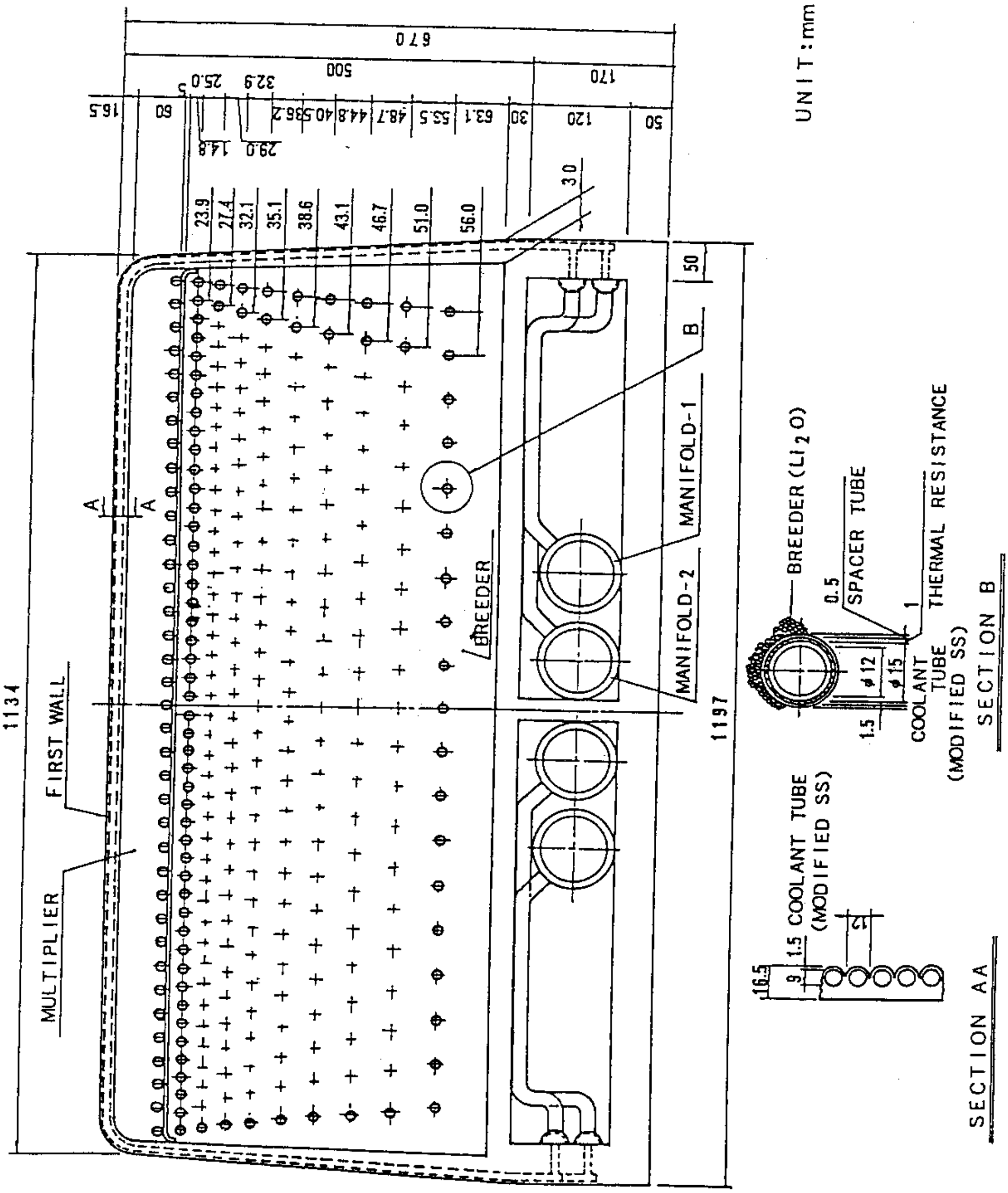


Fig. 2.2.1 Cross-sectional view of PCA/Li_2O/H_2O/Be blanket for DEMO reactor (separate multiplier zone type)

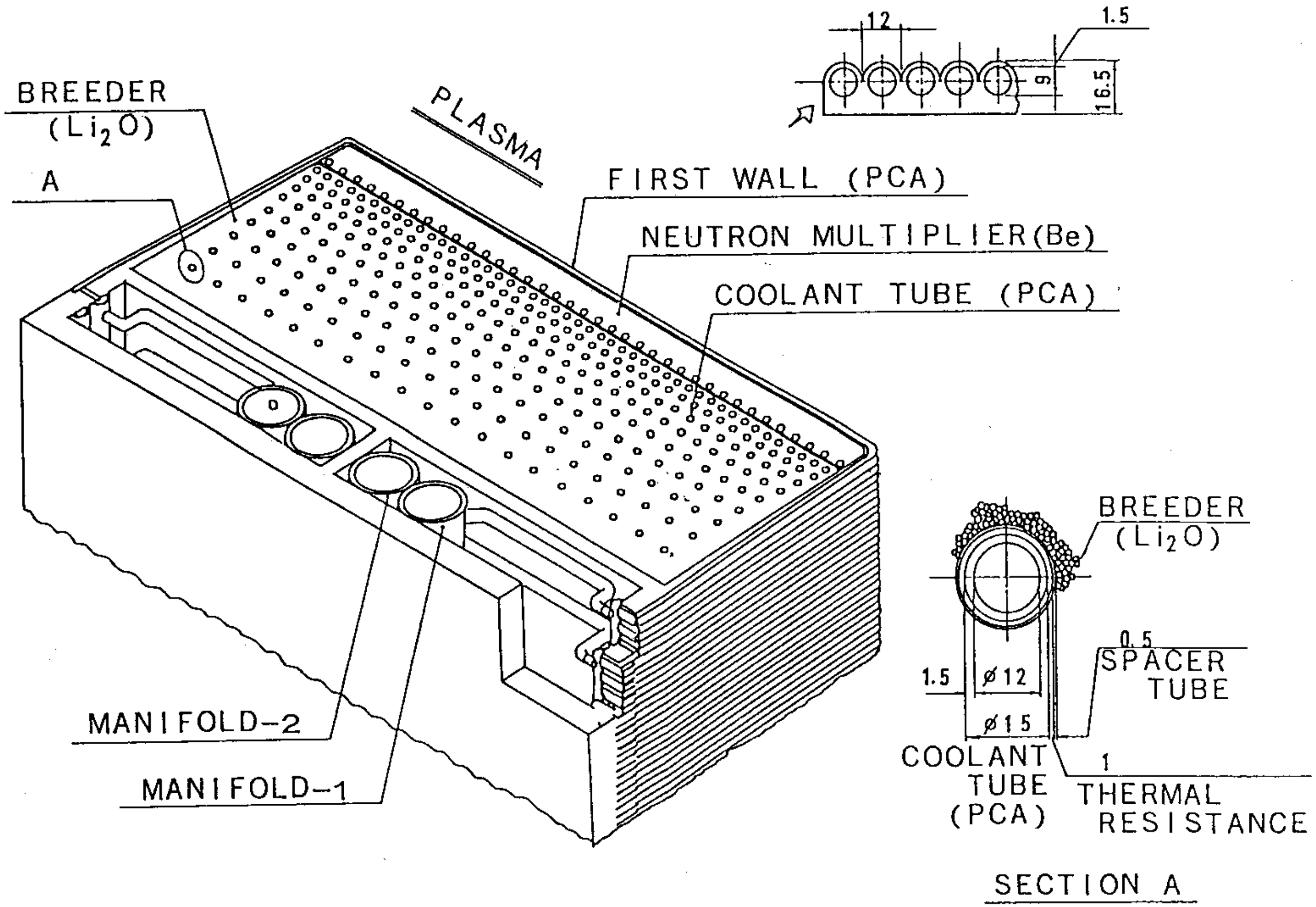


Fig. 2.2.2 Schematic view of PCA/Li_2O/H_2O/Be blanket for DEMO reactor (separate multiplier zone type)

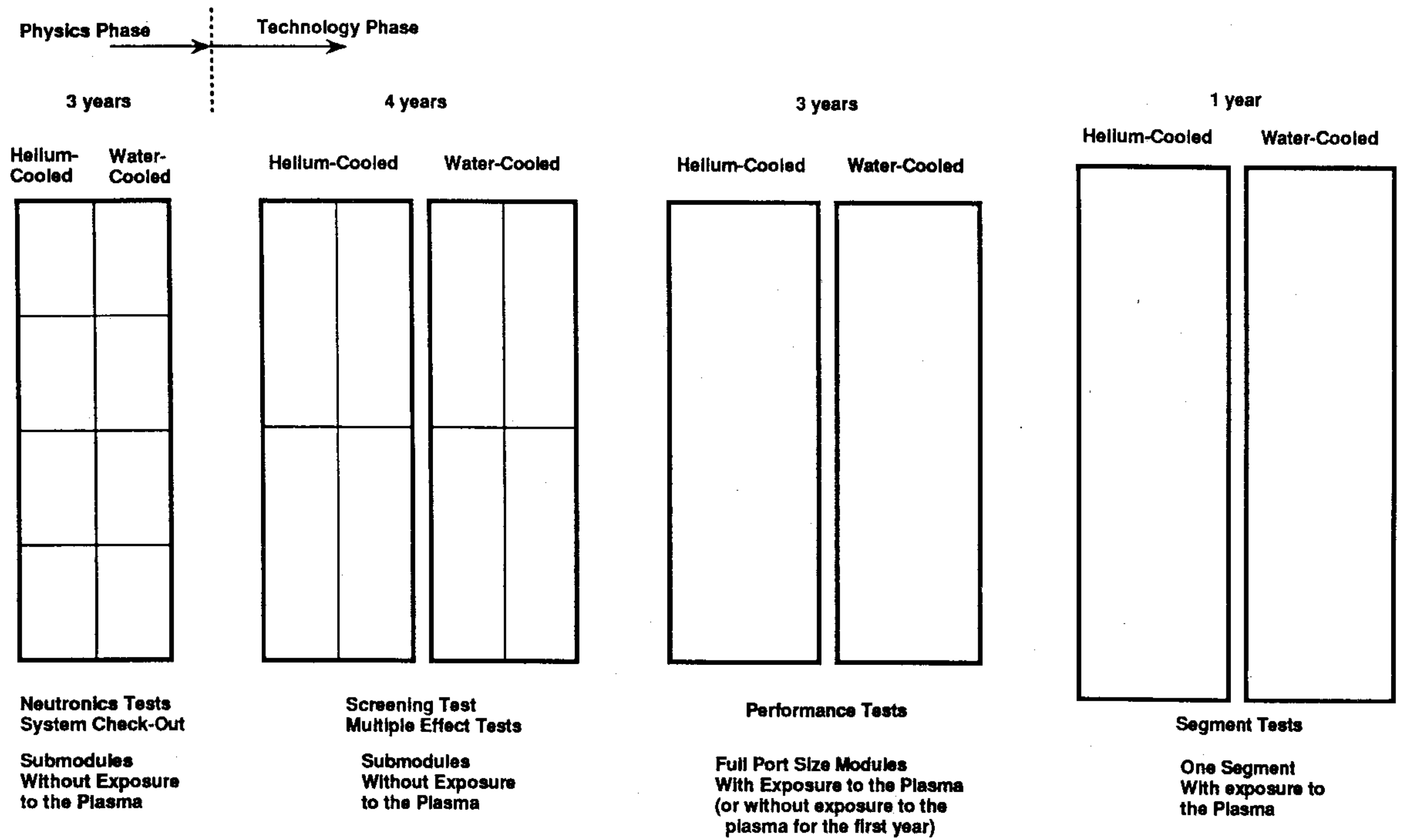


Fig. 2.2.3 Test port allocation to helium- and water-cooled solid breeder blankets

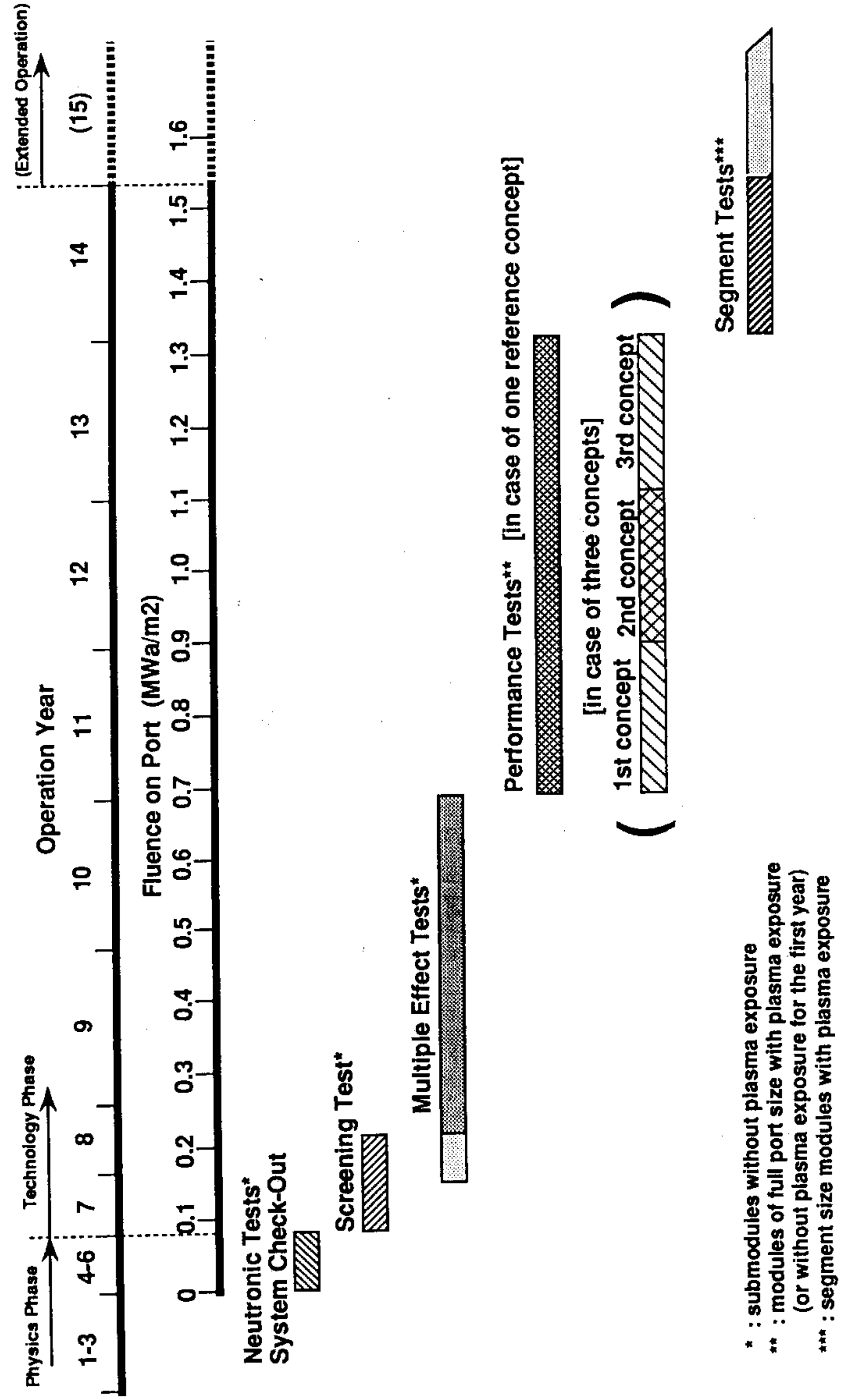


Fig. 2.2.4 Testing schedule for water-cooled solid breeder blanket

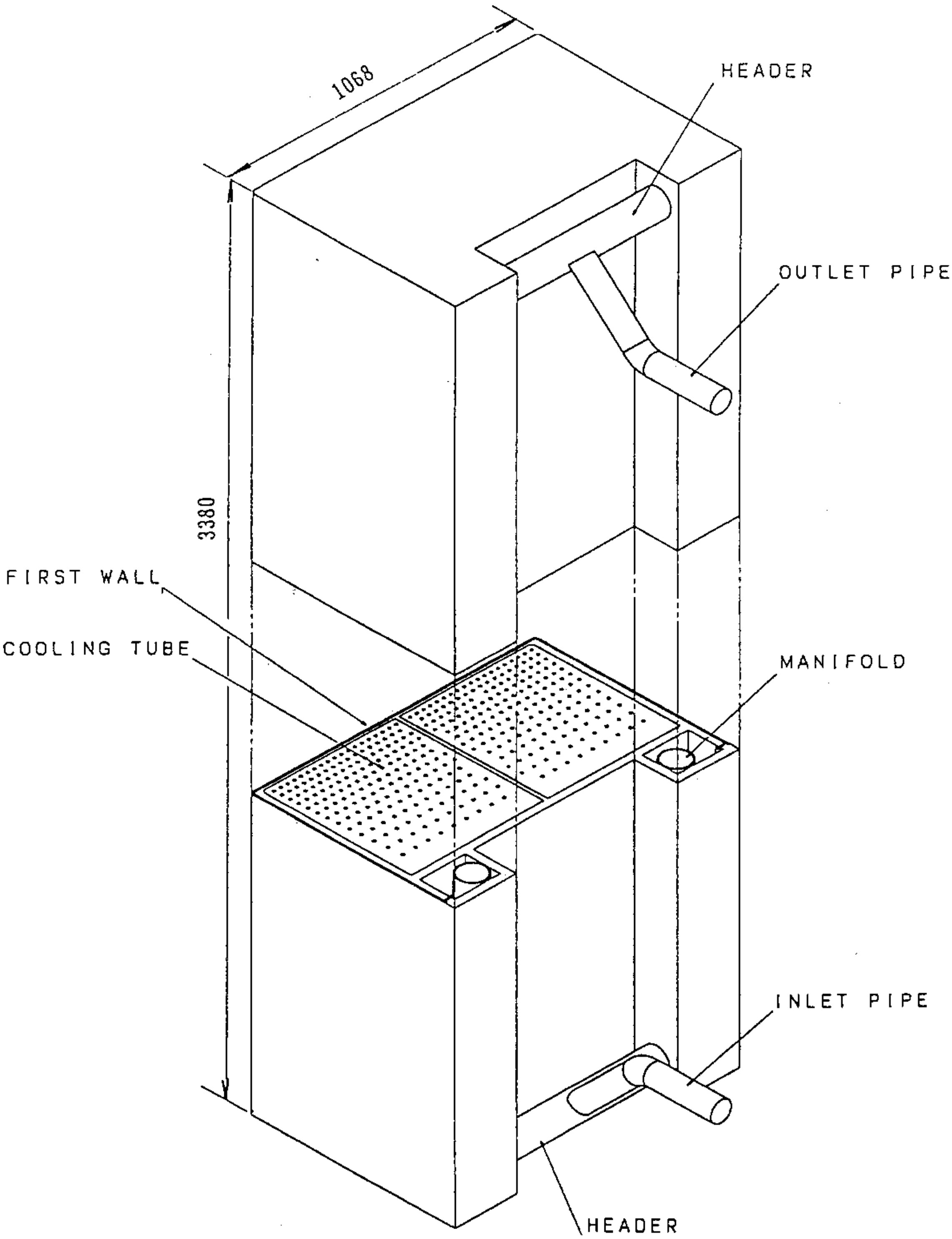


Fig. 2.2.5 Schematic view of water-cooled test module for ITER testing (breeder/multiplier mixture type)

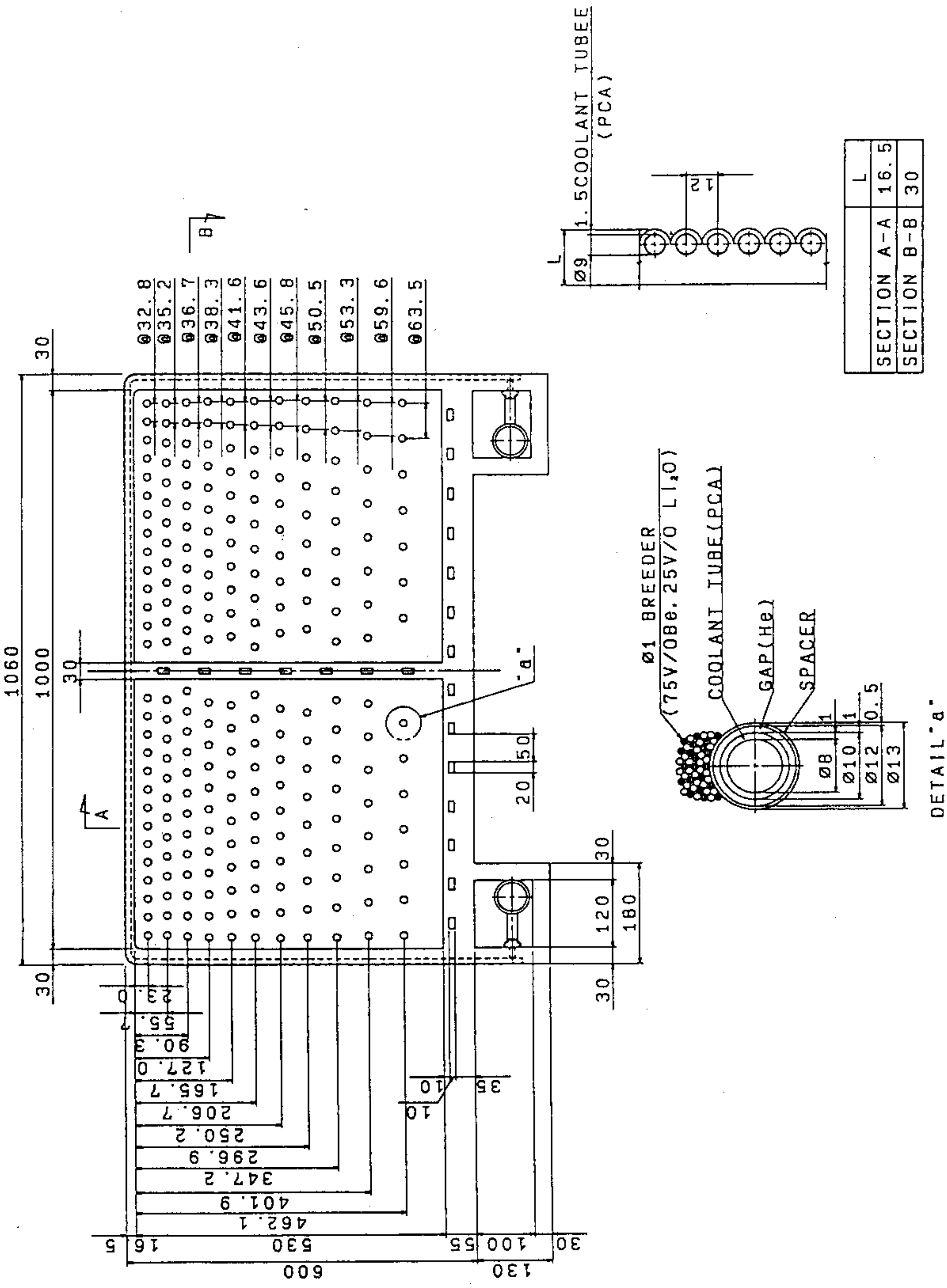


Fig. 2.2.6 Cross-sectional view of water-cooled test module for ITER testing (breeder/multiplier mixture type)

2.3 Separate First Wall Design for Test Article Containment

A structural concept of a test article containment during the performance test phase is illustrated in Fig. 2.3.1. The container and the test port duct are hermetically sealed by lip seal welding at the support flange. The inside of the container is under atmospheric condition. Therefore, the plasma chamber is not contaminated even if some trouble, for example, coolant leakage from test modules would occur. Replacement and setting of the test module can be performed without breaking the vacuum boundary of the reactor.

The container has the same cooling mechanism as the first wall of the reactor. The wall of the container is a pile of ribbed panels which have rectangular coolant paths. The coolant is pressurized water which has the same conditions as those for the main first wall of the reactor, i.e. 1.5 MPa and 60/100°C.

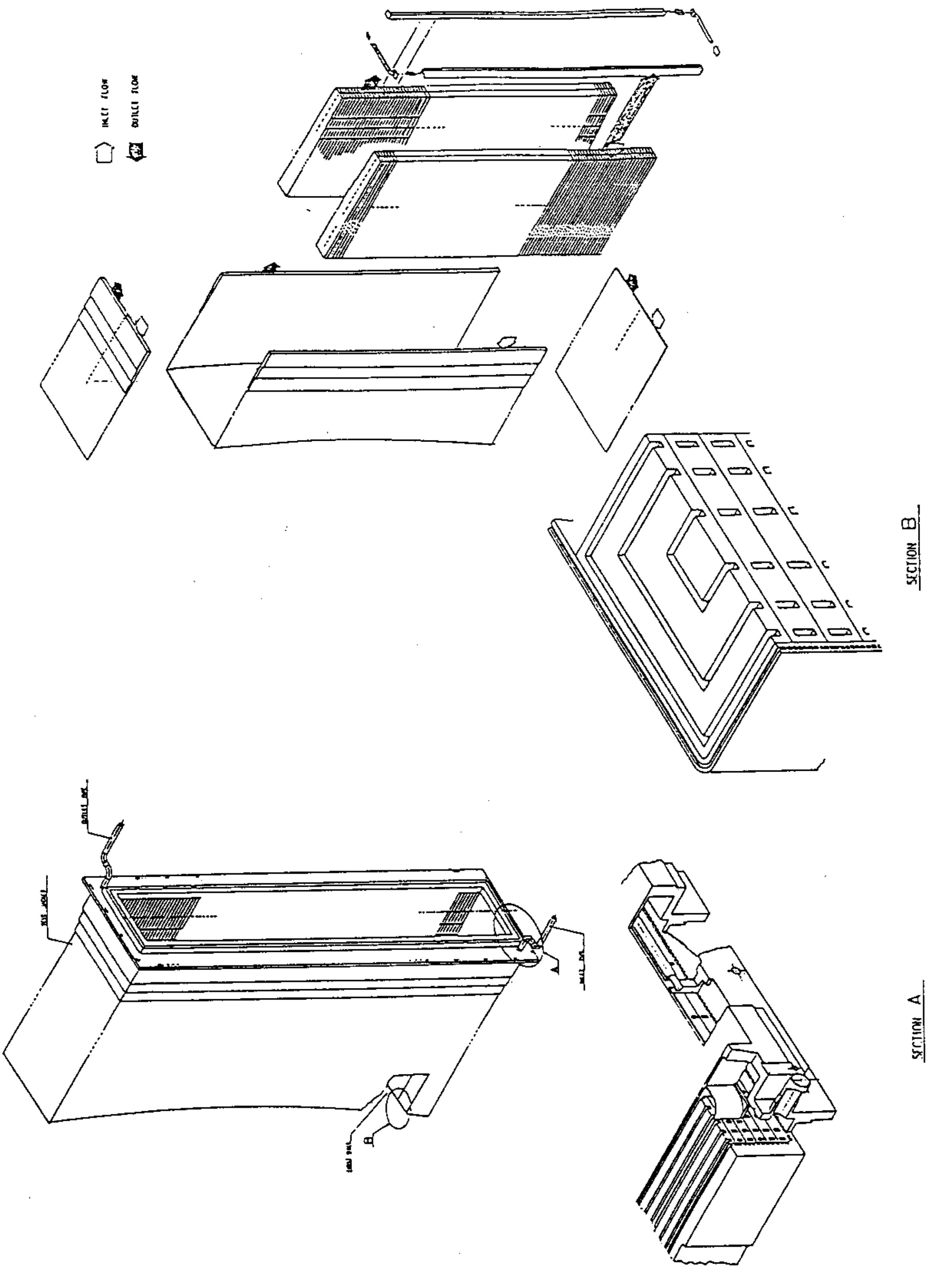


Fig. 2.3.1 Concept of test article containment

3. Ancillary Equipment

3.1 Helium-cooled Blankets

The helium-cooled test module is ancillarily equipped with its own cooling system and tritium recovery system because its test conditions differ from the others. In this section, system configuration, specification of equipments, and layout plan of both systems are described corresponding to a test module of full port size shown in Section 2.1.5.

3.1.1 Test Module Cooling System

3.1.1.1 System Functions

The purposes of test module cooling system are as follows:

(i) to remove heat generated in test module including module box structure, and

(ii) to control the breeder temperature in the projected test condition.

Major parameters of test module cooling system are summarized in Table 3.1.1.

The test module cooling system removes heat generated in the test module, which is evaluated to be 5.93 MW, and transfers the heat to the secondary cooling water loop of main reactor heat removal system.

The test module inlet and outlet temperature of helium coolant are 360°C and 480°C, respectively. The helium coolant pressure is 9 MPa to simulate the DEMO blanket. And the helium coolant flow rate of 10.0 kg/s is derived from the heat balance. These coolant conditions keep the breeder zone temperature between 450°C and 950°C, and the structural material temperature below the maximum allowable temperature.

3.1.1.2 System Description

(1) System Configuration

Figure 3.1.1 shows a schematic flow diagram of the test module cooling system. The test module cooling system is made up of a heat transferring helium gas loop, purification unit, and helium gas make-up unit.

Heat generated in the test module is transferred to the secondary cooling loop of main reactor heat removal system through the helium gas loop. The helium gas loop comprises a main cooler, a filter, main circulators, and a heater. During plasma burning, helium gas leaving the test module flows through main cooler, filter, and a main circulator.

During dwell time or before plasma burning, bypass line of a main cooler is opened and a heater is put into line to make up the test condition.

The purification unit circulates the helium gas in a low flow rate and removes impurities from the helium gas stream. Hydrogen including tritium which permeates through the cooling tubes in the test module is estimated as major impurity in the helium gas. Hydrogen is removed by catalytic oxidation followed by adsorption.

The helium gas make-up unit supplies and stores helium gas at high pressure.

The helium gas hold up of the test module cooling system is estimated to be about 100 kg assuming coolant pipe between the test module and the cooling system room is 100 m in length.

(2) System Performance

Process conditions during plasma burning is shown in Fig. 3.1.1. Heat load to the test module cooling system is 5.93 MW, which is equal to heat load to secondary cooling loop of heat removal system. The test module cooling system circulates helium gas of 9 MPa at a flow rate of 10 kg/s with lowering the helium gas temperature from 480°C to 360°C at the main cooler. Inlet and outlet temperature of water at the main cooler are 35°C and 50°C, respectively, and pressure of water is 2.0 MPa. The required flow rate of water is derived as 345 t/h.

3.1.1.3 Components Description

Major components and units included in the test module cooling system are main cooler, main circulator, heater, gas purification unit, and gas make-up unit. Specifications of these components and units are summarized in Tables 3.1.2 through 3.1.6.

3.1.1.4 System Layout

Layout plan of the test module cooling system is shown in Fig. 3.1.2. Required space for the test module cooling system is 20 m × 10 m × 12 m-height.

3.1.2 Test Module Tritium Recovery System

3.1.2.1 System Functions

The purposes of test module tritium recovery system are as follows:

(i) to recover tritium produced in breeder zone of the test module continuously from helium sweep gas stream,

(ii) to analyze tritium recovered and contained in helium gas, and

(iii) to control atmosphere in breeder zone by varying gas flow rate and adjusting gas composition.

Major parameters of test module tritium recovery system are summarized in Table 3.1.7.

Tritium production rate is evaluated using integrated neutron wall load to the breeder zone and local tritium breeding ratio. Successive pulsed plasma operation of 1000 sec burn and 200 sec dwell is assumed to last 1 week. Helium sweep gas is at atmospheric pressure (0.1 MPa) and its flow rate is determined to keep tritiated species partial pressure less than 5 Pa, which is one-twentieth of equilibrium water pressure at 400°C in Li_2O-H_2O-LiOH system. Since hydrogen gas at a rate of H/T=100 is added to the helium sweep gas to recover tritium efficiently, chemical form of tritium released from the breeder is assumed to be 97% of HT and 3% of HTO based on in-situ tritium recovery experimental data. As water is not used as coolant of the test module, water source to the test module atmosphere is assumed to be tritium in the form of HTO and helium gas flow rate is set to 10 Nm^3/h.

Tritium is recovered with catalytic oxidation followed by adsorption.

3.1.2.2 System Description

(1) System Configuration

Figure 3.1.3 is a schematic flow diagram of the test module tritium recovery system. The test module tritium recovery system is made up of analytical section, tritium recovery section, and gas adjusting section.

Analytical section measures tritium in the form of HT and HTO separately and continuously and is composed of ceramic electrolysis cell, moisture adsorption bed, and ionization chambers.

Tritium recovery section removes tritium from the helium gas stream with catalytic oxidation followed by adsorption. Tritium is recovered as HTO in moisture adsorption bed, which is designed to be able to be in line over one week continuously. In addition to the moisture adsorption beds, liquid nitrogen cooled cold trap is equipped for additional removal of impurities.

Gas adjusting section controls the sweep gas composition by adding hydrogen gas or so on.

(2) System Performance

Process condition during plasma burning is shown in Fig. 3.1.4.

In nominal test condition, hydrogen gas of 2100 ppm, which makes H/T ratio in the test module atmosphere 100, is added to helium gas stream. Tritium production rate is 5.60×10^{-2} g/h and chemical form of tritium is 97% of HT and 3% of HTO. Assuming no water source other than released tritiated water, water concentration in test module atmosphere is 2.3 ppm, which is much less than one-twentieth of water equilibrium pressure of Li_2O-H_2O-LiOH system at 400°C. With oxidation followed by adsorption, tritium removal at an efficiency of over 0.999 per path is attained.

3.1.2.3 Components Description

Major components composing the test module tritium recovery system are catalytic oxidizer, cooler, dryer, pump, and cold trap. Specifications of these components are summarized in Tables 3.1.8 through 3.1.12.

3.1.2.4 System Layout

Layout plan of the test module tritium recovery system is shown in Fig. 3.1.5. As the test module tritium recovery system handles tritium of relatively high concentration, most of the system is contained in glove box of 1.2m × 4.5m × 3.0m-high in dimension.

The test module tritium recovery system is operated remotely. In the course of maintenance and repair, workers enter the system room and access the tritium recovery system in the glove box.

Table 3.1.1 Major parameters of cooling system for helium-cooled test module

Heat Load	
Test Module Box	1.73 MW
Breeding Zone	4.20 MW
Total	5.93 MW
Type of Coolant	Helium Gas
Coolant Flow Mode	Series (Test Module Box → Breeding Zone)
Temperature	
Inlet of Test Module Box	360°C
Outlet of Test Module Box	400°C
Inlet of Breeding Zone	400°C
Outlet of Breeding Zone	480°C
Pressure	9 MPa
Flow Rate	10.0 kg/s (36.0 t/h)
Heat Sink	Secondary Cooling Loop of ITER Heat Removal System
Auxiliary System	Helium Gas Purification Unit Helium Gas Make-up Unit

Table 3.1.2 Specifications of main cooler

Number of Components	1
Type	Water Cooled Shell and Tubes Heat Exchanger
Fluid	
Shell Side	Helium
Tube Side	Water
Flow Rate	
Helium	10.0 kg/s
Water	345 t/h
Pressure	
Helium	9 MPa
Water	2 MPa
Temperature	
Helium (Inlet/Outlet)	480°C/360°C
Water (Inlet/Outlet)	35°C/ 50°C
Heat Load	6.0 MW

Table 3.1.3 Specifications of main circulator

Number of Components	2
Type	Centrifugal Circulator with Gas Bearing
Fluid	Helium Gas
Flow Rate	10.0 kg/s
Pressure at the Inlet	9 MPa
Head	600 kPa
Temperature at the Inlet	360°C
Driving Unit	High-frequency Induced Motor

Table 3.1.4 Specifications of heater

Number of Components	1
Type	Electric Heater
Fluid	Helium Gas
Flow Rate	10.0 kg/s
Pressure at the Inlet	9 MPa
Heating Power	1 MW

Table 3.1.5 Specifications of gas purification unit

Number of Units	1
Fluid	Helium Gas
Flow Rate	0.1 kg/s
Pressure at the Interface	9 MPa
Temperature at the Interface	400°C
Purification Method	Oxidation and (Cryo-)Adsorption

Table 3.1.6 Specifications of gas make-up unit

Number of Units	1
Fluid	Helium Gas
Gas Storage Pressure	15 MPa
Gas Storage Capacity	TBD

Table 3.1.7 Major parameters of tritium recovery system for helium-cooled test module

1. Tritium Production Rate	
Neutron Wall Load	1.2 MW/m^2
Surface Area of Test Module Facing to Plasma	3.5 m^2
Plasma Burn Time	1200 sec
Plasma Dwell Time	200 sec
Duration of Continuous Operation	1 week
Local Tritium Breeding Ratio	1.67
Tritium Production Rate (during Burning)	5.60×10^{-2} g/h
2. Breeding Zone	
Breeder	Li_2O 1 mmϕ Pebble (85%T.D.)
Multiplier	Be 1 mmϕ Pebble
Volumetric Ratio of Breeder/Multiplier	1/3
Packing Fraction	60%
Weight of Breeder	420 kg
Weight of Multiplier	1360 kg
Temperature of Breeding Zone	400 to 1000°C

Table 3.1.7 (Continued)

3. Sweep Gas	
Fluid	Helium Gas
Pressure	0.1 MPa
Temperature	
Outlet of Breeding Zone	400°C
Inlet of Breeding Zone	R.T.
Flow Rate	10 Nm^3/hr to keep tritiated species partial pressure less than 5 Pa
Composition of Impurities at the Inlet	
Hydrogen	Nominal 2100 ppm
Water	less than 1 ppm
Others	TBD
Chemical Form of Released Tritium from Breeder	HT/HTO = 97/3 *)
Release Rate of Impurity	Kr, Xe etc. (not used in design)
4. Tritium Recovery	
Tritium Recovery Method	Oxidation Followed by Adsorption
Tritium Recovery Efficiency	Nominal 0.999 (one path) depending on hydrogen content

*) H. Yoshida et al., Proc. Fusion Reactor Blanket and Fuel Cycle Technology, p158 Tokai, Oct. 1986.

Table 3.1.8 Specifications of catalytic oxidizer

Number of Components	1
Fluid	Helium Gas
Flow Rate	10 Nm^3/h
Pressure	0.1 MPa
Operation Temperature	473 K
Catalyst	Pd-Pt Coated Alumina
Hydrogen Concentration Reduction Factor	10000
Size	100 mmϕ × 1000 mmL

Table 3.1.9 Specifications of cooler

Number of Components	1
Type	Water Cooled Shell and Tubes Heat Exchanger
Fluid	Helium Gas
Flow Rate	10 Nm^3/h
Pressure	0.1 MPa
Size	300 mmϕ × 1000 mmL

Table 3.1.10 Specifications of dryer

Number of Components	2 (used alternatively)
Type	Adsorbent Packed Bed
Fluid	Helium Gas
Flow Rate	10 Nm^3/h
Pressure	0.1 MPa
Operation Temperature	293 K
Duration of Operation	1 week per one dryer
Adsorbent	Molecular Sieve Type 5A
Size	400 mmϕ × 1400 mmL

Table 3.1.11 Specifications of pump

Number of Components	1
Type	Wabble Pump
Fluid	Helium Gas
Flow Rate	Nominal 10 Nm^3/h
Pressure (inlet/outlet)	0.08/0.1 MPa
Size	635 × 350 × 580 mm

Table 3.1.12 Specifications of cold trap

Number of Components	2
Type	LN_2 Cooled Adsorbent Packed Bed
Fluid	Helium Gas
Flow Rate	10 Nm^3/h
Pressure	0.1 MPa
Operation Temperature	77 K
Adsorbent	Molecular Sieve Type 5A
Size	700 mmϕ × 1500 mmL

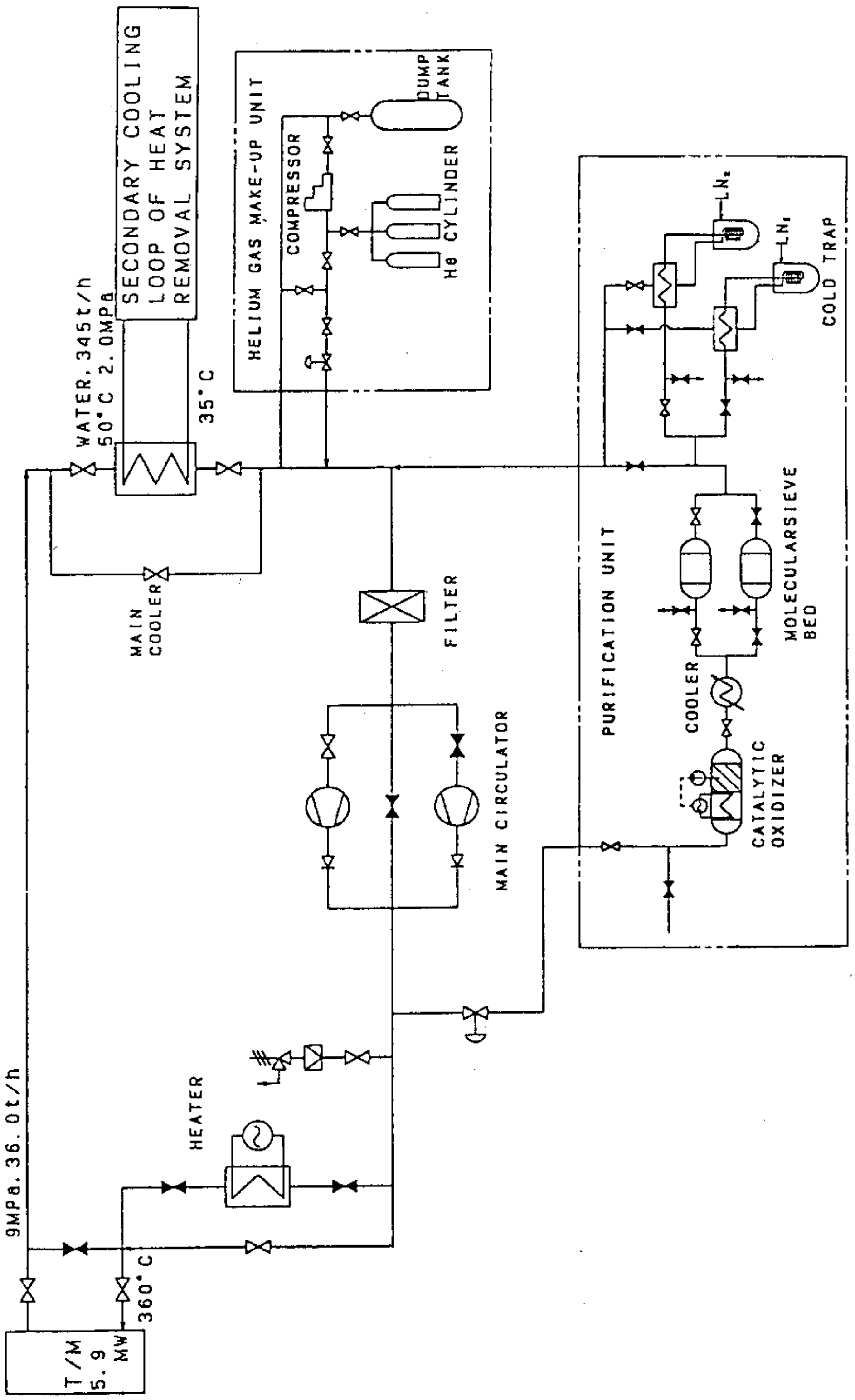


Fig. 3.1.1 Schematic flow diagram of cooling system for helium-cooled test module

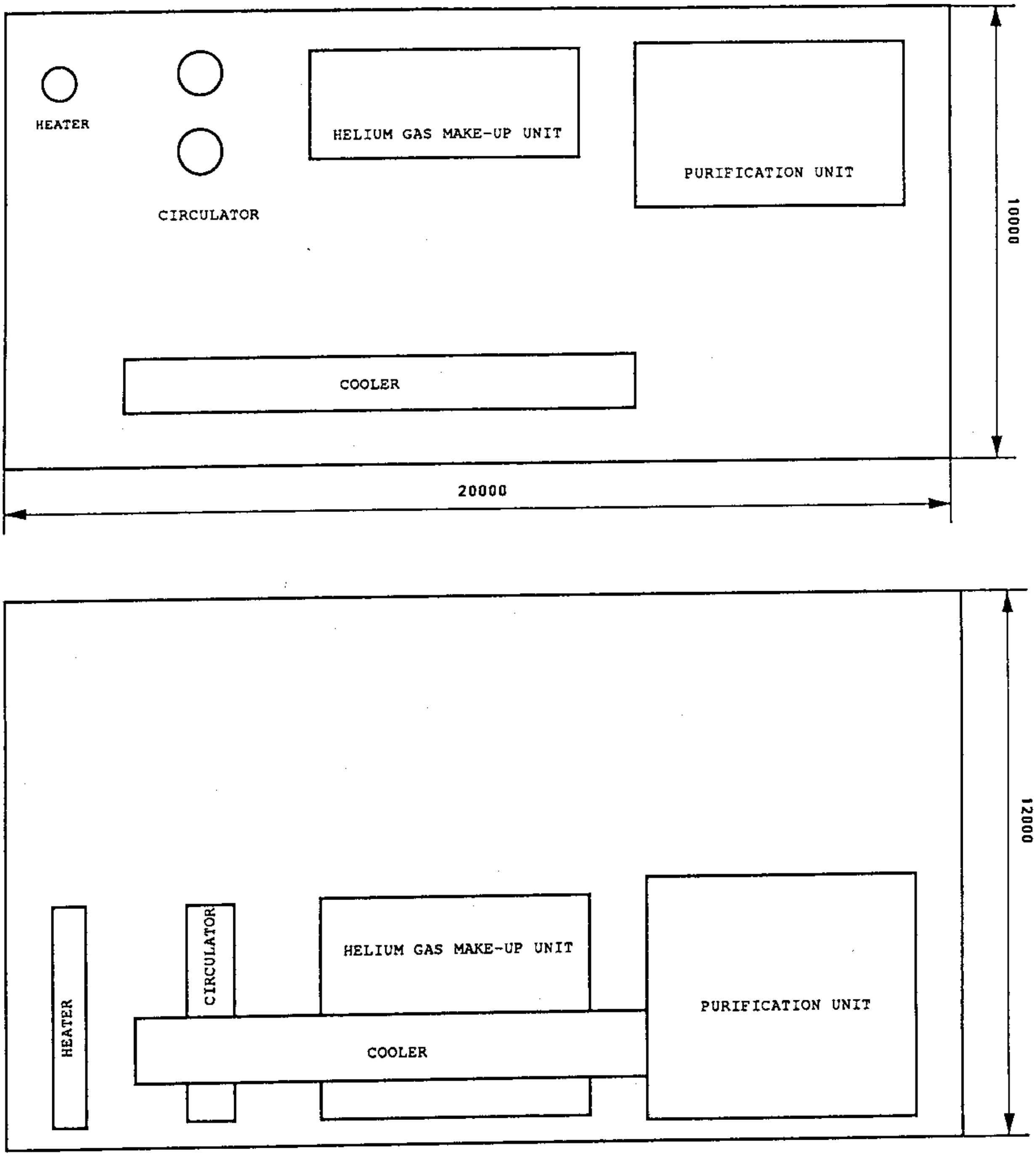
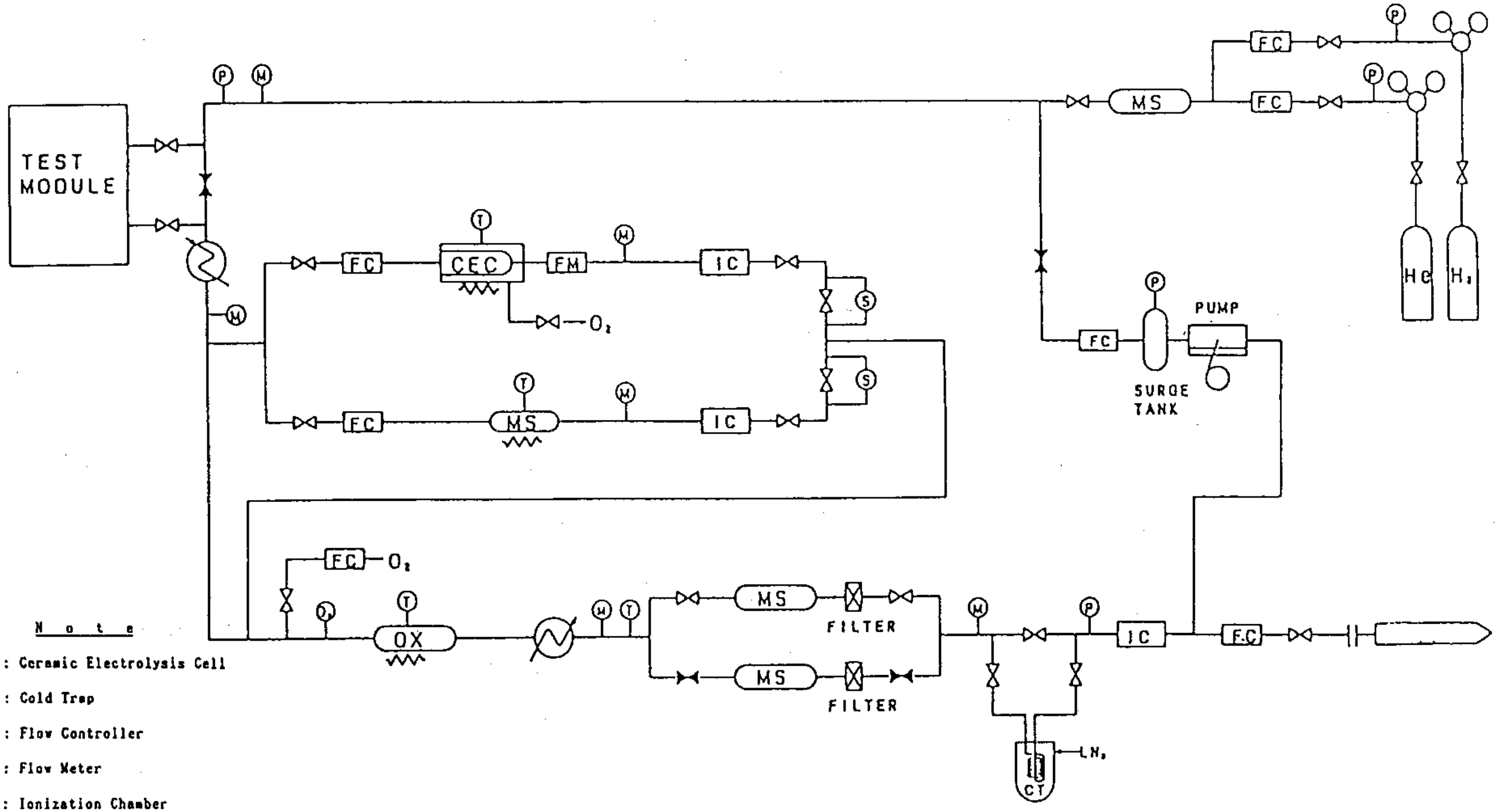


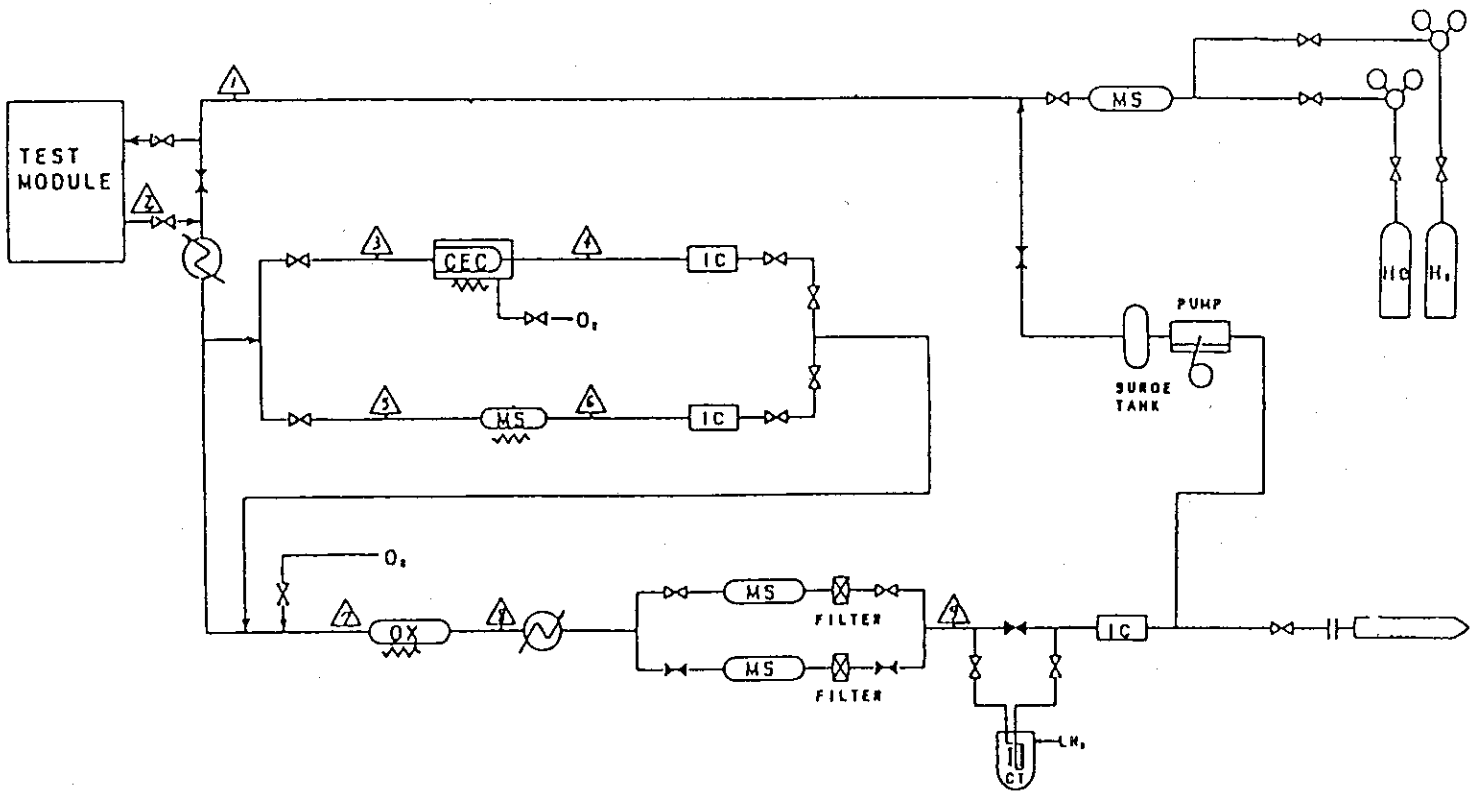
Fig. 3.1.2 Layout plan of cooling system for helium-cooled test module



N o t e

CEC : Ceramic Electrolysis Cell
CT : Cold Trap
FC : Flow Controller
FM : Flow Meter
IC : Ionization Chamber
M : Hygrometer
MS : Moisture Adsorption Bed
OX : Catalytic Oxidizer
P : Pressure Gauge
S : Gas Sampler for Analysis
T : Thermometer

Fig. 3.1.3 Schematic flow diagram of tritium recovery system for helium-cooled test module



Position	1	2	3	4	5	6	7	8	9
	TM Inlet	TM Out	CEC In	CEC Out	MS In	MS Out	OX In	OX Out	MS Out
Temperature (K)	293	673	293	973	293	293	293	473	293
Pressure (MPa)	~0.1	←	←	←	←	←	←	←	←
Flow Rate (Nm^3/H)	10.0	←	$3.0x10^{-3}$	←	$3.0x10^{-3}$	←	10.0	←	←
Hydrogen Conc. (vpm	$2.1x10^3$	$2.1x10^3$	←	$2.1x10^3$	$2.1x10^3$	$2.1x10^3$	$2.1x10^3$	$2.1x10^{-1}$	←
Water Conc. (vpm)	1.0	2.3	←	1.0	2.3	1.0	2.3	$2.1x10^3$	1.0
HT Conc. (Ci/Nm^3)	$5.2x10^{-3}$	$5.2x10^1$	←	$5.3x10^1$	$5.2x10^1$	←	←	$5.2x10^{-3}$	←
HTO Conc. (Ci/Nm^3)	$3.1x10^{-2}$	1.6	←	$7.0x10^{-1}$	1.6	$7.0x10^{-1}$	1.6	$5.4x10^1$	$3.1x10^{-2}$

Tritium Production Rate
5.60×10^{-2} g/h
Chemical Form of Tritium
HT / HTO = 97 / 3
Hydrogen Addition Rate
H / T = 100

Fig. 3.1.4 Chemical flow diagram of tritium recovery system for helium-cooled test module

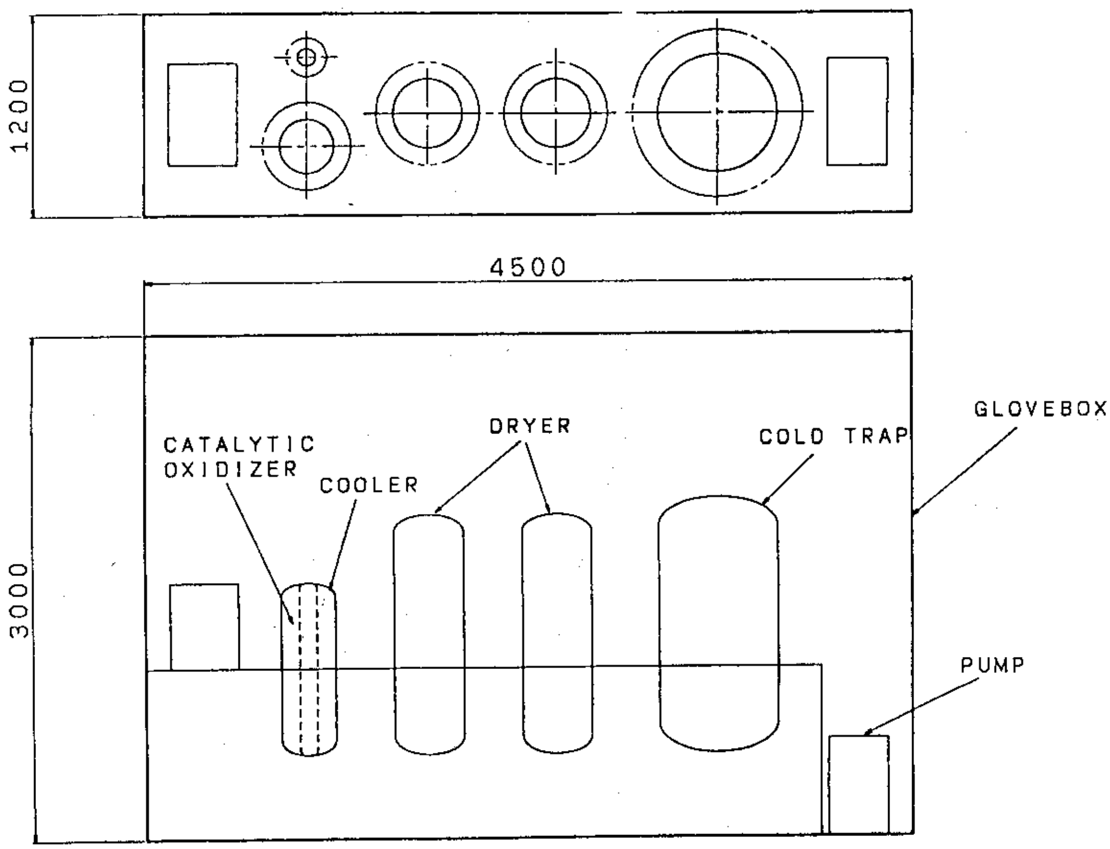


Fig. 3.1.5 Layout plan of tritium recovery system for helium-cooled test module

3.2 Water-cooled Blankets

The water-cooled test module is ancillarily equipped with its own cooling system and tritium recovery system because its test conditions differ from the others. In this section, system configuration, specification of equipments, and layout plan of both systems are described corresponding to a test module of full port size shown in Section 2.2.5.

3.2.1 Test Module Cooling System

3.2.1.1 System Functions

The purposes of test module cooling system are as follows:

(i) to remove heat generated in test module including module box structure, and

(ii) to control the breeder temperature in the projected test condition.

Major parameters of test module cooling system are summarized in Table 3.2.1.

The test module cooling system removes heat generated in the test module, which is evaluated to be 5.81 MW, and transfers the heat to secondary cooling water loop of main reactor heat removal system.

The inlet and outlet water coolant temperatures are 280°C and 320°C, respectively. The coolant pressure is 15 MPa to simulate the DEMO blanket. And the water coolant flow rate of 90 t/h is derived from the heat balance. These coolant conditions keep the breeder zone temperature between 450°C and 950°C, and the structural material temperature below the maximum allowable temperature.

3.2.1.2 System Description

(1) System Configuration

Figure 3.2.1 shows a schematic flow diagram of the test module cooling system. The test module cooling system is made up of a heat transferring water loop and purification unit.

Heat generated in the test module is transferred to the secondary cooling loop of main reactor heat removal system through the water loop. The water loop comprises a cooler, circulators, and a heater. During plasma burning, water leaving the test module flows through cooler and one of circulators and returns to the test module. During dwell time or before plasma burning, bypass line of a main cooler is opened and a heater is put into the line to make up the test condition.

The purification unit circulates the coolant water in a low flow rate compared with main circuit and removes impurities from the water coolant system. Corrosion products, which are estimated as major impurities in the water, are removed by demineralizer.

The water hold up of the test module cooling system is estimated to be about 3000 kg assuming coolant pipe length between the test module and the cooling system room is 100 m each and 200 m in total.

(2) System Performance

Process conditions during plasma burning is shown in Fig. 3.2.1. Heat load to the test module cooling system is 5.81 MW, which is equal to heat load to secondary cooling loop of heat removal system. The test module cooling system circulates water of 15 MPa at a flow rate of 90 t/h with lowering its temperature from 320°C to 280°C at the cooler. Inlet and outlet temperatures of secondary cooling loop water at the cooler are 35°C and 50°C, respectively, and pressure of water is 2.0 MPa. The required flow rate of water is derived as 345 t/h.

3.2.1.3 Components Description

Major components and units included in the test module cooling system are cooler, circulator, heater, and purification unit. Specifications of these components and units are summarized in Tables 3.2.2 through 3.2.6.

3.2.1.4 System Layout

Layout plan of the test module cooling system is shown in Fig. 3.2.2. Required space for the test module cooling system is 15 m × 10 m × 10 m-height.

3.2.2 Test Module Tritium Recovery System

3.2.2.1 System Functions

The purposes of test module tritium recovery system are as follows:

(i) to recover tritium produced in breeder zone of the test module continuously from helium sweep gas stream,

(ii) to analyze tritium recovered and contained in helium gas, and

(iii) to control atmosphere in breeder zone by varying gas flow rate and adjusting gas composition.

Major parameters of test module tritium recovery system are summarized in Table 3.2.7.

Tritium production rate is evaluated using integrated neutron wall load to the breeder zone and local tritium breeding ratio. Successive pulsed plasma operation of 1000 sec burn and 200 sec dwell is assumed to last 1 week. Helium sweep gas is at atmospheric pressure (0.1 MPa) and its flow rate is determined to keep tritiated species partial pressure less than 5 Pa, which is one-twentieth of equilibrium water pressure at 400°C in Li_2O-H_2O-LiOH system. Since hydrogen gas giving H/T ratio of 100 is added to the helium sweep gas to recover tritium efficiently, chemical form of tritium released from the breeder is assumed to be 97% of HT and 3% of HTO based on in-situ tritium recovery experimental data. As water is used as coolant of the test module, water leakage to the test module atmosphere is assumed to be 0.1 g/h. Helium gas flow rate is set to 10 Nm^3/h.

Tritium is recovered with catalytic oxidation followed by adsorption.

3.2.2.2 System Description

(1) System Configuration

Figure 3.2.3 is a schematic flow diagram of the test module tritium recovery system. The test module tritium recovery system is made up of analytical section, tritium recovery section, and gas adjusting section.

Analytical section measures tritium in the form of HT and HTO separately and continuously and is composed of ceramic electrolysis cell, moisture adsorption bed, and ionization chambers.

Tritium recovery section removes tritium from the helium gas stream with catalytic oxidation followed by adsorption. Tritium is recovered as HTO in moisture adsorption bed, and a moisture adsorption bed is designed to be able to be in line over one week continuously. In addition to the moisture adsorption beds, liquid nitrogen cooled cold trap is equipped for additional removal of impurities.

Gas adjusting section controls the sweep gas composition by adding hydrogen gas or so on.

(2) System Performance

Process condition during plasma burning is shown in Fig. 3.2.4.

In nominal test condition, hydrogen gas of 2000 ppm, which makes H/T ratio in the test module atmosphere 100, is added to helium gas stream. Tritium production rate is 5.33×10^{-2} g/h and chemical form of tritium is 97% of HT and 3% of HTO. Assuming water source of 0.1 g/h leaking through the water coolant tubes, water concentration in test module atmosphere is 15 ppm including HTO, which is less than one-twentieth of water equilibrium pressure of Li_2O-H_2O-LiOH system at 400°C. With oxidation followed by adsorption, tritium removal at an efficiency of over 0.999 per path is attained.

3.2.2.3 Components Description

Major components composing the test module tritium recovery system are catalytic oxidizer, cooler, dryer, pump, and cold trap. Specifications of these components are summarized in Tables 3.2.8 through 3.2.12.

3.2.2.4 System Layout

Layout plan of the test module tritium recovery system is shown in Fig. 3.2.5. As the test module tritium recovery system handles tritium of relatively high concentration, most of the system is contained in glove box of 1.2m × 4.5m × 3.0m-high in dimension.

The test module tritium recovery system is operated remotely. In the course of maintenance and repair, workers enter the system room and access the tritium recovery system in the glove box.

Table 3.2.1 Major parameters of cooling system for water-cooled test module

Heat Load	
Test Module Box	1.70 MW
Breeding Zone	4.11 MW
Total	5.81 MW
Type of Coolant	Pressurized Water
Coolant Flow Mode	Parallel
	(→ Test Module Box →)
	(→ Breeding Zone →)
Temperature	
Inlet	280°C
Outlet	320°C
Pressure	15 MPa
Flow Rate	90.0 t/h
Heat Sink	Secondary Cooling Loop of ITER Heat Removal System
Auxiliary System	Purification Unit

Table 3.2.2 Specifications of main cooler

Number of Components	1
Type	Shell and Tubes Type
Heat Load	6.0 MW
Primary Fluid	
Type of Fluid	Water
Inlet Temperature	280°C
Outlet Temperature	320°C
Pressure	15 MPa
Flow Rate	90.0 t/h
Secondary Fluid	
Type of Fluid	Water
Inlet Temperature	35°C
Outlet Temperature	50°C
Pressure	2 MPa
Flow Rate	345 t/h

Table 3.2.3 Specifications of circulators

Number of Components	2
Type	Axial
Flow Rate	60.0 t/h
Pressure of Coolant	9 MPa
Delivery Head	TBD
Temperature of Coolant	280°C

Table 3.2.4 Specifications of heater

Number of Components	1
Type	Immersion Type Electric Heater
Heating Power	1 MW

Table 3.2.5 Specifications of pressurizer

Number of Units	1
Type	Vertical Cylinder with Immersion Heater
Volume	0.5 m^3

Table 3.2.6 Specifications of purification unit

Number of Units	1
Fluid	Water
Flow Rate	1.0 t/h
Pressure at the Interface	15 MPa
Temperature at the Interface	280°C
Purification Method	Deaeration and Demineralization

Table 3.2.7 Major parameters of tritium recovery system for water-cooled test module

1. Tritium Production Rate	
Neutron Wall Load	1.2 MW/m^2
Surface Area of Test Module	
Facing to Plasma	3.5 m^2
Plasma Burn Time	1200 sec
Plasma Dwell Time	200 sec
Duration of Continuous Operation	1 week
Local Tritium Breeding Ratio	1.59
Tritium Production Rate	
(during Burning)	5.33×10^{-2} g/h
2. Breeding Zone	
Breeder	Li_2O 1 mmϕ Pebble (85%T.D.)
Multiplier	Be 1 mmϕ Pebble
Volumetric Ratio of	
Breeder/Multiplier	1/3
Packing Fraction	60%
Weight of Breeder	420 kg
Weight of Multiplier	1360 kg
Temperature of Breeding Zone	400 to 1000°C
3. Sweep Gas	
Fluid	Helium Gas
Pressure	0.1 MPa
Temperature	
Outlet of Breeding Zone	400°C
Inlet of Breeding Zone	R.T.
Flow Rate	10 Nm^3/hr to keep tritiated species partial pressure less than 5 Pa

Table 3.2.7 (Continued)

Composition of Impurities at the Inlet of Breeding Zone	
Hydrogen	Nominal 2000 ppm
Water	less than 1 ppm
Others	TBD
Chemical Form of Released Tritium from Breeder	HT/HTO = 97/3 *)
Release Rate of Impurity	H_2O: 1.0×10^{-1} g/h Kr, Xe etc. (not used in design)
4. Tritium Recovery	
Tritium Recovery Method	Oxidation Followed by Adsorption
Tritium Recovery Efficiency	Nominal 0.999 (one path) depending on hydrogen content

*) H. Yoshida et al., Proc. Fusion Reactor Blanket and Fuel Cycle Technology, p158, Tokai, Oct. 1986.

Table 3.2.8 Specifications of catalytic oxidizer

Number of Components	1
Fluid	Helium Gas
Flow Rate	1.0 Nm^3/h
Pressure	0.1 MPa
Operation Temperature	473 K
Catalyst	Pd-Pt Coated Alumina
Hydrogen Concentration Reduction Factor	10000
Size	100 mmϕ × 1000 mmL

Table 3.2.9 Specifications of cooler

Number of Components	1
Type	Water Cooled Shell and Tubes Heat Exchanger
Fluid	Helium Gas
Flow Rate	10 Nm^3/h
Pressure	0.1 MPa
Size	300 mmϕ × 1000 mmL

Table 3.2.10 Specifications of dryer

Number of Components	2 (used alternatively)
Type	Adsorbent Packed Bed
Fluid	Helium Gas
Flow Rate	10 Nm^3/h
Pressure	0.1 MPa
Operation Temperature	293 K
Duration of Operation	1 week per one dryer
Adsorbent	Molecular Sieve Type 5A
Size	400 mmϕ × 1400 mmL

Table 3.2.11 Specifications of pump

Number of Components	1
Type	Wabble Pump
Fluid	Helium Gas
Flow Rate	Nominal 10 Nm^3/h
Pressure (inlet/outlet)	0.08/0.1 MPa
Size	635 × 350 × 580 mm

Table 3.2.12 Specifications of cold trap

Number of Components	2
Type	LN_2 Cooled Adsorbent Packed Bed
Fluid	Helium Gas
Flow Rate	10 Nm^3/h
Pressure	0.1 MPa
Operation Temperature	77 K
Adsorbent	Molecular Sieve Type 5A
Size	700 mmϕ × 1500 mmL

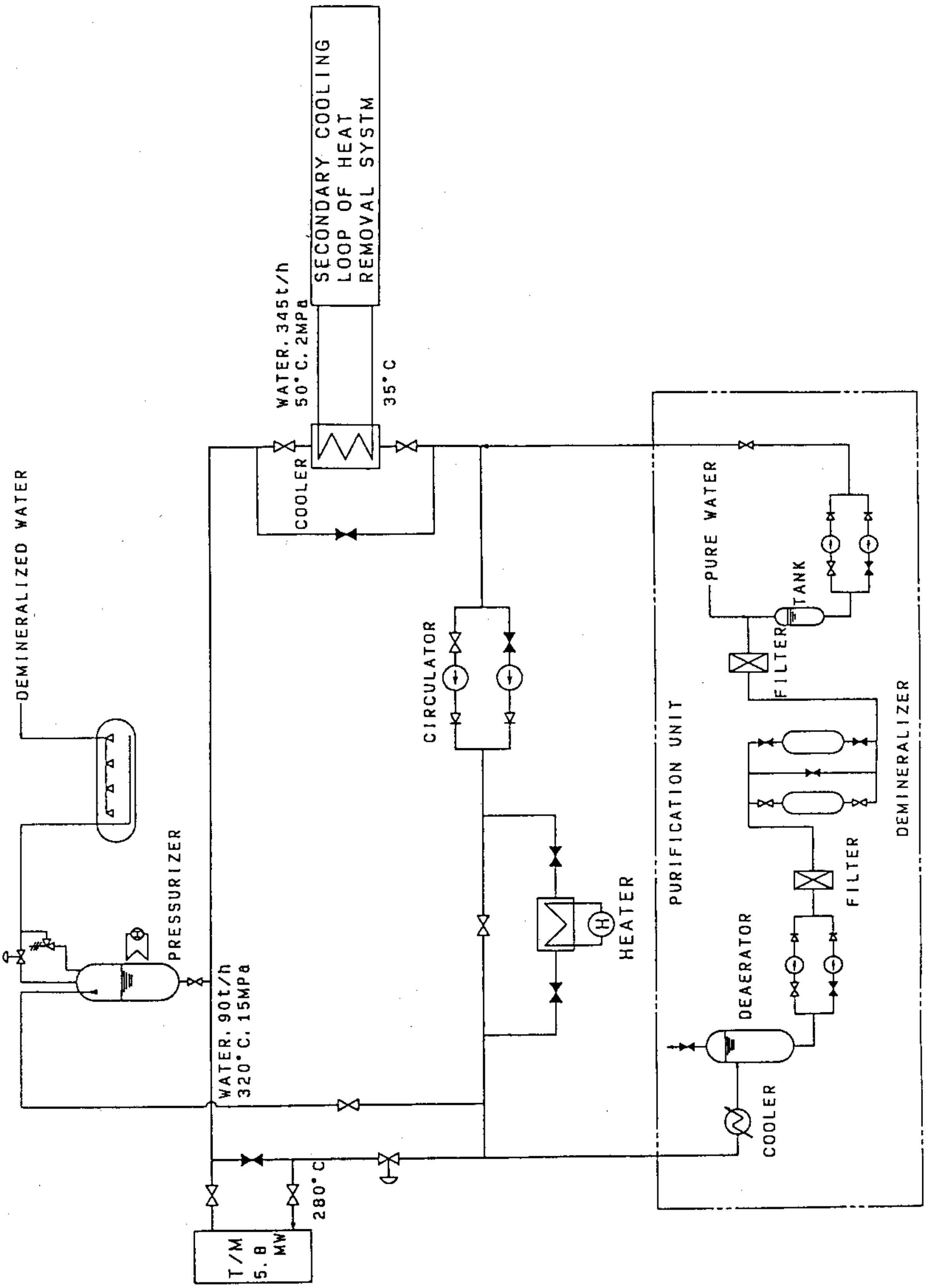
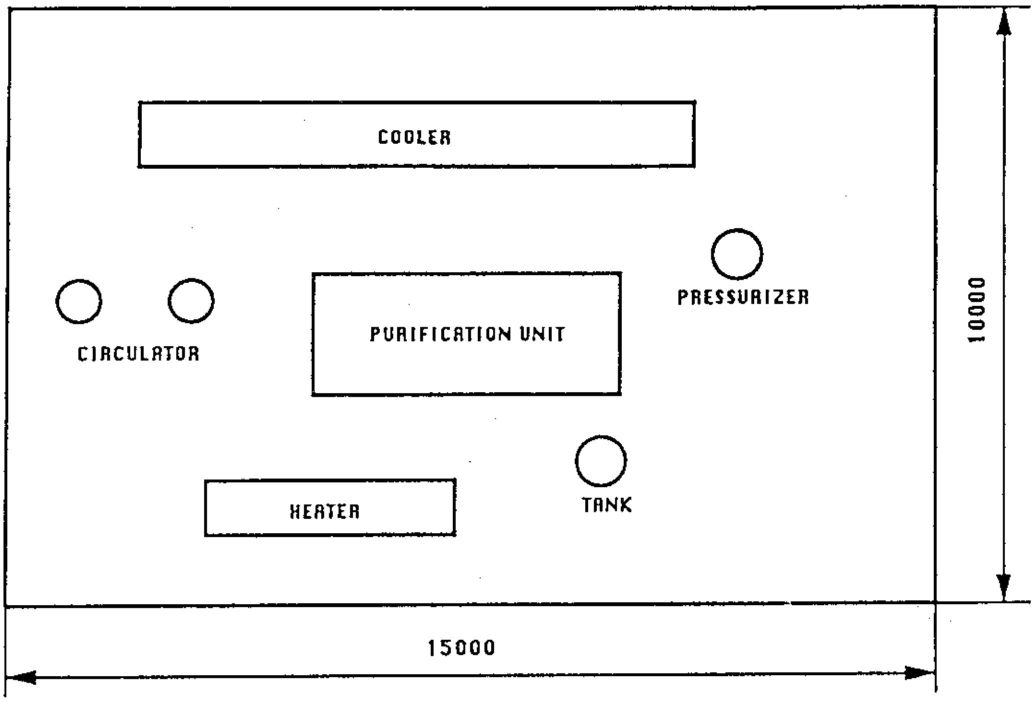


Fig. 3.2.1 Schematic flow diagram of cooling system for water-cooled test module



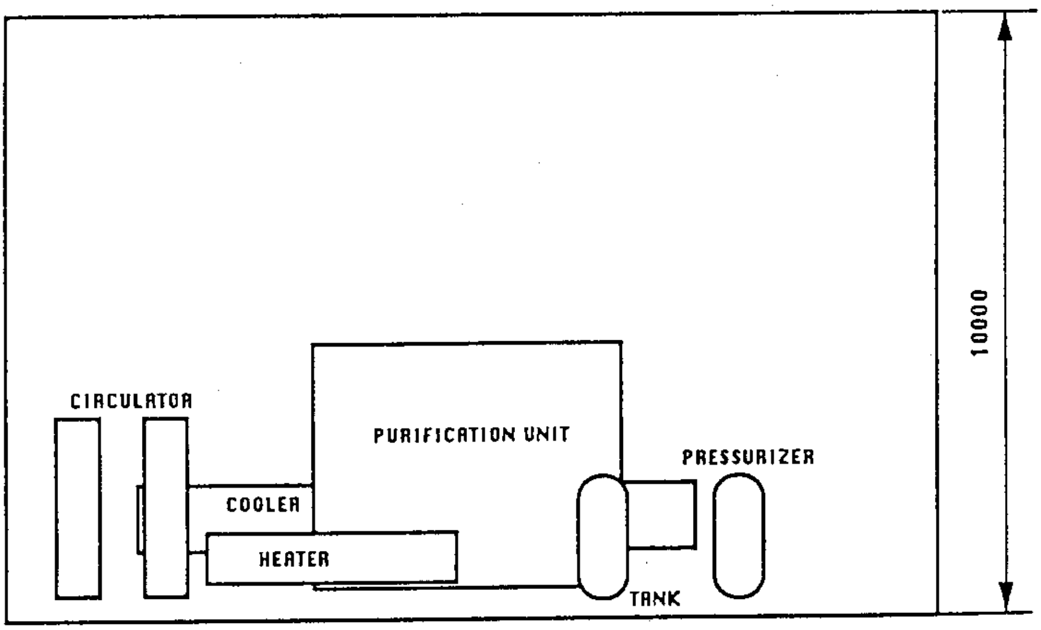
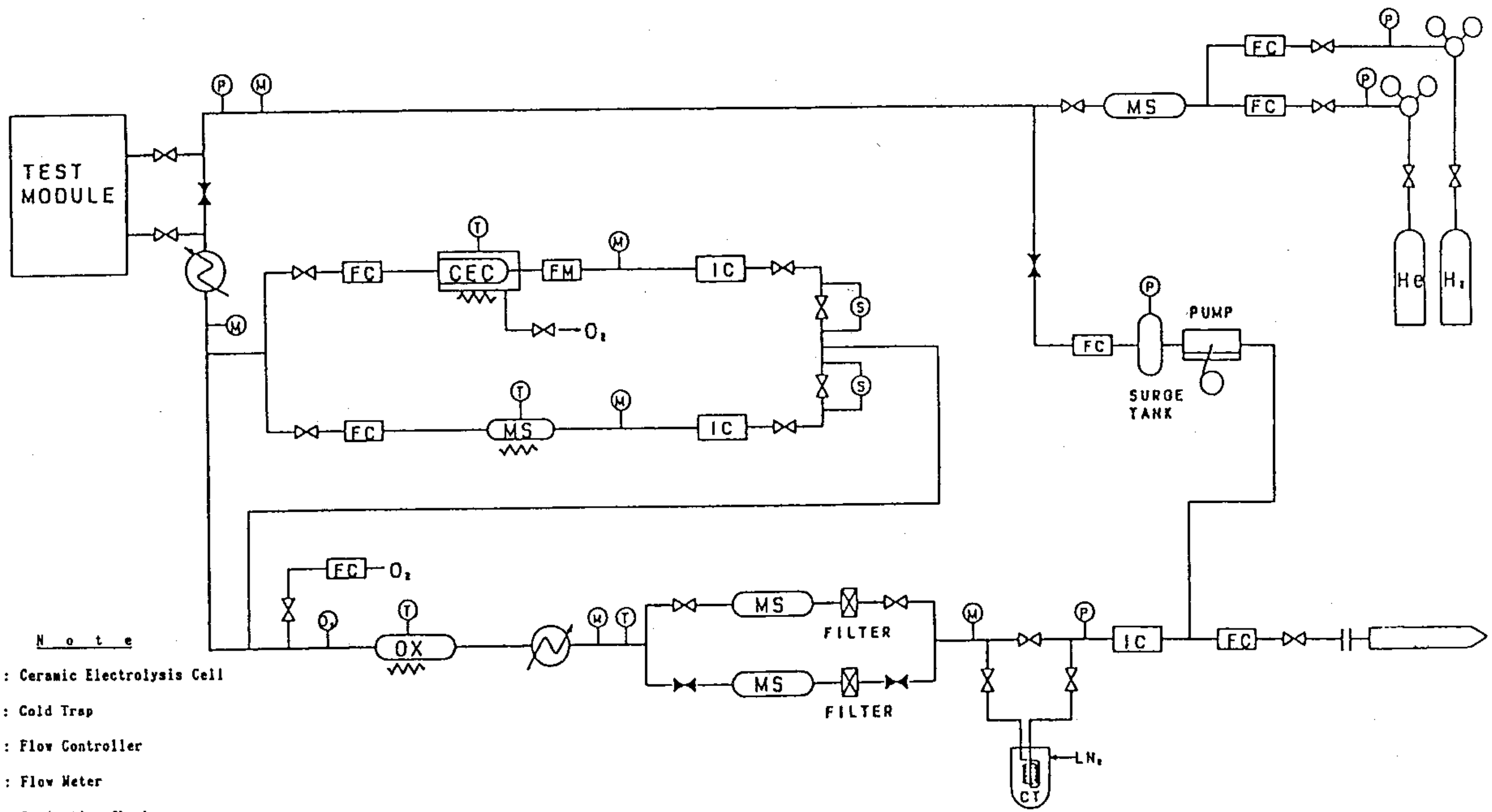


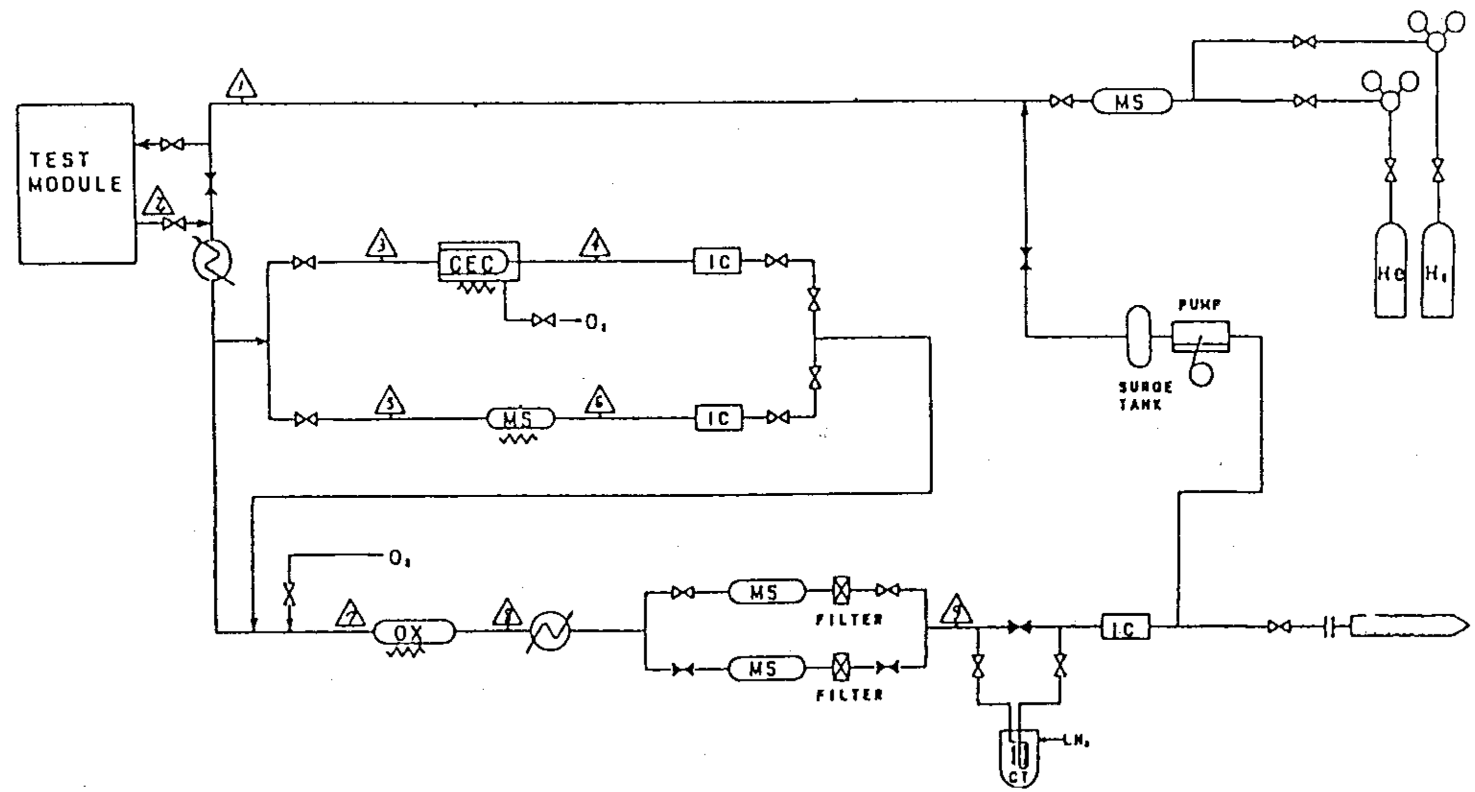
Fig. 3.2.2 Layout plan of cooling system for water-cooled test module



N o t e

CEC : Ceramic Electrolysis Cell
CT : Cold Trap
FC : Flow Controller
FM : Flow Meter
IC : Ionization Chamber
M : Hygrometer
MS : Moisture Adsorption Bed
OX : Catalytic Oxidizer
P : Pressure Gauge
S : Gas Sampler for Analysis
T : Thermometer

Fig. 3.2.3 Schematic flow diagram of tritium recovery system for water-cooled test module



Position	1	2	3	4	5	6	7	8	9
	TM Inlet	TM Out	CEC In	CEC Out	MS In	MS Out	OX In	OX Out	MS Out
Temperature (K)	293	673	293	973	293	293	293	473	293
Pressure (MPa)	~0.1	←	←	←	←	←	←	←	←
Flow Rate (Nm^3/H)	10.0	←	3.0×10^{-3}	←	3.0×10^{-3}	←	10.0	←	←
Hydrogen Conc. (vpm	2.0×10^{3}	2.0×10^{3}	←	←	←	←	←	2.0×10^{-1}	←
Water Conc. (vpm)	1.0	1.5×10^{1}	←	1.0	1.5×10^{1}	1.0	1.5×10^{1}	2.0×10^{3}	1.0
HT Conc. (Ci/Nm^3)	5.0×10^{-3}	5.0×10^{1}	←	5.2×10^{2}	5.0×10^{1}	←	←	5.0×10^{-2}	←
HTO Conc. (Ci/Nm^3)	2.6×10^{-2}	1.6	←	1.1×10^{-1}	1.6	1.1×10^{-1}	1.6	5.1×10^{1}	2.6×10^{-2}

Tritium Production Rate
5.33×10^{-2} g/h
Chemical Form of Tritium
HT / HTO = 97 / 3
Chemical Form of Tritium
H / T = 100
Water Leakage Rate
1.0×10^{-1} g/h

Fig. 3.2.4 Chemical flow diagram of tritium recovery system for water-cooled test module

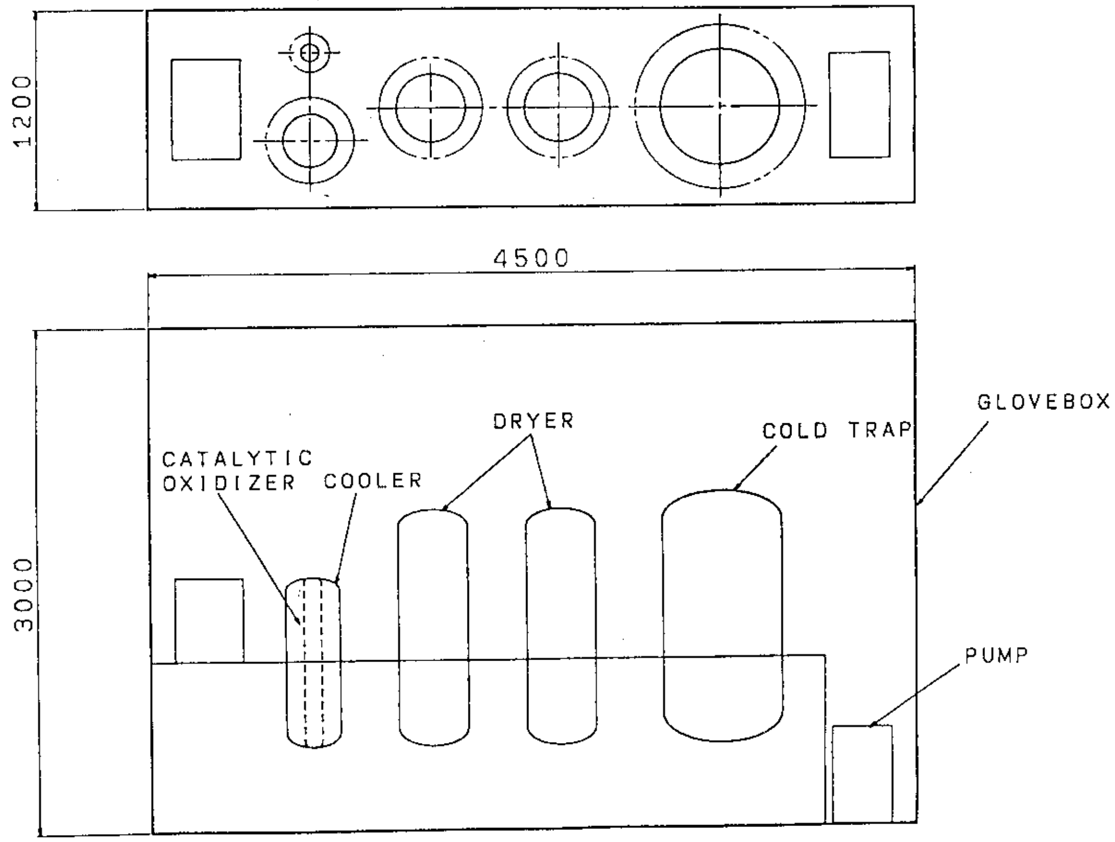


Fig. 3.2.5 Layout plan of tritium recovery system for helium-cooled test module

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