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TEST PROGRAM DEVELOPMENT FOR ITER BLANKET DESIGN

March 1995

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Test programs for water-cooled and helium-cooled blankets have been developed. Following the description of ITER Experimental and DEMO reactors and blankets proposed for it, which are recently developed in Japan, test items and their features are indicated, Tests required for the development of ITER and DEMO blankets consist of neutronics test, scoping test, performance and performance verification tests, reliability enhancement test, and segment demonstration test. Procedure for these tests has been evolved on the assumption that neutron fluences onto test ports during the BPP and the EPP are 0.1 MWa/m^2 and 1-3 MWa/m^2, respectively. Comprehensive tests of the ITER driver blanket during the BPP are also proposed, which consist of neutronics and performance verification tests.

keywords : ITER Blanket Design, DEMO Reactor, Tritium Breeding, Neutron Wall Loading, Design Concept, Water Cooled Blanket

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\+ Department of ITER Project

* Kawasaki Heavy Industries Co.

ＩＴＥＲブランケット設計のための工学試験計画

日本原子力研究所那珂研究所核融合工学部
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（1995年２月10日受理）

核融合実験炉及び原型炉用ブランケットを対象としたＩＴＥＲでの工学試験計画について検討した。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，各ブランケットの原型炉における設計例を示すと共に，ＩＴＥＲでの試験項目，試験仕様，試験手順等についてまとめた。試験は，初期の基本性能段階（ＢＰＰ）から実施するものとし，ニュートロニクス試験およびサーベイ試験，性能試験／性能確証試験，信頼性試験，セグメント試験を拡張性能段階（ＥＰＰ）にかけて順次行うものとした。また，拡張性能段階において装荷が予定されている，ＩＴＥＲ自身の燃料トリチウムを生産するドライバーブランケットについても，ＢＰＰにおける試験の実施を提案した。

この報告はＩＴＥＲ工学Ｒ＆Ｄの1994年タスクの一部として実施されたものである。
（Task no：G16TD55. 1D N0；D2）
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2.1 Design Concept of ITER Blanket

Design and R&D activities on ceramic breeder blanket have been progressed in JAERI (Japan Atomic Energy Research Institute), aiming at application to breeding blankets of fusion experimental reactors and DEMO reactors. Ceramic breeder blanket has a number of attractive features such as abundant data basis, high safety, and good DEMO relevancy. Irradiation degradation of both of breeding materials and neutron multipliers is one of the critical issues for this type of blanket, and usage of these materials as a form of small pebbles has been proposed to accommodate dimensional change by irradiation effects. Pebble-bed concept has also another capability to be convertible from non-breeding to breeding blankets without large hardware changeout.

A layered pebble bed type ceramic breeder blanket with water cooling is a leading candidate concept for the breeding blanket of fusion experimental reactors in JAERI, where beryllium and ceramic breeder (Li_2O as a prime candidate) pebble bed layers are alternately arranged [1]. Design activities have been concentrated on improvement of the design by conducting detailed analyses. This concept can meet the specifications and requirements of International Thermonuclear Experimental Reactor (ITER), and has been proposed for the breeding blanket of ITER Experimental Reactor.

A layered pebble bed type ceramic breeder blanket with water cooling proposed by JAERI has been improved based on detailed analyses and consideration on fabrication procedure. In this concept, beryllium and ceramic breeder (Li_2O as a prime candidate) pebble bed layers are alternately arranged, where the beryllium layer works as a thermal resistance layer between the breeder layer and a cooling panel as well as a neutron multiplier.

Figure 1 shows a cross-sectional view at midplane (a) and a cut-away lateral view (b) of the outboard side module. Breeding region is

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Figure 1 shows a cross-sectional view at midplane (a) and a cut-away lateral view (b) of the outboard side module. Breeding region is divided into three in the poloidal direction to accommodate poloidal distribution of the neutron wall load and to improve fabricability. Cooling channels and helium purge gas lines are connected via intermediate plenums placed between each poloidal region. Rear part of the blanket box is utilized as coolant manifolds for the first wall and cooling panels in the breeding region. The number of the breeder layer is three, as shown in Fig. 1(a), and the breeder pebbles are housed in the steel can of 1 mm thick to avoid compatibility problem with beryllium and also to improve safety by avoiding direct contact with cooling water in case of an accident. Packing fraction of 65% is assumed for both the breeder and beryllium beds, and 50% Lithium-6 enrichment is assumed to obtain higher tritium breeding performance.

Neutronics, thermal and mechanical performances of this type of breeding blanket are described in Ref. [1] together with considerations of Li burn-up and tritium inventory under the conditions of the average neutron wall load of 1MW/m^2, the same as that of ITER/EDA for nominal operation condition with fusion power of 1500MW. Local TBR is given as 1.13 at the outboard midplane.

2.2 Design Concepts of DEMO Reactor and Blanket

Figure 2 shows a recently developed conceptional design for a DEMO reactor, SSTR (Steady State Tokamak Reactor), in Japan, and Table 1 its major parameters [2]. Fusion and thermal powers of this reactor have been estimated to be 3000 MW and 3700 MW, respectively. Maximum neutron wall load and peak surface heat flux to the first wall are 5 MW / m^2 and 0.5 MW/m^2, respectively.

Figure 3 shows a water-cooled blanket conceptional design proposed for the SSTR [3]. Table 2 summarizes major design parameters of this blanket. The coolant is pressurized (15 MPa) light water with inlet/outlet temperatures of 285/325 °C so that power generation by a conventional PWR balance of plant is applicable. As for the structural material, the use of a ferritic steel, F82H, is taken into account in terms of its higher heat load durability (higher thermal conductivity) and higher neutron fluence to the swelling lifetime than austenitic stainless steel for experimental reactors. This blanket consists of a replaceable blanket facing the plasma and a permanent blanket placed behind the replaceable blanket. In the replaceable blanket, small spherical ceramic breeder filled behind the first wall and in front of the back wall. In-between these two breeder zones, Be blocks zone of 10 cm thickness is provided. The permanent blanket contains ceramic breeder also in the shape of small spheres but not Be. The first candidate for the ceramic breeder material is Li_2O. However, other ceramic breeders, e.g., Li_4SiO_4, Li_2ZrO_3 and $LiAlO_2$, will give almost the same performance. The final decision on the breeder to be used will be made through R&D's including tests and experiences in experimental reactors such as ITER.

There are several other concepts of the DEMO blanket and even of the DEMO reactor itself studied in Japan [4,5]. Through detail design studies and R&D's, the concept of blanket to be installed in DEMO reactors will be converged.

2.3 Test Program Description

2.3.1 Tests for DEMO Blanket

Test program in ITER should be closely related to out-of-ITER test programs such as out-of-reactor and in-fission reactor tests to minimize the number of modules to be tested in ITER. Based on this premise, tests in ITER will be performed mainly for investigating blanket characteristics under 14 MeV neutron irradiation (nuclear heat generation and irradiation damage) and demonstrating blanket performance with synergetic effects of nuclear heating generation, surface heat flux from plasma, electro-magnetic force, and irradiation damage.

The following tests should be performed in ITER, of which features are summarized in Table 3:

1) Neutronics test
2) Scoping test
3) Performance and performance verification tests
4) Reliability enhancement test
5) Segment demonstration test.

At the beginning of DT operation in the Basic Performance Phase (BPP), neutronics test can be started. This test is to characterize neutronic environment and obtain data for fission/fusion correlation. System check-out will also be carried out during this test. Neutron fluence required for neutronics test will be very low, e.g., several tens of shots at most.

Following the neutronic test, scoping test will be performed to evaluate basic blanket performance and select 1- 2 candidate concepts for further tests. The performance to be mainly looked at in this test is breeder temperature control and release of tritium generated in the blanket. In case of long off-burn time such as ≥ 1000 sec presently assumed in ITER, burn time > 1000-10000 sec will be required to reach temperature saturation in the blanket. (1000 sec for front part and 3000-10000 sec for back part) This requirement to burn time is the same for above-listed tests except for the neutronics test. Total neutron fluence required for the scoping test will be 0.01-0.02 MW/m^2. Therefore, this test can be performed during the BPP.

Based on the results of the scoping test, performance verification test will be conducted aiming at confirmation of the blanket basic performance with higher neutron fluence resulting in possible indication of irradiation effects on breeder properties. Neutron fluence expected to perform this test is ≥ 0.2 MWa/m^2. However, since the assumption given on the neutron fluence during the BPP is 0.1 MWa/m^2 (that is corresponding to 800 hours of operation time), the test with the used and selected module in the scoping test can be continued up to about 0.1 MWa/m^2 during the BPP as performance test. Then, the performance verification test with module(s) improved based upon the results of the performance test started at the beginning of the EPP.

Reliability verification test will follow the performance verification test. With larger module(s), overall blanket performance will be confirmed with higher neutron fluence in this test. The first wall area of this test module needs to be at least 1 m x 2 m (2 m^2). With this size, the test module might be installed in the horizontal port of ITER, which remarkably ease its replacement. Neutron fluence up to 1 MWa/m^2 will give sufficient experience of blanket operation and performance evaluation under irradiation except for irradiation effects on structural materials.

Segment verification test will be performed at the final stage of the EPP with available neutron fluence of 0.2-1.0 MWa/m^2. The objectives of this test are to demonstrate component integration into tokamak and overall blanket performance with poloidal distribution of wall loads and under electromagnetic loads during plasma disruptions.

2.3.2 Tests for Driver Blanket

Since the driver blanket which produces tritium for ITER EPP operation will be installed in ITER after BPP operation, tests of the driver blanket should be incorporated during the BPP. The ITER driver blanket might use water coolant considering the consistency with the shielding blanket installed during the BPP. Therefore, tests of the driver blanket might be categorized in water-cooled blanket tests. The tests will consist of neutronics test and performance verification test similar to those for DEMO blankets mentioned above. As the available neutron fluence is limited during the BPP, the number of candidates for the driver blanket should be sufficiently converged so that the scoping test in ITER can be eliminated or significantly diminished. The tests of the driver blanket is very important to operate ITER itself and verify the blanket performance.

2.3.3 Test Procedure

Test procedure proposed is shown in Fig. 4, in which above-mentioned tests will be performed in series from the BPP through the EPP. This procedure is based upon the assumption that one test port is allocated to the water-cooled DEMO blanket with ceramic breeder. For the driver blanket test, another port should be allocated to carry out the neutronics and performance verification tests in series through the BPP. If only one port is possible to be used for both of the driver and DEMO blanket tests, a procedure indicated in Fig. 5 can be taken with some risk (shortage of neutron fluence) on the achievement of reliability enhancement test of DEMO blanket. In this case, the performance test of DEMO blanket can be eliminated by being united with the performance verification test.

3. Helium-cooled Blanket

3.1 Blanket Concepts for DEMO Reactor

Figure 6 shows a helium-cooled blanket concept proposed for the SSTR [6]. Table 4 summarizes major design parameters of this blanket. The coolant is not exactly helium but the mixture of helium and solid particulate to enhance heat removal capability. As a candidate material

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for the solid particulate, SiC is taken into consideration. Coolant temperatures are 400/580 °C for the first wall inlet/outlet and 580/700 °C for the breeding region inlet/outlet. As for the structural material, an intermetallic compound, TiAl that is light yet highly heat resistant , is employed. The blanket consists of a replaceable blanket facing the plasma with the function of tritium breeding and a permanent blanket which is a non-breeding shield structure placed behind the replaceable blanket. In the replaceable (breeding) blanket, small spherical ceramic breeder filled in an annular region around the coolant tube. The first candidate for the ceramic breeding material is Li_2O. However, other ceramic breeders, e.g., Li_4SiO_4, Li_2ZrO_3 and $LiAlO_2$, will give almost the same performance. The final decision on the breeder to be used will be made through R&D's including tests and experiences in experimental reactors such as ITER. Surrounding the breeder annulus, Be blocks as neutron multiplier are packed. The breeder and Be are separated by a clad tube placed in-between them.

Figures 7 and 8 also show helium-cooled blanket concepts studied in Japan [7,8]. Table 5 summarizes major design parameters of these blankets. The blanket shown in Fig. 7 is based on tube-in-shell type BOT (Breeder-Out of-Tube) concept, and the other directly-cooled BIT (Breeder-In-Tube) concept. Through design studies in more detail and R&D's, the concept of blanket to be installed in DEMO reactors will be converged.

3.2 Test Program Description

Tests to be performed and their features for helium-cooled DEMO blankets are the same as those for water-cooled blanket described in 1.3.1 and Table 3. Test procedure indicated in Fig. 4 is also applied to the helium-cooled blanket based upon the assumption that one port is allocated for tests of helium-cooled ceramic breeder blanket.

Acknowledgment

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Table 1 Major parameters of Steady State Tokamak Reactor (SSTR)

Fusion power	3000 MW
Thermal power	3710 MW
Major radius	7.0 m
Minor Radius	1.75 m
Elongation	1.8
Plasma current	12 MA
NBI power	60 MW
Average neutron wall load	5 MW/m^2
Peak surface heat flux to first wall	0.5 MW/m^2

Table 2 Major design parameters of water-cooled blanket for SSTR

Structural material	F82H (ferritic steel)
Operating temperature	< 500 °C
Coolant	H_2O
Pressure	15 MPa
In/out temperature	285/325 °C
Particle size	50 mm
Solid to gas mass ratio	13
Breeder material	Li_2O sphere (< 1 mm dia.)
Density	85 % T.D.
^6Li enrichment	natural
Operating temperature	450-950 °C
Multiplier material	Be small blocks
Blanket thickness	
Replaceable	0.2 m
Permanent	0.3 m
Net tritium breeding ratio	1.2

Table 3 Proposed test for water-cooled ceramic breeder blanket -1/6-

Test name
Neutronics test

Test Objectives
- Correlation between fission and fusion environments
- Verification of nuclear data and analyses on nuclear heating and tritium production

Measurements
- Nuclear heating rate
- Tritium generation rate
- Neutron and gamma distributions
- Induced activity

Module width x height x depth
1.0 m x 1.0 m x 0.6 m

Boundary condition with plasma
With plasma exposure

Test features

- Individual test time	:	short
- Number of tests	:	< 5
- Iterations each test	:	< 10
- Total fluence	:	< 0.001 MWa/m^2

Required ancillary system
Cooling system

Table 3 Proposed test for water-cooled ceramic breeder blanket -2/6-

<u>Test name</u>

Scoping test

<u>Test objectives</u>

Evaluation of breeder temperature control, tritium release performance and tritium inventory, and selection of 1(-2) candidate concept(s) for further tests

<u>Measurements</u>

- Breeder/multiplier temperatures
- Tritium release rate
- Tritium inventory

<u>Module width x height x depth</u>

1.0 m x 0.5 m x 0.15-0.6 m

<u>Boundary condition with plasma</u>

With plasma exposure

<u>Test features</u>

- Individual test time	:	1000-10000 sec
- Number of tests	:	3-5
- Iterations each test	:	5-10
- Total fluence	:	0.01-0.02 MWa/m^2

<u>Required ancillary system</u>

- Cooling system and tritium recovery system

<u>Remarks</u>

- Burn time ≥ 1000-3000 sec is required for temperature saturation in case of long dwell time (e.g., ≥ 500 sec).

Table 3 Proposed test for water-cooled ceramic breeder blanket -3/6-

Test name

Performance test

Test objectives

To evaluate tritium breeding and release performance of candidates selected from the scoping test with more operation experience and thermal cycling

Measurements

- Heat generation and removal performance
- Thermal-hydraulic and thermo-mechanical behaviors
- Breeder/multiplier temperatures
- Tritium release rate and confirmation of continuous tritium recovery
- Tritium inventory
- Behaviors due to thermal cycling

Module width x height x depth

1.0 m x 0.5 m x 0.15-0.6 m

Boundary condition with plasma

With plasma exposure

Test features

Individual test fluence	:	~0.1 MWa/m^2
Number of tests	:	1(-2)
Iterations each test	:	1
Total fluence	:	~0.1 MWa/m^2

Required ancillary system

Cooling system and tritium recovery system

Remarks

- Burn time ≥ 1000-3000 sec is required for temperature saturation in case of long dwell time (e.g., ≥ 500 sec).
- Possible to be in succession from the scoping test using the selected module

Table 3 Proposed test for water-cooled ceramic breeder blanket -4/6-

Test name

Performance verification test

Test objectives

To verify tritium breeding and release performance of the candidate(s) and to evaluate these performance with possible indication of some irradiation effects (e.g., cracking, sintering, grain growth)

Measurements

- Heat generation and removal performance
- Thermal-hydraulic and thermo-mechanical behaviors
- Breeder/multiplier temperatures
- Tritium release rate and confirmation of continuous tritium recovery
- Tritium inventory
- Irradiation damage

Module width x height x depth

1.0 m x 1.0 m x 0.6 m

Boundary condition with plasma

With plasma exposure

Test features

Individual test fluence	:	~ 0.2 MWa/m^2
Number of tests	:	1(-2)
Iterations each test	:	1
Total fluence	:	~ 0.2 MWa/m^2

Required ancillary system

Cooling system and tritium recovery system

Remarks

- Burn time ≥ 1000-3000 sec is required for temperature saturation in case of long dwell time (e.g., ≥ 500 sec).

Table 3 Proposed test for water-cooled ceramic breeder blanket -5/6-

Test name

Reliability enhancement test

Test objectives

To obtain more confirmation of overall blanket performance with higher neutron fluence and to show possibility of electricity generation by obtaining sufficient outlet coolant temperature

Measurements

- Heat generation and removal performance
- Thermal-hydraulic and thermo-mechanical behaviors
- Breeder/multiplier temperatures
- Tritium release rate and confirmation of continuous tritium recovery
- Tritium inventory
- Effects of higher Li burn-up in breeders and irradiation damage

Module width x height x depth

~1.0 m x 2.0 m x 0.6 m

Boundary condition with plasma

With plasma exposure

Test features

Individual test fluence	:	~1(-2) MWa/m^2
Number of tests	:	1(-2)
Iterations each test	:	1
Total fluence	:	~1(-2) MWa/m^2

Required ancillary system

Cooling system, tritium recovery system
(electricity generation system)

Remarks

- Burn time ≥ 1000-3000 sec is required for temperature saturation in case of long dwell time (e.g., ≥ 500 sec).

Table 3 Proposed test for water-cooled ceramic breeder blanket -6/6-

Test name

Segment demonstration test

Test objectives

To demonstrate component integration into tokamak and overall blanket performance with poloidal distribution of wall loads and under electromagnetic loads during disruptions

Measurements

- Heat generation and removal performance
- Thermal-hydraulic and thermo-mechanical behaviors
- Breeder/multiplier temperatures
- Tritium release rate and confirmation of continuous tritium recovery
- Tritium inventory

Module width x height x depth

~1.0 m x ~12 m x 0.6 m

Boundary condition with plasma

With plasma exposure

Test features

Individual test fluence	:	0.2-0.5 MWa/m^2
Number of tests	:	1
Iterations each test	:	1
Total fluence	:	0.2-0.5 MWa/m^2

Required ancillary system

Cooling system, tritium recovery system (electricity generation system)

Remarks

- To be performed at the final stage of possible extended operation

Table 4 Major design parameters of helium-cooled blanket for SSTR

Structural material	TiAl
Operating temperature	< 900 °C
Breeder material	Li_2O sphere (< 1 mm dia.)
Density	90 % T.D.
^6Li enrichment	natural
Operating temperature	400-950 °C
Multiplier material	Be metal block
Coolant He + SiC particulate	
Pressure 5 MPa	
In/out temperature	400/580 °C (first wall)
	580/700 °C (breeding zone)
Particle size	50 mm
Solid to gas mass ratio	13
Blanket thickness	
Replaceable (breeding)	0.3 m
Permanent (non-breeding)	0.2 m
Blanket lifetime	
Replaceable	2 years (1.4 FPY)
Permanent	30 years (21 FPY)
Tritium breeding ratio	1.38 (local)
Neutron energy multiplication	1.35

Table 5 Major design parameters of helium-cooled blanket for DEMO reactor

Blanket concept	BOT	BIT
Thermal power [MW]	3720	3510
Average neutron wall load [MW/m^2]	3.3	3.3
Heat flux to first wall [MW/m^2]	0.9	0.9
Nuclear heating rate in the first wall [MW/m^3]	27	27
Structural material	Mo-alloy	Mo-alloy
Maximum temperature [°C]	852	852
Coolant	He	He
Pressure [MPa]	9	9
Inlet/outlet temperature [°C]	400/700	400/700
Breeder material	Li_2O	$LiAlO_2$
^6Li enrichment [%]	30	95
Operating temperature [°C]	400-950	400-725
Multiplier material	Be	Be
Outboard/inboard local tritium breeding ratio	1.37/1.32	1.24/1.17
Net tritium breeding ratio	1.19	1.06
Tritium recovery	continuous by He sweep gas	continuous by He coolant

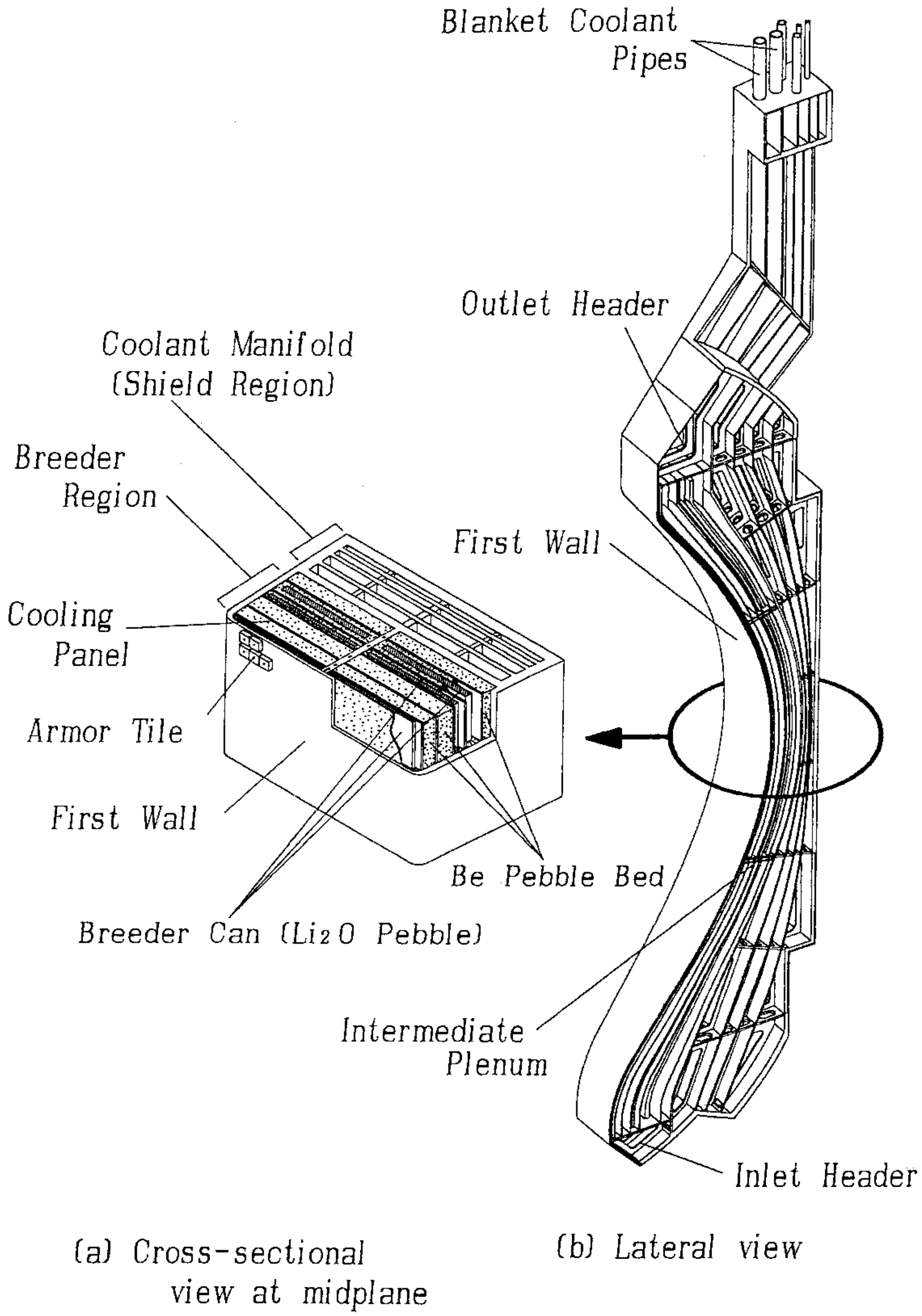

Fig. 1 Cut-away-views of outboard blanket module of layered pebble bed design.

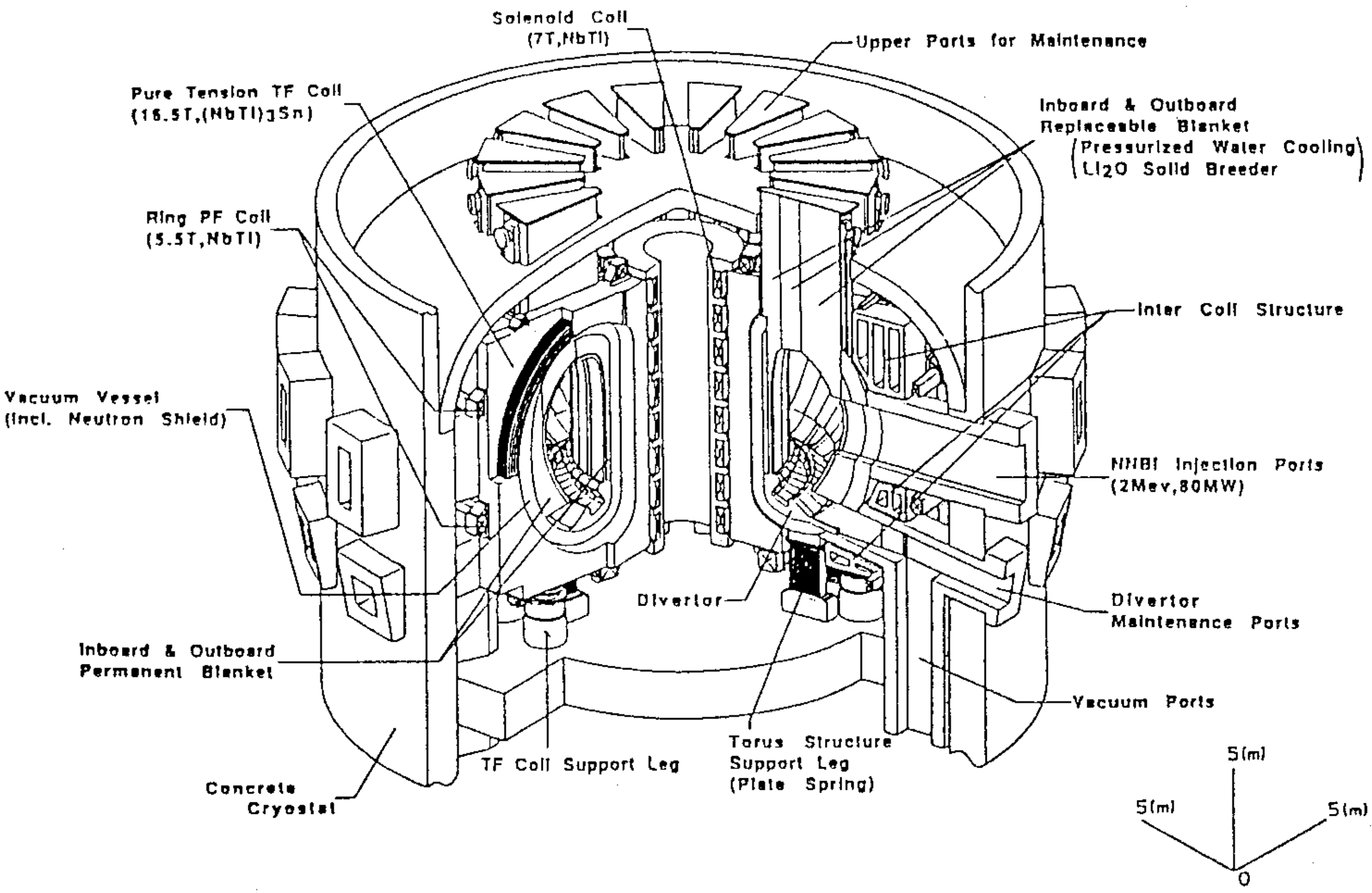

Fig. 2 Steady State Tokamak Reactor (SSTR)

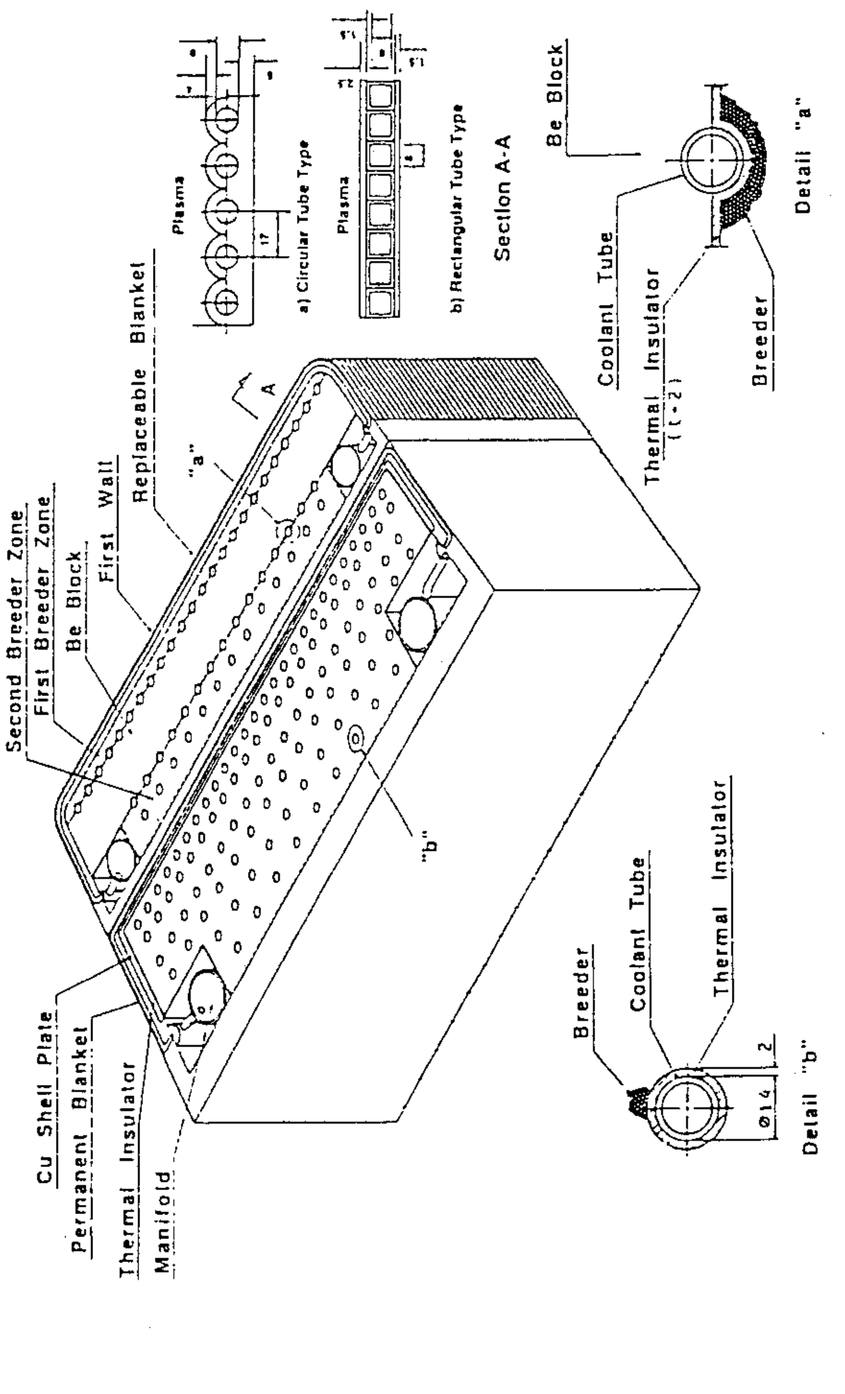

Fig. 3 Water-cooled ceramic breeder blanket for SSTR

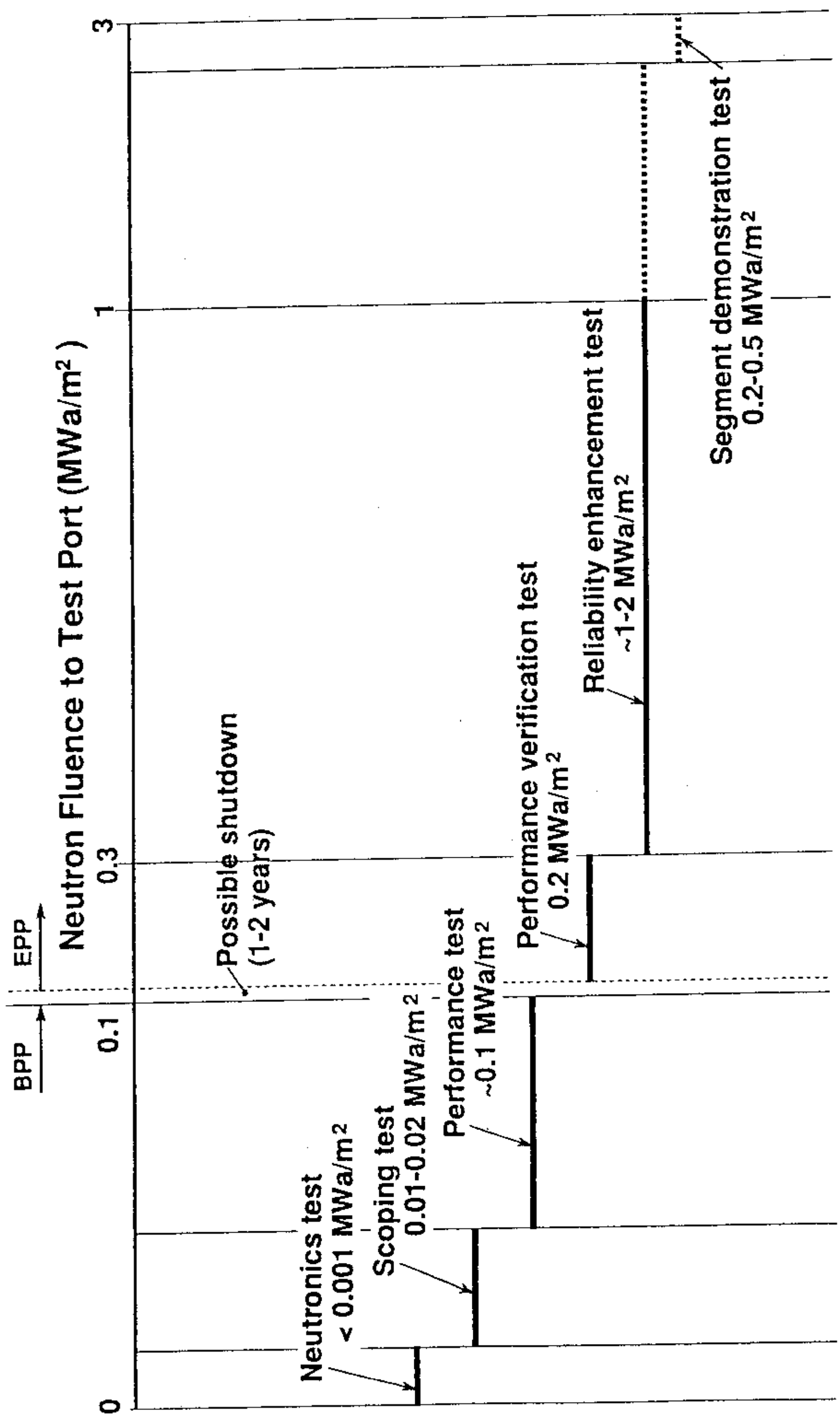

Fig. 4 Test procedure in ITER

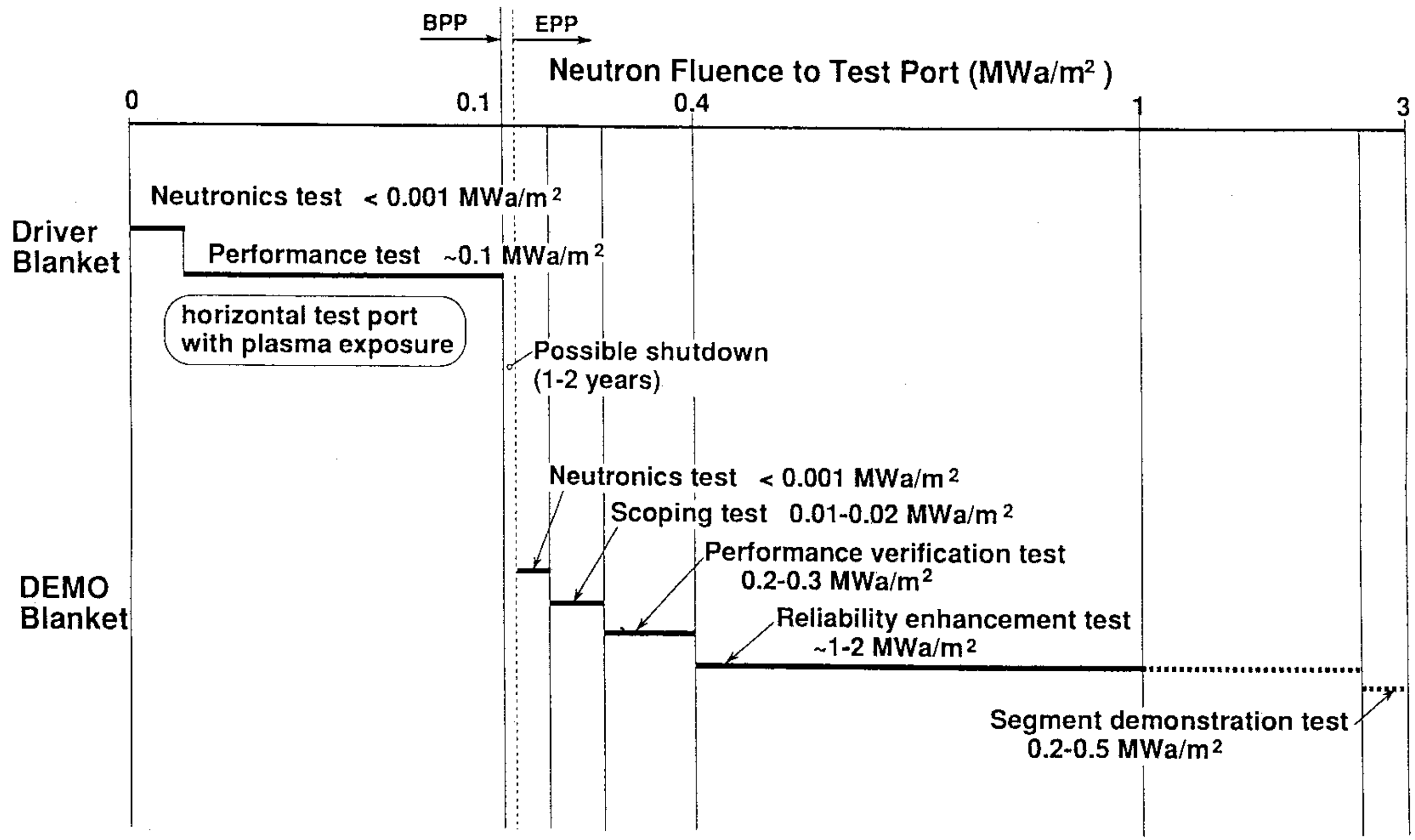

Fig. 5 Test procedure with tests for driver blanket in one port

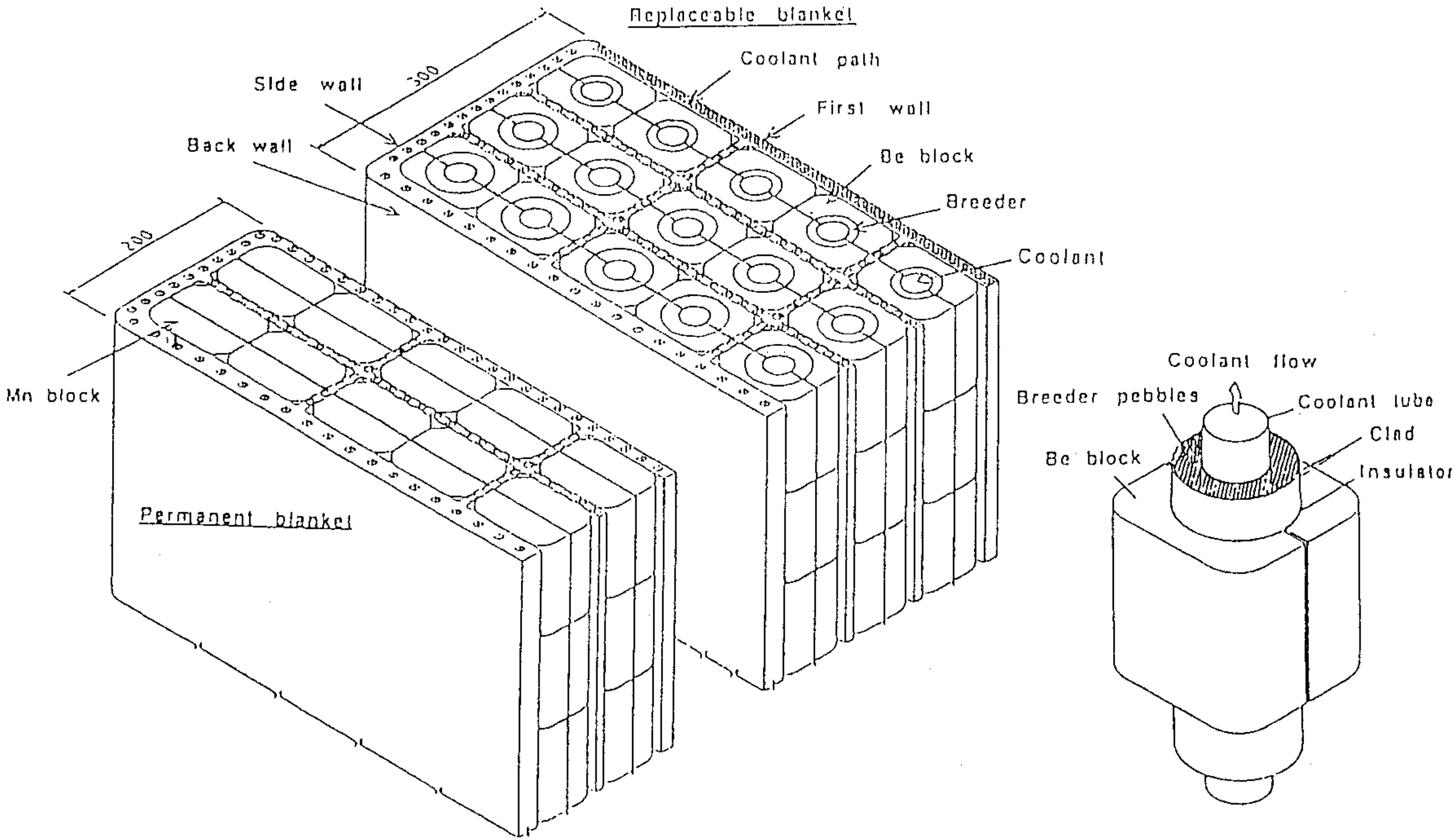

Fig. 6 Helium-cooled ceramic breeder blanket for SSTR

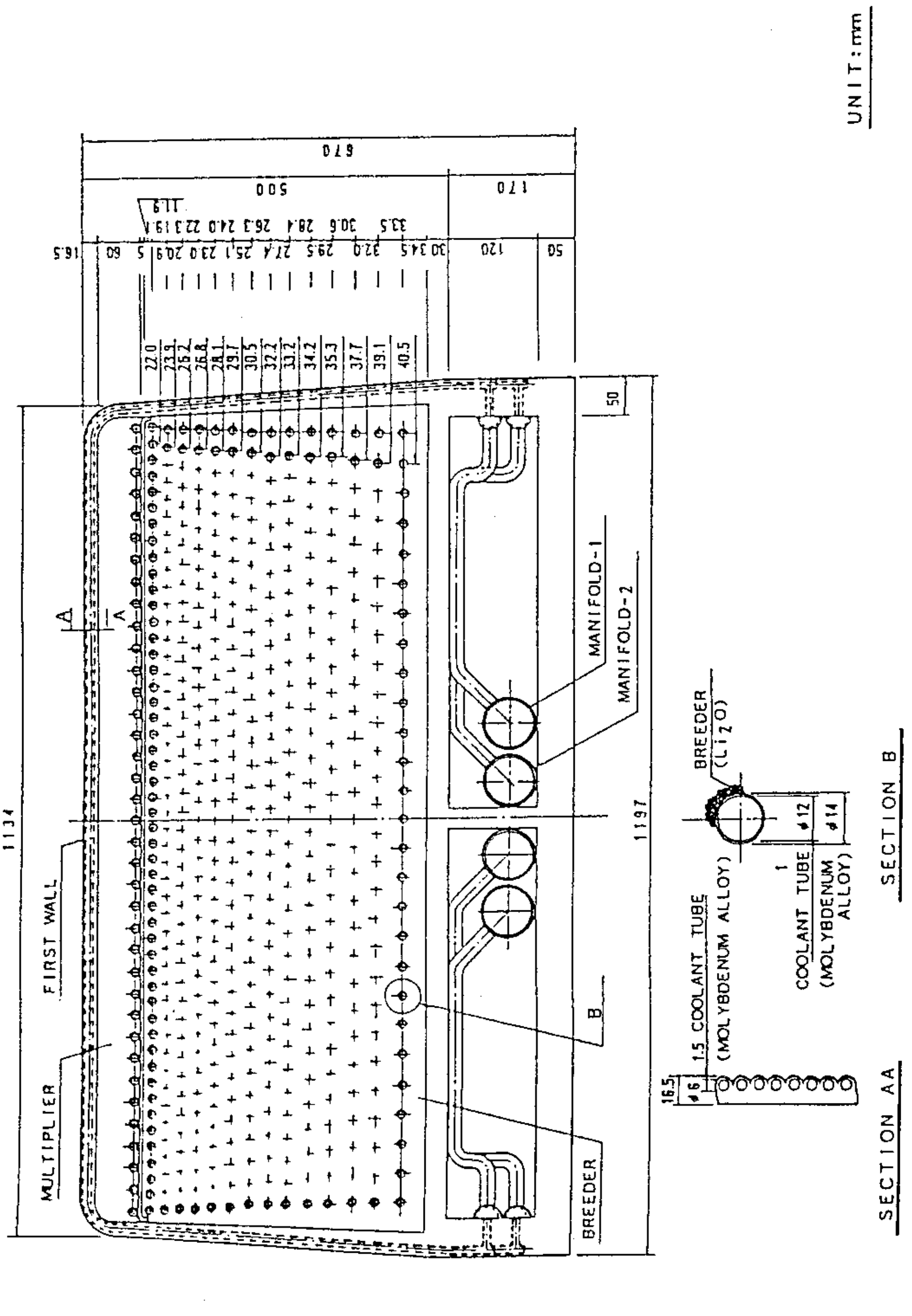

Fig. 7 Cross-sectional view of Mo-alloy/Li_2O/He-BOT blanket for DEMO reactor

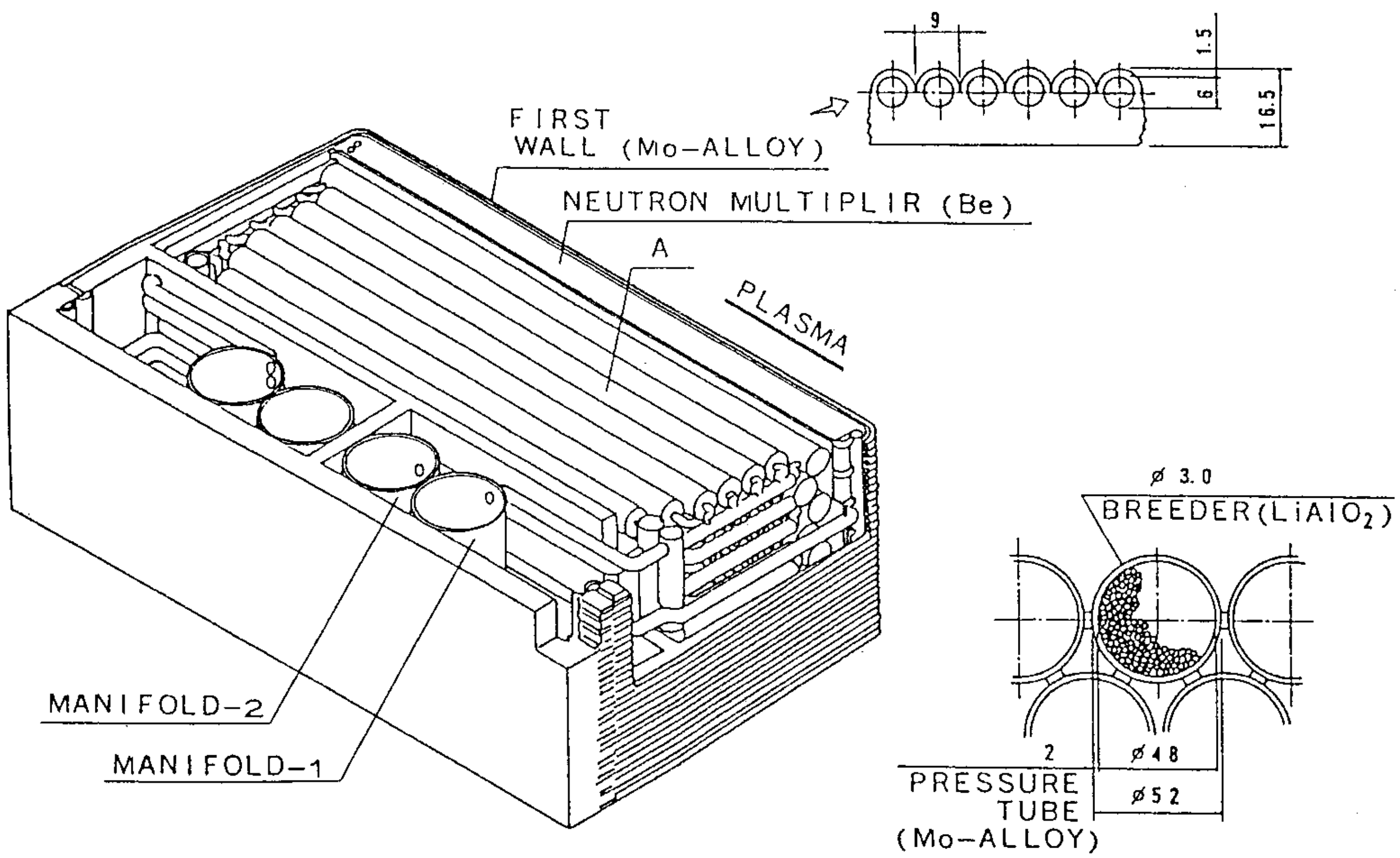

Fig. 8 Mo-alloy/$LiAlO_2$/He blanket for DEMO reactor