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European Helium Cooled Pebble Bed (HCPB) Test Blanket

ITER Design Description Document Status 1.12.1996

Compiled by:
M. Dalle Donne and L. V. Boccaccini

Contributors:
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Forschungszentrum Karlsruhe GmbH
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Abstract

The Helium Cooled Pebble Bed (HCPB) blanket is based on the use of separate small lithium orthosilicate and beryllium pebble beds placed between radial toroidal cooling plates. The cooling is provided by helium at 8 MPa. The tritium produced in the pebble beds is purged by the flow of helium at 0.1 MPa. The structural material is martensitic steel. It is foreseen, after an extended R&D work, to test in ITER a blanket module based on the HCPB design, which is one of the two European proposals for the ITER Test Blanket Programme. To facilitate the handling operation the Blanket Test Module (BTM) is bolted to a surrounding water cooled frame fixed to the ITER shield blanket back plate. For the design of the test module, three-dimensional Monte Carlo neutronic calculations and thermohydraulic and stress analyses for the operation during the Basic Performance Phase (BPP) and during the Extended Performance Phase (EPP) of ITER have been performed. The behaviour of the test module during LOCA and LOFA has been investigated. Conceptual designs of the required ancillary loops have been performed. The present report is the updated version of the Design Description Document (DDD) for the HCPB Test Module. It has been written in accordance with a scheme given by the ITER Joint Central Team (JCT) and accounts for the comments made by the JCT to the previous version of this report.

This work has been performed in the framework of the Nuclear Fusion Project of the Forschungszentrum Karlsruhe and it is supported by the European Union within the European Fusion Technology Program.

Europäisches heliumgekühltes Feststoff-Testblanket ITER Design Description Document. Status 1.12.1996

Kurzfassung

Das HCPB Feststoffblanket ist aus Schichtungen kleiner Lithium-Orthosilikat- und Berylliumkugeln aufgebaut, die durch radial-toroidal verlaufende Kühlplatten voneinander getrennt sind. (HCPB steht für die englische Bezeichnung helium cooled pebble bed.) Die Kühlung erfolgt durch gasförmiges Helium bei einem Druck von 8 MPa. Das in den Kugelschichten erzeugte Tritium wird durch einen separaten Heliumstrom von 0.1 MPa herausgespült. Das Strukturmaterial ist martensitischer Stahl. Das Blanket-Testprogramm für ITER sieht vor, unter anderen, zwei europäische Blanketvarianten in Form von Testeinsätzen (Module) zu untersuchen, darunter auch das HCPB. Um die Handhabung zu erleichtern, wird das Blanket-Testmodul (BTM) in einen wassergekühlten Halterahmen montiert, der seinerseits an der Rückwand des ITER Abschirmblankets befestigt ist. Die Auslegung des Testmoduls erfolgte an Hand dreidimensionaler Analysen für die Neutronik (mit Monte-Carlo-Methoden) sowie für das thermohydraulische und das mechanische Verhalten für beide in ITER geplanten Betriebsphasen, die sogenannte basic performance phase (BPP) und die extended performance phase (EPP). Auch wurde das Störfallverhalten des Testmoduls bei Kühlmittelverlust (LOCA) und Durchsatzstörungen (LOFA) untersucht. Hierzu wurden Konzepte für die erforderlichen Kühl- und Hilfssysteme erarbeitet. Der vorliegende Bericht ist das sogenannte ITER Design Description Document (DDD) für das HCPB Testmodul. Er entspricht in seiner Gliederung dem vom ITER Joint Central Team (JCT) vorgegebenen Schema. Die vom JCT abgegebenen Kommentare zum ersten Entwurf dieses Berichtes wurden berücksichtigt.

Die vorliegende Arbeit wurde im Rahmen des Projekts Kernfusion des Forschungszentrums Karlsruhe durchgeführt und ist ein von der Europäischen Union geförderter Beitrag im Rahmen des Fusionstechnologieprogramms.

Design Description Document
(Status: 1.12.1996)

European Helium Cooled Pebble Bed (HCPB) Test Blanket

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Summary

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

The blanket test program will investigate various design concepts of tritium breeding blankets proposed by the Parties. The Design Description Document(s) of the Test Blanket System addresses the requirements and the design description of the Test Blanket System(s) with the ITER device, auxiliaries, facilities, machine operations, safety, reliability, and maintenance.

The European fusion program proposes two DEMO relevant blanket concepts for testing in ITER. The one is the Water Cooled Lithium Load (WCLL) blanket, the other the Helium Cooled Pebble Bed (HCPB) blanket. Both use martensitic steel as structural material. The present version of the EU DDD covers the design specifications for the HCPB.

During the ITER Basic Performance Phase (BPP) the blanket test module shall occupy half of the test port allocated to the helium cooled blankets, the other half being occupied by the Japanese helium cooled ceramic breeder blanket module. During the Extended Performance Phase the HCPB test modules may occupy the whole or half of one port according to the numbers of blanket concepts to be tested and to the number of ports available for the blanket module testing. Parts of the tests, in particular those related to tritium control and fluiddynamics (helium flow distribution, pressure drop), may be conducted before the first plasma ignition.

The purpose of the tests is to validate the design principles and the operational feasibility for the demonstration blanket system. The tests in ITER include the simultaneous function of all subsystems including first wall, blanket module and shield as well as cooling and tritium systems and - to a lesser extent - blanket specific equipment for remote handling. Safety, reliability, maintenance and dismantling will be equally addressed. To assess those qualities and characteristics,

the test blanket systems are to be exposed directly to the ITER plasma for relatively long, continuous operating periods. These test blanket modules will be replacing shielding blanket modules, thus they must meet all applicable ITER shielding blanket requirements, including heat removal, shielding protection for the Vacuum Vessel welds and Toroidal Field magnets, and reduction of neutron streaming.

Breeding and recovery of tritium are important goals of the test program. Lithium ceramic compounds will be used as the breeder materials to be investigated. Subsystems to recover the bred tritium will be demonstrated along with test facilities to separate and remove the tritium from the coolant or purge steams. Special designs and tritium handling facilities will be required to meet the ITER safety goals and requirements. Generation and extraction of high temperature coolant of helium will demonstrate the suitability of fusion for commercial power generation. The high temperature coolant will transfer the rejected heat to the ITER facility water coolant system.

The Test Blanket Modules will be designed to: (1) conform to the same safety requirements as other in-vessel components, (2) be robust against the thermal and mechanical loads produced on them by disruption, and (3) have a minimum impact on reactor operation and availability due to any unscheduled test module removal. To meet the vacuum requirements, the Test Blanket Systems shall provide the primary vacuum boundary near the inner Vacuum Vessel surface.

The Test Blanket Systems are to be installed and maintained through the horizontal test ports. Standard ITER remote handling equipment and procedures will be used to the maximum extent. All Test Blanket Systems plumbing and instrumentation and control shall be contained within the vacuum chamber horizontal port extensions and pass through the horizontal port or shielding doors. Maintenance rails and other remote handling equipment are to be provided for use within the horizontal ports. Space shall be provided in the region immediately outside the biological shield, near the ports, for tritium handling equipment while the Test Blanket Systems are in place and for storage of test equipment during maintenance actions. Furthermore place shall be provided for the helium coolant loops, but not necessarily near the horizontal ports. Transport from the horizontal ports to the Hot Cells are to be provided as well as facilities in the Hot Cells for storage, maintenance, testing, refurbishing, and dismantling the test articles.

1 Functions and Design Requirements

1.1 Functions

The test blanket system has to fulfill the following tasks:

1.1.1 Tritium breeding to demonstrate the feasibility of the process and to ultimately enable the extrapolation to a full size blanket and the validation of analytical tools.

1.1.2 High grade heat production and removal to demonstrate the feasibility of electricity production.

1.1.3 Remove the surface heat flux and the nuclear heating within the allowable temperature, stress and deformation limits.

1.1.4 Reduce the nuclear responses in the vacuum vessel structural material for the ITER fluence goal.

1.1.5 Contribute to the protection of superconducting magnets from excessive nuclear heating and radiation damage.

1.1.6 Contribute to the passive stabilization of the plasma.

1.1.7 Contribute to the reduction of neutral density between the divertor and the main plasma chamber by ~ 10^4.

1.1.8 Provide a maximum degree of mechanical and structural self-support to: (1) minimize the loads transmitted to the vacuum vessel, and (2) decouple the operating temperature ranges between the test blanket system, the backplate and the vacuum vessel.

1.2 Design Requirements

1.2.1 General Requirements

1.2.1.1 The system must be designed for the power requirements set for ITER (Ref. GDRD Sect 2.2.1.2.1 and 2.2.6.1)

a. Maximum Nominal Fusion Power	1.5 GW
b. Maximum Fusion Power Excursions	± 20%
c. Duration of Excursion Time	~ 10 sec
d. Pulse Duration	1000 sec
e. Pulse Repetition time	2200 sec

1.2.1.2 The primary wall of the Test Blanket shall provide a vacuum tight, cooled barrier between the plasma and the underlying blanket/shield structure capable of removing the surface heat flux (≈ 0.5 MW/m^2) and the highest level of nuclear heating (≈ 15 MW/m^3) [per Ref. GDRD Sect 1.4-3.4].

1.2.1.3 The Test Blanket shall be designed for an average FW boundary fluence of > 0.3 MWa/m^2 if testing is during the BPP and > 1.0 MWa/m^2 if testing is during the EPP.

1.2.1.4 The Test Blanket System shall demonstrate a tritium breeding ratio sufficiently high to perform measurements and to allow reliable extrapolation of the breeding ratio to a full size blanket.

1.2.1.5 The Test Blanket System shall provide adequate neutron shielding protection to the vacuum vessel and magnets (per GDRD 5.5.2.1.2).

1.2.1.6 The Test Blanket System shall demonstrate its capacity to generate high grade heat and to remove the power from the blanket system at reactor relevant coolant conditions (outlet temperatures 450°C).

1.2.1.7 The Test Blanket System shall be designed for installation, routine maintenance, and removal by remote handling equipment through horizontal test ports in the cryostat and vacuum vessel. The time required by these operations shall be minimized.

1.2.1.8 Due to its high level of importance in the successful operation of ITER and its potentially large effect on the overall machine availability, the Test Blanket System design, R&D, procurement, manufacture, test, installation, and operation will be to high quality standards.

1.2.1.9 The Test Blanket System will be designed according to the Test Blanket Program standards and to the applicable codes, manuals, and guidelines specified. The system shall be designed in compliance with the applicable structural design criteria.

1.2.1.10 System and component reliability requirements are TBD pending outcome of FMEA, Reliability, and other System Engineering Studies.

1.2.2 Vacuum Requirements

1.2.2.1 A double barrier with intermediate leak detection will be used as the primary tritium containment boundary at vulnerable locations (i.e. flanges, bellows, etc.). For the Test Blanket System, this boundary will be established at the nominal Vacuum Vessel.

1.2.2.2 The leak rate inside the primary vacuum must be < 10^{-7} Pa m^3 s^{-1}. The Test Blanket System should have a leak rate < 10^{-8} Pa m^3 s^{-1}.

1.2.2.3 The Test Blanket System and its components will have to undergo both hot and cold vacuum leak tests before and after installation. The possibility of repair work shall be foreseen.

1.2.2.4 Materials, design, tolerances and surface finish must be consistent with the generation and maintenance of high quality vacuum and with the ITER outgassing requirements.

1.2.2.5 The design of the Test Blanket System shall enable bake-out of the structures at 240°C before (to avoid plasma pollution) and after an operation period (to avoid risk of hydrogen embrittlement in the test blanket structure).

1.2.3 Structural Requirements

1.2.3.1 The Test Blanket System shall be designed according to the project Structural Design Criteria, (SDC). Using the rules specified in the SDC, stresses and other applicable quantities shall be calculated for different load calculations in nominal and accidental situations. Details are TBD.

1.2.3.2 The Test Blanket System shall be supported by the blanket/shield backplate structure. It shall be designed to withstand the following conditions:

1.2.3.2.1 The external pressure inside the vessel will be 10^{-6} Pa during normal operation, 0.5 MPa for off-normal conditions, and 0.1 MPa for maintenance.

1.2.3.2.2 The pressure of the cooling helium will be 8 MPa in normal operation. The pressures during off-normal conditions and system tests are TBD.

1.2.3.2.3 Electromagnetic loads as defined in 1.2.4.

1.2.3.2.4 Heat loads at maximum power conditions defined in 1.2.5 and the resulting thermal stresses.

1.2.3.3 The shield structure must accommodate the loads resulting from the cooling pressure, the external pressure within the vacuum vessel, and the full range of electromagnetic loads.

1.2.3.4 The Test Blanket System structure must react the range of axisymmetric radial and poloidal loads on the components that it supports. The weight, net vertical, and net toroidal loads will be transmitted to the Shielding Blanket Backplate.

1.2.4 Electromagnetic Requirements

1.2.4.1 The system must be designed to withstand the electromagnetic loads resulting from the interaction of the magnetic fields and eddy currents induced in the system during plasma transient conditions. The combination of these currents and fields existing in the device may result in radial, toroidal, and/or poloidal pressures on different faces of the modules. The direction and magnitude of these loads must be determined based on design dependent factors such as: location, electrical characteristics, size, segmentation, and connection to other components. The loads at all positions must be calculated for:

a. normal operation, including start-up and shut-down

b. the system must be designed to withstand a reduced set of electromagnetic induced loads resulting from centered plasma disruptions and vertical displacement events (VDE's) with the parameters described in GDRD Section 2.2.7 and for the number of disruptions specified in section 2.2.7.4. Specific values are TBD.

1.2.5 Thermal-hydraulic Requirements

1.2.5.1 System Requirements at nominal power of 1.5 GW. The power deposition has been calculated with a 3-D Monte Carlo neutronic model for a full size test module (to account for the presence of the Japanese half module) and a first wall of martensitic steel with a 5 mm thick beryllium protective layer. The two test modules have been assumed to be 50 mm recessed in respect of the contour of the ITER first wall. A surface heat flux of 0.25 MW/m^2 shall be used for the design of the helium coolant system, while a peak value of 0.5 MW/m^2 shall be used for the first wall design.

1.2.5.2 System Requirements in off-normal conditions are TBD

1.2.5.3 The power of the test blanket shall be iteratively recalculated as the design and material data evolve.

1.2.5.4 First Wall and Breeder Zone shall be cooled in series by the high pressure (8 MPa) helium coolant. An overpressure of 20 % and steady state conditions shall be assumed for the coolant loop design. The test blanket power shall be dissipated to a low pressure, low temperature cooling loop provided by ITER. Peak requirements in the heat exchangers are TBD. The loops shall be equipped with a control system enabling to decrease the helium mass flow during the plasma dwell time. The coolant system should be able to raise the temperature of the test blanket modules to the prescribed value in approximately 2 to 4 h.

1.2.5.5 The test blanket systems will be designed to operate at elevated temperatures relative to the shield blanket systems. This will allow the test blanket to demonstrate the tritium breeding capability and generation of high grade heat. The maximum temperatures for the coolant helium will be in the range of 450 °C, structural material in the range 500 - 550 °C and solid breeder material up to 900 °C.

1.2.6 Mechanical Requirements

1.2.6.1 The first wall and blanket portion of the Test Blanket System will be attached to the Shielding Blanket Backplate. This will help maintain critical first wall and module dimensional tolerances and to react imposed electro-mechanical loads transferred from the modules to the Vacuum Vessel. This also allows the Test Blanket and Vacuum Vessel to have two different working temperatures. The whole blanket static and dynamic loads are transmitted to the Vacuum Vessel.

1.2.6.2 The coolant and breeder pipework as well as eventual gas ducts, electrical wires or diagnostic cables will be routed through the horizontal test port Vacuum Vessel ports and will be designed to allow movements during thermal transients.

1.2.6.3 The penetrations of all pipework through the Vacuum Vessel and the Cryostat shall fulfill all requirements of a vacuum and safety boundary.

1.2.6.4 Welds in contact with water and in high fluence and/or stress level regions, such as near the first wall, are subject to irradiation assisted stress corrosion cracking and should be avoided.

1.2.6.5 The Test Blanket First Wall shall be bakeable to 240°C.

1.2.6.6 The Test Blanket Articles shall be designed to be removable (RH Class 2) by remote handling through the horizontal test ports. The required time for this operation shall be minimized.

1.2.6.7 The Test Blanket structural connections shall use remote handling compatible connectors, accessible from the back side.

1.2.7 Electrical Requirements

1.2.7.1 The in-vessel portion of the Test Blanket System shall contribute to meeting the requirements that the combined toroidal resistance of the blanket in-vessel structures and the Vacuum Vessel must be larger than 4 $\mu\Omega$ as specified in GDRD Section 5.3.3.3.1.

1.2.7.2 A continuous electrical connection (poloidal and toroidal) between all FW of adjacent modules is desirable to decrease the above electromagnetic loads at the expense of large localized effects on these connections.

1.2.7.3 The connection from the tokamak assembly to the outside, through the supply pipes of the blanket system, shall have a resistance of TBD.

1.2.8 Nuclear Requirements

1.2.8.1 The Test Blanket System shall provide enough shielding so that the Vacuum Vessel remains reweldable at specific locations until atleast an average fluence of 1 MWa/m^2 is reached on the FW (Ref. GDRD 5.5.2.3.3.1).

1.2.8.2 The Test Blanket System shall be designed so that the nuclear responses for at least 1 MWa/m^2 at the First Wall are limited to a helium production of < 1 appm at all components that may need to be rewelded, such as Vacuum Vessel, blanket components, or piping.

1.2.8.3 The blanket system (including the Test Blanket System), in combination with the vacuum vessel and divertor, shall be designed so that the power dissipated by the attenuated radiation in the cryogenic toroidal magnet remains within the limits specified in GDRD Section 5.3.3.6. The peak insulator dose shall be limited to 3×10^8 rad with neutron fluence of 1 MW a/m^2 at the First Wall.

1.2.8.4 Provisions shall be provided to breed tritium in the test blankets during the Basic and Enhanced Performance Phases with a tritium breeding ratio from which the self-sufficiency in power reactors can be foreseen. Bred tritium will be extracted in-situ from the test blankets.

1.2.9 Remote Handling Requirements

1.2.9.1 All systems inside the biological shield boundary shall be remotely maintainable. The Test Blanket System and its supporting subsystems shall be designed in complete compliance with the remote handling requirements applicable to their respective remote handling classification. All Test Blanket System components are to be considered as Class 2, except the frames interposing between the modules and the backplate, wich are RH Class 3.

1.2.9.2 The Test Blanket System shall be designed for full remote replacement.

1.2.9.3 The in-vessel Test Blanket System components may be removed and installed without disturbing any ITER Blanket Modules.

1.2.9.4 The Test Blanket System and its supporting in-vessel subsystems must be capable of insertion/removal through the horizontal test ports by use of horizontal test remote handling equipment.

1.2.9.5 Welded joints within the plasma chamber and the Vacuum Vessel extensions shall be done, repaired, and leak tested remotely.

1.2.9.6 For any maintenance actions, the more important corrective action should meet the following design goals, see GDRD Section 5.5.1.3.3.3 and 5.19.3.9.3.1.

Test Blanket

a. be able to replace a module in 8 weeks
b. be able to repair a leak at a fluid joint within 6 weeks (not including the time required to locate and isolate the leak)

After a failure of a Test Blanket Module with impact on normal ITER operation

a. be able to correct or remove faulty module or test article within 2 weeks.

b. be able to install repaired module within 4 weeks during scheduled maintenance period

1.2.9.7 At prescribed intervals (TBD) and after significant off-normal, including electromagnetic, events it shall be possible, using existing in-vessel inspection equipment, to:

a. inspect/verify modules position
b. inspect/verify First Wall integrity
c. conduct all specified pre-operational tests

1.2.9.8 Special assembly and maintenance tools shall be provided:
a. for structural attachment of the test blanket article to the back plate:

I for welded connections

wall thickness	TBD cm
speed:	
welding	TBD cm/s
cutting	TBD cm/s
inspection	TBD cm/s

II. for mechanical connections:

end effectors	type and capacity are TBD
tools	type and capacity are TBD

III. for pipe welding, cutting, and inspection of manifolds to blanket module/FW connections:

pipe size	TBD cm OD
wall thickness	TBD cm
position	from inside pipe
speed:	
welding	TBD cm/s
cutting	TBD cm/s
inspection	TBD cm/s

be capable of joining, cutting, and leak testing the breeder and cooling manifolds of the test blanket article.

IV. Others TBD

1.2.9.9 Other in-vessel requirements include:

a. Gripping points must be provided on all replaceable components or assemblies capable of supporting their full weight over the full range of motion required for installation and removal.

b. The structural supports, coolant line joints, instrumentation, and all other interfaces necessary for (dis)assembly must be compatible with the capability of the remotely operated tools.

c. Sufficient space for the insertion and removal of tools must be assured.

d. All liquid and gas pressure bearing joints must be capable of being leak detected by remote means.

e. Mechanical guides should be provided to aid the transporter for final positioning and alignment and to protect adjacent components from damage due to collisions.

f. The maximum Test Article weight to be attached to the Shield Blanket Backplate is < 30 000 kg (TBD).

1.2.9.10 Transporter Requirements

a. The size of the Test Blanket Article, the transportable support equipment shall remain within the internal transporter dimensions accounting for covers as well as α and γ protection walls on the transporter and space occupation of transporter service equipment (requirements are TBD). The maximum transported mass is < 50 000 kg (TBD). Afterheat removal of the test article may be required during the transport time from the test port to the storage room and the hot cell. Remote surveillance may be required (TBD).

1.2.10 Chemical Requirements

The Test Blanket System and its supporting subsystems, in particular breeder and cooling systems, have to be compatible with the materials with which they are in contact. The coolant chemistry shall be defined to limit corrosion, electrochemical, and neutronic effects to acceptable levels over the system lifetime. Neutron absorbers, tritium generating chemicals (e.g. LiOH) and matter with bad activation characteristics as well as toxic and reactive chemicals shall be avoided in the coolant. The presence of hdyrogen isotopes, in particular tritium, in breeder and cooling circuits shall be continuously monitored. The need of double confinement of tritium carrying plumbing is TBD.

1.2.11 Seismic Requirements

The earthquake resistance of the Test Blanket System and subsystems shall be consistent with the specifications adopted for the ITER building. The Test Blanket System shall in particular contribute to the efficient confinement of radioactive material and chemicals during an earthquake so that the allowable release will not be exceeded.

1.2.12 Manufacturing Requirements

The Test Blanket article shall be manufactured according to the ASME code class 1 (TBD) [reference] with particular emphasis on tolerances between the Test Blanket article and the shielding blanket in the following situations:

a. shut-down including installation, shut-down after operation, and shut-down before removal;

b. nominal operation taking into account the pulsed conditions and irradiation effects (e.g. swelling);

c. accidental situations which could lead to deformations of the test article or its surroundings;

The manufacture of the test blanket system shall be accompanied by an approved quality assurance plan and pass an acceptance test prior to shipment. (Other testing requirements see 1.2.15). These acceptance tests are TBD but shall include among others:

- pressure and flow testing of all fluid channels
- vacuum/He leak testing
- NDT certification of structural and seal welds
- certification of bonding of dissimilar melts
- certification of critical dimensions

1.2.13 Construction Requirements

Construction requirements are TBD; however, it is anticipated that specific requirements will be applied during transport, handling, storing and dismantling of the various components of the Test Blanket System.

1.2.14 Assembly Requirements

1.2.14.1 The primary wall of the Shielding Blanket shall be installed within ± 10 mm of the corresponding magnetic surface, as defined in GDRD Section 2.2.4.5 (including ripples) at operating temperature. To help protect the first wall of the Test Blanket, the Test Blanket First Wall will be recessed below the adjacent Shielding Blanket First Wall and, thus, will not have an explicit requirement for alignment to the magnetic surface.

1.2.14.2 The Test Blanket System will also have the requirement to minimize any gap (less than 20 mm) to adjacent modules in order to minimize neutron streaming.

1.2.14.3 The Test Blanket System shall be installed from the horizontal test port using remote handling equipment. The structural support element for the blanket portion of the Test Blanket System shall be attached to the Shielding Blanket Backplate by bolting or welding. Provisions are to be provided to react design basis shear loads.

1.2.14.4 The shielding and vacuum vessel portion of the Test Blanket System may attach to the nominal Vacuum Vessel.

1.2.14.5 All assembly techniques must be compatible with maintaining the vacuum requirements on the system. Handling, cleaning, limits on the use of potential contaminants, etc. must be in compliance with the vacuum specifications.

1.2.15 Testing Requirements

1.2.15.1 The Test Blanket System must pass both a hot and cold leak test after completion of its assembly within the vacuum vessel and prior to start of operation. This will supplement the Test Blanket System full operational test in the Hot Cell prior to installing on the ITER device.

Leak tests

a	internal pressure	TBD MPa with helium
b	external pressure	1 Pa
c	component temperature cold/hot	20°C / 200°C (TBD)
d	leak rate acceptance level into plasma chamber	1×10^{-8} Pa m^3 s^{-1}

1.2.15.2 The system must be pressure tested with operational coolant at 1.38 (TBD) times nominal operating value after welding of the shield and first wall coolant connections to their respective manifolds. Each flow circuit must be flow tested to demonstrate the required flow rate at the design pressure differential.

1.2.16 Instrumentation & Control Requirements

1.2.16.1 Instrumentation required for operation:

(1) Monitor the system temperatures, flow rates, pressure, and stressed /deflections (to insure that they are within prescribed values).

(2) Maintain temperature differentials between different points in the system to prescribed values (TBD) as determined by thermal stress limits

a. Cooling temperature sensors; number and position	TBD
b. Flow sensors; number and position	TBD
c. Others	TBD

1.2.16.2 Instrumentation to signal acceptability to operate or need to shut down:

a. Stresses/ deflection detectors; number and location	TBD
b. Temperature sensors; number, location	TBD
c. Flow sensors; number and position	TBD
d. Leak sensors; number and position	TBD
e. Others	TBD

1.2.16.3 Other: TBD

1.2.17 Decommissioning Requirements

The Test Blanket System shall be designed to minimize the disposal rating. Since the rating criteria are site specific, the specific criteria are TBD.

1.2.18 Electrical Connections/Earthing/Insulation Requirements

The grounding requirements are TBD.

1.2.19 Material Requirements

1.2.19.1 The materials of the in-vessel components will be chosen according to the test blanket requirements, the compatibility between materials, their outgassing requirements and to the physics requirements with the objective of limiting the impurity level inside the machine.

1.2.19.2 The materials of the in-vessel components have to be consistent with the generation and maintenance of a high quality vacuum.

1.2.19.3 Materials shall be used with well characterized mechanical, structural and irradiation properties for their respective service conditions (temperature, stress, irradiation, hydrogen etc.) in order to obtain a high degree of confidence in their performance capability. The materials used in the test blanket are anticipated to be:

• structural material	martensitic steel (grade is TBD)
• first wall structural material	martensitic steel (grade is TBD)
• first wall protection	beryllium
• breeder material	overstoichiometric lithiumorthosilicate, i.e. Li_4SiO_4 plus a small amount of SiO_2 and TeO_2 (alternatives: Li_2ZrO_3 or Li_2TiO_3)
• multiplier material	Beryllium
• shielding	stainless steel (water cooled)
• coolant	helium
• piping	martensitic steel (grade is TBD)

1.2.20 HVAC Requirements

(heating,ventilation, air conditioning) Not directly applicable.

1.2.21 Lay-out Requirements

1.2.21.1 Structural and leak tightness welds shall be removed as far away as possible from high neutron flux locations.

1.2.21.2 Welds shall be isolated from gaps whenever possible. Field welds shall be protected by sufficient shielding to allow rewelding.

1.2.21.3 (deleted)

1.2.21.4 The design shall minimize gaps between modules with a goal of less than 20 mm.

1.2.21.5 Special attention shall be given to gaps between modules. Radiation streaming shall be minimized by the design.

1.2.21.6 The Test Blanket system shall be sized for insertion and removal through the horizontal mid-plane test port and the transporter shall be sized to accommodate the Test Blanket System and/or ancillary equipment (TBD).

1.2.21.7 Wherever structural welding is required, the module arrangement shall include a (TBD) mm space adjacent to welds for remote welding/cutting equipment. This layout must include an unobstructed route, of this cross-sectional size, between the weld and the point of entry for the welding equipment.

1.2.22 Other Services

The ancillary systems for the Test Blanket article depend on a reliable supply of the following infrastructure services that ITER shall provide (details and redundancy are TBD): electrical power, data connections, secondary cooling water, He detritiation, off gas

1.3 Safety Requirements

The safety requirements for the Test Blanket System are derived from the General Safety and Environmental Design Criteria (GSEDC), the General Design Requirements Document (GDRD) and functional safety requirements (confinement, fusion power shutdown, decay heat removal, monitoring, and control of physical and chemical energies) which are generally necessary for ITER. All criteria and requirements build upon the fundamental safety principles stated below:

- Design, construction, operation, and decommissioning shall meet technology-independent radiological dose and radioactivity release limits for the public and site personnel based on recommendations by international bodies such as IAEA and ICRP.
- During normal operation, including maintenance and decommissioning, radiation exposure of site personnel and the public shall remain below the prescribed limits and be kept as low as reasonably achievable (ALARA).
- ITER shall make maximum use of favorable safety characteristics which are inherent to fusion. Uncertainties of plasma physics shall not have an effect on public safety.
- The defense in depth concept shall be applied to all safety activities so that multiple levels of protection are provided to prevent or minimize the consequences of accidents.
- Special attention should be given to passive safety.
- The design shall minimize the amounts of radioactive and toxic materials and the hazards associated with their handling.
- All conventional (non-nuclear) safety and environmental impacts from construction, operation, and decommissioning shall meet common industrial standards for industrial practice. This includes chemical toxins and electromagnetic hazards.

1.3.1 Safety Functions

The Test Blanket System may contain "experimental" components to which no safety function will be assigned. The Test Blanket System may, however, support the safety function "fusion power shutdown" in off-normal situations by passive or active action; however, the definition of and requirements on this type of system are not yet defined.

1.3.2 Safety Classification of Items

The Test Blanket System equipment shall be classified according to its importance to safety into four classes according to Table 4.1.2.-3 "Safety Importance Classification" in [GDRD - Safety v.5(4/21/95)] and the associated rules. The following provisional Safety Importance Classes (SIC) are suggested by the Safety Environmental and Health Division (SEHD):

Component	SiC	Comment
In-vessel part of the Test Blanket system	3 or 4 TBD	No design and related safety analyses are presently available
Ex-vessel part of Test Blanket System and blanket coolant loops	2	SIC-2 for confinement SIC-4 for decay heat removal

1.3.3 Safety Design Limits and Analysis Requirements

The safety limits shall be determined by iterating deterministic and probabilistic safety analyses with the design of the Test Blanket System. The safety analyses shall use the process adopted by the project which aims at systematic identification, modeling, and analysis of the representative event sequences. Depending on the required degree of detail, this process will be graded from qualitative analysis up to detailed simulations and calculations. Accident initiating events will be identified through Failure Modes and Effects Analysis (FMEA) and then grouped into Postulated Initiating Event (PIE) categories. The PIEs will be supplemented by the related accident source terms (tritium, activation products), determined in a conservative manner. Particularly, detailed fault analysis shall be performed where there is a potential for challenging confinement barriers.

Provisional safety design limits are as follows:

- If beryllium (Be) is used as FW armor material, short term temperatures shall stay below 800°C (TBD) to avoid Be-steam ignition scenarios.
- If carbon (CFC) is used as FW armor material, short term temperatures shall stay below 1800°C (TBD); the use of radiatively cooled (i.e. very hot) carbon tiles shall be limited as far as possible.
- If Be is used as first wall armor material, long term (decay heat driven) Be temperatures shall be limited to 500°C (TBD) to avoid excessive H_2 production.
- If CFC is used as first wall armor material, long term (decay heat driven) temperatures shall be limited to 800°C (TBD) to avoid excessive H_2 production.
- Maximum steel temperatures are TBD and depend on the final material choice. Environmental effects (e.g. DBTT or hydrogen embrittlement) shall be accounted for.
- The inventory of Be dust inside the vacuum vessel shall be limited to 100 kg (TBD). This value is provided provisionally fore ease of EDA design.
- The total mobilizable tritium inventory inside the PFCs (first wall, divertor, limiters, launchers) shall be limited to 1 kg.
- The corrosion products in the blanket cooling loops shall be limited to a total of 10 kg (TBD).

1.3.4 Safety Assessment

The safety analyses will include but are not limited to the following events:

- Plasma disturbances (such as disruptions, VDEs, power excursions) resulting in an overload of the Blanket.
- Over-pressure in the VV from water LOCAs causing steam formation and H_2 generation on hot FW armor surfaces.
- Temperature transients of the Blanket due to LOFAs in the primary heat transfer system and fron in- and ex-vessel LOCAs.
- Pressure and temperature transients with related chemical reactions inside the Breeding Blanket due to water ingress by LOCAs.
- Pressure and temperature transients and related chemical reactions at the FW surface due to air ingress into the VV.
- Mechanical loads to the Blanket from magnetic accidents.

1.3.5 Specific Safety Design Requirements

1.3.5.1 The design basis for the Test Blanket System shall take into account the initiating events and potential loads due to accidents as identified by the safety analysis.

1.3.5.2 The design of the blanket module support structure shall react a large portion of the load acting on the modules thus minimizing the load on the Vacuum Vessel, the first radioactivity confinement barrier.

1.3.5.3 The Test Blanket System shall not significantly contribute to the ITER radioactivity source term and the blanket parameters shall be chosen accordingly.

1.3.5.4 The design should minimize the volume of liquid spills from the Test Blanket article into the Vacuum Vessel.

1.3.5.5 The design should assure fast thermal relaxation of an overheated FW to avoid self-sustained chemical reactions between plasma facing materials and coolants/air. This requires the provision of reliable means (such as good thermal contact between FW and bulk blanket) to cool down the hot FW surface in the course of an accident (such as ex-vessel LOCA, LOFA or plasma disturbance). Otherwise the accident may cause an in-vessel LOCA with the related concerns, i.e. mobilization/release of tritium and activation products, and chemical reactions (H_2 production). This requirement is quantified in terms of temperature limits set out in Section 1.3.3.

1.3.5.6 The design should limit the long term (several hours after shutdown) decay heat driven FW temperatures to avoid H_2 concentrations in the Vacuum Vessel which are prone to deflagration/detonation if air ingress in the Vacuum Vessel cannot be excluded. This requirement is quantified in terms of temperature limits set out in Section 1.3.3.

1.3.5.7 It is suggested to segment the Test Blanket System cooling loops so that sufficient independence is provided. This measure would serve the implementation of the single failure criterion.

1.3.5.8 Attention should be paid to potentially asymmetric temperature distributions due to these measures which should not cause thermal stress in the first wall/blanket equipment above permissible limits.

1.3.5.9 Off-normal heat removal should be as passive as possible. It is suggested to design the heat transport system to allow for removal of decay heat by natural coolant circulation. It is suggested further to increase by adequate treatment, if the vacuum requirements allow, the relative emissivity of thermal radiation between the adjacent surfaces of Test Blanket System and Vacuum Vessel to values significantly above the natural ones (such as 0.8 vs. 0.3).

1.3.5.10 (deleted)

1.3.5.11 In general, the design should strive for:

- Limitation of the inventory of radioactive dust inside the Vacuum Vessel.
- Limitation of the mobilizable tritium inventory inside the Test Blanket System.
- Limitation of the corrosion products in the Test Blanket System cooling loops.
- Limitation of the tritium concentration in the Test Blanket System coolant systems.

1.3.5.12 Monitoring shall be provided to indicate whether the above requirements are being met.

1.3.5.13 The design of decontamination, shielding, remote operation, flask transfer functions should minimize the dose to personnel in the course of maintenance and decommissioning.

1.3.5.14 Amounts and radio-toxicity of radioactive waste from operation and decommissioning of the Test Blanket System equipment should be minimized within the limits set by the applicable material. Potentially high radio-toxicity of breeder, multiplier, and braze materials should be considered in this context.

1.3.5.15 The experimental nature of the FW leads to the design requirement for the Vacuum Vessel that failures of the FW should not cause rupture of the vessel which is the first radioactivity confinement barrier.

1.4 R&D Requirements

The R&D requirements for the Test Blanket development are concept dependent except for the development and the complete characterization of a suitable structural steel with martensitic steel being the currently preferred material. The associated R&D is regularly revised to adjust priorities and effort, and to account for the latest technical progress in the different fields of R&D. This program is closely linked to the development of a blanket for a DEMOnstration reactor for which the Test Blanket article should be a representative module.

1.5 Operation and Maintenance

The operational and maintenance requirements for the Test Blanket System are included in Section 1.2.1 and 1.2.9.

1.6 Surveillance and In-Service Inspection

The surveillance and in-service inspection requirements are included in Section 1.2.9.

1.7 Quality Assurance

The quality assurance requirements are included in Section 1.2.1.

1.8 System Configuration & Essential Features

The configuration and essential features are included in Section 1.2.21.

1.9 Interfacing Systems

In order to successfully complete all test objectives, the Test Blanket System must work in cooperation with many of the other ITER systems and facilities. These interrelationships are many and complex, involving both geometric and functional requirements. Below is a list of the systems that have a significant impact on the operational capability of the Test Blanket System. A brief description of the geometric and functional requirements is given for each interfacing system. In the future, a set of interface control documents will be prepared to identify the complete listing of interfaces and define the detailed requirements of each interface.

Vacuum Vessel

The Vacuum Vessel System is to provide twenty horizontal ports for systems to access the plasma chamber. Specifically, this involves ports or access chambers of a particular size and structural capability to properly accommodate the port systems, including ancillary equipment, and the associated remote handling equipment.

The unique requirements imposed by the Test Blanket System will involve the mounting configuration onto the Vacuum Vessel Wall, the structural requirements during operation and maintenance periods, the thermal conditions of the shield and ancillary equipment, and accommodations for routing of plumbing lines.

- number of test ports required
- Horizontal port size/geometry
- Load support requirements
- Thermal requirements
- Coolant plumbing requirements
  - size/location
  - mechanical loads and displacements
  - special seal requirements
  - penetration requirements

Shielding Blanket

The Test Blanket System will work in close cooperation with this system. One of the primary requirements for the Shielding Blanket is to support the static and dynamic loads of the Test Blanket First Wall, and Blanket portion of the Test System. This support will be provided by the Shielding Blanket Backplate. to support the imposed loading conditions, the Backplate may have to be strengthened to provide the additional support. The Backplate will also have to provide provisions to handle the shear loads.

There must be a high level of geometric synergism between these two systems to meet the ITER requirements for the neutronic streaming and not have contact load transfer between system modules.

In order to provide limited protection from direct plasma ion impingement on the Test Blanket First Wall, the Test Blanket, the Test Blanket will be recessed below the general surface level of the surrounding Shielding Blanket First Wall. This will impose additional surface heating requirements on the adjacent Shielding Blanket First Wall components. The temperatures and surface conditions (emissivity, absorptivity, and surface area) of the interfacing surfaces will have to be determined to estimate the anticipated heat transfer.

- geometry
- mechanical loads
- physical loads
- thermal loads

Remote Handling Equipment

Remote handling equipment will be required to install, inspect, and maintain diagnostic, plasma heating, maintenance, test blanket modules, and shield port systems through the horizontal access port. The specific interface requirements for the Test Blanket System will involve unique geometry, weight, positioning, and thermal constraints. The geometry will involve not only the Test Blanket, which may be separated into two elements, but will also include the ancillary equipment that will be positioned behind the blanket in the Vacuum Vessel Extension area. Special-use end effectors will be the responsibility of the Test Blanket System. Some of the interface requirements are listed below:

- maximum supported weight
- positioning accuracy
- kinematics requirements
- inspection requirements
- accomodation of special end effectors
- accomodation of special materials and coolants

Cryostat

The Cryostat System is to provide twenty horizontal ports for access to the Vacuum Chamber. Additionally, the Cryostat is to provide the Second Tokamak Confinement Boundary.

The unique requirements imposed by the Test Blanket System will involve the unique geometry constraints and special maintenance requirements. Plumbing lines shall be accomodated in the port areas.

- number of test ports required
- horizontal port size/geometry
- thermal requirements
- coolant plumbing requirements
  - size/location
  - mechanical loads and displacements

- special seal requirements
- penetration requirements

Primary Heat Transport System

This system is to provide water coolant to remove the heat generated in the test blanket and shield. Detailed information needed;

- Number of loops
- Inlet and outlet temperature for each loop
- Flow rate for each loop

Vacuum Pumping System

The blanket system is partially contained within the primary boundary and effects the volume pumped by the Vacuum Pumping System. As a result, emissions from surfaces and leaks from the blanket system must be within the capability of the pumping system. In addition, the vacuum pumping may include specific components, such as tracer gas sources, for remote leak checking. These components must be permanently mounted on the blanket components near high potential leak sources.

- outgassing requirement
- leakage rate

Tritium Plant

The use of unique materials will effect the Tritium Plant System involving the possible airborne elements.

Tokamak Operations and Control

The Test Blanket System instrumentation needs shall be integrated into the Tokamak Operations and Control System.

Building

The building space external to the cryostat and biological shield shall accommodate the Test Blanket System maintenance scheme. Space and support services (power, cooling water, He, ventilation etc.) shall be provided for operational support equipment near the horizontal test ports. Radial space must be provided to remove the modules form the mid-plane maintenance ports and transport them to the hot cells.

- location and size of needed space
- support services (electrical, I&C, fluids)

Waste Treatment and Storage

The Test Blanket System will impose some additional requirements on the Waste Treatment and Storage system. This will evolve from the use of unique materials (see section 1.2.19) and coolants.

General Testing Equipment

The Test Blanket System will impose some additional requirements on the General Testing Equipment system. This will evolve from the use of unique materials (see section 1.2.19) and coolants.

Hot Cells

The Test Blanket System should be designed in such a way that the following operations can be performed in the hot cells:

1. to separate the components of the Test Blanket Subsystem:
    - remove the Test Blanket Subsystem from ist location in the Transporter;
    - cut the tubes at designed planes;
    - unfasten the bolts between the Shield and the Support Frame;
    - unfasten the bolts of the mechnical connection between the BTM and the Shield.
2. to perform the following operations on the component at the end of the irradiation time foreseen for the HCPB BTM:
    - cut the BTM and remove the beryllium and orthosilicate pebble from the beds for investigation:
        - tritium release test
        - mechanical investigation
        - crush test
        - thermal cycling test
    - cut probes of the structure for investigations.
        - tritium release test
        - swelling test
        - embrittelment test
        - tritium inventory determination
3. to perform the following repairs:
    - weld small leakages in the components;
    - replace tubes;
    - replace damaged instrumentation.

1.10 Codes and Standards

Codes and standards requirements are included in Section 1.2.1.

1.11 Reliability Requirements

Reliability requirements are included in Section 1.2.1.

1.12 Other Special Requirements

TBD

2 Design Description

2.0 Summary Description

2.0.1 General

The European fusion program proposes two DEMO relevant blanket concepts for testing in ITER. The one is the Water Cooled Lithium Load (WCLL) blanket, the other the Helium Cooled Pebble Bed (HCPB) blanket. Both use martensitic steel as structural material. The present version of the EU DDD gives a description of the HCPB Test Blanket modules and of the related supporting subsystems.

The testing foreseen for Demo Test Blanket modules includes the demonstration of a breeding capability that would lead to tritium self-sufficiency in a reactor and the extraction of high-grade heat suitable for electricity generation. To accomplish these goals, the ITER horizontal ports will be used to provide a relevant fusion plasma and the appropriate nuclear environment.

The purpose of the tests is to validate the design principles and the operational feasibility for the demonstration blanket system. This system includes all the basic support functions for the tritium breeding blanket. The supporting subsystems include:

1. Test Blanket Subsystem (first wall, breeding blanket, shield, and structure);
2. Tritium Extraction Subsystem (tritium removal, handling and processing);
3. Helium Coolant Subsystem (heat transfer, heat transport);
4. Coolant Purification Subsystem (helium purification and conditioning);
5. Test Blanket Remote Handling Subsystem (remote handling as related to the test blanket systems).

In addition, the basic properties and operating characteristics of the system's materials will be validated. To assess those qualities and characteristics, the test blanket systems are to be exposed directly to the ITER plasma for relatively long, continuous operation periods. These test blanket modules will be replacing shielding blanket modules; thus they must meet all applicable ITER shielding blanket requirements, including heat removal, shielding protection for the vacuum vessel welds and toroidal field magnets, and reduction of neutron streaming.

Breeding and recovery of tritium are important goals of the test program. Lithium ceramic compounds will be used as the breeder materials to be investigated. Subsystems to recover the bred tritium will be demonstrated along with test facilities to separate and remove the tritium from the coolant or purge streams. Special designs and tritium handling facilities will be required to meet the ITER safety goals and requirements.

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

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

The Test Blanket systems are to be installed and maintained through the horizontal test ports. Standard ITER remote handling equipment and procedures will be used to the maximum extent. All Test Blanket systems' plumbing and instrumentation and control shall be contained within the vacuum chamber horizontal port extensions and shall pass through the horizontal port or shielding doors. Maintenance rails and other remote handling equipment are to be provided for use within the horizontal ports. Space shall be provided in the region immediately outside the biological shield, near the ports, for tritium handling equipment while the Test Blankets systems are in place and for storage of test equipment during maintenance actions. Transport from the horizontal ports to the hot cells is to be provided as well as facilities in the hot cells for storage, maintenance, testing, refurbishing, and dismantling the test modules. The installed Test Blanket Subsystem is to be a complete assembly which can be fully tested prior to the installation. This will facilitate the installation and removal process and increase the reliability of the installation and check out procedure.

The European and the Japanese have collaborated in their approach for testing their helium cooled solid breeder test modules. During the ITER Basic Performance Phase (BPP) the European Blanket Test Module (BTM) shall occupy half of the test port allocated to the helium cooled blankets, the other half being occupied by the Japanese helium cooled ceramic breeder blanket module. The tritium subsystems for the two BTM's will be separate and placed in the pit immediately adjacent to the test port. The helium coolant loops (heat transfer, heat transport and helium purification) will also be separate and will be placed outside the pit, probably in the Tritium Building.

Parts of the BPP tests, in particular those related to tritium control and fluiddynamics (helium flow distribution, pressure drop), may be conducted before the first plasma ignition.

To facilitate handling operations, the two BTM's together with their shield shall be bolted to a water cooled frame. This frame shall be bolted to the shield blanket back plate. The frame will be supplied with water at approximately 3 MPa and 100 °C with a maximum temperature rise of 50 °C under the maximum quasi steady state heat flux. It will be made with the same materials as the main blanket/shield structure.

During the Extended Performance Phase the HCPB test module may occupy the whole or half of one port according to the numbers of blanket concepts to be tested and to the number of ports available for the blanket module testing.

2.0.2 Test Blanket Subsystem

The Test Blanket subsystem contains the DEMO Blanket test module (BTM) to be validated. In addition, it also must perform all the functions of the basic ITER shielding blanket. Thus both of these main functions must be achieved. Listed below are the functional requirements that must be met.

The Test Blanket subsystem must perform the following functions:

- Breed tritium to demonstrate the technical objectives of the Demo test program.
- Produce high-grade heat that is removed with a suitable coolant medium to demonstrate the technical objectives of the Demo test program.
- Remove the surface heat flux and the nuclear heating within the allowable temperature and stress limits.
- Reduce the nuclear responses in the vacuum vessel structural material for the ITER fluence goal.
- Protect the superconducting coils, in combination with the vacuum vessel, from excessive nuclear heating and radiation damage.
- Contribute to the passive stabilization of the plasma.
- Contribute to the reduction of neutral density between the divertor and the main plasma chamber by ~ 10^4.
- Provide a maximum degree of mechanical and structural self-support to: (1) minimize the loads transmitted to the vacuum vessel, and (2) decouple the operating temperature ranges between the test blanket system and the vacuum vessel.

Most of the functional requirements listed above assure that the test blanket modules perform the functions equally as well as the basic shielding blanket - remove the surface heat, thermalize the neutrons, protect the magnets and vacuum vessel, assure minimal leakage of coolant, and react the electromagnetic loads. The first two requirements of tritium production and power production address the new requirements to verify the DEMO Blanket materials and design approaches.

The Helium Cooled Pebble Bed (HCPB) blanket has been developed within the European Program for a DEMO relevant blanket. Forschungszentrum Karlsruhe (FZK), Commissariat a l'Energie Atomique (CEA), ENEA (Ente per le Nuove Tecnologie, l'Energia e l'Ambiente), together with ECN Petten and SCK-CEN Mol are collaborating for the further development of the HCPB DEMO blanket and design and construction of the HCPB Blanket Test Modules (BTM).

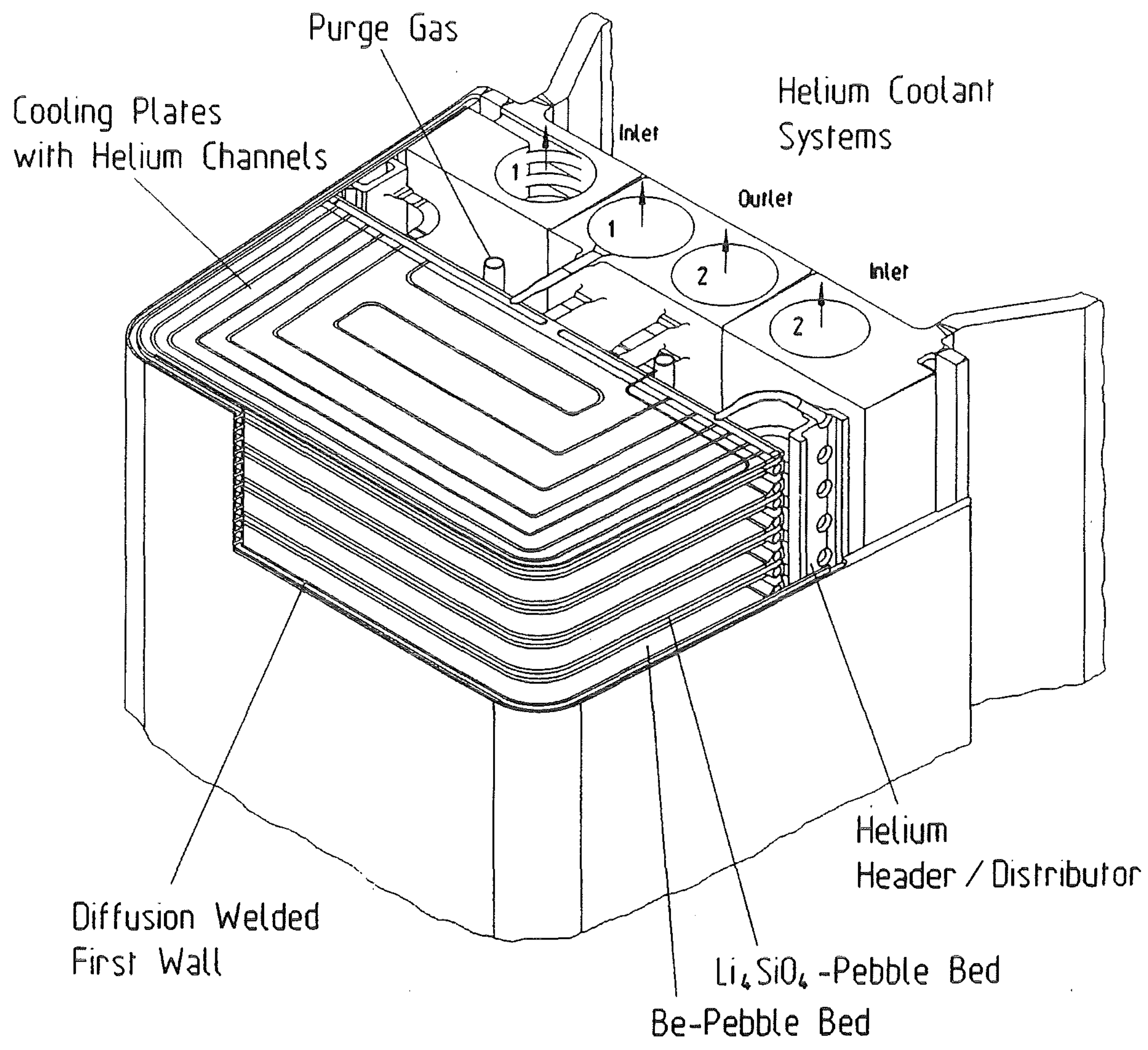


Figure 2.0.2-1 Isometric view of a poloidal portion of the DEMO outboard blanket segment around the torus equatorial plane

Fig. 2.0.2-1 shows an isometric view of the poloidal portion of the DEMO HCPB outboard blanket segment around the torus equatorial plane, where the highest power deposition, highest stresses and temperatures are expected. This portion of the HCPB blanket shall be tested in ITER. The HCPB DEMO blanket exhibits the following basic design features:

1. The ceramic breeder pebbles and the neutron multiplier are contained in a tightly closed box called blanket box.
2. The plasma facing wall of the blanket box is the first wall (FW). The back side of the blanket box is formed by a plate which contains the poloidal helium feeding and collecting manifolds.
3. The blanket box and the blanket structure are cooled by helium at 8 MPa. The coolant flows in series through the blanket box and the blanket structure.

4. The blanket structure consists of 8 mm thick cooling plates placed in toroidal-radial planes. The plates are welded to the front and side wall of the blanket box.
5. Alternatively between the plates there are slits of 11 mm thickness filled by a bed of the reference pebble of Li_4SiO_4, slightly overstoichiometric (+ SiO_2) + a small amount of TeO_2 (or alternatively Li_2ZrO_3 or Li_2TiO_3) of 0.25 to 0.63 mm diameter, and of 45 mm thickness filled by a binary bed of 1.5 to 2.3 mm and 0.1 to 0.2 mm beryllium pebbles.
6. A separate purge gas system at 0.1 MPa carries away the tritium generated in breeding material and in beryllium.
7. For safety reasons, the coolant flow is divided into two completely independent coolant systems, which feed in series the FW cooling channels and then the coolant plates in alternating directions.

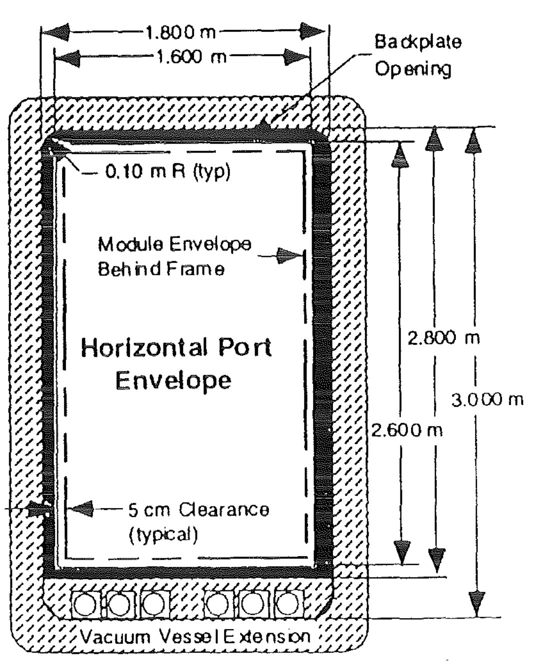


Figure 2.0.2-2 Horizontal Port Dimensions

More information on the HCPB DEMO design can be found in Ref. 2.0.2-1.

The physical size of the test blanket is determined by the constraints of the ITER horizontal port. The governing dimensions for the vacuum vessel port and the backplate opening are given in the GDRD, Section 5.3.3.5.2 and are shown in Figure 2.0.2-2. Provisions for limiter and baffle cooling pipes on the floor reduce the available height by 20 cm within the vacuum vessel extension region (1.800 m x 2.800 m). Allowing some space for the attachment frame yields an opening size for the backplate of 1.600 m x 2.600 m. The maximum size of the mounted two half modules (EU and Ja) will be 2.560 m high by 1.560 m wide after a nominal gap of 20 mm is subtracted. Within the vacuum vessel extension, the shielding should fill the envelope as much as possible. The maximum size of the shield in this region will be the vacuum vessel extension envelope less a 5 cm clearance all around for differential movement between the vacuum vessel and the backplate. The radial depth of the blanket is determined by the blanket design parameters. The first wall of the test module should be planar, without curvature, but should conform as closely as possible to the first wall of the adjacent shield blanket modules.

The ITER Shielding Blanket system is responsible for supporting the static and the dynamic loads generated by any module located within the horizontal port, including the Test Blanket subsystem (Ref. GDRD 5.19.2.2). These loads will be transmitted through a mounting system, which is a series of teeth and bolts surrounding the perimeter of the 2.600 m high by 1.600 m wide opening in the backplate (Ref. GDRD 5.3.3.5.2). The Test Blanket subsystem will provide a mating flange mounting system around the perimeter of the test module to transmit the internal loads to the

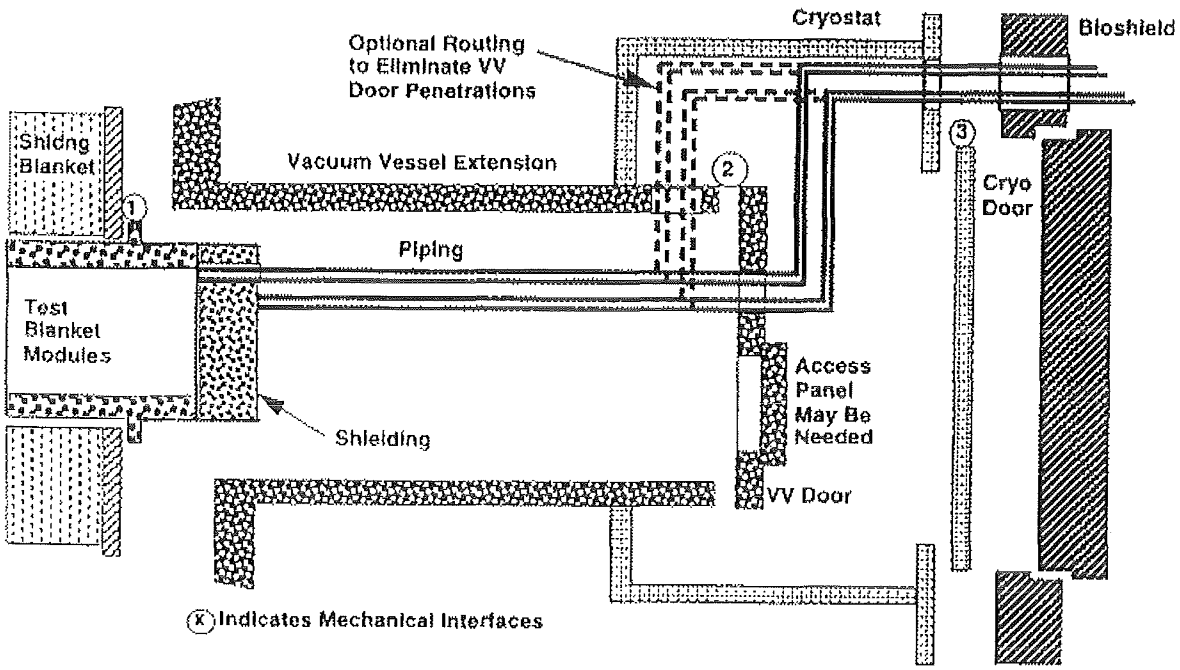


Figure 2.0.2-3 Schematic of Test Blanket Subsystem Elements in Horizontal Port

backplate. The temperature of the flange shall be designed to be compatible with the nominal operating temperature of the backplate of approximately 150 °C to reduce thermal stresses in the attachment mechanism. Electrical conductance across the mounting interface will be determined. Access to the mounting bolts will be provided for the remote handling equipment. Shielding will be provided behind the main blanket to assure the shielding requirements for the vacuum vessel and the magnets are satisfied.

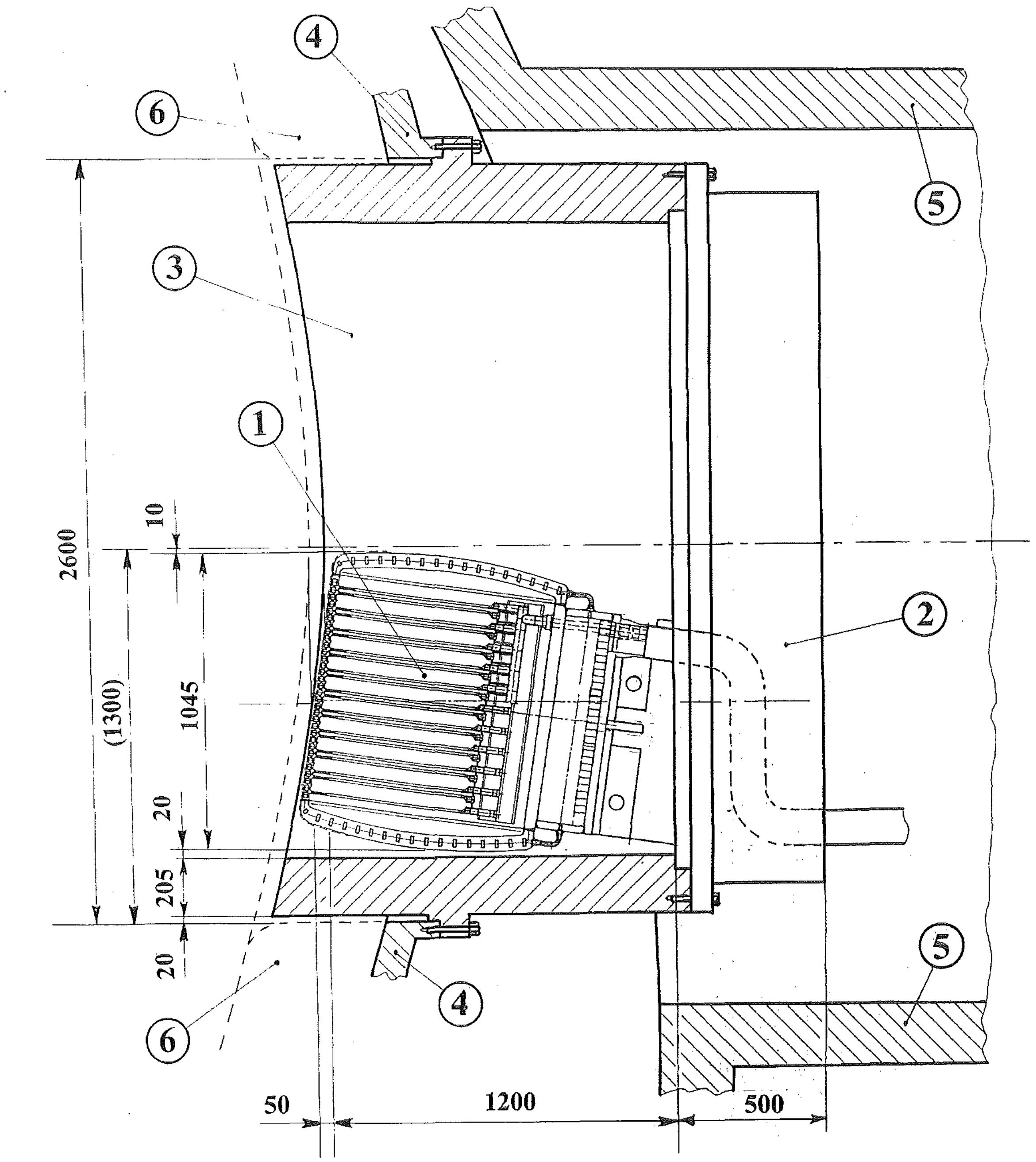
The requirement to be able to conduct blanket module testing in ITER while not adversely impacting the availability puts constraints on the design approach to the system configuration and the attendant remote handling equipment and procedures. The general approach employed in the Test Blanket system is to fully test the largest system that can be handled and installed in the horizontal ports. Figure 2.0.2-3 presents the overall scheme for the Test Blanket system installation. A support frame is mounted onto the backplate, which will carry all the static and dynamic loads and provide the proper dimensional control for alignment of the BTM within the shielding blanket modules. The primary vacuum will be sealed with a standard vacuum vessel door at the end of the vacuum vessel extension. The plumbing pipes associated with the blanket, shielding, and support frame will either penetrate the vacuum vessel closure plate or will penetrate the side wall of the vacuum vessel port wall. If the former approach is used, it will allow a complete assembly to be installed as a pretested unit without welding within the vacuum enclosure. If the latter approach is used, the vacuum vessel door will have no penetrations which will allow easy access to the plumbing and the back of the test module.

To facilitate handling operations the two BTM's (European and Japanese) together with their shield shall be bolted to a water cooled frame. This frame is bolted to the shield blankets back plate and it is designed to transmit the loads generated in the BTM's to the shield blanket back plate. Fig. 2.0.2-4 shows a vertical cross section of the support frame with the European HCPB-BTM attached to its lower part, the upper part being reserved for the Japanese helium cooled BTM.

The HCPB-BTM represents a poloidal portion of the HCPB DEMO blanket. As in the DEMO the radial toroidal cooling plates and the first wall are cooled by helium at 8 MPa flowing first in the first wall and then in the cooling plates. For safety reasons the coolant helium flows in two completely separated loops. In the blanket the coolant helium is flowing alternatively in opposite directions in the first wall and in the adjacent blanket plates. In this way the BTM temperatures are more uniform. In the reference BTM between the blanket plates there are alternatively 11 mm thick ceramic breeder pebble layers and 45 mm thick beryllium pebble layers. The tritium purging gas is helium at about 0.1 MPa flowing in radial direction from the first wall to the back of the module. The plasma side of the first wall is protected by a 5 mm beryllium layer and it recessed from the shield blanket countor by a minimum amount of 50 mm. At the upper and lower ends the HCPB-BTM is closed by covers capable to sustain a pressure of 8 MPa. During normal operation the space in the BTM other than the cooling plates and the FW is at the purge gas pressure of 0.1 MPa. However, in case of a leak from a cooling plate, it can be pressurized up to 8 MPa. Thus the blanket box, and the helium purge system connected to it, have been designed to sustain the full 8 MPa pressure. This is a double barrier against helium leakage from the cooling plates and would allow, in case of need, to wait for the next planned period for the exchange and repair of the module.

The maximum temperature of the structural material and of the ceramic pebble bed for the reference BTM amount to 507 °C an 622 °C, respectively. Power excursions to 120 % of nominal with a duration of 10 s lead to a temperature increase of 22 K in the FW structure. All calculated stresses are below the admissible limits according to ASME and RCC-MC (transients TBD). The thermal time constant of the BTM is much less than the scheduled ITER burn time; i.e. steady-state conditions are prevailing in the BTM during most of the burn time.

The second version of the test module with a modified flow scheme in the FW and an increased thickness of the ceramic pebble bed (BTM-II) allows a significant increase of the helium outlet temperature and of the maximum ceramic bed temperature at about the same FW temperature. BTM- II will be tested in ITER after the BTM-I during BPP. Calculations have also been performed for a third HCPB test module (BTM-III) to be tested during the EPP. These calculations have been performed to size the ancillary loops for the HCPB-BTM, so that the same ancillary loops could be used during the EPP period as well.



1.Blanket Test Module
2.Shield
3.Support Frame
4.Back Plate
5.Vacuum Vessel
6. Shield Blanket

Figure 2.0.2-4 Vertical cross section of the support frame with the European HCPB-BTM

Water cooled BTM Support Frame

The frame shall be made of the same structural material as the back plate and maintained by the cooling water at about the same temperature of the back plate at their contact surfaces. Also the BTM shield will be cooled by water and maintained at the same temperature of the frame at its contact surfaces.

Supply Pipes

A set of two supply and two return pipes will provide the HCPB blanket test module with high pressure helium coolant. A set of one supply and one return pipe will provide the 0.1 MPa helium for the purging of the tritium produced in the BTM.

A simple set of water pipes will be used to cool the Support Frame and the Shield. The diagnostic conduit with Instrumentation and Control System leads will penetrate the BTM Back Plate and VV Shielding Door.

Safety

The safety considerations of the blanket test module focus on the accidental safety aspects. Occupational safety and waste generation issues have not been elaborated so far. The accidental safety concerns are addressed giving first an overview on the material mass inventories and on tritium inventories in the different subsystems and on the energy sources which are the driving elements in any accident sequence. Enveloping events are identified to serve as design basis for the BTM. These design basis events are analysed with view to their short and long term evolution. Finally, the events are evaluated against a given set of safety requirements to demonstrate that the HCPB-BTM design complies with ITER safety criteria.

Two types of transient thermodynamic calculations have been carried out to determine the 3D temperature distribution in a representative section of the BTM in the course of the events with the finite element code FIDAP, and the thermal-hydraulic and heat transport mechanisms in the whole cooling subsystem with the system code RELAP. In addition, some of the events were evaluated qualitatively by deduction and extrapolation.

In the assessment of the enveloping events it is checked whether the requirements are met by the HCPB system. It can be concluded that 9 out of 16 safety requirements are shown to be fulfilled by the proposed design. Some of the postulated events in relation to certain requirements need to be further investigated. Pending these results, no fundamental safety concern has been identified so far that could violate any requirement.

Reliability

The reliability of the BTM including the supply pipes inside the vacuum vessel (VV) has been analysed using usual basic failure rates of the components like welds, pipes, and bends. An overall failure rate of less than 0.01 1/a has been obtained which yields with a BTM replacement time of 8 weeks an availability of more than 99.9 %. The reliability is dominantly determined by leaks of the pipes inside th VV. The BTM itself is very fault-tolerant; this is a consequence of the design concept which allows single failures of most internal weld without affecting the operability of the BTM and of ITER. Radiation effects have not yet been taken into account.

References

[2.0.2-1] M. Dalle Donne et al., „European DEMO BOT Solid Breeder Blanket", KfK 5429, Nov. 1994

2.0.3 Tritium Extraction Subsystem

The tasks of the tritium extraction system (which is also called purge gas system) are:

- Removal of tritium produced in the blanket test module,
- Separation and intermediate storage of the two main chemical forms of tritium, i.e. HTO and HT,
- Purification and conditioning of the purge gas.

It is a main aspect of the design that the system can be operated for one campaign of the reactor (max. 6 days) without intermediate unloading or regeneration of single components. In addition, no valve switching actions, temperature cycling or tritium transfer operations will be needed within this time span.

Principle of Operation (Fig. 2.03-1):

The helium purge gas stream containing 0.1% H_2 is sent through the breeder and beryllium pebble beds to extract the accumulated tritium (mainly by isotopic exchange). After passing an ionization chamber, the gas stream is processed in the tritium extraction system.

Removal of tritum and excess hydrogen from the helium carrier gas is accomplished in two steps:

1. tritiated water (HTO and HO) is frozen out in a cold trap operated at -100°C,
2. molecular hydrogen isotopes (HT,H_2) and gaseous impurities are adsorbed on a molecular sieve bed operated at -196°C.

The clean helium is then sent through a make-up unit where hydrogen is again added to provide a He : H_2 swamping ratio of 1000.

At the end of an experimental cycle, the tritiated water collected in the cold trap is transferred to the Water Detritiation System (WDS) which is part of the installations for the primary fuel cycle[a)]. Desorption of the molecular hydrogen isotopes from the molecular sieve bed is carried out in a secondary helium loop containing a circulation pump and a Pd/Ag diffuser. The pure hydrogen isotopes obtained at the secondary side of the diffuser are stored in uranium getter beds and, later on, transferred to the Isotope Separation System (ISS).

The tritium extraction system is located in a pit adjacent to the port of the blanket test module.

[a)] see J.E. Koonce, O. K. Kveton: Design Description Document (DDD) - Tritium Plant 3.2, Chapter 3.2E and Appendix to this report, Chapter 3.2WE

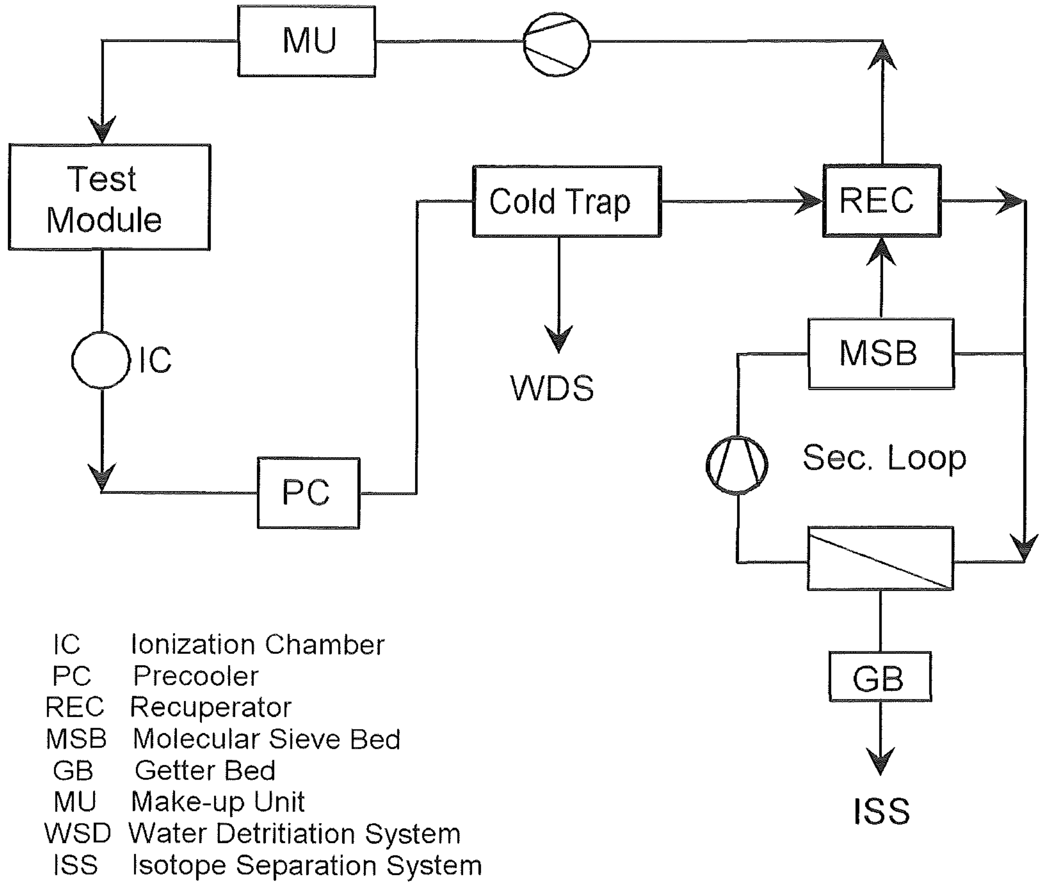


Fig. 2.0.3-1: Block Diagram of the Tritium Extraction Subsytem

2.0.4 Helium Coolant Subsystem

The cooling subsystem is designed for the European helium-cooled pebble bed (HCPB) test module to be installed in the bottom half of an equatorial test port in ITER, presumably port No. 17. It includes the primary helium heat removal loops with all components, and the pressure control subsystem. The secondary water loop subsystem with the ultimate heat sink is part of the ITER cooling system providing water flow at low temperature. A further interface to the cooling subsystem are the connections to the helium purification subsystem, taking a bypass flow of about 0.1 % of the main mass flow rate. Two separate primary heat removal loops of 2 x 50 % heat capacity are foreseen for redundancy purposes in accordance with the DEMO blanket design. The cooling subsystem will be housed in the tritium building, next to the helium purification subsystem, at a floor level about 20 m above the test module, requiring a space of about 700 m^3. A schematic flow diagram is shown in figure 2.0.4-1.

The thermal-hydraulic design parameters are as follows: The maximum heat to be removed from the test module amounts to 2.3 MW. Nominal primary helium coolant conditions are 250 °C and 350 °C (later on 250 and 450 °C) at module inlet and outlet, respectively, and 8 MPa of pressure. The total flow rate in both primary helium loops is 3.7 kg/s. The secondary cooling water provided by ITER has a temperature of 35/75 °C at the heat exchanger inlet/outlet, a pressure of 0.5 to 1.0 MPa, and a maximum mass flow rate 22 kg/s.

Main components in each loop are the heat exchanger, circulator, electrical heater, dust filter, and pipework. The total helium mass inventory in one loop amounts to 21 kg and the overall pressure loss is about 0.27 MPa, most of which occurring in the test module proper. The heat exchanger is assumed to be a straight tube bundle heat exchanger, or alternatively consisting of U-tubes, with high pressure helium flowing inside the tubes. The design specification for the circulator is as follows: temperature 300 °C, pressure 9.6 MPa, mass flow rate 1.9 kg/s at a pumping head of 0.27 MPa at 80 % of maximum speed and at 250 °C inlet temperature, speed variation max/min of at least 4. The electrical heater with an electrical power of 100 kW which is installed in a bypass to the heat exchanger is needed for baking the test module first wall at 240 °C and for heating the whole cooling subsystem. A filter unit is installed in the hot leg of the main loop, accumulating residual dust and particles from fabrication, and erosion particles down to a size of typically 10^{-6} m. For the main pipework an outer diameter of 168.3 mm and a wall thickness of 10 mm have been chosen for the part external to the cryostat. Inside the cryostat smaller pipes are foreseen (114.3 mm outer diameter, 8 mm wall thickness) to limit the size of pipe penetrations in the cryostat to 130 mm. This results in flow velocities of between 15 and 41 m/s. The total pipe length sums up to 120 m per loop. The number of valves in the main loops has been kept at a minimum to avoid inadvertent closure which would mean loss of heat sink. All of the piping and components in the primary cooling subsystem will be constructed of austenitic steel.

The pressure control subsystem is needed for evacuation, helium supply, pressure control, and overpressure protection. The components are conventional and of relatively small size, except for the storage and dump tanks.

Activation of cooling subsystem components installed in the tritium building is expected to be generally low allowing controlled personnel access. Remote handling is envisaged for connection and disconnection of the BTM by the aid of the transporter, and for replacement of the dust filter insert.

All components of the heat removal subsystem such as heat exchangers, circulators, electrical heaters, dust filters, tanks, and valves will be preassembled at the factory and delivered to the site as functional units. High quality assurance standards are applied to all assembly procedures. The large components of the cooling subsystem installed in the tritium building require a crane with a load capacity of about 2 tons.

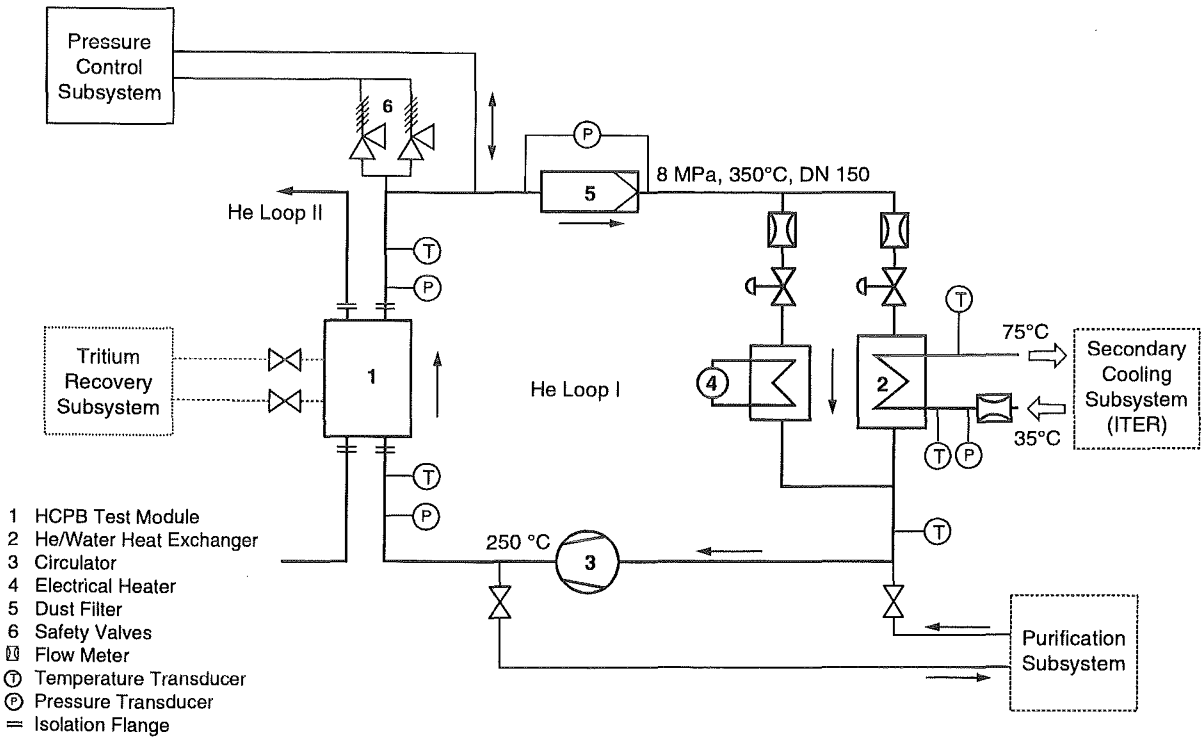


Fig. 2.0.4-1 Helium Coolant Subsystem Flow Diagram

The following preliminary control scheme is proposed for pulsed operation: The principal objective is to keep the test module inlet temperature at 250 °C. The secondary cooling water inlet temperature is kept at 35 °C, the circulator is operated at rated speed, the electrical heaters are turned off, and flow partition through the heat exchanger and heater bypass (with the heater turned off) is controlled as to maintain the inlet temperature close to 250 °C. During longer shutdown periods decay heat removal is achieved at reduced circulator speed, or by natural convection.

2.0.5 Coolant Purification Subsystem

Two coolant purification systems are provided, one for each of the two main coolant systems (Fig. 2.03-1). They are designed to purify 0.1% of the helium coolant stream. The specific tasks of a purification system are:

- to extract hydrogen isotopes as well as solid, liquid or gaseous impurities from the main coolant system;
- to remove condensed water that may be entrained in the cooling gas due to leakages or failures of the heat exchanger tubes.

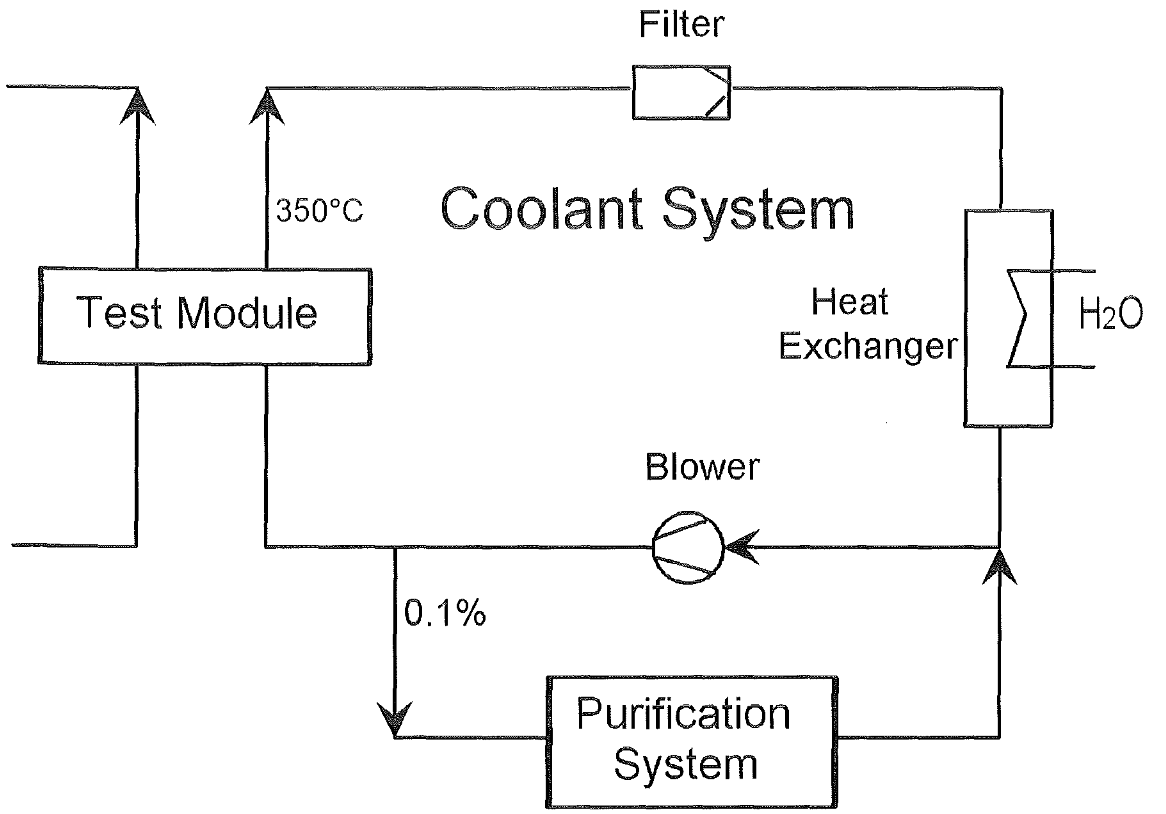


Figure 2.0.5 -1 : Block Diagram showing the interrelation between Helium Coolant Subsystem and Coolant Purification Subsystem

Principle of Operation

The 0.1 % fraction of the coolant gas stream entering the purification system downstream of the coolant blower is first sent through a water separator to remove condensed water that may be present as a consequence of water leakages in the heat exchanger. Then, an oxidizer unit is applied to convert all molecular hydrogen isotopes into water ($Q_2 \rightarrow Q_2O$, Q = H, D, T). This water is frozen out in a cold trap operated at -100°C while the remaining impurities are adsorbed on a molecular sieve bed operated at LN_2 temperature (- 196°C). The pure helium is warmed up again and returned into the main coolant loop upstream of the blower.

The coolant purification system can be operated for one reactor campaign (max. 6 days) without intermediate unloading or regeneration of single components. In addition, no valve switching actions, temperature cycling or tritium transfer operations will be needed during this time span.

At the end of a campaign, the cold trap is warmed up, the liquified water is drained into a transportable water container and then transferred to the Water Detritiation System (available for the primary fuel cycle). The gaseous impurities desorbing from the molecular sieve bed during regeneration are sent to the Radioactive Waste Gas System.

2.0.6 Test Blanket Remote Handling Subsystem

All equipment to be used in the horizontal ports should be designed for radial installation and removal of components through the port extensions. Since the equipment will be inside the Bioshield and will be highly activated after reactor operation, it will be necessary to use remote handling systems for all operations within the Bioshield boundaries. This requirement will apply to the Test Blanket Subsystem.

The design of the remote handling system of the test blanket modules is dependent upon the piping system layout within the port extension. One of the project recommendations is to minimize the amount of remote operations inside the port extension. A concept was developed which combines the blanket modules, the shielding assembly, the coolant pipes and the vacuum vessel closure plate as one super assembly. This allows full functional testing of the assembly prior to installation within the port. This will also reduce the amount of time required to remove and install a test blanket assembly and eliminate remote operations inside the port extension.

The remote handling system for the blanket assemblies will take full advantage of the equipment designed by the JCT to minimize duplication of efforts and to standardize system operations. As a result, the current design of the remote handling system will utilize a series of transporters; each is designed to perform a certain task. All operations that are identical to other ITER operations will use the same ITER system to perform, such as removing the bioshield plug and the cryostat closure plate. Operations that are specific to the test blanket system will be integrated into the overall system design.

2.1 Detailed System Description

2.1.1 General Design Description

The test blanket system is based on the requirements listed in Section 1. The detailed design of the system components will be analyzed to demonstrate that these requirements can be met within allowable engineering parameters. The analyses refer to the reference ceramic breeder material Lithium Orthosilicate. The principal components in the test blanket are:

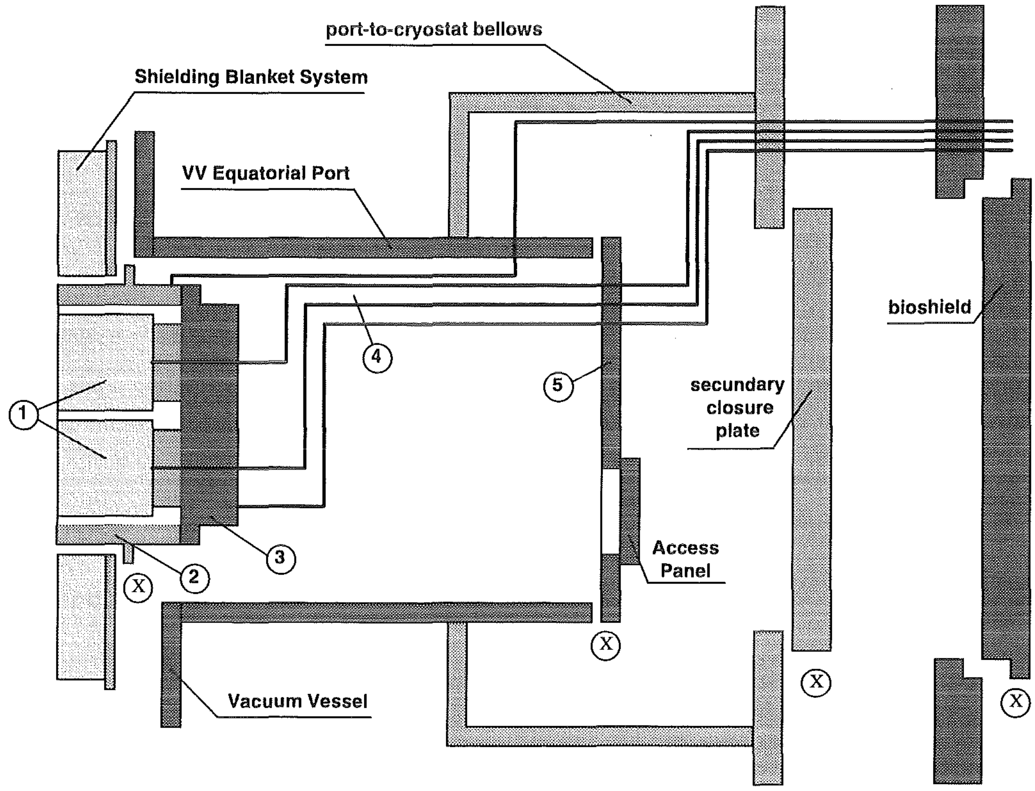
1. the Test Blanket Subsystem (first wall, breeding blanket, shield, and structure);
2. the Tritium Extraction Subsystem (tritium removal, handling, and processing);
3. the Helium Coolant Subsystem (heat transfer and transport);
4. the Coolant Purification Subsystem;
5. the Test Blanket Remote Handling Subsystem (remote handling as related to the test blanket systems).

The test blanket subsystem encompasses the functions of the first wall, breeding blanket, shield, and structure. Like the basic ITER shielding blanket, one of its principal functions is to remove surface heat flux and energy from the plasma during normal and off-normal operational conditions. It also incorporates a shield section designed to reduce the nuclear responses in the vacuum vessel and, together with the vacuum vessel, shield the superconducting coils. In addition to these requirements, the test blankets breed sufficient tritium to demonstrate self sufficiency in a DEMO reactor and to produce and extract high grade heat suitable for electric power production. The test blanket also has structural elements that provide the structural and thermal decoupling from the vacuum vessel while minimizing the eddy currents in the vacuum vessel. The heat generated within the test blanket is removed with a compatible heat removal system that provides both thermal and safety protection. The test blanket subsystem is designed so that it is sufficiently reliable and can be readily removed and replaced so that the availability of ITER is not adversely impacted. It also is designed to be compatible with the primary vacuum and safety boundaries so that the basic ITER safety requirements can be met.

These functions correspond to the requirements documented in Section 2.0.2 of this document. The ITER GDRD also lists the test blanket system functional, configurational, and specific requirements in 5.19.

2.1.1.1 Test Blanket Subsystem Design Description

The test blanket subsystem is the in-vessel subsystem that contains the blanket test modules. The BTMs are connected to the blanket shielding system by means an isolation frame. This frame will provide a standarized interface with the ITER basic structure and a better shielding capability as well as neutronic and thermal isolation from the basic shielding blanket. The design of the frame and in particular the plumbing layout is strongly dependent on the adopted remote handling procedures. If

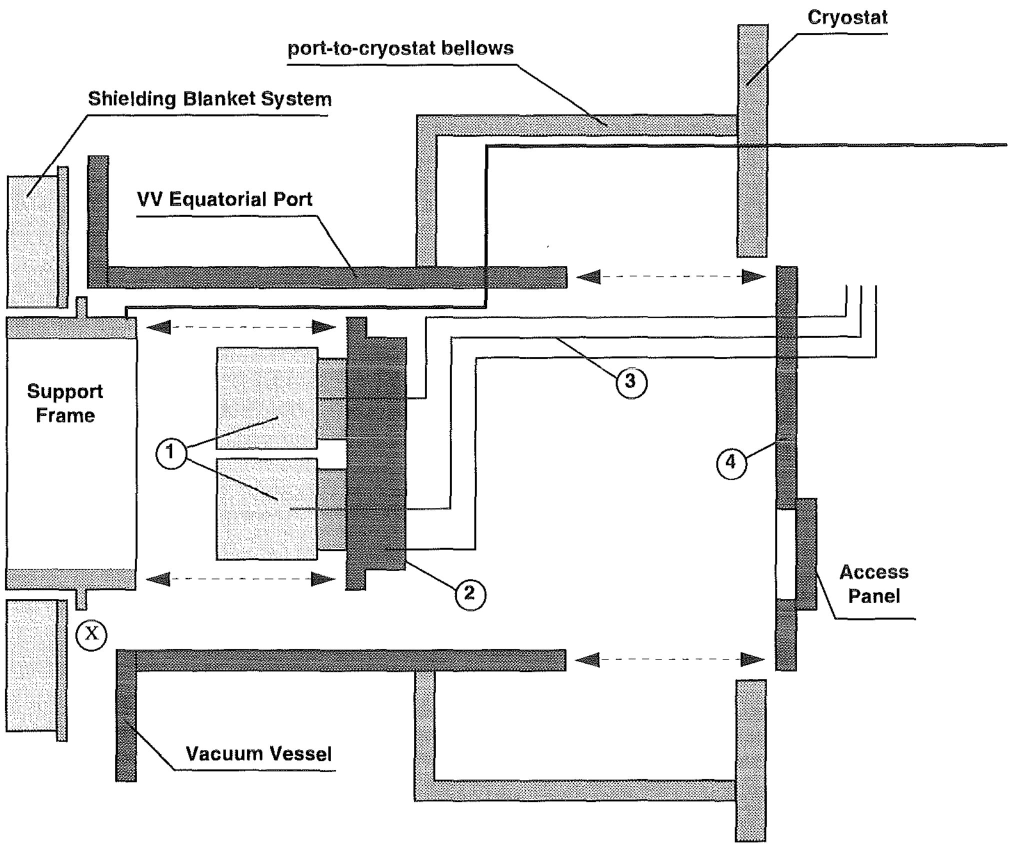


1. Blanket Test Modules
2. Support Frame
3. Shield
4. Plumbing
5. Vacuum Vessel Closure Plate

(x) indicates mechanical interfaces

Figure 2.1.1.1-1 Schematic of the Test Blanket Subsystem in the Horizontal Port

the solution without welding and cutting in the vacuum vessel horizontal port is chosen (as discussed in section 2.0.2), the arrangement of the Test Blanket Subsystem inside the port can be schematically represented by Fig. 2.1.1.1-1. The BTMs are provided with a mating flange mounting system (support frame and shield) around the perimeter to transmit the internal loads to the ITER shielding system. The support frame is bolted to the ITER backplate; the shield with the BTMs attached to it is bolted to the support frame. The plumbing extends through the VV closure plate, which is also part of the subsystem. This plate is a modified version of the ITER closure plate due to the use of a separate hatch to access the vacuum vessel port to disconnect the attach bolts. This arrangement enables the test blanket assembly (BTMs, Shield, related plumbing and VV closure plate) to be a self contained unit (see Fig.2.1.1.1-2). Indeed this assembly can be completely tested prior installation with no welding inside the vacuum vessel. Figures 2.1.1.1-3 is an isometric view of the mid-plane port illustrating the routing of the test blanket piping through the primary closure plate. This approach requires a separate hatch (Fig, 2.1.1.1-4). Note that all penetrations through the vacuum vessel and cryostat boundaries will require vacuum tight flexible connections such as bellows (Fig. 2.1.1.1-4). The design of such connections will be similar to those approved by the JCT.



1. Blanket Test Modules
2. Shield
3. Plumbing
4. Vacuum Vessel Closure Plate

(x) indicates mechanical interfaces

Fig. 2.1.1.1-2 Schematic of the Test Blanket Assembly

During the Basic Performance Phase of ITER two test modules will be tested. The first (BMT-I) will have the same configuration of the DEMO blanket, however with a higher Li^6-enrichment (75 % instead of 25 % of the DEMO). The second (BMT-II) will also have a Li^6-enrichment of 75 %, however the geometry of the blanket will be slightly modified to achieve higher, and thus more relevant, temperatures at the coolant helium outlet and in the ceramic breeder pebbles (see Section 2.1.1.1.2).

Neutronic and thermomechanical design calculations have been performed also for a BTM to be tested during the EPP (BTM-III). Also in this case a half module surrounded by a frame has been assumed. These calculations have been performed to assess the requirements posed by the BTM to the ancillary loops(helium coolant loop, helium purification plant, tritium extraction subsystem) during the EPP.

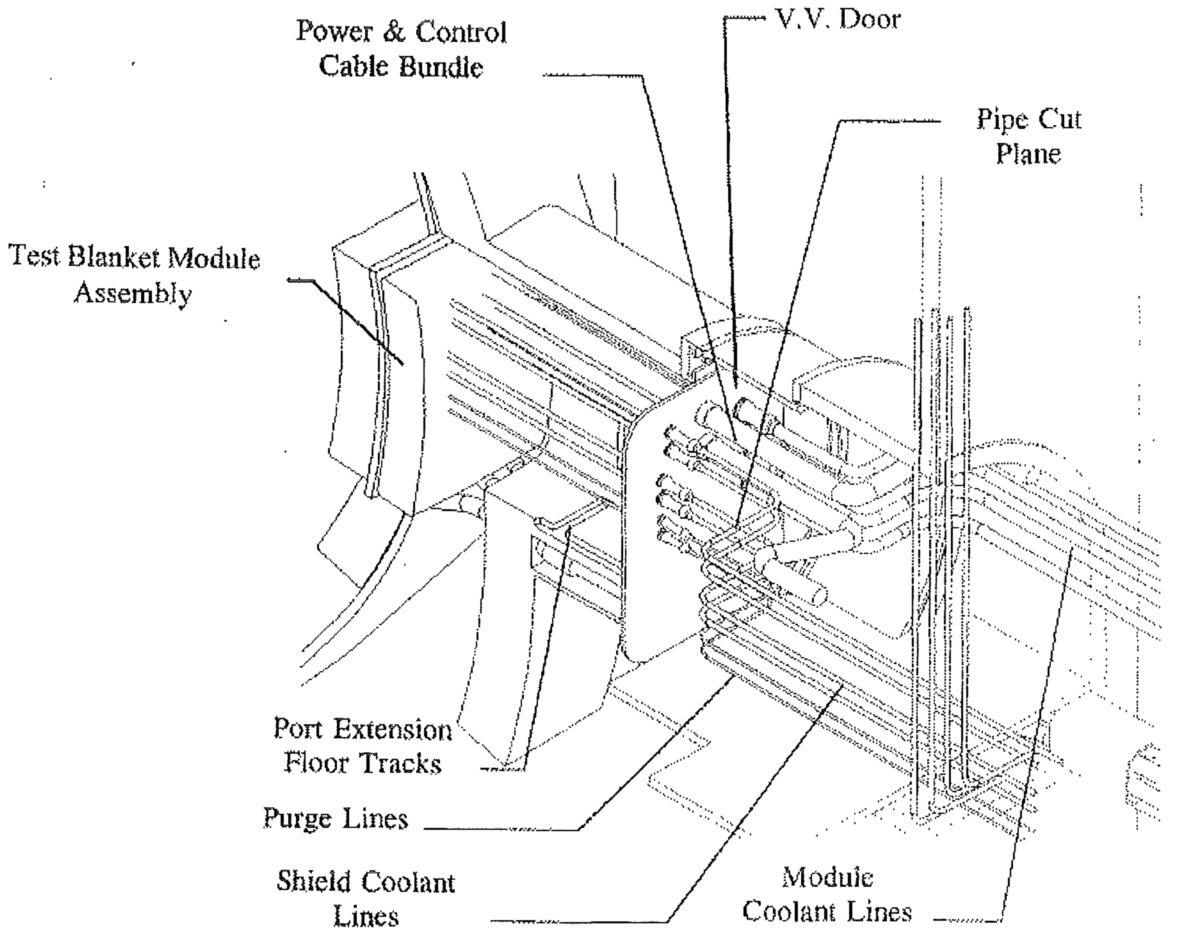


Fig. 2.1.1.1-3 Schematic of the Test Blanket Assembly installed with piping through the Vacuum Vessel Closure Plate.

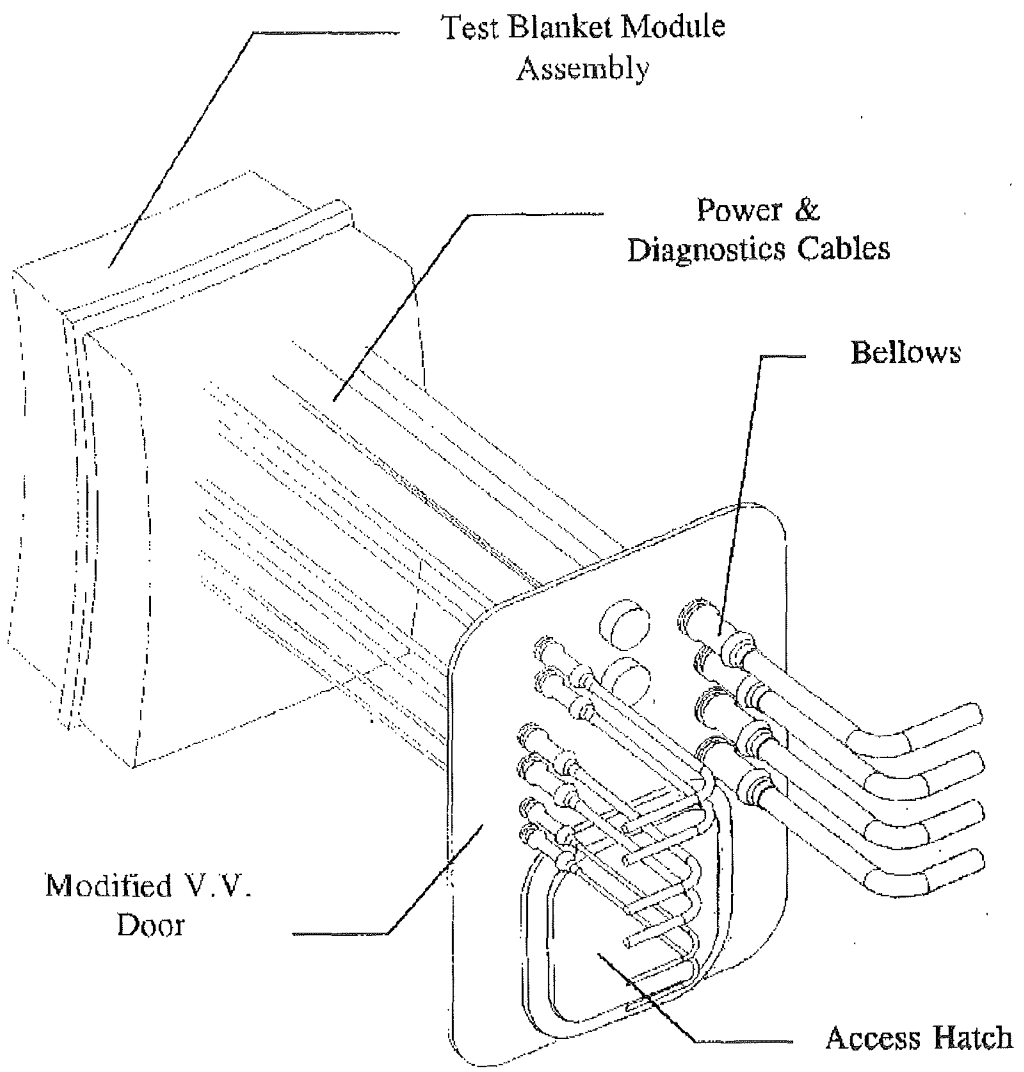


Fig. 2.1.1.1-4 Schematic of the Test Module Installed with Piping through the Vacuum Vessel Closure Plate showing the access hatch.

2.1.1.1.1 Nuclear Design

Calculational procedure and modelling

Three-dimensional Monte Carlo calculations with the MCNP-code [2.1.1.1.1-1] and nuclear cross-sections from the FENDL-1 data library [2.1.1.1.1-2] form the basis of the nuclear design analyses for the HCPB blanket test modules in ITER. Use is made of a 9° torus sector model of ITER provided by the ITER-JCT nuclear analysis group for the current 20 TF-coil design. A horizontal outboard test blanket port has been inserted into this model with the proper dimensions of 2.6 m times 1.60 m at the level of the blanket back plate. Inside the horizontal port, a water-cooled steel support frame and two test modules of the HCPB-type were integrated. The upper one simulates the Japanese Helium Cooled Solid Breeder Blanket module (assumed with the same geometry and material composition as the HCPB-BTM), while the lower one is the actual HCPB-BTM. The HCPB -BTM model has been adopted from the DEMO blanket model [Ref. 1 of Section 2.0.2] with the following exceptions: in toroidal direction, the BTM is rectangular instead of trapezoidal in order to reduce neutron streaming between the module and the steel frame; in poloidal direction, the modules follow the contour of the ITER shielding blanket first wall with an outward recess of 5 cm. As for ITER, there is a 5 mm beryllium protection layer for the HCPB-BTM first wall. Vertical and horizontal cross-sections of the model are displayed in figs. 2.1.1.1.1- 1 through 2.1.1.1.1- 3.

Three-dimensional transport calculations have been performed to obtain the nuclear heating and the tritium production in the lower BTM and to assess its shielding performance with regard to the radiation loads on the TF-coil and the vacuum vessel adjacent to the test blanket port by taking into account the streaming effect. In addition, the neutron wall loading was calculated at the first wall of the lower BTM. A fusion power of 1500 MW was assumed in the calculations.

Three different cases were considered in the nuclear analysis: BTM-I and -II for tests in the basic performance phase (BPP), BTM-III for test in the enhanced performance phase (EPP). BTM-I essentially shows the same configuration as the DEMO blanket, however, with a Li^6-enrichment of 75 at% instead of 25 at%. BTM-II has a slightly increased thickness of the ceramics pebble bed layer (14 mm instead of 11 mm) to achieve higher temperatures in the breeder and the helium coolant. BTM-III is identical to BTM-I in its geometrical configuration, but is foreseen to be tested in the ITER EPP. This has a strong impact on the nuclear performance of the BTM since ITER is equipped with a breeder blanket during the EPP that is neutronically very similar to the HCPB -BTM. During the BPP, however, ITER uses only shielding blanket modules with a detrimental effect on the HCPB-BTM. For all three cases, the Support Frame was assumed to be composed of a 80% steel, 20% H_2O homogenised mixture.

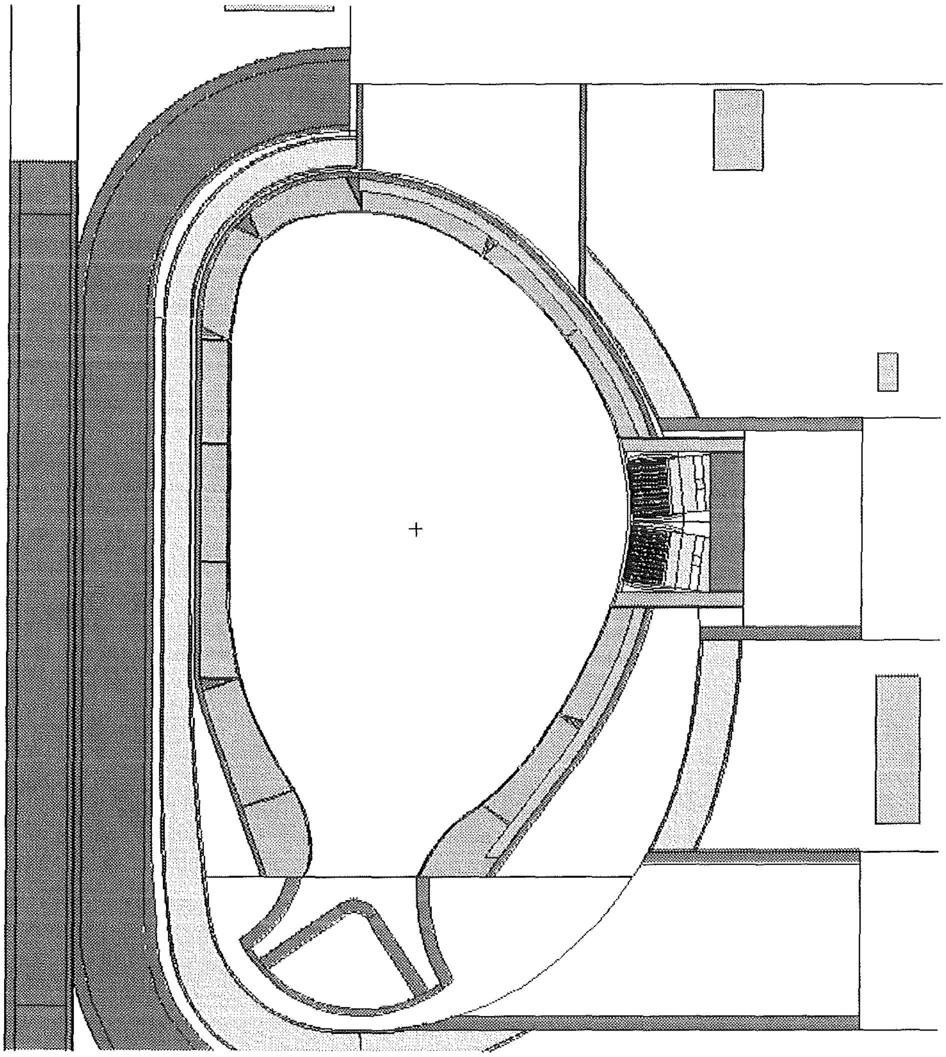


Fig. 2.1.1.1.1- 1: Vertical cross-section of the MCNP torus-sector model

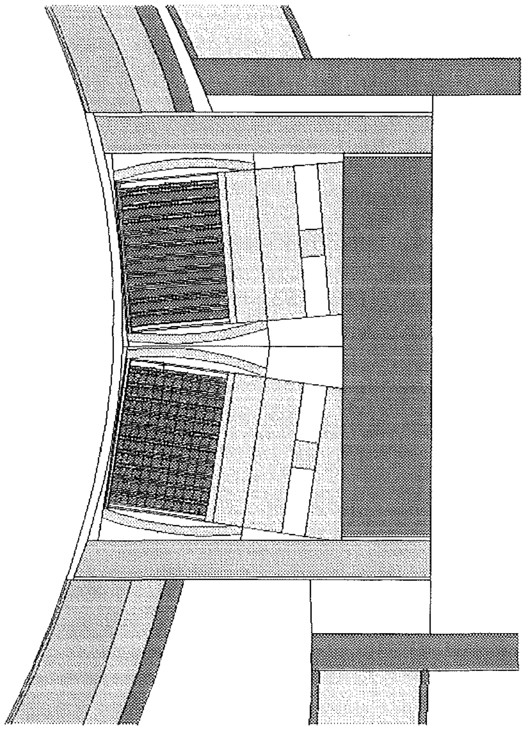


Fig. 2.1.1.1.1- 2: Vertical cut through port section of the MCNP model

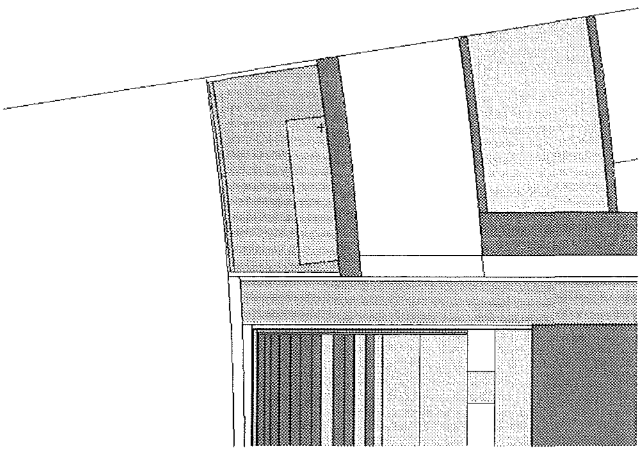


Fig. 2.1.1.1.1- 3: Horizontal cut through the test blanket port section.

Calculational conditions

The presented neutronic results have been obtained throughout by three-dimensional Monte Carlo calculations performed on an IBM RS/6000 workstation cluster. For the nuclear heating calculations, typically 1.5 million (BPP) and 600,000 (EPP) source neutron histories were followed to arrive at accurate statistical scorings. For the shielding calculation, about 4 million source neutron histories were tracked while geometry splitting with Russian Roulette was applied for variance reduction purposes.

Calculational results

Neutron wall loading

A 1.94% fraction of 14-MeV source neutrons is impinging onto the lower HCPB-BTM first wall. This results in a neutron wall loading of 1.197 MW/m².

Tritium production

The tritium production in the test module amounts to 0.221, 0.225 and 0.265 g/day for BTM-I, -II and -III, respectively. Based on the 1.94% fraction of source neutrons entering the BTM, this would result in local tritium breeding ratios of 0.99, 1.01, and 1.19, respectively. Note that there is a strong albedo effect of the ITER shielding (BTM-I, -II) and breeding blankets (BTM-III) on the HCPB-BTM. For BTM-I, the radial tritium production rate, averaged over the poloidal and toroidal extension of the BTM, is presented in table 2.1.1.1.1-2.

Nuclear power production

The total nuclear power production in the HCPB test blanket module is 1.52 MW for BTM-I and -II (BPP) and 1.84 MW for BTM-III (EPP). The corresponding energy multiplication amounts to 1.24 for the BPP and 1.50 for the EPP. Table 2.1.1.1.1-1 shows the breakdown of the power production for the test blanket module components. In any case the power generated in the steel frame, being composed of a 80% SS-316, 20 % H_2O mixture, exceeds the power generation in the test blanket module.

Nuclear power density distribution

The nuclear power density distribution has been calculated on a fine radial-toroidal segmentation grid. In poloidal direction the nuclear power density is averaged over the poloidal height of the BTM which amounts to 84 cm. Note that the neutron wall load is at its maximum at this poloidal height. Table 2.1.1.1.1-2 reproduces the radial power density and the tritium production rate distribution for BTM-I. The displayed date are toroidal average values. The actual variation in toroidal direction is only weak. The maximum power densities are displayed in table 2.1.1.1.1-3 for BTM-I, -II and III.

Table 2.1.1.1.1-1 Power production [MW] in the HCPB- BTM and the steel frame.

	BPP		EPP
	BTM-I 11 mm breeder layer thickness	BTM-II 14 mm breeder layer thickness	BTM-III 11 mm breeder layer thickness
First wall	0.18	0.18	0.26
Beryllium	0.39	0.37	0.42
Breeder	0. 47	0.50	0.56
Structure	0.48	0.47	0.60
Total Nuclear Power BTM	1.52	1.52	1.84
Energy Multiplication	1.24	1.24	1.50
Steel frame	2.19	2.18	3.17

Table 2.1.1.1.1-2 Power densities and tritium production rate distribution in the HCPB- BTM-I.

	Nuclear power density [W/cm^3] in the HCPB-BTM			Tritium production rate [cm^{-3}s^{-1}]
Radial distance to first wall (cm)	Beryllium	Ceramics	Steel	
0.5	8.05	-	-	
1.00	-	-	10.0	
2.40			9.54	
3.00			8.86	
6.00	4.71	18.5	7.55	1.87E+13
9.00	3.89	15.6	6.67	1.59E+13
13.0	3.11	13.6	5.35	1.42E+13
18.0	2.31	11.9	4.25	1.29E+13
23.0	1.60	9.71	3.25	1.08E+13
28.0	1.12	7.55	2.31	8.57E+12
33.0	0.776	5.46	1.63	6.27E+12
38.0	0.550	4.54	1.32	5.30E+12
43.0	0.391	3.35	0.998	3.92E+12
48.0	0.276	2.56	0.604	3.03E+12
53.0	0.201	2.11	0.580	2.52E+12
57.3	0.152	1.63	0.508	1.96E+12
61.3			0.325	
78.3			0.229	
1.005			0.128	

Table 2.1.1.1.1-3 Maximum power densities [W/cm^3] in the HCPB- BTM's

	BPP		EPP
	BTM-I 11 mm breeder layer thickness	BTM-II 14 mm breeder layer thickness	BTM-III 11 mm breeder layer thickness
First wall/beryllium	8.05	8.02	9.21
First wall/steel	10.0	10.1	15.7
Be pebble bed	4.71	4.69	5.12
Ceramics pebble bed	18.5	16.2	29.9
Cooling plate/steel	7.55	7.96	5.12

Shielding efficiency

Neutron streaming through the test blanket port will deteriorate the shielding efficiency of the blanket/shield system. The geometrical and material configuration of the test blanket port, therefore, has been optimised for a maximum shielding efficiency by reducing void gaps to a minimum and by designing the steel support frame as efficient radiation shield.

The shielding efficiency was assessed by calculating the radiation loads to the vacuum vessel and the TF-coil adjacent to the test blanket port. An integral operation time of 3 years was assumed in these calculations. This corresponds to a total BTM first wall fluence of 3.6 MWa/m^2. The nuclear responses obtained in the Monte Carlo shielding calculations are again for the poloidal extension of the lower BTM which amounts to a height of 84 cm. At this poloidal level, the neutron wall load is at its maximum. In toroidal direction, the responses apply for those parts which are located closest to the test blanket port. For the TF-coil, the most crucial radiation loads are the fast neutron fluence to the superconductor, the peak nuclear heating in the winding pack, the radiation dose to the copper insulator in terms of accumulated dpa and the radiation dose absorbed by the epoxy resin insulator. For the vacuum vessel, the helium gas production at the weldings are of concern when reweldability of SS-joints is considered. The assumed allowed limit is about 1 appm helium.

Table 2.1.1.1.1-4 shows the calculated radiation loads to the TF-coils by assuming the BTM-I present in the test blanket port. For most of the responses, the test blanket system can meet within a safety factor 2 the required radiation load limits as being specified in the GDRD for the TF-coil.

The accumulated helium production in the SS-316 front plate of the vacuum vessel amounts to $\cong$ 0.39 appm for the assumed 3 FPY operation which satisfy the joint reweldability criterion within a safety factor 2.

Table 2.1.1.1.1-4 Radiation loads to the TF-coil, 3 FPY operation, 1500 MW fusion power

	BTM-I 3d -calculation	Radiation load limits (GDRD)
Peak dose to electrical insulator (Epoxy) [rad]	$3.5 \cdot 10^{9}$	$5 \cdot 10^{9}$
Peak displacement damage to copper stabiliser [dpa]	$1.1 \cdot 10^{-3}$	$6 \cdot 10^{-3}$
Peak fast neutron fluence (E>0.1 MeV) to the NB_3Sn superconductor [cm^{-2}]	$2.9 \cdot 10^{18}$	$1 \cdot 10^{19}$
Peak nuclear heating in winding pack [$mWcm^{-3}$]	0.47	1.0

References

[2.1.1.1.1-1] J. F. Briesmeister (ed.): MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A, LA-12625-M, November 1993

[2.1.1.1.1-2-] FENDL/MC-1.0 - Library of Continuous Energy Cross-Sections in ACE Format for Neutron-Photon Transport Calculations with the Monte Carlo N-Particle Transport Code System MCNP4A, generated by R. E. MacFarlane by processing FENDL/E-1.0, Summary Documentation by A. B. Pashchenko, H. Wienke and S. Ganesan, IAEA-NDS-169, Rev. 3, November 1995

2.1.1.1.2 Thermohydraulic Design

Ref. [2.1.1.1.2-1] illustrates in detail the methods of the thermohydraulic and mechanical stress calculations for the DEMO blanket. Here, it will suffice to recall the groundrules used for the present calculations and the obtained results.

Basic calculational assumptions

The groundrules can be summarized as follows:

1. The effective thermal conductivity of the bed of Li_4SiO_4 pebbles has been measured at FZK. For the bed of 0.25 - 0.63 mm Li_4SiO_4 pebbles in helium the measured effective thermal conductivity data may be correlated by the equation K_e [W/mK] = 0.708 + 4.51 x 10^{-4} T + 5.66 x 10^{-7} T^2 with T in degree centigrades [2.1.1.1.2-2]. The heat transfer coefficient between pebble bed and containment wall is equal to 0.6 W/cm^2K according to the Schlünder correlation [2.1.1.1.2-3].
2. The effective thermal conductivity of the binary beryllium bed (1.5 - 2.3 mm and 0.1 - 0.2 mm beryllium pebbles) has been obtained by interpolating the experimental results of similar beryllium and Li_4SiO_4 pebble beds [2.1.1.1.2-2]. The used correlations are:

$$k_e\,[W/mK] = 6.235\left\{1+353\left[\alpha_{Be}(T_m-T_o)-\alpha_{Ma}(T_{Ma}-T_o)+\left(\left(1+\frac{\Delta V}{V}\right)^{1/3}-1\right)\right]\right\}$$

$$\alpha[W/m^2K] = 3308\left\{1+383.1\left[\alpha_{Be}(T_m-T_o)-\alpha_{Ma}(T_{Ma}-T_o)+\left(\left(1+\frac{\Delta V}{V}\right)^{1/3}-1\right)\right]\right\} x\left[1+9.239\,x\,10^{-4}T_w\right]$$

where

where	k_e [W/mK]	= effective thermal conductivity of the bed
	α [W/m^2K]	= heat transfer coefficient between pebble bed and containment wall
	T_m [°C]	= average temperature of the pebble bed
	T_o [°C]	= temperature at which the bed filling operation has been performed ≈ room temp.
	T_{Ma} [°C]	= average temperature of the bed containing wall of MANET
	T_W [°C]	= local wall temperature
	α_{Be} [K^{-1}]	= thermal expansion coefficient of beryllium at Tm [2.1.1.1.2-4]
	α_{Ma} [K^{-1}]	= thermal expansion of MANET at T_{Ma} [2.1.1.1.2-4]
	$\Delta V/V$	= volume swelling of beryllium under neutron irradiation

For the present calculations the Beginning Of Life (BOL) situation has been considered where the highest pebble bed temperatures are expected, thus $\Delta V/V$ = 0. This is a pessimistic assumption as the beryllium swelling increases the bed thermal conductivity. Furthermore in the present calculation the term in T_W for the calculation of the wall heat transfer coefficient α has been neglected to simplify the calculations. This term has not a large effect on α and in any case it is pessimistic to neglect it.

3. The radial and poloidal power density distribution has been obtained by the three-dimensional neutronic calculations (Section 2.1.1.1). For the calculation of the blanket power a heat flux on the first wall of 0.25 MW/m^2 has been assumed, while for the first wall maximum temperature calculations the local heat flux of 0.5 MW/m^2 has been assumed.

Results

The two-dimensional steady-state temperature calculations have been performed for a radial-toroidal section with a poloidal height of six FW coolant channels corresponding to four blanket coolant plates (Fig. 2.1.1.1-1). The opposite directions of the coolant helium flow and the resulting helium temperature differences at the radial-toroidal boundary sufaces of the model have been accounted for by an iterative adjustment of the temperatures of these surfaces. The calculations have been performed with the FE computer code ABAQUS for the three modules BTM-I, BTM-II and BTM-III (see Section 2.1.1.1). In all these three cases, it has been assumed that the BTM will occupy only half of the total height of the port. This means that, if during the ITER EPP, a full port should be available for the HCPB blanket, two modules could be tested at the same time.

Table 2.1.1.1.2-1 shows the results of the 2D steady-state calculations compared with the corresponding DEMO values. In all the cases Li^6-enrichment of 75 % has been chosen for the BTM's, although for the DEMO a Li^6-enrichment of 25 % is sufficient. In case of the module BTM-I, with exactly the DEMO blanket geometry, the maximum FW temperature approaches that of the DEMO. However the temperatures in the blanket are considerably lower than in the DEMO. This is due to the fact that the neutron load, and thus the power densities, are considerably smaller than in the DEMO, while the maximum heat flux on the first wall is the same. The BTM-II is proposed to obtain higher, and thus more DEMO relevant, temperatures in the blanket. In this case, however, the BTM geometry scheme is slightly different from that in the DEMO. Namely the coolant helium flows in series through two FW coolant channels before entering the blanket region, rather than in one as in the DEMO blanket. Furthermore the thickness of the ceramic pebble layers has been increased from 11 to 14 mm. Fig. 2.1.1.1.2-1 to 2.1.1.1.2-3 show the temperature distributions for the three cases.

Table 2.1.1.1.2-1 Results of the thermal-hydraulic calculations for the three BTM's

	ITER Test Module			DEMO
	BPP		EPP	
	BTM-I	**BTM-II**	**BTM-III**	**Blanket**
Li^6-enrichment	75 %	75 %	75 %	25 %
Total Power [MW]	1.85	1.9	2.2	2500
Total helium mass flow [kg/sec]	3.6	2.1	3.6	2400
Helium pressure [MPa]	8	8	8	8
Helium pressure drop in BTM [MPa]	0.19	0.22	0.19	0.24
Helium inlet/outlet temp. [°C]	250/350	250/420	250/366	250/450
Max. power density [MW/m^3] in				
structural material	10	10	16	25
beryllium pebble bed	5	5	5	15
ceramic pebble bed	19	16	30	37
Maximum temperatures [°C]				
structural material	509	507	526	520
beryllium pebble bed	403	457	421	637
ceramic pebble bed	618	742	778	907

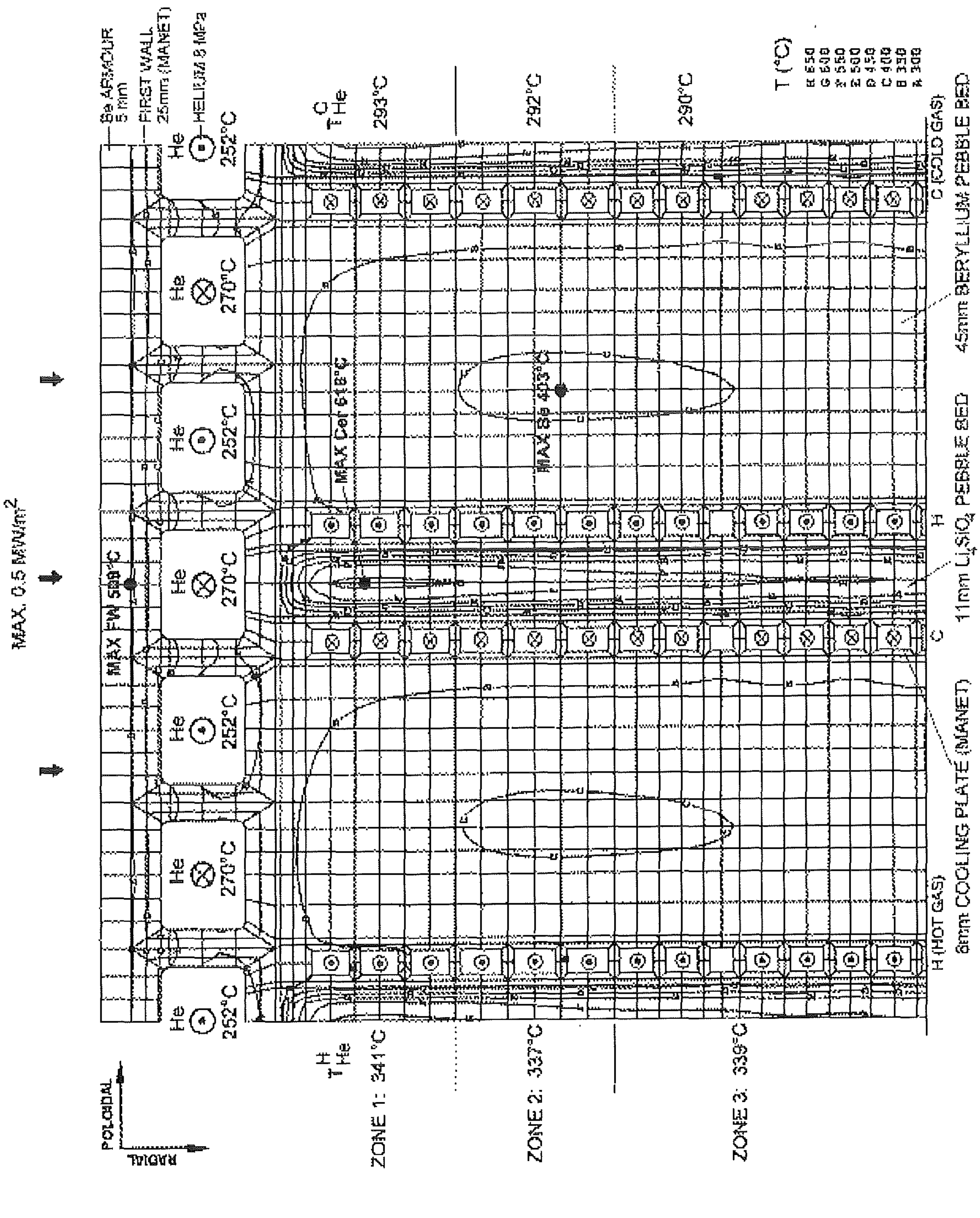


Figure 2.1.1.1.2-1 Temperature distribution in BTM-I

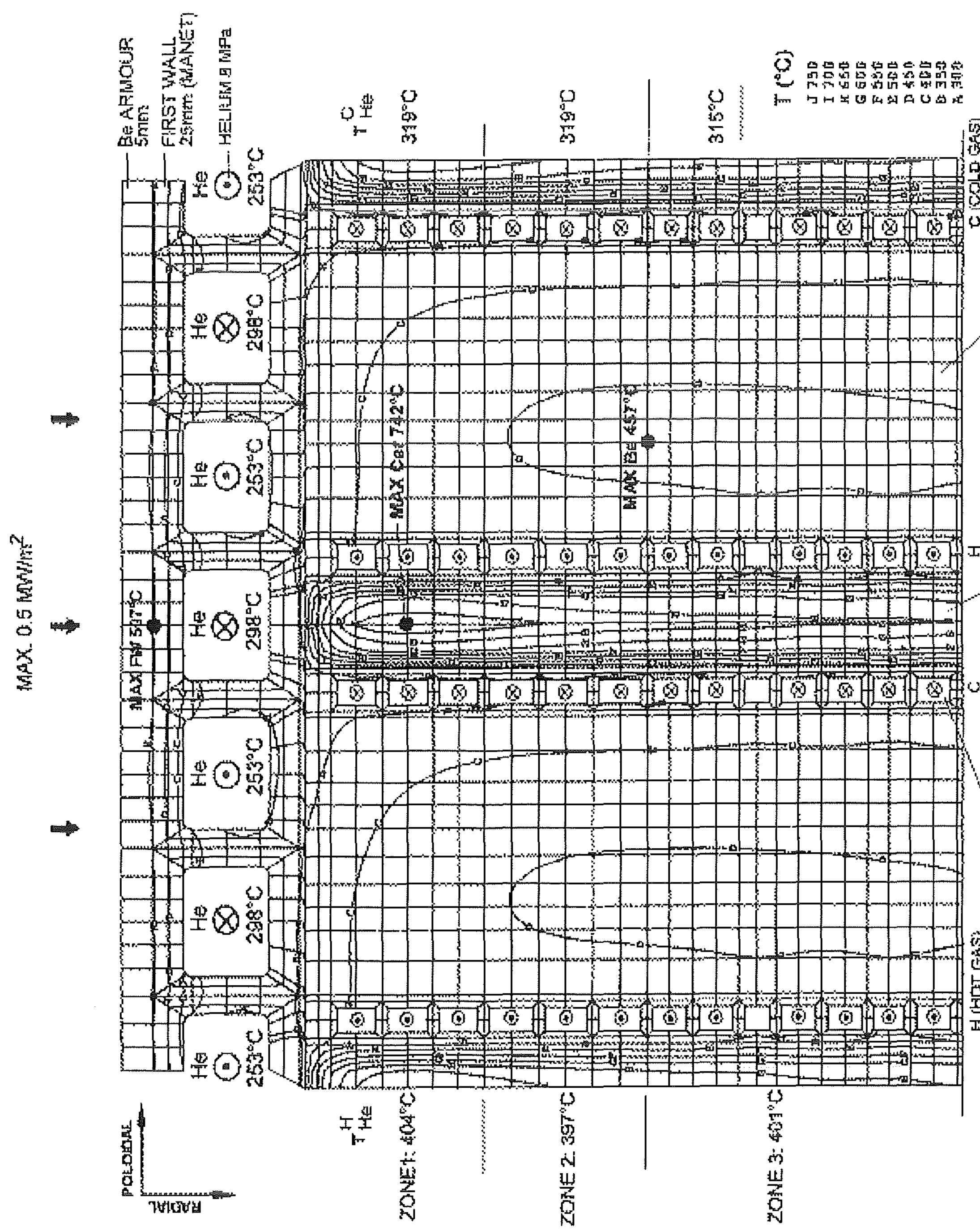


Figure 2.1.1.1.2-2 Temperature distribution in BTM-II

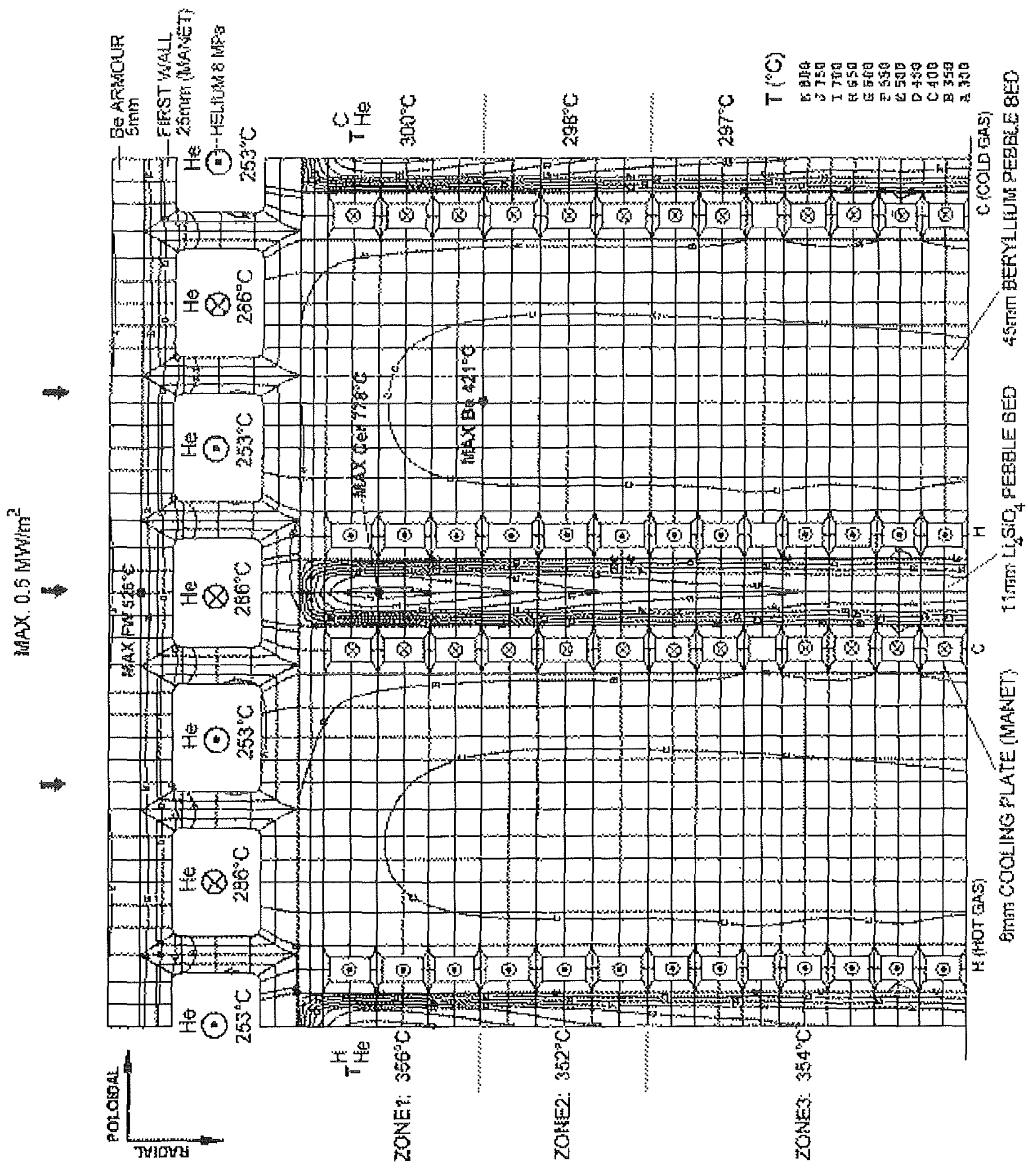


Figure 2.1.1.1.2-3 Temperature distribution in BTM-III

Additionally, three-dimensional calculations have been carried out with the FE code FIDAP [2.1.1.1.2-5] to determine the temperature distribution in the BTM-I during operational transients. In these calculations a section of the blanket box with three FW cooling channels and the neightbouring breeding zone with two cooling plates and two cooling channels each was analysed .The length of the model corresponds to the length of the FW cooling channels, i.e. the FW and the side walls of the box. The alternating flow directions of the two helium systems were taken into account. The input data correspond to the steady-state calculations, with the exception that for the surface heat flux a value of 0.5 MW/m^2 was assumed for the whole plasma-facing surface of the box. Two cases were considered:

a) The cyclic operation of ITER with the following power history (for surface heat flux and internal heat sources): Linear power ramp-up within 50 s; full power burn time 1000 s; linear power ramp-down within 100 s; pulse repetition time 2200 s.

b) An instantaneous power excursion (surface flux and internal heat sources) to 120 % of nominal with a duration of 10 s.

For both cases the inlet temperature was kept constant (250 °C).

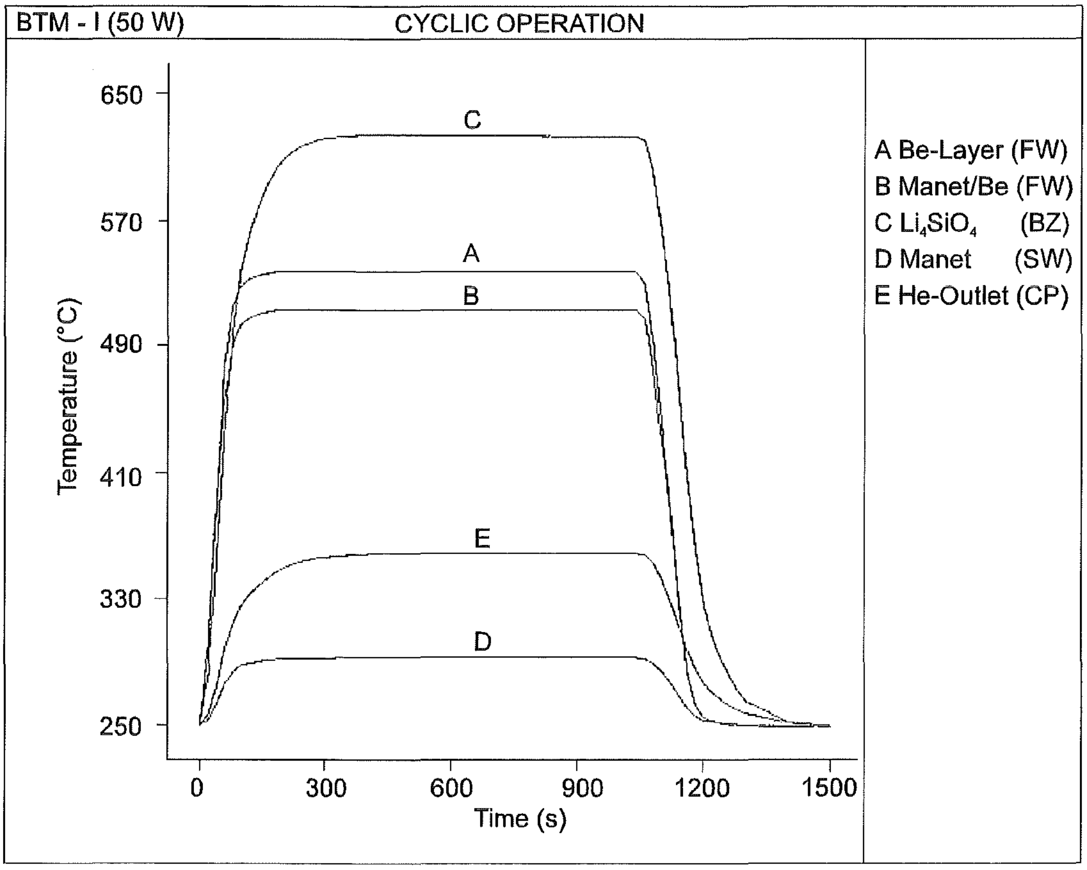


Figure 2.1.1.1.2-4 Blanket temperatures during cycling operation (nominal operation).

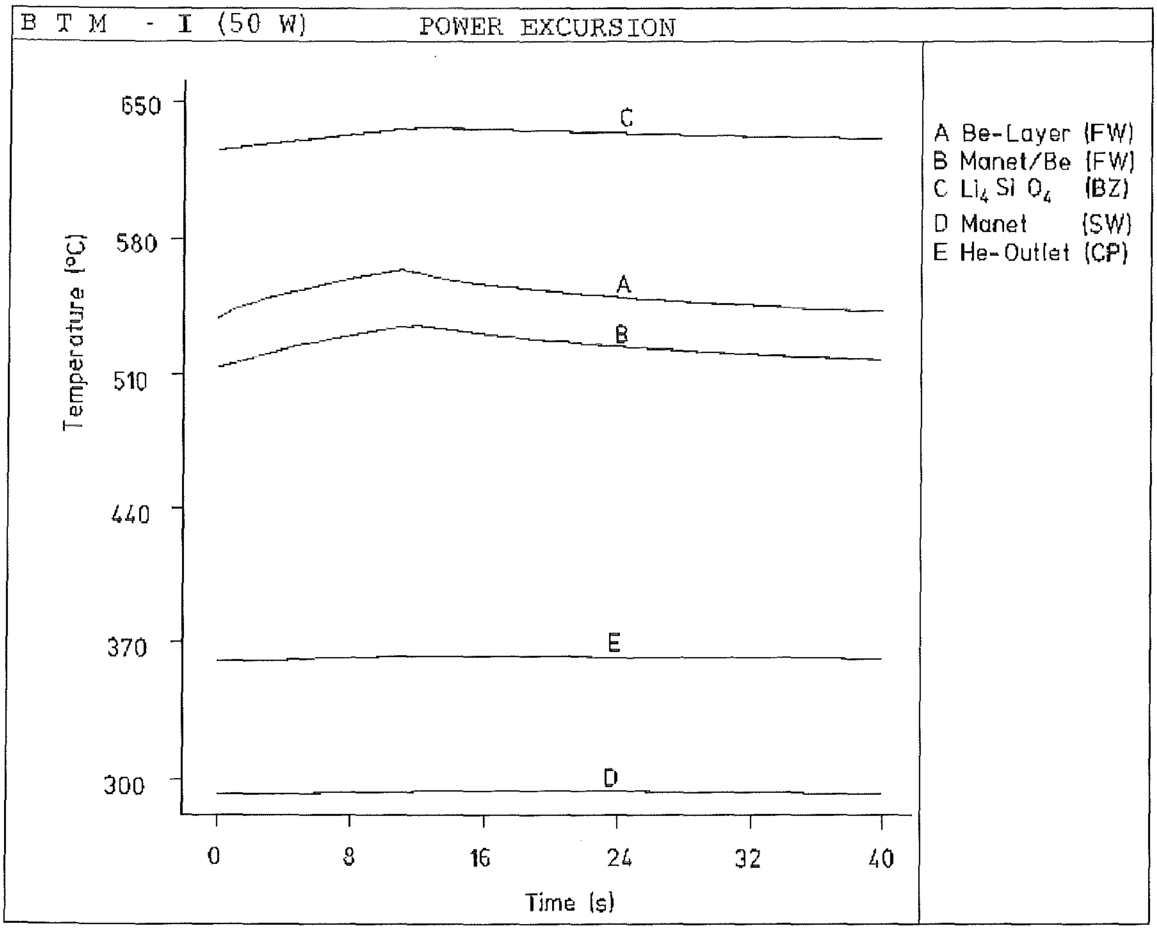


Figure 2.1.1.1.2-5 Blanket temperatures during 20 % overpower excursion operation.

The results of the FIDAP calculations are presented in Fig. 2.1.1.1.2-4 (case a) and Fig. 2.1.1.1.2-5 (case b) which show the course of the maximum beryllium (A) and steel temperature of the FW (B), the steel temperature of the side wall (D), the maximum temperature of the Li_4SiO_4 pebble bed (C), and the helium outlet temperature (E). The main information derived from the figures can be summarized as follows:

-The time constant of the plasma-facing side of the box is rather short, e.g. 25 s after establishing full power the FW has reached 90 % of the temperature rise under steady-state conditions.

-The maximum temperature ramp rate of the breeder material is 4.5 K/s during ramp-up, and 3.0 K/s during ramp-down.

-The power excursion leads to a temperature increase of 22 and 25 K in the FW (steel and beryllium, respectively), and 10 K in the breeder zone. The steel temperature of the side wall is only slightly affected.

To investigate in more detail the time response behaviour to the cyclic power operation, a complementary FIDAP analysis has been carried out for different locations of the breeder zone (BZ). The temperature decay during and following power ramp-down was calculated using a three-dimensional model consisting of two cooling plates, one ceramic pebble bed layer, and one Be pebble bed layer,and using periodic temperature boundary conditions in poloidal direction. The results are presented in Fig. 2.1.1.1.2-6, which shows, for different locations of the BZ, the time needed for the decay of the temperatures to 10 % of the steady-state ΔT (difference between considered temperature and helium temperature at the test module inlet). The behaviour of the ceramic pebble bed is of particular interest: The time delay of 88 s obtained for the BZ near to the FW is comparable to the value obtained for the

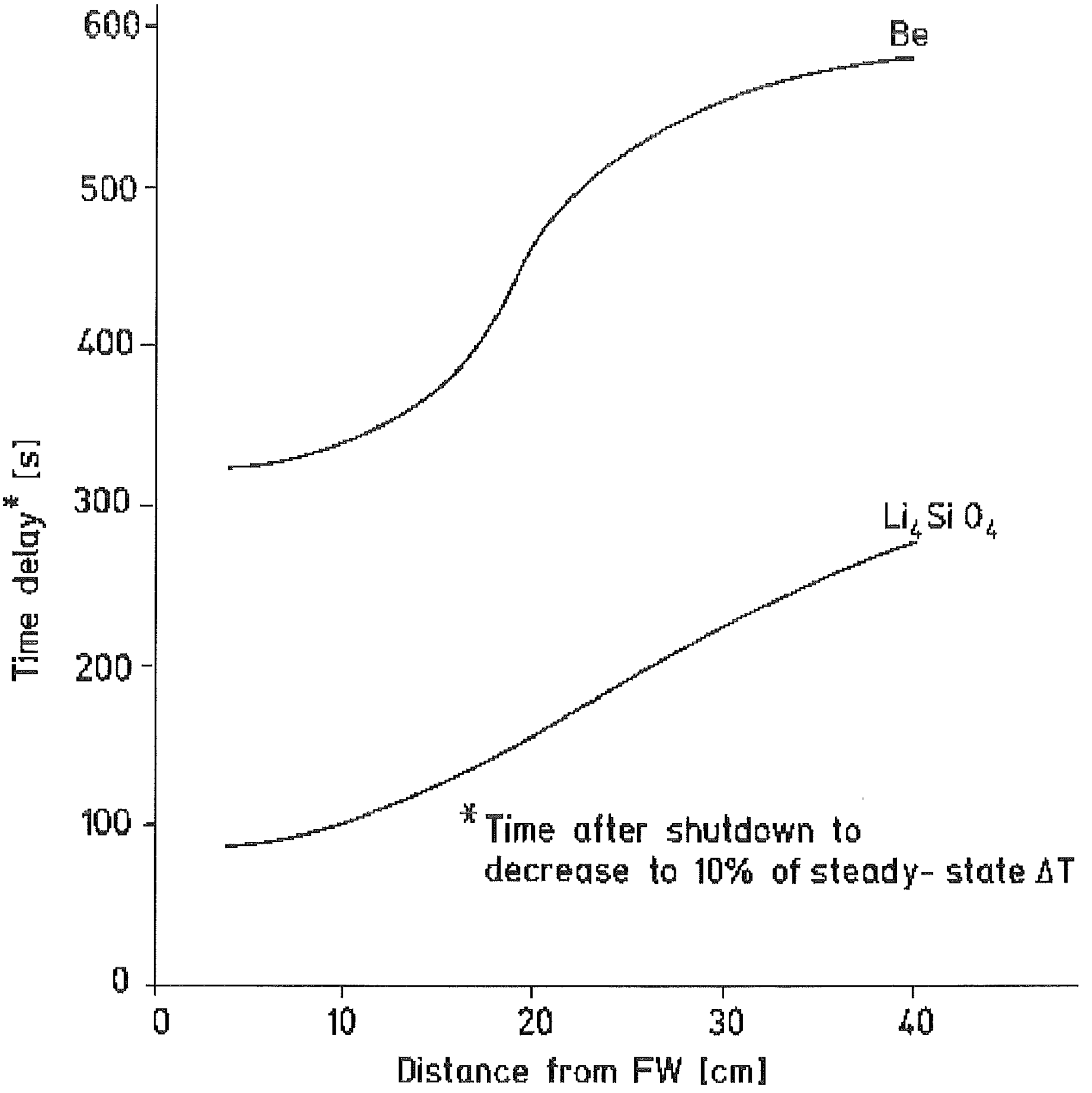


Fig. 2.1.1.1.2-6 Response in time of the breeder zone to cyclic power operation

ramp-up (see above). The time constants increase with the distance from the FW. In the rear part a value of 280 s is attained. In general, it can be stated that the time constants of the BZ are sufficiently low for reaching thermal equilibrium in the BTM within the power cyling times specified for ITER.

References:

[2.1.1.1.2-1] P. Norajitra, Thermohydraulics Design and Thermomechanics Analysis of Two European Breeder Blanket Concepts for DEMO, FZKA 5580 (1995)

[2.1.1.1.2-2] M. Dalle Donne et al., Heat Transfer and Technological Investigation on Mixed Beds of Beryllium and Li_4SiO_4 Pebbles, Proc. ICFRM 6, Stresa, Italy, Sept. 27, -Oct 1 (1993)

[2.1.1.1.2-3] E.U. Schlünder, Particle Heat Transfer, Proc 17th Int. Heat Transfer Conf. Munich (1982), Vol. 1, RK 19, 195

[2.1.1.1.2-4] M. Küchle, Material Data Base for the NET Test Blanket Design Studies, Test Blanke Advisory Group, KfK, Feb. (1990)

[2.1.1.1.2-5] FIDAP 7.0 - FIDAP Users Manual, Fluid Dynamics International (1993)

2.1.1.1.3 Structural Design

Various low activation martensitic steels are being investigated within the European Fusion Technology Program. The choice of the reference structural material will be made by the end of 1998. In the mean time, as in the case of the DEMO blanket design, the martensitic steel MANET has been chosen for the present calculations. The relevant properties of MANET are from Ref. [2.1.1.1.3-1] and [2.1.1.1.3-2].

Two and three-dimensional calculations have been performed with the FE computer code ABAQUS and compared with the ASME and RCC-MR (see Appendix B) codes. The 2D model is a radial/poloidal cut through the FW region of the blanket box and includes 6 FW cooling channels. The results are shown in Table 2.1.1.1.3-1 (BTM -I and II) and Table 2.1.1.1.3-2 (BTM-III). The admissible stresses according to ASME and RCC-MC are likewise included. The comparison of the results in both tables shows that all calculated stresses are below the admissible limits. The distribution of the von Mises stresses (primary and primary plus secondary stresses) in the first wall of the three test modules are shown in Fig. 2.1.1.1.3-1 and Fig. 2.1.1.1.3-2.

In the 3D ABAQUS calculations a full radial/torodial section of the BTM with a poloidal height of 48 mm including two FW channels and two cooling plates was investigated. In a preliminary analysis rather high stresses were obtained for the side wall of the box between the breeder zone and the helium manifold in the case of full helium pressure in the whole blanket box. For this reason two radial/poloidal stiffening ribs were introduced which connect the manifold with the cooling plates. With this modification the stress in the side wall of the box is reduced to 210 MPa. The stress in the stiffening ribs amounts to 233 MPa. Both values are below the admissible stress which is - because of the lower local temperatures - above 300 MPa (see Table 2.1.1.1.3-1(a)). The total stress analysis for the 3D model as well as the analysis for transients have still to be carried out.

Stress analysis of the beryllium plasma facing coating shows that direct use of solid beryllium coating results in the failure of the coating due to fatigue. Thermal stresses can be significantly reduced by cutting a square mesh of slots in the beryllium layer, thus avoiding beryllium failure.

Table 2.1.1.1.3-1 Results of the stress calculations for BTM-I and BTM-II (Basic Performance Phase)

a) Maximum vom Mises primary stresses [MPa]: occurs at the corners of the plasma side of the FW cooling channels (T = 400 °C):

	p = 8 MPa	1.2 p = 9.6 MPa	Admissible limit by ASME (Class 1) and RCC-MR (Class A) for 1000 hours*
Normal operation (pressure only in cooling channels)	56	67	300
Leakage from cooling plates (pressure in the whole blanket box)	131	157	300

b) Maximum von Mises primary plus secondary stresses [MPa]: occurs at the FW interface between FW and plasma facing beryllium layer (T = 509 °C):

	p = 8 MPa	1.2 p = 9.6 MPa	Admissible limit by ASME (Class 1) and RCC-MR (Class A) for 1000 hours*
Normal operation (pressure only in cooling channels)	364	367	458
Leakage from cooling plates (pressure in the whole blanket box)	411	424	458

* During the Basic Performance Phase the operation of one BTM will be less than 1000 full power hours.

Table 2.1.1.1.3-2 Results of the stress calculations for BTM-III (Extended Performance Phase)

a) Maximum von Mises primary stresses [MPa]: occurs at the corners of the plasma side of the FW cooling channels (T = 413 °C)

	p = 8 MPa	1.2 p = 9.6 MPa	Admissible limit by ASME (Class 1) and RCC-MR (Class A)
Normal operation (pressure only in cooling channels)	56	67	290* for 3600 hours
Leakage from cooling plates (pressure in the whole blanket box)	131	157	292** for 1000 hours

b) Maximum von Mises primary plus secondary stresses [MPa]: occurs at the FW interface between FW and plasma facing beryllium layer (T = 526 °C)

	p = 8 MPa	1.2 p = 9.6 MPa	Admissible limit by ASME (Class 1) and RCC-MR (Class A)
Normal operation (pressure only in cooling channels)	370	374	394* for 3600 hours
Leakage from cooling plates (pressure in the whole blanket box)	416	430	430** for ≈ 900 hours

* If operation foreseen for the whole EPP (3 Mwa/m^2≈20000 h), the first wall should operate at a lower temperature (higher helium cooling mass flow)

** Within 900 full power hours operation is very likely possible to change the leaking module during a planned plasma shut-down.

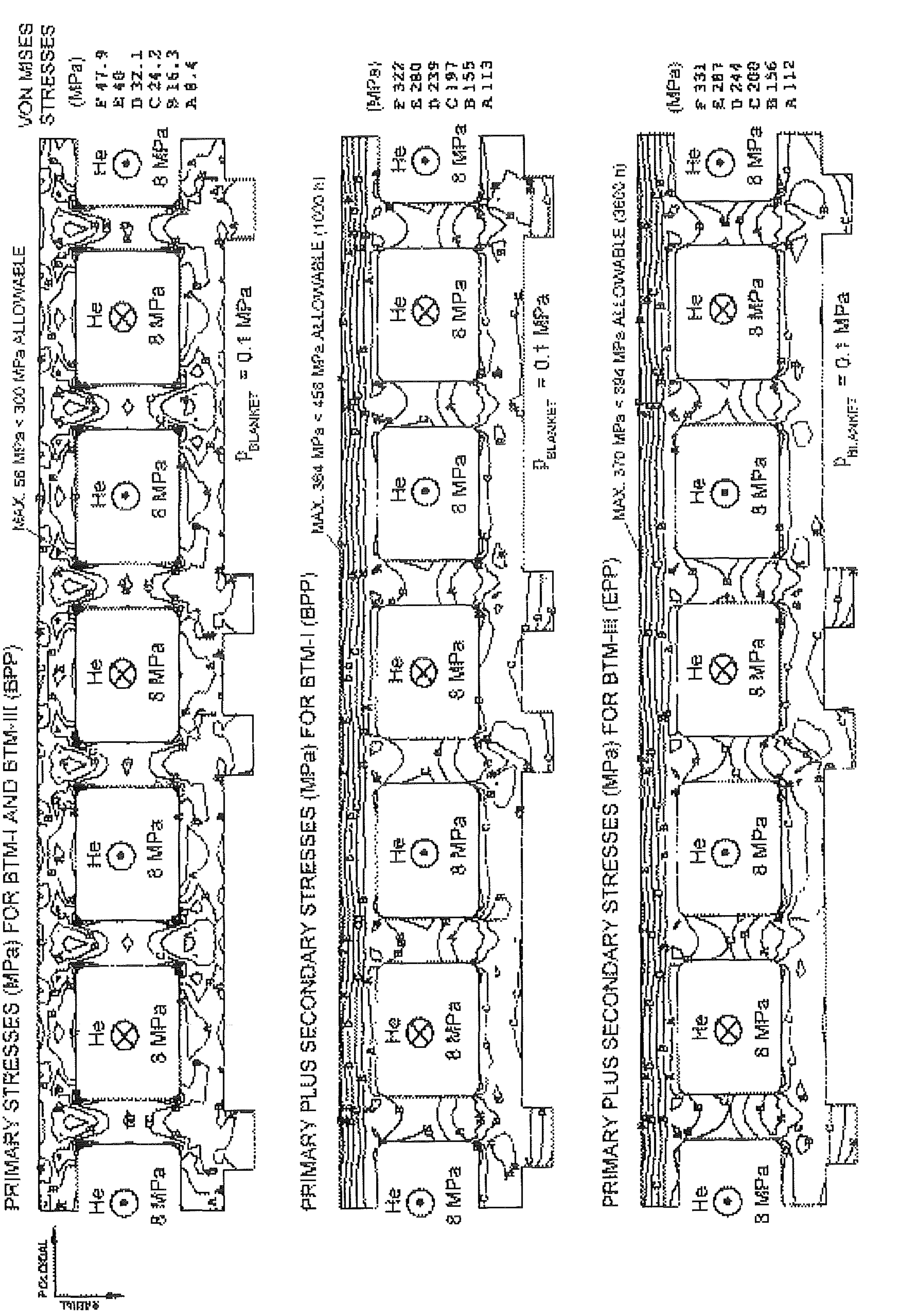


Fig. 2.1.1.1.3-1 Von Mises stress distribution (MPa) in the First Wall in a normal operation (pHe=8 MPa; pBlanket=0.1 MPa).

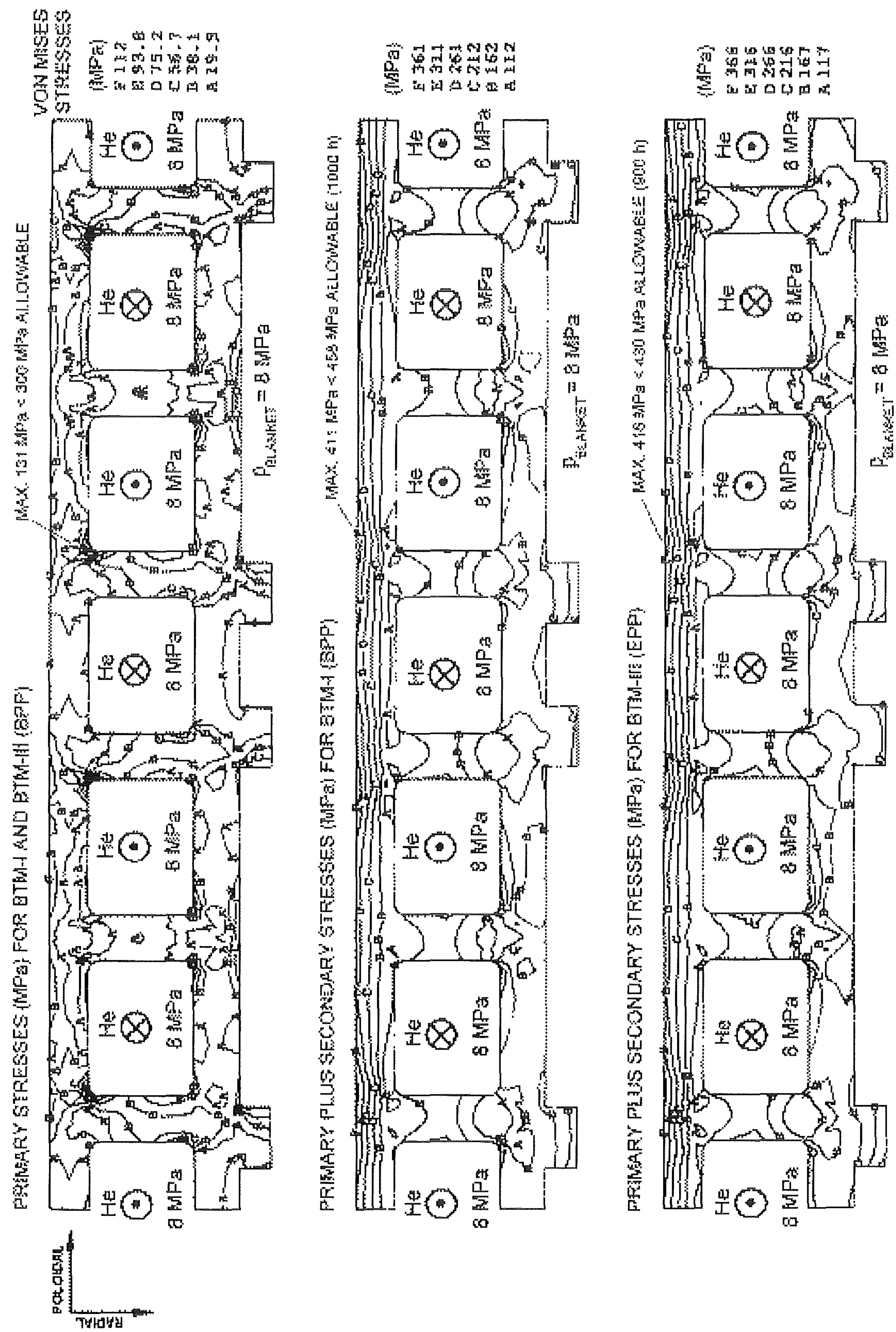


Fig. 2.1.1.1.3-2 Von Mises stress distribution (MPa) in the First Wall in an accidental case (pHe=8 MPa; pBlanket=8 MPa).

Electromagnetic Forces

A preliminary electromagnetic (EM) calculation has been performed to evaluate the magnetic forces acting on the various components of the Test Blanket Subsystem (see Appendix A). The results constitute a basis for the structural analysis of the connection between BTM and Shield, and between the whole Test Blanket Subsystem and the ITER backplate. The results form partially also a basis for the structural analisys of the ITER backplate for load conditions caused by the presence of the Test Blanket Subsystem. These values are only partial because the increase of the EM load acting on the backplate, caused by the presence of the test blanket, is not considered in the presented calculations. The increase is due to the distortion of the eddy current patterns on the backplate.

The analysis has been performed with the code AENEAS [2.1.1.1.3-3]. This 3D finite element program, developed at the Research Center of Karlsruhe can calculate eddy currents and magnetic forces in presence of ferromagnetic materials like MANET.

At the present only the effects of a centered disruption of 10 ms have been calculated. The results, resultant forces and torques, are summarized in Table 2.1.1.1.3-3.

Table 2.1.1.1.3-3: Resultant forces and torques caused by a centered disruption.

	Lower BTM	Upper BTM	Frame and Shield	Total
x-coordinate (*) [m]	11.768	11.768	12.291	11.658
y-coordinate (*) [m]	0.000	0.000	0.000	0.000
z-coordinate (*) [m]	0.823	1.887	1.355	1.355
Force x [MN]	-0.035	0.029	-1.975	-1.981
Force y [Mn]	0.200	-0.085	0.229	0.344
Force z [Mn]	0.693	-0.201	1.108	1.600
Torque x [MNm]	2.007	2.010	-0.243	3.927
Torque y [MNm]	-0.104	0.377	-0.409	-1.085
Torque z [MNm]	-0.961	1.164	-1.742	-1.869

(*) According to the torus Coordinate System

References:

[2.1.1.1.3-1] M. Küchle, Material Data Base for the NET Test Blanket Design Studies, Test Blanke Advisory Group, KfK, Feb. (1990)

[2.1.1.1.3-2] K. Ehrlich, Internal KfK Report, May (1986)

[2.1.1.1.3-3] P. Ruatto, „Entwicklung einer Methode zur Berechnung der elektromagnetischen Kräfte durch Magnetfeldänderungen in ferromagnetischen

Strukturen und Anwendung dieser Methode auf den Plasmaabbruch in einem Tokamakreaktor", FZKA 5683, Forschungszentrum Karlsruhe (1996).

2.1.1.1.4 Mechanical Design

The test blanket module is installed radially into the horizontal test port to replace a portion of the shielding blanket. Thus, it must act similarly to the shielding blanket and not adversely impact the operation of the shielding blanket.

The clear opening through the shielding blanket backplate and the blanket modules is 1.600 m wide and 2.600 m high, centered on the horizontal port. Figure 2.1.1.1.4-1 illustrates the available space in the backplate with the shear teeth shown. They are conformal with the backplate so that the backplate will have sufficient clearance for rotation during installation.

There is a gap allowance of 20 mm completely around the perimeter of the test blanket. The first wall of the test module is planar, without curvature, but it is conform as closely as possible to the first wall of the adjacent shield blankets modules (Fig. 2.1.1.1.4-2 and 2.1.1.1.4-3). The deviation from the primary first wall is in the range of 50-65 mm. The thermal conditions on the side wall of the test blanket module need not be identical to the wall conditions of the facing wall of the shielding blanket due to the presence of the water cooled frame in between.

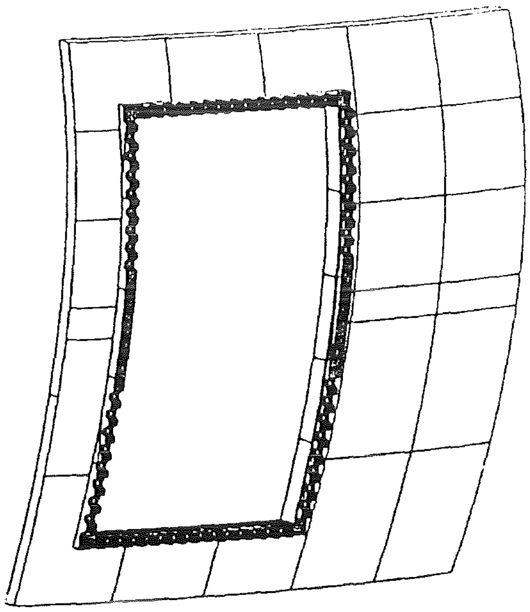
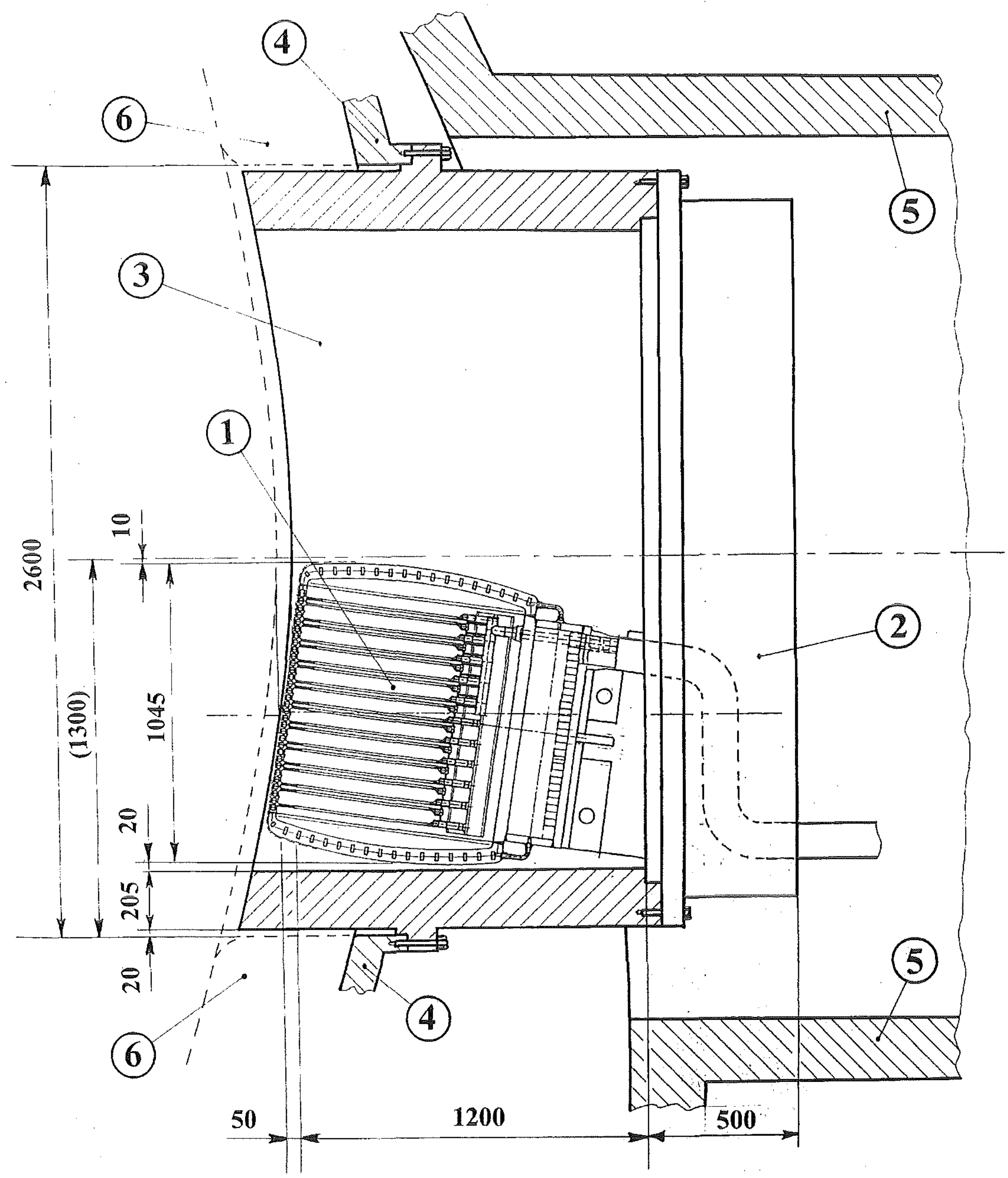


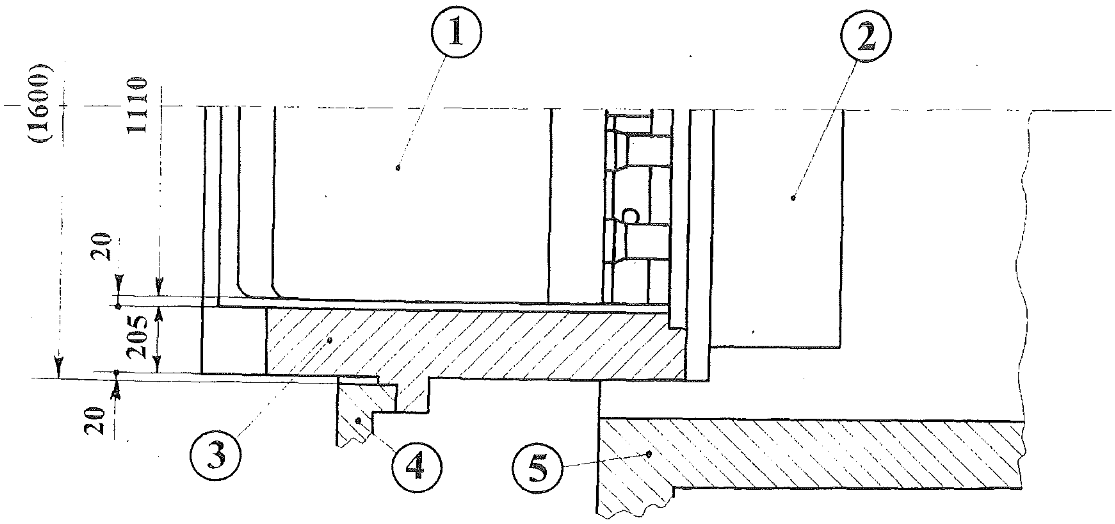
Figure.2.1.1.1.4-1: Opening in the Backplate

The test blanket module is located and supported by the shielding blanket backplate. The opening in the backplate contains a toothed-frame provided by ITER, which is illustrated in Figure 2.1.1.1.4-1. There is a mating frame around the test blanket as shown in Figure 2.1.1.1.4-4. The test module frame dimensions adjacent to the regular shielding blanket modules are 1.560 m wide by 2.560 m high. This frame has a series of teeth to react the static and dynamic loads induced in the test module. The teeth have bolts to engage and secure the toothed-frame system and react the normal loading conditions. The frame structure will be integrated into the blanket module to react all the internal loads. The shielding blanket backplate will transmit the loads through a predetermined load path to the vacuum vessel. The connection between the teeth is to provide an electrical current path for currents generated in the module. Although the nominal temperature of some of the test blankets may be in the neighborhood of 300°C to 500°C, the mounting frame of the test blanket must be cooled to approximately 150°C to prevent high thermal stresses in the mounting flange.



1. Blanket Test Module
2. Shield
3. Support Frame
4. Back Plate
5. Vacuum Vessel
6. Shield Blanket

Figure.2.1.1.1.4-2 Vertical cross section of the Support Frame with the European HCPB-BTM



1. Blanket Test Module
2. Shield
3. Support Frame
4. Back Plate
5. Vacuum Vessel

Figure.2.1.1.1.4-3: Horizontal cross section of the Support Frame with the European HCPB-BTM

Because the breeding blankets are more neutron transparent, additional shielding is needed behind and to the side of the breeding module to protect the vacuum vessel. This will be provided by the water cooled Support Frame. The wall thickness of the Support Frame has been chosen so that it allows a suitable BTM size and at the same time a good neutron shielding capability (see Section 2.1.1.1.1).

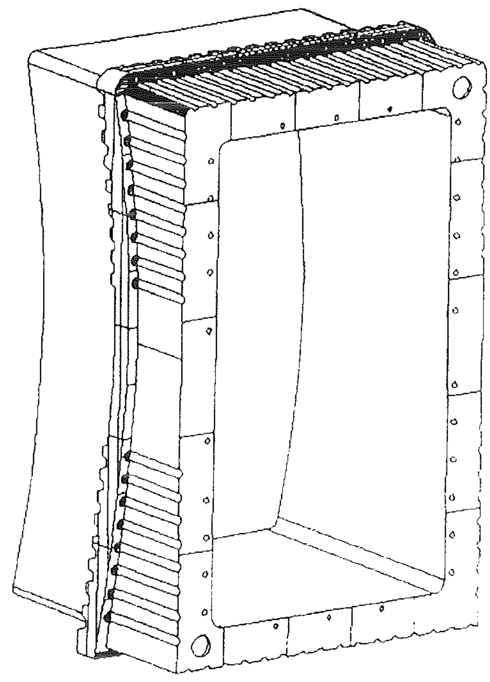


Figure 2.1.1.1.4-4. Support Frame

The BTM's are being designed to withstand a combination of coolant pressure loads, thermal loads from the plasma, electromagnetic loads, thermal stresses due to temperature gradients inside the BTM and especially at interfaces of parts cooled by the water.

Basic parameters of the frame cooling are similar to ones of the ITER Shielding Blanket Modules.

The European and the Japanese have collaborated in their approach for testing their helium cooled solid breeder test modules. During the ITER Basic Performance Phase (BPP) the European Blanket Test Module (BTM) shall occupy half of the test port allocated to the helium cooled blankets, the other half being occupied by the Japanese helium cooled ceramic breeder blanket module. The Tritium Extraction Subsystems for the two BTM's will be separate and placed in the pit immediately adjacent to the test port. The helium coolant loops (heat transfer, heat transport and helium purification) will also be separate and will be placed outside the pit, probably in the Tritium Building.

To facilitate handling operations the two BTM's (European and Japanese) together with their shields shall be bolted to the water cooled support frame. This frame is bolted to the shield blankets back plate and it is designed to transmit the loads generated in the BTM's to the shield blanket back plate. Fig. 2.1.1.1.4-2 shows a vertical cross section of the support frame with the European HCPB-BTM attached to its lower part, the upper part being reserved for the Japanese helium cooled BTM.

The HCPB-BTM (see section 2.4.2.2 for Reference BTM drawings) represents a poloidal portion of the HCPB DEMO blanket. As in the DEMO the radial toroidal plates and the first wall are cooled by helium at 8 MPa flowing first in the first wall and then in the blanket plates. For safety reasons the coolant helium flows in two completely separated loops. In the blanket the coolant helium is flowing alternatively in opposite directions in the first wall and in the adjacent blanket plates. In this way the BTM temperatures are more uniform. In the reference BTM between the blanket plates there are, as in the DEMO blanket design, alternatively 11 mm thick ceramic breeder pebble layers and 45 mm thick beryllium pebble layers. The tritium purging gas is helium at about 0.1 MPa flowing in radial direction from the first wall to the back of the module. The plasma side of the first wall is protected by a 5 mm beryllium layer and it recessed from the shield blanket countor by a minimum amount of 50 mm to a maximum of circa 65 mm. At the upper and lower ends the HCPB-BTM is closed by covers capable to sustain a pressure of 8 MPa. During normal operation the space in the BTM other than the cooling plates and the FW is at the purge gas pressure of 0.1 MPa. However, in case of a leak from a cooling plate, it can be pressurized up to 8 MPa. Thus the blanket box, and the helium purge system connected to it, have been designed to sustain the full 8 MPa pressure. This is a double barrier against helium leakage from the cooling plates and would allow, in case of need, to wait for the next planned period for the exchange and repair of the module.

For the mechanical connection between the BTM and the shield a groove-and-tongue design is proposed which holds the BTM in the poloidal and toroidal direction (Fig. 2.1.1.1.4-5). Fixation in the radial direction is assured by four bolts with slightly slotted holes. This design allows the transmission of loads and torques in all directions and planes - in particular the large torque vector in the radial direction during disruptions - , and the accomodation of the differential expansions resulting from the different materials and temperatures in the BTM and the shield.

The heat generated inside the supporting structure is transmitted by heat conduction to the cooled regions of the BTM and the shield, respectively, without excessive temperature rises. Heat transfer across the contact areas of the support is not needed for heat removal.

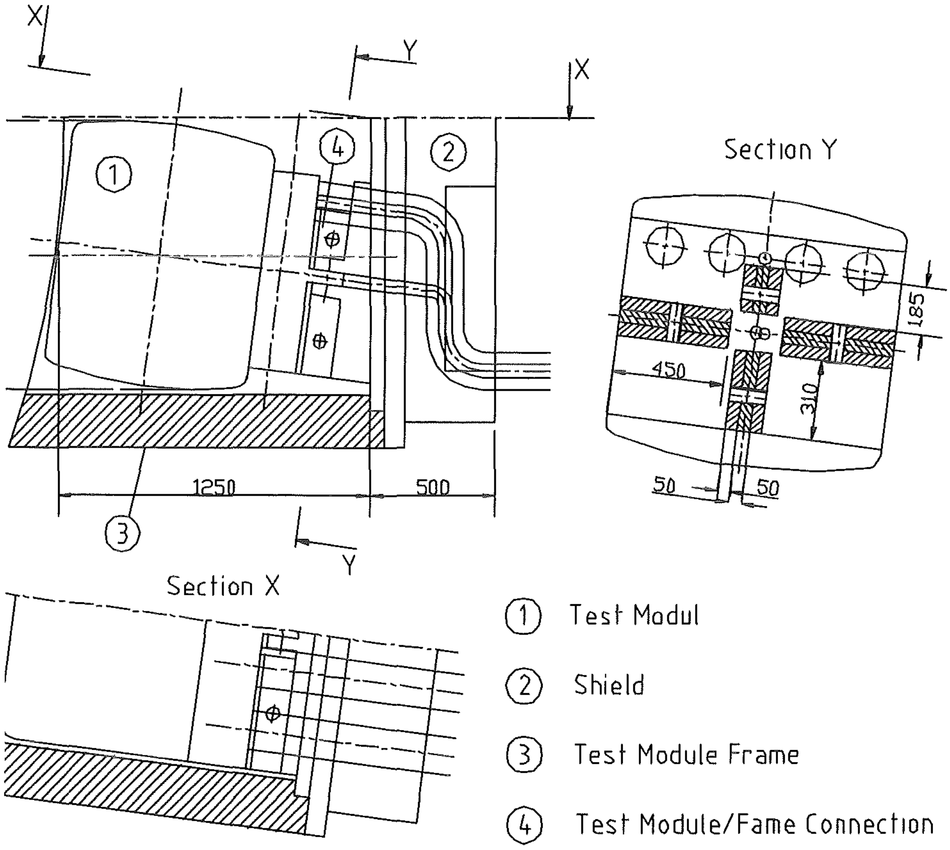


Figure 2.1.1.1.4-5. HCPB-BTM with mechanical connection to the BTM Shield

2.1.1.1.5 Maintenance/Remote Handling

The BTMs are designed to be removable through the horizontal test port by use of remote handling equipment. The BTMs size is limited to the port dimensions. Since the weight of full BTM is higher than that of the regular Shielding Blanket Module, special attention should be paid to remote handling equipment for BTMs. The mass distribution of the current HCPB-BTM design is as follows:

HCPB-BTM	3.0 t
Support Frame (without coolant)	12.5 t
Shield (without coolant)	10.7 t
Plumbing	1.0 t
Primary Closure Plate	11 t
Total (without Japanese BMT)	38.2 t

Replacement frequency is estimated as several times (TBD) during BPP. Maintenance procedure description is:

- duration is no more than 8 weeks;
- drainage of water and helium depressurization;
- removal of cryostat plug to the maintenance cell;
- further steps will depend on the decision of JCT and TBWG on general approach for design of BTMs, all the interfaces and piping, handling equipment for blanket test modules .

2.1.1.1.6 Assembly

The size and configuration of the test blanket assembly is dependent upon the decision on penetrations through the side wall of the vacuum vessel extension or the vacuum vessel closure plate. It will also depend upon the decision of the JCT and the TBWG on the general approach of design of the BTMs, all the interfaces, piping, and handling equipment for the blanket test modules. In any case, the blanket module will be a self-contained module encased in a frame. This is shown in Figure.2.1.1.1.4-2. The blanket modules, the frame, and all the associated piping will be assembled and fully tested in a remote location. Then the complete unit (with or without the vacuum vessel door) will be brought to the test port with a transporter. Rails will align and position the assembly within the vacuum vessel. The unit will then be installed on the flange and the assembly bolts tightened. Then the piping will be welded and inspected. Then the vacuum vessel closure plate will be closed, followed by the cryostat closure plate and the bioshield plug.

2.1.1.1.7 Materials

According to the available information the Blanket Test Module Design is based on the use of the following materials:

- The structural material used in the breeding blanket will be a low activation martensitic steel. The choice of the type will be made within the European Fusion Technology Program by the end of 1998. For the present design the martensitic steel MANET has been chosen. This is a fully martensitic steel containing 10-11% Cr and additions of approximately 0.6% Mo, 0.65% Ni, 0.25% V and 0.15% Nb.
- The reference ceramic breeder is overstoichiometric lithium orthosilicate (Li_4SiO_4) with a small amount of TeO_2. Alternative material is Li_2ZrO_3 or Li_2TiO_3
- Beryllium is used in form of pebbles as multiplier and in form of a plasma facing layer protecting the first wall.

In the following the different materials for each component are listed with the estimated weight:

HCPB-BTM		
structural material	MANET	2500 Kg
breeder material	bed of 0.25-0.63 mm of reference ceramic breeder	100 Kg
multiplier	binary bed of 1.5 to 2.3 and 0.1 to 0.2 mm beryllium pebbles	400 Kg
plasma facing material	5 mm beryllium layer	1 Kg
coolant	helium (8 MPa, 250°-450°C)	neglegible
Support Frame		
structural material	316LN-IG	11900 Kg
heat sink	copper alloy	205 Kg
plasma facing material	10 mm beryllium layer	28 Kg
coolant	water	
Shield		
structural material	MANET	10700 Kg
coolant	water	

Plumbing	helium coolant / water coolant/ diagnostics	
structural material	MANET / 316LN-IG / 316LN-IG	TBD Kg
coolant	helium / water / none	TBD Kg
VV Closure Plate	TBD	TBD

Waste characterization of all materials is being done. The tritium inventory has been estimated at 0.2 g (not included tritium in plasma facing beryllium) mainly contained in beryllium.

2.1.1.1.8 Safety

The safety considerations of the blanket test module focus on the accidental safety aspects to the extent that conceivable failures of the BTM system can impede the safe operation of ITER. On the other side, occupational safety and waste generation issues have not been elaborated so far, since detailed activation data are not yet available. In addition, the favourable features of the gas-cooled concept and the small size of the test module relative to the ITER machine don't seem to raise extra safety problems in the latter respects.

The accidental safety concerns are addressed in the following way. At first an overview is given on the material mass inventories and on tritium inventories in the different subsystems of the HCPB-BTM system. Next the energy sources are compiled which are the driving elements in any accident sequence. Following the identification and categorisation of enveloping events to serve as design base for the BTM, these design events are analysed with view to their short and long term evolution and consequences thereof. Finally, the events are evaluated against a set of safety requirements set out in section 1.3.5 to demonstrate that the HCPB-BTM design complies with ITER safety criteria. Please note that the analysis is still incomplete.

2.1.1.1.8.1 Materials and Toxic Materials Inventory

To give an overview of the types of materials and masses involved, and their coarse distribution in the BTM system the inventories are summarised in this section. It is distinguished between material masses in technical terms (e.g., structural material, breeder, multiplier, coolant) and two radio-toxic categories, i.e., tritium and activation products (AP). Only those inventories are considered which may influence in some way the sequence of an accident, for instance, via chemical reaction, mechanical energy release, thermal inertia, or liberation of radioactivity.

In Table 2.1.1.1.8-1 are given the mass inventories in the HCPB-BTM subsystems relevant in accident analysis. The test module proper (excluding the water cooled shield plug) includes 2700 kg of structural material, 450 kg of beryllium pebbles, and 120 kg of breeder pebbles. The cooling subsystem (2 loops plus components) contains an additional steel mass of 15000 kg, constituting a relatively large heat capacity. Note that also the water-cooled shield plug has a large steel mass of 11000 kg (section 2.1.1.1.7). The helium enclosed in the main loops amounts to 43 kg under operating conditions (8 MPa, 300 °C average temperature).

The tritium inventory in the BTM has not yet been evaluated in detail. A preliminary assessment for the structure and breeder material is made by scaling down the inventories obtained for the outboard of the DEMO blanket after 2.5 years of full power operation, corresponding to a total neutron fluence at the outboard first wall at torus midplane of about $10^{23}/cm^2$. A scaling factor of 1/480 (one tenth of one out of 48 outboard segments) is used, resulting in the tritium inventories given in Table 2.1.1.1.8-2. As can be seen, the tritium content in the BTM is in the 20 mg range in structural and breeder materials each. For the beryllium multiplier a tritium inventory at the end of BPP and EPP of 0.14 g and 4 g, respectively, has been estimated.

The primary coolant contains only 1 mg (10 Ci) of tritium when the purification process is in equilibrium, otherwise even less, provided that the purification system is operating while the ITER power is on. The cold traps of the helium purification subsystem will accumulate up to 3 mole of Q_2O (≈3 g of tritium) within six days (section 2.1.1.4). After this period they will be regenerated.

Table 2.1.1.1.8-1: Mass Inventory in the BTM System Relevant in Accident Analysis

Subsystem and type of material	Inventory (kg)
Inventories in the BTM proper	
Structural material (MANET)	
First wall (FW)	120
Breeding zone (BZ)	1100
Manifold zone (MZ)	1500
Shield Zone (SH)	11000
Plasma facing material (beryllium, 5 mm)	10
Breeder (Li_4SiO_4) in BZ	120
Multiplier (beryllium pebbles) in BZ	450
Coolant (helium, operating conditions)	1.2
Purge gas (helium, operating conditions)	0.01
Inventories in cooling subsystem (2 main loops)	
Structural material (type 316L)	14800
Coolant (helium, operating conditions)	43
Inventories in helium purification subsystem	
Structural material	tbd
Helium	tbd
Inventories in tritium extraction subsystem	
Structural material	tbd
Helium (purge gas, main loop only)	1.1

The tritium inventory in the plasma facing beryllium layer has been calculated for ITER (here assumed three continuous full power years) as 0.088 g tritium per kg of beryllium. Assuming zero release, this would yield 0.88 g of tritium in the 10 kg beryllium contained in the BTM plasma facing layer. Since the first wall beryllium will not last for three full power years the tritium content will only be a proportion of that amount, e.g., ~0.09 g at the end of the BPP with 0.3 full power years.

Activation products inventories in the BTM materials (breeder, multiplier, and structural material, and in the primary cooling loops) are tbd.

Table 2.1.1.1.8-2: Estimates of Tritium Inventory in BTM subsystems (see text for neutron fluence or operating time)

Tritium containing material	Tritium Inventory (g)
BTM including primary coolant	
Structural material	
First wall	8×10^{-3}
Breeding zone	10×10^{-3}
Manifold zone	0.6×10^{-6}
Plasma facing material (Be, 5 mm)	0.88
Breeder material (Li_4SiO_4)	15×10^{-3}
Multiplier (beryllium pebbles) BPP / EPP	0.14 / 4
Primary coolant (helium, 2 loops)	1×10^{-3}
Helium purification subsystem (in cold traps maximum)	3
Tritium extraction subsystem	tbd

2.1.1.1.8.2 Energy Sources

Energy sources in upset or accidental conditions are seen in (a) plasma disruptions, (b) delayed plasma shutdown after a cooling disturbance, (c) afterheat, (d) work potential of pressurised coolants, and (e) exothermic chemical reaction. This section summarises the energy sources for the HCPB-BTM based on the inventories described in section 2.1.1.1.8.1. An overview of the energy quantities (a) to (e) is given in Table 2.1.1.1.8-3. The values are explained in the paragraphs to follow.

For short and severe disruptions due to alpha-particle induced instabilities a maximum energy up to 0.3 MJ/m^2 (GDRD section 2.2.6.10) is assumed corresponding to an energy of 0.35 MJ to be applied to the BTM first wall.

The energy due to delayed plasma shutdown is the time integral of surface power and internal power for a given shutdown scenario. For a normal shutdown sequence (1 s delay , 100 s ramp-down, see Table 2.1.1.1.8-4) this energy deposited in the BTM is 97 MJ. Any second in shutdown delay would add 1.9 MJ. The accelerated and the fast shutdown scenarios defined in Table 2.1.1.1.8-4 yields an energy of 21 and 4.8 MJ, respectively.

The afterheat power generation needs tbd in section 2.1.1.1.1. As a first approximation it has been deduced for safety analyses from results obtained for the DEMO blanket by applying a scaling factor to the radial distribution of the afterheat power density at mid-plane of the DEMO outboard blanket. This scaling factor is taken as the ratio of the power density at normal operation in the BTM to that in the DEMO outboard which turned out to be about 1:4. Thus, the spatially integrated afterheat power for the whole test module after 1 minute, 1 hour, 1 day, 1 month yields values of 23.6, 16.5, 1.8, and 1.5 kW, respectively, leading to the time integrated energy as displayed in Table 2.1.1.1.8-3.

The main helium cooling subsystem contains about 43 kg of helium at 8 MPa and a mean temperature of 300 °C. The work potential relative to ambient conditions amounts to 63 MJ.

Table 2.1.1.1.8-3: Energy sources in the HCPB-BTM

Energy source	Energy (MJ)
a) Plasma disruptions	0.35
b) Delayed plasma shutdown	
normal: 1 s delay, 100 s ramp-down	97
accelerated: 1 s delay, 20 s ramp-down	21
fast: 1 s delay, 20 s ramp-down for surface heat (tab. 2.1.1.1.8-4)	4.8
c) Afterheat integrated over:	
1 minute	1.5
1 hour	69
1 day	445
1 month	4510
d) Work potential of helium coolant (both loops)	63
e) Chemical energy	
beryllium/water reaction	18000
beryllium/air reaction	30000

The exothermic reaction per kg of beryllium with water or oxygen generates 40 MJ or 67.4 MJ, respectively, resulting in a total chemical energy potential of the beryllium pebble beds (450 kg) of 18000 MJ for a beryllium/water reaction and 30000 MJ for a beryllium/air reaction. This is vast compared to the other energies and must not be neglected in the accident analysis should there be a chemical reaction involved.

2.1.1.1.8.3 Identification of Events for Design Basis

Events for design basis of the HCPB blanket test module have been identified in Table 2.1.1.1.8-5. They are postulated enveloping events representing the most demanding design requirements to the BTM which will require shutdown of ITER. A category has been assigned to each event according to the classification scheme given in GDRD Table 4.1.1-1. The event sequence categories correspond to typical annual expected frequency ranges, f, as follows: category I: f>1/a (operational), II: $f>10^{-2}/a$, III: $10^{-2}/a > f > 10^{-4}/a$, IV: $10^{-4}/a > f > 10^{-6}/a$, V: $f<10^{-6}/a$ (hypothetical).

Depending on the severeness of the event, different plasma shutdown requirements are postulated for accident analysis as define in Table 2.1.1.1.8-4, i.e., normal, accelerated, and fast shutdown. Following the accident, a delay time is assumed for the safety system to initiate plasma shutdown which then occurs as a linear transition from full power to zero power within the ramp-down time specified.

The events listed in Table 2.1.1.1.8-5 pertain to three families: (a) the loss of coolant family with leaks in one or both loops at different locations, (b) the loss of flow family with the loss of forced convection flow in one or both loops, and (c) the loss of heat sink family postulating complete failure of the secondary heat removal system for one or both main loops.

Table 2.1.1.1.8-4: Postulated Plasma Shutdown Scenarios for Accident Analysis

Shutdown Scenario	Delay time (s)	Ramp-down time from full power to zero (s)	
		of internal heat	of surface heat
Normal	1	100	100
Accelerated	1	20	20
Fast	1	0	20

Table 2.1.1.1.8-5: Characterisation of Events for HCPB-BTM Design Basis

Name and Category	Enveloping Event	Event Characterisation
LOCA1EX III	LOCA ex-vessel in one cooling loop	Loss of coolant due to spontaneous guillotine break of cold leg at circulator outlet, normal plasma shutdown, pressurisation and contamination of affected compartment in tritium building, afterheat removal via intact loop.
LOCA1IN III	LOCA in-vessel in one cooling loop	Loss of coolant due to guillotine break in feeders inside VV, fast shutdown (inherently), pressurisation of VV, integrity of VV confinement boundary maintained, afterheat removal via intact loop.
LOCA2EX V	LOCA ex-vessel in both cooling loops	Loss of coolant, e.g., due to earthquake-induced guillotine break of hot legs of both loops, accelerated shutdown, pressurisation and contamination of affected compartment in tritium building, integrity of VV confinement boundary maintained, afterheat removal by forced circulation of air via broken loops with ultimate heat sink provided by ventilation system and supported by purge gas system.
LOCA2IN III	LOCA in-vessel in both cooling loops	Loss of coolant due to overpower, disruption forces, or material degradation induced spontaneous rupture of FW involving both cooling loops, fast shutdown, pressurisation of VV, integrity of VV confinement boundary maintained, afterheat removal via low-pressure circulation of VV atmosphere through broken loops supported by purge gas heat removal and free convection at module surface, ultimate heat sink is cooling system of ITER in-vessel components.
LOCA1HX II	LOCA in one heat exchanger	Loss of coolant in one heat exchanger from primary to secondary loop, normal plasma shutdown, pressurisation of secondary cooling subsystem and discharge of primary helium into compartment housing the secondary cooling subsystem, afterheat removal via intact loop.

Table 2.1.1.1.8-5: Characterisation of Events for HCPB-BTM Design Basis

Name and Category	Enveloping Event	Event Characterisation
LEAK1TM III	Leak inside test module of one cooling loop	Leak inside test module with subsequent pressurisation of BTM box and purge gas system, normal plasma shutdown to check for continuation of operation, pressure boundary (affected loop plus tritium removal subsystem) maintained, full heat removal capability retained, ITER operation not affected until planned shutdown.
LEAK2TM IV	Leak inside test module of both loops	Leak inside test module with subsequent pressurisation of BTM box and purge gas system, normal plasma shutdown, pressure boundary (affected loops plus tritium removal subsystems) maintained, afterheat removal via 1 or 2 loops at reduced system pressure.
LOFA1 I or II tbd	LOFA in one cooling loop	Loss of flow due to pump coast down or inadvertent valve closure, normal plasma shutdown, cooling loop integrity maintained, afterheat removal via intact loop.
LOFA2 II	LOFA in both cooling loops	Loss of flow in both cooling loops due to loss of pump power, normal plasma shutdown, afterheat removal via natural circulation in both loops.
LOFA2A IV	LOFA in both cooling loops	Loss of flow in both cooling loops due to inadvertent valve closure, accelerated plasma shutdown, afterheat removal tbd.
LOHS1HX II	LOHS in one heat exchanger	Loss of heat sink in one heat exchanger, normal plasma shutdown, cooling loop integrity maintained, afterheat removal via intact loop.
LOHS2HX III	LOHS in both heat exchangers	Loss of heat sink in both heat exchangers due to failure in secondary cooling systems, normal plasma shutdown, cooling loops' integrity maintained, temporary afterheat removal at reduced pressure by means of heat capacity of both cooling subsystems until thermal equilibrium is obtained with ultimate heat sink tbd, supported by purge gas system.

2.1.1.1.8.4 Safety Analysis

The analysis has been performed for the enveloping events postulated in Table 2.1.1.1.8-5 in a deterministic way. Two types of transient thermodynamic calculations have been carried out to determine (a) the 3D temperature distribution in a representative section of the BTM in the course of the event with the finite element code FIDAP, and (b) the thermal-hydraulic and heat transport mechanisms in the whole cooling subsystem with the system code RELAP, where , by the nature of the 1D RELAP code, the temperatures in the structures are approximate. In addition, some of the events were evaluated qualitatively by deduction and extrapolation.

Emphasis was placed in the analysis on the following items, while considering the conceivable plasma shutdown scenarios defined in Table 2.1.1.1.8-4.

- short term (several tens of seconds) temperature evolution at critical points of the BTM like first wall (FW), protection layer, pebble beds (with FIDAP)
- short term coolant temperature, pressure, and mass flow transients in the cooling loop(s) affected (with RELAP)
- long term (hours) heat transport and temperature evolution in loop components, in particular with view to afterheat removal (with RELAP)
- short term pressure transients and activation products release in confinement compartments (estimates, by mass balance).

The results obtained for each enveloping event are summarised in the following paragraphs in the order of event characterisation from Table 2.1.1.1.8-5. An assessment of the results in terms of adherence to safety requirements is given in section 2.1.1.1.8.5.

LOCA ex-vessel in one cooling loop (LOCA1EX)

The event has been analysed with FIDAP for the first 125 s into the transient, assuming the normal plasma shutdown scenario (see Table 2.1.1.1.8-4) from 1.9 MW nuclear power in the BTM and 0.5 MW/m^2 of surface heat flux. The radial power distribution at the start of the transient has been derived from the power density profiles in the different materials specified in Table 2.1.1.1.1-2. The decay heat power generation was ignored in this case. The model represented the blanket box with three first wall cooling channels and the neighbouring breeding zone as described in the thermal-hydraulic design, section 2.1.1.1.2. The depressurization of the broken loop was analysed with RELAP and the results were used as boundary conditions in FIDAP. Figure 2.1.1.1.8-1 gives an overview of the temperature evolution at distinct points of the BTM which are explained below.

The temperature in the hottest spot of the first wall steel (curve D) increases gradually, reaching a peak value of 610 °C after 36 s, that is 105 °C above the temperature at steady state. This occurs at the toroidal side of the BTM at the interface between MANET and beryllium layer opposing the affected first wall channels. For comparison, the temperature change in the corresponding nodes opposing the unaffected channels is only 35 °C (curve C). Maximum temperatures in the cooling plates which are of interest to the weldings reach a value of less than 500 °C after 60 s (not shown).

A similar behaviour as for the first wall structure is observed at the 5 mm beryllium protection layer. The temperature in the hottest spot reaches a peak value of 620 °C after 36 s, that is 90 °C above the steady state value (curve B). The maximum temperature in the nodes opposing the unaffected channels goes up to 570 °C, i.e., 35 °C above steady state (curve A). After the peak values have been passed the surface temperature decreases rather fast, reaching the long term limit of 450 °C within 100 s.

Temperatures in the breeder and beryllium pebble beds are little affected by this type of cooling disturbances. They generally tend to decrease (curve G for breeder and curves H, I for beryllium pebbles). Later in the event they will stabilise at a level below the nominal steady state temperature regime. Typical spatial peak values in the ceramic breeder pebbles are 625 °C at nominal steady state and presumably less than 400 °C after reaching quasi equilibrium during afterheat removal by one operating loop. Corresponding values for the beryllium pebble bed are 460 °C and less than 400 °C if the coolant inlet temperature is kept below 250 °C.

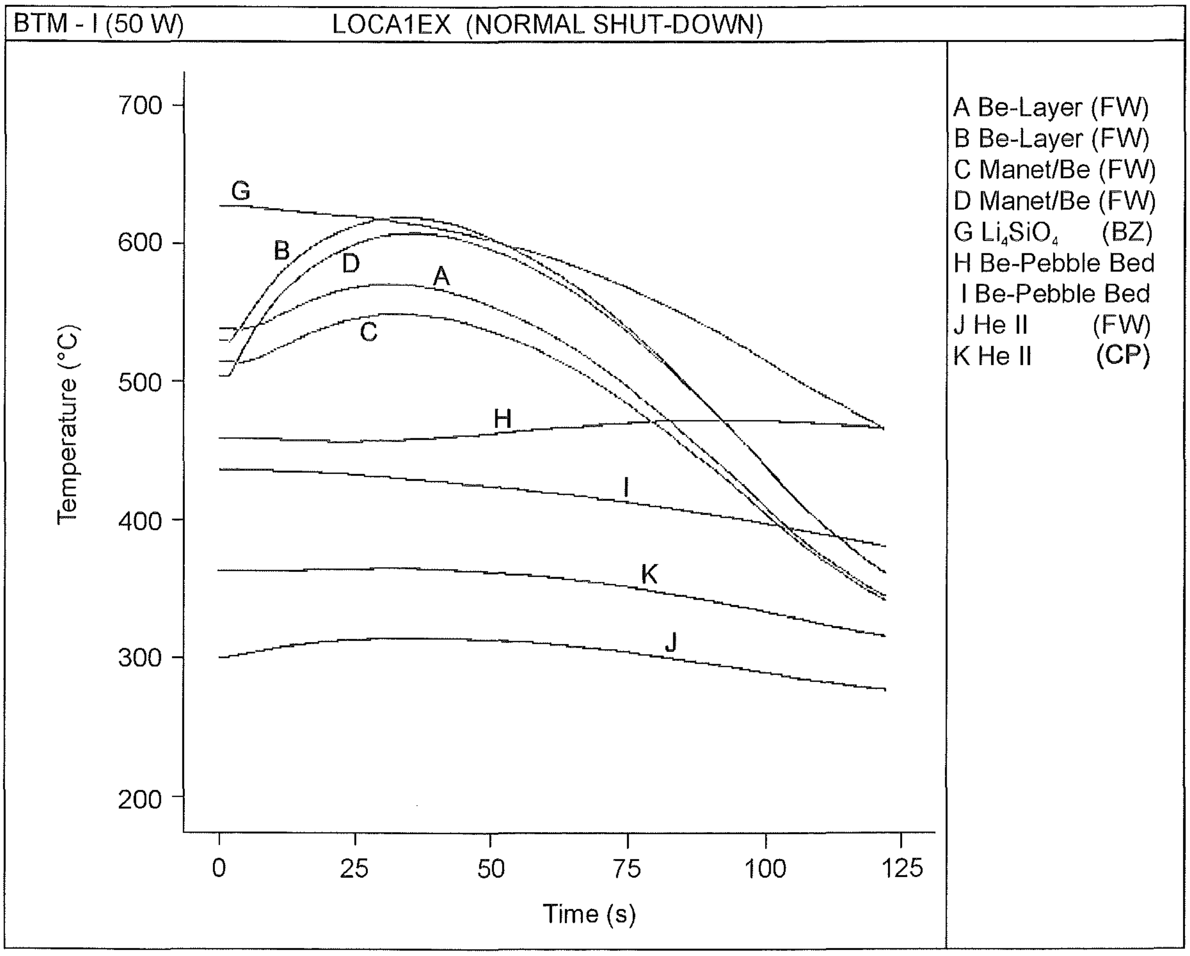


Figure 2.1.1.1.8-1: Temperature transients at hot spots of the BTM following the LOCA1EX event.

The thermal-hydraulic analysis performed with RELAP (although for a guillotine break in the hot leg instead of in the cold leg as specified in Table 2.1.1.1.8-5) yielded a fast pressure transient. For instance, the 'half-life' time of the helium pressure in the inlet and outlet manifold to the test module is 0.25 s and 0.1 s, respectively. Accordingly, the mass flow rate in the affected first wall channels experiences a brief burst of about 400 % of nominal at 0.1 s, followed by a decay with a half-life time of 0.25 s. During this phase there is a substantial cooling effect at the first wall which can be seen at curves A to D in Figure 2.1.1.1.8-1. However, the effect ceases after about 2 s and has, therefore, little impact on the temperature evolution in general. This may change in cases with more realistic leaks (smaller than the guillotine breaks in the largest diameter pipe) towards lower peak temperatures.

The initial jet forces acting at the broken pipe ends with an inner diameter of 148 mm amount to about 55 kN per side.

LOCA in-vessel in one cooling loop (LOCA1IN)

In terms of temperature evolution at critical points of the test module the results obtained for the LOCA1EX event (Figure 2.1.1.1.8-1) constitute a bounding case, since the in-vessel leak will inherently initiate a fast shutdown. The internal heat will be down within about a second and the surface heat will decay within several seconds, presumably 20 s. Figure 2.1.1.1.8-1 suggests that the temperature hump at the first wall and in the protection layer will stay at less then 50 °C above the steady state values.

In regard to pressure transients the process will be equally fast as in the LOCA1EX event (about 2 s) despite the fact that the blowdown now occurs into vacuum rather than to atmospheric pressure. Mass balance yields a pressure rise in the vacuum vessel of about 0.01 MPa given the helium inventory of 21.5 kg in one loop, a vacuum vessel free volume of 2000 m^3, and an assumed gas temperature of 300 °C.

LOCA ex-vessel in both cooling loops (LOCA2EX)

The short term temperature evolution has been analysed with FIDAP for the first 35 s into the transient. In this case an accelerated plasma shutdown scenario (see Table 2.1.1.1.8-4) was postulated, since a test run with the normal shutdown sequence had indicated, that the first wall temperatures would exceed allowable limits. Otherwise the modelling and the initial power distribution were the same as described for the LOCA1EX event.

As a result of the accelerated plasma shutdown the temperature in the hottest spot of the first wall increases more moderate as compared to the LOCA1EX case. The first wall steel reaches a peak temperature of 550 °C after 15 s, that is 38 °C above the temperature at steady state. This occurs at the toroidal side of the BTM at the interface between MANET and beryllium layer opposing one of the hot channels (the ones where the toroidal side regarded is in downstream direction). The temperature change in the nodes opposing the cold channels is little higher, 47 °C, and both temperature curves approach each other after 21 s while descending.

A similar behaviour as for the first wall steel is observed at the 5 mm beryllium protection layer with peak temperatures of 560 °C after 15 s, i.e., about 30 °C above steady state.

A series of computations with prolonged delay times of 10, 20, and 30 s suggest that in the LOCA2EX event a delay time of tbd s can be tolerated without exceeding the peak temperatures obtained in the LOCA1EX event.

The long term consequences need tbd with RELAP. In particular, it should be shown that the afterheat can be removed by forced circulation of air via one or both of the broken loops at tolerable temperatures. In defining temperature limits credit can be taken from the fact that this is a category V event, i.e., hypothetical with extremely low frequency of occurrence.

The pressure and mass flow transients are equal to those described with the LOCA1EX event, i.e., the two loops will be emptied within about 2 s, releasing 43 kg of helium, including about 1 mg (10 Ci) of tritium, into the tritium building. A certain amount of activation products trapped in the dust filters of the main loops will also be liberated. Analyses of hazards to the public in case of release of these source terms into the environment need tbd but with view to the small inventories they are not expected to become a safety problem.

LOCA in-vessel in both cooling loops (LOCA2IN)

With regard to short term temperature evolution at critical points of the test module the results obtained for the LOCA2EX event constitute a bounding case, since the in-vessel leak will inherently initiate a fast shutdown. Assuming the breach to occur in the first wall (rather than a simultaneous break of both feeders, which is extremely unlikely) the blowdown phase ot the helium coolant lasts for several tens of seconds providing effective cooling during the whole shutdown period. Thus, no short term temperature excursions are expected.

The long term afterheat removal before replacement of the BTM needs tbd. The concept is to circulate the vacuum vessel atmosphere through the open loops by circulators at reduced speed. Part of the afterheat will also be dissipated by radiation and convection inside the vacuum vessel and to the support frame. Further, the purge gas flowing at a rate of 0.85 g/s and at inlet/outlet temperatures of 20/450 °C has a heat capacity of about 1.9 kW, equivalent to the afterheat power obtained one day after shutdown.

The pressure rise in the vacuum vessel at the end of the blowdown will be twice as high as discussed with the LOCA1IN event, that is about 0.02 MPa. This is below the setpoint of any pressure relief device which assures the integrity of the vacuum vessel confinement boundary. The introduction of radioactive material into the vacuum vessel (1 mg of tritium plus tbd Bq residing in the helium-born activation products) are not deemed to affect the ITER operation after replacement of the test module. There is one exception: This type of postulated heavy first wall failure implies that the damaged box may give way to an injection of part of the beryllium and breeder pebbles into the vacuum vessel. Conceivable amounts are about 10 percent of the inventory, i.e., 12 kg of Li_4Si0_4 and 45 kg of beryllium. The routine dust clean-up procedure provided by ITER will be used to remove the particles from the vacuum vessel.

LOCA in one heat exchanger (LOCA1HX)

The heat exchangers (HX) are designed in a way that the high pressure helium flows inside the cooling tubes and the water flows on the shell side (section 2.1.1.3.2). The HX shell is designed to sustain the full primary helium pressure. Therefore, the maximum conceivable damage inside the HX is a guillotine break of one (or at most a few) cooling tubes. The outflowing helium will then pressurise the secondary cooling loop until the relief valve opens, discharging secondary water and primary helium into the secondary loop compartment.

Because of the restricted outflow of primary helium, the discharge process will last for several tens of seconds providing sufficient cooling of the BTM during the normal plasma shutdown. Therefore, a significant overshoot of structural temperatures in the BTM can be ruled out in the course of the transient.

The long term afterheat removal is assured by the intact loop at a low temperature level as indicated by the trends shown in Figure 2.1.1.1.8-1 for the LOCA1EX event.

The radiological implications in the compartment housing the secondary cooling system need tbd based on the building design. They are judged to be of the same quality as briefly discussed in the LOCA2EX event.

Leak inside test module of one cooling loop (LEAK1TM)

A large leak inside the BTM will pressurise the first wall box and all breeder and beryllium pebble beds, and thus, the main loop of the tritium extraction subsystem. According to the volumes and mean temperatures in the main cooling loop (2.7 m^3, 300 °C) and in the components of the tritium extraction loop (3.9 m^3, ≈-75 °C, estimated from data given in section 2.1.1.2) the pressure would balance at 1.6 to 1.9 MPa, depending on how much helium would be recharged by the pressure control system from the buffer tank of the main loop. In any way, the primary pressure in the affected loop will temporarily drop to about 0.2 MPa affecting the heat removal in the BTM. However, the pressure balance process will be inherently slow (due to the small size of the pipework connecting the BTM with the tritium extraction subsystem) so that a significant temperature increase in the BTM during the normal plasma shutdown sequence is not expected. It is certainly less than in the LOCA1EX event.

The long term afterheat removal is assured by the intact loop.

It is intended to operate the BTM, after the normal shutdown and check-up of affected subsystems, until the next ITER outage. If this situation lasts for more than a few days the consequences of the leak in terms of continued tritium extraction, tritium transport into the main loop, tritium removal capability from the main loop by the helium purification subsystem, and tritium permeation through the main loop boundary must be monitored.

Leak inside test module of both cooling loops (LEAK2TM)

The event sequence will essentially be the same as the one described under the subheading LEAK1TM, but the pressure would balance at 2.8 to 3 MPa and the transients would be further retarded. Thus, no problems are expected with short term temperature excursions which, however, needs to be proved (tbd).

There is also no doubt that the afterheat can be removed by forced convection at reduced pressure via one or both loops. Whether natural circulation is sufficient to keep the temperatures in the BTM at appropriate levels needs tbd.

In contrast to the LEAK1TM event it is not intended to continue ITER operation after a LEAK2TM event without replacing the HCPB-BTM. Note that the frequency of occurrence is judged to be extremely low (category IV), since no single failure can lead to the LEAK2TM event.

LOFA in one cooling loop (LOFA1)

This loss of flow event can be caused by pump coast down or by inadvertent valve closure. The latter is the more unfavourable initiator, since the flow in the affected loop will be completely stopped within a few seconds (unless the heater bypass is partially open).

The short term temperature response in the BTM will be comparable with the LOCA1EX event (Figure 2.1.1.1.8-1).

The afterheat can be removed by natural circulation through the intact loop, for which case the equilibrium temperature levels need tbd. It is likely that natural circulation can be restored soon in the affected loop.

LOFA in both cooling loops (LOFA2)

The event has been analysed with RELAP for up to 1400 s into the transient for an assumed pump coast-down and normal shutdown scenario (see Table 2.1.1.1.8-4) from 1.9 MW nuclear power in the BTM and 0.5 MW/m^2 of surface heat flux. After ramp-down of the internal power the afterheat history was used as heat source (1.5 % of the initial power profile declining to 1.36 % in the time interval analysed). No flow control action was assumed during the transient, neither in the primary loops nor in the secondary loops. Figure 2.1.1.1.8-2 gives an overview of the temperature evolution in selected nodes of the RELAP model. The results are discussed below.

Due to the loss of pump power the mass flow rate in the loops fades from the nominal value (1.85 kg/s per loop) at a half-life time of about 3 s, reaching natural circulation driven equilibrium of 5.3 % of the nominal flow rate after 60 s. Thereafter the coolant temperature stabilises at 125 °C/420 °C at test module inlet/outlet.

The temperature in the hottest nodes of the first wall (curve MANET/Be, FW1) reaches a peak value of 625 °C after 70 s, that is 90 °C above the temperature at steady state. This occurs at the toroidal side (downstream) of the BTM, ignoring in this 1D model the recuperative heat transfer between adjacent coolant channels with alternating flow direction. Also shown are the temperature histories at the first wall/helium interface (curve MANET/He, FW1) and at the back side of the first wall (curve MANET/Back, FW2). The latter is of interest with regard to the welds connecting the first wall to the cooling plates. It increases from 280 °C to 470 °C before it declines and stabilises at a low level.

The temperature in the beryllium protection layer (curve Be-Layer, FW1) reaches a maximum value of 635 °C with the overshoot lasting for 100 s.

The temperatures in the hottest nodes of the breeder and beryllium pebble beds essentially decrease from the beginning of the event and stabilise beyond the calculation interval at a level below 300 °C. The largest temperature transient in the Li_4SiO_4 pebble bed amounts to 2 °C/s.

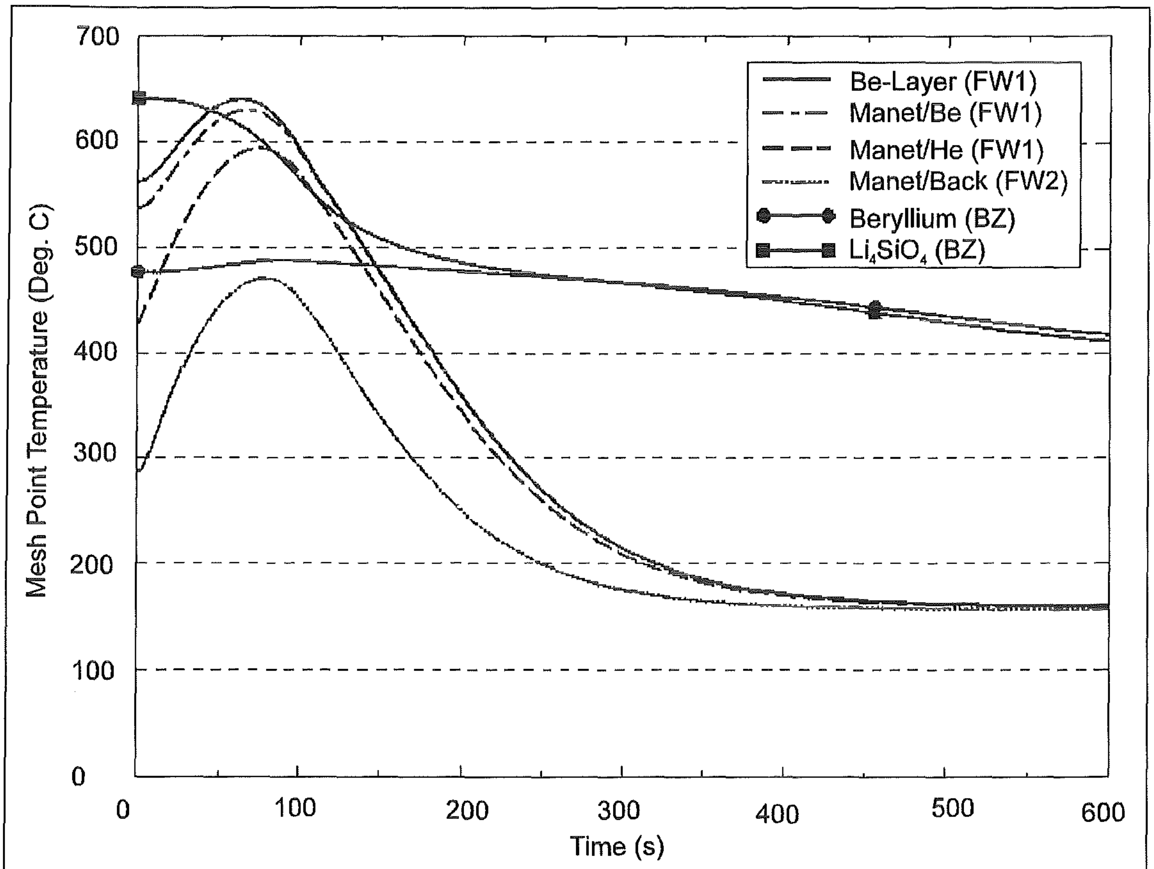


Figure 2.1.1.1.8-2: Temperature transients at hot spots of the BTM following the LOFA2 event.

It should be noted that the structural temperature goes down to about 150 °C if no flow control is provided. It can be kept at a higher level, if needed, by throttling the flow through the heat exchanger and/or by turning on the heaters. It should also be noted that the thermal capacity of all the loop components outside the BTM proper has not yet been included in the model. When taken into account, this will affect the long term trends.

LOFA in both cooling loops due to valve closure (LOFA2A)

In this variant to the LOFA2 event the initiator is the coherent inadvertent valve closure in both cooling loops rather than a loss of pump power. The valve closure, including the valves in the heater bypass lines, is assumed to occur within a few seconds (say 3 s).

With regard to short term temperature transients this event is equivalent to the LOCA2EX scenario, provided an accelerated plasma shutdown is achieved. Then the temporary temperature increase at the first wall is about 50 °C (see LOCA2EX). Again, it has tbd, what delay time to shutdown the plasma can be tolerated without exceeding temperature limits.

For the long term afterheat removal the following strategy will be employed: (1) restore natural circulation in at least one loop by re-opening (if needed manually) the main valves, (2) if action 1 fails, open valves in heater bypass lines to make use of heat capacity and heat losses in loop components to slow down heat-up of the BTM, (3) if actions 1 and 2 fail, flood the vacuum vessel to enhance in-vessel heat dissipation, (4) after about 1 day heat removal capability of the purge gas should be sufficient to control long term temperatures. Details are tbd.

LOHS in one heat exchanger (LOHS1HX)

Loss of heat sink can occur if the secondary loop fails or if the ultimate heat sink (both are part of ITER ancillaries) gets lost.

For the short term temperature transients the LOCA1EX event represents a bounding case (Figure 2.1.1.1.8-1). It is expected that the hump of the first wall temperature curves are significantly reduced due to the heat capacity of the failed loop. Details have tbd.

The long term afterheat removal is assured by forced flow in the other loop at reduced circulator speed or even by natural circulation.

It is conceivable that the mean helium temperature in the affected loop will rise beyond the steady state value (≈300 °C). The pressure control system will then keep the pressure within design limits assuring loop integrity.

LOHS in both heat exchangers (LOHS2HX)

Simultaneous loss of heat sink in both heat removal chains is generally considered as an extremely unlikely event (or even a hypothetical event) if redundant systems are provided. The degree of redundancy of ITER ancillaries are not yet known. Therefore, a category III (unlikely sequence) is assigned to the LOHS2HX event.

In regard of short term temperature transients in the case of normal plasma shutdown the results obtained for the LOFA2 event (Figure 2.1.1.1.8-2) can probably be taken as a guide line. Details need tbd. should the excess temperatures turn out to be beyond design limits, an accelerated shutdown sequence would be demanded.

The long term behaviour has tbd, taking into account the large heat capacity of the cooling loops as well as heat losses through component insulation. The following estimate gives an idea of the heat-up rate of the steel contained in both loops, assuming adiabatic and isothermal heat-up caused by afterheat from the BTM. The approximately 15000 kg of stainless steel in the loops plus 2600 kg of ferritic steel in the test module (Table 2.1.1.1.8-1) would be heated up at a rate of 15 °C/h, 6.5 °C/h, and 0.7 °C/h after decay times of 0 seconds, 1 hour, and 1 day, respectively. The integrated temperature rise would be about 40 °C during the first day. This is a slow transient and allows (but also requires) to operate the circulators at minimum speed and minimum power. Please note that the rated power of both circulators amounts to about 200 kW which is much more than the afterheat produced in the BTM. Detailed analyses will show how the LOHS2HX event can be handled best, including the strategy of vacuum vessel flooding and heat removal by purge gas as outlined with the LOFA2A event.

2.1.1.1.8.5 Safety Assessment

In the course of development of the HCPB the fundamental safety principles formulated in section 1.3 were kept in mind. In short, these principles are: dose and radioactivity release limits, exposure of site personnel, account for uncertainties of plasma physics, multiple levels of protection (defence in depth), passive safety, minimisation of radio-toxic materials, implementation of common industrial standards.

In addition to these fundamental safety principles a number of specific safety requirements have been defined in section 1.3.5. In this assessment it will be checked whether these requirements are met by the HCPB system. In particular, each enveloping event defined and analysed in the previous section is assessed against all of the requirements. Table 2.1.1.1.8-6 represents the checking matrix to serve as overview. A discussion of the assessment and conclusions, by requirements, is given below.

Requirement 1: Take into account postulated events

The design basis for the BTM system takes into account the initiating events as identified by the safety analysis (section 2.1.1.1.8.4). The list of enveloping events (Table 2.1.1.1.8-5) is judged to be complete as far as the test module proper, including the cooling subsystem, is concerned. Initiating events generated in the helium purification subsystem and in the tritium extraction subsystem are not considered.

Requirement 2: Support structure shall be capable to react loads in a way as to minimise the load on the vacuum vessel

This is primarily a design requirement with the largest loads to be accounted for being cause by the dead weight of the BTM and by plasma disruption forces. Of the events considered only LOCA1IN and LOCA2IN might generate extra jet forces on the support structure which, however, would be comparably small. Analysis needs tbd by the support structure and vacuum vessel design groups.

Requirement 3: Small contribution to ITER source terms

Only the ex-vessel LOCAs (LOCA1EX, LOCA2EX) and the LOCA in one heat exchanger (LOCA1HX) can lead to release of radioactive material into the building and, thereby, add to potential ITER source terms. Since the inventories are small (section 2.1.1.1.8.1) the requirement is fulfilled, pending the quantification of activation products release in the events mentioned.

Requirement 4: Minimise liquid spills into the vacuum vessel

There are no liquids in the BTM system, except for the water in the secondary cooling loops. Only the in-vessel LOCAs (LOCA1IN, LOCA2IN) concurrently with a major leak in the heat exchanger could lead to a limited cold water spill into the vacuum vessel. The event sequence is considered as beyond design basis accident. Thus, the safety requirement is met.

Requirement 5: Assure fast thermal relaxation of overheated first wall

The analyses have shown that the short term temperatures of the beryllium protection layer stay below about 650 °C in all initiating events, provided the shutdown is achieved as specified. This is far below the limit to avoid Be-steam ignition scenarios (provisionally 800-1300 °C according to section 1.3.3). The requirement is met.

Requirement 6: Assure long term first wall temperature limits

The long term temperature limit of ≈450 °C set in section 1.3.3 to avoid excessive H_2 production has to be verified for the initiating events involving both cooling loops, i.e., LOCA2EX, LOCA2IN, LOFA2A, and LOHS2HX. Several back-up measures have been indicated to support or restore afterheat removal. Major problems are, therefore, not expected.

Requirement 7: Segmentation of BTM cooling loops

Two separate heat removal loops of 2x50 % heat capacity are foreseen both for the main cooling subsystem and the helium purification subsystem. This serves the implementation of the single failure criterion and makes the more critical scenarios, where both loops are involved (see requirement 5), unlikely or extremely unlikely events. The requirement is met.

Requirement 8: Avoid asymmetric temperature distribution

The staggered flow pattern of the two cooling streams in the BTM box as well as in the cooling plates of the breeding zone (alternating loops and alternating flow direction in subsequent flow channels and cooling plates) minimises gross asymmetric temperature distributions in the BTM. The resulting local temperature variations have been analysed and are found to be acceptable in terms of thermal stresses. The requirement is met.

Requirement 9: Guarantee passive off-normal heat removal

Heat removal by natural circulation has been demonstrated to be very effective (compare LOFA2 event in section 2.1.1.1.8.4), aided by the elevation of the heat sink relative to the heat source (≈20 m). The effectiveness at distorted conditions, e.g., reduced system pressure, broken loop, a single loop operating, will be analysed in connection with the long term first wall temperature limits addressed in requirement 6. No treatment is envisaged to increase the emissivity of BTM surfaces to enhance thermal radiation.

Requirement 10: Minimise afterheat within material limits

The choice of MANET as structural material is preliminary and may change in the course of development (section 2.1.1.1.7). With view to afterheat and activation products MANET may not be the optimal solution but is a viable choice. The requirement is met.

Requirement 11: Strive for limitation of dust, tritium, corrosion products

Beryllium as protection layer at the first wall is the current ITER reference choice. The contribution of the BTM to the dust production inside the vacuum vessel is negligible. The tritium inventory in the BTM is small (Table 2.1.1.1.8-2), in particular, the mobilisable proportion in the cooling subsystem. Corrosion and corrosion product transport are generally not a problem with helium cooling systems. Nevertheless, the last point will be further evaluated concerning LOCA events. The requirement is met.

Requirement 12: Provide monitoring to meet requirements

The instrumentation and control system has not yet been elaborated. The minimum required monitoring for process control is mentioned in sections 2.1.1.2, 2.1.1.3, and 2.1.1.4. In terms of signal generation to initiate plasma shutdown the requirements are not very demanding, since most of the events analysed tolerate normal plasma shutdown (Table 2.1.1.1.8-5). Tolerable delay times have to be determined for almost each event (Table 2.1.1.1.8-6) where a trade-off between delay time and ramp-down time should be accepted.

Requirement 13: Minimise the dose to personnel

The design of decontamination, shielding, remote operation, flask transfer functions concerning minimal dose to personnel in the course of maintenance and decommissioning has to be elaborated. No special problems are seen with BTM compared with general ITER procedures. Specific questions related to activation products release for several LOCA events need to be analysed.

Requirement 14: Minimise radio-toxicity of radioactive waste

Activation calculations are not yet performed. Extrapolations of the findings of analyses performed for the DEMO blanket indicate that the breeder and multiplier can be classified as low and intermediate level waste according to IAEA interpretations. The small amount of structural material in the first wall of the BTM may have to be classified as high level waste. No extra braze or cladding materials are used. Hence, the requirement is judged to be fulfilled, pending a determination of activation products.

Requirement 15: Failure of the first wall must not cause rupture of the vacuum vessel

It is shown for the LOCA in-vessel events that the vacuum vessel can be pressurised to at most 0.02 MPa by the primary coolant (LOCA2IN), which is not critical. Limited water ingress from the secondary cooling system, simultaneously with the LOCA, is hypothetical and will not add significantly to the risk of vacuum vessel overpressurisation. The requirement is met.

Summary and conclusion

Summing up the assessment, it can be concluded that 9 out of 15 safety requirements specified in section 1.3.5 are shown to be fulfilled by the proposed

HCPB-BTM design. Some of the postulated events in relation to certain requirements need to be further investigated as indicated by the 'tbd' label in Table 2.1.1.1.8-6. Pending these results, no fundamental safety concern has been identified so far that could violate any requirement.

Table 2.1.1.1.8-6: Checking postulated events against safety design requirements

Safety Design Requirements according to 1.3.5 (shortened)		Event Family LOCA (and LEAK)							Event Family LOFA			Event Family LOHS	
No	Description	1EX	1IN	2EX	2IN	1HX	1TM	2TM	1	2	2A	1HX	2HX
1	Take into account postulated events	*	*	*	*	*	*	*	*	*	*	*	*
2	Support structure capability to react loads	*	tbd	*	tbd	*	*	*	*	*	*	*	*
3	Small contribution to ITER source term	tbd	*	tbd	*	tbd	*	*	*	*	*	*	*
4	Minimise liquid spills into the VV	*	*	*	*	*	*	*	*	*	*	*	*
5	Fast thermal relaxation of overheated FW	*	*	*	*	*	*	*	*	*	*	*	*
6	Long term FW temperature limits met	*	*	tbd	tbd	*	*	*	*	*	tbd	*	tbd
7	Segmentation of BTM cooling loops	*	*	*	*	*	*	*	*	*	*	*	*
8	Avoid asymmetric temperature distribution	*	*	*	*	*	*	*	*	*	*	*	*
9	Passive off-normal heat removal assured	*	*	tbd	tbd	*	*	tbd	tbd	*	tbd	tbd	tbd
10	Minimise afterheat within material limits	*	*	tbd	tbd	*	*	*	*	*	tbd	*	tbd
11	Limitation of dust, tritium, corrosion prod.	tbd	*	tbd	*	tbd	tbd	*	*	*	*	*	*
12	Monitoring provided to meet requirements	tbd	*	tbd	*	*	tbd	tbd	tbd	tbd	tbd	tbd	tbd
13	Minimise the dose to personnel	tbd	*	tbd	tbd	tbd	*	*	*	*	*	*	*
14	Minimise radio-toxicity of radioactive waste	tbd	tbd	tbd	tbd	tbd	tbd	tbd	tbd	tbd	tbd	tbd	tbd
15	Failure of FW does not cause rupture of VV	*	*	*	*	*	*	*	*	*	*	*	*

*Safety design requirement met or not applicable in case of event

2.1.1.1.9 Reliability Analysis

The availability of the HCPB-BTM including the supply pipes inside the vacuum vessel (VV) has been analyzed in order to get preliminary quantitative estimates. No experience is available on the reliability of equipments like the BTM; therefore, the failure rate has to be determined "synthetically" using the failure rates of the basic components of the BTM like welds, tubes, and pipe bends. Data for these components are available from other technologies, combined with assumptions based on expert opinion.

The BTM is designed such that certain types of failures like small internal leaks can be tolerated and will not necessarily lead to shut down of the facility. This is taken into account by two failure modes: the "leak" failure mode and the "loss of integrity" failure mode. The failure rates given in Table 2.1.1.1.9-1 represent the leak mode. The loss of integrity mode is applied to those welds where leaks can be tolerated. In these cases values are used which are one order of magnitude less. This procedure corresponds to the approach applied in the analyses carried out for the DEMO blankets [2.1.1.1.9-1, -2], and is in agreement with the recommendations given in [2.1.1.1.9-3].

A repair of the BTM inside the vacuum vessel is not envisaged. Hence, a loss of operability necessitates the exchange of the BTM. According to [2.1.1.1.9-4] it has been asssumed that 1344 h (8 weeks) are needed for this operation. This "repair time" (MTTR) determines - together with the failure rates - the unavailability of the BTM which is defined as the probability for the inoperability of the BTM when it should be operable.

The FMEA has shown that four failure effects have to be considered: helium ingress into the VV, distortions of the BTM coolability, distortions of the purge gas flow to the pebble beds inside the BTM, and loss of the structural integrity of the BTM. The quantitative evaluation of the failure rates and availabilities has been carried out using standard methods of fault tree analysis. The results are compiled in Table 2.1.1.1.9-2. The BTM reliability is dominantly determined by the failure effect "helium ingress into the VV" with a contribution of 93 % to the total unavailability. The main cause are leaks of the supply pipes inside the VV. The low contribution of the BTM itself to this case is due to the high degree of fault tolerance against internal leaks. The contribution of the failure effect "loss of BTM structural integrity" to the overall unavailability amounts to only 2 %. The total failure rate of the BTM is less than 0.01 a^{-1}, and the corresponding unavailability less than 0.1 %. The increase of reliability compared to the DEMO blanket system is a consequence of the size of the BTM which represents only a very small fraction of the DEMO blanket.

A parameter variation has shown that an increase of the failure rates given in Table 2.1.1.1.9-1 by one order of magnitude would lead to an overall unavailability of about 1 %. This shows that a margin is available to accommodate effects which are not yet included in the analysis, e.g. irradiation effects.

References

[2.1.1.1.9-1] H. Schnauder, Forschungszentrum Karlsruhe, unveröffentlichter Bericht, August 1995

[2.1.1.1.9-2] H. Schnauder, C. Nardi, M. Eid, Comparative Availability of the four European DEMO-Blanket Concepts in view of the Selection Exercise, accepted for publication in: Fusion Engineering and Design.

[2.1.1.1.9-3] R. Bünde, S. Fabritssiev, V. Rybin; Reliability of welds and brazed joints in blankets and its influence on availability; Fusion Engineering and Design 16 (1991) 59-72, North-Holland

[2.1.1.1.9-4] Test Blanket System (WSB 1.x), U.S. Version, Design Description Document, Draft 9/29/95

Table 2.1.1.1.9-1: Failure rates for the components of the BTM

Failure component		Failure rate
Pipe failure		$3.0 \cdot 10^{-9}$
Collector failure		$1.0 \cdot 10^{-8}$
EB weld	*[1/mh]*	$1.0 \cdot 10^{-9}$
TIG weld	*[1/mh]*	$1.0 \cdot 10^{-9}$
Diffusion weld	*[1/mh]*	$1.0 \cdot 10^{-8}$
Longitudinal weld	*[1/mh]*	$1.0 \cdot 10^{-9}$
Butt weld	*[1/h]*	$1.0 \cdot 10^{-9}$
Pipe bend (180º)	*[1/h]*	$1.0 \cdot 10^{-8}$
Pipe bend (90º)	*[1/h]*	$5.0 \cdot 10^{-9}$

Table 2.1.1.1-2: Results of the BTM reliability analysis

Failure effects	**Failure rate** *[1/a]*	**Unavailability** *(absolute/percent)*
He ingress into vacuum chamber	$3.7 \cdot 10^{-3}$	$\mathbf{5.8 \cdot 10^{-4}}$ / 93
Distortion of the BTM coolability	$1.6 \cdot 10^{-4}$	$\mathbf{2.5 \cdot 10^{-5}}$ / 4
Distortion of purge gas flow inside BTM	$2.6 \cdot 10^{-5}$	$\mathbf{4.0 \cdot 10^{-6}}$ / 1
Loss of structural integrity of BTM	$6.1 \cdot 10^{-5}$	$\mathbf{9.5 \cdot 10^{-6}}$ / 2
Total	$4.0 \cdot 10^{-3}$	$\mathbf{6.2 \cdot 10^{-4}}$ / 100

2.1.1.1.10 Electrical Design

The BTM is electrically connected to the blanket back plate, that mostly defines the boundary conditions for induced electric currents determination during transient electromagnetic events.

Diagnostic cables will be provided between the test blanket module and the ITER Instrumentation and Control Systems and the Test Blanket Control Console. Power and control will be required at the transporter for the maintenance operations. The diagnostic conduit linking the BTM to the Test Blanket Control Console includes electrical lines to the following measuring and monitoring devices:

a) coolant helium, structural material, beryllium pebble bed, ceramic pebble bed temperature sensors,
b) coolant helium flow sensors,
c) coolant helium pressure and pressure drop sensors,

Number and location of the systems:

a) sensors location in BTM: (TBD)
b) measuring system location: ancillary rooms outside the biological shield

2.1.1.2 Tritium Extraction Subsystem

The main design data of the tritium extraction system are given in Table 2.1.1.2-1. The system has to be designed for a pressure of 8 MPa because leakages from the coolant system can lead to a pressure increase. In addition, relief valves are forseen to cope with the case of over-pressurization. For radiological safety reasons, the system must be installed in a secondary containment.

Table 2.1.1.2-1 : Main Design Data for the Tritium Extraction System

Power in Test Module (EPP)	2.3 MW [a)]
He Mass Flow	0.85 g/s = 17 Nm3/h
Swamping Ratio	He : H_2 = 1000
Tritium Production Rate	0.15 g / day
Partial Pressures[b)] p (H_2) p (HT+HTO) p (H_2O)	110 Pa 0.29 Pa[c)] 0.18 Pa[c)]
Extraction Rates H_2 HT H_2O / HTO	18.36 mole /day 0.04 mole /day ≈ 0.9 g / day
Temperature of Purge Gas at Test Module Outlet at Test Module Inlet	 450 °C 20 °C
Pressure of Purge Gas at Test Module Outlet at Test Module Inlet Pressure Drop in Test Module	 0.106 MPa 0.120 MPa 0.014 MPa

a) 1.9 MW in BPP 2.3 MW in EPP (including 0.3 MW in First Wall in both cases)

b) Average values at Test Module outlet (accounting for plasma pulse dwell time)

c) about 80 % of HTO is assumed to be converted to HT + H_2O by isotopic exchange, no HTO / H_2O is considered to be reduced by the steel walls

Process Description

A flow sheet of the tritium extraction system is shown in Figure 2.1.1.2-1. Instrumentation for process control like sensors for temperature, pressure, flow rate, etc. are not included in this figure.

After passing an ionization chamber the purge gas stream is sent through a cooler and then through a particulate filter (No. 3) to remove particulate material which might be carried out from the blanket zone. The arrangement of these filters has to be planned in such a way that they are easily exchangeable (within 1-2 hours). Downstream of the particulate filter, there is a bypass line leading to the compressor (No.8). This line can be used for initial scavenging of the blanket test module.

The Q_2O content (Q = H, T) of the gas is frozen out in the cold trap (No.4) operated at ≤ -100°C. The residual Q_2O concentration is < 0.015 ppm. The amount of ice accumulated within 6 days is of the order of a few grams (max. 6 g). This is advantageous for two reasons: (a) the construction of the trap is relatively simple, and (b) it is not necessary to exchange the water collector (Volume ≤ 200 ml) after each test run.

The purge gas is further cooled down by a recuperative heat exchanger (No. 5) and then passed through an adsorber bed (No. 6a) operated at liquid nitrogen (LN_2) temperature (-196°C). The bed is filled with 5A zeolite pellets which adsorb molecular hydrogen as well as gaseous impurities and residual moisture. The bed contains filters on the down-stream and upstream side to prevent particulate material from being transferred during loading or unloading operations. In addition, the bed is equipped with a LN_2 chiller and an electrical heater. The second bed provides additional adsorption capacity; it can be also used when the first bed is being unloaded or regenerated.

The clean gas leaving the adsorber bed is utilized in the heat exchanger mentioned above to precool the gas coming from the cold trap. It is then further warmed up by an electrical heater. The next components are the purge gas blower (No.8) coming in contact only with clean gas at room temperature, and the helium make-up unit (No. 9) where hydrogen is added to provide a He : H_2 swamping ratio of 1000 for the gas reentering the blanket test module.

Ancillary Installations

The main tasks of the ancillary installations are:

- to facilitate the transfer of tritium from the main components of the purge gas system to the isotope separation system (ISS) and the water detritiation system (WDS), respectively;
- to prepare the components of the main purge gas loop for the next tritium extraction campaign.

Fig.2.1.1.2-1 Flow sheet of the Tritium Extraction Subsystem

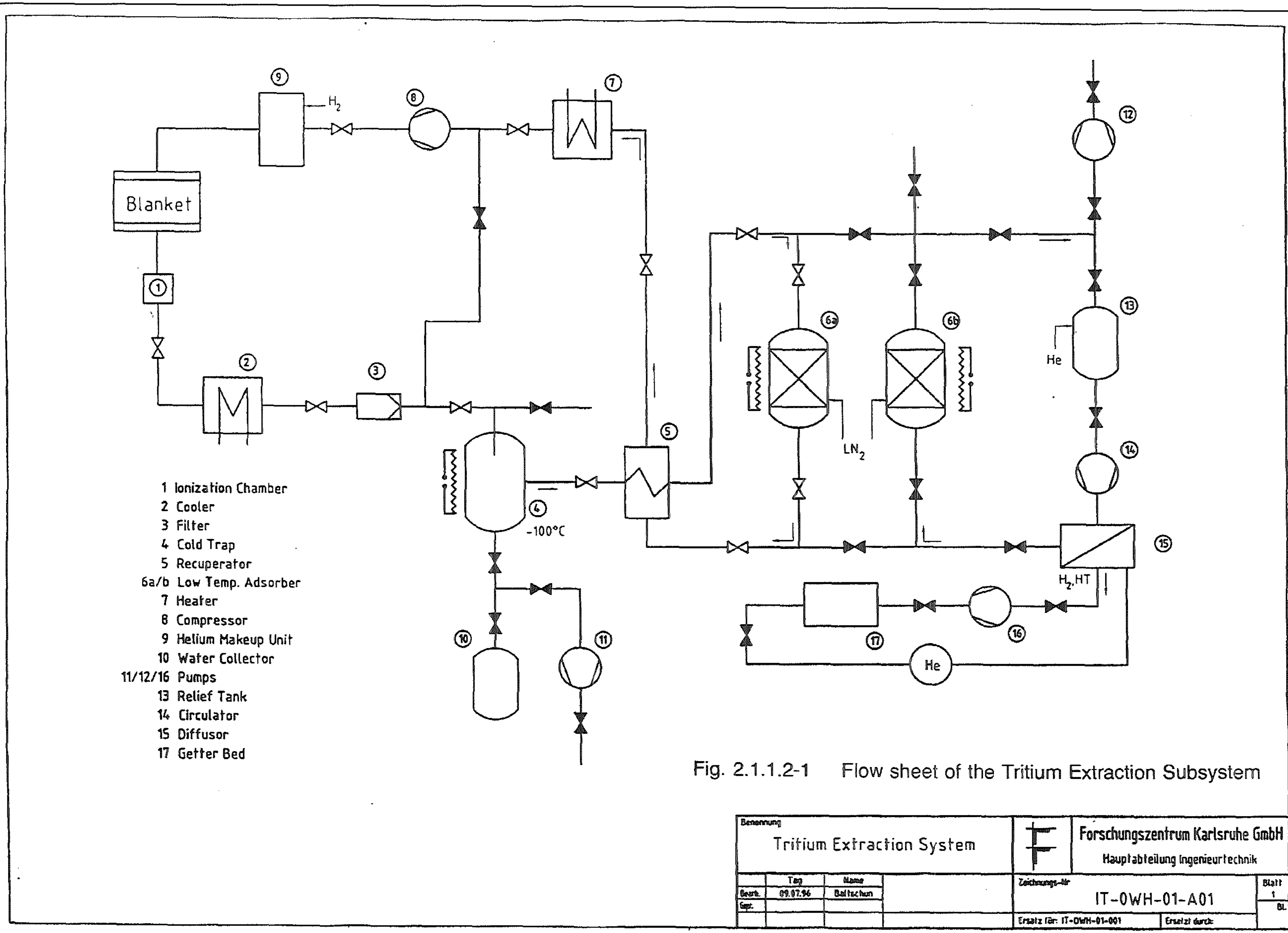


Fig. 2.1.1.2-1 Flow sheet of the Tritium Extraction Subsystem

Unloading of the cold trap:

After closing the inlet and outlet valves the ice is thawn with the help of an electrical heater. The liquefied water is drained into an evacuated water collector (No.10) which is later on transferred to the water detritiation system and replaced by an empty collector vessel.

Unloading and regeneration of a low temperature adsorber bed:

It is important to consider the pressure increase during desorption in the warm-up phase; the maximum amount of adsorbed H_2/ HT is 110 mole. A relief tank with a volume of 2 m3 is available, therefore, to be connected with the adsorber bed before the temperature increase is started (from -196 °C to about + 20°C). The pressure will remain below 0.2 MPa even though the tank has to be prefilled with 50 kPa helium which is needed as a carrier gas for the hydrogen isotopes during the further unloading process. With the help of a circulation pump, the desorbed gas is circulated several times through the adsorber and through a Pd/Ag diffuser where the hydrogen isotopes are separated from the helium carrier gas. At the low pressure side of the diffuser, there is a small helium loop containing a circulation pump and several uranium getter beds for storage of the hydrogen isotopes. When the loading capacity of these beds is reached they will be transferred to the Isotope Separation System and replaced by fresh beds. This is expected to be needed after 6 days of nominal operation with back-to-back pulses.

At the end of the unloading cycle, the relief tank is evacuated (by use of pump 12 leading to the Waste Gas System) and refilled with 50 kPa.

If the adsorber bed is to be fully regenerated it must be heated to about 300 °C. The desorbing moisture and impurities will then be sent to the Waste Gas System (via pump 12).

Analytical Tools

The processes of tritium extraction and purge gas purification are controlled by continuous measurement of the tritium concentration at several points of the gas loop and by taking gas samples for chemical analysis.

a) Ionization chambers are used at the following points:

- at the loop inlet (see Fig. 2.1.1.2-1),
- downstream of the cold trap (after removal of HTO, the fraction of HT can be determined and - as the total activity is known - also the HT/HTO ratio),
- downstream of the adsorber bed (to control the removal efficiency of the bed),
- downstream of the relief tank (to control the unloading process as well as the removal efficiency of the Pd/Ag diffuser and the tritium concentration of the gas sent to the Waste Gas System at the end of the deloading / regeneration process).

b) Two gas chromatographs (GC) are used to measure the gas composition at the inlet and the outlet of the tritium extraction system (in the first case downstream of filter unit 3, in the second case downstream of the helium make-up unit 9). In both cases, small gas transfer pumps are needed to transport gas samples to the GC where they are quantitatively analyzed with respect to Q_2, N_2, CO, etc.

A moisture detector which would be very difficult to operate at Q_2O concentrations less than 1 ppm appears not to be necessarry since the corresponding information can be obtained from the amount of water collected in the cold trap and / or from the results of the measurements carried out with the ionization chambers.

Due to the presence of varying magnetic fields with intensities up to 0.2 T at the location of the tritium extraction system it may be necessarry to consider screening measures for the analytic instrumentation as well as for the signal cables to the remote control station. In particular, magnetic valves have to be avoided unless they have been tested under comparable electromagnetic field conditions.

Space Requirements

It is intended to install the purge gas system in the pit adjacent to the port of the blanket test module. A space of 2.55(1.4)m x 8m and a height of 7.5m is available on one side of the transporter corridor in the pit. The size of the main components has been estimated and listed in Table 2.1.1.2-2. The integral space requirement in the pit is about 105 m3. Not included here is the space for a control station and for electrical cabinets which should be placed at the Test Blanket Control Console outside the pit. It is also expected that supply and diposal facilities are externally available. Respective wall penetrations are listed in Table 2.1.1.2-3, while Table 2.1.1.2-4 summarizes the requirements to be supplied from locations outside of the pit. Figure 2.1.1.2-2 shows the arrangement of the components of the tritium extraction system in the pit.

Fig. 2.1.1.2-2 Layout of the Tritium Extraction Subsystem

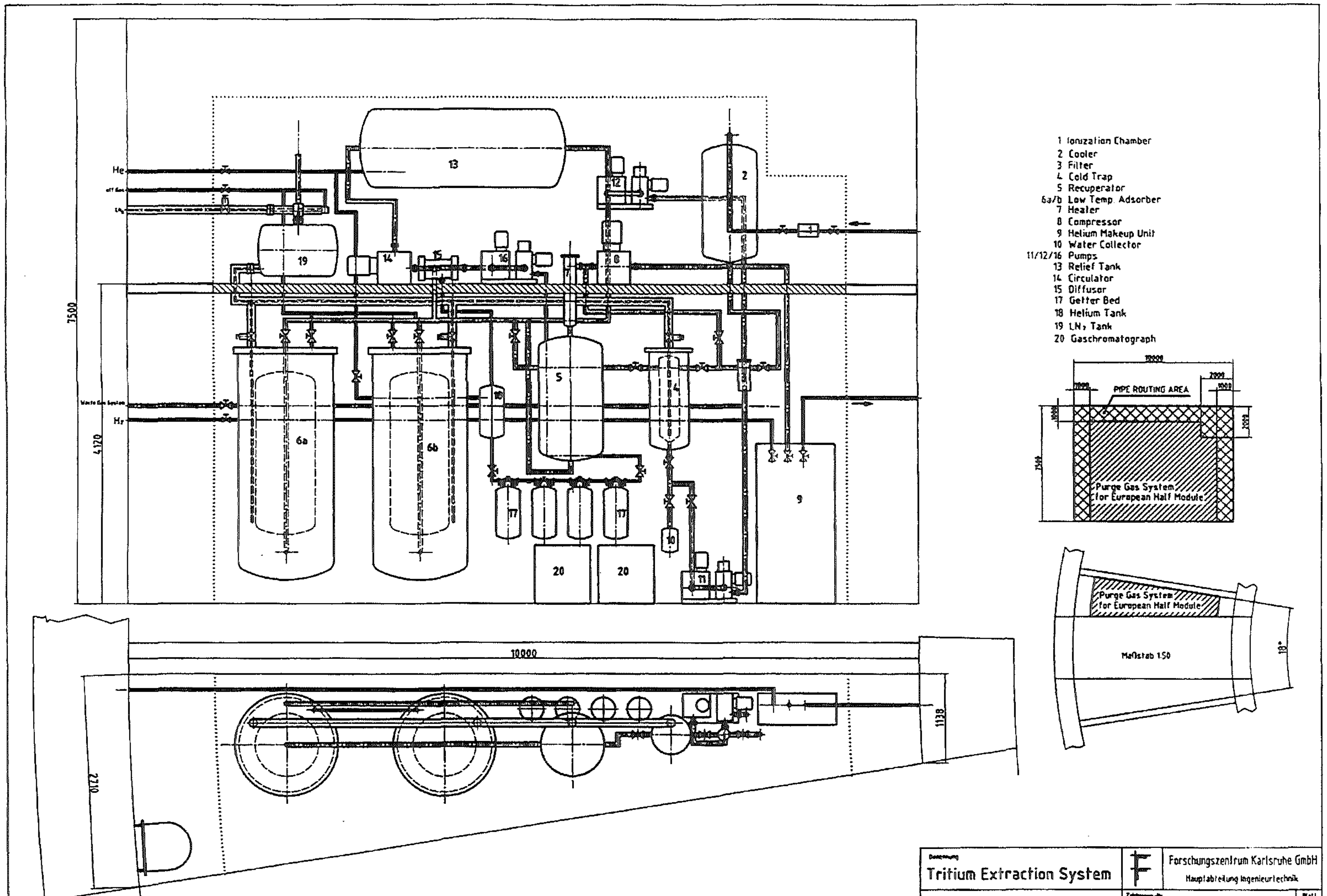


Fig. 2.1.1.2-2 Layout of the Tritium Extraction Subsystem

Table 2.1.1.2-2: Size of the Main Components

No.	Component	Size a)		
1a/b	Ionization Chamber	D: 200	H: 400 (each)	
2	Cooler	D: 700	H: 1500	
3	Filter	D: 100	H: 400	
4	Cold Trap	D: 500	H: 1300	
5	Recuperator	D: 800	H: 1600	
6 a/b	Low Temp. Adsorber	D: 1600	H: 3000	
7	Heater	D: 100	H: 900	
8	Compressor	L: 600	W: 590	H: 790
9	Helium Makeup Unit	L: 1000	W: 400	H: 2000
10	Water Collector	D: 200	H: 320	
11/12/16	Pumping Sets	L_1: 400 L_2: 500	W_1: 350 W_2: 240	H_1: 480 H_2: 460
13	Relief Tank	D: 1000	H: 2600	
14	Circulator	L: 800	W: 390	H: 410
15	Diffusor	D: 300	H: 500	
17	Getter Bed (4 Units)	D: 300	H: 670 (each)	
19	LN_2-Supply Tank	D: 650	W:1100	H: 1150
20	Gaschromatograph (2)	D: 700	W:700	H: 700

a) D = Diameter, L = Length, W = Width, H = Height
all dimensions in mm and including thermal insulation where needed

Table 2.1.1.2-3: Wall Penetrations

	Medium	Number of Tubes	Diameter (mm)[a]
Torus Side	Purge Gas from Blanket	2	45
	Purge Gas to Blanket	2	45
Opposite Side	Liquid Nitrogen	1	76
	Nitrogen Off-gas	1	33.7
	Cooling Water in / out	2	42.4
	Helium	1	17.2
	Hydrogen	1	17.2
	Waste Gas	1	60.3
	Pressurized Air	1	17.2
	Secondary Containm. Cover Gas	2	105
	Electrical Cables for Process Control and Power Supply	TBD	Feed-Through Box 700 x 500

[a] including thermal insulation where needed

Table 2.1.1.2-4: Requirements for Space outside of the Pit and for External Supply

Type of Requirement	Space / Amount
Electrical Cabinets	4 m x 0.5 m x 2 m
Control Panel, Computer, Table	4 m x 3 m
Evacuation System	TBD
Liquid Nitrogen	TBD
Gases (He, H_2)	TBD
Pressurized Air	TBD
Waste Gas and Waste Disposal System	TBD
Water Detritiation System	TBD
Cooling Water	TBD
Electrical Power Supply (including emergency power and uninterruptible power)	TBD

2.1.1.3 Helium Coolant Subsystem

The cooling subsystem is designed for the European helium-cooled pebble bed (HCPB) test module to be installed in the bottom half of an equatorial test port in ITER, presumably port No. 17. The first wall area of the test module corresponds to the test module frame with an opening of 1.15 m wide by 1.075 m high which is separately cooled by water provided by ITER and is thus not considered here. The cooling subsystem includes the primary helium heat removal loops with all components and the secondary heat removal loops. It is assumed that the secondary water loop subsystem is part of the ITER cooling system providing water flow at low temperature of about 35 °C and low pressure of 0.5 MPa. A further interface to the test module are the purge gas lines to the tritium recovery subsystem which is separate from the cooling stream and with negligible thermal coupling between both subsystems in terms of heat removal. The cooling subsystem together with the helium purification subsystem will be housed in the tritium building at a floor level about 20 m above the test module. Two separate primary heat removal loops of 2 x 50 % heat capacity are foreseen for redundancy purposes in accordance with the DEMO blanket design. Figure 2.1.1.3-1 shows a flow diagram of the primary heat removal loops and the interfaces to ancillary equipment. Figure 2.1.1.3-2 shows the arrangement in the tritium building with an estimated space required of about 700 m^3, excluding the helium purification subsystem.

2.1.1.3.1 Thermal-hydraulic design

Design conditions and assumptions are summarised in Table 2.1.1.3-1. The table contains two columns of values, the nominal values pertaining to the present layout of the first test module, and the design values including margin for uncertainties in control and for later options.

Table 2.1.1.3-1: Nominal and Design Conditions of the Cooling Subsystem for HCPB Blanket Test Module

	Unit	Nominal Value	Design Value
Size of module first wall facing the plasma	(m x m)	1.11 x 1.035	1.11 x 1.035
Surface heat flux	(MW/m^2)	0.25	0.5
Neutron wall loading	(MW/m^2)	1.2	1.2
Total heat to be removed	(MW)	1.9	2.3
Primary coolant		helium	helium
Temperature module in/out	(°C)	250/350	250/450
Pressure	(MPa)	8	9.6
Number of circuits		2	2
Mass flow rate (both circuits)	(kg/s)	3.7	3.7
Secondary coolant		water	water
Temperature heat exchanger in/out	(°C)	35/75	35/75
Pressure	(MPa)	0.5	1.0
Number of circuits		2	2
Mass flow rate (both circuits)	(kg/s)	11.4	13.8

The heat to be removed from the test module amounts to 1.9 MW nominal and 2.3 MW design value, based on 0.25 and 0.5 MW/m^2 of surface heat flux, respectively, nuclear heating due to neutron wall loading of 1.2 MW/m^2 and the first wall area facing plasma of 1.11 m wide by 1.035 m high. Nominal primary helium coolant conditions are 250 °C and 350 °C (later on 250 and 450 °C) at module inlet and outlet, respectively, and 8 MPa of pressure. The total flow rate of the primary helium is 3.7 kg/s. Two identical loops of 2x50 % of heat capacity are foreseen.

The thermal power of the test module is removed to the ITER secondary cooling water with assumed conditions of 35/60 °C at the heat exchanger inlet/outlet, 0.5 to 1.0 MPa and a mass flow rate of 18.2 kg/s (with a maximum of 22 kg/s). Detailed thermal-hydraulic data for loop components are given in 2.1.1.3.2.

2.1.1.3.2 Primary Heat Removal Loops

A flow diagram of one of the two separate cooling loops for the HCPB test module is shown in Fig. 2.1.1.3-1. Main components in each loop are, besides the test module, a heat exchanger, circulator, electrical heater, dust filter, and pipework. The primary loop is directly connected to the helium purification subsystem (see 2.1.1.4) via small pipes taking a bypass flow of about 0.1 % of the main mass flow rate. Further interfaces are shown in the flow diagram to the pressure control subsystem needed for subsystem evacuation, helium supply, and protection against overpressure. Also shown is the minimum required instrumentation for process control. An overview of thermal-hydraulic data such as pressure loss, helium volume, and helium mass inventory in the different components is displayed in Table 2.1.1.3-2. The total helium mass inventory in one loop amounts to 20.8 kg at nominal operating conditions and the overall pressure loss is about 0.27 MPa most of which occurring in the test module proper.

Table 2.1.1.3-2: HCPB Cooling Loop Pressure Loss and Helium Inventory, nominal conditions

Component	Press. loss (Pa)	Volume per Loop (m^3)	Mass per Loop (kg)
Hot leg pipework	13100	0.94	5.66
Cold leg pipework	10800	0.94	6.92
Main pipe elbows	14000	incl. in pipes	incl. In pipes
Bypass to heat exchanger	-	0.038	0.252
Valves	6000	0.007	0.046
Heat exchanger	4000	0.061	0.377
Circulator	-	0.025	0.184
Electrical heater	500	0.042	0.275
Dust filter	30000	0.2	1.206
Buffer tank	-	0.255	5.326
Test module	190000	0.009	0.6
TOTALS	268400	2.6	20.85

The main primary loop components are described below with a summary of the main dimensions displayed in Table 2.1.1.3-4.

Blanket Test Module: The HCPB blanket test module (BTM) is described in section 2.1.1.1. Similar to the DEMO blanket design the volume fraction of the primary helium relative to the total module volume is about 16 %. With this ratio and a test module volume of 1.11 x 1.035 x 1 m^3 the helium volume amounts to 0.18 m^3, i.e., 0.09 m^3 pertaining to each loop. The pressure loss is evaluated as 0.19 MPa.

Heat Exchanger: A first layout has been performed assuming a straight tube bundle heat exchanger (HX) with high pressure helium flowing inside the tubes and low pressure water flowing outside. The required tube bundle data along with the primary and secondary loop flow parameters are listed in Table 2.1.1.3-3, again showing the nominal values for the first test module and the design values considering anticipated options with sufficient margin. (The design values are regarded as maximum values for component layout and are, thus, not consistently pertaining to a certain operating mode.) For the design size the helium volume in one HX would be 0.06 m^3 (0.02 m^3 in the tubes and 0.04 m^3 in the end domes). Alternatively a U-tube HX could be envisaged which would not significantly alter the design data.

Table 2.1.1.3-3: Heat Exchanger Layout for HCPB Test Module Cooling Subsystem

	Unit	Nominal Value	Design Value
Type		Straight tube bundle	
Number of heat exchangers (HX)		2	2
Heat to be removed per HX	(MW)	0.95	1.15
HX tube size (outer/inner diameter)	(mm)	18/14	18/14
Number of tubes per HX		96	96
Tube bundle diameter x length	(m)	0.27 x 0.9	0.27 x 1.2
Overall HX dimensions diameter x length	(m)	0.35 x 1.8	0.35 x 2.2
Primary coolant		helium inside tube	
Temperature in/out	(°C)	250/350	250/450
Pressure	(MPa)	8	9.6
Mass flow rate per HX	(kg/s)	1.9	<1.9
Flow velocity in/out	(m/s)	20.8/17.5	14.6/10.6
Heat transfer coefficient in/out	($W/(m^2K)$)	2500/2300	1700/1500
Secondary coolant		water outside tube	
Temperature in/out	(°C)	35/75	35/75
Pressure	(MPa)	0.5	1.0
Mass flow rate per HX	(kg/s)	5.7	6.9
Flow velocity	(m/s)	0.16	0.2
Heat transfer coefficient in/out	($W/(m^2K)$)	1500/1900	1800/2200

Circulator: One variable speed helium circulator will be installed in the cold leg of each primary loop operating at 250 °C in normal operation. Including some margin during heating and baking phases the design temperature is set to 300 °C and the design pressure to 9.6 MPa (20 % above nominal for overpressure control). An encapsulated type circulator with vertical shaft is envisaged where the type of bearing (gas lubricated or magnetic) has still to be decided upon. The design specification for the circulator is as follows: temperature 300 °C, pressure 9.6 MPa, mass flow rate 1.9 kg/s at a pumping head of 0.27 MPa at 80 % of maximum speed and at 250 °C inlet temperature, speed variation max/min of at least 4. Under these conditions the electric power of the drive motor would be 100 kW. The helium volume contained in the circulator is estimated as 0.025 m^3 and the overall dimensions of the circulator and drive unit are expected to be 0.5 m diameter times 1.8 m height.

Electrical Heater: This component is needed for baking the test module first wall at 240 °C and for heating the whole cooling subsystem including the test module to operating temperatures after maintenance or repair periods. It is positioned in a bypass to the HX, assuming that the HX is isolated during heating phases and the circulator is operating at reduced speed. The required electrical power has been estimated to 100 kW. This would enable to heat the whole subsystem at a rate of about 10 to 20 °C per hour. The main dimensions of the helium volume are 0.3 m diameter times 1.2 m height, approximately half of which being occupied by the heating rods. This yields a helium volume of 0.042 m^3. The estimated pressure loss is small, ca. 500 Pa. The overall dimensions are assumed to be 0.5 m diameter times 1.7 m height.

Dust Filter: A filter unit is installed in the hot leg of the main loop, accumulating residual dust and particles from fabrication, and erosion particles down to a size of typically 10^{-6} m. To some extent even the much smaller sputter products evolving in the neutron field in the BTM may be trapped which otherwise are expected to be deposited mainly in the heat exchanger and at pipe walls. The array of small-diameter filter tubes, or plates in a grid format, forms a removable filter cartridge of 0.3 m diameter and 3 m length, giving a helium volume of approximately 0.2 m^3. The pressure loss is expected to be less than 0.03 MPa.

Pipework: For the main pipework, i.e., hot leg and cold leg, an outer diameter of 168.3 mm and a wall thickness of 10 mm have been chosen for the part external to the cryostat. Inside the cryostat smaller pipes are foreseen (114.3 mm outer diameter, 8 mm wall thickness) to limit the size of pipe penetrations in the cryostat to 130 mm. This results in a flow velocity in normal operation of between 15 and 41 m/s and in small pressure losses. The pipe length is determined by space allocation in ITER, leading first to 20 m of horizontal (radial) run (10 m of which are assumed to be of the small size), followed by a 20 m vertical part, followed by 10 m of horizontal (tangential to the torus) run and completed by about 10 m of pipework connecting the components in the tritium building. This sums up to 60 m of pipework each for cold and hot legs. The number of elbows is assumed to be 16 per leg (6 per leg of the small diameter). Overall the pipework contributes with 14 % to the pressure losses in the loop (Table 2.1.1.3-2).

The flow rate during baking and heating will be reduced by a factor of about 4 compared to the rated mass flow rate. Thus the bypass to the HX can be smaller than the main pipework. The outer diameter has been set accordingly to 82.5 mm. In normal operation the bypass is supposed to be almost closed during burn times and open during dwell times (see 2.1.1.3.6).

Valves: The number of valves in the main loops has been minimised to avoid inadvertent closure which would mean loss of heat sink in the affected loop. Hence, only one valve is installed before the HX in the main loop and another one in the bypass line before the electrical heater. These two valves are needed for temperature control in normal cyclic operation and must be position controlled.

Pressure Control Subsystem: This is a combination of equipment needed for cooling subsystem evacuation, helium supply, pressure control, and overpressure protection. The components are conventional and of relatively small size, except for the storage and dump tanks. The subsystem is essentially isolated from the main cooling loops during normal operation, however in case of a pressure drop, like a loss of coolant accident, the buffer tank will discharge into the main loop.

The evacuation subsystem is needed for the first start-up as well as after repair of the main cooling loops and consist of dust filter, booster pump, cooler, vacuum pump, oil filter, and instrumentation and control equipment. The exhaust is released to the air detritiation subsystem of the tritium building.

The helium supply and storage subsystem consists of storage tanks, buffer tank, compressor, and pressure regulators. The battery of storage tanks is sized as to take the whole helium inventory of the loop excluding the one in the buffer tank (about 16 kg) at about 50 °C, 14 MPa, resulting in a tank volume of 0.75 m^3. This can be achieved by, e.g., two tanks of 0.5 m diameter, 2.3 m long with 20 % reserve. A multi-stage compressor and cooler will be needed to load the storage tanks for emptying the main loops.

Pressure control in the main loops during normal operation is achieved in the following way: The storage tanks are kept at low pressure ($\approx$1.5 MPa) so that the main loops can discharge to the storage tanks via the pressure regulator if the set point "pressure high" is reached. The buffer tank, on the other hand, has to compensate for the loop pressure if the set point "pressure low" is reached. As it discharges to the loop it will be recharged by a compressor from the storage tanks. A buffer tank volume of 10 % of the loop volume is chosen, that is about 0.26 m^3, and a maximum operating pressure of 14 MPa.

The overpressure protection of the cooling loops consists of two redundant safety valves or a combination of one safety valve plus a burst disc. The safety valves discharge into a dump tank which is kept at controlled low pressure (near atmospheric) during normal operation. This avoids releasing contaminated helium into the building and to completely depressurise the main loop in case the valve would fail to close. The dump tank is sized for the event that the primary loop was inadvertently pressurised to the nominal pressure (8 MPa) at room temperature and the whole subsystem was subsequently heated up to 250 °C, the nominal operating inlet temperature. If in this case the pressure regulator would fail to open, the safety valve would respond. In order to limit then the pressure to 8 MPa requires a dump volume of about 70 % of the loop volume, i.e., 1.82 m^3. This is about twice to what is needed as storage volume and can be achieved by, e.g., four tanks of 0.5 m diameter, 2.3 m length.

Table 2.1.1.3-4: Primary Heat Removal Loop Components Main Dimensions (for 1 of 2 loops)

Component	Number per loop	Diameter (m)	Length (m)
Helium/water heat exchanger	1	0.35	2.2
Circulator and drive (vertical shaft)	1	0.5	1.8
Electrical heater	1	0.5	1.7
Dust filter (without shield)	1	0.4	3.5
Main pipework (hot leg and cold leg)	1	0.168	120
Helium storage tanks	2	0.5	2.3
Helium dump tanks	4	0.5	2.3
Buffer tank for pressure control	1	0.5	1.3
Elevation of HX above test module	-	-	22

2.1.1.3.3 Secondary Heat Removal Loops

The secondary heat removal subsystem is considered as part of the ITER cooling system. For the layout a constant water flow with nominal inlet temperature of 35 °C at the secondary side of the heat exchanger (see 2.1.1.3.1) and a flow rate of ~7 kg/s per HX are assumed, leading to pipe dimensions of 50 mm outer diameter, 3 mm wall thickness, at a velocity of ~5 m/s. The outlet temperature will then vary according to the burn and dwell cycles between 75 °C and 35 °C. Flow, pressure, and temperature monitoring are needed. No significant migration of tritium from the primary coolant to the secondary side is expected.

2.1.1.3.4 Maintenance/Remote Handling

Activation of cooling subsystem components installed in the tritium building is expected to be generally low allowing controlled personnel access. An exemption may be the dust filters which require extra shielding. In-service inspection such as examination of selected welds by different methods (visual, eddy current, ultrasonic), inspection of circulator internals, functional tests of valves, leak tightness of heat exchangers etc. occur during test module changeout or during planned or unplanned machine shutdown periods. Remote handling is envisaged for connection and disconnection of the BTM. There are isolation flanges with welding lips connecting the module with the rest of the cooling subsystem. One or two such flanges for each of the four main pipes have to be remotely connected (or disconnected in case of test module replacement). One set of flanges will be placed close to the BTM, e.g., immediately behind the shield. The other set of isolation flanges (if needed) will preferably be located in the wedge-shaped pit housing the tritium removal subsystem. Remote handling is also envisaged for replacement of the dust filter insert. In the case of any defects in heat exchangers, electrical heaters, or dust filter casings replacement of the whole component may be more appropriate than a repair.

2.1.1.3.5 Assembly

All components of the heat removal subsystem such as heat exchangers, circulators, electrical heaters, dust filters, tanks, and valves will be preassembled at the factory and delivered to the site as functional units. Connection of the components will be performed on site by conventional means. An exception is the installation of the test module. It will be brought in place by the aid of a special transporter being aligned with the test port. It is equipped with all tools needed for positioning, aligning, locking, connecting the module in the 8 m long tunnel of the test port, and cooling. The other large components of the cooling subsystem installed in the tritium building require a crane with a load capacity of about 2 tons (the heaviest component, the HX, has a weight of about 800 kg). Field welded joints will be subjected to surface and/or volumetric inspection, followed by pressure and leak tests. Thermal insulation will be installed after leak testing of the loops.

2.1.1.3.6 System Start-up, Control, and Shutdown

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

- Subsystem evacuation to $<10^2$ Pa within about 24 hours
- Subsystem floodling with helium and pressurisation to approximately 4.2 MPa at 25 °C
- Heating to 300 °C within 10 - 20 hours at reduced circulator speed ($\approx$25 % of nominal), HX closed
- Establishing secondary cooling water flow in HX
- Establishing temperature control at desired baking temperature (about 240 °C at circulator outlet) by controlling the flow through HX, heater power still on
- Keeping subsystem stable for baking period
- Bringing circulator to nominal speed
- Establishing temperature and pressure control at stand-by level: 250 °C, 8 $\pm$ 0.3 MPa at circulator outlet, heater power on. Subsystem is now ready for operation.

For cooling subsystem control in normal operation the typical ITER load cycle is envisaged, i.e., pulse duration of 1000 s and repetition time of 2200 s with specified power ramp-up and ramp-down. The power removed by the cooling loops thus varies between about 2 MW and 0.2 MW, the latter coming mainly from circulator and decay heat (or 0.4 MW if the heaters are kept on), that is a ratio of 10:1 (respectively 5:1). Because of the given large mean temperature difference in the HX between the primary and secondary side the heat removed in the HX can most effectively be influenced by primary helium flow control. Hence, the following preliminary subsystem control scheme is proposed for pulsed operation, the details of which, considering the thermal inertia of all components and the heat transfer, have still to be analysed.

- The principal objective is to keep the test module inlet temperature at 250 °C.

- The secondary cooling water inlet temperature should be kept at 35 °C.
- The circulator should be operated at rated speed.
- The electrical heaters are turned off.
- Flow partition through the HX and heater bypass is controlled as to maintain the inlet temperature as close as possible to 250 °C.

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

Complete shutdown of the HCPB-BTM including removal or replacement of the BTM will be accomplished by the aid of the transporter. In this case one of the cooling loops will be emptied and disconnected, while the other loop maintains decay heat removal. The disconnected piping of the BTM will then be connected to the transporter's cooling system which overtakes decay heat removal. Subsequently, the second cooling loop will be treated in the same way as the first one, before the BTM is removed. It is assumed that during the whole procedure an inert gas atmosphere with a pressure of about 0.1 MPa and a temperature of less than 200 °C be introduced into the vacuum vessel. Under these conditions it is expected from analyses performed for DEMO that the decay heat of the BTM can be dissipated to the surrounding without extra active cooling, constituting a back-up means to the transporter's temporary decay heat removal mission.

2.1.1.3.7 Materials

All of the piping and components in the primary cooling subsystem will be constructed of austenitic steel. The test module will be made of ferritic steel (section 2.1.1.1.7) with the interface being at the isolation flanges next to the test module. All of the piping and components will be equipped with 5 to 10 cm of a mineral thermal insulation.

2.1.1.3.8 Safety

The main safety concerns with the cooling subsystem are the loss of coolant accident with regard to tritium and activation products release, and the loss of flow or loss of heat sink accident in both loops with respect to decay heat removal. They are assessed in section 2.1.1.1.8.

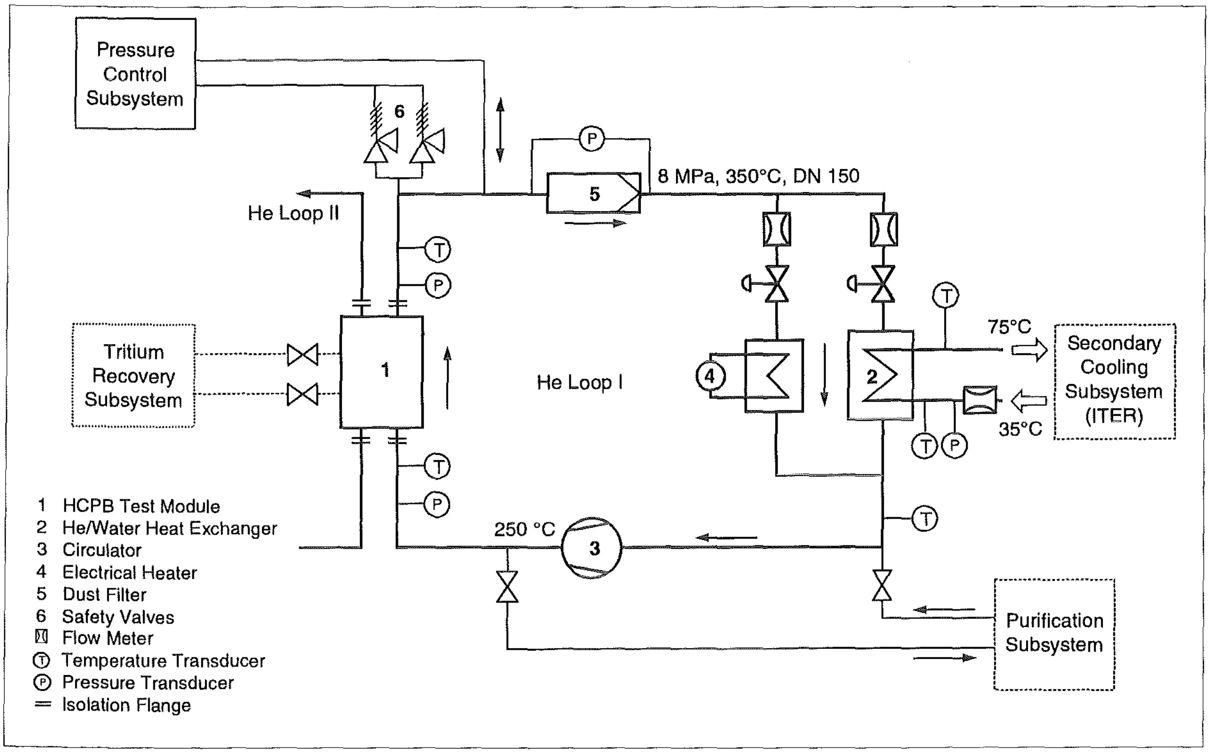


Figure 2.1.1.3-1: Helium Coolant Subsystem Flow Diagram for the HCPB Blanket Test Module

Height required ca. 7m

ELEVATION VIEW tbd

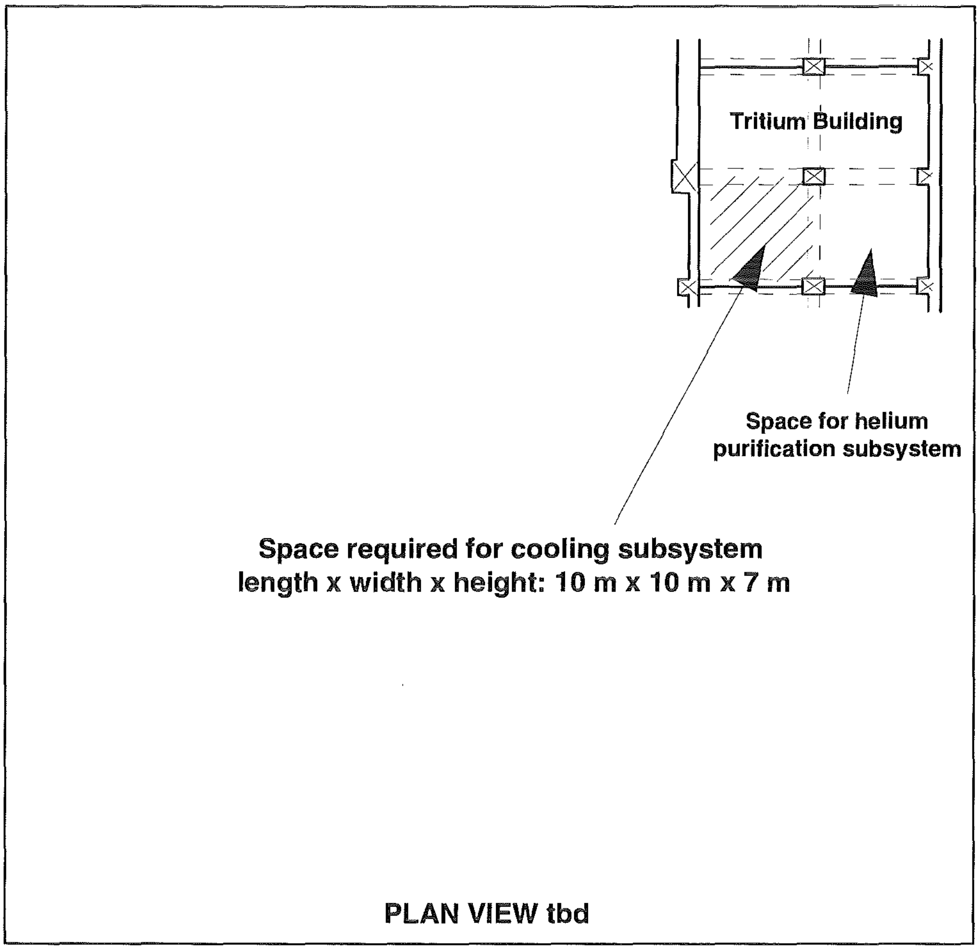


Figure 2.1.1.3-2: Helium Cooling Subsystem Arrangement of HCPB-BTM in Tritium Building

2.1.1.4 Coolant Purification Subsystem

For each of the two coolant systems of the blanket test module one purification system is provided to purify a fraction of 0.1% of the helium coolant stream, i.e.

- to extract hydrogen isotopes as well as solid, liquid or gaseous impurities from the main coolant, and
- to remove condensed water that may be entrained in the cooling gas due to leakages or failures of the heat exchanger tubes.

The main design data of the purification systems are given in Table 2.1.1.4-1, a flow sheet is shown in Figure 2.1.1.4-1.

The thermal power removed in each coolant system is 0.95 MW, the corresponding mass flow rate is 1.85 kg He/sec. The tritium content in the coolant is caused by permeation from the First Wall and from the Purge Gas System. In analogy to calculations performed for DEMO [2.1.1.4-1], it has been assumed that an average value for the tritium permeation rate into the coolant is

$$\mathbf{m_p} = 0.05 \cdot 10^{-3} \text{ mole T / h}$$

As 0.1 % of the coolant is continuously purified with an efficiency of 95 %, the tritium concentration in the coolant will increase to an equilibrium concentration c_e which is reached when

Removal rate $\mathbf{m_r} = 0.001 \cdot 0.95 \cdot \mathbf{m_T}$ = Permeation rate $\mathbf{m_p} = 0.05 \cdot 10^{-3}$ mole T/h

with $\mathbf{m_T}$ = mass flow rate of tritium in the coolant loop under equilibrium conditions.

This leads to $\mathbf{m_T} = 0.05$ mole T/h

and $c_e = \mathbf{m_T}/\mathbf{m_{He}} = 0.03 \cdot 10^{-6}$ ($\mathbf{m_{He}} = 1.665 \cdot 10^{6}$ mole/h)

This is equivalent to a HT partial pressure of 0.24 Pa. The concentration of H_2 in the coolant is calculated from the HT partial pressure by taking into account the much higher concentration of H_2 in the purge gas (about a factor 600 in comparison to HT), the fractional contribution of the tritium permeation from the purge gas system (about 25% of the total permeation), and the higher permeation rate of H in comparison to T (factor 1.73). The resulting partial pressure of H_2 is 62 Pa.

The partial pressures and extraction rates of HTO / H_2O given in Table 2.1.1.4-1 have been calculated assuming a leakage of 3 g water per day from the heat exchanger into the coolant loop. The corresponding values for N_2 were obtained for a concentration of 1 ppm in the coolant.

Table 2.1.1.4-1: Main Design Data for each of the Coolant Purification Systems

He Mass Flow in Purification System		1.85 g/s = 37.3 Nm^3/h
Pressure		8 MPa
Total Amount of He Coolant		23 kg
Partial Pressures [a]	p (H_2)	62 Pa
	p (HT)	0.24 Pa
	p (DT)	0.24 Pa
	p (Q_2O)	35 Pa
	p (N_2)	8 Pa
Extraction Rates	Q_2O	0.47 mole / day [b]
	N_2	0.04 mole / day
Tritium Extraction Efficiency		≥ 95 %
Temperature of the Coolant at Coolant Purification Inlet / Outlet		250 °C / 50 °C

[a] under equilibrium conditions (obtained after about 24 hours) in the coolant
[b] due to catalytic oxidation, H_2, HD, and HT are extracted as Q_2O (Q = H, D, T)

System Description (Fig. 2.1.1.4-1)

The gas stream entering the coolant purification system downstream of the coolant loop circulation pump is first sent through a water separator to remove condensed water that may be present as a consequence of larger leakages than anticipated above. The gas is then warmed up to 450 °C by an electrical heater and transferred to an oxidizer unit containing a precious metal catalyst (Pd or Pt on alumina). An over-stoichiometric amount of oxygen is added to obtain a quantitative conversion of Q_2 to Q_2O (Q = H, D, T). The high temperature of the gas is favorable for the kinetics of the oxidation process.

Now the gas temperature is reduced to room temperature by a water cooler. The remaining humidity is frozen out in a cold trap operated at -100°C. The amount of water extracted under the conditions described in Table 2.1.1.4-1 is 8.5 g/day.

Finally, the gas is passed through a recuperator and then to a 5A molecular sieve bed cooled with liquid nitrogen (LN_2) to adsorb gaseous impurities like N_2; any hydrogen isotopes that have not been oxidized are also adsorbed. The inlet and the outlet side of the bed are equipped with mechanical filters to prevent a carry-over of particulate material during normal operation (downward flow) and regeneration (upward flow).

Fig.2.1.1.4-1 Flow sheet of the Coolant Purification System

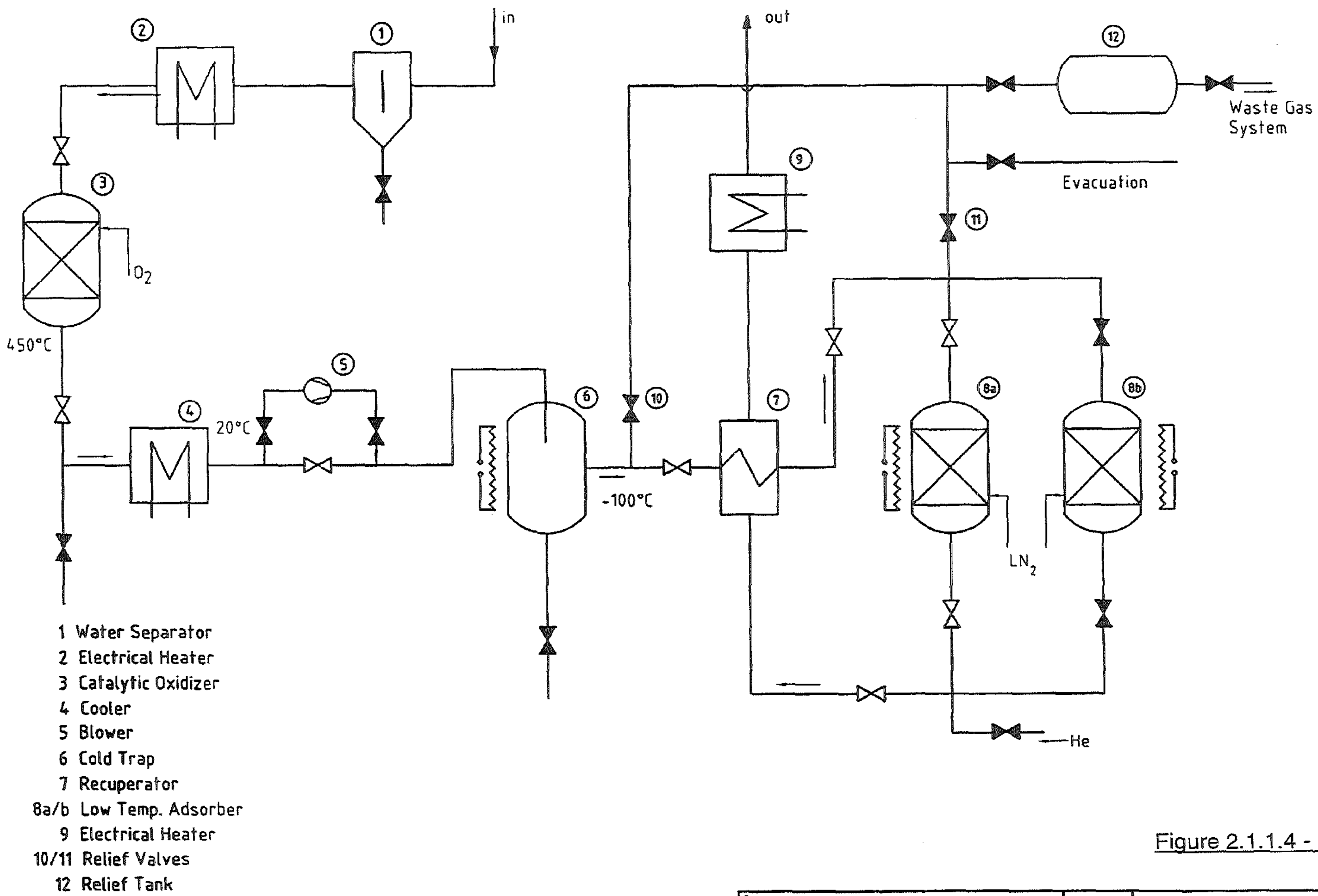


Figure 2.1.1.4 - 1:

Benennung			
Coolant Purification System			Forschungszentrum Karlsruhe GmbH Hauptabteilung Ingenieurtechnik
	Tag	Name	Zeichnungs-Nr
Bearb.	10.07.96	Ballschun	IT-OWH-02-001
Gepr.			Blatt 1
Maßstab	1:1		Ersatz für / Ersetzt durch

Fig. 2.1.1.4-1 Flow sheet of the Coolant Purification Subsystem

The second bed provides additional adsorption capacity; it may be used when the first bed has not been unloaded or regenerated.

The pure helium is carried back through the recuperator, further warmed up by an electrical heater, and then returned into the main coolant loop upstream of the circulation pump. Thus, it should be possible to operate the purification system without an additional compressor or circulation pump. Nevertheless, a corresponding pump (No.5) will be available on demand. The total inventory in the main coolant remains below 0.1 mol Q_2 and 0.05 mol Q_2O under the conditions described in Table 2.1.1.4-1. It will be sufficient, therefore, to continue the operation of the purification system for about 12 hours after reactor shutdown to arrive at a reasonably low concentration of hydrogen isotopes in the coolant.

Regeneration

Some components must be regenerated before their retention capacity has been reached. The particulate filters will be transferred to the waste disposal system after exchange.

The cold trap loaded with ice is depressurized (via relief valve 10) and warmed up to room temperature to liquefy the water which is then drained into a water container and sent to the Water Detritiation System.

The adsorber beds are first depressurized like the cold trap (via relief valve 11). During a normal unloading operation they are warmed up to room temperature, and the desorbing impurities are sent to the Waste Gas System. A complete regeneration is achieved by heating to 300 °C and purging with clean helium.

Analytical Tools

The tritium extraction is controlled by continuous measurement of the tritium concentration at several points of the loop. 4 ionization chambers are used for this reason:

No.1: At the loop inlet upstream of the electrical heater,

No.2: Downstream of the cold trap (information about satisfactory function of the oxidizer and the cold trap),

No.3: At the loop outlet (under proper conditions, the reading should be the same as of No.2),

No.4: Downstream of valve 11 (to monitor the effluent gases).

In addition, the composition of the coolant gas is analyzed with the help of a gas chromatograph by taking gas samples at the inlet and the outlet of the loop.

Space Requirements

The size of the main components has been estimated and listed in Table 2.1.1.4-2. A first proposal for the geometrical arrangement is given in Figure 2.1.1.4.-2. The integral space requirement of the facility is about 50 m^3. Additional space of at least 15 m^2 in front of the facility will be needed for installation, manual operations, repair, etc. This number does not include the space for a control station and for electrical cabinets. It is also expected that supply and diposal facilities are available. Table 2.1.1.4-3 summarizes these requirements.

References :

[2.1.1.4-1] L. Berardinucci and M. Dalle Donne, „Tritium control in the European Helium Cooled Pebble Bed Blanket“, 19th SOFT, Sep. 16-20 (1996), Lisbon, Portugal.

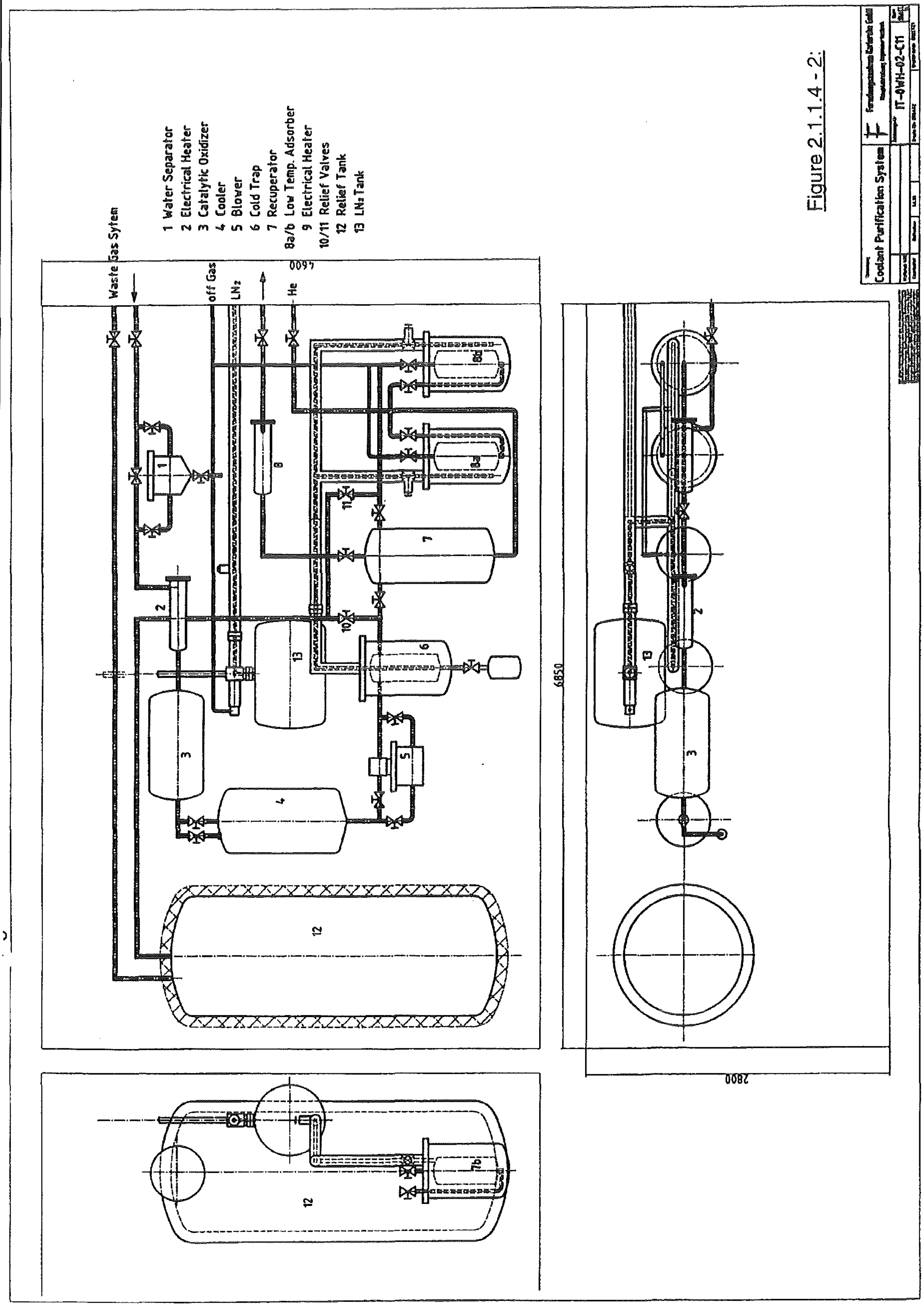


Fig.2.1.1.4-2 Layout of the Coolant Purification Subsystem

Table 2.1.1.4-2: Size of the Main Components

No.	Component	Size [a]	
1	Water Separator	D: 200	H: 320
2	Electrical Heater	D: 100	H: 900
3	Catalytic Oxidizer	D: 500	H: 1000
4	Cooler	D: 600	H: 1200
6	Cold Trap	D: 500	H: 800
7	Recuperator	D: 500	H: 1200
8 a/b	Low Temp. Adsorber	D: 500	H: 800
9	Electrical Heater	D: 100	H: 900
10 /11	Relief Valves		
12	Relief Tank	D: 1300	H: 3200

[a] D = Diameter, H = Height; all dimensions in mm and including thermal insulation where necessary

Table 2.1.1.4-3: Requirements for Additional Space and for External Supply

Type of Requirement	Space / Amount
Electrical Cabinets	4 m x 0.5 m x 2 m
Control Panel, Computer, Table	4 m x 3 m
Evacuation System	TBD
Liquid Nitrogen	TBD
Gases (He, O_2)	TBD
Pressurized air for pneumatic valves	TBD
Waste Gas and Waste Disposal System	TBD
Water Detritiation System	TBD
Cooling Water	TBD
Electrical Power Supply (including emergency power and uninterruptible power)	TBD

2.1.1.5 Test Blanket Remote Handling Subsystem

All equipment to be used in the horizontal ports should be designed for radial installation and removal of components through the port extensions. Since the equipment will be inside the Bioshield and will be highly activated after reactor operation, it will be necessary to use remote handling systems for all operations within the Bioshield boundaries. This requirement will apply to the Test Blanket Subsystem. The design of the remote handling system of the test blanket modules is dependent upon the piping system layout within the port extension. One of the project recommendations is to minimize the amount of remote operations inside the port extension. A concept was developed which combines the BTM's, the Shield, the related plumbing and the vacuum vessel closure plate as one superassembly, („Test Blanket Assembly" or TBA) as already shown in section 2.1.1.1. This allows full functional testing of the assembly prior to installation within the port. As a result, the remote handling system was adapted to handle this assembly. Figure 2.1.1.5-1 shows a schematic of the TBA inside the horizontal port. The overall length of this assembly is about 5 meters which is well within the allowed limits of the transporter of 8 meters. The weight of the assembly is in the range of 30-35 tonnes. The center of gravity of the assembly is anticipated to be about at the geometrical center of the Test Blanket Asssembly.

Figure 2.1.1.5-2 shows coolant pipes being routed through the vacuum vessel closure plate, then through the cryostat wall and bioshield wall. The advantage of such layout is to minimize the amount of remote operations required to remove and install the blanket modules, and to eliminate remote operations inside the VV port. This will also reduce the amount of time required to remove and install the TBA.

The remote handling system for the Blanket Test Modules will take full advantage of the equipment designed by the JCT to minimize duplication of efforts and to standardize system operations. As a result, the current design of the remote handling system will utilize a series of transporters each is designed to perform a certain task. The transporter is the standard JCT design with overall dimensions of 8 m long, 3.8 m wide and 5 m high. All operations that are identical to other ITER operations will use the same ITER system to perform, such as removing the bioshield plugs and the cryostat closure plate. As a result the piping system for the test blanket modules will be designed to meet the requirements of this tool, such as minimum bend radii. Operations that are specific to the test blanket system will be integrated into the overall system design.

Removing and installing a test blanket assembly will involve a number of steps. Prior to removing the blanket test modules, procedures will be established to prepare the modules for removal such as breaking the vacuum in the vessel, releesing the coolant helium to the helium storage, draining the coolant water from the BTM shielding and frame and purging the system to reduce the amount of residual tritium inside the modules. The next step would be to disconnect the tritium extraction system from the blanket modules and clear the way for the blanket removal operations to start. A top level procedure of the remote handling process is outlined below. Using this procedure will help in identifying the special equipment needed for the test blanket handling system. This procedure is based on the assumption that the tritium extraction processing system is housed in the pit adjacent to the test module (see Section 2.1.1.2). Also this procedure starts after an inert gas atmosphere (nitrogen) has been introduced to the vacuum vessel.

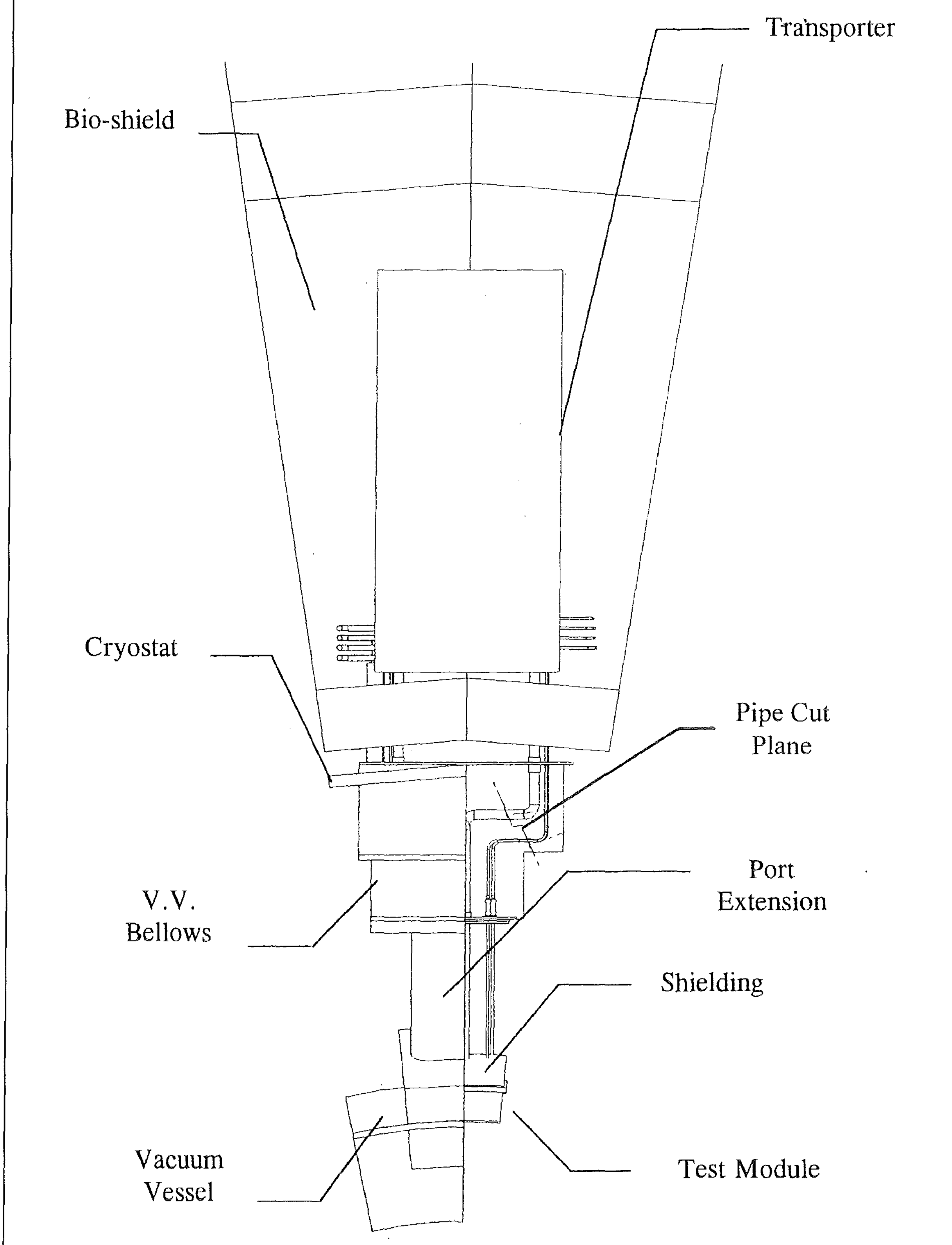


Figure 2.1.1.5-2 Proposed Plumbing Routing (Plan view).

Basic TBA Replacement Concept [2.1.1.5-1]

The initial and final steps of TBA replacement procedures are performed by hands-on techniques using conventional radiation and contamination control procedures. This involves the handling of the bioshield plug and the removal and installation of the TBA service lines feeding through the Cryostat secondary closure plate. The secondary closure plate is replaced with a special contamination door used in the RH portions of the procedure.

The primary remote steps to remove the TBA include 1) the disassembly of the primary closure plate access door for the insertion of RH tools to the TBA, 2) the unbolting of the TBA from the support frame, 3) the mechanical attachment of the shield to the primary closure plate, 4) the disassembly of primary closure plate and 5) the withdrawal of the TBA into the RH transfer cask. These operations are performed using RH systems operating from within a transfer cask that docks directly to the Cryostat port flange.

The RH transfer cask includes a double door system for contamination control during these operations. When the cask is not attached to the port, one door remains in place at the port while the other seals the cask opening. After docking and sealing the cask to the port by bolting from within the cask, the two doors are combined (bolted together) to keep exterior surfaces free of contamination. The combined door assembly is then moved into the cask to gain access to the port. After removal of the TBA, the double doors are closed, are bolted to the cask and the port, and are separated (unbolted) from one another. The transfer cask is then undocked from the port and the assembly is delivered to the Hot Cell. The same cask and RH systems are used to replace the TBA with either a spare unit or the original unit following its repair.

TBA replacement procedure [2.1.1.5-1]

The basic removal steps for the TBA are listed below. Installation is performed by reverse order.

<u>Hands-on preparation</u>

1. Remove the TBA systems and service lines outboard the bioshieldplug and remove the bioshield plug to gain access to the port interface at the Cryostat docking flange and secondary closure plate.

2. Using contamination control barriers (e.g., glove box), disconnect and remove service lines penetrating the secondary closure plate. (Note: service lines should be disconnected close to the TBA's primary closure plate in order to minimize port assembly length and maximize access.)

3. Replace the secondary closure plate with the port maintenance door (combines with RH cask door). A contamination control liner may be installed inside the port secondary containment space to minimize contamination of this volume.

4. Drain the TBA of cooling water and internally cut (inlet and outlet) cooling pipes close to the primary closure plate.

Note: Draining cooling water from the TBA will increase radiation levels at the port interface and may preclude personnel access to the port interface without the addition of temporary shielding.

Remote steps for TBA removal

5. Dock and seal the RH transfer cask to the Cryostat port flange. Combine and seal the cask door with the maintenance door, and move the door assembly into the cask. Seals are leak tested between the cask and the port before opening the doors.

6. Unbolt the access cover in the primary closure plate (M36 x TBD bolts) and cut the lip-seal weld - performed by dexterous manipulator and RH tools.

7. Engage the TBA support rollers with rails located on port wall or floor (rollers cannot be engaged during machine operation due to relative motion requirements between Shield and VV port).

8. Deploy the in-cask rail and RH tools to unbolt the TBA at the Support Frame (M24 x 64 bolts-TBD) and remove tool.

9. Engage bracing system to attach closure plate and Shield.

10. Unbolt the primary closure plate (M36 x 40 bolts) and cut the lip-seal weld - performed by dexterous manipulator and RH tools.

11. Withdraw the TBA from the port into the cask.

Note: Removing the TBA will increase radiation levels at the port substantially and prevent personnel access to the vicinity of the port. The port area will be shielded by installing portable shield blocks or doors at the pit wall openings in this location so access in adjacent work areas of the pit are possible.

12. Close the double doors, separate the 2 doors and undock the RH cask from the port. Leak checks are performed to verify tightness of seals. The TBA is then transferred to the hot cell.

Remarks

This remote handling process is based on removing the full test blanket assembly as a whole unit. Removing an individual submodule within the port is not possible under this scenario. Implementing such a capability into the remote handling system will further complicate the operation and will increase the time required for removal and installation. Since one of the requirements of the test program is to perform maintenance operations during scheduled reactor shutdowns, every effort is taken to minimize the number of operations required for blanket removal and installation. As noted earlier, the test blanket remote handling system will take full advantage of the ITER designed equipment. The vacuum vessel closure plate used on the test ports will be a modification of the standard vacuum vessel closure plate and will have an access hatch built into it. This hatch is designed to allow access into the port extension prior to removing the main vacuum vessel plug.

The outline of the remote handling process mentions some special operations that are specific to the test blanket modules. This operations will utilize special tools designed specifically to perform this task. Description and function of the special equipment will be presented in the Section 2.1.1.5.1.

The test port extension will be designed with built in tracks on the bottom floor to allow the bolting tool and the blanket removal vehicle to be deployed inside the port extension. Additional tracks will also be built into the port extension walls (side walls, floor and ceiling) near the blanket modules to allow the bolting tool to move along the walls and be able to reach all the fasteners of the blanket module.

The remote bolting tool will utilize the tracks in the port extension to reach around the shield and bolt/unbolt the shield from the support frame. The blanket module frame is equipped with special teeth designed to accurately locate it at the support frame and to also support the blanket modules during installation and removal. Once all bolts are removed, the blanket support vehicle is introduce into the port extension. This vehicle is designed to connect to the blanket modules at the shield, and to connect to the vacuum vessel closure plate. Since the only connection between the blanket modules and the vacuum vessel closure plate is the piping, additional support is needed between the two components to insure stability during transportation and to minimize damage to the piping system. This vehicle will serve as the structural support during installation and removal of the blanket modules. It will utilize the tracks inside the port extension as well as the temporary track between the port door and the transporter to move the blanket modules into the transporter.

References

[2.1.1.5-1] R. Haange, T. Burgess, Comments for Sept. 2, DDD overview, 17 October 1996.

2.1.1.5.1 Mechanical Design

The remote handling system for the test blanket modules will consist of a number of components designed to perform certain tasks. A number of the tasks required for the removal and installation of the test blanket modules are identical to ITER tasks, such as bioshield plug installation, cryostat plug interface and the internal pipe cutting and welding operation. For those tasks, ITER equipment will be used. Special equipment such as the test blanket support vehicle, the remote bolting tool and test module assembly transporter will be designed by the test blanket group with interface data inputs from ITER. A brief description of each of the special equipment is included below, although concept drawings will be provided as soon as they become available.

Test blanket remote handling transporter - This transporter will be based on the standard transporter design as it is developed by ITER. This transporter will be designed to handle a number of tasks. As noted in the remote handling process procedure in the previous section, the transporter should be equipped with special tools to perform a number of specific operations. A manipulator is needed to plug and unplug the power and diagnostics cable bundles without damaging them. This manipulator will serve other tasks by exchanging the end effector tools to fit a specific task. Some of those tools include a fastening tool to handle the vacuum vessel plug bolts. Another tool is needed to cut and weld the lip seal weld of the vacuum vessel plug and the access hatch. Deployment of the temporary tracks between the cryostat door and the vacuum vessel is also handled with this manipulator. Other tools include inspection equipment and possibly viewing equipment such as a camera to perform remote visual inspection.

The transporter is also designed to contain the test blanket assembly. Internal tracks are installed to allow the blanket support vehicle to travel into and out of this transporter. Room should be provided within the transporter to store the temporary tracks when not in use. Monitoring equipment designed to monitor the status of the test blanket modules during transport to/from the test port should be built into the transporter with capabilities to transmit important or emergency status data to the control room. Emergency recovery operations should be designed and built into this transporter to enable it to recover from certain emergency conditions without interrupting the operation of ITER. Other equipment stored inside the transporter include the remote bolting tool and the blanket support vehicle. Active cooling inside the transporter for the test blanket module may is required.

Remote bolting tool - As noted in the previous section, the blanket assembly may consist of the blanket module, shielding, piping and the vacuum vessel door. While this particular arrangement has the advantage of eliminating remote operations inside the port extension, it creates an accessibility problem for fastening the blanket module to the back plate. To gain access to the back plate while the blanket assembly is installed, an access hatch is built into the vacuum vessel door. This access hatch is bolted and seal welded to the vacuum vessel door. A remote manipulator is used to cut the seal weld and remove the fasteners. The access hatch is then removed into the transporter. A remote bolting tool designed to fit through this access hatch is then dispatched into the port extension. The main task for this tool is to be able to reach all the blanket module fasteners and either install or remove them. This is achieved by having a set of tracks built into the port structure. Radial tracks on the floor of the port extension are used for the tool deployment. Once it reaches a certain distance away from the blanket module, the bolting tool engages a set of crossing track that allows it to move toroidally until it reaches the side walls of the port extension. Tracks along the side walls are designed to allow the bolting tool to move up and down allowing it access to the side wall bolts. Tracks in the ceiling of the port extension are designed to engage the tool and allow it to hang as it traverses across the top to reach the upper set of fasteners. In order to provide positive traction for the bolting tool as it moves along the tracks, a rack and pinion gear set will be built into the design. This feature provides the accurate positioning required for the bolting tool to engage with the blanket fasteners. As a result, this bolting tool will be capable of engaging and disengaging the tracks from all sides. In addition, it is also equipped with the special tools to provide the required preload on the blanket module fasteners. Emergency conditions and recovery scenarios must be studied and analyzed. As a result this device may have to have interchangeable tool heads to perform special tasks for recovery operations.

Blanket Support Vehicle - The test blanket assembly, which consists of the blanket module, shielding, piping and the vacuum vessel door is expected to weigh 30-35 tons. Most of this weight is concentrated in the front section of the assembly at the blanket modules. Moving this assembly through the port extension requires a special vehicle capable of supporting and moving this weight. Another feature of this vehicle would be to provide the structural support between the blanket modules and the vacuum vessel plug. Since the two are only connected through the coolant pipes, a structural support system should be provided to avoid damage to any component of this assembly. The blanket support vehicle is designed to fit through the access hatch and to ride on the floor track in the port extension. The front end of this vehicle will be attached to the blanket modules and shielding assembly, while the back end extends through the access hatch and bolts to the vacuum vessel door. This creates a rigid connection between the two components and enables moving the full assembly. A heavy duty set of wheels are used to support the vehicle on the port tracks as it moves through the port extension.

This vehicle would be inserted into the port extension after the bolting vehicle has been withdrawn. It is attached to the blanket modules and to the vacuum vessel door. Then the vacuum vessel door bolts are removed. The next step is to pull the support vehicle out into the transporter. This is achieved either by having this vehicle self propelled or using a manipulator to push and pull this vehicle as needed. This vehicle will also serve as a platform for assembling the blanket assembly in the hot cell prior to installing it in the tokamak. Additionally, it will also be used to transport the blanket assembly through the various stations in the hot cell during assembly, testing and inspection.

Vacuum Vessel Closure Plate- The vacuum vessel closere plate used for the test port is a modification of the standard ITER port. It is equipped with an access hatch to provide access to the port extension while the main door remains closed. It is attached to the main structure using bolts, and a lip seal on the periphery of the access hatch provides the vacuum seal required.

2.1.1.5.2 Electrical Design

During ITER operation, the blanket test modules are monitored and controlled separately through a dedicated control system. The monitoring and control systems are designed to insure proper blanket operation, and provide warning in case of malfunction. In addition, it is used as a data acquisition for recording relevant information for analysis purposes.

Monitoring of the blanket modules and the shielding assembly involves reading temperature, pressure and flow rates of all fluids used. Temperature monitoring of critical surfaces, such as the first wall surface, is required for safety purposes. Controlling the operating parameters of the blanket assembly requires active control systems to maintain pressures and flow rates to keep the blankets and shield operating within their specified design conditions. Control devices such as flow meters, control valves and pressure regulators are anticipated to be outside the bioshield in the pit area. This allows manned access to these devices for maintenance or replacement. Measuring instruments within the blanket assembly are considered a part of the overall assembly. They are installed and maintained in the hot cell during blanket maintenance. Additional instruments are installed on the piping system between the blanket module and the door. Additional pipe monitoring instruments include leak detection devices. Those devices will serve the whole blanket assembly as they are designed to monitor leaks within the port extension. Another area where leak detection is required is between the cryostat and the bioshield. Since cutting and welding of pipes will take place in this area, monitoring devices are a critical item in the safe operation of the blanket modules.

Data acquisition for analysis purposes consists of reading temperatures, pressures, flow rates and tritium content. The instruments used for this purpose can be used as a backup to the control system to provide redundancy and enhance safety. Power and control cable are connected to the blanket assembly in their respective locations. They are then bundled together and routed to a vacuum sealed bulkhead connection at the vacuum vessel door. This is considered part of the blanket assembly. Another plug and a wire bundle is used on the other side of the vacuum vessel door to connect the instruments to the outside monitoring system. A manipulator is used to disconnect the wire bundles prior to removing the blanket assembly.

Data acquisition and control systems are located in the pit area near the port opening, or they could be located inside the tritium processing transporter should this concept be adopted. Another required link will be provided by ITER is the communication line between the data acquisition system and the ITER control room.

2.1.2 Supporting Design Documents

Supporting design documents are listed in the Appendix A-B.

Appendix A	Preliminary Electromagnetic Analysis
Appendix B	9Cr Rcc-MR properties data

2.2 System Performance Characteristics

2.2.1 Operating State Description

The Test Blanket System has TBD operational states. At the moment only nominal data for the 3 versions of the BTM are summarized

2.2.2 Operating State Data

Normal Plasma Operating Data (stady state during 1000 s pulse)

Test Module	BTM-I	BTM-II	BTM-III
Total power [MW]	1.85	1.9	2.2
Total helium Mass flow [Kg/s]	3.6	2.1	3.6
Helium Pressure [MPa]	8	8	8
Helium Pressure drop in BTM [Mpa]	0.19	0.22	0.19
Helium inlet/outlet temperature [°C]	250 / 350	250 / 420	250 / 366
Max. power density [MW/m^3] in			
structural material	10	10	16
beryllium pebble bed	5	5	5
ceramic pebble bed	19	16	30
Max. temperatures [°C] in			
structural material	509	507	526
beryllium pebble bed	403	457	421
ceramic pebble bed	618	742	778
Max von Mises primary stresses [MPa]	56	56	56
Max von Mises primary plus secondary stresses [MPa]	364	364	370

Support Frame	BTM-I	BTM-II	BTM-III
Total power [MW]	3.3	3.3	4.7
Water mass flow [Kg/s]	14.3	14.3	22.5
Water Pressure [MPa]	3	3	3
Water Pressure drop [Mpa]	TBD	TBD	TBD
Water inlet/outlet temperature [°C]	100 / 150	100 / 150	100 / 150

Shield	BTM-I	BTM-II	BTM-III
Total power [KW]	35	35	36.5
Water mass flow [Kg/s]	0.17	0.17	0.17
Water Pressure [MPa]	3	3	3
Water Pressure drop [Mpa]	TBD	TBD	TBD
Water inlet/outlet temperature [°C]	100 / 150	100 / 150	100 / 150

Helium Coolant Subsystem	BTM-I	BTM-II	BTM-III
Primary coolant	helium		
number of circuits	2		
pressure [MPa]	8		
hot leg temperature [°C]	350		
cold leg temperature [°C]	250		
mass flow rate (both circuits) [kg/s]	3.7		
flow velocity [m/s] hot/cold leg	20.8 / 17.5		
loop pressure drop (BTM included) [MPa]	0.27		
total mass of coolant (both loops) [kg]	42		
Secondary coolant	water		
number of circuits	2		
pressure [MPa]	0.5		
temperature in/out [°C]	35 / 75		
mass flow rate (both circuits) [kg/s]	11.4		
flow velocity [m/s]	0.16		

Tritium Extration Subsystem	BTM-I	BTM-II	BTM-III
Purge flow:			
pressure at BTM outlet [MPa]			0.106
pressure at BTM inlet [MPa]			0.120
temperature at BTM outlet [°C]			450
temperature at BTM inlet [°C]			20
mass flow rate [g/s]			0.85
Tritium production rate g/day			0.15
Swamping Ratio (He/H_2)			1000
Partial Pressure [Pa]			
$p(H_2)$			110
p(HT+HTO)			0.29
$p(H_2O)$			0.18
Extration Rates:			
H_2 [mole/day]			18.36
HT [mole/day]			0.04
H_2O / HTO [g/day]			≈ 0.9

Coolant Purification Subsystem	BTM-I	BTM-II	BTM-III
Coolant Purification Circuit:			
pressure [MPa]	8		
temperature inlet / outlet [°C]	250/50		
mass flow rate [g/s]	1.85		
Partial Pressure [Pa]			
$p(H_2)$	62		
p(HT)	0.24		
p(DT)	0.24		
$p(Q_2O)$	35		
$p(N_2)$	8		
Extration Rates [mole/day]:			
Q_2O	0.47		
N_2	0.04		
Tritium Extration Efficiency	≥95 %		

2.3 System Arrangement

The Test Blanket Subsystem consist of five main subsystems:

1. the Test Blanket Subsystem (first wall, breeding blanket, shield, and structure);
2. the Tritium Extraction Subsystem (tritium removal, handling, and processing);
3. the Helium Coolant Subsystem (heat transfer and transport);
4. the Coolant Purification Subsystem;
5. the Test Blanket Remote Handling Subsystem (remote handling as related to the test blanket systems).

The system arrangement has been already described in section 2.1. In the following, subsystem location, related drawings and subsection in which each subsystem is described, are summarized.

Subsystem	Location	Drawings	Section
Test Blanket	VV Horizontal Port	2.1.1.1.4-2, -3	2.1.1.1
Tritium Extraction	pit adjacent to the test port	2.1.1.2-2	2.1.1.2
Helium Coolant	Tritium Building	2.1.1.3-2	2.1.1.3
Coolant Purification	Tritium Building	2.1.1.4-2	2.1.1.4
Blanket Remote Handling		2.1.1.5-1, -2	2.1.1.5

2.4 Component Design Description

2.4.1 List of Components

In order to provide the required functions, the European HCPB Test Blanket System is composed of the following subsystem and their components:

1. Test Blanket Subsystem
    - Blanket Test Module
    - Support Frame
    - Shield
    - Plumbing
    - Vacuum Vessel Closure Plate (if the option with pipe penetrations through this door is chosen)
2. Tritium Extraction Subsystem
    - Ionization Chamber
    - Cooler
    - Filter
    - Cold Trap
    - Recuperator
    - Low Temperature Adsorber
    - Heater
    - Compressor
    - Helium Make-up Unit
    - Water Collector
    - Pumps
    - Relief Tank
    - Blower
    - Diffusor
    - Getter Bed
    - Helium Tank
    - LN_2Tank
    - Gaschromatograph
3. Helium Coolant Subsystem
    - Heat Exchangers
    - Circulator
    - Electrical Heater
    - Dust Filter
    - Pipework
    - Valves
    - Pressure Control Subsystem
      - Helium Storage Tank
      - Helium Dump Tank

- Buffer Tank
- Compressor
- Pressure Regolators
- Safety Valves

4. Coolant Purification Subsystem
    - Water Separator
    - Electrical Heater
    - Catalytic Oxidizer
    - Cooler
    - Blower
    - Cold Trap
    - Recuperator
    - Low Temperature Adsorber
    - Relief Valves
    - Relief Tank
5. Test Blanket Remote Handling Subsystem
    - Test Blanket Remote Handling Transporter
    - Remote Bolting Tool
    - Blanket Support Vehicle

The components and systems are extensively described in section 2.1. This section contains only a short description of all the components with some additional information.

2.4.2 Test Blanket Subsystem

2.4.2.1 General Description

2.4.2.2 Blanket Test Module

The HCPB-BTM represents a poloidal portion of the HCPB DEMO blanket. As in the DEMO the radial toroidal plates (see Fig. 2.4.2.2-1 and 2.4.2.2-2) and the first wall are cooled by helium at 8 MPa flowing first in the first wall and then in the blanket plates. For safety reasons the coolant helium flows in two completely separated loops. In the blanket the coolant helium is flowing alternatively in opposite directions in the first wall and in the adjacent blanket plates. In this way the BTM temperatures are more uniform. In the reference BTM (BTM-I) between the blanket plates there are alternatively 11 mm thick ceramic breeder pebble layers and 45 mm thick beryllium pebble layers. The tritium purging gas is helium at about 0.1 MPa flowing in radial direction from the first wall to the back of the module. The plasma side of the first wall is protected by a 5 mm beryllium layer and it recessed from the shield blanket countor by a minimum amount of 50 mm. At the upper and lower ends the HCPB-BTM is closed by covers capable to sustain a pressure of 8 MPa. During normal operation the space in the BTM other than the cooling plates and the FW is at the purge gas pressure of 0.1 MPa. However, in case of a leak from a cooling plate, it can be pressurized up to 8 MPa. Thus the blanket box, and the helium purge system connected to it, have been designed to sustain the full 8 MPa pressure. This is a double barrier against helium leakage from the cooling plates and would allow, in case of need, to wait for the next planned period for the exchange and repair of the module.

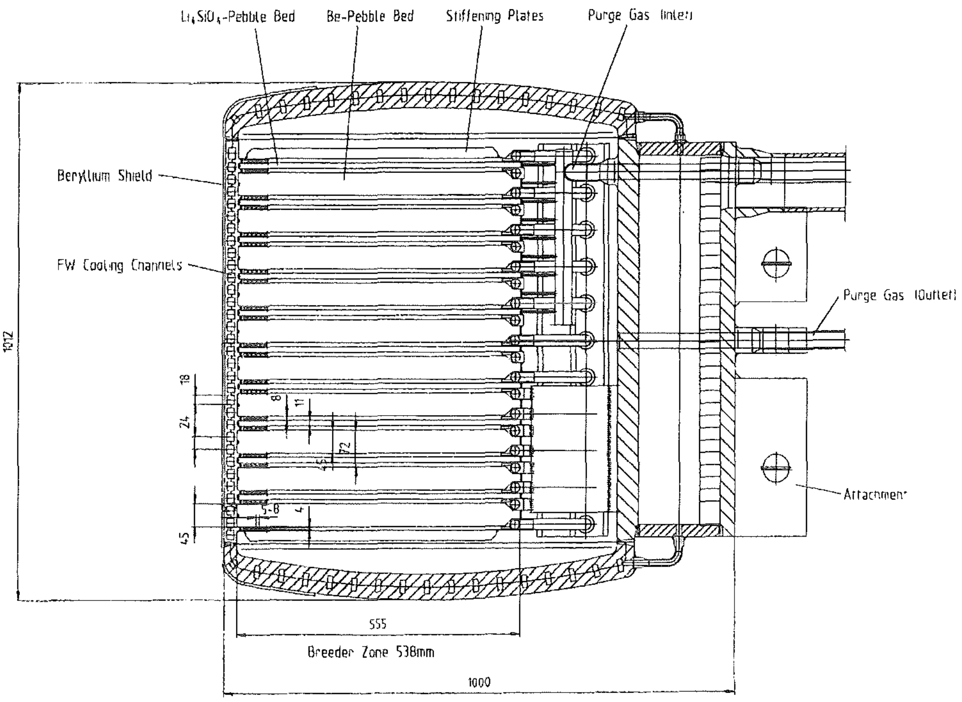


Fig. 2.4.2.2-1 Vertical section of the Reference BTM.

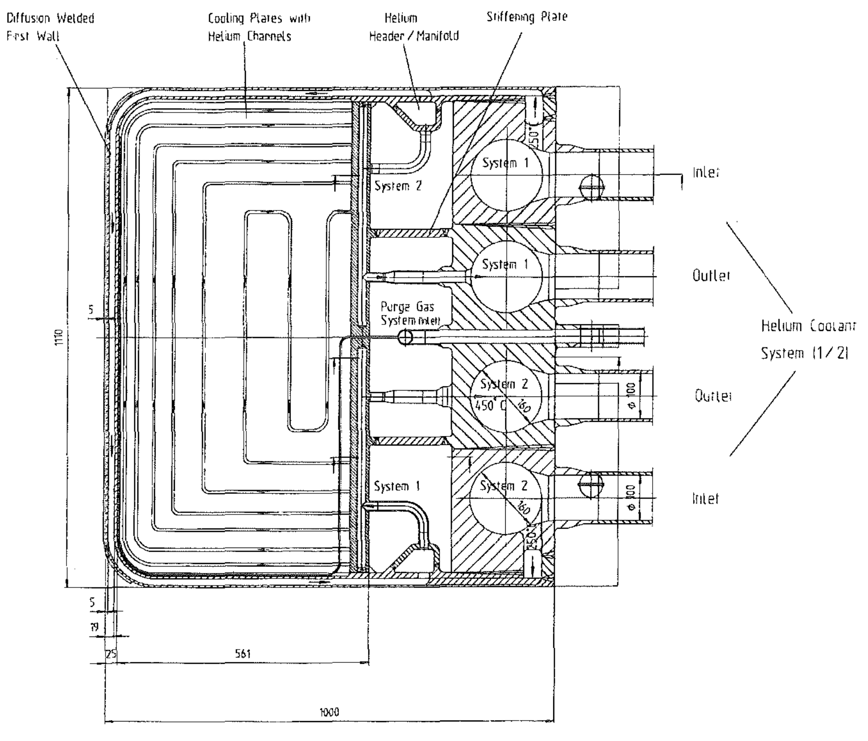


Fig. 2.4.2.2-2 Horizontal section of the Reference BTM

2.4.2.3 Support Frame

The Support Frame is made of the same structural material (316LN-IG) as the back plate and maintained by the cooling water at about the same temperature of the back plate at their contact surfaces. The part of the frame facing the plasma is covered by a layer of beryllium as armour material and an heat sink of 316LN-IG steiless steel tubes embedded in a copper alloy layer. The design will be similar to the ITER shielding Blanket.

The power produced in the frame is 3.3 and 4.7 MW during the BPP and the EPP operation respectively

2.4.2.4 Shield

The Shield is cooled by water and maintained at the same temperature of the Support Frame at its contact surfaces. It is made of 316LN-IG. The power produced in the shield is 35 and 36.5 kW during the BPP and the EPP operation respectively.

2.4.2.5 Plumbing

In Table 2.4.2.5-1 number, size, description and material of the pipes are summarized. A set of two supply and two return pipes provides the HCPB Blanket Test Module with high pressure helium coolant. A set of one supply and one return pipe provides the 0.1 MPa helium for the purging of the tritium produced in the BTM.

A simple set of water pipes is used to cool the Support Frame and the Shield, separately. A conduit is reserved for the diagnostics.

The plumbing sistem has to be completed by the helium coolant and purge pipes for the Japanese BTM

2.4.2.6 Vacuum Vessel Closure Plate

If the solution with plumbing penetration through the Vacuum Vessel Closure Plate is adopted, this Plate is part of the Test Blanket Subsystem. In this case, the Closure Plate used for the test port is a modification of the standard ITER plate. It allows the penetration of the plumbing of the Test Blanket Assembly. Furthermore, it is equipped with an access hatch to provide access to the port while the main plate remains closed. It is attached to the main plate using bolts, and a lip seal on the periphery of the access hatch provides the vacuum seal required.

Table 2.4.2.5-1 Plumbing Description

Pipe description (1)	No.	Size (2)	Material	Pipe Carrier (Oper. Cond.)
Helium Coolant Supply	2	115	MANET	helium (8 MPa, 250°C)
Helium Coolant Return	2	115	MANET	helium (8 MPa, 350-420°C)
Purge Gas Supply (3)	1	25	MANET	helium (0.1 MPa, 20°C)
Purge Gas Return (3)	1	25	MANET	helium (0.1 MPa, 450°C)
Diagnostics Conduct	1	250	316LN-IG	-
Shield Coolant Supply	1	50	316LN-IG	water (3 MPa, 100°C)
Shield Coolant Return	1	50	316LN-IG	water (3 MPa, 150°C)
Frame Coolant Supply	1	50	316LN-IG	water (3 MPa, 100°C)
Frame Coolant Return	1	50	316LN-IG	water (3 MPa, 150°C)

(1) The helium coolant and purge pipes for the Japanese BTM are not included in the table.
(2) Outer Diameter (in mm) not accounting for thermal insulation (5 - 10 cm).
(3) Two concentrical tubes with detection gap.

2.4.3 Tritium Extraction Subsystem

The main design data of the tritium extraction system are given in Table 2.4.3-1. The system has to be designed for a pressure of 8 MPa because leakages from the coolant system can lead to a pressure increase. In addition, relief valves are forseen to cope with the case of over-pressurization. For radiological safety reasons, the system must be installed in a secondary containment.

Table 2.4.3-1 : Main Design Data for the Tritium Extraction System

Power in Test Module (EPP)	2.3 MW a)
He Mass Flow	0.85 g/s = 17 Nm3/h
Swamping Ratio	He : H_2 = 1000
Tritium Production Rate	0.15 g / day
Partial Pressures b) p (H_2) p (HT+HTO) p (H_2O)	110 Pa 0.29 Pa c) 0.18 Pa c)
Extraction Rates H_2 HAT H_2O / HTO	18.36 mole /day 0.04 mole /day ≈ 0.9 g / day
Temperature of Purge Gas at Test Module Outlet at Test Module Inlet	 450 °C 20 °C
Pressure of Purge Gas at Test Module Outlet at Test Module Inlet Pressure Drop in Test Module	 0.106 MPa 0.120 MPa 0.014 MPa

a) MW in BPP 2.3 MW in EPP (including 0.3 MW in First Wall in both cases)
b) Average values at Test Module outlet (accounting for plasma pulse dwell time)
c) about 80 % of HTO is assumed to be converted to HT + H_2O by isotopic
d) exchange, no HTO / H_2O is considered to be reduced by the steel walls

In the following table (Table 2.4.3-2) the components of the Subsystem are listed with remarks on their development status.

Table 2.4.3-2 Component development and design

Component	Available from Industry	Remarks
Cold Trap	Yes	a)
Molecular Sieve Beds	Yes	a)
Getter Beds	Yes	b)
Diffuser	Yes	a)
Circulator	Yes	b)

a) These components can be purchased from the industry; however, as they will be operated under conditions characterized by high gas flow rates and extremely low concentrations of Q_2O and Q_2 (Q = H,T) which are not at all typical for industrial applications, it appears indispensable to carry out an experimental test program to optimize/modify the design of the components, i.e. to avoid over-dimensioning and to demonstrate the desired removal factors. In addition, such tests are needed to develop appropriate means for process control and analytical measurements.

b) These components can be purchased from the industry or from special suppliers without the need of experimental testing as described above.

2.4.3.1 Ionization Chamber

2.4.3.2 Cooler

2.4.3.3 Filter

The particulate filter removes particulate material which might be carried out from the blanket zone. The arrangement of these filters has to be planned in such a way that they are easily exchangeable (within 1-2 hours).

2.4.3.4 Cold Trap

The cold trap operated at ≤ -100°C frozes out the Q_2O content (Q = H, T) of the gas. The residual Q_2O concentration (Q = H, T) is < 0.015 ppm. The amount of ice accumulated within 6 days is of the order of a few grams (max. 6 g). This is advantageous for two reasons: (a) the construction of the trap is relatively simple, and (b) it is not necessary to exchange the water collector (Volume ≤ 200 ml) after each test run. Perodically, the cold trap loaded with ice is depressurized and warmed up to room temperature to liquefy the water which is then drained into a water container.

2.4.3.5 Recuperator

The recuperator use the clean gas leaving the adsorber to precool the gas coming from the cold trap.

2.4.3.6 Low Temperature Adsorber

Two adsorber beds filled with 5A zeolite pellets which adsorb molecular hydrogen as well as gaseous impurities and residual moisture, operate at liquid nitrogen (LN_2) temperature (-196°C). The beds contains filters on the down-stream and upstream side to prevent particulate material from being transferred during loading or unloading operations. In addition, each bed is equipped with a LN_2 chiller and an electrical heater. The second bed provides additional adsorption capacity; it can be also used when the first bed is being unloaded or regenerated. Periodically, the adsorber beds are regenerated. During a normal unloading operation they are depressurized and warmed up to room temperature, and the desorbing impurities are sent to the Waste Gas System. A complete regeneration is achieved by heating to 300 °C and purging with clean helium.

2.4.3.7 Heater

2.4.3.8 Compressor

The purge gas blower comes in contact only with clean gas at room temperature.

2.4.3.9 Helium Make-up Unit

In the helium make-up unit hydrogen is added to provide a He : H_2 swamping ratio of 1000 for the gas reentering the blanket test module.

2.4.3.10 Water Collector

The liquefied water from the cold trap is drained into an evacuated water collector which is later on transferred to the Water Detritiation System and replaced by an empty collector vessel.

2.4.3.11 Pumps

2.4.3.12 Relief Tank

A relief tank with a volume of 2 m3 is available to avoid a pressure increase during desorption in the warm-up phase of the adsorber; in fact, the maximum amount of adsorbed H_2/ HT is 110 mole. The pressure will remain below 0.2 MPa even though the tank has to be prefilled with 50 kPa helium which is needed as a carrier gas for the hydrogen isotopes during the further un-loading process. At the end of the unloading cycle, the relief tank is evacuated (leading to the Waste Gas System) and refilled with 50 kPa.

2.4.3.13 Blower

2.4.3.14 Diffuser

The Pd/Ag diffuser separates the hydrogen isotopes from the helium carrier gas.

2.4.3.15 Getter Bed

Several uranium getter beds are provided for storage of the hydrogen isotopes. When the loading capacity of these beds is reached they will be transferred to the Isotope Separation System and replaced by fresh beds. This is expected to be needed after 6 days of nominal operation with back-to-back pulses.

2.4.3.16 Helium Tank

2.4.3.17 LN_2Tank

2.4.3.18 Gaschromatograph

2.4.4 Helium Coolant Subsystem

The cooling subsystem is designed for the European helium-cooled pebble bed (HCPB) test module to be installed in the bottom half of an equatorial test port in ITER, presumably port No. 20. Two separate primary heat removal loops of 2 x 50 % heat capacity are foreseen for redundancy purposes in accordance with the DEMO blanket design. Figure 2.1.1.3-1 shows a flow diagram of the primary heat removal loops and the interfaces to ancillary equipment. Figure 2.1.1.3-2 shows the arrangement in the tritium building with an estimated space required of about 700 m^3, excluding the helium purification subsystem (see Section 2.1.1.3).

In the following table (Table 2.4.4-1) the components of the Subsystem are listed with remarks on their development status.

Table 2.4.4-1 Component development and design

Component	Remarks
Heat Exchangers	special design (tritium leaktightness)
Circulator	special design (operating temperature, leaktightness and bearing
Electrical Heater	special quality assurance program (tritium leaktightness)
Dust Filter	special design (efficiency and grain size)
Pipework	conventional
Valves	special design (flow control performance)
Pressure Control Sys.	conventional (presence of tritium)

2.4.4.1 Heat Exchanger

A first layout has been performed assuming a straight tube bundle heat exchanger (HX) with high pressure helium flowing inside the tubes and low pressure water flowing outside. The required tube bundle data along with the primary and secondary loop flow parameters are listed in Table 2.1.1.3-3, showing the nominal values for the first test module and the design values considering anticipated options with sufficient margin. For the design size the helium volume in one HX would be 0.06 m^3 (0.02 m^3 in the tubes and 0.04 m^3 in the end domes). Alternatively a U-tube HX could be envisaged which would not significantly alter the design data.

2.4.4.2 Blower

One variable speed helium circulator will be installed in the cold leg of each primary loop operating at 250 °C in normal operation. Including some margin during heating and baking phases the design temperature is set to 300 °C and the design pressure to 9.6 MPa (20 % above nominal for overpressure control). An encapsulated type circulator with vertical shaft is envisaged where the type of bearing (gas lubricated or magnetic) has still to be decided upon. The design specification for the circulator is as follows: temperature 300 °C, pressure 9.6 MPa, mass flow rate 1.9 kg/s at a pumping head of 0.27 MPa at 80 % of maximum speed and at 250 °C inlet temperature, speed variation max/min of at least 4. Under these conditions the electric power of the drive motor would be 100 kW. The helium volume contained in the circulator is estimated as 0.025 m^3 and the overall dimensions of the circulator and drive unit are expected to be 0.5 m diameter times 1.8 m height.

2.4.4.3 Electrical Heater

This component is needed for baking the test module first wall at 240 °C and for heating the whole cooling subsystem including the test module to operating temperatures after maintenance or repair periods. It is positioned in a bypass to the HX, assuming that the HX is isolated during heating phases and the circulator is operating at reduced speed. The required electrical power has been estimated to 100 kW. This would enable to heat the whole subsystem at a rate of about 10 to 20 °C per hour. The main dimensions of the helium volume are 0.3 m diameter times 1.2 m height, approximately half of which being occupied by the heating rods. This yields a helium volume of 0.042 m^3. The estimated pressure loss is small, ca. 500 Pa. The overall dimensions are assumed to be 0.5 m diameter times 1.7 m height.

2.4.4.4 Dust Filter

A filter unit is installed in the hot leg of the main loop, accumulating residual dust and particles from fabrication, and erosion particles down to a size of typically 10^{-6} m. To some extent even the much smaller sputter products evolving in the neutron field in the BTM may be trapped which otherwise are expected to be deposited mainly in the heat exchanger and at pipe walls. The array of small-diameter filter tubes, or plates in a grid format, forms a removable filter cartridge of 0.3 m diameter and 3 m length, giving a helium volume of approximately 0.2 m^3. The pressure loss is expected to be less than 0.03 MPa.

2.4.4.5 Pipework

For the main pipework, i.e., hot leg and cold leg, an outer diameter of 168.3 mm and a wall thickness of 10 mm have been chosen for the part external to the cryostat. Inside the cryostat smaller pipes are foreseen (114.3 mm outer diameter, 8 mm wall thickness) to limit the size of pipe penetrations in the cryostat to 130 mm. This results in a flow velocity in normal operation of between 15 and 41 m/s and in small pressure losses. The pipe length is determined by space allocation in ITER, leading first to 20 m of horizontal (radial) run (10 m of which are assumed to be of the small size), followed by a 20 m vertical part, followed by 10 m of horizontal (tangential to the torus) run and completed by about 10 m of pipework connecting the components in the tritium building. This sums up to 60 m of pipework each for cold and hot legs. The number of elbows is assumed to be 16 per leg (6 per leg of the small diameter). Overall the pipework contributes with 14 % to the pressure losses in the loop (Table 2.1.1.3-2).

The flow rate during baking and heating will be reduced by a factor of about 4 compared to the rated mass flow rate. Thus the bypass to the HX can be smaller than the main pipework. The outer diameter has been set accordingly to 82.5 mm. In normal operation the bypass is supposed to be almost closed during burn times and open during dwell times (see 2.1.1.3.6).

2.4.4.6 Valves

2.4.4.7 Pressure Control Subsystem

This is a combination of equipment needed for cooling subsystem evacuation, helium supply, pressure control, and overpressure protection.

The helium supply and storage subsystem consists of storage tanks, buffer tank, compressor, and pressure regulators. The battery of storage tanks is sized as to take the whole helium inventory of the loop excluding the one in the buffer tank (about 17 kg) at about 50 °C, 14 MPa, resulting in a tank volume of 0.81 m^3. This can be achieved by, e.g., two tanks of 0.5 m diameter, 2.6 m long with 20 % reserve. A multi-stage compressor and cooler will be needed to load the storage tanks for emptying the main loops. A buffer tank volume of 10 % of the loop volume is chosen, that is about 0.25 m^3, and a maximum operating pressure of 14 MPa.

The overpressure protection of the cooling loops consists of two redundant safety valves or a combination of one safety valve plus a burst disc. The safety valves discharge into a dump tank which is kept at controlled low pressure (near atmospheric) during normal operation. The dump tank is sized for the event that the primary loop was inadvertently pressurised to the nominal pressure (8 MPa) at room temperature and the whole subsystem was subsequently heated up to 250 °C, the nominal operating inlet temperature. If the pressure regulator would fail to open, the safety valve would respond. In order to limit then the pressure to 8 MPa requires a dump volume of about 70 % of the loop volume, i.e., 1.84 m^3. This is about twice to what is needed as storage volume.

2.4.5 Coolant Purification Subsystem

For each of the two coolant systems of the blanket test module one purification system is provided to purify a fraction of 0.1% of the helium coolant stream, i.e.

- to extract hydrogen isotopes as well as solid, liquid or gaseous impurities from the main coolant, and
- to remove condensed water that may be entrained in the cooling gas due to leakages or failures of the heat exchanger tubes.

The main design data of the purification systems are given in Section 2.1.1.4.

In the following table (Table 2.4.5-1) the components of the Subsystem are listed with remarks on their development status.

Table 2.4.5-1 Component development and design

Component	Available from Industry	Remarks
Oxidizer	Yes	a)
Cold Trap	Yes	a)
Water Separator	Yes	b)
Molecular Sieve Beds	Yes	a)

a) These components can be purchased from the industry; however, as they will be operated under conditions characterized by high gas flow rates and extremely low concentrations of Q_2O and Q_2 (Q = H,T) which are not at all typical for industrial applications, it appears indispensable to carry out an experimental test program to optimize/modify the design of the components, i.e. to avoid over-dimensioning and to demonstrate the desired removal factors. In addition, such tests are needed to develop appropriate means for process control and analytical measurements.
b) These components can be purchased from the industry or from special suppliers without the need of experimental testing as described above.

2.4.5.1 Water Separator

It removes condensed water that may be present in the helium stream as a consequence of larger leakages.

2.4.5.2 Electrical Heater

Two electrical heater warmers for the helium stream are present in each loop.

2.4.5.3 Catalytic Oxidizer

Oxidizer unit containing a precious metal catalyst (Pd or Pt on alumina). An over-stoichiometric amount of oxygen is added to obtain a quantitative conversion of Q_2 to Q_2O (Q = H, D, T).

2.4.5.4 Cooler

The water cooler reduces the helium temperature to room temperature. It is assumed that the secondary water loop system is part of the ITER cooling system providing water flow at low temperature and pressure.

2.4.5.5 Blower

It should be possible to operate the purification system without an additional compressor or circulation pump. Nevertheless, a corresponding pump will be available on demand.

2.4.5.6 Cold Trap

A cold trap operated at -100°C frozes out the remaining humidity. The amount of water extracted under the conditions described in Table 2.1.1.4-1 is 8.5 g/day. Perodically, the cold trap loaded with ice is depressurized and warmed up to room temperature to liquefy the water which is then drained into a water container and sent to the Water Detritiation System.

2.4.5.7 Recuperator

2.4.5.8 Low Temperature Adsorber

Two 5A molecular sieve beds cooled with liquid nitrogen (LN_2) to adsorb gaseous impurities like N_2; any hydrogen isotopes that have not been oxidized are also adsorbed. The inlet and the outlet side of the beds are equipped with mechanical filters to prevent a carry-over of particulate material during normal operation (downward flow) and regeneration (upward flow). The second bed provides additional adsorption capacity; it may be used when the first bed has not been unloaded or regenerated. Periodically, the adsorber beds are regenerated. During a normal unloading operation they are depressurized and warmed up to room temperature, and the desorbing impurities are sent to the Waste Gas System. A complete regeneration is achieved by heating to 300 °C and purging with clean helium.

2.4.5.9 Relief Valves

2.4.5.10 Relief Tank

The Relief Tank stores the desorbing impurities, which are released by the cold bed during a regeneration operation. These impurities are periodically sent to the Waste Gas System.

2.4.6 Test Blanket Remote Handling Subsystem

The remote handling system for the test blanket modules will consist of a number of components designed to perform certain tasks. A number of the tasks required for the removal and installation of the test blanket modules are identical to ITER tasks, such as bioshield plug installation, cryostat closure plate interface and the pipe cutting and welding operation. For those tasks, ITER equipment will be used. Special equipment such as the test blanket support vehicle, the remote bolting tool and test module assembly transporter will be designed by the test blanket group with interface data inputs from ITER. A brief description of each of the special equipment is included below, although concept drawings will be provided as soon as they become available.

2.4.6.1 Test blanket remote handling transporter

This transporter will be based on the standard transporter design as it is developed by ITER. This transporter will be designed to handle a number of tasks. As noted in the remote handling process procedure in the previous section, the transporter should be equipped with special tools to perform a number of specific operations. A manipulator is needed to plug and unplug the power and diagnostics cable bundles without damaging them. This manipulator will serve other tasks by exchanging the end effector tools to fit a specific task. Some of those tools include a fastening tool to handle the vacuum vessel plug bolts. Another tool is needed to cut and weld the lip seal weld of the vacuum vessel plug and the access hatch. Deployment of the temporary tracks between the cryostat closure plate and the vacuum vessel is also handled with this manipulator. Other tools include inspection equipment and possibly viewing equipment such as a camera to perform remote visual inspection.

The transporter is also designed to contain the test blanket assembly. Internal tracks are installed to allow the blanket support vehicle to travel into and out of this transporter. Room should be provided within the transporter to store the temporary tracks when not in use. Monitoring equipment designed to monitor the status of the test blanket modules during transport to/from the test port should be built into the transporter with capabilities to transmit important or emergency status data to the control room. Emergency recovery operations should be designed and built into this transporter to enable it to recover from certain emergency conditions without interrupting the operation of ITER. Other equipment stored inside the transporter include the remote bolting tool and the blanket support vehicle. Active cooling inside the transporter for the test blanket module may is required.

2.4.6.2 Remote bolting tool

As noted in the previous section, the blanket assembly may consist of the blanket modules, shield, related piping and the vacuum vessel closure plate. While this particular arrangement has the advantage of eliminating remote operations inside the port extension, it creates an accessibility problem for fastening the test blanket assembly to the back plate. To gain access to the back plate while the blanket assembly is installed, an access hatch is built into the vacuum vessel door. This access hatch is bolted and seal welded to the vacuum vessel door. A remote manipulator is used to cut the seal weld and remove the fasteners. The access hatch is then removed into the transporter. A remote bolting tool designed to fit through this access hatch is then dispatched into the port extension. The main task for this tool is to be able to reach all the blanket module fasteners and either install or remove them. This is achieved by having a set of tracks built into the port structure. Radial tracks on the floor of the port extension are used for the tool deployment. Once it reaches a certain distance away from the blanket module, the bolting tool engages a set of crossing track that allows it to move toroidally until it reaches the side walls of the port extension. Tracks along the side walls are designed to allow the bolting tool to move up and down allowing it access to the side wall bolts. Tracks in the ceiling of the port extension are designed to engage the tool and allow it to hang as it traverses across the top to reach the upper set of fasteners. In order to provide positive traction for the bolting tool as it moves along the tracks, a rack and pinion gear set will be built into the design. This feature provides the accurate positioning required for the bolting tool to engage with the blanket fasteners. As a result, this bolting tool will be capable of engaging and disengaging the tracks from all sides. In addition, it is also equipped with the special tools to provide the required preload on the blanket module fasteners. Emergency conditions and recovery scenarios must be studied and analyzed. As a result this device may have to have interchangeable tool heads to perform special tasks for recovery operations.

2.4.6.3 Blanket Support Vehicle

The test blanket assembly is expected to weigh circa 30 tons. Moving this assembly through the port extension requires a special vehicle capable of supporting and moving this weight. Another feature of this vehicle would be to provide the structural support between the shield and the vacuum vessel closure plate. Since the two are only connected through the coolant pipes, a structural support system should be provided to avoid damage to any component of this assembly. The blanket support vehicle is designed to fit through the access hatch and to ride on the floor track in the port extension. The front end of this vehicle will be attached to the shield, while the back end extends through the access hatch and bolts to the vacuum vessel closure plate. This creates a rigid connection between the two components and enables moving the full assembly. A heavy duty set of wheels are used to support the vehicle on the port tracks as it moves through the port extension.

This vehicle would be inserted into the port extension after the bolting vehicle has been withdrawn. It is attached to the blanket modules and to the vacuum vessel door. Then the vacuum vessel closure plate bolts are removed. The next step is to pull the support vehicle out into the transporter. This is achieved either by having this vehicle self propelled or using a manipulator to push and pull this vehicle as needed. This vehicle will also serve as a platform for assembling the blanket assembly in the hot cell prior to installing it in the tokamak. Additionally, it will also be used to transport the blanket assembly through the various stations in the hot cell during assembly, testing and inspection.

2.5 Instrumentation and Control

Test Blanket Subsystem

The Test Blanket Subsystem requires instrumentation to monitor temperature, pressure, pressure drop and mass flow of the coolants, temperature of the pebble beds, and temperature and local stresses of the structural materials. Additionally the measurement of the tritium concentration at the test module outlet and inlet of the helium coolant and of the purge gas loops should be performed to allow to make a tritium balance and assess the tritium permeation losses.

BTM

cooling plates	temperature, local stresses
first wall	temperature, local stresses
pebble beds	temperature
helium coolant	temperature, pressure, pressure drop
purge gas	temperature, pressure, pressure drop

Support Frame	TBD
Shield	TBD
Plumbing	temperature, pressure, pressure drop, tritium and protium concentrations

Specific Instrumentation for the BTM is listed in Table 2.5-1.

Table 2.5-1 Instrumentation Requirements for the Blanket Test Module

Instrument Type	Range	Accuracy	Number of Sensor	Sensor Location	Read out Location
Thermocouple	0-600°C	±10°C	≈TBD	first wall	Control Room
Thermocouple	0-600°C	±10°C	≈TBD	cooling plates	Control Room
Thermocouple	0-600°C	±10°C	≈TBD	helium cooling system	Control Room
Thermocouple	0-600°C	±10°C	≈TBD	purge gas system	Control Room
Thermocouple	0-1000°C	±10°C	≈TBD	ceramic pebble beds	Control Room
Thermocouple	0-800°C	±10°C	≈TBD	beryllium pebble beds	Control Room
Pressure Tapping	0-10MPa	±TBD MPa	≈TBD	helium coolant system	Control Room
Pressure Tapping	0-10MPa	±TBD MPa	≈TBD	purge gas	Control Room
Differential pressure Tapping	0-0.1MPa	±TBD MPa	≈TBD	purge gas	Control Room
Differential pressure Tapping	0-0.1MPa	±TBD MPa	≈TBD	helium coolant system	Control Room
Strain gage	0-400MPa	±TBD MPa	≈TBD	first wall	Control Room
Strain gage	0-400MPa	±TBD MPa	≈TBD	cooling plates	Control Room

<u>Tritium Extraction Subsystem</u> TBD

<u>Helium Coolant Subsystem</u> TBD

<u>Coolant Purification Subsystem</u> TBD

<u>Test Blanket Remote Handling Subsystem</u> TBD

2.6 System Interfaces

In order to successfully complete all the test objectives, the Test Blanket System must work in cooperation with many of the other ITER systems and facilities.

2.6.1 Plasma Interface

The first wall of the Blanket Test Module and of the Support Frame is recessed from the shield blanket contour by a minimum amount of 50 mm. The first wall of the test module is planar, without curvature, but is conform as closely as possible to the first wall of the adjacent shield blanket modules. The plasma side of the first wall of the Blanket Test Module and the Support Frame is protected by a 5 mm beryllium layer in order to avoid that high atomic weigt particles can reach the plasma.

The test Blanket System has a leak rate of coolant Helium lower than 10^{-8} Pa m^3 s^{-1}.

The first wall of the Blanket Test Module and the Support Frame has been design in order to remove the surface heat flux and the nuclear heating within the allowable temperature and stress limits.

The in vessel portion of the Test Blanket System contribute to meeting the requirement that the combined toroidal resistance of the blanket in-vessel strctures and the Vacuum Vessel must be larger than $4\mu\Omega$. The Support Frame and the Shield assure a continuous electrical connection (poloidal and toroidal) of the back plate. This contribute to the passive stabilization of the plasma

The presence of a large amount of ferromagnetic material - as structural material for the Blanket Test Module - in the proximity of the plasma magnetic boundary causes a local slight distortion of the toroidal magnetic field. However this distortion remains below the allowable limits [2.6.1-1].

References:

[2.6.1-1] Statement of R.Aymar at the Test Blanket Working Group Meeting of 16-17th Januar 1996.

2.6.2 Blanket System Interface

The Test Blanket Subsystem is mounted onto the back plate, wich will carry all the static and dynamic loads and provide the proper dimensional control for alignment of the Test Blanke Subsystem within the shielding blanket modules. A list of the additional loads caused by the presence of the Test Blanket System is presented in Table 2.6.2-1.

The opening in the back plate at the horizontal port contains a toothed-frame. The Support Frame has a series of teeth to react the static and dynamic loads induced in the Test Blanket Subsystem. The teeth have bolts to engage and secure the toothed-frame system and react the normal loading conditions. The connection between the teeth is to provide an electrical current path for current generated in the module. The Supporting Frame is cooled to approximately 150°C to prevent high thermal stresses in the mounting flange. There is a gap allowance of 20 mm completely around the perimeter of the test blanket.

The first wall of the Test Blanket System is recessed below the general surface level of the surrounding Shield Blanket First Wall. This imposes additional surface heating requirements on the adjacent Shield Blanket First Wall components.

Table 2.6.2-1 Back plate additional loads due to the Test Blanket System

Loads (*)	x (**)	y (**)	z (**)
Weigth (with coolant):			
application point [m]	12.19	0.00	1.355
resultant force [kN]	-	-	289
Centered Disruption (***):			
application point [m]	11.658	0.0	1.355
resultant force [MN]	-1.981	0.344	1.600
resultant torque [MNm]	3.927	-1.085	-1.869
VDE (***):	TBD	TBD	TBD

(*) The loads caused by the Japanese BTM are included in the presented values. They are extimated on the basis of the HCPB BTM design.

(**) Torus coordinate system.

(***) The part of the electromagnetic loads acting on the backplate self caused by the presence of the Test Blanket System are not included.

2.6.3 Vacuum Vessel Interface

The Test Blanket plumbing can either extend through the vessel door or exit through the extension wall (see Section 2.1.1.1). The first approach eliminates any penetrations through the vacuun vessel wall. The vacuum vessel plate used for this solution is a modification of the standard ITER VV plate. The alternative approach requires penetrations through the vessel wall. For number, size and description of the pipes see Table 2.6.3-1.

Note that all penetrations through the vacuum vessel boundaries will require vacuum tight flexible connections such as bellows (Fig. 2.1.1.1-3). The design of such connections will be similar to those approved by the JCT.

The test port extension will be designed with built in tracks on the bottom floor to allow the bolting tool and the blanket removal vehicle to be deployed inside the port extension. Additional tracks will also be built into the port extension walls (side walls, floor and ceiling) near the blanket modules to allow the bolting tool to move along the walls and be able to reach all the fasteners of the blanket module (see Section 2.1.1.5).

Table 2.6.3-1 Vacuum Vessel Penetration List

Pipe description (1)	No.	Size (2)	Material	Pipe Carrier (Oper. Cond.)
Helium Coolant Supply	2	115	MANET	helium (8 MPa, 250°C)
Helium Coolant Return	2	115	MANET	helium (8 MPa, 350-420°C)
Purge Gas Supply (3)	1	25	MANET	helium (0.1 MPa, 20°C)
Purge Gas Return (3)	1	25	MANET	helium (0.1 MPa, 450°C)
Diagnostic Conduct	1	250	316LN-IG	-
Shield Coolant Supply	1	50	316LN-IG	water (3 MPa, 100°C)
Shield Coolant Return	1	50	316LN-IG	water (3 MPa, 150°C)
Frame Coolant Supply	1	50	316LN-IG	water (3 MPa, 100°C)
Frame Coolant Return	1	50	316LN-IG	water (3 MPa, 150°C)

(1) The helium coolant and purge pipes for the Japanese BTM are not included in the table.
(2) Outer Diameter (in mm) not accounting for eventual thermal insulation
(3) Two concentrical tubes with detection gap.

2.6.4 Remote Handling Interface

The remote handling system for the Test Blanket System will take full advantage of the equipment designed by the JCT to minimize duplication of efforts and to standardize system operations. All operations that are identical to other ITER operations will use the same ITER system to perform, such as removing the bioshield plugs and the cryostat plugs. Also internal pipe cutting and welding operation will use the JCT's bore tool design.

Operations that are specific to the Test Blanket System will be integrated into the overall system design. In particular the ITER Handling System must be able to perform operations 8 to 17 as listed in Section 2.1.1.5. Some of the interface requirements are listed below:

- Maximum supported weight 40 t
- position accuracy TBD
- kinematic requirements TBD
- inspection requirements TBD
- accomodation of special end effectors TBD
- accomodation of special material and coolants TBD

2.6.5 Cryostat Interface

The plumbing system from the Vacuum Vessel boundary penetrates the Cryostat Test boundary as shown in Fig. 2.1.1.1-2. Dimensions, size and description of the pipes are listed in Table 2.6.5-1.

All the penetrations through the cryostat boundaries will require vacuum tight flexible connections such as bellows. The design of such connection will be similar to those approved by the JRC.

Table 2.6.5-1 Cryostat Penetration List

Pipe description (*)	No.	Size (**)	Material	Pipe Carrier (Oper. Cond.)
Helium Coolant Supply (1)	2	115	316LN-IG	helium (8 MPa, 250°C)
Helium Coolant Return (1)	2	115	316LN-IG	helium (8 MPa, 350-420°C)
Purge Gas Supply (2)	1	25	316LN-IG	helium (0.1 MPa, 20°C)
Purge Gas Return (2)	1	25	316LN-IG	helium (0.1 MPa, 450°C)
Diagnostic Conduct (1) (2)	1	250	316LN-IG	-
Shield Coolant Supply (3)	1	50	316LN-IG	water (3 MPa, 100°C)
Shield Coolant Return (3)	1	50	316LN-IG	water (3 MPa, 150°C)
Frame Coolant Supply (3)	1	50	316LN-IG	water (3 MPa, 100°C)
Frame Coolant Return (3)	1	50	316LN-IG	water (3 MPa, 150°C)

(*) The helium coolant and purge pipes for the Japanese BTM are not included in the table.

(**) Outer Diameter (in mm) not accounting for eventual thermal insulation (5-10 cm).

(1) Helium Coolant Subsystem

(2) Tritium Extration Subsystem

(3) ITER Primary Heat Transfer System

2.6.6 Primary and Secondary Heat Transfer Interface

The Support Frame and the Shield of the Test Blanket Subsystem are cooled by water provided by the Primary Heat Transfer System of ITER. The design requirements are listed in Table 2.6.6-1.

The secondary heat removal system of the Helium Coolant Subsystem is considered as part of the ITER cooling system. For the layout a constant water flow with nominal inlet temperature of 35 °C at the secondary side of the two heat exchangers (see 2.1.1.3.1) and a flow rate of 11 kg/s per heat exchanger are assumed, leading to pipe dimensions of 57 mm diameter, 3 mm wall thickness, at a velocity of 5 m/s. The outlet temperature will then vary according to the burn and dwell cycles between 75°C and 35°C. Flow, pressure, and temperature monitoring are needed. No significant migration of tritium from the primary coolant to the secondary side is expected.

2.6.6-1 Design requirements for the ITER Primary Heat Transfer Interface

Parameters	Support Frame	Shield
Power [Mw]: BPP/EPP	3.3 / 4.7	0.035 / TBD
Loop no.	1	1
Pressure [Mpa]	3	3
Coolant temperature:		
inlet	100°C	100°C
outlet	150°C	150
Flow rate [kg/s]: BPP/EPP	15.8 / 22.5	0.2 /
Outer Pipe diameter (without insulation)		
inlet	50 mm	50 mm
outlet	50 mm	50 mm
Tritium generation	TBD	TBD

The secondary loop of the two water cooler (2.4.5.4) of the Coolant Purification Subsystem is considered as part of the ITER cooling system. For the layout a constant water flow with nominal inlet temperature of 35 °C and a flow rate of TBD kg/s per cooler are assumed, leading to pipe dimensions of TBD mm diameter, TBD mm wall thickness, at a velocity of TBD m/s. The outlet temperature will then vary according to the burn and dwell cycles between 60 °C and 35 °C. No significant migration of tritium from the primary coolant to the secondary side is expected.

The secondary loop of the two water cooler (2.4.3.2) of the Tritium Extraction Subsystem is considered as part of the ITER cooling system. For the layout a constant water flow with nominal inlet temperature of 35 °C and a flow rate of TBD kg/s per cooler are assumed, leading to pipe dimensions of TBD mm diameter, TBD mm wall thickness, at a velocity of TBD m/s. The outlet temperature will then vary according to the burn and dwell cycles between 60 °C and 35 °C. No significant migration of tritium from the primary coolant to the secondary side is expected. Only in case of leakage through the heat changer a not neglegible ammount of tritium can reach the secondary side.

2.6.7 Vacuum Pumping and Leak Detection Interface

The test Blanket System has a leak rate of coolant Helium lower than 10^{-8} Pa m^3 s^{-1}.

In case of loss of all the Helium coolant present in the Test Blanket System (46 Kg, in two independent loops) the pressure in the Vacuum Vessel will remain less of 5 bar.

2.6.8 Tritium Plant Interface

Molecular hydrogen isotopes (HT,H_2) are stored in uranium getter beds (2.4.3.17) of the Tritium Extration Subsystem.When the loading capacity of these beds is reached they will be transfer to the Isotope Separation System. This is expected to be needed after 6 days of nominal operation with back-to-back pulses. The rate of HT extracted is 0.04 mole/day and of H_2 18.36 mole/day.

The cold trap (2.4.3.4) of the Tritium Extraction Subsystem is periodically depressurized and warmed up to room temperature to liquefy the water which is then drained into a water container and sent to the Water Detritiation System. The rate of water extracted is ≈0.9 g/day

The adsorber beds (2.4.3.6) of the Tritium Extraction Subsystem are periodically depressurized and warmed up to room temperature. The desorbing impurities are sent to the Waste Gas System.

The cold traps (2.4.5.6) of the Colant Purification Subsystem are periodically depressurized and warmed up to room temperature to liquefy the water which is then drained into a water container and sent to the Water Detritiation System. The rate of water extracted under Design Conditions (see Table 2.1.1.4-1) is 17 g/day.

The adsorber beds (2.4.5.8) of the Colant Purification Subsystem are periodically depressurized and warmed up to room temperature. The desorbing impurities are sent to the Waste Gas System.

2.6.9 Tokamak Operations and Control Interface Interface

The BTM will remain inside the Test Port for few years.

An interruption of at least 2 days of the plasma operation is required to exchange the uranium Getter Beds of the Tritium Extraction Subsystem (see 2.1.1.2). This operation is, in the present design, required after 6 days of nominal operation with back-to-back pulses. The period of operation could be longer but this would require to increase the number of Getter Beds.

2.6.10 Building and Transporter Interface

It is intended to install the Tritium Extraction Subsystem in a so-called pit adjacent to the port of the blanket test module. A space of 2.55(1.4)m x 8m and a height of 1.65m is available in this pit. The size of the main components has been estimated and listed in Table 2.1.1.2 - 2. The integral space requirement in the pit is about 105 m^3. Not included here is the space for a control station and for electrical cabinets which should be placed at the Test Blanket Control Console outside the pit. It is also expected that supply and diposal facilities are externally available. Respective wall penetrations are listed in Table 2.1.1.2 - 3, while Table 2.1.1.2 - 4 summarizes the requirements to be supplied from locations outside of the pit. Figure 2.1.1.2. - 2 gives a preliminary arrangement of the components of the Tritium Extraction Subsystem.

The Coolant Purification Subsystem is installed in the Tritium Building.The size of the main components has been estimated and listed in Table 2.1.1.4-2. A first proposal for the geometrical arrangement is given in Figure 2.1.1.4.-2. The integral space requirement of the facility is about 50 m^3. Additional space of at least 15 m^2 in front of the facility will be needed for installation, manual operations, repair, etc. This number does not include the space for a control station and for electrical cabinets. It is also expected that supply and diposal facilities are available. Table 2.1.1.4-3 summarizes these requirements.

The Helium Coolant Subsystem will be housed in the tritium building at a floor level about 20 m above the test module. Figure 2.1.1.3-2 shows the arrangement in the tritium building with an estimated space required of about 700 m^3, excluding the helium purification subsystem.

During the operations of removing and installing of the Test Blanket Subsystem the ITER transporters are used to perform the operations that are identical to other ITER operations, such us the removing of the cryostat and bioshield plug. A special transporter (Test Blanket Remote Handling Transporter, see Section 2.4.6.1) have to be design for the operation involving the managment of the Test Blanket Subsystem. This transporter will be based on the standard transport design of ITER.

2.6.11 Hot Cell Interface

The hot cells have to handle the Blanket Remote Handling Transporter wich contains the Test Blanket Subsystem from ist location in ITER reactor. This Transporter fulfills the standard JCT design with overall dimensions of 8 m long, 3.8 m wide and 5 m high.

The following operations have been to perform in hot cell to separate the components of the Test Blanket Subsystem:

- remove the Test Blanket Subsystem from ist location in the Transporter;
- cut the tubes at designed planes;
- unfasten the bolts between the Shield and the Support Frame;
- unfasten the bolts of the mechnical connection between the BTM and the Shield.

At the end of the irradiation time foreseen for the HCPB BTM, the following operations have to be performed on the component:

- cut the BTM and remove the beryllium and orthosilicate pebble from the beds for investigation:
    - tritium release test
    - mechanical investigation
    - crush test
    - thermal cycling test
- cut probes of the structure for investigations.
    - tritium release test
    - swelling test
    - embrittelment test
    - tritium inventory determination

The following repairs have to be performed too:

- weld small leakages in the components;
- replace tubes;
- replace damaged instrumentation.

2.6.12 Other Interfaces

The particulate filters will be transferred to the waste disposal system after exchange.

Appendix A. Preliminary Electromagnetic Analysis

A-1.0 Introduction

The mechanical design of the Test Blanket Subsystem components (Blanket Test Module, Support Frame and Shield) is strongly influenced by the magnitude of electromagnetic loads due to plasma disruptions. As these components are supported by the ITER backplate, their electromagnetic loads affect also the design of this ITER component.

In this section a preliminary electromagnetic analysis in case of a centered disruption is presented. Results of the calculation are the magnetic forces acting on the Test Blanket Subsystem components.

A-1.1 Analysis Method

Eddy currents and electromagnetic forces have been calculated by means of AENEAS [1], a three-dimensional finite element method (FEM) code. This allows electromagnetic calculations in presence of non-linear ferromagnetic materials.

A-1.2 Materials

The electrical and magnetic properties of the materials taken into account in the electromagnetic calculation are listed in Table A-1.1. In particular the martensitic steel MANET is a ferromagnetic material. The magnetization curve used in the calculation is shown in Fig.A-1.1. The magnetic properties of MANET have been measured at the RWTH Aachen [2]. For 316LN Steel the vacuum permeability is assumed in the calculation

Table A-1.1: electrical and magnetic properties.

	Electrical resistivity [$\mu\Omega$m]				**magnetic properties**
	100°C	200°C	300°C	400°C	
MANET	0.657	0.732	0.806	0.881	see Fig.A-1.1
316LN	0.915	0.884	0.949	1.002	vacuum (μ_0)

A-1.3 Model Description

Fig. 2.1.1.1.4-2 and 2.1.1.1.4-3 show the geometrical model on the basis of which the electromagnetic calculation has been performed. The HCPB Blanket Test Module is placed in the lower half of the frame. The upper half is reserved for the Japanese BTM. In default of a detailed description of the Japanese BTM, the upper position is supposed to be occupied by another HCPB-BTM.

Fig.A-1.2 shows the electromagnetic FEM model used in the calculation. 1/20 of the ITER reactor has been modelled. Vacuum vessel, backplate, PF-coils and Test Module System are taken into account in the model. The vacuum vessel electrical resistivity has been chosen in order to achieve a toroidal resistance of 13.2 $\mu\Omega$. The backplate is assumed to be electrically insulated from the vacuum vessel. Its electrical resistivity results in a toroidal resistance of 7.4 $\mu\Omega$. This value takes into account the contribution of the shield blankets supported by the backplate and electrically connected to it.

The model of the Test Blanket Subsystem is shown in detail in Fig.A-1.3. The Support Frame is electrically connected to the backplate with a toothed-flange. The shields are both electrically connected to the Frame by means of a flange bolted to the back side of the Frame. The structural material of the Support Frame and Shield is 316LN steel at an average temperature of 150°C. The Test Modules are electrically insulated from the other components of the Test Module Subsystem. The structural material MANET has been considered at an average temperature of 350°C.

In Table A-1.2 the most important parameters of the FEM model are summarized. The calculation has been performed with a relatively small amount of degrees of freedom 2482 (964 for the description of the current and 1464 of the magnetization vector). A relatively rough mesh has been used for vessel and backplate; a finer mesh has been used for the test port region.

Table A-1.2.Mesh parameters

Component	Node No.	Element No.	current d.o.f.	magnetic d.o.f.
Lower BTM	511	268	301	732
Upper BTM	511	268	301	732
Vessel	414	176	172	-
Backplate, Frame and Shield	503	230	239	-
PF-coils	120	20	10	-
Total	2059	962	1023	1464

For the centered disruption a filamentary model derived from the plasma equilibrium is used. The plasma current center has a radius of 8.2 m and a height above the equatorial plane of 1.47.m. The plasma current of 21 MA decays to zero linearly in 10 ms. The poloidal field coil currents are taken at end-of-burn (EOB). The Pf-coil dimensions and current values are summarized in Table A-1.3. A toroidal field of 5.7 T at a radius of 8.1 m is also included.

Table A-1.3: PF-coils

Coil	R (mm)	Z (mm)	ΔR (mm)	ΔZ (mm)	Current (MAt)
CS Module 1	2038,0	0,0	224,0	12071,0	-37,55
CS Module 2	2263,5	0,0	205,0	12071,0	-40,33
CS Module 3	2531,1	0,0	308,0	12071,0	-61,20
PF-2	5858,0	9981,0	1553,0	1528,0	-5,76
PF-3	12892,0	7389,0	657,0	1186,0	-7,05
PF-4	15364,0	-2090,0	587,0	2222,0	-13,15
PF-5	13198,0	-7943,0	1236,0	1743,0	-4,55
PF-6	9703,0	-9583,0	1035,0	1385,0	4,47
PF-7	5859,0	-9981,0	1553,0	1528,0	8,68
PF-8	15177,0	3691,0	273,0	556,0	0,00

A-1.4 Analysis Results

The resultant forces and torques acting on the BTM at end of disruption (t=10 ms) are summarized in Table A-1.4. The values are given for the Lower BTM, the Upper BTM, the Support Frame (together with the Shield) and for the total Subsystem. The resultant torques are calculated at the geometric center of the correspondent component. The total torque is calculated at the backplate location.

Table A-1.4: Resultant forces and torques

	Lower BTM	Upper BTM	Frame and Shield	Total
x-coordinate (*) [m]	11.768	11.768	12.291	11.658
y-coordinate (*) [m]	0.000	0.000	0.000	0.000
z-coordinate (*) [m]	0.823	1.887	1.355	1.355
Force x [MN]	-0.035	0.029	-1.975	-1.981
Force y [Mn]	0.200	-0.085	0.229	0.344
Force z [Mn]	0.693	-0.201	1.108	1.600
Torque x [MNm]	2.007	2.010	-0.243	3.927
Torque y [MNm]	-0.104	0.377	-0.409	-1.085
Torque z [MNm]	-0.961	1.164	-1.742	-1.869

(*) According to the torus Coordinate System

References:

[1] P. Ruatto, „Entwicklung einer Methode zur Berechnung der elektromagnetischen Kräfte durch Magnetfeldänderungen in ferromagnetischen Strukturen und Anwendung dieser Methode auf den Plasmaabbruch in einem Tokamakreaktor", FZKA 5683, Forschungszentrum Karlsruhe (1996).

[2] K.A. Hempel, W. Salz, personal communication, RWTH Aachen, January (1995).

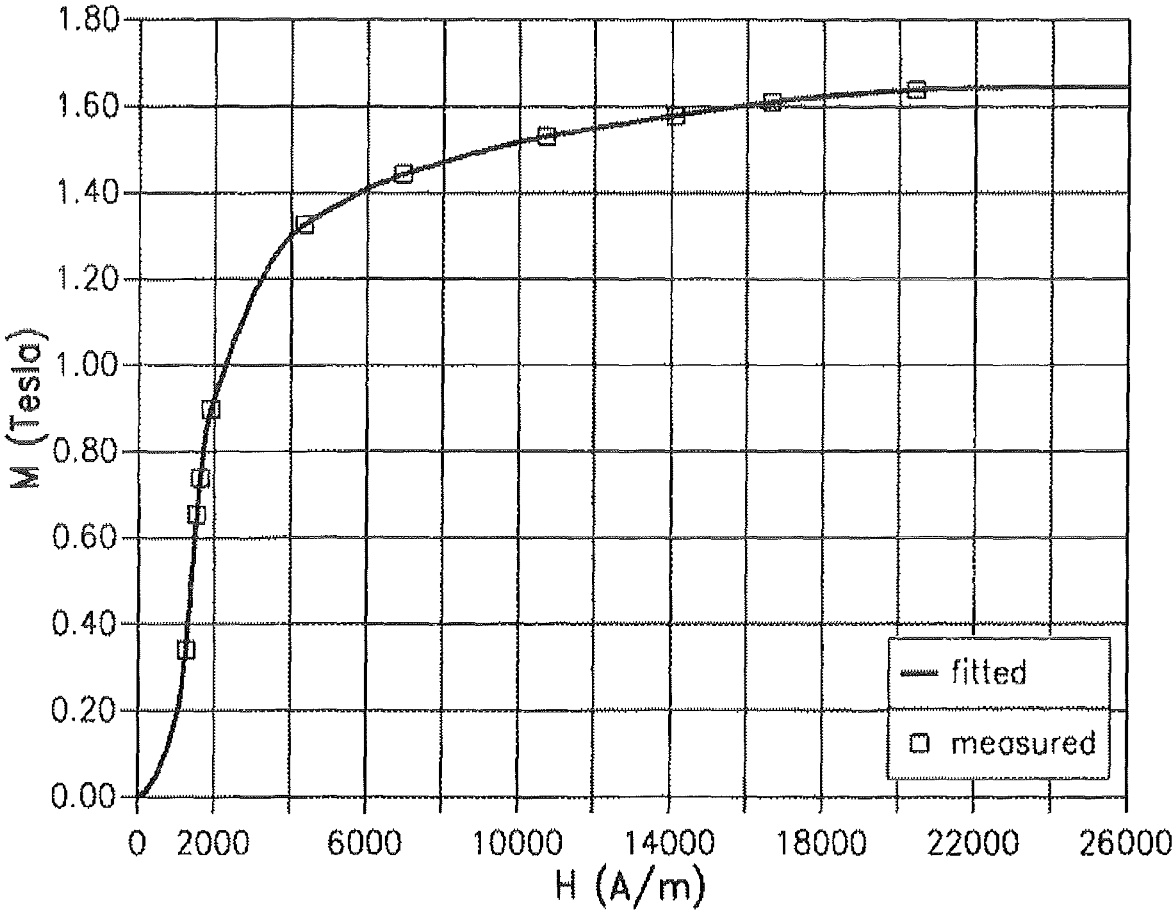


Fig. A-1.1 Magnetization curve of MANET used in the calculation

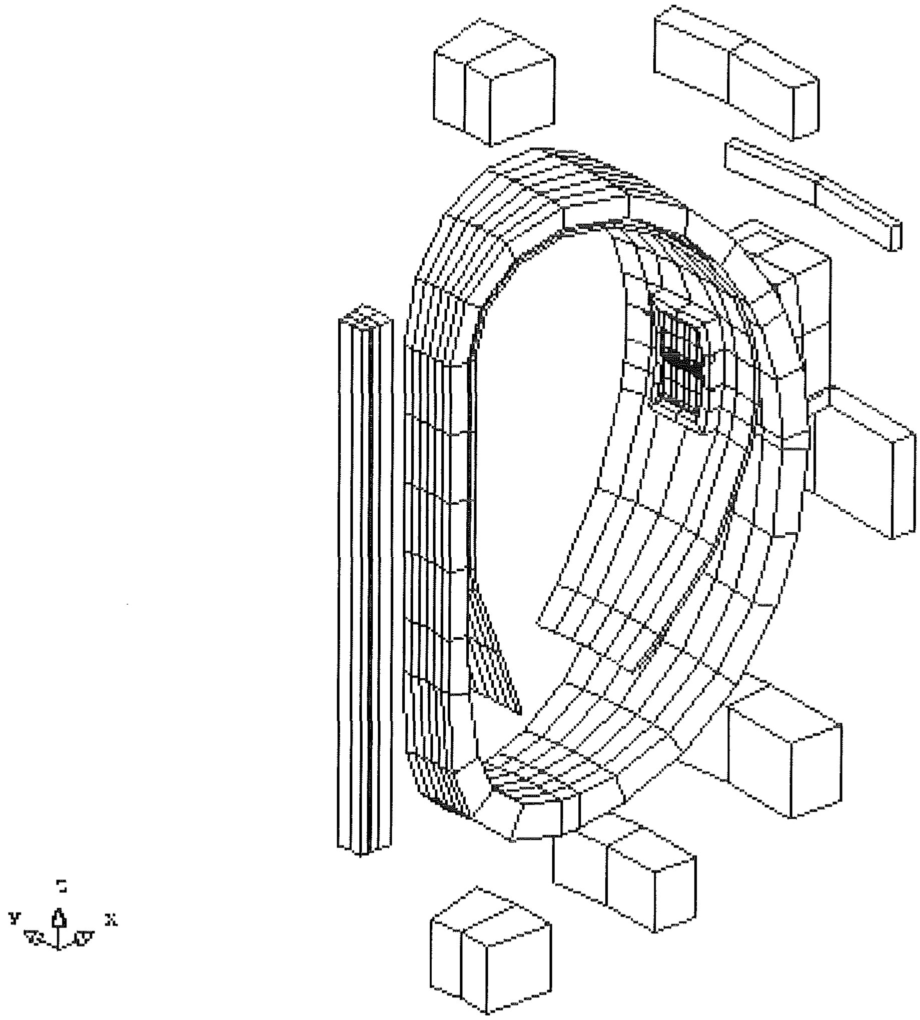


Fig. A-1.2 Electromagnetic FEM model

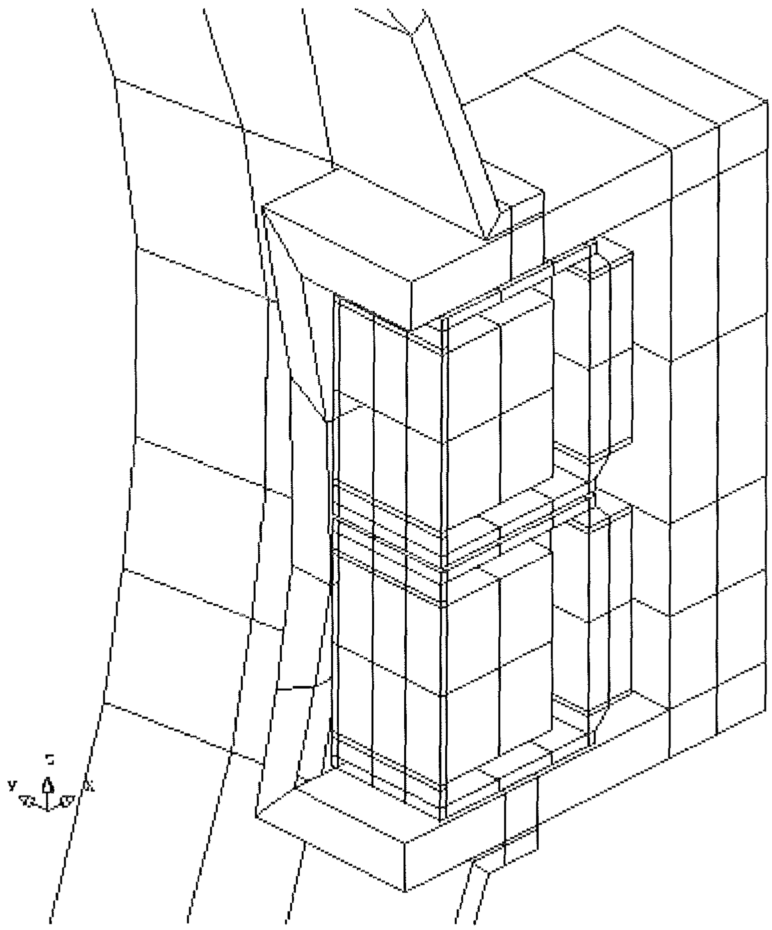


Fig. A-1.3 Electromagnetic FEM model; detail of the Test Blanket Subsystem

Appendix B. 9Cr RCC-MR properties data

Energieonderzoek Centrum Nederland
Netherlands Energy Research Foundation ECN

ECN NUCLEAR ENERGY

Forschungszentrum karlsruhe
Prof. M. Dalle Donne
Postfach 3640
D-76021 KARLSRUHE

Direct line: +31 224 56 4665
Telefax : +31 224 56 3490
Our ref. : 71099/BvdS/mh/1996/003187
Your ref. : -
Subject : 9Cr RCC-MR properties data

Petten, 2 August 1996

Dear Mario,

For your information I have attached the data on 9Cr steel mechanical properties as given in the RCC-MR. Before using the data a short explanation is needed. In the 1987 version of the code three product forms of 9Cr steel have been included in the Part Procurement Specifications:

RM 2422: Z10 CDVNb 9-1 alloy steel forgings for steam generator tube plates.
RM 2431: Z10 CDVNb 9-1 alloy steel thick plates for steam generator tube plates.
RM 2442: Z10 CDNbV 9-1 beamless chromium-molybdenum alloy steel tubes for FBR steam generator bundles.

The chemical specification for these materials is given in the 1987 issue of the RCC-MR as follows:

C	0.08 - 0.12
Mn	0.3 - 0.5
P_{max}	0.02
S_{max}	0.01
Si	0.2 - 0.6
Ni_{max}	0.2
Cr	8.00 - 9.00
Mo	0.85 - 1.05
V	0.18 - 0.25
Cu_{max}	0.1
Nb	0.06 - 0.10
Al_{max}	0.04
N	0.03 - 0.07

ECN

- 2 -

These three forms shall comply with the Properties Group 18 s as given in RCC-MR '87. This properties group 18 s I have attached to this letter. Additionally I have added the fatigue data of properties group 17 s. The reason is that in the 18 s version the fatigue limits are not issued. Originally this 9Cr steel complied with Properties Group 17 s. There fatigue data are included. In order to give you an idea of the fatigue trend I have included the relevant part of 17 s, but formally it is not given in 18 s.
The Properties Group 18 s data have been compared with large data bases of similar 9Cr steels under CEC study contract RAI-0199-VK by Curbishley et al. in January 1995. Some discrepancies have been observed for long term creep exposures at 823 K. Since this is not a relevant temperature domain for the ITER test object I do still support the use of the 18 s Properties Group. The new low activation Cr steel have the potential of improving the "classical" properties given in the RCC-MR, which is another reason not to bother with the findings of the study contract.
I fear that the fax processing affects the quality of the figures. The information is therefore sent to you by surface mail to make sure it can be read. I apologize for the resulting delay.
If you require additional information or explanations, please do not hesitate to contact me.

Yours sincerely,

ir. B. van der Schaaf
Member Management Team
ECN Nuclear Energy

TABLE OF CONTENTS

A3.18S.1 INTRODUCTION

A3.18S.2 PHYSICAL PROPERTIES

A3.18S.2.1 COEFFICIENT OF THERMAL EXPANSION

This is the average coefficient of linear thermal expansion α between 20 °C and the temperature indicated θ.

α is given as a function of θ by the following:

. Table 2.1.1

θ (°C)	20	100	200	300	400	500	600	700
$\alpha\ 10^{-6}.(°C)^{-1}$	10,4	10,8	11,2	11,6	11,9	12,2	12,5	12,7

. Figure 2.1.2

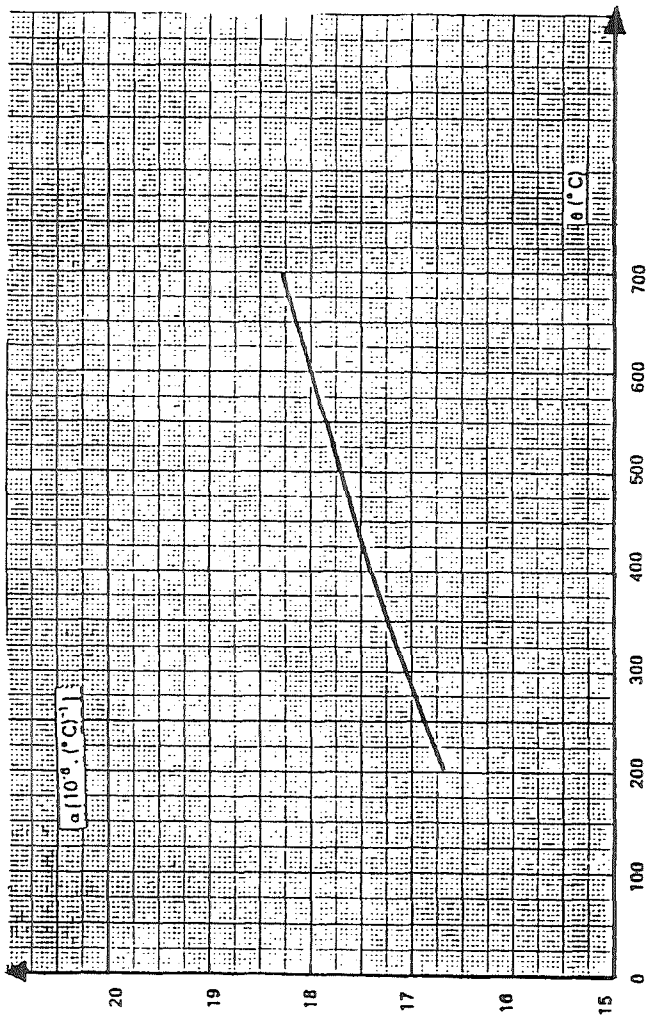


Figure A3.18S.2.1.2

AVERAGE COEFFICIENT OF LINEAR THERMAL EXPANSION

A3.18S.2.2 YOUNG'S MODULUS

Young's modulus, E (MPa) is given as a function of the temperature θ by the following:

. Formula 2.2.1

$$E = 207300 - 64{,}58\ \theta \qquad 20 \le \theta\,°C \le 500$$
$$E = 235000 - 120\ \theta \qquad 500 \le \theta\,°C \le 600$$

. Table 2.2.2

θ °C	20	100	200	300	350	400	450	500	550	600
E (10^3)N/mm2)	206	201	194	188	185	181,5	178	175	163	151

. Figure 2.2.3

A3.18S.2.3 POISSON'S RATIO

This is taken as 0,3 within the elastic range.

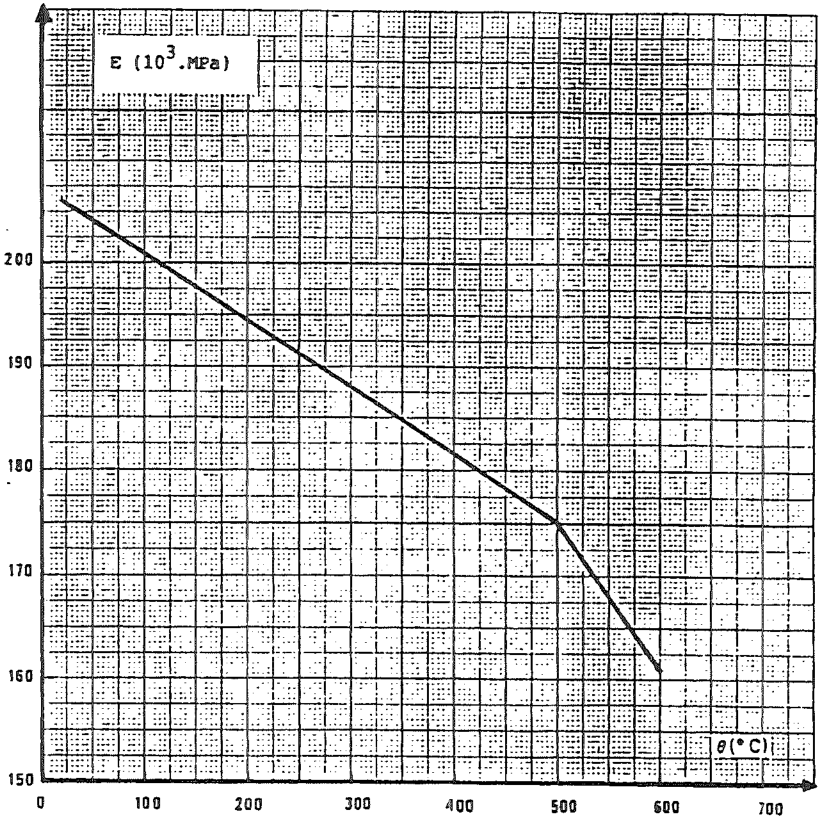


Figure A3.18S.2.2.3

YOUNG'S MODULUS

A3.18S.3 TENSILE STRENGTH PREPERTIES

A3.18S.3.1 MINIMUM AND AVERAGE YIELD STRENGTH AT 0,2% OFFSET

The minimum yield strength at 0,2% offset $(R_{0,002})_{min}$ is given as a function of temperature θ by the following:

. Formula 3.1.1

$$(R_{0,002})_{min} = 409,56 - 0,51595\,\theta + 1,9521 \cdot 10^{-3}\,\theta^2 - 2,7776 \cdot 10^{-6}\,\theta^3$$

This formula is applicable for $20 \le \theta\ °C \le 700$

. Table 3.1.2

θ (°C) / $(R_{0,002})_{min}$ MPa	20	50	100	150	200	250	300	350
Tubes	420	400	375	367	362	359	355	349
Plates	400	390	375	367	362	359	355	349

θ (°C) / $(R_{0,002})_{min}$ MPa	400	450	500	550	600
Tubes	338	320	292	254	203
Plates	338	320	292	254	203

. Figure 3.1.3

The average yield strength at 0,2% offset is given as a function of temperature θ by:.

. Formula 3.1.4

$$(R_{0,002})_{avg} = 564,25 - 0,7108\,\theta + 2,6894 \cdot 10^{-3}\,\theta^2 - 3,8267 \cdot 10^{-6}\,\theta^3$$

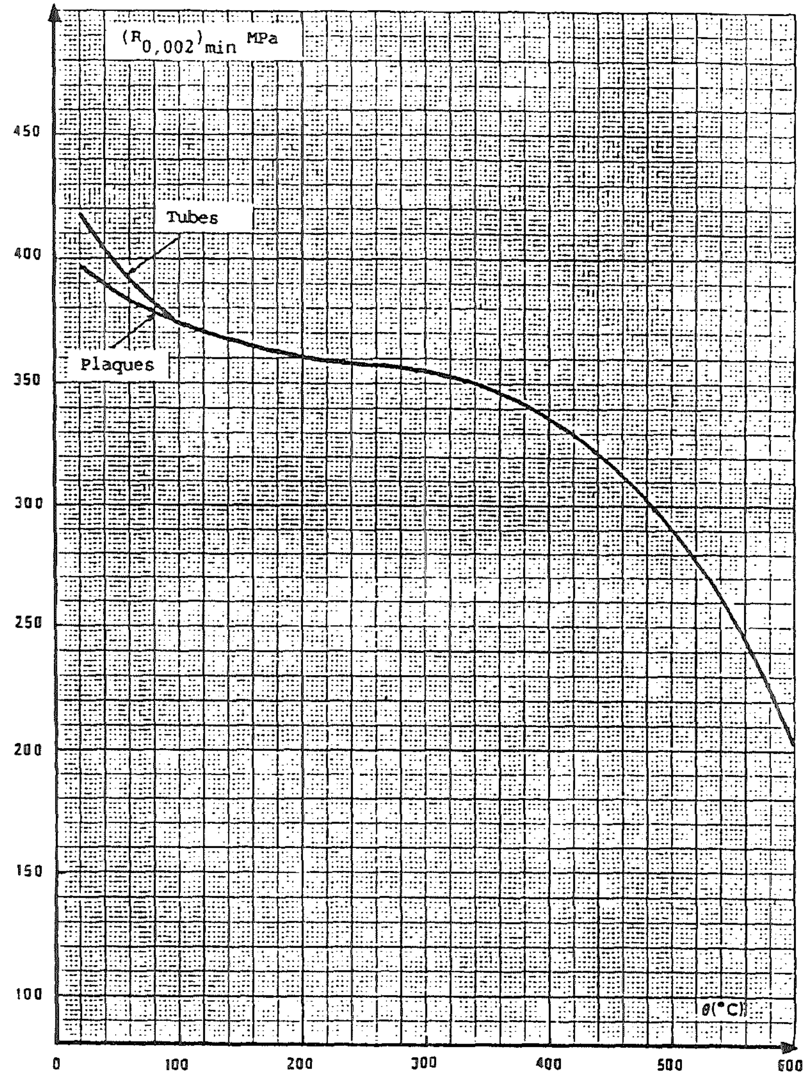


Figure A3.18S.3.1.3

YIELD STRENGTH AT 0,2% OFFSET

A3.18S.3.2 MINIMUM AND AVERAGE TENSILE STRENGTH

The minimum tensile strength $(R_m)_{min}$ is given as a function of temperature θ by the following:

. Formula 3.2.1

$$(R_m)_{min} = 598{,}06 - 0{,}9922\,\theta + 4{,}6386.10^{-3}\,\theta^2 - 9{,}199.10^{-6}\,\theta^3 + 4{,}535.10^{-9}\,\theta^4$$

This formula is applicable for 20 < θ°C 600

. Table 3.2.2

θ (°C)	20	50	100	150	200	250	300	350	400
$(R_m)_{min}$ MPa	580	559	536,5	525	519	514	506	493	471

θ (°C)	450	500	550	600
$(R_m)_{min}$ MPa	439	395	340	273

. Figure 3.2.3

The average tensile strength $(R_m)_{avg}$ is given as a function of temperature θ by:

. Formula 3.2.4

$$(R_m)_{avg.} = 722{,}02 - 1{,}198\,\theta + 5{,}60.10^{-3}\,\theta^2 - 11{,}06.10^{-6}\,\theta^3 + 5{,}475.10^{-9}\,\theta^4$$

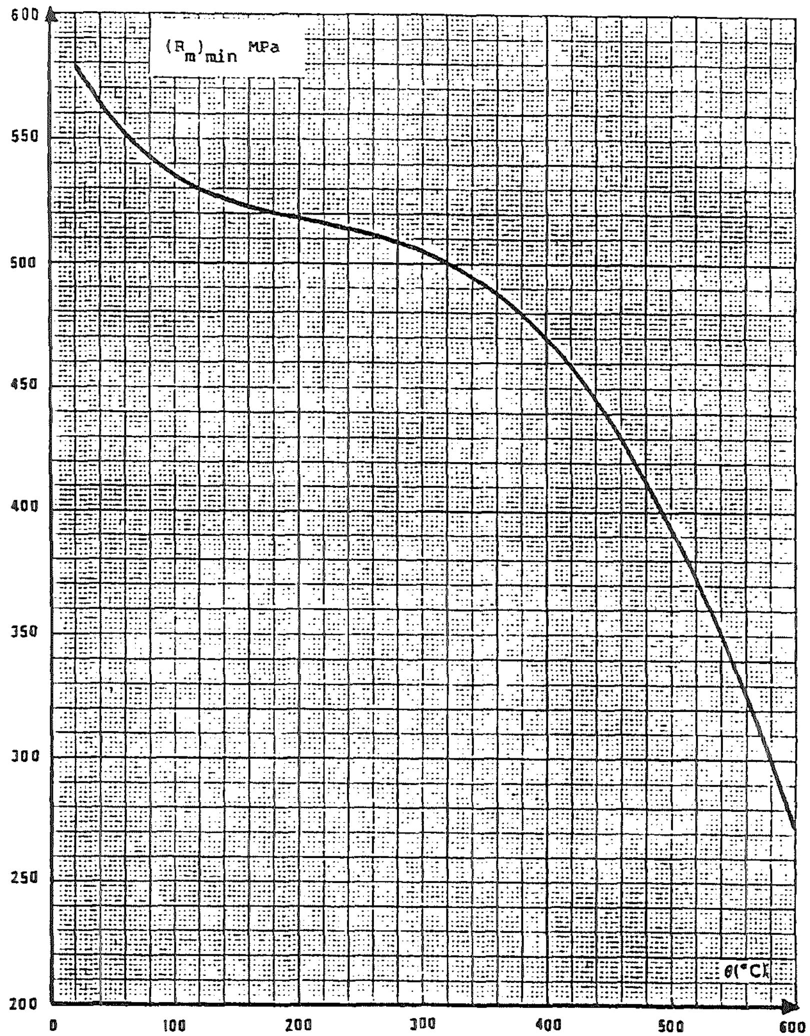


Figure A3.18S.3.2.3

MINIMUM TENSILE STRENGTH

A3.18S.4 NEGLIGIBLE CREEP CURVE (To be issued)

A3.18S.5 ANALYSIS DATA

A3.18S.5.1 VALUES OF S_m

The maximum allowable stress S_m is given as a function of temperature θ by the following:

. Table 5.1.1

θ (°C)	20	50	100	150	200	250	300	350	400	450	500	550	600
S_m (MPa)	193	193	193	193	192	190	187	183	174	163	146	126	101

. Figure 5.1.2

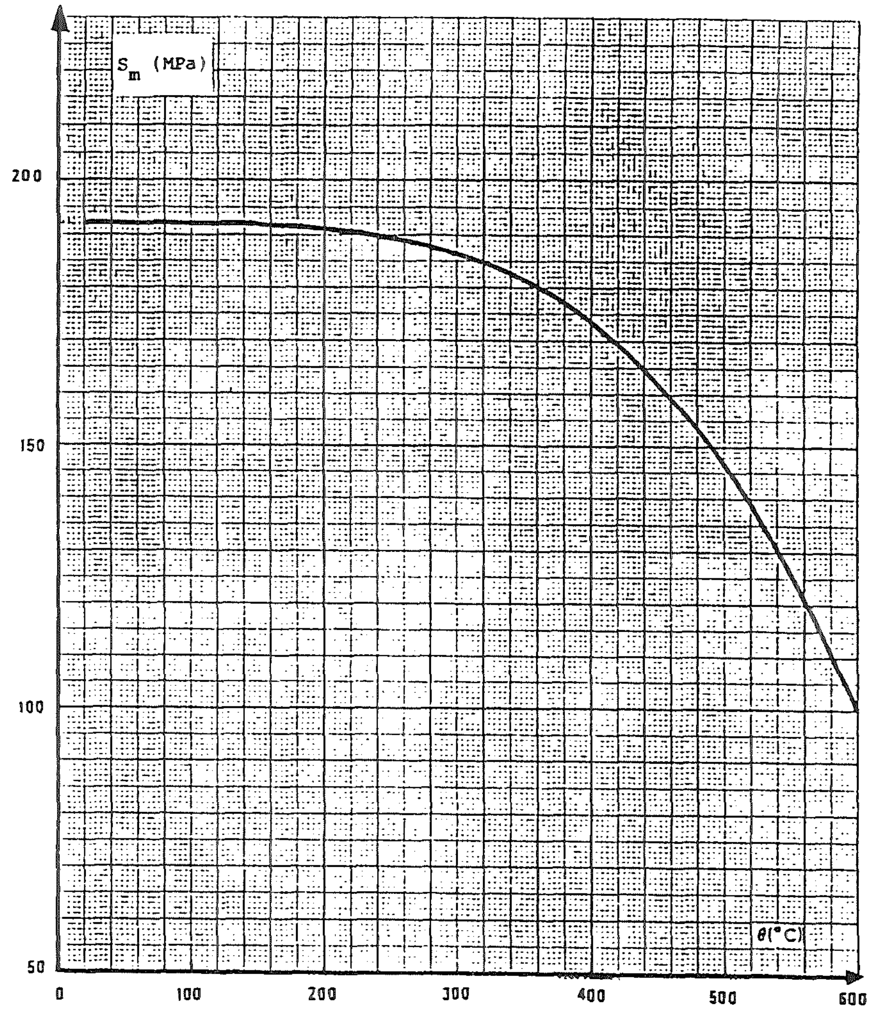


Figure A3.18S.5.1.2

ALLOWABLE STRESSES S_m

A3.18S.5.2 VALUES OF St

The maximum allowable stress S_t is given as a function of the temperature θ and time t by the following table or figure in which S_t is expressed in MPa, θ in °C and t in h:

. Table 5.2.1

. Figure 5.2.2

t(h) θ°C	1	10	30	10^2	$3x10^2$	10^3	$3x10^3$	10^4	$3x10^4$	10^5	$3x10^5$
425		293	293	287	271	255	241	227	216	204	194
450		277	261	245	232	219	207	196	186	176	167
475		238	225	212	201	189	180	169	161	151	143
500		207	196	185	175	166	157	146	138	129	121
525		181	171	161	152	142	134	125	117	108	101
550		157	149	139	131	121	113	105	97	89	81
575		137	129	119	111	102	94	86	78	70	63
600		118	110	101	93	84	76	68	61	53	47
625		100	92	83	75	67	59	52	45	39	33
650		83	75	66	57	51	45	38	33	27	23
675		67	59	51	45	37	32	27	22		

Table A3.18S.5.2.1

MAXIMUM ALLOWABLE STRESS S_t IN MPa

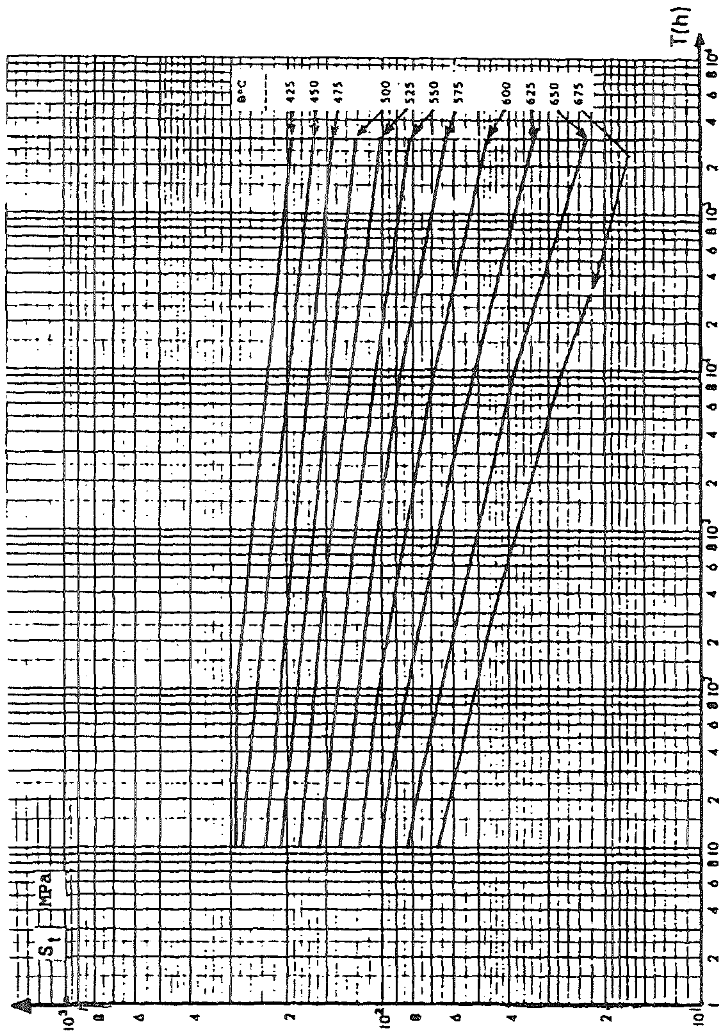


Figure A3.18S.5.2.2

MAXIMUM ALLOWABLE STRESSES S_t IN MPa

A3.6S.5.3 CREEP RUPTURE STRESS : S_r

The minimum value of the creep rupture stress S_r is given as a function of temperature θ and time t by the following:

- Table 5.3.1
- Figure 5.3.2

t(h) / θ°C	1	10	30	10^2	3×10^2	10^3	3×10^3	10^4	3×10^4	10^5	3×10^5
425		439	439	430	406	382	362	341	324	307	292
450		415	392	368	348	328	311	294	279	264	251
475		357	338	318	301	284	270	254	241	227	215
500		311	294	277	262	247	233	219	207	194	182
525		272	257	241	228	213	201	187	175	162	151
550		238	224	209	196	182	170	157	145	133	122
575		206	193	179	167	153	141	129	117	105	95
600		177	165	151	139	126	114	102	91	80	71
625		150	138	124	112	100	89	78	68	59	50
650		124	112	99	88	77	67	57	49	40	34
675		100	89	77	67	56	48	40	33		

Table A3.18S.5.3.1

MINIMUM VALUES OF CREEP RUPTURE STRESS S_r (MPa)

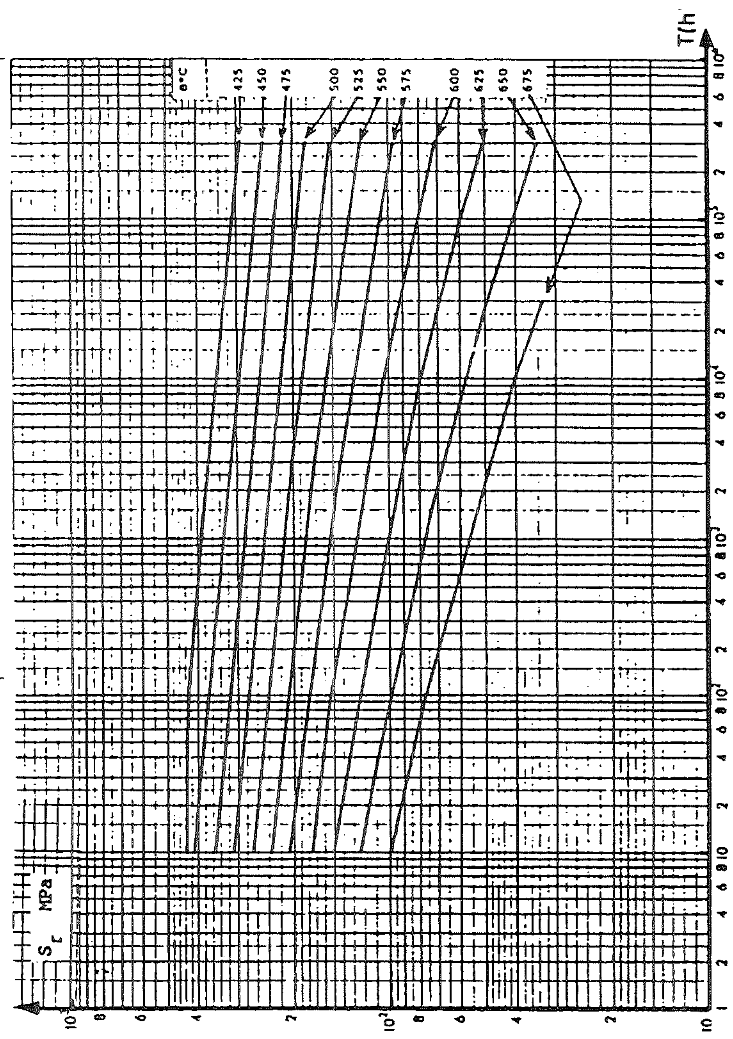


Figure A3.18S.5.3.2

MINIMUM VALUES OF CREEP RUPTURE STRESS Sr (MPa)

A3.18S.5.4 SATURATION FATIGUE CURVES

To be issued.

A3.18S.5.5 ISOCHRONOUS CURVES, CREEP STRAIN

To be issued.

A3.18S.5.6 STRESSES S_{Rh} AND S_{Rc}

To be issued.

A3.18S.5.7 SYMMETRIZATION COEFFICIENT K_s

To be issued.

A3.18S.5.8 FATIGUE - CREEP INTERACTION DIAGRAM

To be issued.

A3.18S.5.9 CYCLE CURVES, VALUES OF K and K

To be issued.

A3.6S.6 ANALYSIS DATA

A3.6S.6.1 MONOTONIC TENSILE HARDENING RULE

A3.6S.6.1.1 For plastic strain limited to 1,7%

The average tensile hardening rule is given by the following:

. Formula 6.1.1.1

$$(\sigma)_{avg} = (R_{0,002})_{avg} \times C\ (\theta) \times \varepsilon_p^{\,n}$$

with:

$$(R_{0,002})_{avg} = 564,25 - 0,7108\,\theta + 2,6894.10^{-3}\,\theta^2 - 3,8267.10^{-6}\,\theta^3$$

In this formula, ε_p (%) designates the plastic strain induced by stress σ (MPa) at temperature θ (°C).

This formula is applicable for:

θ	20	200	300	400	450	500	525	550	550
C	1,0855	1,088	1,0985	1,1085	1,119	1,0985	1,0985	1,0605	1,0435
n	0,0511	0,0525	0,0585	0,0640	0,0690	0,0582	0,0451	0,0367	0,0264

The formula expressing the minimum hardening rule is obtained by replacing $(R_{0,002})_{avg}$ by $(R_{0,002})_{min}$ in the above formula.

. Table 6.1.1.2

. Figures 6.1.1.3

A3.18S.6.1.2 For total strain reaching the distributed elongation

To be issued.

θ = 20 °C		
σ (MPa)	ε_p (%)	ε_t (%)
400	0,00038	0,19455
425	0,00124	0,20755
450	0,00381	0,22225
475	0,01097	0,24155
500	0,02993	0,27265
510	0,04409	0,29166
520	0,06448	0,31690
530	0,09360	0,35068
540	0,13494	0,39708
550	0,19324	0,46023
560	0,27493	0,54678
570	0,38874	0,66544
580	0,54635	0,82790
590	0,76340	1,04981
600	1,06070	1,35197
610	1,46580	1,76192
615	1,71971	2,01825

θ = 200 °C		
σ (MPa)	ε_p (%)	ε_t (%)
350	0,00023	0,18030
375	0,00087	0,19379
400	0,00297	0,20876
425	0,00942	0,22807
450	0,02798	0,25949
460	0,04252	0,27917
470	0,06404	0,30584
480	0,09563	0,34257
490	0,14162	0,39371
500	0,20807	0,46531
510	0,30339	0,56577
520	0,43914	0,70666
530	0,63113	0,90383
540	0,90103	1,17884
550	1,27790	1,56086
560	1,80104	2,08914

θ = 300 °C		
σ (MPa)	ε_p (%)	ε_t (%)
350	0,00065	0,18690
375	0,00210	0,20165
400	0,00632	0,21918
425	0,01781	0,24398
450	0,04731	0,28678
460	0,06888	0,31366
470	0,09946	0,34957
480	0,14253	0,39796
490	0,20274	0,46343
500	0,28633	0,55240
510	0,40163	0,67302
520	0,55967	0,83639
530	0,77499	1,05703
540	1,06663	1,35399
550	1,45945	1,75213
555	1,70353	1,99888

Table A3.1BS.6.1.1.2

MEAN STRESS σ INDUCING PLASTIC STRAIN ε_p AND TOTAL STRAIN ε_t AT TEMPERATURE θ

θ = 400 °C		
σ (MPa)	ε_p (%)	ε_t (%)
300	0,00021	0,16554
325	0,00074	0,17984
350	0,00235	0,19523
375	0,00689	0,21355
400	0,01888	0,23931
420	0,04044	0,27190
440	0,08361	0,32609
450	0,11875	0,36674
460	0,16736	0,42086
470	0,23414	0,49315
480	0,32526	0,58978
490	0,44879	0,71882
500	0,61522	0,89076
510	0,83811	1,11917
520	1,13493	1,42149
530	1,52801	1,82008
535	1,76927	2,06410

θ = 450 °C		
σ (MPa)	ε_p (%)	ε_t (%)
300	0,00075	0,16907
325	0,00240	0,18475
350	0,00703	0,20341
370	0,01574	0,22334
390	0,03377	0,25259
410	0,06973	0,29977
420	0,09889	0,33454
430	0,13910	0,38036
440	0,19413	0,44100
450	0,26890	0,52139
460	0,36982	0,62792
470	0,50515	0,76885
480	0,68547	0,95478
490	0,92433	1,19925
500	1,23891	1,51944
510	1,65095	1,93709
512	1,74734	2,03461

θ = 500 °C		
σ (MPa)	ε_p (%)	ε_t (%)
275	0,00028	0,15743
300	0,00126	0,17269
320	0,00382	0,18668
330	0,00648	0,19506
340	0,01082	0,20511
350	0,01780	0,21780
360	0,02888	0,23459
370	0,04623	0,25766
380	0,07308	0,29023
390	0,11417	0,33703
400	0,17635	0,40492
410	0,26949	0,50377
420	0,40763	0,64763
430	0,61061	0,85632
440	0,90620	1,15763
450	1,33301	1,59015
456	1,61166	1,87156

Table A3.1BS.6.1.1.2 (cont. 1)

MEAN STRESS σ INDUCING PLASTIC STRAIN ε_p AND TOTAL STRAIN ε_t AT TEMPERATURE θ

θ = 525 °C		
σ (MPa)	ε_p (%)	ε_t (%)
275	0,00016	0,16289
300	0,00114	0,17865
320	0,00476	0,19411
330	0,00943	0,20469
340	0,01829	0,21947
350	0,03479	0,24189
360	0,06502	0,27803
370	0,11942	0,33836
380	0,21583	0,44069
390	0,38413	0,61490
395	0,50963	0,74336
400	0,67374	0,91043
405	0,88761	1,12726
410	1,16542	1,40803
415	1,52515	1,77071
417	1,69687	1,94362

θ = 550 °C		
σ (MPa)	ε_p (%)	ε_t (%)
270	0,00017	0,16581
280	0,00045	0,17223
290	0,00118	0,17909
300	0,00298	0,18702
310	0,00727	0,19746
320	0,01728	0,21360
330	0,03999	0,24245
335	0,06026	0,26578
340	0,09025	0,29884
345	0,13437	0,34603
350	0,19891	0,41364
355	0,29283	0,51062
360	0,42875	0,64961
365	0,62448	0,84841
370	0,90493	1,13192
375	1,30480	1,53486
380	1,87227	2,10540

θ = 600 °C		
σ (MPa)	ε_p (%)	ε_t (%)
230	0,00013	0,15244
240	0,00063	0,15957
250	0,00297	0,16853
255	0,00628	0,17515
260	0,01310	0,18529
265	0,02696	0,20245
270	0,05472	0,23353
275	0,10965	0,29177
280	0,21699	0,40242
282	0,28413	0,47089
284	0,37135	0,55943
286	0,48442	0,67383
288	0,63076	0,82148
290	0,81980	1,01185
292	1,06358	1,25696
294	1,37740	1,57210
296	1,78070	1,97672

Table A3.18S.6.1.1.2 (cont. 2)

MEAN STRESS σ INDUCING PLASTIC STRAIN ε_p AND TOTAL STRAIN ε_t AT TEMPERATURE θ

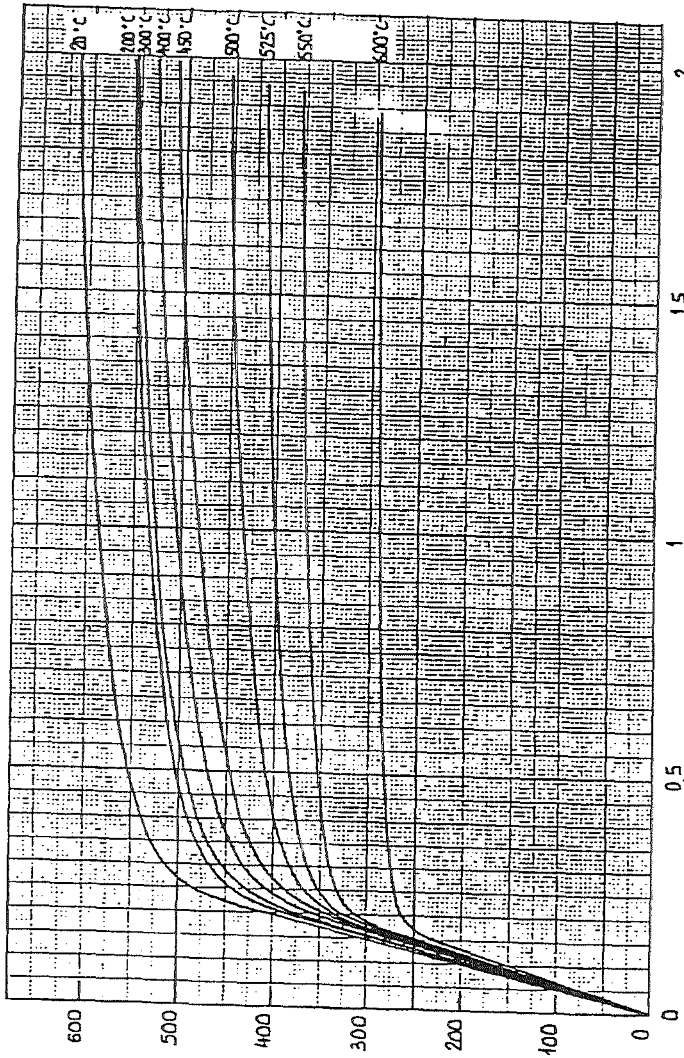


Figure A3.18S.6.1.1.3

MEAN TENSILE CURVES

A3.18S.6.2 Bilinear curves

To be issued.

A3.18S.6.3 Creep-strain law

The mean value of the minimun creep rate is given by:

. formula 6.3.1

$$\log \varepsilon_m = 27{,}3 + 0{,}025\,265\,\sigma + 3{,}2172 \log \sigma - 35{,}594/T$$

where

ε_m = minimum creep rate in % h^{-1}

σ = stress in MPa

T = temperature in degrees Kelvin $T = 273 + \theta$

This formula is applicable for $480 \le \theta\ °C \le 700$

$10^{-5} \le \varepsilon_m\ (\% h^{-1}) \le 10^{-1}$

A3.18S.6.4 Fatigue curve

To be issued.

A3.18S.6.5 Maximum allowable strain : D_{max}

D_{max} = 1%.

A3.17S.6 ANALYSIS DATA (Cont.)

A3.17S.6.1 TENSILE HARDENING RULE

(Not supplied)

A3.17S.6.2 BILINEAR CURVES

(Not supplied)

A3.17S.6.3 CREEP STRAIN LAW

(Not supplied)

A3.17S.6.4 FATIGUE CURVES

The allowable strain range is given as a function of temperature and the number of cycles by the following:

- Table 6.4.1 (to be issued)
- Figure 6.4.2

Where applicable, the fatigue curves may be extrapolated for numbers of cycles greater than 10^6 by a straight line passing through the last two points of each curve in the N $\overline{\Delta\varepsilon}$ log-log diagram.

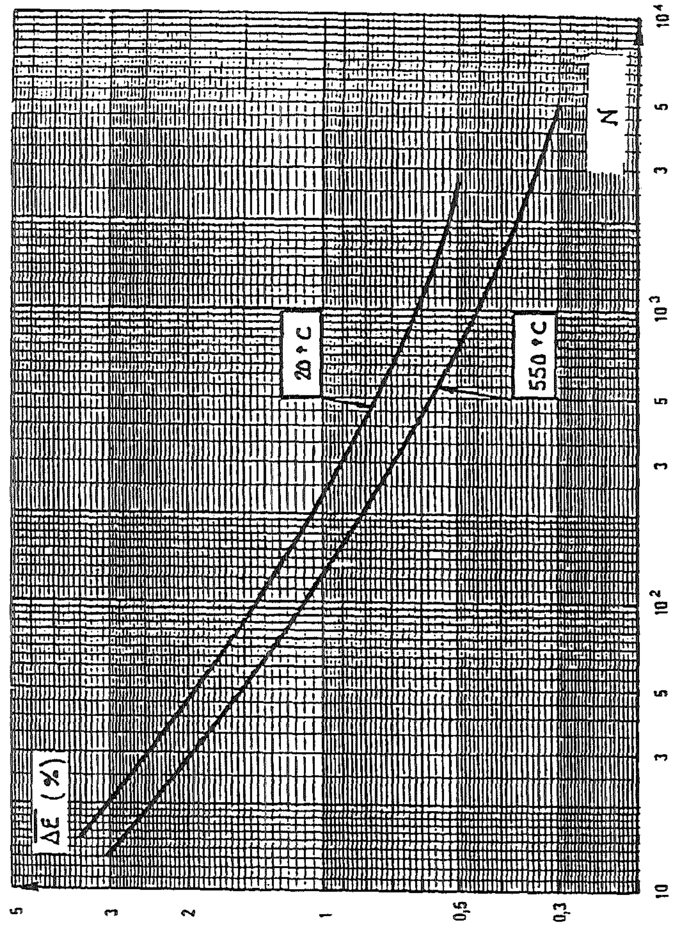


Figure A3.17S.6.4

FATIGUE CURVES

$\overline{\Delta\varepsilon}$ IN %, FUNCTION OF NUMBER OF CYCLES N