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Design Description Document for the U.S. Helium-Cooled Solid Breeder Test Blanket Module

Report to the ITER Test Blanket Working Group (TBWG)

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A. Ying, M.A. Abdou, P. Calderoni, S. Sharafat, M. Youssef, Z. An, A. Abou-Sena, E. Kim and G. Wen
University of California Los Angeles

S. Reyes
LANL, Livermore, California

S. Willms
LANL, Los Alamos, New Mexico

R. Kurtz
PNL, Richland, Washington

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0. SUMMARY

Recognizing that a final selection between solid and liquid breeders cannot be made prior to fusion testing, the US has selected a helium-cooled solid breeder concept with ferritic steel structure and beryllium neutron multiplier as one of the candidate breeder blankets for ITER TBM testing. The Objectives of solid breeder blanket testing during the first phase of ITER operations are focused on exploration of fusion break-in phenomena and configuration scoping. Specific emphasis is placed on first wall structural response, evaluation of neutronic parameters, assessment of thermomechanical performance and characterization of tritium release behavior. The concept is based on the use of lithium ceramic pebbles as a breeder material, whose complex thermomechanical interactions inside an integrated blanket system can be addressed only in a fusion environment. The US ITER testing approach for this concept is to design unit cell/submodule test articles rather than testing a fully independent TBM. The test program emphasizes international collaboration, including collaborative R&D, sharing of common ancillary equipment, and possible co-development of TBMs. The design operating conditions of the main helium coolant for the proposed unit cell/submodules are similar to those of other neighboring modules and submodules, in which any special requests to the coolant operating conditions (such as temperatures) will be handled through a much smaller component, such as a helium coolant conditioner located in the port cell area. This leads to one coolant supply line and one coolant return line running between the port area and the TCWS building per half port. To maximize ITER testing, the tritium concentration and gas composition from each breeder purge gas line will be analyzed at the port cell area before merging with other purge gas lines for tritium extraction at the tritium building.

The unit cell/submodule test article designs focus on particular technical issues of interest to all parties. A unit cell occupies a port area of about 19.5 x 21 cm and is housed behind another party's structural box, while a submodule takes up a testing space of a quarter port 73 x 91 cm and has its own structural box[1-2]. Two distinct design approaches have been considered to fulfill their testing objectives: 1) design the unit cell/submodule for low temperature operation, or a look-alike approach; and 2) refer to a reactor blanket design and use engineering scaling to reproduce key parameters under ITER wall loading conditions, so that phenomena under investigation can be measured at their reactor-like level. The two approaches result in two different sets of operational parameters, the low temperature scenario being used for neutronics assessment and the high temperature scenario for thermo-mechanic and tritium release performance evaluation.

The design and analysis for the US ITER solid breeder blanket test articles are discussed in Chapter 3, while the engineering description of the proposed solid breeder test program is presented in Chapter 2. R&D plan is discussed in Chapter 4.

1. FUNCTIONS AND REQUIREMENTS

1.4 INTERFACE REQUIREMENTS

The ITER testing for the US helium-cooled solid breeder concept with ferritic steel structure option is not to have independent ancillary equipment but rather a partial or complete sharing of the helium line and auxiliary systems. The sharing of the auxiliary system also includes tritium extraction subsystems. This reduces the number of helium lines that will connect the port with the TWCS vault and the tritium building, while also reducing the space requirements for the mechanical attachment, the TWCS vault and the tritium building. It is, however, desired to have independent coolant temperature control from the neighbouring TBM unit cells/submodule as well as in-situ tritium measurement from the purge lines. This engages the use of helium conditioning components (such as valve and controller, heater, and sensors) and tritium measurement systems to be placed in the port cell area using a cask solution. This scheme moves the interface between the service lines and the ITER building (such as TCWS and tritium building) to the piping integration cask. Other interfaces that impact a test blanket module involve interface between test blanket object and the machine, interface between test blanket performance/data and machine control, and interface between test blanket object and hot cell. Concerning the piping and auxiliary equipments for the submodule, three interfaces as listed in Table 1 have been defined. The interface 1 concerns the pipe joints, where the connection/disconnection operations will be performed during the replacement of the submodule. This interface located at the back of the submodule consists of piping for various service and instrumentation lines as shown in Figure 1, including three cooling pipes (one inlet, one outlet and one by-pass line) providing the submodule with high pressure helium coolant (~8 MPa) and three sets of one supply pipe and one return pipe providing the 0.1 MPa helium for the purging of the tritium produced in the submodule. The back side of the submodule needs to make room for the shear keys and for the flexible support cartridges. The size of each pipe is listed in Table 2. A detailed evaluation of the space requirements, taking into account space for tool access during a TBM exchange, extra shielding requirement for bends, and thermal insulation, is still needed. The 2nd interface is defined at the boundary of the back of the port plug structure, which forms the unit to be transferred to the ITER hot cell building for refurbishment/removal and installation of the old and new submodules.

The interfaces are subdivided into the following categories as summarized in Table 1: interface with ITER machine; interface with services lines; interface with diagnostics and control; and interface with TBM integration. The interface with the ITER machine is through port plug and bioshield plug, which provides functions of vacuum and radiation shielding, respectively. The port plug structure, which is still under design, consists of frame, shield plug and associated cooling lines. It provides the vacuum boundary between the TBM and ITER machine and mechanical support to the TBM. The port plug structure is designed according to the port and associated test plan.

The bioshield plug is a part of the bioshield, which is installed in the port and is designed for removal in pieces. It provides access for remote maintenance activities during a TBM exchange inside the vacuum vessel. The bioshield plug design is TBM test port specific.

The TBM will be integrated into the port plug for installation and during a TBM exchange at the hot cell building. It has been proposed to attach the port plug into a mounting port, which has the same connection interface at the back of the port plug structure. This allows various tools (including bore welding tools) to fasten/tighten/weld all necessary pipe connections.

The operating parameters from the TBM are connected to ITER control system through TBM Data Management System (TBM DMS) to ensure proper performance of TBM and to protect the machine by triggering FPSS (fusion power shutdown system) in the case of any catastrophic failure. The TBM DMS acquires performance and diagnostic data, displays alarms, creates the TBM operational database, and communicates with ITER Control System.

TABLE 1.1

Summary of Interface Requirements for the US Solid Breeder Submodule TBM Pipe dimensions at the front of the shield block

Interface Category	Location/component	Remark
Service Line Interfaces	Back of the Submodule	Operational boundary for submodule replacement
	Back of the Port Plug Structure	Operational boundary for transferring the submodule to the hot cell building
	Piping Integration Cask	Operational boundary for replacing the services/ instrumentation lines
Machine Interface	Bioshield Plug	As part of ITER bioshield for radiation protection, this is easily removable to allow tool access to cut/weld pipes during a TBM exchange process
	Port Plug Structure (should be designed according to port and associated test plan)	As a vacuum boundary between the TBM and ITER machine to support the TBM mechanically and accommodate piping penetrations
TCWS and Tritium building interfaces	Port cell area through the piping integration cask	Helium purge gases from different lines are merged into one line to process into tritium building. Helium coolant lines from TBMs are merged into one coolant return line connected to TCWS
Diagnostics and Control Interface	ITER Control System	To communicate with ITER control system through monitoring of the operation state of each TBM subsystems to ensure it is operating within the proper operational envelope or to issue a protective action in case of a catoptrical failure in TBM
TBM Integration Interface	Hot Cell Building	TBM port plug mounting stand to facilitate TBM integration and removal from/installation into the port plug
Plasma Interface	First wall	A 2 mm-thick beryllium layer

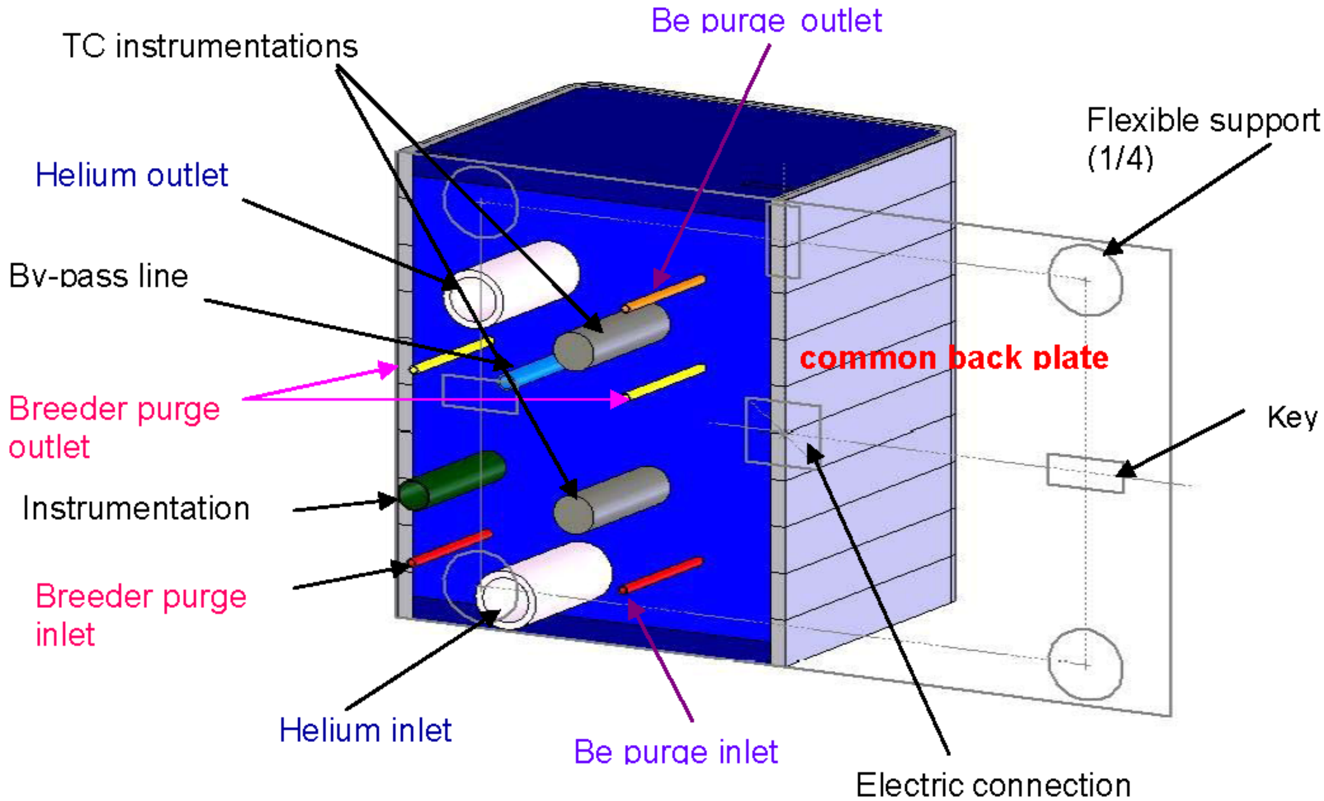

Figure 1.1 Interface between the back of the module and the front of the shield plug

TABLE 1.2

Legend	Pipe Dimension (OD/ID), mm*	Note
Quarter port submodule	Back plate area: 73 x 91 cm^2	
Helium-coolant inlet	73.0/66.9	Pipe is curved/bent inside the shield plug; maximum velocity 37 m/s
Helium-coolant outlet	73.02/66.9	Pipe is curved/bent inside the shield plug; maximum velocity 50 m/s
Helium coolant by-pass	33.41/27.86	Max. velocity 25 m/s
Breeder purge, inlet	13.716/10.414	
Breeder purge, outlet 1	13.716/10.414	
Breeder purge, outlet 2	13.716/10.414	
Multiplier purge, inlet	13.716/10.414	
Multiplier purge, outlet	13.716/10.414	
Instrumentation, TC connector 1	76.2	10 pairs TCs
Instrumentation, TC connector 2	76.2	10 pairs TCs
Instrumentation, neutronics	50	

* The pipe size has not yet included any thermal insulation layer

1.6 PROPOSED US SOLID BREEDER TEST PLAN IN ITER

The US strategy for ITER TBM includes participation in testing a helium-cooled solid breeder concept with FS structure and Be neutron multiplier. All ITER Parties have such a solid breeder concept as one of their options. In this case, the US will not provide an independent TBM, but rather will collaborate with the EU and Japan using their ancillary equipment. The US will contribute unit cells and sub-module test articles that focus on particular technical issues of unique US expertise and are of interest to all parties. A unit cell will occupy a port area of about 19.5 x 21 cm and be housed behind another party's structural box (Fig. 1.2), while a submodule will take up a testing space of a quarter port 73 x 91 cm and have its own structural box (Fig. 1.3). The unit cell approach includes testing three unit cells simultaneously, which provides multiple test data and allows statistical significance on test results to be analyzed. The decision to test one or both of these two options will be made in a few years and will coincide with the TBWG test program and the US budgetary situation. These test blanket units will be designed and inserted into the helium-cooled ceramic breeder test port (Port A) according to the testing strategy;,during the first phase of which, ITER testing synchronizes ITER operational characteristics. It includes:

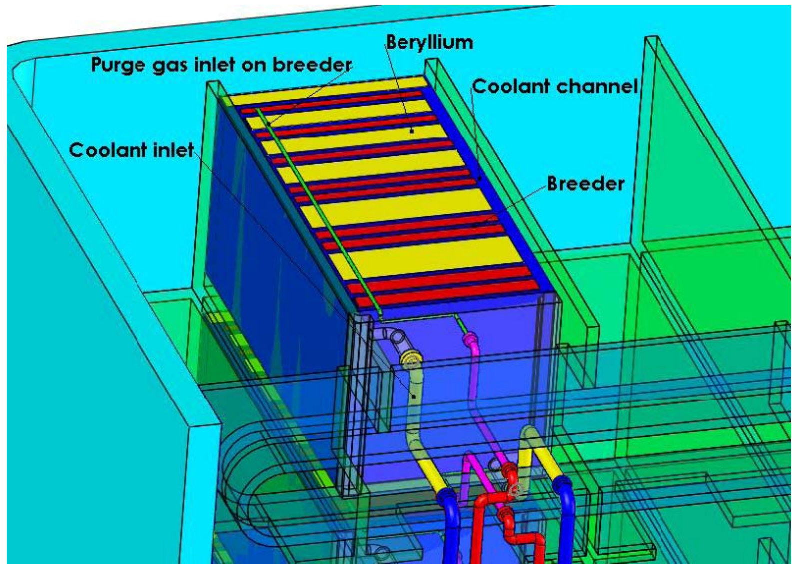

Figure 1.2 Proposed solid breeder unit cell array for neutronics tests

- First wall performance and transient electro-magnetic tests during the H-phase (EM/S). Testing objectives during H-Phase focus on the evaluation of predictive capability on the test blanket module's structural thermomechanical performance in response to an integrated fusion load of electromagnetic, thermal, and mechanical forces. In addition, neutronics and tritium production rate prediction tests will be performed during the D-D phase.
- Neutronics and tritium production rate prediction tests will be performed during the early DT-phase (NT)**.** Testing objectives focus on the evaluation of tritium breeding performance and the validation of neutronic code prediction and nuclear data.
- Tritium breeding, release and thermomechanics explorations tests during the D-T phase (TM). The objectives are to study configuration effects on tritium release and

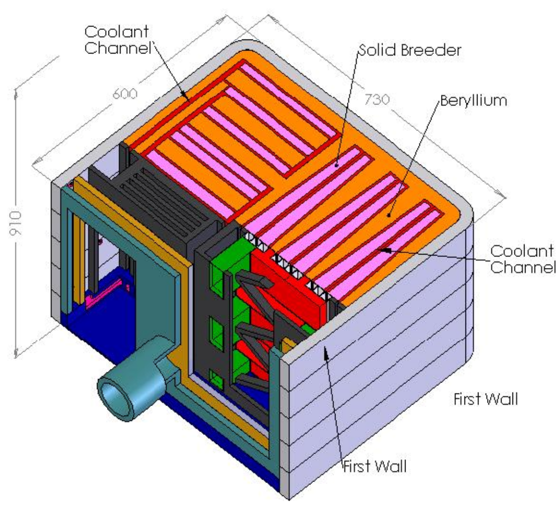

Figure 1.3 Proposed quarter-port test blanket submodule

pebble bed thermomechanical performance. The data can be used to optimize configuration aspects of solid breeder blanket designs.

• Initial study of irradiation effects on performance during the DT-phase (PI). Since several thermo-physical properties of breeding materials show the largest changes after initial exposure to irradiation, understanding their impacts on blanket performance will guide the design.

This testing strategy, as illustrated in Figure 1.4, calls for three to four unit cells (Fig. 1.5) /submodules (Fig. 1.6) to be sequentially inserted into the designated test port (Port A) from day 1 of ITER operation. However, not only due to the limited space available in the TCWS building, but also because the operating conditions of the helium coolant for the proposed submodules are similar to that of other neighboring modules/submodules, it is planned to share the same helium loop with the neighboring party or parties. Any special requests to the coolant operating conditions (such as temperatures) are handled through a helium coolant conditioner located near the port area. This leads to only one coolant supply and one coolant return line running between the port area and the TCWS building per half of a port. To maximize the use of ITER testing, the tritium concentration and gas composition from each breeder purge gas line will be analyzed at the test port area before merging it with other purge gas lines for tritium extraction process at the tritium building.

Operational Year	-2	-1	1	2	3	4	5	6	7	8	9	10	11	12
ITER Master Schedule			H-H			D	Low Duty D-T			High Duty D-T			Refurbish-ment	
# of burn pulses/year						1	750	1000	1500	2500	3000	3000		
1) Structural &EM unit cell/submodule (EM/S)														
Fabrication and qualification														
Operation in ITER														
Post Examination														
2) Neutronics unit cell/submodule (NT)														
Fabrication and qualification														
Operation in ITER														
Post Examination														
3) Thermomechanics-tritium unit cell/submodule* (TM)														
Fabrication and qualification														
Operation in ITER														
Post Examination														
4) Partially integrated module* (PI)														
Fabrication and qualification														
Operation in ITER														
Post Examination														

* Test 3 TM and Test 4 PI can be combined into one test, which utilizes ITER Operational Year 7 to 10.

Figure 1.4. Test plan for the US Solid Breeder Test Blanket Program

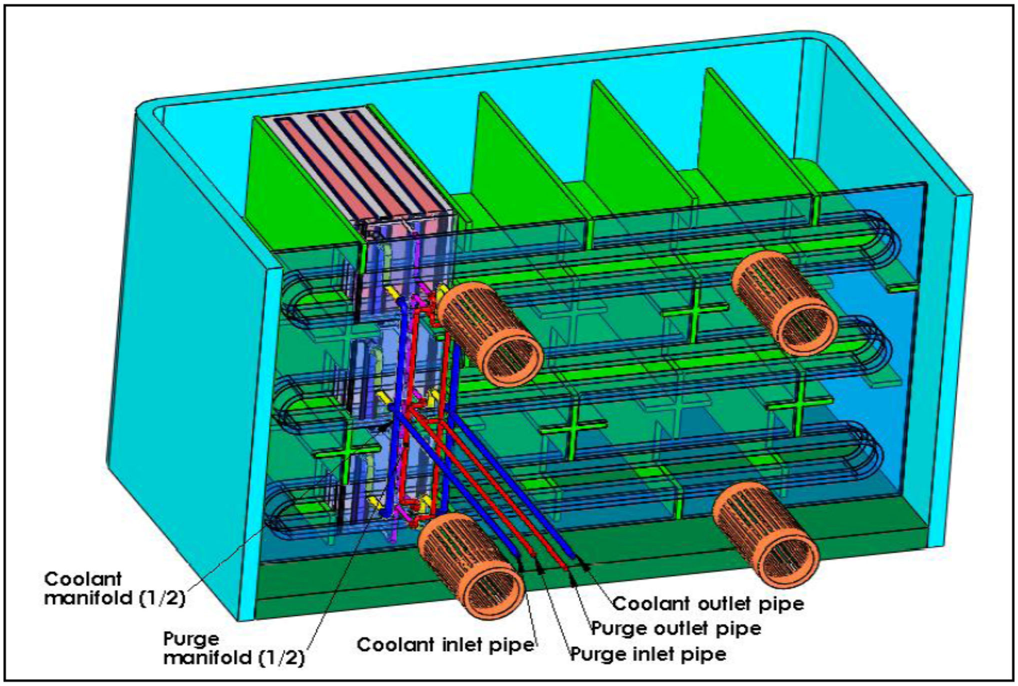

Figure 1.5. Proposed solid breeder unit cell array for thermomechanics tests

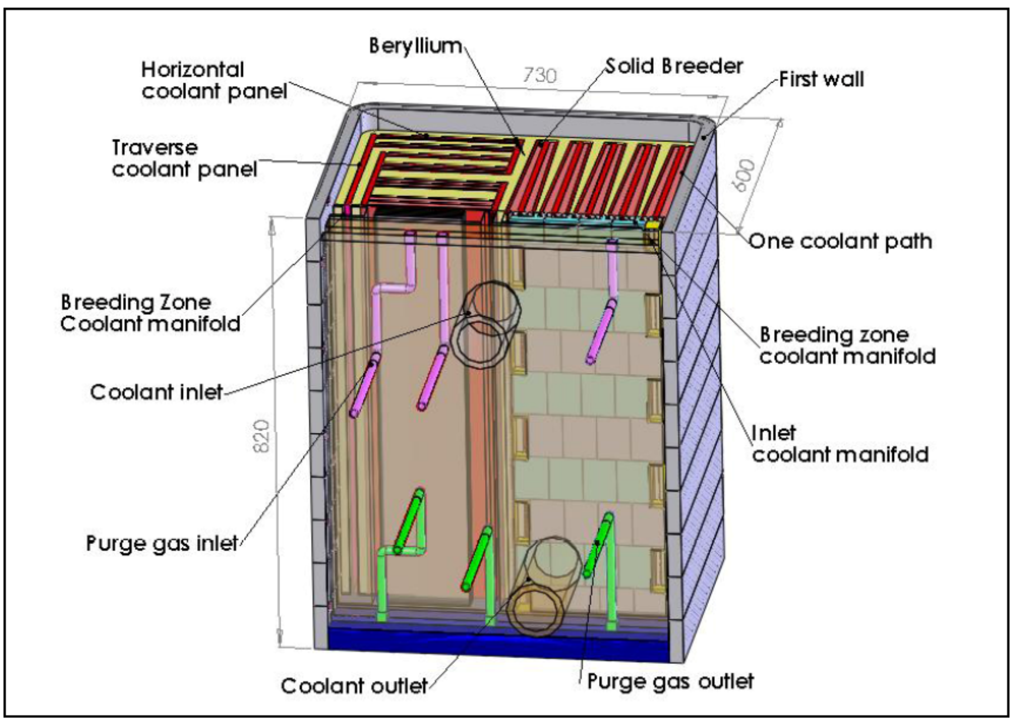

Figure 1.6. Proposed solid breeder submodule design for neutronics tests

2. ENGINEERING DESCRIPTION

Summary

The proposed ITER TBM for the helium-cooled solid breeder concept with ferritic steel structure option is not to have US independent ancillary equipment but rather to have a partial or complete sharing of other parties' helium line and auxiliary systems. The sharing of the auxiliary system also includes tritium extraction subsystems. This implies that the US plans to collaborate with EU and JA on the development and installation of helium cooling and tritium extraction systems, although details of such collaboration are yet to be defined. Technically, this helps reduce the number of helium lines that will connect the port with the TWCS vault and with the tritium building, the space requirements for the mechanical attachment, the TWCS vault, and the tritium building. However, an effective integrated scheme is needed to ensure that each party's needs are taken into consideration. For example, it is desired to have independent coolant temperature control as well as tritium measurement from the purge lines of the US' test blanket unit cells/submodule, which calls for helium coolant conditioning components and tritium measurement systems to be installed in the port cell area (or a cask solution). This scheme for the proposed ancillary equipment arrangement is illustrated in Figure 2.1. As shown, there are three cooling pipes (one inlet, one outlet and one by-pass line) that provide the submodule with a high pressure helium coolant (~8 MPa). Three sets of one supply pipe and one return pipe provide the 0.1 MPa helium for the purging of the tritium produced in the submodule.

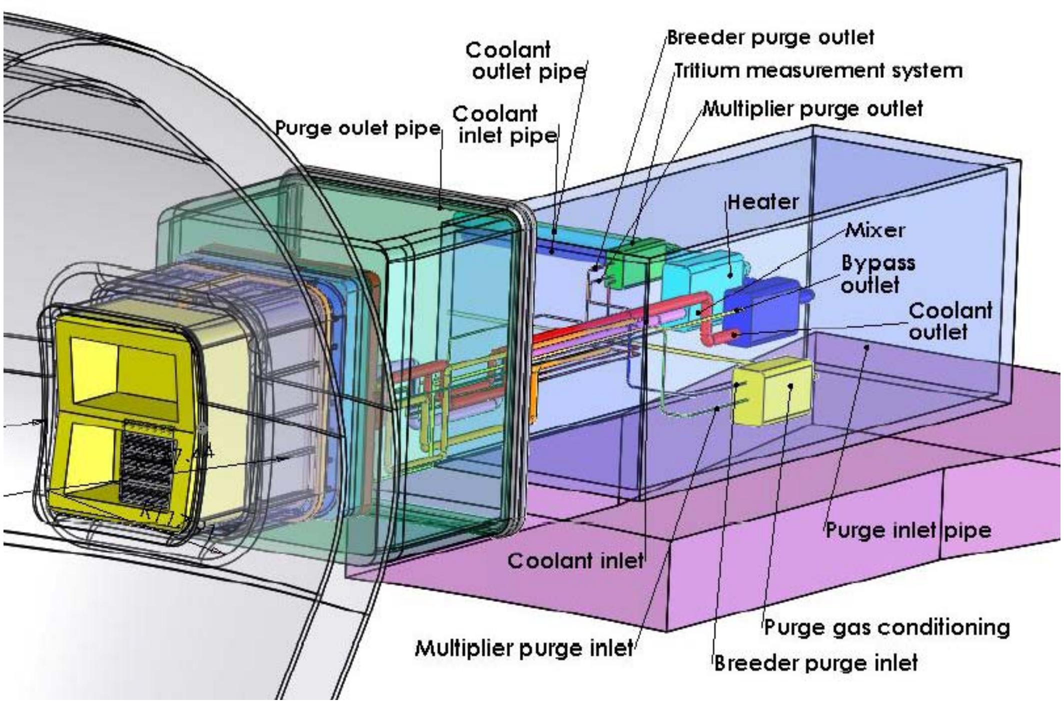

Figure 2.1. Schematic view of the US solid breeder TBM with ancillary equipment at the port cell area

The principal components in the US helium cooled solid breeder test blanket system include:

1. Test Blanket Unit Cell/Submodule (TBM) and associated auxiliary lines
2. Helium Coolant Conditioner System
3. Tritium Measurement System (installed after year 4 of ITER operations)

In addition, a neutronic measurement system, designed to perform dedicated measurement of tritium production and neutron fluxes and spectra, will be installed in the port area.

4. Neutron Measurement System (present only during years 4 and 5 of ITER operations).

2.1 TEST BLANKET UNIT CELL/SUBMODULE
2.1.1 SYSTEM DESCRIPTION AND INTERFACE

Two development approaches have been considered for unit cell/submodule designs, which lead to two distinct configurations as shown in Figures 2.2 and 2.3, including: 1) design the unit cell/submodule for low temperature operations, or a look-alike approach; and 2) refer to a reactor blanket design and use engineering scaling to reproduce key parameters under ITER wall loading conditions, so that phenomena under investigation can be measured at a reactor-like level. A unit cell will occupy a port area of about 19.5 x 21 cm and be housed behind EU's structural box (Fig. 1.XX). In addition, within each phase of testing, three unit cells occupying a column will be tested simultaneously to allow statistical significance on test results to be analyzed. Two unit cell configurations have been proposed according to the test objectives, including a low temperature operation neutronics unit cell (Fig. 2.2), and an act-alike thermomechanics unit cell (Fig. 2.3). The unit cell to be considered for the 1st TBM electro magnetic tests will have a configuration similar to that of the neutronics unit cell. A stream of coolant from the primary loop will be fed into the unit cell array common manifold located at the back manifold region of the structural box and is subsequently divided into three paths for cooling three unit cells. The detailed design of the array manifold is a subject to be discussed once a collaborative agreement is settled. This is also applied to the design of the purge gas lines. The dimensions and arrangement inside a US submodule involves a total toroidal width of 73 cm and a poloidal height of 91 cm, as shown in Figure 2.4. These dimensions are based on a frame structure design with a 10 cm width. The first and side walls of the TBM have a thickness of 2.8 cm and are made of low activation ferritic steel (F82H) with content of ~53% by volume. The helium coolant is routed toroidally through the first and side walls in alternating directions. The US solid breeder test blanket unit cell/submodule will be inserted into the horizontal port No. A, together with the EU and JA's Helium Cooled TBMs. This proposal calls for an international collaboration, which requires a collaborative agreement among the involved parties to execute and resolve all the interface and integration related issues including port plug support and shield structures, piping layouts and space

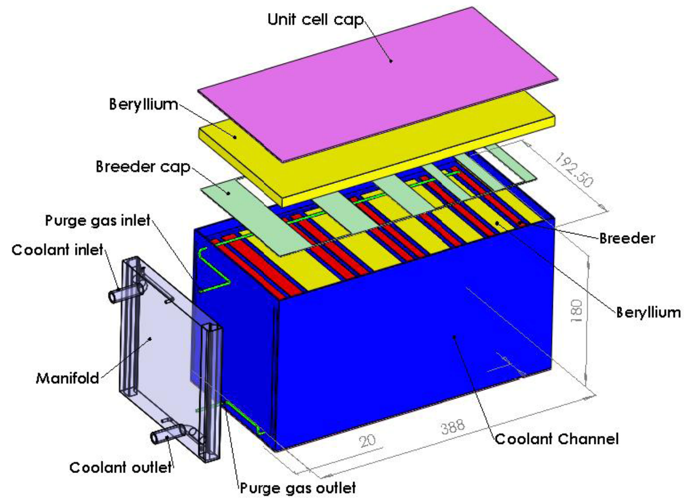

Figure 2.2. Low temperature option Neutronics unit

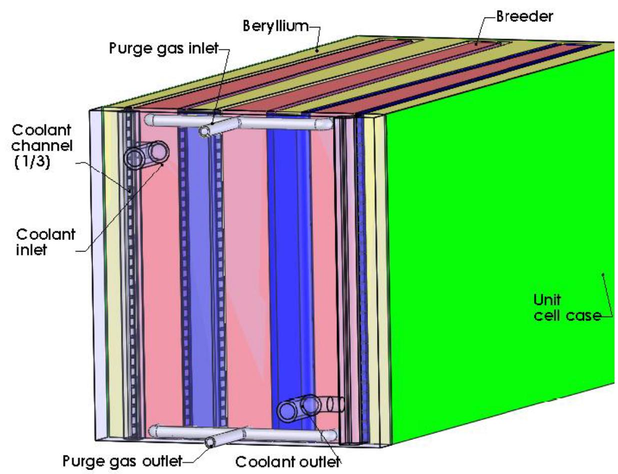

Figure 2.3. Reactor-relevant temperature option Thermomechanics unit cell

sharing around the port cell area, sequences of remote handling operations that are required for removal and re-installation of TBMs, etc.

In the submodule design, the breeding zones are housed behind a ferritic steel U-shaped FW structural box, as shown in Figure 2.4. The overall FW thickness is 28 mm, including a coolant channel of 16 mm x 14 mm and a front wall thickness of 5 mm (Figure 2.5). The pitch between the coolant channels is 18. mm. The FW is designed to remove a total deposited heat of 0.307 MW, based on the contribution of the average surface heat flux of 0.3 MW/m$_2$ and nuclear heating deposition on the front and side walls of the FW structures with a neutron wall load of 0.78 MW/m$_2$.

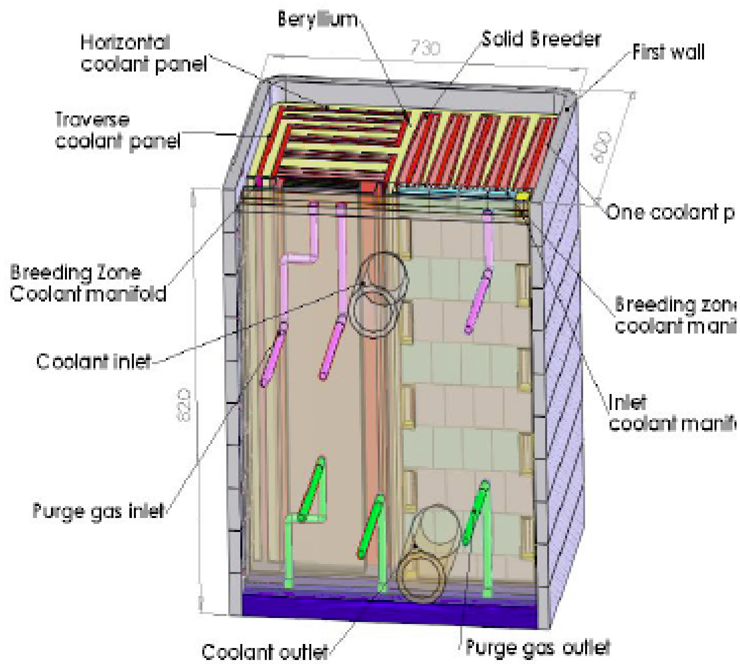

Figure 2.4. Neutronics quarter-port submodule

Because a relatively high velocity is needed to ensure an adequately high heat transfer coefficient for locally removing a surface heat load of 0.5 MW/m$_2$, the first wall design features a reduced coolant flow area by grouping five coolant flow channels in a series into one coolant flow path (bottom picture of Figure 2.5). In the low temperature scenario, the 8 MPa helium coolant enters the submodule at a rate of 0.755 kg/s and a temperature of 100$_o$C and is subsequently distributed into 2 paths to remove the heat generated in the breeder region, which amounts to 0.45 MW. In the low temperature operation design, the helium flows first in the breeder region channels, then in the first wall, since the goal is to keep the breeding material temperature low. In the high temperature design, the scheme is reversed, since the main challenge becomes the cooling of the first wall structure. The 8 MPa helium coolant enters the submodule at a rate of 0.9 kg/s at a temperature of 300ºC (a typical value of helium coolant inlet temperature applicable to any helium cooled blanket designs with F S as a structural material) and is subsequently distributed into 10 first wall cooling paths for surface heat removal. However, this high coolant flow rate gives a lower coolant outlet temperature as compared to typical values of 500$_o$C needed for achieving a high thermal efficiency in helium-cooled FS blanket designs, thus about 10% of the flow is by-passed away from the breeding zones after the first wall cooling. The remaining coolant in the submodule is divided into four paths for cooling upper and lower

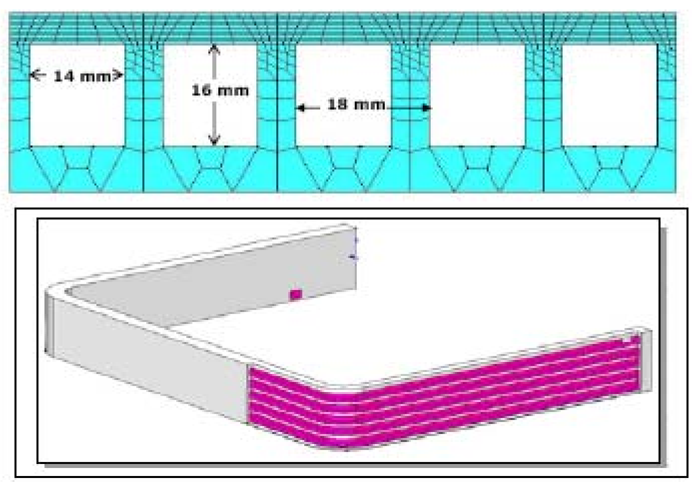

Figure 2.5. Cross-Sectional View of the FW (top) and 5-Channel Pass Detail (bottom)

caps and two breeding configurations. The design parameters as listed in Table 2.1 show that the temperature and stress magnitudes of the first wall are within the maximum allowable limits of FS structural material with the design parameters described. The helium coolant entering into the unit cell may be coming directly from the supply line of the helium loop, and thus it may be necessary to raise its temperature from 300 ^{o}C to 350 ^{o}C using an external heater located in the port cell area in order to reproduce coolant operating temperatures and replicate prototype breeder temperature levels so that the exit temperature reproduces a typical prototype helium outlet temperature of 500^{o}C. The total heat generated inside a unit cell is about 35.8 kW, which is removed by a coolant flow rate of 0.046 kg/s. This implies that a total flow rate of 0.138 kg/s is needed to cool the proposed US unit cell test array.

TABLE 2.1 Key Operating Parameters for Solid Breeder Submodule

Parameter	Design value	
Test objective	Thermomechanics	Neutronics
Test article	Submodule	Submodule
Test article size, m^3	0.73x0.91x0.6	0.73x0.91x0.6
Surface heat flux, MW/m^2	0.5 (maximum) 0.3 (average)	0.5 (maximum) 0.3 (average)
Neutron wall load, MW/m^2	0.78	0.78
Helium coolant pressure, MPa	8	8
Helium inlet/outlet temperature, °C	300/500	100/300
Total power to be removed, MW	0.785	0.785
Mass flow rate to test article, kg/s	0.9	0.755
Helium temperature rise from first wall, °C	53	76
Bypass mass flow rate, kg/s	0.08	0
Mass flow rate to breeding zone, kg/s	0.82	0.755
He temperature rise from breeding zone, °C	146	124
First wall maximum temperature, °C	538	484
First wall maximum stress, MPa	268	268

Within a submodule, two design configurations are housed behind the first wall structural box to maximize the testing goals. In one configuration, both Be and breeder beds are placed perpendicular to the FW facing the plasma region. In the second configuration a parallel layout is considered. The later option resembles the blanket concept considered in the US ARIES-CS and HAPL designs. This allows the effect of configuration on tritium breeding performance to be studied from these submodule tests. The layer configuration inside a neutronics submodule layout shown in Figure 2.4 consists of a number of ceramic breeder (CB) and Be multiplier packed bed layers separated by cooling panels and arranged parallel to the first wall. The helium coolant goes through a series of 3 toroidal passes, each pass consisting of a parallel-flow configuration through several parallel cooling panels (PCP) that are 6 mm-thick. There are 4 PCP in the first pass, 5 PCP in the 2nd pass, and 6 PCP in the 3rd pass. The internal manifolds for these passes consist of four 12 mm thick traverse cooling panels (TCP). Both the PCP and the TCP have 53% by volume F82H structure content. There are nine CB beds whose thicknesses in the radial direction are 9, 10, 10, 10, 11, 13, 14, 18, and 18 mm, respectively, whereas the six horizontal beryllium beds have thicknesses of 20, 22, 25, 30, 40, and 48 mm, respectively, following the FW. There are two traverse Be beds on the right side of the sub-module that are 20 mm-thick each, as shown in Fig. 2.4. The edge-on sub-module consists of five (5) canister units arranged to be perpendicular to the FW. At the front edge, these units are separated by a distance of 23.23 mm. The far left unit (unit#1) is at a distance of 14.96 mm from the side wall of the TBM. Each unit is composed of two side and one central TCP, one PCP, and two CB beds. The TCP and PCP are 6 mm-thick while the central TCP is 12 mm-thick. They have F82H structure content of ~55% and helium coolant is routed and returned in the radial direction through these panels. The thickness of the CB traverse beds is 10 mm at the front end of each unit and gradually increases as we move towards the back of the sub-module. At the interface with the back manifold, the units are separated from each other with a distance of 9 mm. Thus, the amount of CB increases, while the amount of Be decreases, as one moves towards the back locations. There is a central Be bed which separates the parallel and the edge-on sub-modules, which has a thickness of 24.12 mm at the front and 7 mm at the back. In both sub-modules, single-sized pebbles are assumed for the CB and beryllium beds with a packing fraction of 60%. Tentatively, lithium ortho-silicate (Li_4SiO_4) with 75% Li6 enrichment is selected for the CB.

The helium mainly flows toroidally in the layered configuration layout and flows mostly radially in the edge-on configuration. In the edge-on configuration, the coolant inside the breeding zone is subdivided into two paths; one path enters the breeding zone from the far left subunit and the other from the far right subunit. Both streams flow radially to the front, make a turn and flow radially back to the manifold. Each stream is guided through the breeding coolant manifold to the next breeder unit until it cools the last breeder unit before merging into the outlet channel through the outlet manifold. This cooling scheme and associated manifolds are illustrated in Figures 2.6 and 2.7.

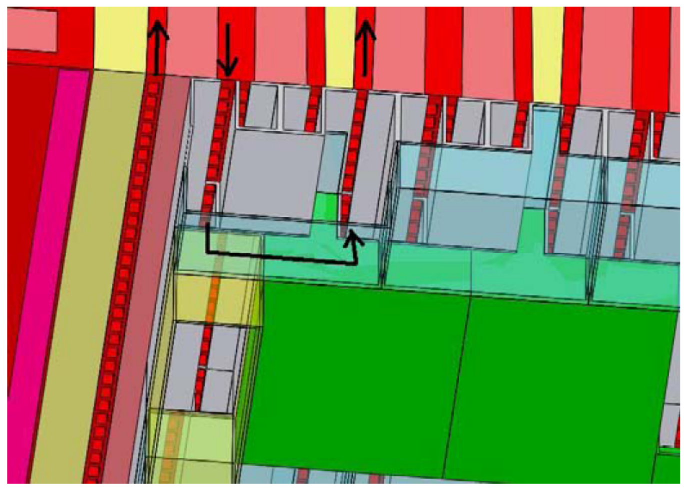

Figure 2.6. Coolant manifold schemes for Edge-on configuration

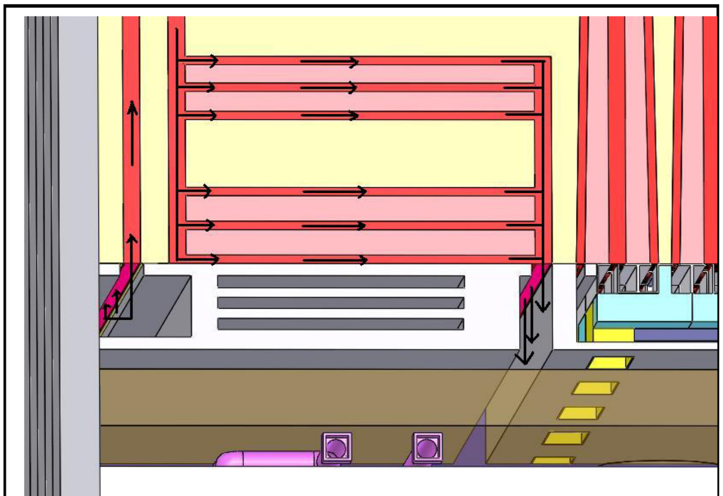

Figure 2.7. Coolant manifold schemes for Layer design configuration

These breeder unit arrangements create very different breeder temperature profiles. For example, the temperature gradient is mainly in the direction perpendicular to the coolant plates (radial) in the breeder unit of the layered configuration, while there are two temperature gradients (radial and toroidal) found in the breeder unit of the edge-on configuration. The effect of a two dimensional temperature gradient on pebble bed thermomechanical interaction and dimensional stability and their consequent impacts on thermal and tritium release performance is one of the key feasibility issues that only fusion testing can resolve. The overall breeder temperature profile in each configuration has an impact on tritium release, which can be studied by analyzing the tritium concentration inside the helium purge gas. For this purpose, each breeder configuration is equipped with its own tritium purge gas line so that the tritium collected from breeder units can be traced. All data obtained from this submodule test would help establish an optimal configuration for FS helium cooled solid breeder blanket designs for further analysis, in particular in the area of the irradiation effect on overall blanket performance.

Three (or two) purge gas streams enter the submodule through the pipes connected at the back manifold region. Each purge gas is then directed to the upper end cap purge gas manifold and is sub-distributed into different breeding units including beryllium pebble bed. The purge gas passes through the packed bed region, collects at the bottom end cap manifold and is then directed into the purge gas outlet pipe. There are three outlet purge gas pipes: one carries all the tritium generated in the layered configuration breeding zones, one carries all the tritium generated in the edge-on configuration breeding zones, and the third carries all the tritium generated in the beryllium zones. The purge gas flow scheme and associated manifold are illustrated in Figures 2.8, 2.9, and 2.10.

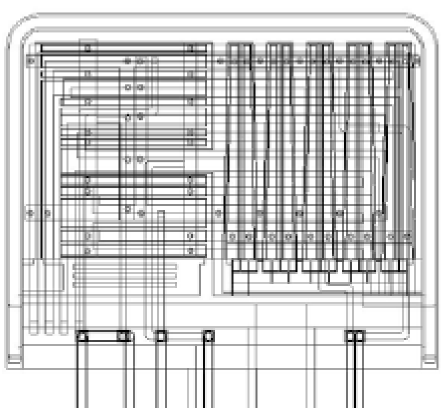

Figure 2.8. Purge gas inlet/outlet nozzle locations at different breeding elements

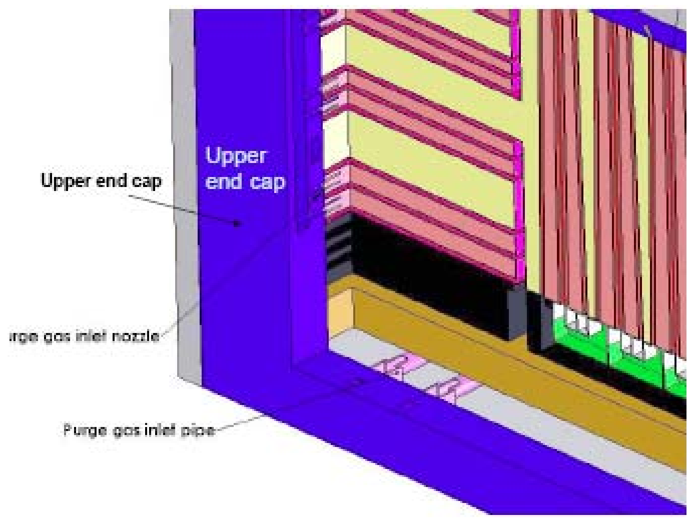

Figure 2.9. Purge gas is fed into breeding zones through purge gas nozzles located at the upper end cap (see illustration)

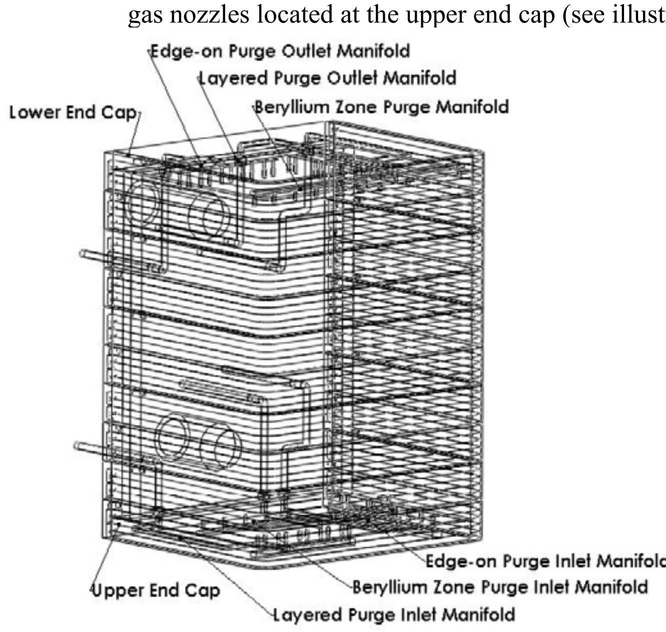

Figure 2.10. Schematic views of various purge gas manifolds located at upper and lower end caps

2.1.2 COMPONENT DESCRIPTION

Besides an 8 MPa high pressure helium coolant and a 0.1 MPa helium purge gas, a solid breeder test blanket unit cell/submodule consists of a ferritic steel structural box, coolant panels and manifold, lithium ceramic breeding pebble materials, and beryllium pebbles as multiplier. The design of the submodule involves a look-alike concept and an act-alike concept to cope with different testing needs. The resultant main difference between these two design concepts is the weight compositions between different blanket elements. A look-alike submodule is a geometric replicate of a Demo blanket design, while an act-alike submodule is designed to reproduce Demo operating parameters under ITER's neutron wall load. Two look-alike submodules are planned to be inserted into ITER at year 1 and year 5, respectively. The structure of the EM/S submodule should look like, and be built from the same FS structural material and using the same fabrication techniques as later NT and TM submodules in order to validate that structure's ability to resist disruption loads in particular, and also to measure the effect of the FS on perturbation of the local magnetic fields. It is possible that breeding materials such as beryllium will not be used in the EM/S if safety is more of a concern due to the uncertainties involved in the plasma operations. The EM/S should, however, have electrical characteristics similar to those of more integrated TBMs as well, so that the induced eddy current and its distributions are simulated. The unit cell/submodule will use about 39 and 143 kg of lithium ceramic material and 22 and 116 kg of beryllium pebbles, respectively as shown in Table 2.2. Typical design parameters including instrumentation are summarized in Table 2.3.

TABLE 2.2 Materials and their amount in the proposed unit cell/submodule TBM

Parameters	Unit Cell	Submodule (TM)
Size, m^3	0.1925 x 0.211 X 0.6	0.73 x 0.91 x 0.6
Total breeding volume (0.4 m)	0.016247	0.26572
Number of units	3	1
Breeder volume per unit, m^3	0.00633	0.0702
Beryllium volume, m^3	0.0066314	0.10399
Total ferritic steel volume, m^3	0.020089	0.147
Total breeder weight, kg (packing fraction 60%) (pebble density = 98%)	3450 x 0.98 x 0.60 x 0.00633 x3=38.5	3450 x 0.98 x 0.6 x 0.0702= 143.08
Total beryllium weight, kg (1 mm pebble 60% packing)	1850 x 0.62 x 0.0066314 x 3 =22.	1850x0.6 x 0.104 =115.4
Total ferritic steel weight, kg	154.6 x 3= 464	1132

TABLE 2.3 Typical Design Features and Operating Parameters for Different Submodules

Quarter Port Submodule	EM/S-TBM	NT-TBM	TM-TBM
ITER Master Schedule	H-H and D-D	Earlier D-T	D-T
ITER Operational Year	1-4	5	6-10
Delivery Year	-1	2	5
Ancillary Equipments Helium Loop	To Share	To Share	To Share
Ancillary Equipment Tritium Processing	To Share	To Share	To Share
Auxiliary Components in Port Cell Area	ICC, OCM, DAS	ICC, OCM, DAS, TMS	TMS, ICC, OCM, DAS
Space Required in Port Area	1x 1 x 1 m^3	1x 1 x 1 m^3	2 x (1x 1 x 1 m^3)
Total Helium Mass Flow Rate [kg/s]	TBD	0.9	0.9

Helium Pressure [MPa]	8	8	8
Helium Pressure Drop in TBM [MPa]	< 0.01	< 0.01	< 0.01
Helium inlet/outlet temperature [$_o$C]	300/350 H-H 100/300 D-D	100/300	300/500
Design Maximum temperature [$_o$C]			
FW Beryllium (2 mm)	< 346	346	545
FW Structure	< 340	340	539
Coolant Plate Structure	< 200	200	550
Beryllium Pebble Bed	300 (D-D)	300	650
Ceramic Breeder Pebble Bed	350 (D-D)	350	900
Helium Purge Gas Pressure [MPa]	NA	NA	0.1
Total Helium Purge Gas Flow Rate [g/s]	NA	NA	0.3g/s [6 Nm$_3$/s]
Purge inlet/outlet temperature [$_o$C]	NA	NA	TBD/450
Diagnostics	Field coils, Rogoski coils, pressure and displacement transducer	Thermocouples, neutron detectors,	Thermocouples, displacement sensors,
Special feature	Instrumented with activation foils capsules		
ICC: Inlet coolant conditionner; OCM: Outlet coolant mixer; DAS: Data acquisition system; TMS : Tritium measurement system			

2.1.3 Procurement Packaging

TBD

2.2 HELIUM COOLANT CONDITIONING SYSTEM
2.2.1 SYSTEM DESCRIPTION

The helium coolant conditioning system includes valves, a heater, and a mixer. The system is housed in the piping integration cask located behind the bioshield plug. The purpose of this coolant conditioning system is to divide the main coolant into a number of cooling streams and to regulate the temperature according to the flow condition required for the subunits. In addition, a by-pass pipe has been proposed by injecting an excess amount of flow to cope with the uncertainties in the surface thermal loading condition in ITER. This excess amount of flow bypasses the breeder zones and is removed after the first wall cooling. The temperature of the by-pass line can be about 150K lower than that of the outlet. One of the ideas is to mix this by-pass helium flow with the normal outlet flow in the mixer located in the port cell area rather than to run two pipes into TCWS building. However, this idea may have caused flow instability with fatigue loadings to the pipes and pipe junctions. Further investigation on this and other schemes to mix by-pass low temperature stream with high outlet temperature stream is underway. Figure 2.11 illustrates the proposed scheme.

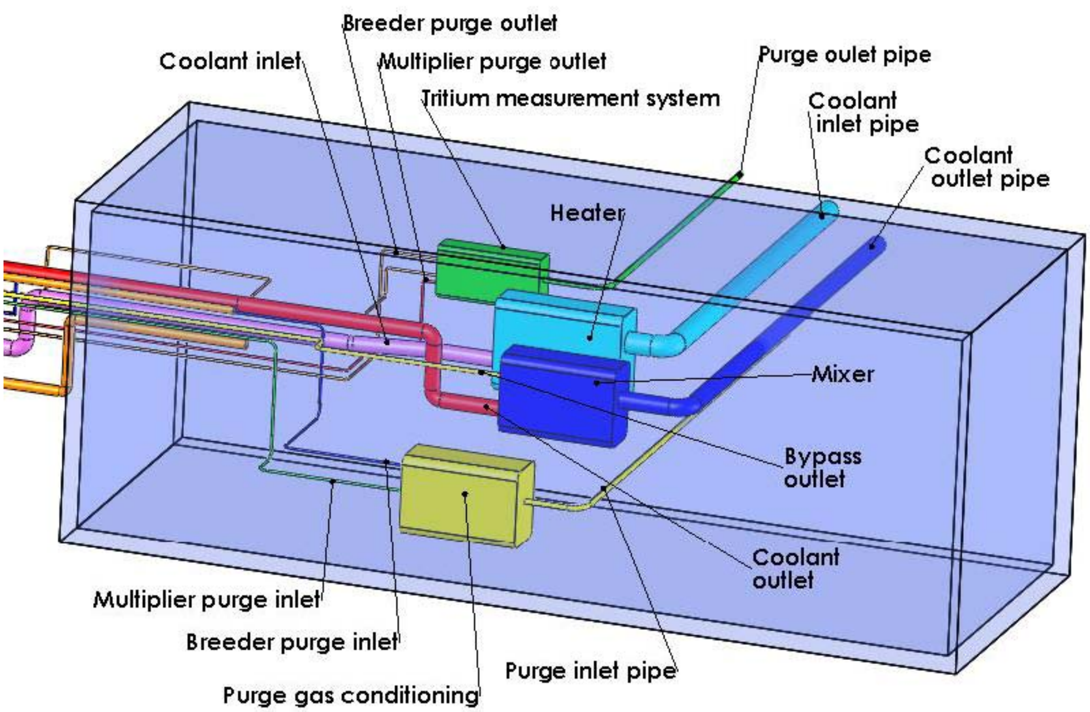

Figure 2.11. Schematic view of helium coolant conditioning

2.2.2 COMPONENT DESCRIPTION (TBD)
2.2.3 PROCUREMENT PACKAGING (TBD)

2.3 TRITIUM MEASUREMENT SYSTEM
2.3.1 SYSTEM DESCRIPTION

The tritium measurement system is installed in the port cell area to perform measurement of tritium concentration and compositions in the purge gas stream before the purge gas proceeds to the tritium extraction system. The US plans to share with EU and/or JA on the tritium extraction system, thus only tritium measurement system is described here. The tritium extraction is achieved with the help of a helium purge gas containing up to 0.1 % vol. H_2; the addition of hydrogen is needed to facilitate the tritium release by isotopic exchange. (The hydrogen is added to the clean helium purge line through a make-up unit to provide a He : H2 swamping ratio of 1000.) Removal of tritium and excess hydrogen from the helium carrier gas is performed in the extraction systems installed in the proposed glove box (4m x 1.2m x 5.5m) in the Tritium Plant. The proposed tritium measurement scheme is illustrated in Figure 2.12. It consists of a dryer, a hygrometer, ionization chambers, residual gas analyze and an associated Turbo and backing pump. The measurement system measures total tritium concentration as well as tritium concentration of HT and HTO forms. The concentrations of HT and HTO will be measured at the first ionization chamber; the water is then removed by the dryer and the concentration of HT will be measured again at the second ionization chamber.

2.3.2 COMPONENT DESCRIPTION

TBD

2.3.3. PROCUREMENT PACKAGING

TBD

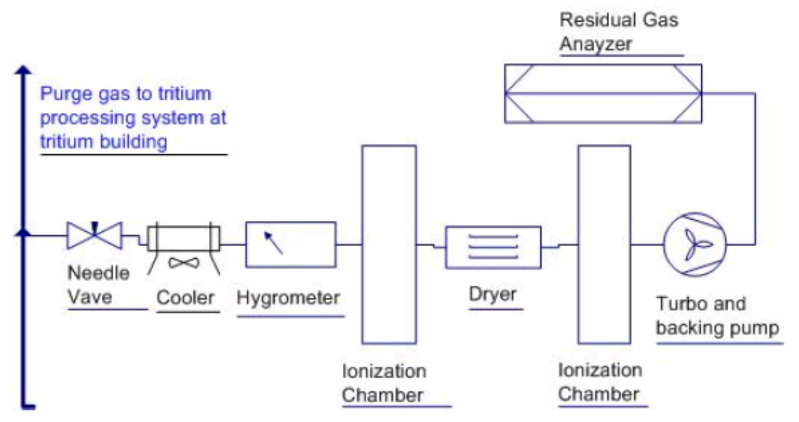

Figure 2.12. Tritium measurement system diagram (the RGA station is optional if the space availability at the port area becomes a concern)

2.4 NEUTRONICS MEASUREMENT SYSTEM

Another diagnostic system to be employed in evaluating the accuracy associated with the prediction of the nuclear environment inside the TBM is the neutronics measurement system. The objective is to compare the calculated neutronics parameters (e.g. neutron flux, neutron and gamma spectra, tritium production rate, TPR, heating rate, etc.) to the measured valued. These tests could be performed during the D-D phase and possibly the very beginning of the Low Duty Cycle D-T phases. These dedicated tests aim at examining the present state-of-the-art neutron cross-section data, various methodologies implemented in transport codes, and system geometrical modeling as to the accuracy in predicting key neutronics parameters such as neutron/gamma spectra, tritium production rate (TPR), and nuclear heating rates. The first campaign could be devoted to quantifying the nuclear field through neutron and gamma spectra measurements. Multi-foil activation pellets (MFA) can be used for that purpose. The second campaign is for TPR measurements (can be performed with some techniques such as using lithium glass scintillators for detecting TPR from Li-6 and Li-7, lithium foils/pellets, etc.). Generally, all neutronics parameters, except activation and damage parameters can be measured at one of two fluence levels (see Ref. 2.1-2), namely: low fluence level (~1 W.s/m^2) and very-low fluence level (~1 mW.s/ m^2) . These levels are the minimum fluence requirements, but higher levels are generally desirable for improving measuring statistics. The low fluence level could be realized, for example, with a wall load of 1 MW m^2 and 1-s pulse, or alternatively, a wall load of 0.0025 MW/ m^2 and 400 s pulse, as is the case in ITER. Clearly the NWL of ~0.78 MW/ m^2 at the TBM is much larger than the 0.0025 MW/ m^2 needed to achieve the required low fluence level for these types of neutronics tests. Therefore, most of these measurements can be made *within the duration of a single pulse*. Performing the measurements of spectra and TBR measurements during the D-D pulses will give information on the prediction uncertainties under a typical nuclear environment of incident neutrons whose average energy is ~2.5 MeV. It is therefore desirable to repeat these measurements during the low duty D-T phase to derive more representative estimates to the prediction uncertainties under a typical fusion environment. Furthermore, we must have accessibility to reach several locations in the TBM to perform the aforementioned tests. This can be carried out by inserting two or three radial measuring tubes in which we place the foils/pellets (in a train) and be able to retrieve them after irradiation from behind (mechanically, or by a rabbit system) without removing the TBM and/or interrupting the operation of ITER. Details of this system will have to be designed.

References:

2.1-1 L.V. Boccaccini (ed.): "European Helium Cooled Pebble Bed (HCPB) Test Blanket. ITER Design Description Document. Status 1.12.1998", FZKA 6127, March 1999.

2.1-2 M.Z. Youssef and M.E. Sawan, "On the Strategy and Requirements for Neutronics Testing in ITER, To be Published in *Fusion Science & Engineering*, April 2005.

3. PERFORMANCE ANALYSIS
3.1 NUCLEAR ANALYSIS

In addition to modeling qualitative interpretation of system response, ITER test results can be used to benchmark/calibrate any numerical codes or analytical methods, or to make quantitative predictions of the prototype response. The neutronic submodule is designed to comply with this purpose as shown in Fig. 3.1, in which its design criterion is determined by the geometrical size requirements to maintain a high spatial resolution for any specific measurement and allow complexity to maximize code validation. The submodule represents two "look-alike" helium-cooled solid breeder test blanket configurations proposed and placed inside a TBM side-by-side to the Japanese ceramic breeder TBM (Fig. 3.2) has been analyzed and compared. One of the configurations is based on layered ARIES CS design[3.1-1] in which the breeder pebble beds are parallel to the FW of the TBM whereas an edge-on configuration as in EU HCPB [3.1-2] is considered for the other sub-module where the breeder beds are placed perpendicular to the FW. The comparison is made for integrated as well as local values (profiles) of key parameters such as tritium production rate (TPR) and heat deposition rate under ITER operation conditions. The total volumetric heat load that must be removed from the TBM is estimated for both sub-modules. Locations where steep gradients are found for these key parameters are identified in order to make recommendations for the most appropriate locations where neutronics measurements and tests can be performed with the least perturbation from the surroundings. The analysis also provided tritium production rate and heat generation rate for subsequent thermal-hydraulic, thermomechanics, and tritium management analyses.

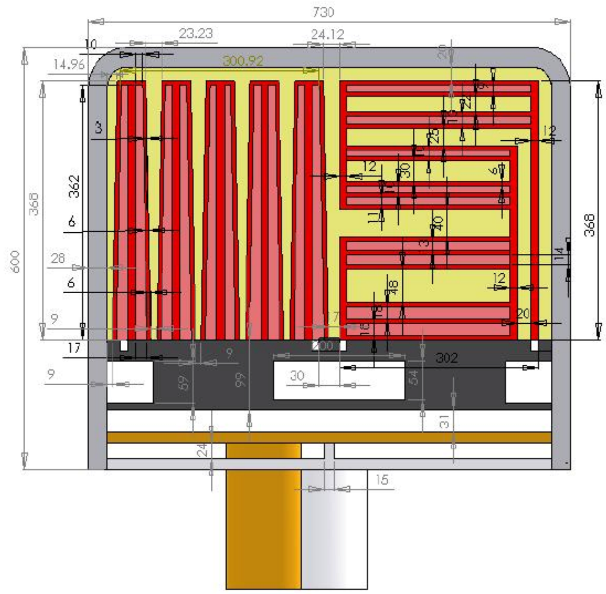

Figure 3.1. The proposed US neutronics submodule

3.1.1 CALCULATIONAL PROCEDURES AND MODELING

An R-θ model was used to describe the geometrical arrangement of the ITER basic shielding blanket, the vacuum vessel (VV), the magnet, the US and Japan TBM[3.1-3], and the detail of the test port at the mid plane of ITER machine. An isometric view of the material assignment of this model is shown in Fig. 3.2, while the Japan TBM details in the analysis is shown in Figure 3.3. The thickness of the VV at the outboard (OB) is 75 cm (33.7 cm at the

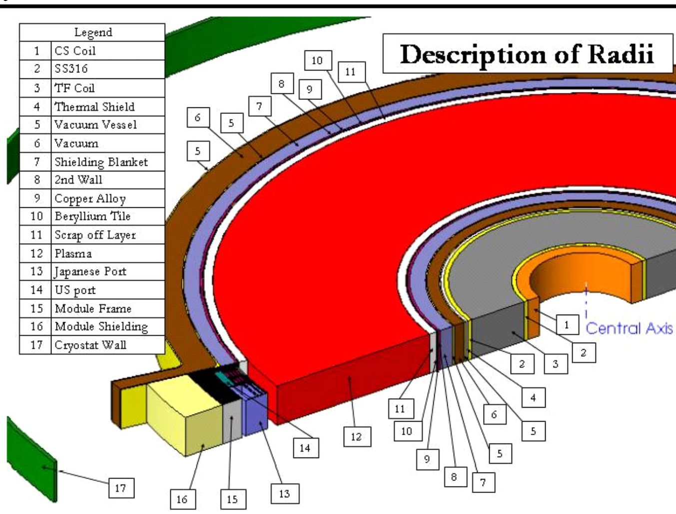

Figure 3.2. Isometric view showing material assignments in the calculation model

inboard, IB), including 3 cmthick SS316ln walls. The VV is cooled with water (40% H2O, 60% SS316ln). The magnet in the IB is 140.8 cm thick surrounded by 10 cm-thick SS316lw outer layers. This magnet is placed immediately after the central solenoid (CS) which is 30 cmthick; its inner radius is 125 cm from the center of the torous. There is a 3 cm-thick gap (void) between the IB magnet and the VV. The thickness of the Be tile, Cu-alloy zone (79% SS316ln, 21% H2O), 2nd FW (81% SS316ln, 19% H2O) and ITER shielding blanket (72% SS316ln, 28% H2O) is 1, 2.2, 4.9, 36.9 cm (35.5, IB), respectively. In the calculation model, the front edge of the Be tile in the IB is placed at a radial distance of 356.5 cm from the center of the torus (850 cm for the OB). Figure 3.4 shows the details of the model at the port where the two TBMs are placed and mounted on a 10 cm-thick steel frame (60% SS316ln, 40% H2O). There is a 10 cm-thick gap behind the manifolds of the two TBMs. A port shield (72% SS316ln, 28% H2O) with a 90 cm thickness, is placed behind the back of the frame. The two TBMs are placed at a distance of 855 cm from the center of the torus. The recess of 5 cm depth is prescribed in ITER design for the test ports.

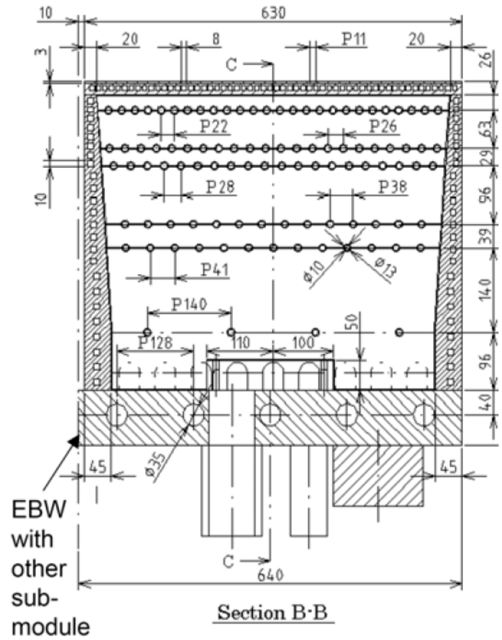

Figure 3.3. A Japanese test blanket

The discrete ordinates transport code DORT [3.1-4] was used in the calculation of the R-θ model with P5S8 approximation. The 46 neutron-21 gamma group library used is based on FENDL-2 data base[3.1-5]. In the model, the entire machine in the θ direction was considered (θ varies between 0 and 360 degrees). The numbers of meshes considered in the model are 336 and 162 in the toroidal and radial direction, respectively. In the edge-on configuration, the thickness of the breeder unit (canister) at its front tip (4.4 cm) is assumed to be the same (constant) as we move towards its base in the radial direction. The front breeder unit tip in this configuration is placed at a radial distance of 859.8 cm from the center of the torus and the subtended θ angel is 0.00515 radian. Thus, in the R-θ model used, the breeder unit thickness at its base (located at a radial distance of 896.6 cm from the center of the torus) would be ~4.6 cm, a 4.5% increase from its nominal value (4.4 cm) due to the model adopted in the present analysis.

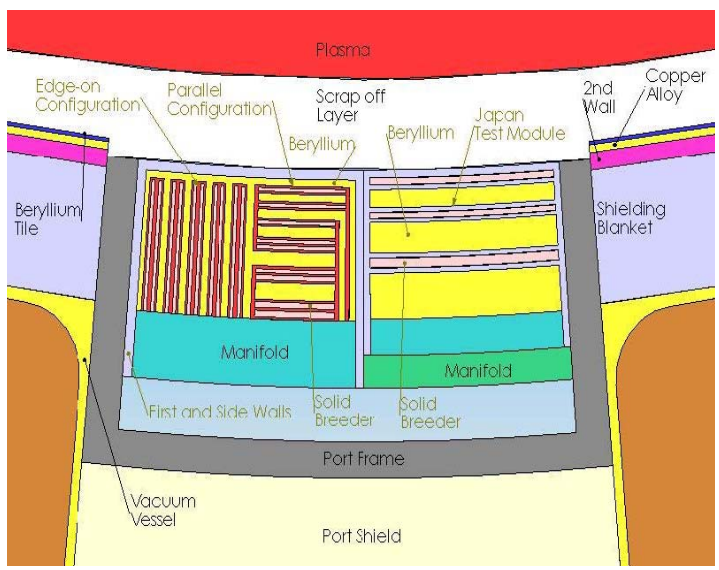

Figure 3.4. Top view showing the two submodules placed in the port with the surrounding ITER basic shielding blanket

3.1.2. PROFILES OF NUCLEAR HEAT DEPOSITION RATES

The nuclear heating rates in the FW of the US TBM are shown in Fig. 3.5 for an average neutron wall load (NWL) of 0.78 MW/m2 at the TBM. Except for the irregularities in these profile in the left edge-on configuration in which more heterogeneities are found in the toroidal direction, one can observe that the profiles are nearly flat over a toroidal distance of ~15-20 cm (right configuration) and ~10-15 cm (left configuration). This flatness maintains its length in the toroidal direction at the three locations shown in the figure. The values in the right configuration at these flat regions are nearly the same as those in the left configuration. The heating rate measurements could be performed over these flat regions with no concern for error due to uncertainty in location definition. The profiles are steep near the left side wall of the TBM over a toroidal length of ~16 cm due to the presence of the 10 cm-thick frame which is cooled with water and which produces significant amounts of gamma generation. The heating rates there are a factor of 1.12, 1.17, and 1.25 larger than those at the inner regions at distances of 0, 8, and 16 mm from the front edge of the FW, respectively.

The nuclear heating rates in the beryllium layer located just behind the FW in the toroidal direction are shown in Fig. 3.6. The profiles are flat over the entire layer. As shown, the heating rates in structure content of the adjacent side

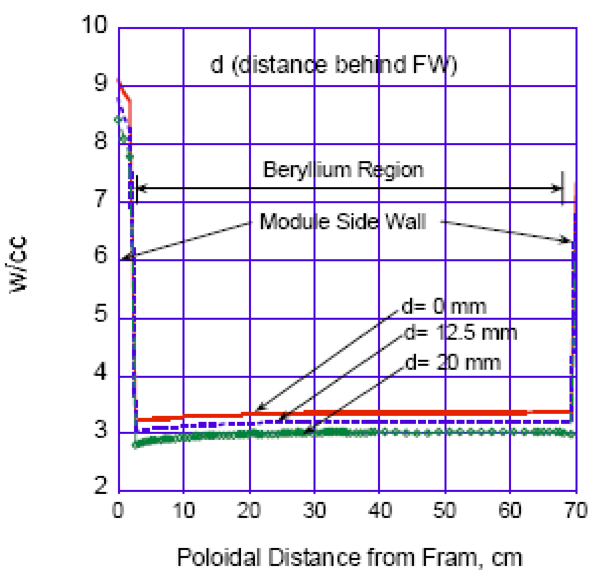

Figure 3.5. Nuclear heating in the beryllium region behind FW in the US HCPB Test Blanket Module

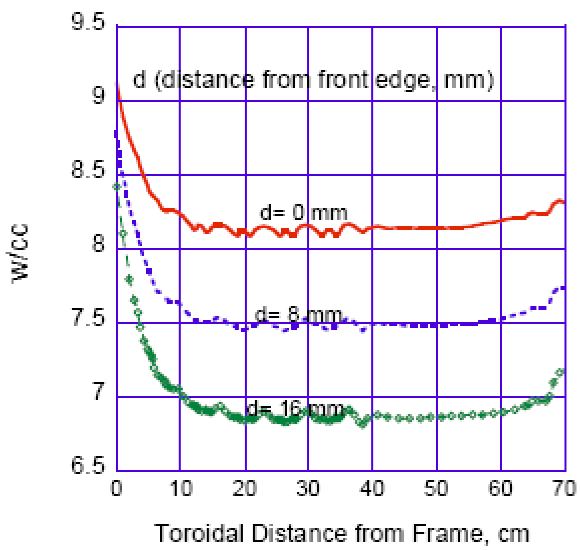

Figure 3.6. Radial heating rate (W/cc) in the First Wall of the US HCPB Test Blanket Module

wall of the TBM are much higher (with a factor of ~3) than those in the beryllium layer. Notice that the beryllium layer at this location extends over the entire width of the TBM that covers both the left and right configur ations. The radial heating rate profiles inside the left traverse cooling panel (TCP) of the five breeder units of the edge-on configuration are shown in Fig. 3.7. The heating rates in the left TCP of the far left breeder unit (unit#1), are the largest among the other TCPs at the front locations by the virtue of being the closet to the water-cooled TBM frame. However, at the back locations, the heating rates in the TCP of the far right unit (unit#5) seems to be slightly larger than those found in the other TCPs. The heating rates in the other inner TCPs are comparable. Note that the radial gradients of the heating rate are not steep inside these TCPs. They vary by a factor of ~10 over a radial depth of 36.8 cm (see Fig. 3.7).

The gradient of radial heating rate in the ceramic breeder is more pronounced than those found inside the structure in the TCPs. This is shown in Fig. 3.8 where the radial heating rates are given inside the left bed of the ceramic breeder in each breeder unit. The radial heating rates inside the left bed as well as the right bed of the innermost unit (unit#5) are shown for comparison. The steepness (as well as [6] absolute values) of the curve for the left bed of the far left unit (unit#1) is the largest among all these breeder beds since it is placed closer to the beryllium left beds and to the side wall of the TBM where more reflection occurs for slow neutrons, which in turn contribute to the heat generation via Li-6(n,α)t reactions. If heating rate $_2$ measurements are intended inside the ceramic breeder, one should avoid making these measurements near the front edge of these beds closer to the plasma side. Rather, it is recommended to undertake these measurements at the inner most beds at rear locations where the steepness of these curves is the least. One can also notice from Fig. 3.8 that the local values of these profiles are the largest for unit#1 and unit#5 whereas the values inside the breeder beds of unit#2, unit#3, and unit#4 are lower tha n those found in unit#1 and unit#5, and are increasing, in that order. Additionally, the heating rates inside the ceramic breeders are, in general, larger than those attributed to the heating inside the structure content of the TCPs (by ~ a factor that can be as large as 2). The nuclear heating rates across the two sub-modules of the US TBM in the toroidal direction and at a depth 11.6 cm behind the FW are shown in Fig. 3.9. The features discussed above for the heating inside the ceramic breeder beds in comparison to the heating in the TCPs of the edge-on configuration are still shown in this figure. They are larger by a factor of 2 for the most-left breeder bed (unit#1) and by a factor of 1.7 for the rightmost bed (unit#5). These factors are 1.25, 1.31, and 1.43 for unit#2, Unit#3, and unit#4, respectively. The heating rates in the beryllium beds of the edge-on configuration are the least at this radial depth (~1.2-1.4 W/cc) and they are nearly flat. On the other hand, the toroidal profile inside the breeder of the parallel configuration sub-model is nearly flat over a toroidal width of ~ 10 cm but the profile starts to peak towards the right side wall of the TBM due to the presence of the beryllium layers, the right side TCPs and the side wall of the TBM. The heating rate at this parallel layer (3rd layer in Fig. 3.4), is, on average, higher than the values found inside the breeder beds of the edge-on configuration (they vary between 6-8 W/cc).

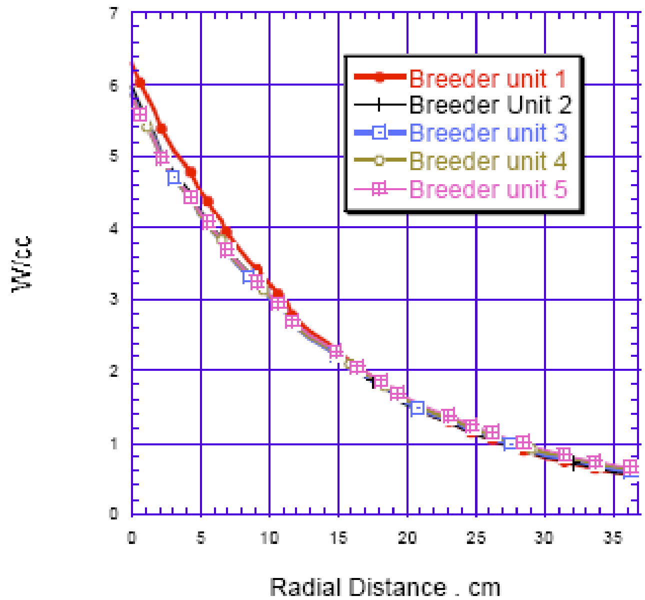

Figure 3.7. Radial heating rate in left traverse cooling panel of each breeder unite in the edge-on configuration

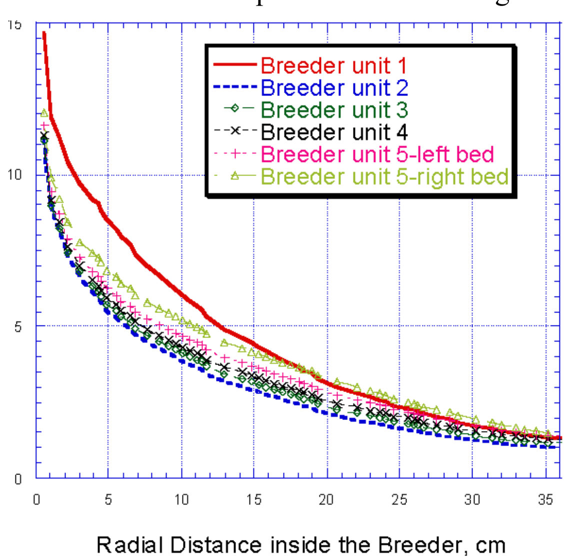

Figure 3.8. Radial heating rates inside the breeder in each unit of the edge-on configuration

3.1.3. PROFILES OF TRITIUM PRODUCTION RATE

The TPR profiles (Tritons/cm3.sec) in the radial direction inside the left breeder beds of the edge-on configuration are shown in Fig. 3.10. The similarity in the features of the heating profiles discussed for Fig. 3.8 and those shown for the TPR in Fig. 3.10 is clear. This is expected since the main contributor to the heating in the ceramic breeder is attributed to the Li-6 (n,α)t reactions which is also the main contributor to the TPR in the ceramic beds (75% Li-6 enrichment). The steepness of the TPR profile in the most-left bed (unit#1) is due to the reasons discussed earlier. The large contribution to the total TPR from the inner-most bed layers in the edge-on configuration (unit#5) is also apparent in the figure.

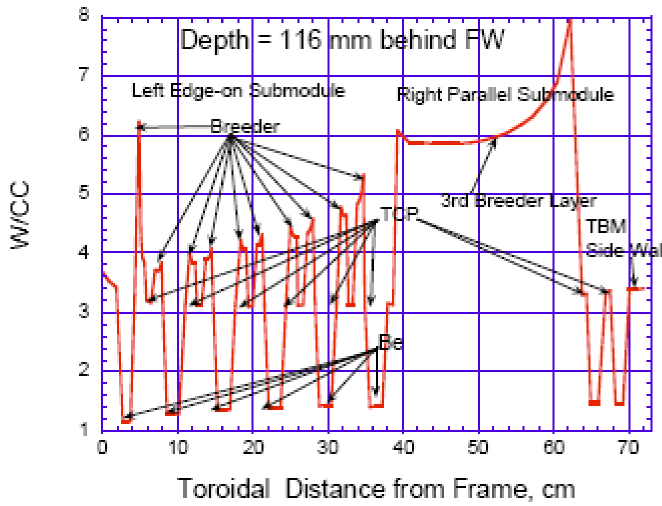

Figure 3.9. Nuclear Heating Across the U.S Two Test Blanket Sub modules in the Toroidal Direction

We discussed earlier that one of the objectives of the present work is to identify those locations where TPR measurements can be performed inside the breeder with the least perturbation from the surround ing heterogeneities. Measured neutronics data require high spatial resolution and consequently this requirement necessitates that the measured quantity be as flat as possible in the innermost locations. We examine in Fig. 3.11 how this requirement can be met. The figure shows the toroidal variation in the TPR inside each parallel layer in the layered configuration and across the breeder beds in the edge-on configuration at various depth "d" in the radial direction behind the FW. As shown, each unit of the edge-on configuration the TPR profiles are nearly flat over a toroidal width of ~15-20 cm in the parallel layered configuration. Furthermore, in previous work [20] we have shown that in this configuration (in which the breeder layers have thicknesses that vary between 0.9-1.8 cm), the profiles of the TPR are much steeper in the radial direction than in the toroidal direction. However, it was shown that the radial distance over which the TPR changes by 5% from its lowest value is limited to ~1 cm. Therefore, to achieve high resolution, the TPR measurements should be performed within this 1 cm range and inside those layers that are far from the plasma side (last

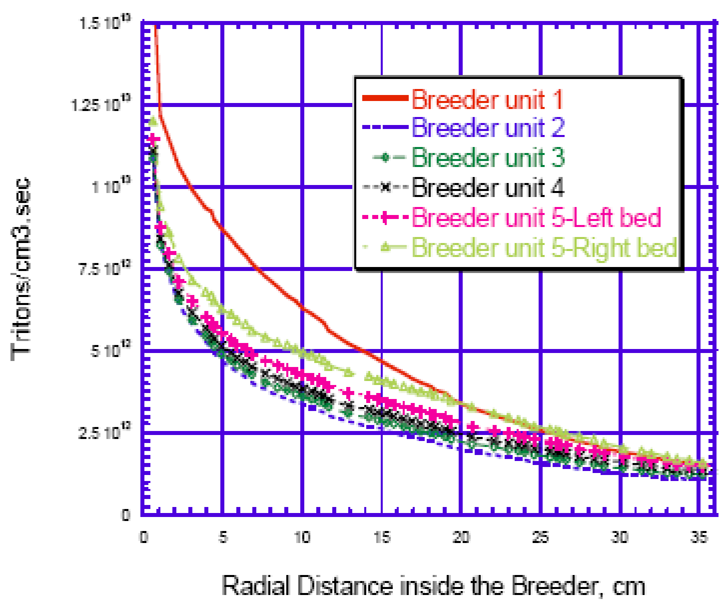

Figure 3.10. Tritium production rate inside the breeder in each unit of the edge-on configuration

layers).

The characteristics of the TPR profiles in the edge-on configuration are different. The thickness of the breeder beds in this configuration is ~1 cm. In the calculations, four spatial meshes were considered inside each bed in the toroidal direction. As shown in Fig. 3.11, the profiles of the TPR are extremely steep in these beds at the fr ont locations near the FW; this is apparent especially inside the left bed of the most-left breeder unit (unit#1) where the maximum-to-lower value is ~1.5 over a 1 cm toroidal width. This steepness is shown to vanish gradually as we move toward the base of each breeder unit and away from the plasma side. It is therefore recommended to perform TPR measurements at those locations which exhibit the least peaking in TPR values. For example, the variation in the TPR is almost null in the breeder unit #2, unit#3, and unit#4 at radial depth d= 26.6, 34, and 36.6 cm, as can be seen from Fig. 3.11. There is, however, a concern that the local TPR values at these locations are a factor of 6 lower than those found at front locations which may require longer time to accumulate enough tritons that can be measured with present measuring techniques[3.1-6].

3.1.4. INTEGRATED NUCLEAR HEATING RATE

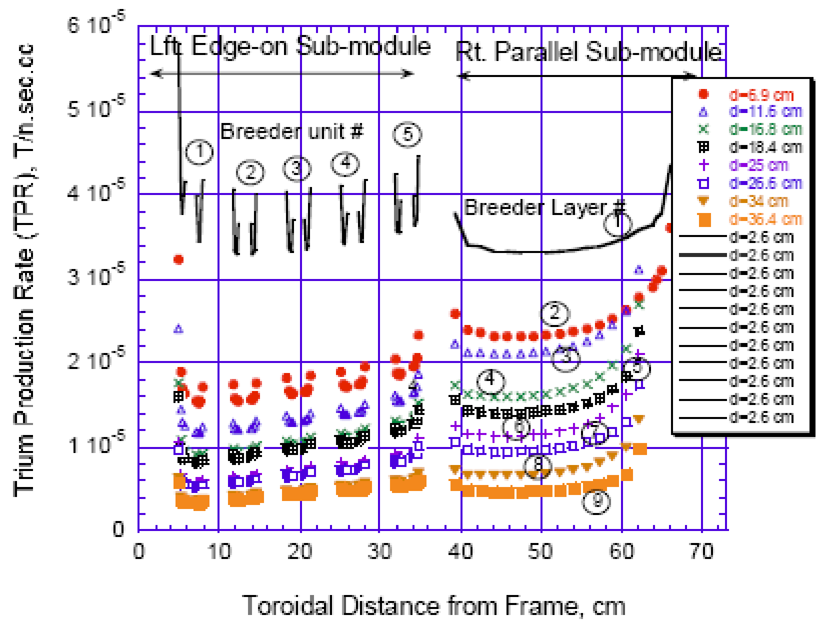

Figure 3.11. Toroidal profile of tritium production rate in the two configurations at various radial distance "d" behind the first wall

The TBM of the US and Japan are placed in the lower half of the test port dedicated to testing ceramic breeders. The height of that half port (excluding the separating frame in the poloidal direction) is 91 cm. We introduce in Table 3.1 the heat load (in kW) in several material zones in both configurations, along with the corresponding volumes (in cm^2). Excluding the heat deposition in the first and side walls of the TBM as configurations at various radial distance "d" behind the first wall well as in the TBM manifolds (shared by both configurations), the heat load in all the internals of the edge-on configuration and in the parallel configuration are 225.35 kW and 224.42 kW, respectively.

TABLE 3.1: Integrated nuclear heat deposition rate, kW*

Left Sub module-Edge-On Configuration			
Material Zone	*Nuclear Heating, kW*	*Volume, cm^3*	Heating Rate per cm^3
Unit 1- Structure	10.44	8365.90	1.25E-03
Unit 2- Structure	10.22	8363.99	1.22E-03
Unit 3- Structure	10.26	8365.90	1.23E-03
Unit 4- Structure	10.35	8363.99	1.24E-03
Unit 5- Structure	10.47	8365.90	1.25E-03
Sub-total- Structure	***51.75***	***41825.69***	***1.24E-03***
Unit 1-Breeder	20.78	6769.67	3.07E-03
Unit 2-Breeder	19.97	6769.67	2.95E-03
Unit 3-Breeder	21.32	6767.85	3.15E-03
Unit 4-Breeder	22.58	6771.58	3.33E-03
Unit 5-Breeder	24.87	6767.85	3.67E-03
Breeder: sub-total	***109.52***	***33846.63***	***3.24E-03***
Front Be Layer	17.82	5720.44	3.11E-03
All other Be zones	46.27	46117.89	1.00E-03
Beryllium: sub-total	***64.08***	***51838.33***	***1.24E-03***
Sub-module-Total Heat Deposition Rate, kW	*225.35*		1.77E-03
Sub-module-Total Volume, cm^3		*127510.66*	
Right Sub module-Parallel Configuration			
1st Horizontal Cooling Panel-Structure	4.76	1527.07	3.12E-03
2nd Horizontal Cooling Panel-Structure	4.22	1529.71	2.76E-03
All other Horizontal Cooling Panels-Structure	21.68	19019.00	1.14E-03
Vertical Cooling Panels-Structure	13.35	10928.19	1.22E-03
Structure: sub-total	***44.01***	***33003.97***	***1.33E-03***
1st Breeding Layer-Breeder	23.45	2547.27	9.21E-03
All other Breeding Layers-Breeder	83.26	23168.60	3.59E-03
Breeder: subtotal	***106.72***	***25715.87***	***4.15E-03***
Front Be Layer	17.46	5520.79	3.16E-03
All Other Be Layers	56.24	52403.26	1.07E-03
Beryllium: sub-total	***73.69***	***57924.05***	***1.27E-03***
Sub-module Total Heating Rate, kW	*224.42*		***1.92E-03***
Sub-module-Total Volume, cm^3		*116643.89*	
First and Side Walls	108.03	48849.71	2.21E-03
Manifold	26.93	119555.80	2.25E-04
Total Heating Rate in the US TBMs, kW	*584.73*		***1.42E-03***
Total Volume of the US TBMs, cm3		*412560.06*	
Port Shield	13.51	1471925.00	9.18E-06
Port Frame	263.96	297733.80	8.87E-04

* For an average wall load of 0.78 MW/m^2 and 91cm TBM height in the poloidal direction

The central beryllium bed separating the two configurations is considered part of the edge-on configuration. Table 3.2 gives the contribution (in %) to these heating loads (and volumes) from each material zone. As shown, 48.6% of the total internal heating is attributed to the breeder in the edge-on configuration (47.55% in the parallel configuration). Total heating in beryllium is appreciable in both sub-modules and is ~28.44% and 32.84% in the edge-on and parallel configuration, respectively. As for the structure contents, the heat deposited in the cooling panels contributes ~22.96% and 19.61%, respectively. One notices that heating in the structure is the highest from unit#1 and unit#5 among other breeder units in the edge-on sub-module, as was discussed earlier with relation to Figs. 7 and 9. Also the heat deposited in the breeder of unit#5 is the largest (~11%) as compared to the other units in the same sub-module. This also was apparent in Figs. 8 and 9. One can see that the highest heat deposition rate per unit volume takes place inside the breeder of the 1st layer in the parallel configuration (0.009 kW/cm^3) whereas the corresponding value in the beryllium multiplier is as low as ~0.0012 kW/cm^3.

TABLE 3.2: Percent heat deposition rate in each material zone in the US TBMs

Left Sub module-Edge-On Configuration		
Material Zone	*% of Total Heating*	*% of Total Volume*
Unit 1- Structure	4.63	6.56
Unit 2- Structure	4.53	6.56
Unit 3- Structure	4.55	6.56
Unit 4- Structure	4.59	6.56
Unit 5- Structure	4.65	6.56
Sub-total- Structure	*22.96*	*32.80*
Unit 1-Breeder	9.22	5.31
Unit 2-Breeder	8.86	5.31
Unit 3-Breeder	9.46	5.31
Unit 4-Breeder	10.02	5.31
Unit 5-Breeder	11.04	5.31
Breeder: sub-total	*48.60*	*26.54*
Front Be Layer	7.91	4.49
All other Be zones	20.53	36.17
Beryllium: sub-total	*28.44*	*40.65*
Sub-module-Total Heat Deposition Rate	*100.00*	
Right Sub module-Parallel Configuration		
1st Horizontal Cooling Panel-Structure	2.12	1.31
2nd Horizontal Cooling Panel-Structure	1.88	1.31
All other Horizontal Cooling Panels-Structure	9.66	16.31
Vertical Cooling Panels-Structure	5.95	9.37
Structure: sub-total	*19.61*	*28.29*
1st Breeding Layer-Breeder	10.45	2.18
All other Breeding Layers-Breeder	37.10	19.86
Breeder: subtotal	*47.55*	*22.05*
Front Be Layer	7.78	4.73
All Other Be Layers	25.06	44.93
Beryllium: sub-total	*32.84*	*49.66*

Sub-module Total Heating Rate	*100.00*	
First and Side Walls	18.47	11.84
Manifold	4.61	28.98

When the total nuclear heat deposited in the first and side walls as well as in the manifold are accounted for, the total volumetric heat load in the US TBM (excluding surface heating) is ~585 kW. This amount of heat should be removed from the TBM with proper design of heat exchangers and the ancillary system. About ~18% to this integrated value is attributed to heating in the TBM walls that surrounds and houses the two sub-modules, while the heat deposited in the manifold is minimal (~4.6%). The integrated heating rate in the port shield and the port frame are also shown. These values are for a poloidal height of 91 cm but clearly the port shield is designed to shield the four quadrants of the port where other TBMs are placed in the upper half of the port. This is also true for the port frame, and therefore, the values shown should be modified to account for the actual port size (roughly twice as much as the values shown). The least heating rate per unit volume is found in the port shield (~9.2x10-6 kW/cm^3) as expected since it is located at the back of the TBM and it occupies a large volume (~1.5 m^3 for a 91 cm poloidal height).

3.1.5. INTEGRATED TRITIUM PRODUCTION RATE

The details of the integrated tritium production in the ceramic breeder beds and in the beryllium multiplier are shown in Tables 3.3 and 3.4, respectively. The volumes of the breeder in the edge-on configuration and in the parallel configuration are 33847 cm^3 and 25716 cm^3, respectively. The corresponding volumes of the beryllium multiplier are 51838 cm^3 and 57924 cm^3, respectively. Thus the volume ratios of the multiplier to the breeder are 1.531 and 2.252, respectively. Thus, the amount of Be in the edge-on configuration is less than in the parallel configuration (by ~10%) while the reverse is true for the ceramic breeder (larger in the edge-on configuration by ~32%). The fact that more neutron multiplication is taking place in the parallel configuration (under the present design) lends itself to a larger triton production rate per unit volume of the breeder with a value of 4.10x1012 Tritons/cm^3•sec as compared to 2.96x1012 Tritons/cm^3•sec in the edge-on configuration. This translates to an increase in the tritium generation rate per unit breeder volume by ~38% although the absolute values of tritium generation rate are comparable (1.0x1017 tritons/sec vs. 1.05x1017 tritons/sec). A similar argument can be applied to the tritium generation rate in the beryllium but the values are ~two orders of magnitude less than tritium bred in the ceramic breeder, as can be seen from Table 4. If a fluence of 0.3 MWa/m^2 is to be reached at the end of ITER operation that has an average wall load of 0.57 MW/m^2, the total amount of tritium generated in the US TBM is ~16.5 g. Only about 0.17 g of tritium will be generated in the beryllium multiplier.

TABLE 3.3: Tritium production rate in the ceramic breeder beds*

Left Sub-module-Edge-On Configuration		
	Volume, cm3	TPR, Tritons/sec
Breeder unit 1	6769.672	1.931E+16
Breeder unit 2	6769.672	1.773E+16
Breeder unit 3	6767.852	1.918E+16
Breeder unit 4	6771.583	2.062E+16
Breeder unit 5	6767.852	2.346E+16
Sub-module-Total	*33846.631*	*1.003E+17*
Sub-module Total TPR per unit volume, tritons/cm3.sec		2.963E+12

Right Sub-module-Parallel Configuration		
1st Parallel Breeder Layer	2547.272	2.142E+16
All other Parallel Breeder Layers	23168.6	8.394E+16
Sub-module-Total	*25715.872*	*1.054E+17*
Sub-module Total TPR per unit volume, tritons/cm3.sec		4.097E+12
Total TPR in the US TBMs, tritons/sec		*2.057E+17*
Total tritium production at fluence 0.3 Mwa/m2, tritons		3.414E+24
Total tritium production at fluence 0.3 Mwa/m2, g		*1.6549E+01*

For an average wall load of 0.78 MW/m^2 and 91cm TBM height in the poloidal direction.

TABLE 3.4: Tritium production rate in the beryllium beds*

Left Sub-module-Edge-On Configuration		
	Volume, cm3	TPR, Tritons/sec
Front Beryllium layer	5720.442	3.513E+14
All other beryllium zones	46117.89	6.487E+14
Sub-module-Total	*51838.332*	*1.000E+15*
Sub-module Total TPR per unit volume, tritons/cm3.sec		1.929E+10
Right Sub-module-Parallel Configuration		
Front Beryllium layer	5520.788	3.429E+14
All other beryllium zones	52403.26	7.399E+14
Sub-module-Total	*57924.048*	*1.083E+15*
Sub-module Total TPR per unit volume, tritons/cm3.sec		1.869E+10
Total TPR in Beryllium the US TBMs, tritons/sec		*2.083E+15*
Total tritium production at fluence 0.3 Mwa/m2, tritons		3.457E+22
Total tritium production at fluence 0.3 Mwa/m2, g		*1.675E-01*

For an average wall load of 0.78 MW/m^2 and 91cm TBM height in the poloidal direction.

3.1.6. SUMMARY AND CONCLUSIONS

Unlike the case of using engineering scaling to reproduce DEMO-relevant parameters in an "act-alike" test module, dedicated neutronics tests require a "look-alike" test module for a given blanket concept. For that purpose, the US is proposing two "look-alike" sub-modules to be tested in the same test blanket module (TBM) that will occupy a quarter of a port in ITER and placed next to the Japanese TBM. The US modular TBM has a totoidal width of 73 cm, radial depth of 60 cm (including the back manifold) and a poloidal height of 91 cm. Helium-cooled pebble beds (HCPB) of ceramic breeder (Li4SiO4, 75% Li-6) and beryllium multiplier are arranged in two configurations housed inside the TBM. A layered configuration is considered in the first sub-module where the ceramic and Be beds are placed parallel to the FW with thicknesses varying in the radial direction. An edge-on configuration is considered in the second sub-module in which the breeder and multiplier beds are perpendicular to the FW facing the plasma. In the present work, we carried out comprehensive two-dimensional calculations in an R-θ model that accounts for the presence of the ITER shielding blanket and the surrounding frame of the port. The objectives are: (1) to examine the profiles of heating and tritium production rates in the two sub-modules, both in the radial and toroidal direction, in order to identify locations where neutronics measurements can be best performed with least perturbation from the surroundings (i.e. profiles should be flat as possible over a reasonable spatial range), (2) to provide both local and integrated values for heating generation rates (required as input for subsequent thermo-mechanics analysis and design of hear removal system) and total tritium production rate in the breeder and beryllium (for safety assessment), and (3) to compare the tritium production capabilities of two variants for HCPB blanket concepts, mainly the parallel and the edge-on configurations.

In the parallel configuration, both heating and tritium production rates are shown to be flat inside the breeder layers over a range that can be as wide as ~20 cm in the toroidal direction, especially at the layers located in the back of the sub-module. Previous work has shown that the profiles can be measured over a range of ~ 1 cm with a variation not exceeding 5%, provided the radial width is ~2 cm or more. In the edge-on configuration, the situation is not as favorable as in the parallel sub-module by the virtue of the presence of two types of gradients; one in the radial direction that shows large steepness near the FW, and the other across the breeder beds in the toroidal direction where extremely steep profiles are found across the 1-cm thick beds. This steepness decreases gradually towards the back locations. Accordingly, it is recommended to perform neutronics measurements in the inner most beds at the back locations of the sub-module although the absolute values are a factor of 6 less than those at front locations, which may require longer exposure time to the neutron source.

The total volumetric heat load generated in the edge-on and the parallel configurations are 225, and 224 kW, respectively. If we account for the heat generated in the first and side walls and in the manifold, the heat removal system should be designed to remove a total of ~585 kW (excluding surface heating). Most of this heat is generated in the breeder (~37%) via Li-6(n,α)t reactions whereas the first and side walls generate ~18% of this heat load

According to the proposed design of the two HCPB sub-modules, the amount of Be in the edge-on configuration is less than in the parallel configuration (by ~10%) while the amount of the breeder is larger in the edge-on configuration by ~32%. This led to a factor of ~1.4 larger tritium production rate per unit volume of the breeder in the parallel configuration in comparison to the edge-on one. However, the absolute tritium production rate (tritons/sec) is comparable in the two configurations. Whether the edge-on configuration is more favorable neutronically than the parallel configuration (or vice versa) is still not clear because, as the results show, the total heat and tritium generated in the two configurations are comparable. However, the parallel configuration shows a wider spatial range for performing measurements, which makes it a more favorable design from this point of view. Other factors, such as manufacturing, maintenance, limitations on material resources, etc, could be the determining factors that favor one concept over the others.

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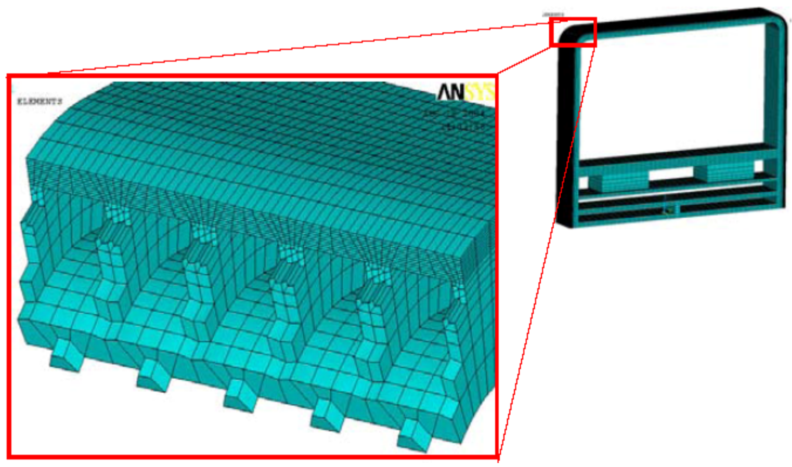
3.2 FIRST WALL THERMOMECHANICAL STRUCTURED ANALYSIS

The model created for this simulation is a subset of the full blanket module (Figure 3.12). It represents one full pass of fluid through the unit and contains five channels. For increased accuracy and to facilitate the eventual transient analysis of the first wall heating process, the unit is modeled with ten quadrilateral elements across the first wall. This cross section is constant around the outer perimeter of the model. The section in the center is solid and also meshed with solid quadrilateral elements. The model is held in place by a “sliding” condition on the back face of the model. The model is fixed across the back face from movement in the perpendicular direction. A single node in the center is also constrained from motion in the X and Y directions. This holds the model without either over-constraining or introducing unnecessary stress concentrations. The computational model employs 280,570 elements.



The material used for this analysis is F82H Steel and the properties were taken from Tavassoli et al.[8] A summary of the properties is as follows: Density (g/mm^3): 7.871E-006 Young’s Modulus (GPa): 217.26 at 20 °C; 177.59 at 600 °C Specific Heat (J/g-k): 470 at 20 °C; 810 at 700 °C Thermal Expansion (ppm/C): 1.04e-5 at 20 °C; 1.24e-5 at 700 °C Thermal Conductivity (W/mm-K): .033

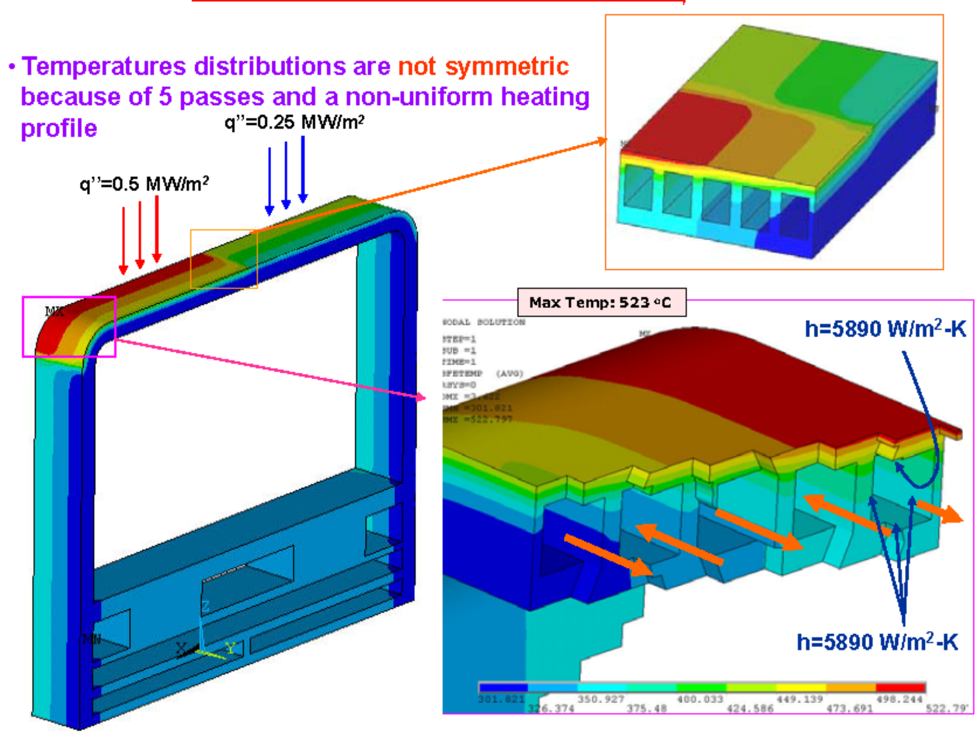


Figure 3.13. Temperature profile seen in one first wall segment of 5 channels coolant path

The analysis is done in two phases. First a thermal analysis is performed by applying a heat flux of .0.5 MW/m^2 to one half of the front wall and a heat flux of .0.25 MW/m^2 to the other half (assumed for ITER first wall design) as shown in Figure 3.13. The total thickness: 28 mm (including 5 mm front and 7 mm back). The coolant channel dimension is 16 x 13 mm^2 with a pitch of 18.2 mm.To simulate the coolant flowing through the five channels, a convection condition is applied to the internal walls of the channels. The convection coefficient for the faces directly opposite the first wall is given as 5890 W/m^{2}K (Table 3.5). The result of the thermal analysis shows a maximum temperature of 522.8ºC (Figure 3.13), which is below the maximum allowable temperature of 550ºC. The calculated temperature profile in the first wall reflects combined features of the cooling scheme and a non-uniform heat flux distribution, in which first wall temperature gradually increases as the helium moves from the first coolant channel to the last (the fifth) channel and is hotter on the high heat flux side.

Table 3.5. Key Operating Parameters for Example Solid Breeder Submodule

Parameter	Design value
Submodule size	0.73x0.91x0.6 m^3
Surface heat flux	0.25- 0.5 MW/m^2
Neutron wall load	0.78 MW/m^2
Helium coolant pressure	8 MPa
Helium inlet/outlet temperature	300/500°C
Mass flow rate to first wall	0.9 kg/s
Helium temperature rise from first wall	53°C
First wall coolant velocity	
Coolant hydraulic diameter	538 $^{\circ}$C
Heat transfer coefficient	5890 W/m^{2}K

Following the thermal analysis, a structural analysis was performed using the output nodal temperatures as an applied thermal load for expansion calculations. In addition to this applied thermal loading, a pressure of 8 MPa was applied to the inside of the five channels. The results of the structural analysis show that the maximum stresses lie in the large radius at the top of the model. The peak stresses shown in Figure 3.14 are around 268 MPa. This maximum stress is similar to the maximum stress magnitude reported for JA's Demo design, which is below the yield strength at 550°C of 380 MPa, while the maximum allowable stress for piping design is 2 times of yield strength (ASME). For simplification this model was created with square channels, which induced certain stress concentrations in the corners. A round corner design will be considered in later analysis. The maximum displacement was in the hot side of the unit and was calculated to be 3.51 mm, as shown in Figure 3.15. The nonuniform characteristics of the displacement found in the present analysis is due to a non-uniform ITER surface heat flux profile. The maximum displacement is 3.51 mm.

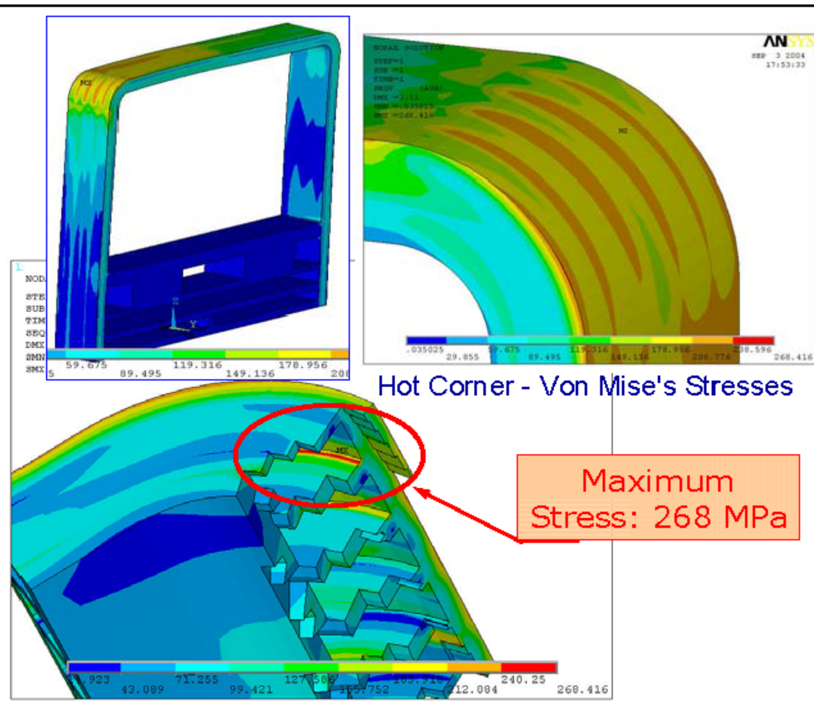


Figure 3.14. Stress distributions near the hot side of the unit (maximum stress is found at the corner of the inner wall)

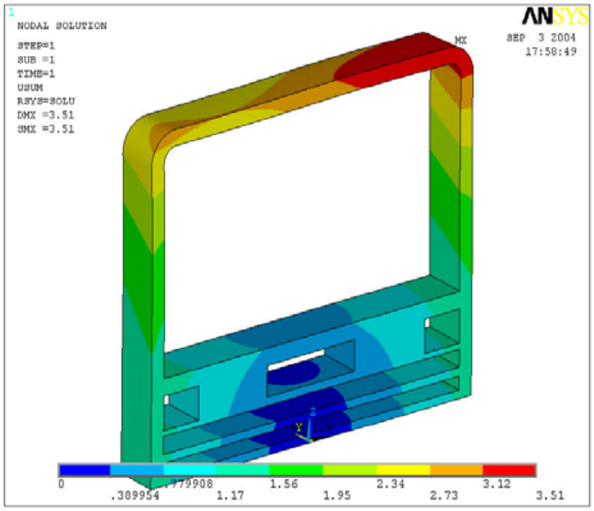


Figure 3.15. Calculated displacement profiles

3.3 BREEDING ZONE THERMAL-HYDRAULICS AND THERMAL ANALYSIS

The heating generation rates obtained from the neutronics calculation were used as inputs for subsequent thermal analysis and design of the heat removal system for the low and high temperature scenarios mentioned in the Introduction. The total heat to be removed from this submodule is 0.784 MW, including the heat deposited on the first wall from the surface heat flux of 0.3 MW/m^2. In the low temperature scenario, the 8 MPa helium coolant enters the submodule at a rate of 0.755 kg/s and a temperature of 100^{o}C and is subsequently distributed into 2 paths to remove the heat generated in the breeder region, which amounts to 0.45 MW. In the low temperature operation design, the helium flows first in the breeder region channels, then in the first wall, since the goal is to keep the breeding material temperature low. In the high temperature design, the scheme is reversed, since the main challenge becomes the cooling of the first wall structure. With the proposed scheme, the exit temperature of the coolant from the breeder zones is 224^{o}C. The combination of a low coolant temperature and a much thinner breeding zone thickness results in the temperatures in the breeder zone falling below 350^{o}C. This ensures that tritium will not be released from the breeder material because of its low diffusion coefficient. The coolant then cools the first wall and leaves the first wall at about 300^{o}C. The resulting first wall maximum temperature is calculated to be 484 C at the highest heat flux location of 0.5 MW/m^2. Since first wall cooling is not an issue for low temperature operation, the submodule can be designed without using by-pass flow, and all the helium flowing in the breeder channels is routed into the first wall structure. The thermal-hydraulic design parameters for different submodule designs are summarized in Table 2.1.

In the high temperature submodule designed for thermomechanics tests, the thickness of the ceramic breeder material bed is increased in order to operate at temperature windows that are typical of power reactors[3.3-1]. This scaling analysis compensates for the lower nuclear heating rates in ITER while increasing thermal resistance and thus the temperature gradient. The helium coolant inlet temperature is now 300^{o}C, to maintain all components at higher temperatures. Under this condition, the main issue is the control of the ferritic steel first wall structure facing the plasma under the maximum heat flux scenario of 0.5 MW/m^2. This is achieved with the higher cooling mass flux of 0.9 kg/s, which ensures a maximum first wall temperature of 538^{o}C (Table 3.5). The higher flow rate must then be reduced before it is sent to the breeder region cooling channels to allow higher temperature operations. This is done with a by-pass line which reduces the main cooling flow rate by 10%[3.3-1]. The helium mainly flows toroidally in the layered configuration layout and flows mostly radially in the edge-on configuration. These different arrangements create very different temperature profiles inside the ceramic breeder bed material. The exit temperature of the helium coolant from the submodule is 500^{o}C.

In the edge-on configuration, the breeder unit width is about 1.8 cm near the first wall and about 3 to 4 cm at 35 cm from the first wall. A two-dimensional thermal analysis of the submodule has been performed with the finite element code ANSYS. The analysis is based on effective properties of the pebble beds, which are treated as continuous materials. The effective thermal conductivity of the lithium orthosilicate material is treated as temperature dependent, with a typical value of 1 W/mK within 400–500^{o}C [3.3-2]. The beryllium pebble beds' effective thermal conductivity depends on the temperature and the stress/strain, and has a typical value of 3-4 W/mK in the same temperature range at a stress value of 0.5 MPa[3.3-2]. The analysis considers volumetric heat generation based on the nuclear heating rates from 2-D nuclear analysis. The description of this analysis is given in the section below.

Thermal Analysis of the Quarter-Port Thermomechanics Submodule

This section presents thermal analysis (steady state and transient) of the proposed US thermomechanics submodule. The objective is to study the thermal profiles of the thermo-mechanics submodule under thermal loads relevant to ITER operating conditions. In general, two configurations are used to arrange the solid breeder and beryllium pebble beds inside the TBMs. In the first configuration (parallel or layer configuration), the solid breeder and beryllium pebble beds are arranged parallel to the first wall, which faces the plasma. In the second one (edge-on configuration), the pebble beds are placed perpendicular to the first wall. The US Quarter-Port Submodule (QPS) features a layer configuration in its left half and an edge-on configuration in its right half, see Figure 3.16. Figure 3.17 shows the dimensions and geometry of the US Quarter-Port Submodule. The cross section area (a) in Figure 3.17 is the cross section area of the first wall. While the cross section areas (b) and (c) belong to the helium channels inside the breeding zone. The QPS has a length of 730mm in the toroidal direction and a width of 600mm in the radial direction. The first and side walls of the QPS have a thickness of 28mm and are made of reduced activation ferritic steel. The front first wall is coated with 2mm-thick layer of beryllium (as a plasma-facing material) to protect the structure. The coolant (helium) is circulated through the first and side walls. Helium mainly flows toroidally in the layer configuration and flows mostly radially in the edge-on configuration.

The six SB pebble beds have radial widths of 17, 20, 20, 20, 22 and 26mm respectively, moving from the front (near the first wall) to the back structure. The five Be pebble beds have radial widths of 36, 40, 41, 40 and 40mm respectively moving from the front to the back. There are two traverse beryllium pebbles beds, at the left side of the layer configuration, which are 20mm in thickness, see Figure 3.17. The edge-on configuration consists of three canister units arranged to be perpendicular to the front first wall. At the front, these units are sepa rated by a distance of 52mm and the far-most right unit is at a distance of 26.5mm from the side first wall. At the interface with the back structure, the units are separated from each other with a distance of 24mm. The central area, which lies between the layer and edge-on configurations, is filled with Be pebble bed. This central area has a thickness of 41.5mm at the front and a thickness of 27.5mm at the back.

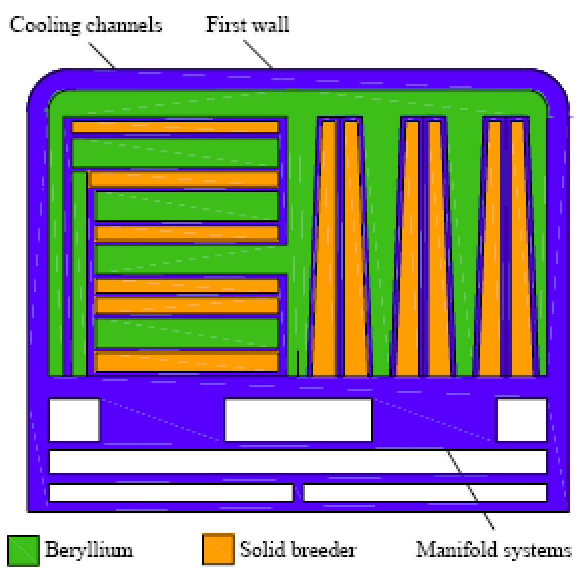


Figure 3.16. : Schematic drawing of the US Quarter-Port Submodule

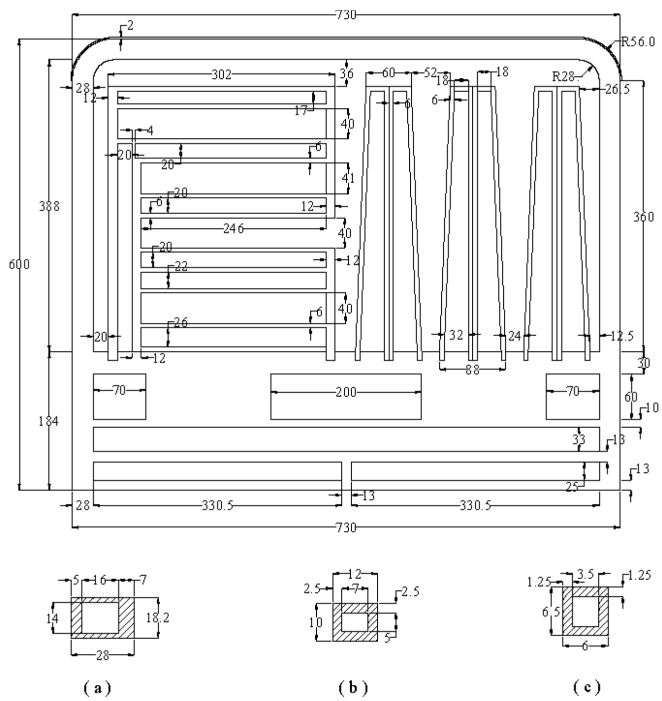


Figure 3.17. Dimensions and geometry of the US Quarter-Port Submodule

Finite element analysis, using ANSYS, was utilized to study the thermal performance and temperature distribution of each part of the QPS. The main steps of performing the thermal analysis of the quarter-port submodule are:

(i) building the model, where the geometry and dimensions of the Quarter-Port Submodule are defined
(ii) meshing the model, where the shape and number of elements of each part in the model were defined: see Figure 3.18,
(iii) applying the boundary conditions and different loads, and
(iv) olving the model and reviewing the results.

<u>Materials Properties</u>
Properties of Beryllium

The first wall of the QPS is protected by a 2mm-layer of beryllium. The properties of this beryllium are summarized as follows:

. • Specific heat, C_p (J/kg.K) = 2432 + 0.6378 T – 0.711 T^{-2} where 300 ≤ T ≤ 1556 K.
. • Density = 1779 kg/m^3. Thermal conductivity is given in Table 1 as a function of temperature.

TABLE 3.6: Thermal conductivity of beryllium

Temperature (°C)	20	100	200	300	400	500	600
k (W/m.K)	160	134.6	114.6	108	100	90	90

Properties of EUROFER-97

The reduced activation ferritic martensitic (RAFM) steel was selected to serve as a structural material. EUROFER-97 is the European product of this RAFM steel. Table 3.7 presents the thermal properties of EUROFER-97.

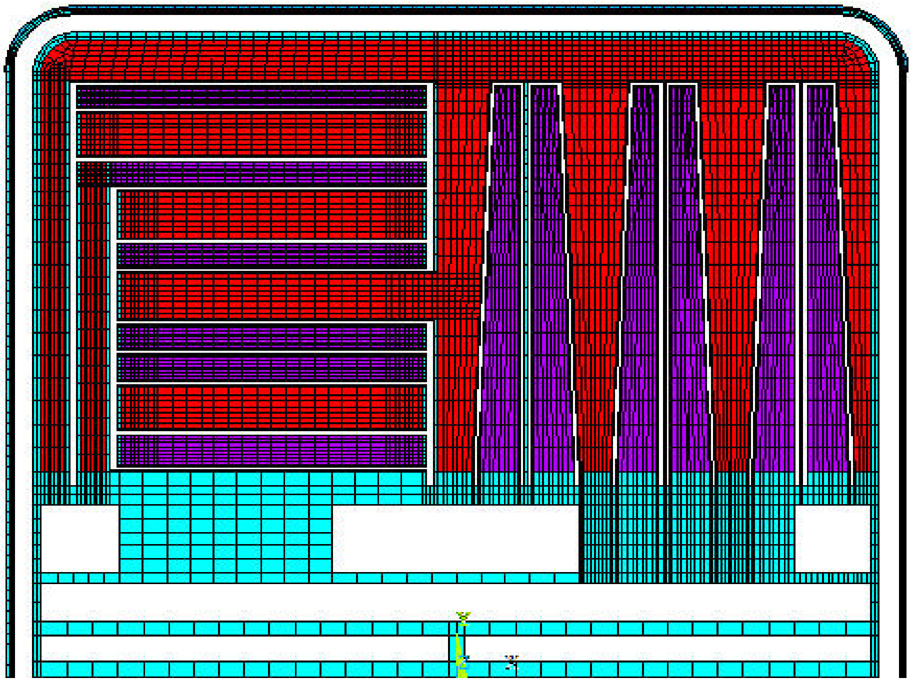


Figure 3.18. ANSYS snapshot showing model after being meshed into elements]

TABLE 3.7: Thermal properties of EUROFER-97 [3.3-2]

Density		Thermal conductivity		Specific heat	
T (ºC)	ρ(kg / m3)	*T* (ºC)	*k* (W / m ºC)	*T* (ºC)	c_p (J / kg ºC)
0	7730.00	0	25.900	0	448.85
20	7730.00	20	25.900	20	448.85
100	7710.00	100	27.000	100	484.11
200	7680.00	200	28.100	200	523.04
300	7650.00	300	28.800	300	562.69
400	7610.00	400	29.200	400	609.96
500	7580.00	500	29.000	500	671.75
600	7540.00	600	28.500	600	754.96
800	7540.00	800	28.500	800	754.96

Properties of Lithium Titanate Pebble Bed

Lithium titanate, in its pebble form, was selected to serve as a solid breeder. Table 3.8 shows specific heat values of lithium titanate pebble bed as a function of temperature. The thermal conductivity of lithium titanate pebble bed, used in this analysis, was taken as an average value of 1.00 W/m.K. The density used is 1483.71 kg/m^3 and it does not change with temperature[3.3-2].

TABLE 3.8: Specific heat of lithium titanate pebble bed [3.3-2]

T (ºC)	Specific heat (J / kg ºC)		
0	1062.00	500	1444.63
20	1080.00	550	1474.35
60	1116.11	600	1502.52
100	1150.90	650	1529.14
200	1233.68	700	1554.21
300	1310.21	750	1577.72
400	1380.52	800	1599.68
300	1310.21	850	1620.09
350	1346.14	900	1638.95
400	1380.52	950	1656.25

Properties of Beryllium Pebbles Bed

Beryllium, in its pebble form, is selected to serve as a neutron multiplier. Table 3.9 shows properties of the beryllium pebble bed. The values of thermal conductivity of beryllium pebble bed are given as a function of temperature and considering the presence of 0.5 MPa compressive stresses.]

TABLE 3.9: Properties of beryllium pebble bed [3.3-2]

T (°C)	Density (kg/m3)	*k* (W/m.°C)	Specific heat (J/kg°C)
0	1166.72	3.45	1741.80
50	1164.48	3.62	1900.97
100	1162.19	3.79	2045.53
200	1157.46	4.13	2294.66
300	1152.54	4.49	2496.83
400	1147.42	4.84	2659.71
500	1142.11	5.18	2790.93
600	1136.61	5.53	2898.14
650	1133.78	5.73	2945.13

Loads and Boundary Conditions

After the model was built and meshed, the appropriate thermal loads and boundary conditions were applied to the model. The analysis type and options should be defined as should the load step options. The thermal loads applied to the model (quarter-port submodule) are relevant to ITER operational conditions, such as: a surface heat flux ranging from 0.25 to 0.50 MW/m^2 and a neutron wall load of 0.78 MW/m^2. The 8MPa-helium coolant enters the first wall at a rate of 0.9kg/s at a temperature of 300°C (a typical value of helium inlet temperature used in many helium cooled blanket designs with ferritic steel as a structural material) for surface heat removal. A relatively high velocity is needed to provide high heat transfer coefficient for removing the surface heat load of 0.5 MW/m^2 [3.3-1].

The loads and boundary conditions can be summarized in the following points:

1. Heat generation was applied everywhere in the model to simulate the nuclear heating. The nuclear heating values were applied as a function of the radial position. These calculations are one-dimensional (radial direction) and are based on ITER operational conditions (neutron wall load of 0.78 MW/m^2).
2. A heat flux of 0.3MW/m^2 was applied to the front first wall.
3. Convection (with h = 6000 W/m^2.K and bulk temperature = 325°C) was applied on the internal walls of the first wall (front and side).
4. Convection (with h = 1000 W/m^2.K and bulk temperature = 400°C) was applied in all helium coolant channels.
5. Convection (with h = 2000 W/m^2.K and bulk temperature = 500°C) was applied in the back coolant channels. Figure 3.19 shows the thermal loads and boundary conditions applied to the Quarter-Port Submodule. Also, Figure 3.20 shows an example of the nuclear heating profiles used in the thermal analysis of the QPS. The shown nuclear heating values are for the solid breeder pebble beds (Edge-on configuration).

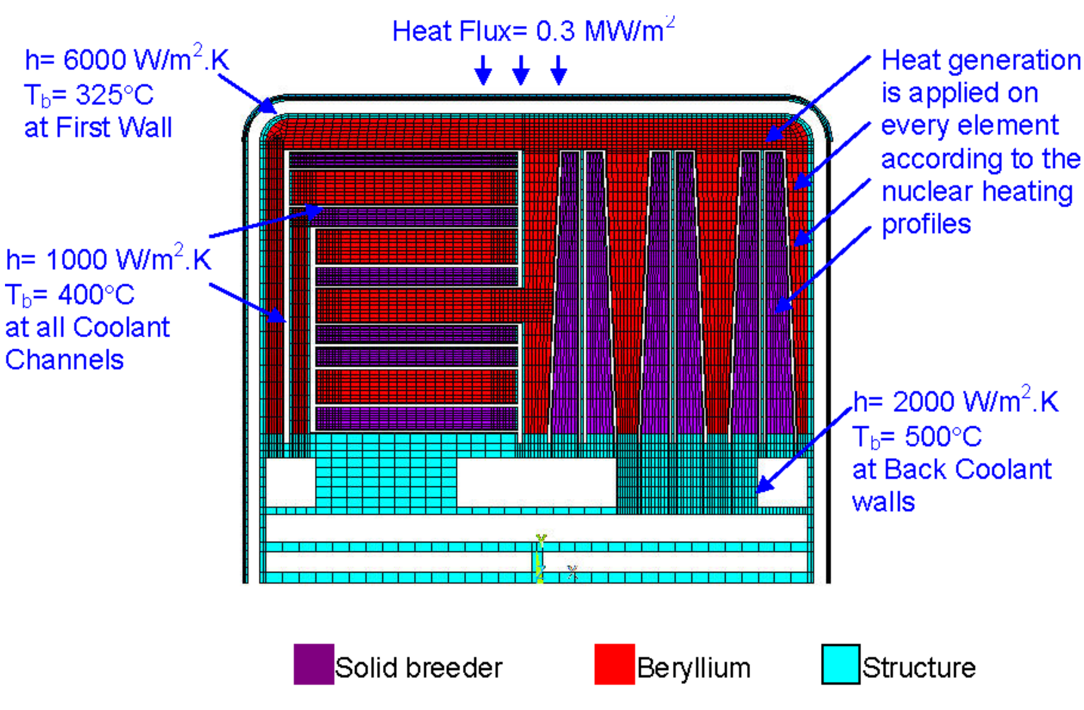


Figure 3.19. Thermal loads and boundary conditions applied to the model

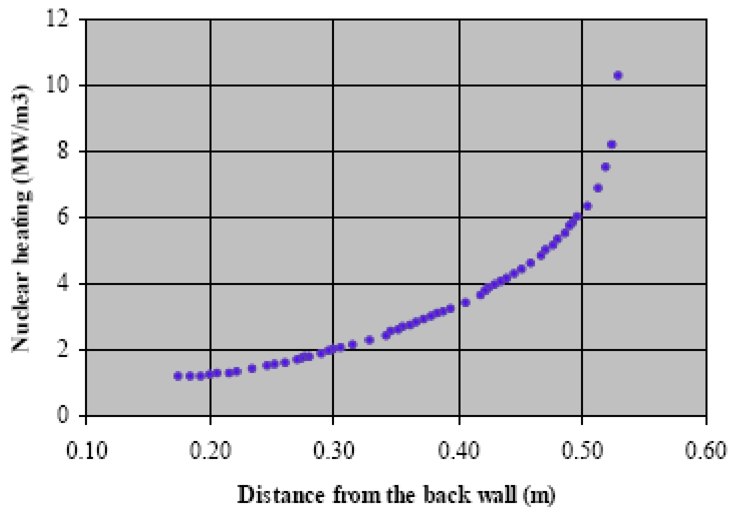


Figure 3.20. Nuclear heating values in the solid breeder pebble beds (Edge-on configuration)

Results of the Steady State Thermal Analysis

In this section, the results of the steady state thermal analysis of the Quarter-Port Submodule are presented. Figure 3.21 shows the temperature distribution (thermal profile) of the Quarter-Port Submodule. The maximum temperature is 790.76°C and it occurs at the solid breeder pebble bed (the closest bed to the front first wall) at the left half (edge-on configuration) of the model. The minimum temperature is 325.125°C and it occurs mostly at structure of side first walls and the inner structure of the front first wall. The calculated temperatures, in the solid breeder and beryllium pebble beds, are below the typical operating temperature limits of 850°C for solid breeder pebble beds and 600°C for beryllium.

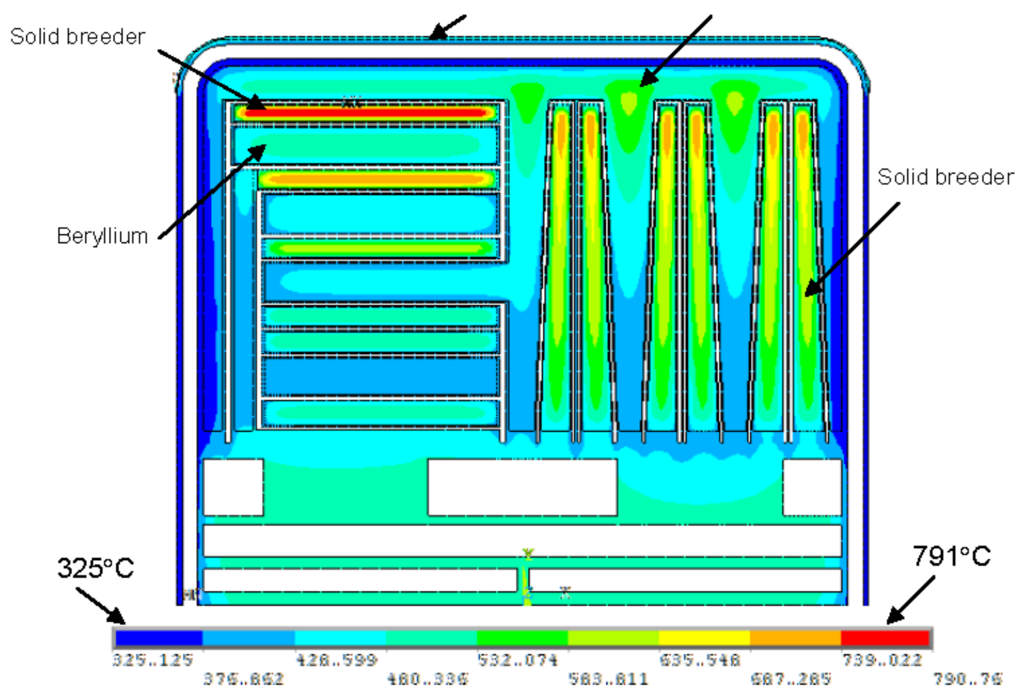


Figure 3.21. Temperature distribution in the Quarter-Port Submodule

For the Thermo-Mechanics Test Blanket Module (TM-TBM), the beryllium layer attached to the front first wall has a design maximum temperature of 545°C. By checking the temperature values in Figure 6, it is noted that the temperatures of the beryllium layer are below the design temperature limit. Also, the temperatures of the first walls' structure are below the design maximum temperature (539°C) for the TMTBM. The temperature contours show that the temperatures of the coolant channels in both configurations (edge-on and layer) are below the design temperature limit (550°C) for the TM-TBM. For the TM-TBM, the beryllium pebble beds have a design maximum temperature of 600-650°C. Figure 3.22 shows the temperature distribution of the beryllium pebble beds inside the Quarter-Port Submodule. The temperature contours limits were specified to: 320-650°C in this figure. For the beryllium pebble beds in the edge-on configuration, the temperature difference is 240°C over a thickness of about 30mm. By scanning all the areas of the beryllium pebble beds inside the model, one can see that the temperatures of the beryllium pebble beds are below the design temperature limit (600-650°C). The hottest spots of the beryllium pebbles beds can be seen in the right half (edge-on configuration) of the model. These hot spots are near the front first wall of the edge-on configuration. The maximum temperature in these hot regions is about 613°C, which is really close to the maximum design limit in some solid breeder blankets. Therefore, these hot regions should be taken into consideration in any future evaluation of the Quarter-Port Submodule.

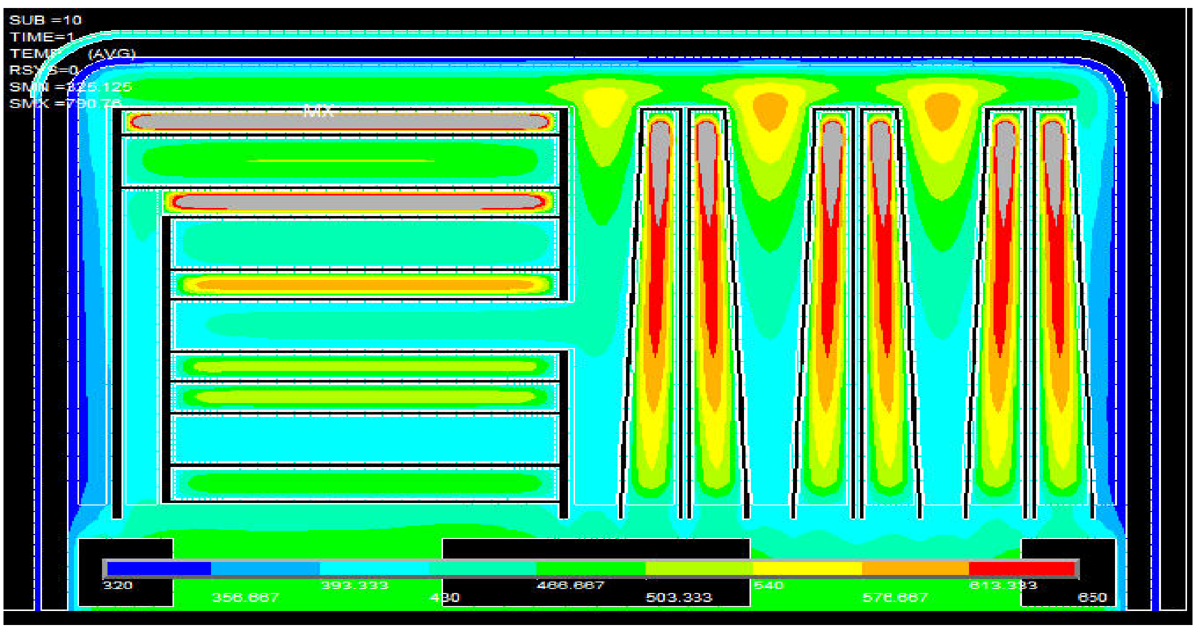


Figure 3.22. Temperature distribution in the beryllium pebble beds (temperature contour limits: 320-650ºC)

For the TM-TBM, the solid breeder pebble beds have a design maximum temperature of 850°C and minimum temperature of 400°C. Figure 3.23 shows the temperature distribution of the breeder pebble beds where temperature contours limits were specified to 400-791°C. This figure shows that the temperatures of the breeder pebble beds are within the design range (400-850°C). Figure 8 shows that the hottest region in all breeder pebble beds is located at the first (closest to the front first wall) bed in the edge-on configuration. The solid breeder pebble beds have different temperature profiles and ranges inside the quarter-port submodule. For example, the temperature gradient is mainly in the radial direction (normal to the coolant channels) in the breeder beds of the layer configuration. On the other side (edge-on configuration), two temperature gradients (radial & toroidal) are found in the breeder beds. These two-dimensional temperature gradients impact the thermal performance of the solid breeder pebble beds.

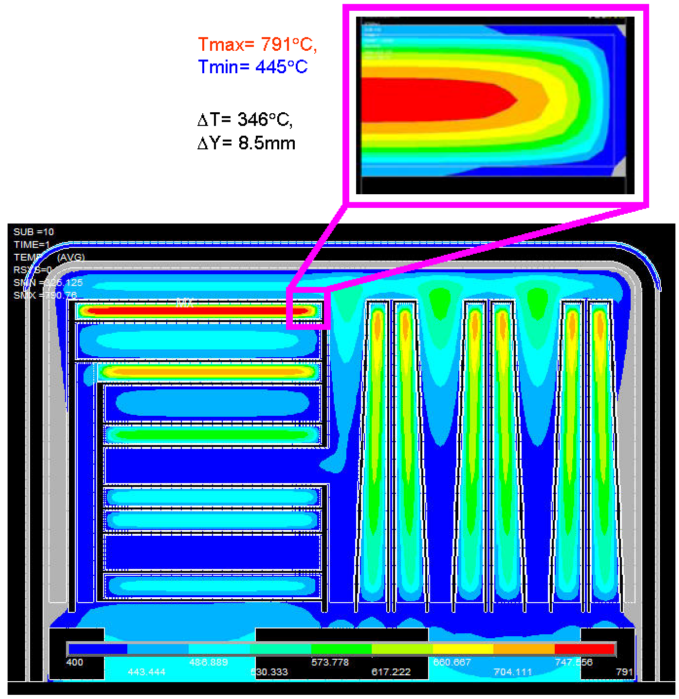


Figure 3.23. Thermal profile of the solid breeder pebble beds (temperature contour limits: 400-791°C)

Figure 3.23 also shows the thermal profile of the first (closest to the front first wall) solid breeder pebble bed in the left half (layer configuration) of the model. The temperature gradient (in the radial direction) is 346°C over a thickness of ~9mm in this solid breeder bed. On the other side, the solid breeder pebble beds in the edge-on configuration have a temperature gradient (in the toroidal direction) of 320°C over a thickness of ~10mm. These temperature gradients are significant and contribute to the thermal behavior of the solid breeder pebble beds.

Results of the Transient Thermal Analysis

A transient thermal analysis was also performed for the Quarter-Port Submodule. The objective of this transient thermal analysis is to evaluate the equilibrium state of temperatures of the QPS under ITER one cycle pulse operation. Also, it helps to determine how long the burn cycle should be in order for the whole QPS to reach equilibrium. Figure 3.24 shows the ITER pulse load cycle used in the transient thermal analysis of the QPS. Figures 25 to 28 show the thermal profile of the QPS with the pulse load after 30, 430, 490, and 800 seconds respectively. Also, Fig. 3.29 shows the temperature-time curves for six different locations inside the QPS (the corresponding locations and numbers are shown in Fig. 3.28).

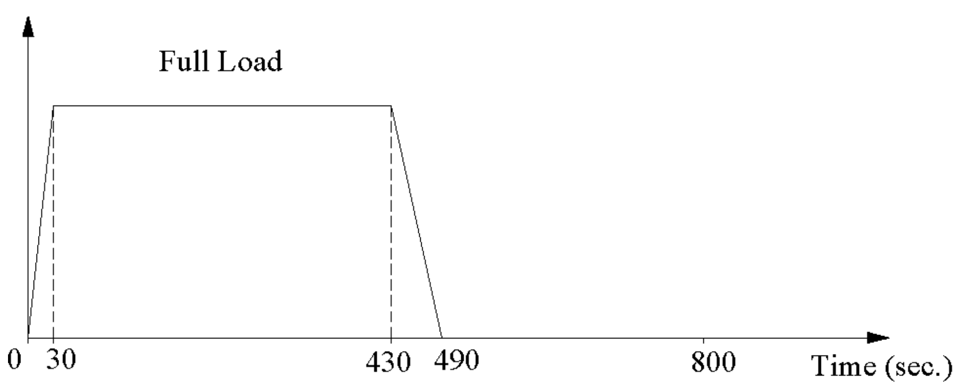


Figure 3.24. ITER pulse load cycle

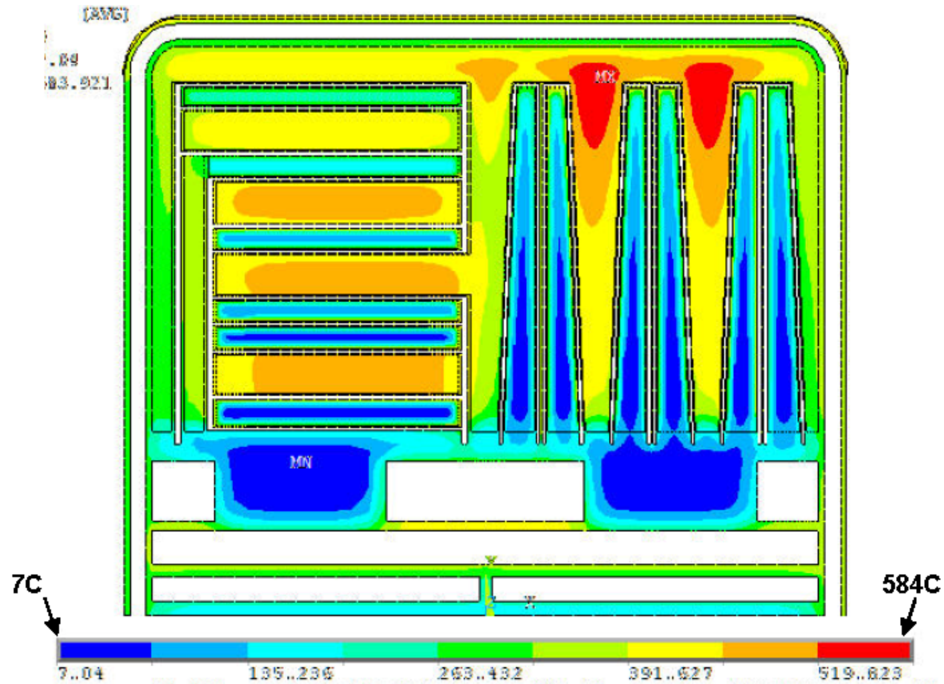


Figure 3.25. Thermal profile of the QPS after 30sec. of the pulse load

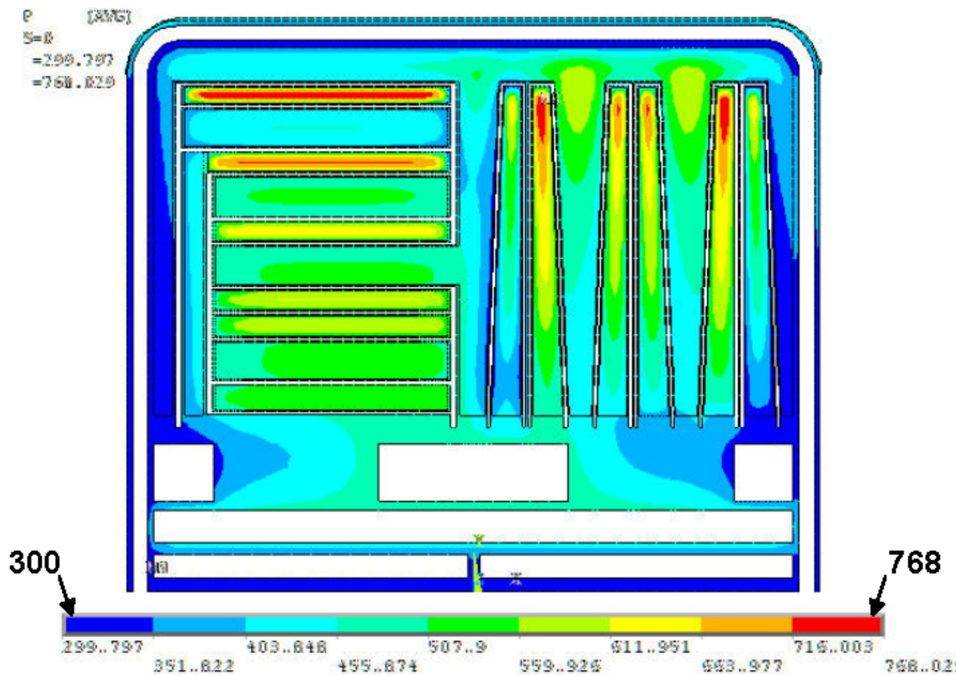


Figure 3.26. Thermal profile of the QPS after 430sec. of the pulse load

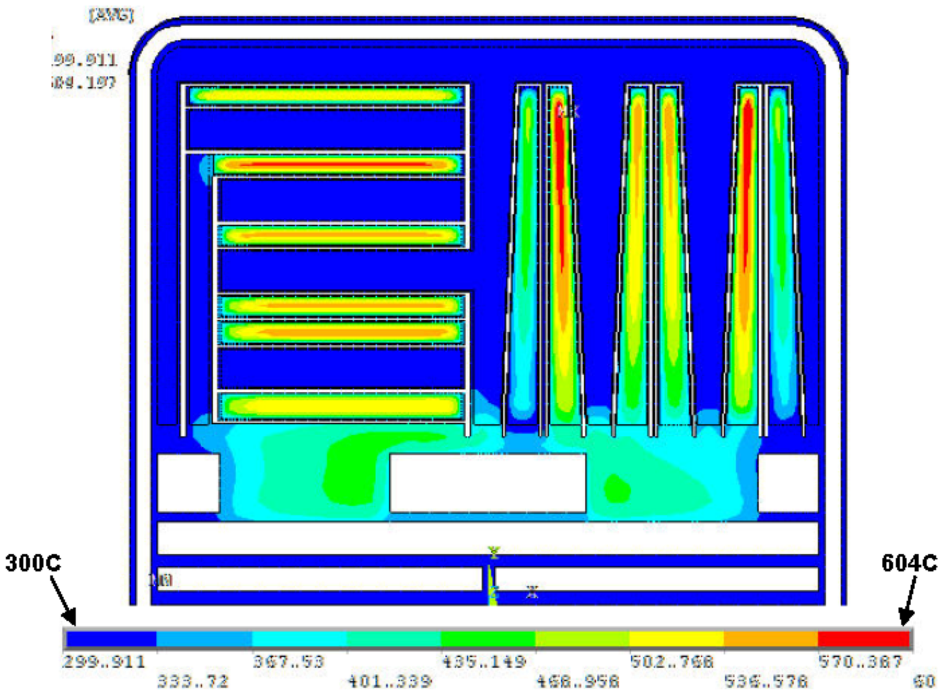


Figure 3.27. Thermal profile of the QPS after 490sec. of the pulse load

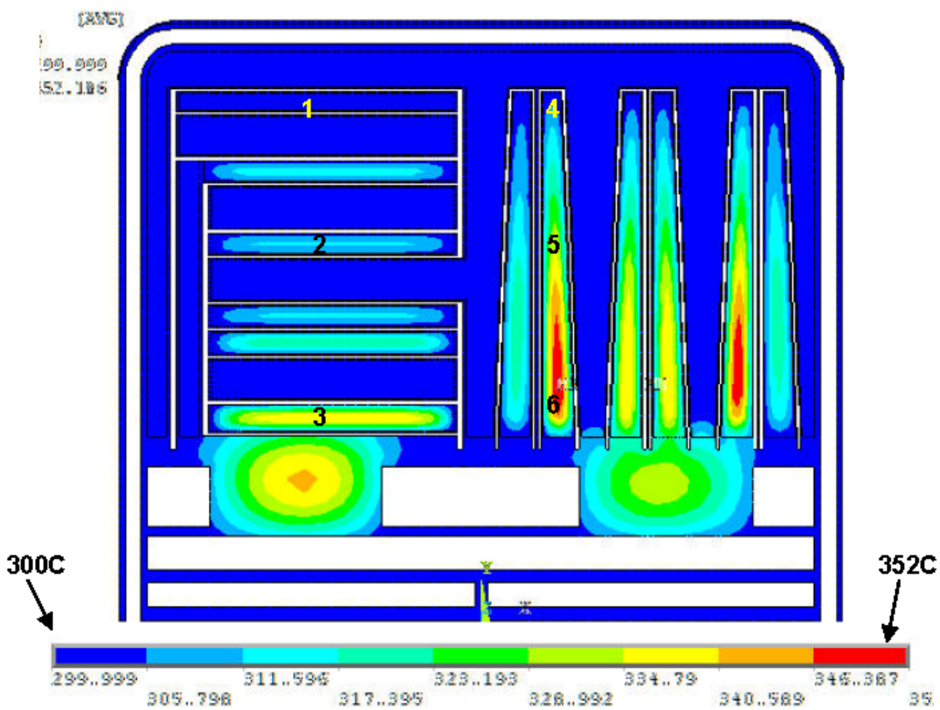


Figure 3.28. Thermal profile of the QPS after 800sec. of the pulse load

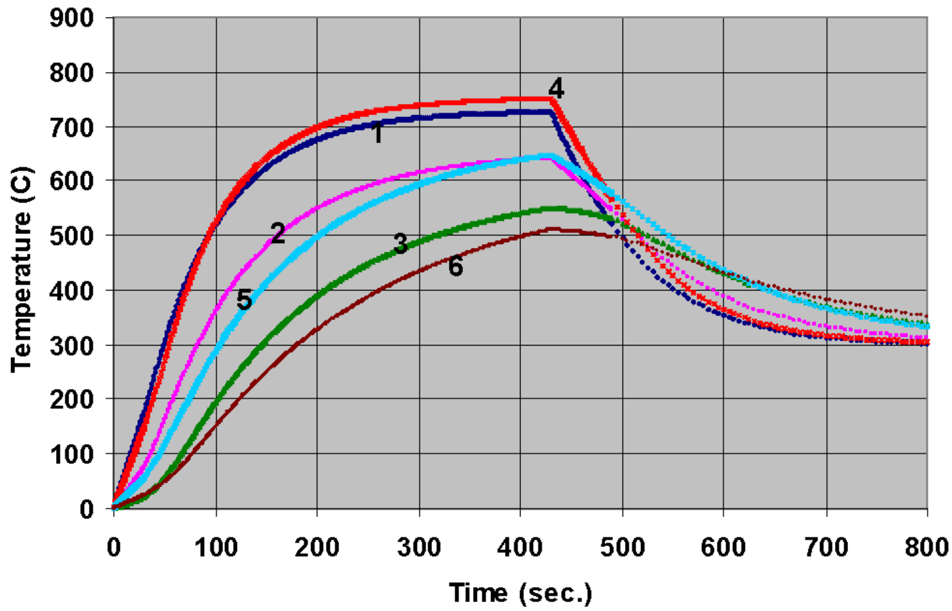


Figure 3.29. Temperature history for different locations inside the QPS after 800sec.

3.3.1 Pebble Bed Thermo-Mechanics Analysis

The thermo-mechanical behavior of the lithium-based ceramic pebble bed at high temperature is very complicated to simulate numerically. First, the material properties depend on the temperature and on the stress and strain condition. Second, the contact area among the pebbles and with the structure is essentially zero initially because of the pebbles' spherical shape, and increases in time at high temperature because of the material plastic deformation. This impacts the thermal conductivity of the bed as a whole and generates local points with very high stress that are not accountable with a continuous material model with effective properties. Also, since creep compaction plays an important role at temperatures above 650°C in the ceramic breeder materials the analysis must include the effect of repeated operative cycles along an extended period of time. As a first approach, the thermomechanics submodule has been analyzed with the

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finite element code MARC[3.3-3. The objective of the analysis is to characterize the general pebble thermomechanics behavior of the submodule as a function of time, and specifically to look at the effect of creep compaction on stress relaxation and the possible formation of gaps between the pebbles and the structure, which could lead to the formation of hot spots. In the continuous approach, the pebble bed is treated as a continuous material with the same effective thermo-physical properties introduced for the thermal analysis. In addition, the effective elastic modulus and creep compaction of ceramic breeder *(E_C and ε_c)* and beryllium *(E_B and ε)* pebble beds are related to stress and temperature levels by the expression [3.3-2,3.3-4 to 3.3-5]:

$$E_C = 314\, x\sigma^{0.75} \text{ and } E_B = 1772x\sigma^{0.83}\ MPa \tag{1}$$

and

$$\varepsilon_C = 1.6x11.41x(\sigma)^{0.4}t^{0.2}e^{-9741/T}$$

and

$$\varepsilon_B = 6.40\varepsilon_C$$

where σ is the axial stress in MPa, T temperature in °C, and t time in seconds.

The calculation has been performed for a breeder/beryllium unit representing a sub-unit found in the edge-on configuration as in the unit cell design (Figure 3.30). As shown, the calculational domain represents half of the unit cell breeder unit using symmetric boundary condition at the center line of the second breeder pebble region. The unit cell is designed to address the issue associated with the pebble bed thermomechanical integrity. The design incorporates features of an edge-on blanket configuration with an attempt to minimize the use of beryllium by increasing the breeder width as it moves toward the back of the blanket region. Specifically, an engineering scaling has been applied to reproduce prototypical ceramic breeder pebble bed thermo-mechanics behavior. Since ITER neutron wall load (0.78 MW/m^2) is much smaller than that of a prototype fusion power reactor (i.e. 3 MW/m^2), attention must be paid to correctly modeling the temperatures because of the much lower nuclear heating rates generated in the scale model. Replicating prototype temperature levels requires scaling up the breeder unit dimension by a factor of roughly the square root of the ratio of the neutron wall load between the scale and prototype models. On the other hand, since the coolant temperature determines the minimum operating temperature encountered in the blanket elements, reproducing coolant operating temperatures serves as a starting point for the engineering scaling process. However, the helium coolant entering into the unit cell may be coming directly from the supply line of the helium loop, and thus it may be necessary to raise its temperature from 300oC to 350oC using an external heater located in the port cell area. The exit temperature reproduces a typical prototype helium outlet temperature of 500oC. The total heat generated inside a unit cell is about 35.8 kW, which is removed by a coolant flow rate of 0.046 kg/s.

The calculated stress profiles at the *x-y* plane of this sub-unit, resulting in a combined effect of temperature gradient, differential thermal expansion and structural constraint, is shown in Figure 3.31. Without taking into account thermal creep effect, the calculated von Mises stress profile of this sub-unit, resulting from a combined effect of temperature gradient, differential thermal expansion and structural constraint, shows a maximum stress level of greater than 10 MPa located inside the beryllium pebble bed near the coolant plate (Fig. 3.31). Whether or not this high stress is accurately predicted is the subject of the current research effort. The maximum stress inside the breeder pebble bed of 1.0 MPa is found ~6.5 cm away from the first wall. The stress profiles at the centerline of the breeder zone as a function of distance at different burn times is shown in Fig. 3.32 for analyses with creep and without creep. The peak stress levels drops to 0.5 MPa at the end of the ITER burn cycle when the thermal creep is coupled into the analysis.

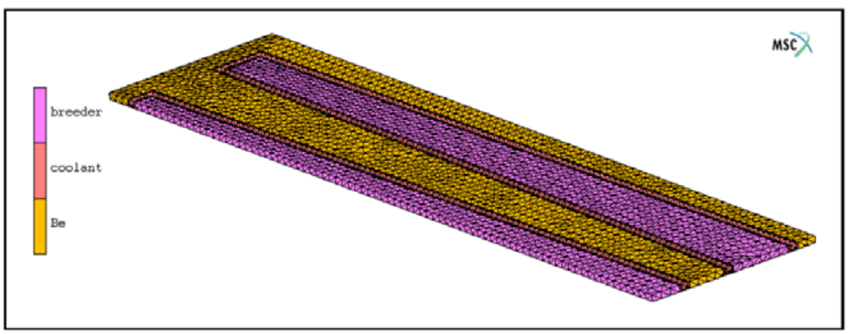

Figure 3.30. Calculation model for a breeding blanket unit cell including breeder (pink color) and beryllium multiplier (yellow color) zones as well as the coolant channel structures

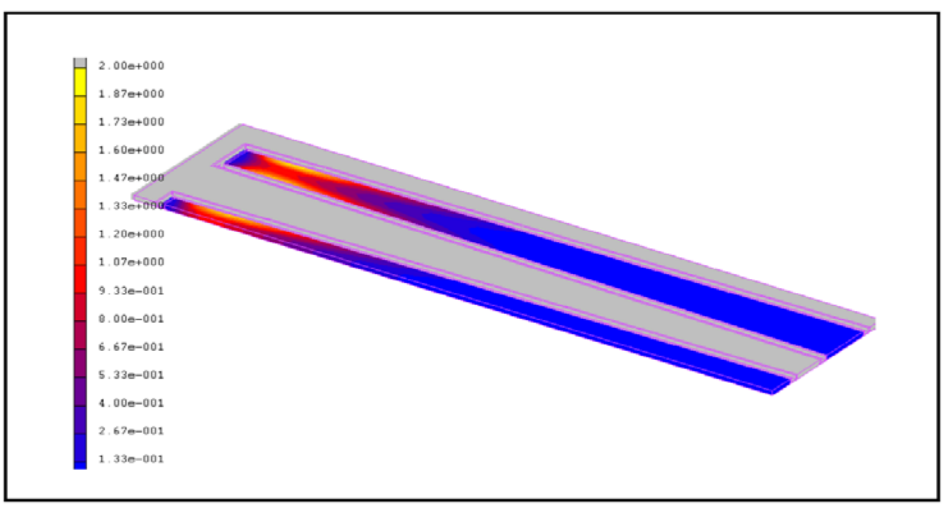

Figure 3.31. Calculated Von Mises stress profile in the breeding unit (emphasizing stress distribution inside the breeder zone.)

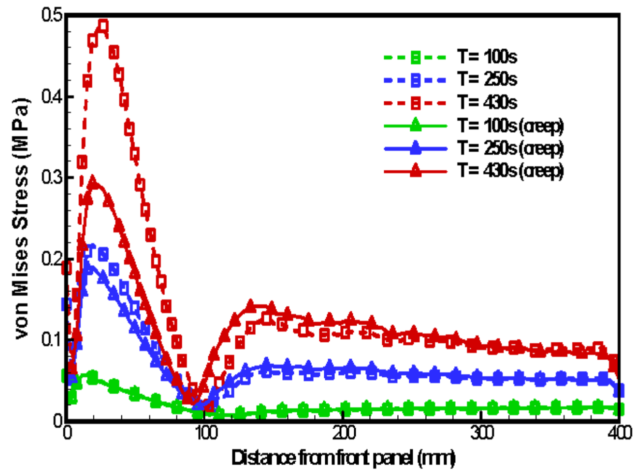

Figure 3.32. Calculated stress histories along the centerline of the breeder pebble bed with and without thermal creep effect

Distance fromfront panel(mm)

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3.4 Tritium Management Analysis

Numerical simulation of tritium permeation from breeding zones to the coolant in the helium cooled pebble-bed blanket has been performed. 2-D and 3-D convection-diffusion models are developed to account for the effects of purge stream convection. Incompressible transient Brinkman model with variable permeability is used in flow calculation, and transient diffusion and convection equations are simulated for the tritium permeation analysis. Tritium partial pressure, concentration and permeation flux are evaluated. The influence of convection on permeation is evaluated under different flow conditions.

There are two main sources of tritium found in the helium coolant. One is permeation into the first wall cooling helium by implantation from the plasma; another source is through the cooling tubes from the breeding zones[3.4-1]. Experimental and analytical studies have shown that under ITER-like plasma conditions, tritium saturation phenomena will take place in which damage to the beryllium surface will enhance the return of the implanted tritium to the plasma and inhibit uptake of tritium. Tritium inventories due to implantation in plasma-facing materials and tritium permeation through the components to the coolant will be reduced. Thus, tritium permeation into the cooling by implantation is very small and can be neglected compared to other sources with a 2–5 mm thick beryllium protective layer[3.4-2-3.4-4].

The problem is defined in three regions as shown in Fig. 3.33: a pebble bed breeding region (a layer design configuration) (1) with helium purge gas flowing through it, a coolant tube structure (2) and a helium coolant region (3). The Navier-Stokes equation, convective and conductive heat transfer equations and convective and diffusive mass transfer equations are solved simultaneously in these three regions.

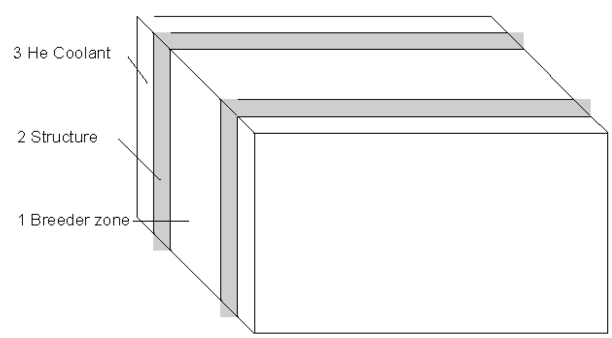

Figure 3.33. Schematic view of computational & physical domain

In the model, the purge gas region has a toroidal length of l = 20cm, a pebble bed radial width of a = 2cm and height h = 1m, a structure thickness of b = 1mm, and a coolant channel width c = 3mm. Other operating parameters include a helium coolant inlet temperature of T_0 = 673K, nuclear heating Q_T = 10^7w/m^3, a tritium production rate Q_C = $8.5 \cdot 10^{-6}$mol/m$^3\cdot$s, He purge gas inlet velocity of u_0 = 3cm/s, and He coolant velocity u = 7.5m/s.

In the purge gas region, the wall effect, which reflects the variations of porosity and permeability in the bed near the wall regions, is considered through the Brinkman model incorporating a variable permeability in the flow equation. The non-dimensional governing equation based on the Brinkman model for the velocity distribution of a fully developed flow in a packed bed is[3.4-5]:

$$\frac{\partial u^*}{\partial t^*} + u^* \nabla u^* = \frac{1}{\mathrm{Re}\varphi^*}\nabla^2 u^* - \nabla p^* - \frac{u^*}{\mathrm{Re}K^*} \tag{1}$$

$$\varphi^* = \varphi_\infty^* \{1 + C_1 \exp[-N_1(a/2 - |y|)/d_p]\} \tag{2}$$

$$K^* = K_\infty^* \{1 + C_2 \exp[-N_2(a/2 - |y|)/d_p]\} \tag{3}$$

where $\varphi_\infty^* = 0.4$ $\quad K_\infty^* = 1.185\times10^{-3} d_p^2$ and $\mathrm{Re} = \frac{\rho u_0 a}{\mu}$

$C_1=1 \quad N_1=2 \quad C_2=20 \quad N_2=4$

where φ^* and K^* are porosity and flow permeability, φ_∞^* and K_∞^* are the porosity and flow permeability at the bulk of the packed bed. The non-dimensional convection-conduction equation and convection-diffusion equation in region 1 are:

$$\frac{\partial T^*}{\partial t^*} + u^* \nabla T^* = \frac{1}{\mathrm{Re}\,\mathrm{Pr}}\nabla^2 T^* + Q_T^* \tag{4}$$

$$\frac{\partial c^*}{\partial t^*} + u^* \nabla c^* = \frac{1}{Pe}\nabla^2 c^* + Q_c^* \tag{5}$$

where $\mathrm{Pr} = \frac{\mu c_p}{k}$ and $Pe = \frac{u_0 a}{D_e}$. Diffusivity of tritium in the purge gas is taken from[3.4-6]. Since tritium transport in the purge is dominated by the ordinary molecular diffusion, the effective diffusivity is estimated as:

$$D_e = D_{AB} / \tau$$

where the ordinary diffusivity is defined as $D_{AB} = (c/P)T^n$ with $c = 4.2\times10^{-9}$ atm·m^2/s for T_2 and $c = 4.6\times10^{-9}$ atm·m^2/s for HT, $n = 1.823$ for HT and T_2. The tortuosity is evaluated within $1/\varepsilon_1 < \tau < 2/\varepsilon_1$. If the total porosity is 0.4, τ changes within 2.5 and 5. In the calculation, τ is taken as 2.5. Sometimes, the effective diffusivity is defined as $D_e = \frac{\varepsilon_1}{\tau} D_{AB}$ [3.4-7]. Under this definition, the diffusivity is reduced by a factor of 6, which makes it more difficult for tritium to diffuse.

At equilibrium, the tritium concentration in the Eurofer at the purge-steel interface is modeled by Sieverts' law

$$C_s = KP^{1/2} \tag{6}$$

where K is solubility constant and P is tritium partial pressure estimated at the purge gas side. The physics properties of different materials used in the calculations are shown in Table 3.7.

Fig. 3.34 shows the velocity profile in the purge gas region if the wall effects are considered; as shown, the velocity profiles achieve equilibrium in a relatively short distance. The change of velocity magnitude, along with the distance from the wall, is shown in Fig. 3.35 to give a better appreciation at such an effect. The velocity near the wall reaches 4 times the inlet velocity.

TABLE 3.10 Physics properties

Helium purge	Density	$0.0724\ \mathrm{kg/m^3}$
	Viscosity	$3.48\times10^{-5}\ \mathrm{kg/m\cdot s}$
	Diffusivity[3.4-4]	$2.403\cdot10^{-4}\ \mathrm{m^2/s}$ at $673K$
Pebble bed	Density	$1.484\times10^{3}\ \mathrm{kg/m^3}$
	Conductivity	$1.065\ \mathrm{w/m\cdot K}$
	Heat capacity	$2.22\times10^{3}\ \mathrm{J/kg\cdot K}$
Eurofer	Density	$7.58\times10^{3}\ \mathrm{kg/m^3}$
	Conductivity	$29.0\ \mathrm{w/m\cdot K}$
	Heat capacity	$671.8\ \mathrm{J/kg\cdot K}$
	Diffusivity	$D = 8.74\cdot10^{-8}\exp(-0.145\mathrm{eV}/\,kT)\ \mathrm{m^2/s}$
	Solubility	$K = 0.364\exp(-0.277\mathrm{eV}/kT)\ \mathrm{mol/m^3\cdot Pa^{0.5}}$
Helium coolant	Density	$5.80\ \mathrm{kg/m^3}$
	Conductivity	$0.271\ \mathrm{w/m\cdot K}$
	Heat capacity	$5.192\times10^{3}\ \mathrm{J/kg\cdot K}$
	Diffusivity[3.4-4]	$7.51\cdot10^{-6}\ \mathrm{m^2/s}$ at $673K$

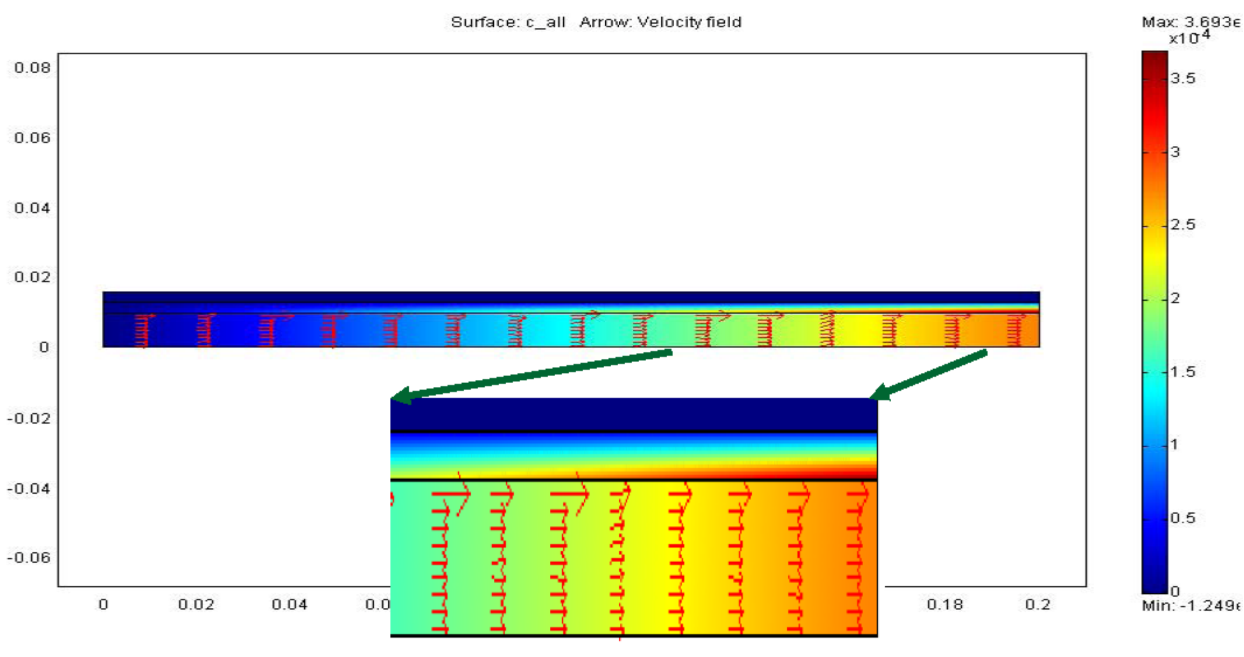

Figure 3.34. Purge gas velocity profile along the purge gas flow path

Fig. 3.36 shows the calculated tritium partial pressure profile in the breeder region along the purge flow direction in comparison with the no-permeation case. The calculated tritium partial pressure at the outlet is about 0.26 Pa, while it reaches 0.31 Pa if no permeation occurs from the purge gas stream. In addition, there is a slight difference between the calculated tritium partial pressure profiles for the realistic velocity and the uniform velocity cases. The partial pressure from the realistic velocity profile is higher than that found from the uniform velocity profile. This shows that the jet velocity profile near the wall region in a porous flow has a slight benefit in preventing tritium permeation and results in a slightly higher tritium partial pressure, but in the given conditions, this additional advantage is relatively small.

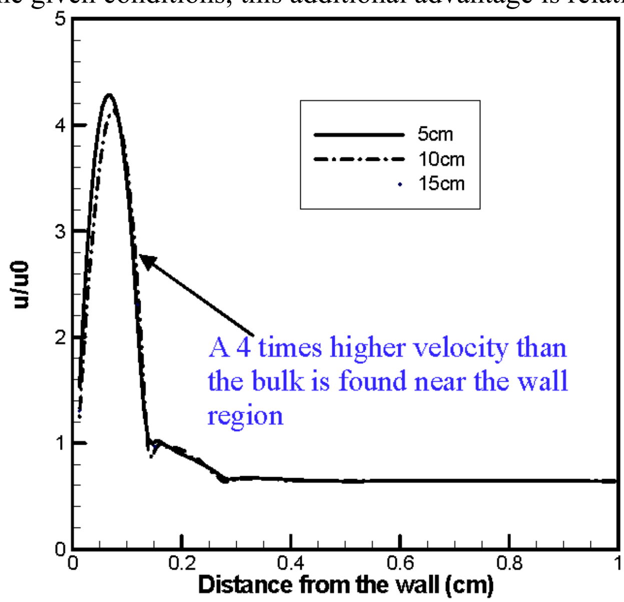

Figure 3.35. Velocity variation with distance from the wall

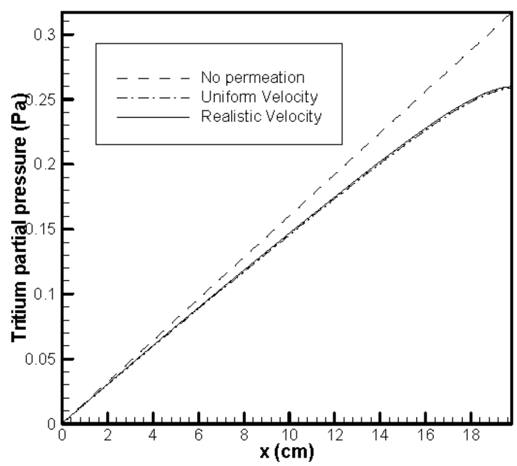

Figure 3.36. Average tritium partial pressure along purge gas flow direction

For a realistic porous flow velocity case, the tritium concentration and partial pressure vary with the Y axes (pebble bed width) direction. Fig. 3.37 shows the tritium concentration distribution with Y at X = 5 cm. The concentration is lower near the wall and higher in the center, while for the uniform velocity case the concentration remains constant except for a small decrease near the wall. The concentration distribution reflects the influence of the velocity profile, but this influence is very small at the parameter ranges studied. Because the purge gas convection has only small effects on temperature distribution, the temperature profile in the breeder region is parabolic. The total permeation in the calculated region is 1.281 mg/day for a realistic porous velocity and 1.283 mg/day in a uniform velocity flow case, nearly 7% of the total production.

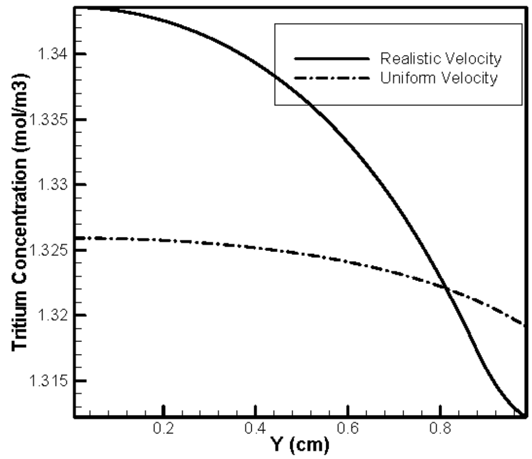

Figure 3.37. Tritium concentration distributions at X = 5cm

To evaluate the convection effect on the tritium permeation, three different inlet velocities are chosen for comparison. The conditions are set to keep the outlet tritium partial pressure the same if no permeation occurs. The results are sumamarized in Table 3.8. Fig. 3.38 shows the tritium partial pressure change along flow direction with different inlet velocities. It can be seen from the table and the figure that for both realistic and uniform flows, if the inlet velocity is increased, the tritium partial pressure at the outlet increases and approaches the value found in the no-permeation case. This shows that convection plays an important role in reducing tritium permeation. The penalty is in the increase of purge gas hydraulic pressure drop and subsequent pumping power. Certainly, there is room for optimizing the purge gas flow conditions from both points of view.

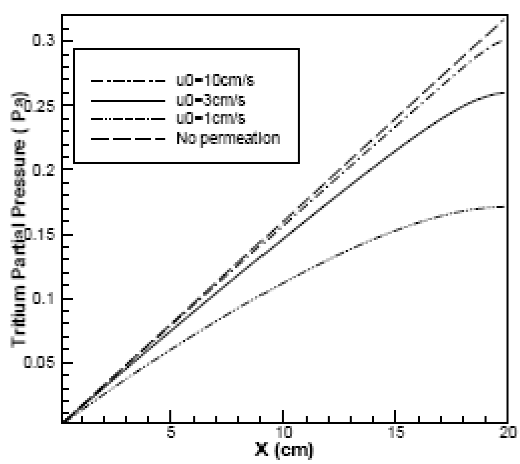

Figure 3.38. Tritium partial pressure with different inlet velocities

TABLE 3.11 Summary Table Calculated permeation rate appears high and unacceptable without taking into account isotope swamping effects or using permeation reduction barriers.

		Without H_2		With 100 wppm H_2	
	Purge gas velocity	**Fractional permeation**	PHT at 1 m downstream	**Fractional permeation**	PHT at 1 m downstream
T= 673 K Eurofer	0.01 m/s	5.62%	4.23 Pa	0.80%	4.46 Pa
	0.03 m/s	3.3%	1.50	**0.271%**	1.56
	0.05 m/s	2.55%	0.92		
	0.1 m/s	1.8%	0.47		
F82 H	0.03 m/s			**0.56%**	1.55
T= 773 K Eurofer	0.01 m/s	13.81%	4.33	2.21%	4.95
	0.03 m/s	8.18%	1.63	0.716%	1.77
	0.05 m/s	6.36%	1.01	**0.417%**	1.08
	0.1 m/s	4.49%	0.52		
F82H	0.05 m/s			**0.88%**	1.07

Summary

- New analysis, taking into account convective and isotope swamping effects, shows that the operating window exists where the permeation can be low and acceptable (Fractional permeation < 0.5%) without using a permeation reduction barrier for tritium permeation control in solid breeder blanket designs. The operating conditions include a 3 cm/s purge gas velocity with 100 wppm H2 addition at 400°C. A higher velocity of 5 cm/s is needed if the purge gas is running at 500°C.
- Existing data of tritium solubility shows a larger permeation for F82H.
- Additional data on tritium (deuterium) solubility, permeability at lower pressure regimes (<10 Pa) with flowing conditions, will help to resolve this important issue.
- The jet velocity profile found in the packed bed configuration slightly reduces permeation. The effect is less significant due to a much faster tritium diffusion time.
- The next step of analysis is to include flow in the complex geometry, temperature distribution, and tritium production profiles.

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3.5 Electromagnetic Analysis (TBD)

3.6 SAFETY ANALYSIS

As stated in Appendix A of the ITER-FEAT Generic Site Safety Report[3.6-1], the safety assessment to date of TBMs has addressed a number of concerns or issues that are directly caused by TBM system failures. Also, some effort has been made to address behavior of the TBMs under hypothetical accident scenarios to assess the ultimate safety margins of the TBMs. Three groups of accidents are judged to cover all accident scenarios envisaged in incidents and accidents involving the TBMs, which are: (1) in-vessel TBM coolant leaks, (2) in-TBM breeder box coolant leaks, and (3) ex-vessel TBM ancillary coolant leaks. These events were selected to address, where applicable, the following ITER-FEAT reactor safety concerns: (1) VV pressurization, (2) vault pressure build-up, (3) purge gas system pressurization, (4) temperature evolution in the TBM, (5) decay heat removal capability, (6) tritium and activation products release from the TBM system, and (7) hydrogen and heat production from chemical reactions.

The US Home Team test objectives regarding the helium-cooled pebble bed (HCPB) blanket module call for a series of test articles of different designs and missions [see Section 2.1]. Of the various US HCPB modules for ITER testing, the so-called Neutronics and Tritium Production (NT-TBM) is the one for which more design information is available to-date[3.6-2]. This module can be also used as the Plant Integration (PI) module, as seen in Table 3.9 of this document, and therefore it represents the enveloping case in term of loads, duration, fluence and tritium production. Hence, the safety assessment for the HCPB type series of TBMs is at this stage restricted to the US NT/PI-TBM.

A more detailed safety analysis of the US HCPB TBM can be conducted when the design of this TBM and the corresponding ancillary systems matures. However, the impact on ITER safety from the proposed TBM concept can be inferred from results already reported for the European and Japanese HCPB designs[3.6-3, 3.6-4], which show that all of the effects such as pressurization, heat production and radioactive inventory, are inherently small and do not add significant safety hazards to the basic ITER machine.

3.6.1. SYSTEM DESCRIPTION AND SOURCE TERMS INVOLVED

System description: The US Home Team is proposing two “look-alike” blanket sub-modules, based on the helium-cooled pebble bed (HCPB) ceramic breeder, to be tested in the same test blanket module (TBM) that will occupy a quarter of a port in ITER and will be placed next to the Japanese TBM. One of the US sub-modules is based on a layered configuration in which the breeder pebble beds are parallel to the FW of the TBM, whereas an edge-on configuration is considered for the other sub-module where the breeder beds are placed perpendicular to the FW. The TBM has a toroidal width of 73 cm, a radial depth of 60 cm and a poloidal height of 91 cm. The ceramic breeder is made of Li4SiO4 with 75% Li-6 enrichment (60% packing factor) and beryllium is used as the multiplier. The depth of the FW/Ceramic breeder/Be zone is 41.6 cm in both submodules. The first and side walls of the TBM have a thickness of 2.8 cm and are made of low activation ferritic steel (F82H) with content of ~53% by volume. The helium coolant at high pressure (8 MPa) is routed toroidally through the first and side walls in alternating directions, and then to the TBM cooling panels. This He flow enters the module at a 300°C and exits at a temperature of 500°C. A separate purge gas system at low pressure (0.1 MPa) carries away the tritium generated in breeding material and in beryllium and keeps the partial pressure of tritium in the beds sufficiently low to avoid excessive permeation of tritium in the main coolant system.

The parallel configuration sub-module layout consists of a number of ceramic breeder (CB) and Be multiplier packed bed layers separated by cooling panels and arranged parallel to the first wall. The helium coolant goes through a series of 3 toroidal passes, each pass consisting of a parallel-flow configuration through several parallel cooling panels (PCP) that are 0.6 cm thick. The internal manifolds for these passes consist of four traverse cooling panels (TCP) whose thickness is 1.2 cm. Both the PCP and the TCP have an F82H structure content that is 53% by volume. There are nine CB beds whose thicknesses in the radial direction vary from 0.9 to 1.8 cm, whereas the six horizontal beryllium beds have thicknesses varying from 2 to 4.8 cm. There are two transverse Be beds on the right side of the sub-module that are 2 cm-thick each.

The edge-on sub-module consists of five canister units arranged to be perpendicular to the FW. At the front edge, a distance of 2.3 cm separates these units. The far most left unit is at a distance of 1.5 cm from the side wall of the TBM. Each unit is composed of two sides TCP, one central TCP, one PCP, and two CB beds. The TCP and PCP are 0.6 cm-thick while the central TCP is 1.2 cm-thick. They have F82H structure content of ~55% and helium coolant is routed and returned in the radial direction through these panels. The thickness of the CB traverse beds is 1 cm at the front end of each unit and gradually increases as we move towards the back of the sub-module. At the interface with the back manifold, the units are separated from each other with a distance of 0.9 cm. Thus, the amount of CB increases, while the amount of Be decreases, as one moves towards the back locations. There is a central Be bed which separates the parallel and the edge-on sub-modules which has a thickness of 2.4 cm at the front and 0.7 cm at the back. In both sub-modules, single sized pebbles are assumed for the CB and beryllium beds with a packing fraction of 60%. Lithium ortho-silicate (Li_4SiO_4) with 75% Li-6 enrichment is selected for the CB. Table 3.9 shows the total material masses for the various components of each sub-module and a comparison between the total amounts in the US HCPB TBM compared to the European concept[3.6-3]. A more detailed description of the HCPB TBM and its ancillary systems can be found in Section 2.1 of this report.

TABLE 3.12. Total mass (kg) for sub-module components and comparison to the EU HCPB TBM

Material mass (kg)	Edge-on submodule	Parallel submodule	Total US HCPB	EU HCPB
SS structure	330	260	976	1186
Be mult	96	107	203	251
Breeder	67	51	117	62
Be FW			2.5	3.5

The estimated heat load in the edge-on configuration and in the parallel configuration are 225.35 kW and 224.42 kW, respectively. When the total nuclear heat deposited in the first and side walls as well as in the manifold are accounted for, the total heat load in the US TBM (excluding surface heating) is ~585 kW. This amount of heat should be removed from the TBM with proper design of heat exchangers and an ancillary system. Most of this heat is generated in the breeder (~37%) via Li-6(n,α)t reactions whereas the first and side walls generate ~18% of this heat load.

Tritium inventory: Assuming a fluence of 0.3 MWa/m^2 and average wall load of 0.57 MW/m^2, the total amount of tritium generated in the US TBM is ~16.5 g. Only about 0.17 g of tritium will be generated in the beryllium multiplier. This is equivalent to a production rate of 0.09 g per full-power day, which is similar to the production rate of 0.1 g/d estimated for the EU HCPB design[3.6-3]. This tritium will be collected and processed in the corresponding Tritium Extraction System. The tritium inventory in the US HCPB TBM structures has not been evaluated in detail. However, since the tritium production rate and structural material masses for the EU HCPB concept[3.6-3]are similar to the US HCPB TBM, a preliminary assessment can be made as an upper bound by using the inventory obtained for the EU HCPB TBM. Using this approach, the tritium inventory TBM structure is expected to be ~20 mg, and 40 mg are expected to be present in the beryllium protection layer. For the beryllium multiplier and breeder material a tritium inventory of 18 mg and 3.5 mg, respectively, has been estimated to build up in the EU-TBM during a two-year period with 28 full power days' operation[3.6-3]. The primary coolant contains less than 1 mg of tritium when the partial pressure of HT is kept at a level of 0.3

Pa. This is the maximum concentration envisaged; most of the time it will be kept much lower by the coolant purification subsystem. Future analyses will be made with the TMAP code[3.6-5] to more accurately estimate tritium inventories. However, even when these tritium sources are combined, the total tritium inventory is less than 100 mg. This inventory is 4500 times less than the estimated mobilizable ITER VV tritium inventory of 450 g [3.6-6] produced within the VV by normal operation of ITER.

Structural material radioactive inventory:
Activation analyses for the EU HCPB TBM reveal that the TBM structure surrounding the breeding zone dominates the activity generated in the TBM[3.6-3]. For initial comparison with such analyses we have simulated the activation of the US HCPB TBM FW by imposing an average TBM neutron wall load of 0.78 MW/m^2. Activation assessments for the rest of the TBM components are currently underway, but the results of these assessments should be bounded by the results obtained for the EU HCPB TBM given in reference 3.6-3. Each ITER pulse is composed of 400 s on and 1800 sec of cooling time between pulses. The estimated number of pulses to reach a fluence of 0.3 MWa/m^2 is 41494 pulses. The activation behavior of the TBM material is affected to a large extent by impurities and other minor elements. For the activation and afterheat calculations, it is therefore mandatory to take proper account of impurities of the materials being considered. Table 3.10 displays the elemental composition of F82H assumed for the activation and afterheat calculations of the US HCPB test blanket module.

TABLE 3.13. Elemental composition of structural material F82H used in US HCPB TBM

Element	At. density (at/cc)	Element	At. Density (at/cc)
C	3.90E+20	Ag	6.95E+15
Si	4.17E+20	Cd	2.08E+15
V	2.30E+19	Eu	1.54E+15
Cr	8.11E+21	Dy	1.44E+15
Mn	4.26E+20	Ho	1.42E+15
Fe	7.39E+22	Er	1.40E+15
Co	2.70E+18	Ta	1.81E+19
Ni	3.21E+19	W	5.10E+20
Nb	2.02E+17	Os	4.93E+14
Mo	3.42E+18	Ir	1.22E+15
Pd	7.93E+15	Bi	1.12E+15

Radioactive isotopes will be generated within the TBM F82H structural material during operation as a consequence of neutron irradiation. At shutdown, the isotopes that dominate the FW activity are Fe-55, Mn-56, Mn-54, Cr-51 and W-185, summing up 85% of the total $5.2x10^{15}$ Bq. However, these isotopes are in a form that is difficult to mobilize. One mobilization mechanism that can occur during the accident scenarios being considered by this safety assessment is the oxidation of the F82H structural material in a steam environment. Oxidation data for F82H steel in steam does not presently exist. However, steam oxidation data for ferritic steel HT-9 (12Cr-1Mo) has been taken by Reference[3.6-7]. This data gives temperature-dependent alloy constituent mobilization rates that can be used to estimate the mobilization of F82H radioactive isotopes during accident conditions. Once these mobilization estimates are made based on this data, the dose at the site boundary can be calculated by assuming no confinement holdup and by applying the dose conversion factor calculated by Reference[3.6-8] for stacked releases during average weather conditions. Results from the US DCLL TBM safety analysis[3.6-9], showed that radioactivity mobilization from the F82H structural material led a dose rate at the site boundary ~6x10^{-3} mSv/d at 700ºC. If the releases were to be stacked, a more realistic scenario, the dose rate would be a factor of ~10 less. Given this information, it appears there should be little safety concern regarding this source of radioactivity during accident conditions. However, more detailed analyses for the HCPB will have to be completed in the future to confirm this finding. Table 3.11 shows a comparison between the FW decay heat for the EU and US concepts. The results for the US design appear to be a factor of 3 lower than those reported for the EU HCPB TBM. This is in part due to the difference in FW material masses (the area sustended by the US FW is 70% of the area occupied by the EU HCPB TBM) and most importantly, to the fact that the US HCPB TBM results are based on pulsed irradiation scenario whereas reference[3.6-3] used a steady state simulation. Preliminary activation calculations for the US TBM show an overestimation in the activation results at shutdown when using continuous versus pulsed irradiation by a factor of about 2. This demonstrates again that a preliminary scaling from the results in[3.6-3], according to structural material fractions, is a conservative approach for our purposes.

Table 3.14. Evolution of TBM FW decay heat after ITER shutdown

	Shutdown	1s	1min	1h	1d	30d
EU HCPB	3.05E-03	3.05E-03	2.98E-03	2.27E-03	3.67E-04	1.50E-04
US HCPB	9.91E-04	9.88E-04	9.17E-04	5.60E-04	9.32E-05	5.90E-05

Chemical Energy and Hydrogen Sources:

Energy sources during accidental conditions can be produced by plasma disruptions, delayed plasma shutdown after a cooling disturbance, decay heat, work potential of pressurized coolants, and exothermic chemical reactions. An overview of the energy quantities is given in table 3.12. The decay heat has been scaled by volume fraction from that reported for the EU HCPB TBM[3.6-3]. This should be a conservative assumption due to the use of a steady state approximation instead of the pulsed irradiation, as was shown in the previous section. Detailed activation calculations for the US HCPB concept will be performed in the future. In reference[3.6-3] it was also noted that after one day of cooling, the structural ferritic steel produces 99% of the heat. Hence, the decay heat generated in beryllium and breeder pebbles can be neglected for times longer than one day after shutdown. In the case of the enthalpy from the helium coolant, we have assumed a mass of 10 kg He at 8 MPa and at a temperature of ~ 375°C, to obtain a contribution of 2 MJ. Finally, regarding the chemical energy from reactions of the beryllium with water and air, we have scaled the results provided in[3.6-4] according to the Be inventory present in the US HCPB TBM.

TABLE 3.15. Energy sources (MJ) for the HCPB

Plasma disruption (1.8 MJ/m^2 on a 0.7 m^2 surface)	1.26
Delayed plasma shutdown (normal: 3 s delay, 1 s ramp-down)	2.5
Decay heat integrated over:	
1 minute	0.5
1 hour	25
1 day	240
1 month	2640
Enthalpy of helium coolant (assuming 10 kg at 375°C, 8 MPa)	2
Chemical energy (203 kg Be in pebble beds plus 2.5 kg at FW)	
beryllium/water reaction	8300
beryllium/air reaction	14100

3.6.2 ACCIDENT ANALYSIS

This section contains an overview of accident analyses for the following three scenarios: (1) in-vessel TBM coolant leaks, (2) in-TBM breeder box coolant leaks, (3) ex-vessel TBM ancillary coolant leaks, including a complete loss of active TBM cooling. It must be noted here that most of the conclusions in this section have been inferred from results already reported in Section 2.5 of the European HCPB Design Description Document[3.6-3], which show that all of the effects such as pressurization, heat production and radioactive inventory, are inherently small and do not add significant safety hazards to the basic ITER machine. These results should be an upper bound for the US HCPB design safety assessment, given the similarity between the two concepts and the fact that the US module has an overall smaller radioactivity source term, and smaller energy sources (lower decay heat values and also lower chemical energy). However, more detailed safety analysis of the US HCPB TBM will be conducted when additional design information becomes available.

3.6.2.1 IN-VESSEL TBM COOLANT LEAKS

Identification of causes and accident description: The postulated accident is a multiple break of TBM FW cooling channels with the blow down of the high pressure primary helium coolant into the vacuum vessel (VV). This accident has been classified as a reference event for the TBM (e.g., probability of occurrence > 10^{-6}/a). This type of failure is conceivable to evolve from a relatively small leak, which initiates an intense plasma disruption that deposits 1.8 MJ/m^2 of plasma stored thermal energy onto the TBM FW over a period of time assumed to be 1 s in duration[3.6-6]. The disruption in turn produces high local stress and/or runaway electron damage not only to the TBM but also to other in-vessel components. Consequently, a simultaneous blow-down of TBM FW helium coolant and ITER FW water coolant occurs, injecting helium and water/steam into the ITER VV. This pressurization causes the VV pressure suppression system to open in an attempt to contain the pressure below the VV safety limit of 0.2 MPa. It is to be noted that only the implications associated with the TBM system are considered here. Pressurization of the VV by steam, and effluent release from other than TBM systems are covered elsewhere. At the beginning of the accident, it is assumed that ITER is operating at full power (500 MW) and there is a uniform surface heat flux of 0.25 MW/m^2 at the TBM. A 1-hour loss of off-site power is assumed to occur coincidently with the initiation of this accident, resulting in pump and circulator coast downs for ITER and the TBM ancillary cooling loops. The coincident loss of off-site power is equivalent to a loss of heat sink in the TBM and ITER cooling systems, resulting in in-vessel component temperatures rising due to decay heating. The VV cooling system is assumed to operate in the natural convection mode, maintaining the VV inner surface temperature at or below 135°C.

There is a hypothetical variant to the above base case, that consists of simultaneous failure of a FW helium cooling channel into the TBM breeding zone, allowing steam from the VV to come into contact with the breeder and multiplier, either in situ or with a certain amount of material spilled into the VV (depending on the relative pressure difference between the VV and TBM interior). The objectives and purposes of these scenarios are to:

- Assess VV pressurization caused by the release of TBM coolant
- Show that decay heat is removed passively
- Show that no excessive chemical (Be-steam) reactions occur

Transient analysis results: More information on break sizes and calculational models are included in reference[3.6-3], where analyses are described in detail to address the three objectives mentioned above. These results showed that the contribution to the pressure buildup in the VV is small (8700 Pa). Also, it was found that the TBM FW temperature remains below 500°C by passive radiation and conduction to colder structures. Even in case of additional failure of the FW/breeder zone interface with unlimited steam access to the pebble beds, the estimated hydrogen production is of the order of 12 g, and therefore the chemical heat generated during the transient is insignificant. The tritium and activation products release from the TBM into the VV was also found to be negligible[3.6-3].

3.6.2.2 LOSS OF COOLANT INSIDE BREEDER BOX

Identification of causes and accident description: This accident has been classified as a reference event for the TBM (e.g., probability of occurrence > 10^{-6}/a). The postulated accident is the break of the largest helium cooling tube inside the TBM, resulting in the pressurization of the TBM breeding zones and cooling system. This would allow primary helium to enter the purge gas collection chamber and, thereby, to penetrate the pebble beds. The result would be pressurization of the blanket box and of the tritium extraction subsystem (TES). To prevent overpressure in the blanket box, one or two burst disks are foreseen that vent into the VV at a specified box pressure. Pressurization of the TES will be prevented by fast isolation valves and by an additional pressure regulator in the purge gas return line (TBM to TES) and by a check valve in the purge gas feed line (TES to TBM). Upon rupture of the burst disk(s) the primary helium coolant discharges into the VV and triggers a disruption. The pressure in the loop will then balance out with the pressure in the VV. At the beginning of the accident, it is assumed that ITER is operating at full power (500 MW) and there is a uniform surface heat flux of 0.25 MW/m^2 at the TBM. Due to the internal leak, the pressure in the purge gas chamber rises and propagates into the TES until this system is isolated from the TBM. This happens when the set point of the TES pressure regulator (0.2 MPa) is reached. At that point, the burst disk(s) opens and helium is spilled into the VV and fusion power is terminated by coolant ingress. At the same time, a loss of off-site power is assumed, leading to pump coast down in the main loop. Thus, after the coolant inventory is lost, the FW will be cooled by radiation to the VV and other FW/shield modules, and perhaps by steam convection. As in the previous scenario, the VV cooling system is assumed to operate in natural circulation mode. The objectives and purposes of this scenario are to:
- Assess TBM box and VV pressurization caused by release of TBM coolant
- Demonstrate TES protection from pressurization
- Show that decay heat is removed passively
- Show that no excessive chemical reactions occur
- Show how fusion power shutdown affects the transient

Transient analysis results: Additional details on assumptions and calculational models are included in the EU HCPB TBM design description document[3.6-3]. According to the results reported in this reference, it was found that the peak box pressure remains below design limits of about 2 MPa, with an additional pressure build-up in the VV below 10 kPa. Also, it was demonstrated that pressure pulses in the TBM purge gas chamber will not affect the TES. Finally it was shown that the TBM decay heat can be passively removed by heat transfer to the machine, and that the energy from chemical reactions is insignificant[3.6-3].

3.6.2.3 EX-VESSEL TBM COOLANT LEAKS

Identification of causes and accident description: This accident has been classified as an ultimate safety margin event for the TBM (e.g., probability of occurrence < 10^{-6}/a). In this scenario, a double-ended pipe break in the TBM cooling loop is postulated to occur in a large diameter pipe of the primary loop discharging coolant into the TCWS vault during plasma burn.

At the initiation of the accident, ITER is operating at full power (500 MW). In order to assess how the fusion power shutdown affects the transient, two different values were assumed for TBM surface heat flux: 0.1 to 0.25 MW/m^2. Instead of active plasma shutdown, plasma burn is supposed to be passively terminated once the FW has reached melting temperature of the beryllium protection layer (1290°C). If it should turn out that melting of the FW is not obtained within the current power pulse, then the dwell time followed by the next power pulse and so forth are to be considered until the melting temperature is reached. At a surface heat flux of 0.25 MW/m^2, the FW surface temperature approaches the melting point of beryllium at the end of the first pulse, but reaches the shutdown condition only in the middle of the next pulse. In the course of this scenario, the beryllium pebbles in the front nodes of the TBM get very hot (1100 to 1150 °C). If a nominal surface heat load of 0.1 MW/m^2 is applied, the situation becomes even more critical as the melting point is only reached at the end of the third pulse and the beryllium pebbles in the front nodes would almost assume the FW temperature. In both of these cases, a plasma disruption is postulated as a consequence of FW melting. This disruption will further damage the TBM FW and other in-vessel components such that a steam atmosphere is present inside the VV, which may cause chemical reactions. After the coolant inventory is lost, the FW and the whole TBM will be cooled by radiation to the surrounding in-vessel components, where internal heat transport in the TBM will be taken into account. As in the previous cases, a loss of off-site power coincides with the disruption and the VV cooling system transits to the natural convection mode, maintaining the VV inner surface temperature at or below 135°C. Additional details on assumptions and models for this analysis are included in the EU HCPB TBM Design Description Document[3.6-3].The objectives and purposes of these scenarios are to:
- Show that the pressure transient inside the vault stays within design limits
- Show that post accident cooling is established to a safe shutdown state
- Show that in-vessel hydrogen generation is limited to 2.5 kg
- Show how fusion power shutdown affects the transient

Transient analysis results: Results reported in reference[3.6-3] show that the pressure transients inside the vault stay within design limits with large margin. On the other hand, a signal “pressure high” for leak detection cannot be obtained. The undetected shutdown at FW beryllium melting leads to long shutdown times over more than one pulse sequence and very high temperatures in the TBM with up to 1100°C in the breeder zone. A scenario with stagnant steam or restricted flow to the TES for a hypothetical accident results in less than 1-kg hydrogen generation.

3.6.3 SUMMARY AND CONCLUSION

The safety results described here have been inferred from analyses included in the European HCPB Design Description Document[3.6-3]. These results should be an upper bound for the US HCPB design safety assessment, and show that all of the effects such as pressurization, heat production and radioactive inventory, are inherently small and do not add significant safety hazards to the basic ITER machine. However, more detailed safety analysis will be conducted in the near term for the US HCPB TBM concept.

References

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4.0 DELIVERY AND REQUIRED R & D PLANS PRIOR TO ITER

The R&D prior to fusion testing in ITER is viewed as essential to the ITER TBM program from the following two perspectives: 1) the need for qualification to demonstrate safe performance and acceptable availability, and 2) the need to acquire adequate knowledge to interpret data from ITER testing. It is necessary to eliminate any uncertainties existing in the proposed TBM.

The most important design uncertainties for solid breeder blanket concepts resulting from these issues related to tritium breeding are tritium permeation and recovery and breeder thermomechanical behavior. In particular, the integrity of the solid breeder/clad interface plays a key role impacting solid breeder thermal and tritium release performance. A better understanding of the occurrence of a gap at the interface, the impact of this gap, and the potential and subsequent consequences of particle breakage remains as the near-term focus through continuous efforts on material properties characterizations and consecutive models derivations, as well as the benchmarking of experimental data resembling fusion relevant breeder unit operating conditions. The next stage of model development will focus on identification and quantification of potential failures/limiting factors related to pebble bed material system thermomechanics interactions under cyclic effects.

It is foreseen that a joint task of off-normal tests, including disruption EM forces, would be performed to demonstrate the capability of the TBM to survive off-normal events in ITER. However, since the forces depend upon specific designs, this test can be scheduled towards the end of the R&D when the specific designs are available for testing. The structure for the EM/S module will most likely be the structure resembling the structures of the other modules. Thus, continued design and analysis of TBM for all testing phases, taking into account different testing objectives, could be the most optimum strategy to finalize the EM/S module design. A near-term R&D on the structural material will focus on the issues of fabrication and bonding, database evaluation and design code development. The effort includes incorporating thermal-physical and mechanical properties of the welds and joints into the state-of-the-art finite element thermal and structural codes to ensure that the performance of critical areas of the design is adequately addressed. The irradiation effects should also be considered at the later stage of development. The neutronics work will focus on the development of measuring techniques and instrumentation needs, and performing sophisticated 3-D neutronics calculations to help investigate the effects of the boundary conditions on the TBM designs and results. The near-term focus on the thermomechanics module R&D is the continued development of a predictive capability to address the cyclic effect on the integrity of the pebbles and dimensional stability at the interface, and the modeling of the inter-relationship between the formation of the gap and subsequent temperature and stress responses. To complete the design, continued helium flow stability analysis in the complex distributing and collecting manifolds is needed and a small scale manifold test could be performed to verify the calculations. As a part of the R&D program, an integrated computer code which models the integrated behavior will be developed for design and result analysis. Furthermore, it is highly desirable to develop and advance available tritium production, heating rate, and neutron spectrum measurement techniques for reliable and meaningful neutronics tests in ITER and this may require further instrumentation R&D effort.

It appears that the R&D for the next 3-5 years will continue to focus on the development of the database necessary for TBM design and fabrication and performance prediction. In the meantime, efforts will be made to coordinate with the WSG-1 community to develop qualification criteria and technology needed for the test program of the helium-cooled ceramic breeder blanket concepts. For the remaining years prior to the ITER testing, R&D will focus on the fabrication and testing of the TBM as well as the preparation of the auxiliary systems needed for the testing. During this time, it is the US' intention to involve industrial teams to play a lead role on the TBM fabrication and testing.

APPENDIX A
UP-DATED TBM FUNCTIONS AND DESIGN REQUIREMENTS FROM 1997 US-TEST BLANKET PROGRAM [A-1], 1998 EU-HCPB [A-2] AND 1997 EU-WCLL REPORTS [A-3]

1.0 FUNCTIONS AND DESIGN REQUIREMENTS

1.1 FUNCTIONS

The Test Blanket System(s) performs the following functions:

1.1.1. Breed tritium to demonstrate the technical objectives of the test program.

1.1.2. Produce high-grade heat that is removed with a suitable coolant medium to demonstrate the technical objectives of the test program.

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

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

1.1.5. Protect the superconducting coils, in combination with the vacuum vessel, from excessive nuclear heating and radiation damage.

1.1.6. 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.

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

a.	Nominal Fusion Power	0.5 GW
b.	Maximum Fusion Power Excursions	+20% of 0.3 MW/m^2
c.	Burn time/cycle time	400 s per 2000s
d.	Average surface heat flux	0.3 MW/m^2
	10% of the surface heat flux can be	0.5 MW/m^2
e.	Neutron wall loading	0.78 MW/m^2
f.	Disruption heat load	0.55 MJ/m^2 for 40 ms, 300 cycles per year
g.	Duty factor	0.25

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 and the highest level of nuclear heating as specified above.

1.2.1.3 The Test Blanket shall be designed for a FW boundary fluence of $\geq$ 0.3 MW a/m^2.

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 design.

1.2.1.5 The Test Blanket System shall provide adequate neutron shielding protection to the vacuum vessel and magnets.

1.2.1.6 The Test Blanket System shall generate high grade heat and remove the heat from the blanket system with reactor-relevant coolant conditions comparable or higher than PWRs.

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.

1.2.1.10System 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 / sec. The Test Blanket System should have a leak rate $<10^{-8}$ Pa-m^3 / sec.

1.2.2.3 The Test Blanket System will have to undergo both hot and cold vacuum leak tests.

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

1.2.3. Structural Requirements

1.2.3.1 The Test Blanket System shall be designed to withstand stresses in nominal and accidental situations according to ITER Structural Design Criteria. Details are TBD.

1.2.3.2 The Test Blanket System shall be supported by the vacuum vessel extension and be cantilevered into its nominal position using an appropriate support 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 helium coolant pressure will be less than:

- Normal operation 10 MPa
- Off-normal conditions TBD MPa
- During system test TBD MPa

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 Vacuum Vessel Extension depending on their respective strength.

1.2.4. Electromagnetic Requirements

The system must be designed to withstand the electromagnetic loads resulting from the interaction of the magnetic fields and eddy current 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 resulting from plasma disruptions at 0.55 MJ/m^2, duration of 1 ms and 300 cycles per year, and vertical displacement events (VDE's) with the parameters described in (ITER design guideline TBD) and for the number of disruptions specified in (ITER design guideline TBD). Specific values are TBD.

1.2.5. Thermo-Hydraulic Requirements

1.2.5.1 System Requirements at nominal fusion power of 0.5 GW

1.2.5.1.1 DCLL Design. The blanket is to design to a neutron wall loading of 0.78 MW/m^2 and an average surface heat flux of 0.3 MW/m^2. A peak heat flux of 0.5 MW/m^2 covering 10% of the module surface shall be used for the design and lay-out of the first wall and its coolant circuit. The module will have to be able to withstand transient effects like disruption and VDE as specified by ITER. The nuclear power deposition distribution shall be specified by 3-D neutronics model calculations, results are to be used for detail thermalhydraulic calculations and design. Specific coolant and breeder operating conditions will be selected by the testing objectives of the DCLL TBM design according to the milestones of development to be further evolved.

1.2.5.1.2 HCPB Design. The thermohydraulic system is designed to remove all the heat deposited in the proposed test unit cells/submodules. The heat to be removed is 0.1074 MW in the unit cell option and 0.785 MW if the submodule option is considered. The main coolant is 8 MPa helium. The coolant inlet and outlet temperature are further guided by the testing objectives. In the low temperature operational scenario, the inlet and outlet temperatures are set at 100 and 300ºC, respectively, while in the high temperature operational tests, they are 300 and 500ºC, respectively.

1.2.5.2 System Requirements in off-normal conditions are specified in Heat Loads and Operational Conditions of the TBM document [A-4].

1.2.5.3 The power of the test module shall be iteratively recalculated as the material and specific TBM design evolve.

1.2.5.4 The Test Blanket System first wall and blanket circuit design will be determined by the specific blanket option

1.2.5.4.1 DCLL Design. First wall and Pb-17 Li breeder zone shall be separately cooled by independent cooling circuits. The first wall and structure to be cooled by 8 MPa helium coolant. The breeder zone is cooled by the circulation of Pb-17Li. An over power of 20% and steady state conditions shall be assumed for the circuit design. The shielding, vacuum vessel, and support structure will be cooled with a low-temperature helium coolant or a compatible fluid. These coolant fluids will be preheated by external means to a temperature (300ºC TBD) significantly higher than the Pb-17Li melting point (235ºC) for keeping the Pb-17Li in liquid form and for material degassing. The duration of the heating shall be in the order of TBD hours, the test module shall be able to withstand temperature for degassing and for Pb-17Li fill at least (one TBD) day. The helium coolant circuit shall be equipped with a control system enabling to keeping the coolant inlet temperatures approximately constant during the ITER pulsed operation. The high pressure and the chemical energy of these coolant fluids along with the requirement to isolate and contain tritium-bearing fluids will require the use of intermediate heat exchangers. These heat exchangers will be located near to the test port openings, location TBD. The secondary or perhaps tertiary coolant fluids will interface with the ITER plant systems. The processed tritium streams will interface with the ITER Tritium Plant.

For the last integrated testing phase of the DCLL TBM program, the module will be designed to operate at elevated temperature $\geq 650^{o}$C to demonstrate the generation of high-grade heat. A by-pass coolant system will be coupled to the heat exchanger to assure that the external circulating Pb-17Li temperature away from the TBM will not exceed the compatibility temperature limit of 475ºC (TBD) between the Pb-17Li and FS and the maximum allowable temperature of 550ºC for the FS.

1.2.5.4.2 HCPB Design. The proposed ITER TBM for the helium-cooled solid breeder concept with FS structure option is not to have US independent ancillary equipments, but rather to have a partial or complete sharing of other parties' helium line and auxiliary systems. The main helium coming from and returning to the TCWS is regulated in the helium coolant conditioning system, which is equipped with valves, a heater, and a mixer. The system is housed in the piping integration cask located behind the bioshield plug. The purpose of this coolant conditioning system is to divide the main coolant into a number of cooling streams and regulate the temperature according to the flow condition required for sub-units. The proposed unit cell TBM does not have its own first wall structure. It is housed behind the EU's FW structural box. In this scheme, a stream of helium of about 0.138 kg/s is extracted from the main coolant and fed to cool the three breeder test units. During the thermomechanics tests, this coolant will be preheated to a temperature of 350^{o}C (from 300 ^{o}C) before entering into the units. For optional submodule tests, a much larger amount of 8 MPa helium coolant will be needed (about 0.9 kg/s). The various helium streams from the TBMs will be merged in a mixer into one stream before it is sent back to the TCWS.

1.2.6. Mechanical Requirements

1.2.6.1 The TBM system including its support structure shall be supported by the vacuum vessel extension and the be cantilevered into its nominal position using an appropriate support structure.The corresponding dimensional tolerances and loads (mechanical, thermomechanical and electromagnetical) are TBD. The shield support shall be determined when the TBM support frame, penetration and TBM transport systems are better defined.

1.2.6.2 The coolant and helium purge lines 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 the coolant, breeder, tritium, electrical and diagnostics lines will be routed through the horizontal ports through the Vacuum Vessel, and will be designed to fulfill all requirements of a vacuum and safety boundary, and allowing spatial displacement during thermal transients.

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

1.2.6.5 The Test Blanket First Wall shall be bakeable to ≥ 240^{o}C.

1.2.6.6 The Test Blanket Articles shall be designed to be removable (RH Class 1) by remote handling through the horizontal test ports.

1.2.6.7 The Test Blanket structural connections shall use remote handling compatible connectors, accessible from the back side. The time requirements for this operation shall be minimized.

1.2.7. Electrical Requirements

1.2.7.1 The in-vessel potion of the Test Blanket system shall contribute to meeting the requirement that the combined toroidal resistance of the blanket in-vessel structures and the Vacuum Vessel must be larger than 4μΩ. as specified in (GDRD Section 5.3.3.3.1, update TBD)

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 an average fluence of 1 MWa / m^2 is reached on the FW (Ref. GDRD 5.5.2.3.3.1, update TBD)

1.2.8.2 The Test Blanket System shall be designed so that the nuclear responses for 0.3 MW a/m^2 at the First Wall are limited to a helium production of < 1 appm at all components behind the shield that may need to be rewelded, such as Vacuum Vessel, blanket ancillary 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, update TBD). The peak insulator dose shall be limited to 3x10^8 rad with 0.3 MWa/m^2 at the First Wall.

1.2.8.4 Tritium shall be bred in the test blanket during the DD and DT phases with a tritium breeding ratio from which the self-sufficiency in a power reactor can be extrapolated. 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 handling classification. All Test Blanket System components are to be considered as RH Class 1, except the frames interposing between the modules and the back plate, which are RH Class 2.

1.2.9.2 The Test Blanket System may be removed and installed without disturbing ITER Blanket/shield Modules and ITER operation.

1.2.9.3 The Test Blanket System and its supporting ancillary systems inside the shield must be capable of insertion/removal through the horizontal test ports by fully remote handling.

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 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, update TBD)

Test Blanket:

a. be able to replace a module in 4-8 weeks during scheduled maintenance period

b. be able to repair a leak at a fluid joint within 6 weeks

(Not including time required to locate and isolate leak)

1.2.9.6 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

1.2.9.7 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 TBD
tools	type and capacity TBD

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

pipe size	5-50 mm ID (TBD)
wall thickness	TBD mm
position	from inside pipe

speeds:

welding	TBD cm / s
cutting	TBD cm / s
inspection	TBD cm / s

be capable of joining, cutting and leak testing the of the TBM system

iv. others TBD

1.2.9.8 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 aide the transporter for final positioning and alignment and to protect adjacent components from damage due to collisions.

f. The maximum mass to be supported by the VV extension shall not exceed TBD kg.

1.2.9.9 Transporter Requirements:

a. The size of the TBM and the transportable supporting and ancillary equipment shall be within the transporter dimensions. Maximum transported mass is <50,000 kg (TBD.) Afterheat of (TBD) MW of heat removal capability will be provided by the Transporter, hot cell and /or storage facility. Active cooling of the test articles will be required during the transport. Remote surveillance and monitoring may be required (TBD).

b. The transporter shall be designed with adequate flexible for the accommodation and easy change out of different TBM concepts.

c. The transporter shall be designed to allow intervention or repair in case of the transporter itself or the transporter equipment fail.

1.2.10. Chemical Requirements

The Test Blanket System and its supporting subsystems must be compatible with the breeder and coolant chemistry. The chemistry will be specified with the requirement to limit corrosion, electrochemical, tritium containment and other effects to acceptable levels over the life of the system. Specifications 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 TBM shall be manufactured according to the RCC-MR code class 1 (TBD) with particular emphasis on tolerances between the TBM and interface frame as well as between interface frame and shielding blanket in the following situations:

a. installation and shut-down after operation;

b. nominal operation taking into account the pulsed conditions and irradiation effects (e.g. swelling) on ITER blanket, interface frame and TBM;

c. accidental situations which could lead to deformations of the TBM 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 installation. (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 leak testing
- NDT certification of structural 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 to the transportation, handling, and storing of the various components of the TBM system.

1.2.14. Assembly Requirements

1.2.14.1 The alignment of the Test Blanket First Wall to the magnetic surface of the shielding blanket is TBD. At the equatorial level, a maximum recess of 50 mm with respect to the magnetic surface of the primary first wall can be used for the first wall of the test blanket.

1.2.14.2 The Test Blanket will also have the requirement (TBD) to minimize any gap to adjacent modules in order to minimize neutron streaming.

1.2.14.3 The Test Blanket System shall be installed from the horizontal test ports 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 to all design basis loads.

1.2.14.4 The shielding and vacuum vessel portion of the Test Blanket System shall 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 installation on the ITER device.

Cold leak tests

a.	Internal pressure	TBD MPa w/helium
b.	External pressure	1 Pa
c.	Component temperature	$20^oC/300^oC$ (TBD)
d.	Total leak rate acceptance level	$\leq 1x10^{-8}$ Pa m^3/s

Hot leak tests

a.	Internal pressure	TBD MPa w/helium
b.	External pressure	1 Pa
c.	Component temperature	200-700^oC (TBD)
d.	Total leak rate acceptance level	$\leq 1x10^{-8}$ Pa m^3/s

1.2.15.2 The system must be pressure tested with operational coolant according to the applicable rules for pressure vessels 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 The instrumentation (number and location TBD) of the breeder and cooling circuits shall include:

1.2.16.1.1 DCLL design

- pressure (absolute pressure and pressure drops)
- temperature (at various locations in the Test Blanket system)
- radioactivity in primary and secondary coolant and Pb-17Li
- hydrogen isotope concentration (protium, deuterium, tritium in primary and secondary cooling water and Pb-17Li)
- mass flow-rates in coolant and Pb-17Li
- tritium concentration in Pb-17Li
- gas detection in Pb-17Li
- level (water and Pb-17Li)
- neutron detector in test blanket
- γ scan wires in test blanket
- surveillance of component integrity (TBD)
- leak detection
- positioning and deformation
- stress
- others (TBD)

1.2.16.1.2 HCPB design

- pressure gauges (absolute pressure and differential pressure gauges)
- thermocouples (at various locations in the test blanket units and auxiliary systems)
- tritium measurement system
- flow meter
- neutron detector in test blanket

- γ scan wires in test blanket
- strain gauges inside the test blanket units
- others (TBD)

1.2.16.2 Redundant control systems are required for flow rate, temperature and pressure control in the breeder and coolant circuits. Details are TBD.

1.2.16.3 A data acquisition system shall process the measurement values and shall issue alarm messages in case of abnormal indications (TBD) which shall lead to reactor shut-down in case of confirmed abnormal behaviour. The response time between the detection of an abnormal event and reactor shut-down (< TBD sec) shall be optimized. The development of a licensed safety strategy is TBD.

1.2.17. Decommissioning Requirements

The 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, and 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.

1.2.19.3.1 The materials used in the DCLL test blanket are anticipated to be:

Table 1.2.19-1
Summary of Structural Material Requirements

Structural material	9% Cr martensitic steel (grade is TBD)
First wall structural material	9% Cr martensitic steel (grade is TBD)
First wall protection	Be (form and attachment are TBD)
Breeder material	Pb-17Li, ^{6}Li enrichment 90%
Flow coolant insert	SiC-composite
Shielding	stainless steel (water cooled)
First wall and structure coolant	8 MPa helium
Piping	9% Cr martensitic steel for in-vessel components (grade is TBD), stainless steel for other ancillary equipment

1.2.19.3.2 The materials used in the PCPB test blanket are anticipated to be:

Table 1.2.19-2
Summary of Structural Material Requirements

Structural material	9% Cr martensitic steel (grade is TBD)
First wall structural material	9% Cr martensitic steel (grade is TBD)
First wall protection	2 mm Be (form and attachment are TBD)
Breeder material	Li_4SiO_4 or Li_2TiO_3 pebbles (TBD)
Neutron multiplier	Be pebbles
Shielding	stainless steel (water cooled)
First wall and structure coolant	8 MPa helium
Piping	9% Cr martensitic steel for in-vessel components (grade is TBD), stainless steel for other ancillary equipment

1.2.20. HVAX Requirements

Not directly applicable

1.2.21. Layout 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 The main coolant headers shall be minimum of TBD cm diameter with TBD minimum bend radius everywhere along the required traveling route of in-pipe welding equipment to be determined by the method and approach of cutting/welding.

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

1.2.21.5 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 corresponding ancillary equipment.

1.2.21.6 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 corss-sectional size, between the weld and the point of entry for the welding equipment. Welding and cutting of the coolant manifolds may be from the interior of the pipes.

1.2.21.7 The Test Blanket System shall be designed to be safely drained of all liquids.

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 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 depend on the TBM behavior in off normal conditions and are treated in section 3.6 (Safety Analysis)..

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 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 the 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 analysis 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 in 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 potential for challenging confinement barriers.

Provisional safety design limits are as follows:

- Because beryllium (Be) is used as FW armor material, short term temperatures shall stay below 800^{o}C (TBD) to avoid Be-steam ignition scenarios.
- Long term (decay heat driven) Be FW armor material temperatures shall be limited to 500^{o}C (TBD) to avoid excessive H_2 production.
- The maximum allowable H_2 production inside the Vacuum Vessel is 5 kg (TBD).
- 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 for 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).
- The tritium concentration in the water cooling loops, resulting from coolant activation, permeation through surfaces or leakage, shall stay below 1 Ci/kg.

The consequences for the TBM are TBD.

1.3.4 Safety Assessment. The safety analyses will include but are not limited to the following events (cf. section 3.6 Safety analysis):

- Plasma disturbances (such as disruptions, VDEs, power excursions) resulting in an overload of the TBM.
- Over-pressure in the VV from water LOCAs causing steam formation and H_2 generation on hot TBM FW armor surfaces.
- Temperature/pressure transients of the Test Blanket due to LOFAs or in-/ex-vessel LOCAs in one or both primary heat transfer systems and from in- and ex-vessel LOCAs with eventual chemical interaction between water and Pb-17Li.
- Pressure and temperature transients with related chemical reactions inside the TBM due to water ingress by LOCAs.
- Pressure and temperature transients and related chemical reactions at the FW surface due to air or water ingress into the VV.
- Loss of Heat Sink Accident in one or both cooling circuits
- Loss of Heating Event with undesired solidification of Pb-17Li.
- Loss of Breeder Material Event and related chemical reactions at the FW surface or elsewhere due to Pb-17Li ingress into the VV.
- Mechanical loads to the TBM from magnet accidents and disruptions.

- Others (TBD)

1.3.5 Test Blanket System Safety 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 (cf. 3.6).

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 temperature limits specified in 1.2.22.3 shall be respected by an appropriate design to avoid in-vessel LOCA with the related concerns (radioactivity release, hydrogen production). For this purpose, the cooling circuits for the First Wall and the Breeder Zone shall be separate and independent, each ensuring the cooling of the complete Test Blanket article from an accidental situation to a stable and safe condition. Redundancy/diversity/spatial separation of cooling circuits and components are TBD and a trade-off between component reliability and system availability.

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.

1.3.5.7 The cooling loops are segmented into two independent systems (redundancy, diversity, spatial separation are TBD), see also 2.2.1.

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. The envisaged heat-exchanger location in the pit area limits, however, the possibilities for decay heat removal by natural coolant circulation. It is suggested further to increase by adequate surface treatment (selective coatings), 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 In general, the design should limit:

- the inventory of radioactive, chemically reactive, or toxic dust inside the Vacuum Vessel
- the mobilizable tritium inventory inside the Test Blanket System
- the corrosion products in the Test Blanket System cooling loops
- the tritium concentration in the Test Blanket System coolant system
- the activation products in involved materials
- the toxicity of the involved materials or derivatives formed during an accident
- the chemical energy release potential
- the physical energy release potential

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

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

1.3.5.13 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.

1.3.5.14 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.

1.3.5.15 Other Requirements

1.5 Other Requirements (R&D, maintenance, inspection, code & standard, reliability, etc...)

1.5.1 R&D Requirements. Several aspects of the Test Blanket System require special development, demonstrations, or testing in order to adequately assure that the design satisfies system requirements. The R&D programs will be necessary to: (1) support the blanket system design by confirming its basic viability, influencing the design details in critical areas, determining irradiated and non-irradiated material properties; and (2) to help determine detail design and fabrication parameters, procedures, and specifications. Key design inputs required from the R&D program are described in section 4.1.

1.5.2. Operation and Maintenance

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

1.5.3. Surveillance and In-Service Inspection

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

1.5.4. Quality Assurance

The quality assurance requirements are included in Section 1.2.1.

1.5.5. System Configuration & Essential Features

The configuration and essential features are included in Section 1.2.21

1.5.6. Codes and Standards

Codes and standards requirements are included in Section 1.2.1. The ITER Structural Design Code shall be used wherever applicable.

1.5.7. 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 are given to 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.

1.5.7.1 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 requirement
- 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 will have to be strengthened to provide additional support. The Backplate will also have to provide provisions to handle the to-be-specified shear loads (e.g. shear keys).

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

In order to provide limited protection from direct plasma ion impingement on the test blanket first wall, the Test Blanket shall 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 Properties
- Thermal Loads

1.5.7.2 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 ports. 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 and 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
- Accommodation of special end effectors
- Accommodation of special materials and coolants

1.5.7.3 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 accommodated 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

1.5.7.4 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

1.5.7.5 Vacuum Pumping System. The blanket system is partially contained within the primary vacuum boundary and affects 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

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

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

1.5.7.8 Building. The building space external to the cryostat and biological shield shall accommodate the Test Blanket System maintenance scheme. Space and support services shall be provided for operational support equipment near the horizontal test ports. Radial space must be provided to remove the modules from 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.

1.5.7.9 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.

Codes and standards requirements are included in Section 1.2.1.

Reliability requirements are included in Section 1.2.1

1.5.8. Other Special Requirements

1.5.8.1 Both the cooling and the breeder circuits are expected to contain tritium (from both breeding and permeation) as well as activation products. Purification (on-line or batch) is foreseen for both circuits. A suitable confinement of the ancillary circuits is therefore required to meet safety and maintenance requirements.

1.5.8.2 Guard heating of the complete Pb-17Li circuit to keep the liquid metal liquid. A moderate increase in melting point (due to Li depletion by Pb-17Li/water interaction) should be taken into account for the dimensioning.

1.5.8.3 The Pb-17Li and the coolant circuits shall be thermally insulated against their environment and comply with the requirements of the areas they penetrate (e.g. vacuum, impurities).

1.5.8.4 Tritium carrying fluids shall have a double confinement with leak detection from the TBM to the circuit caissons (TBD).

1.5.8.5 Other Requirements are TBD

References for Appendix A

[A-1] U.S. Contribution to Test Blanket Program, Draft ITER Test Blanket Program GDRD Test Blanket System DDD, U.S. Proposal on Solid Breeder Blanket Test Program, Test Program Proposal for U.S. Liquid Breeder (Li/V) and U.S. Proposal on Neutronics Test, UCLA-FNT-132, October 1995.

[A-2] European Helium Cooled Pebble Bed (HCPB) Test Blanket, ITER Design Description Document Status, 1.12.1998, Forschungszentrum Karlsruhe, FZKA 6127, 1999.

[A-3] European Water Cooled PbLi (WCLL) Test Blanket, 1997. Personal communication from Dr. Yves Poitevin of EFDA.

[A-4] Heat Loads and Operational Conditions of the Test Blanket Modules (TBM), by K. Ioki, TBWG meeting, 9-11 March 2004.