System Engineering and ITER Integration of the EU HCPB Test Blanket Module System

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Abstract

The Helium Coolant Pebble Bed Blanket (HCPB), a DEMO concept of a solid breeder blanket, has been selected for testing in ITER. The engineering design of the test systems is ongoing with the objective to design, construct, qualify and install this system in ITER for the first plasma operation. This paper presents an overview of the status of this work and in particular of the recent achievements. The design of the Test Blanket System is mostly concluded with the recent analysis of the FW performances; the work addresses now the fabrication technologies and the integration of the diagnostics. The conceptual design of the Helium Coolant System has been performed with the choice of the basic layout and of the main components. A concept of integration in ITER has been presented and now its feasibility is under investigation; a programme to assess and qualify remote handling tools and procedures has been started. Finally, safety studies are ongoing to accomplish the requirements for the licensing of the testing system.

1. Introduction

The Helium Coolant Pebble Bed Blanket (HCPB) is a concept of solid breeder blanket that is supported by the EU as candidate for the DEMO reactor [1]. This concept will be tested in ITER to gain information on in-rector behaviour under typical fusion conditions. The Test Blanket Module (TBM) will be located to occupy a half of a horizontal port of ITER; the TBM will be supported by several auxiliary systems dedicated to supply the coolant helium, to extract the tritium produced in the solid breeder beds, to purify the losses of tritium in the main coolant and to perform several measurement campaigns [2].

The HCPB TBM is assumed inserted in a frame-Port Plug at one of the equatorial port of ITER. As the frame will be shared with another TBM in an up-down configuration, the resulting dimensions for the HCPB TBM will be of 1208 mm x 710 mm x ~700 mm (width x height x depth). With these dimensions the test object in ITER will allow to reproduce a relevant portion of the DEMO blanket, keeping the typical dimensions of the breeder units (about 20 cm x 20 cm). In general the TBM is built with the same architecture philosophy of the corresponding DEMO blanket [3]; in particular the design of the first wall and of the breeder zone will use fabrication technologies that are in discussion for the DEMO blanket.

The Testing Programme foresees the testing of 4 TBMs during the first 10 years of ITER operation; dedicated experiments will be performed with the TBM system to study and compare its response in a relevant fusion environment with the analyses in the fields of neutronics, thermo-mechanics of the solid breeder and multiplier, and tritium handling.

In the last years the engineering design of this system is progressed; several papers can illustrate this development [2, 4]. This paper presents an overview of the status of this work with particular reference to the most recent achievements of this project.

2. Design of the HCPB TBM and performance Analysis

The present design of the HCPB TBM is illustrated in Figure 1. The TBM is constituted basically by the following subcomponents: first wall (FW), caps, stiffening grid, breeding units (BU), back plate/manifolds and attachment system. The FW is U shaped and has cooling channels in radial - toroidal direction; to keep in the testing the same mechanical features of the DEMO component i.e. the possibility to withstand a 8MPa over-pressurisation of the box, the dimensioning of the FW (thickness and channels dimension) is very close to that used in DEMO. The box is closed at the top and bottom with caps about 40 mm thick. For cooling they are equipped with cooling channels similar to the first wall; the cooling channels have different length in order to adjust the cooling capability to the reduction of the specific power released.

The stiffening grid allows the modular DEMO blanket to enforcing the box against an accidental internal over-pressurisation of 8 MPa; this feature is kept also in the TBM design. The stiffening grid plates necessitate to be actively cooled; coolant Helium flowing in internal channels will remove the heat generated in the steel plates self but also from the breeding

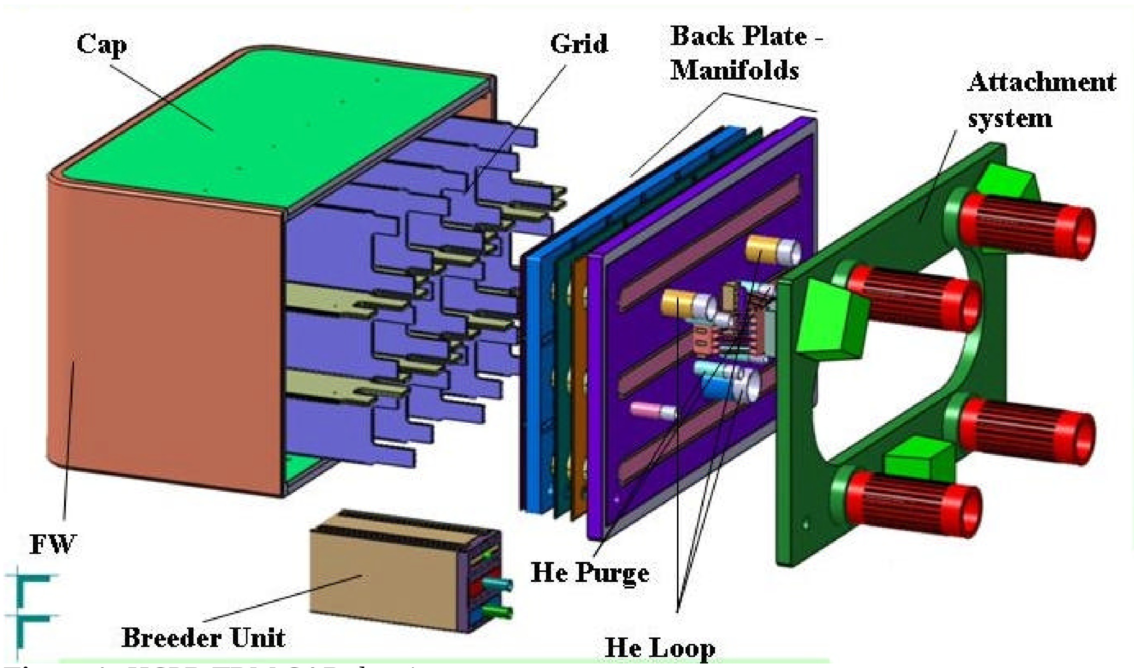

Figure 1: HCPB TBM CAD drawing

region. The breeding unit for the HCPB consists of breeding ceramic canisters each made by two cooling plates connected by a wrap and supported by an individual back plate. The canisters are filled with ceramic breeder (Li_4SiO_4 or Li_2TiO_3) pebbles of diameters in the range 0.2-0.6 mm and 0.8-1.0 mm, respectively. The space between the canister outside surface and the stiffening grid is filled with Beryllium pebbles having a diameter of about 1 mm; the estimated packing factor for the various beds will be in the order of 63%. Ceramic breeder will be use at a ^{6}Li enrichment varying from 40% (like for Li_4SiO_4 in DEMO) to 90% in order to increase the power density partially compensating the lower neutron wall load in ITER.

The back plate closes the TBM box from the rear side, provides the support for the mechanical attachment at the interface with the ITER Port Plug and forms a high pressure manifold system for the Helium feeding the different part of the TBM (FW, Caps, Grid and Breeding Units). The back plate is made by two thick plates (about 40 mm) connected by ribs. The place between these two plates is divided by thin plate and by the rib system in several chambers to accommodate the necessary manifolds. The TBM attachment, made from three shear keys and four flexible cartridges, is an adaptation from the ITER design. The shear keys have to cope with forces and torques in the toroidal - poloidal plane and the flexible cartridges

have to cope with radial forces and moments in the radial poloidal plane. The keys are arranged such that thermal differential expansion between the test blanket module and the supporting frame are possible around a central fix point. The flexible cartridges fasten the TBM and allow for thermal differential expansion.
The interface to the ITER is completed by the piping system; several lines will be necessary to feed the high pressure coolant (inlet, outlet and by-pass with internal diameter in the range of 60-90 mm) and to remove the tritium produced in the blanket (two 30-mm-pipes); additional small pipes (about 4 lines of 30 mm) will be necessary for special testing purposes (instrumentation, additional He lines, etc.). Electrical connection (grounding) and cable connections will complete the interface.
The engineering design of the TBM is almost completed; studies have been started for its optimisation according to the adopted fabrication technologies and the diagnostics systems that should be integrated in the design.
Recently the performance analyses of the system have addressed the thermo-hydraulic and structural design of the FW. The design of this sub-component is a challenge. It has to cope with a wide range of plasma heat fluxes (from 100 kW/m^2 to 500 kW/m^2) and the temperature limit of 550 °C for the material EUROFER. This requires very good heat transfer coefficients in the coolant channels. To achieve this, using Helium as the coolant, the first wall is 'U' shaped and has cooling channels in radial - toroidal direction. The channels are supplied from the rear part of the TBM and each channel has three toroidal passages. The shape of the channel cross section is iterated between analysis and design to fulfil the requirements within the material properties given. The three sweep design allows high velocity of Helium coolant in the channels (that means high heat transfer coefficient) adapting the DEMO geometry of the FW to the ITER conditions; the drawback is the increased friction losses.
The analysis of heat transfer based on three-dimensional simulations of fluid and heat flow in the first wall has been performed [5] demonstrating the capability of the selected design and helium parameters (8 MPa, 1.3 kg/s, 300°C) to fulfil the required control of temperatures in the first wall (under the design limit of 550°C). Computed pressure drop is 0.3 MPa, that is compatible to the overall design and performance of the Helium Coolant Systems. .
Structural analyses [5] of the first wall have been performed to assess the stress limit in the structure. A one fourth of TBM model has been used in the analyses; the model includes caps, stiffening grid, thick plates of coolant manifolds and the back plate of breeding units, while the breeding units itself and other internal structures have been omitted. Coolant channels in caps and stiffening grids have not been modelled, but their presence is taken into account through a reduced anisotropic stiffness of plates.
The stress analyses are performed for the following three scenarios: a) Normal with heat flux on plasma side of 270 kW/m2; b) Upset with heat flux on plasma side of 500 kW/m2 and c) Faulted with heat flux on plasma side of 500 kW/m2 and pressure of 8MPa in TBM box that occurs due to an internal leakage of the cooling channels. The assessment of the stress have be done according to the ITER Structural Design Criteria for in-vessel components using the European material database for EUROFER as "Appendix A: material design limit data". Case Normal and Upset are assessed according to Criteria Level A, whilst Faulted to D. The analyses have shown that these criteria are met for operational cyclic loads. The same is true for the accidental pressurisation of the TBM breeding zone with 8 MPa.

3. The Helium Coolant System (HCS)

The primary HCPB TBM Helium Cooling Loop [7] has to remove the heat which is generated in the TBM due to the plasma radiation (~1 MW for ITER operation at 500 MW fusion power). The loop has to provide a Helium mass flow up to 1.3 kg/s at the inlet of the TBM, a pressure of 8 MPa and an inlet temperature varying from 100°C to 300°C according to the requirement of different testing campaigns. The flow diagram of the HCS is presented in Figure 3. The selected lay-out of the loop has an "8"-shape design with a cold and an hot section separated by an economiser; this component divides the two part of the loop transferring heat from the hot helium coming from the test section to the helium coming from the circulator. This design allows the compressor to operate at low temperature. The cooler placed in front of the compressor controls the temperature of the He in order to maintain it

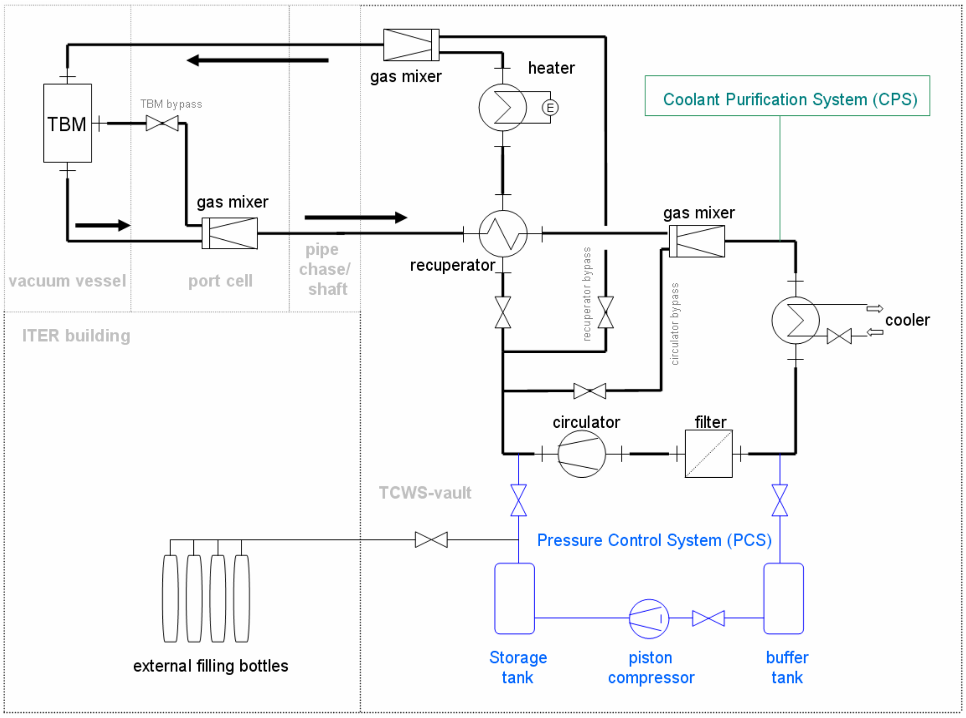

Figure 2: Simplified flow scheme of the Helium cooling loop for the HCPB- TBM

below 50?C which is the limit for the compressor inlet.

The secondary loop of the cooler is supported by water of the ITER Heat Rejection System, HRS (inlet temperature 35°C, pressure of 0.1MPa and mass flow in between 4.7 and 5.6 kg/s dependent on the heat load of the current campaign). In the hot section, an electric heater is used to increase the helium temperature coming from the economiser at the required TBM inlet value. A by-pass from the cold section around the economizer gives the possibility to feed the TBM with temperatures in the range of 100°C. Additionally the HCPB TBM cooling loop will be equipped with a Coolant Purification System (CPS) and a Pressure Control System (PCS).

Function of the CPS is to remove the tritium that can permeate in the coolant and other gaseous impurities (like O_2 and N_2), and control the gas composition in the HCS (partial pressure of H_2 and H_2O). The PCS serves to keep circuit pressure on the nominal value; it consists of a low pressure tank, a high pressure tank, a piston compressor, pipework and several control valves. Coolant is emitted from the circuit into the low pressure tank of the PCS as the circuit average temperature increases. The low pressure tank is connected to a piston compressor which charges the high pressure tank within the duration of the plasma transient that the Helium can be fed back into the circuit during the dwell time as soon as the circuit average temperature decreases again after the plasma pulse is finished. Both tanks have to be designed to cope with a maximum mass fluctuation of the coolant inventory of ~5 kg from plasma transient to hot standby during the most demanding campaign. The PCS has to

keep the pressure of the Helium loop within defined values of 7.9 – 8.05 MPa. A schematic drawing of the PCS is included in Figure 2.
The design of the primary Helium cooling loop in ITER is mainly based on the experience gained from HELOKA in terms of component selection and plant design. HELOKA [7] is an experimental facility currently developed at the FZK for testing of 1:1 TBM mock-ups under ITER relevant conditions. Figure 3 shows the layout of the HCPB TBM cooling loop indicating the components of the loop as well as their accommodation inside of the ITER building. Most of the components are accommodated inside of the TCWS vault. These components are connected to the loop components located in port cell 16 by pipes which are routed through the ITER building and the vertical shaft of the port cell.
The space availability in the ITER building is a critical issue for integration of the circuit, especially in the environment of the TCWS vault where most components of the HCS including PCS and CPS require accommodation. Investigation is ongoing for the design optimisation taking into account dimensions, thermal insulation, support and accessibility of components for maintenance or refurbishment, as well as presence of electrical cabinets, power and water supply. Several components of the HCS inside the vault require access during maintenance sequences especially the circulator, the dust filter and the electric heater which requires the possibility to replace faulty heater rods from the component. An other

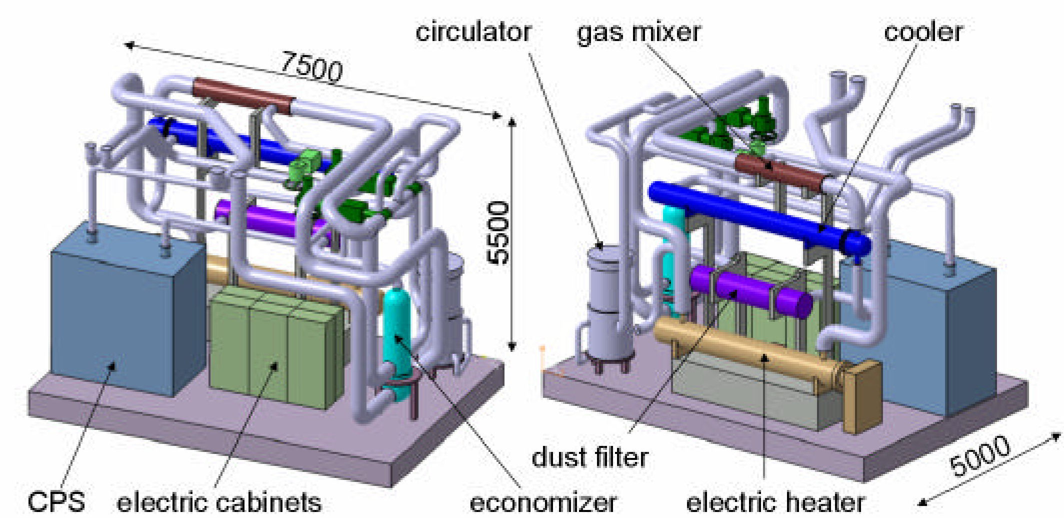

Figure 3: CAD- Model of HCS circuit components inside TCWS- vault (without PCS)

issue for the arrangement of the components is the interaction of the different Helium loops for other TBMs like for example in terms of a common Helium fill line and the requirement of a common system of spare components (like circulator or piston compressor) to ensure reliable operation of the cooling system even if one single component of one Helium loop fails. This topic requires further investigation.

4. Other Systems

Additional auxiliary systems are required to accomplish other basic functions of the TBM, in particular the extraction of the tritium that is produced inside the TBM. A Tritium Extraction System (TES) has to be designed in order to collect and to keep account of the produced tritium. This system should be designed for a peak T flow of about 1.3 µg/s and for a maximum operational time of 6 days at an assumed availability of about 20% (back-to-back pulse regime). The TES is allocated in the Tritium Building. To the tritium systems belongs

also the CPS (already mentioned in section 3). The conceptual design of these systems is under revision; it should be completed this year.
Several diagnostics systems should be allocated in the Port Cell area: in particular a system for neutronics measurements and one for tritium accountancy.

5. Integration in ITER

The installation of the TBM and its auxiliary systems in the ITER building requires a complex engineering of the involved systems, management of the several interfaces and the development of dedicated tools and procedures for the handling of theses systems [8]. Two areas are mainly involved in this integration process, the area around the Vacuum Vessel (VV) Horizontal Port and the ITER Hot Cell. The succession of different TBM campaigns

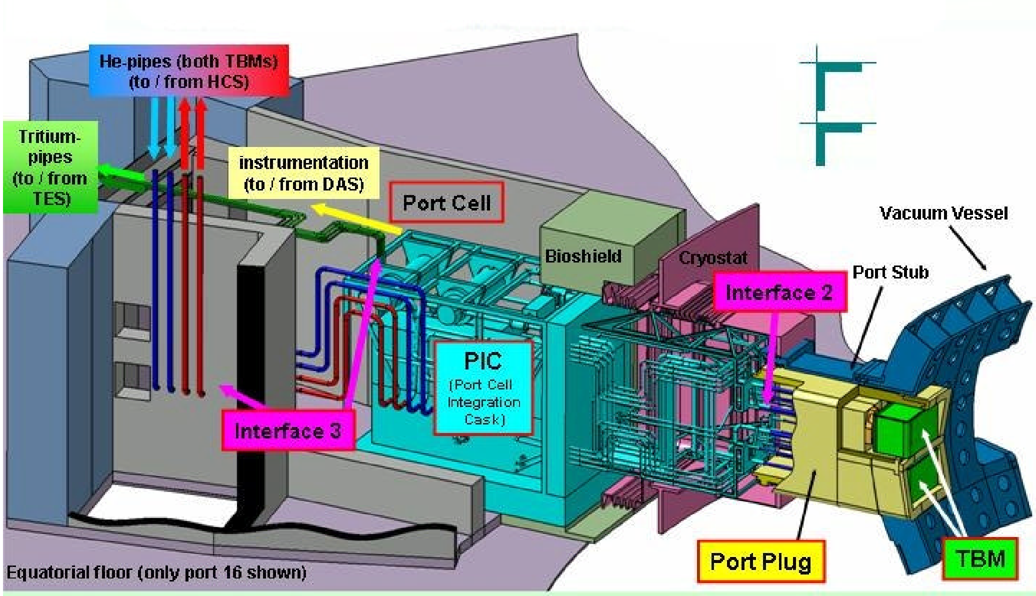

Figure 4: integration of the TBM Systems in the area around the TBM Horizontal Port (Port 16)

requires a TBM replacement strategy which allows a quick and simple TBM exchange standardised for all TBMs which are used subsequently according to the test schedule. In Figure 4 the area around the Horizontal Port of the ITER VV is illustrated. Two TBMs (the upper one is assumed to be the HCPB TBM) are allocated in the VV inside a structure, named TBM Port Plug that is cantilevered in the horizontal Port No.16. The connection is by means of a flange in the VV Extension that assures mechanical attachment and vacuum boundary between VV and Interspace/Port Cell. The replacement of the TBMs is made removing the Port Plug from the horizontal Port and transporting it to the Hot Cell. Successively, the refurbished Port Plug is mounted again in its position in the VV.
The different equipments present in the Port Cell and in the Inter-space region is arranged in an unique device (called PIC, Port Cell Integration Cask) that is supported by a transport vehicle like the standard cask used for RH operation at the VV Ports. This concept reduces the interfaces that should be operated during the TBM replacement to a minimum of two (interface 2 and 3 in Figure 3). The interface 2 should be handled with RH tools because the Inter-space region is exposed to too high radiation level for human access. This interface is constituted for the HCPB TBM by large (about 8 cm ID) helium pipes, purge lines for tritium removal, access lines for instrumentation and tests equipments. The necessary RH system is being developed in cooperation between the FZK and the KFKI RMKI in Hungary [9] in the frame of an EU task.

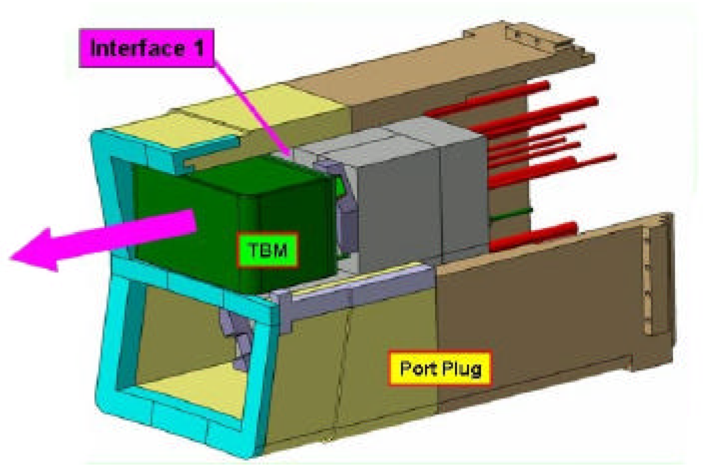

Figure 5: integration of the TBM Systems in the Port Plug; Hot Cell operations

For the operation in Hot Cell a specific RH-system is necessary for the refurbishment of the Port Plug, replacing the irradiated TBM with a new one for the new testing campaign (see Figure 5). Currently two different concepts of port plugs are under discussion [8]; one concept is based on in-board tools used to operate the interface inside the Port Plug, while the other foresees the removal of the TBM including part of the Port Plug shielding. The concept using a removable port plug shield gives more flexibility for operations during TBM exchange. Design work is ongoing to examine the feasibility of the two systems.

6. Safety analysis

Three groups of LOCAs have been studied judged to cover all accident scenarios envisaged in Cat II to IV events involving the TBMs. Some Cat.V scenarios have been evaluated to assess the ultimate safety margins of the TBMs. If the potential risks connected to the TBM itself are low due to the limited inventory of T and activation products, the possibility to jeopardise the ITER safety concept should be considered, especially regard to: i) the pressurisation of the VV, with failure of the containment; b) the H_2 production with the risk of a deflagration in the VV; c) the by-pass of the ITER first confinement with air ingress in the VV. The assessment [10] has addressed a number of concerns or issues that are directly caused by the TBM system failure, namely:

a) The VV pressurisation upon release of the total helium inventory from the TBM system is relatively small (0.026 MPa) in comparison to the failure limit of the VV (0.2 MPa). However the total inventory considered in this accident (34 kg) should be carefully considered and not further increased with the risk to endangers the pressure suppression function in the VV (45 kg of He is considered as maximum value);

b) In case of blow-down outside the vessel, the pressurisation of the TCWS vault has been calculated by 1500 Pa; this stays within design limits with large margin.

c) The pressurisation of the purge system (that is designed only for <0.3 MPa) is protected by design with a combination of insulating, check and pressure reduction valves; a failure of the protection system will cause a blow down of the Coolant Helium in the Tritium Building with the potential release of the T collected in the tritium systems (max 1g).

d) The temperature level reached in the TBM first wall during transients is dictated by the delay time needed to shutdown the plasma. In case of failure of the shut down system, the temperatures in the FW steel arise up to 920°C (with a peak at the plasma side of ~1280°C); for this temperature the integrity of the FW cannot be more accounted (see Figure 6).

e) The detailed heat transport model showed that passive decay heat removal from the TBM is assured in all cases. The effect of the additional chemical heat due to stem/Be and air/Be

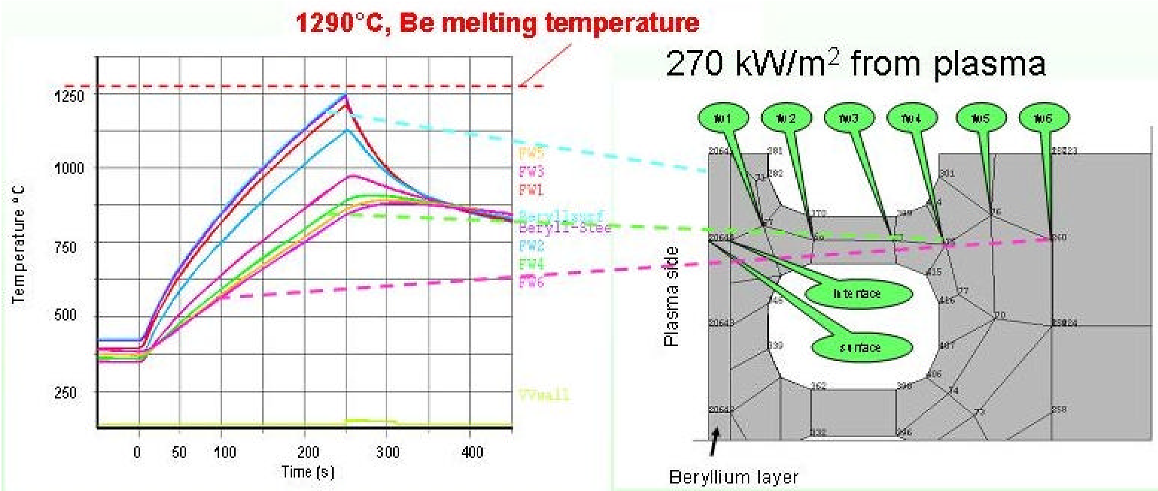

Figure 6: Temperature evolution in FW in case of delayed plasma shut-down.

reaction is considered in point g) and h), respectively.
f) The tritium release from the TBM system is inherently small. The most mobile fraction of the order 1 mg only is carried with the helium coolant. The tritium in ceramic is the second source that can be easily mobilized; its inventory is limited to 16 mg. The amount of tritium which could be liberated from the beryllium pebbles is bounded by the inventory to 58 mg, but its release is caused only by high temperatures exposure of the bed for long time tbd. Activation products in the helium cooling system are expected to be small. 1g T in the TES can be partially mobilised in case of accidental over-pressurisation of the system.
g) Hydrogen production has been calculated assuming an initial steam ingress rate of 18 mol/h into the Be pebble beds. The H_2 production rate is depending on the Be temperatures, and these have been calculated for the in vessel LOCA taking into account the energy released due to oxidation. The results show a production of H_2 of ~2500 g in the first four days of accident. The consequences of Be oxidation on the first wall is negligible due to the limited Be amount.
(h) Oxidation of beryllium beds with air has found to be uncritical, even if unlimited air access is assumed there is sufficient cooling for the beds.

7. Summary and conclusion

The engineering design of the HCPB TBM systems and its ancillary loops has been carried on in these last years and it is mostly concluded and the priority is now on the development and qualification of the fabrication technologies. From calculations point of view, the last modelling effort has addressed the thermal-hydraulic performances and the mechanical resistance in operational and accidental conditions of the first wall and on the manifold systems.
The conceptual design of the Helium Coolant System (HCS) has been completed with the chose of the loop lay-out and the analysis of its performances during operation.
Integration issues of the TBM System has been addressed with a first definition of systems, procedures and tools that have to be used in the Port Plug, Port Cell and ITER Hot Cell. In particular the arrangement of the components in the ITER building is critical; the design has to provide for the integration of several components, instrumentation, control systems in different locations (Vacuum Vessel, Port Cell, TCWS vault, Tritium Building, etc.) and to

cope to the remote handling requirements for the Port Plug and Hot Cell. Furthermore, all these systems require mechanical, hydraulic and electrical connections to the supporting structures, the water and power supply lines in ITER.
The development of diagnostics and measurement systems is a strategic task to gain the required information from the tested systems. Their design, the compatibility with ITER conditions (electromagnetic, neutronic, thermal compatibility), as well the integration in the TBM design are issues. Finally, safety and licensing analyses necessary are ongoing with the goal to include the TBM systems in the ITER preliminary safety report (RPRS).

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