TW1-TTBA-001-D03

Task Title: TBM ADAPTATION TO NEXT STEP MACHINE Adaptation of mechanical performances to ITER specifications

INTRODUCTION

The objective of the deliverable was initially to assess innovative design solutions aiming at reducing the amount of steel in the Water Cooled Lithium Lead (WCLL) Blanket and, by consequence, in the TBM in order to optimise their neutronic performance. This activity was then strongly linked to the shape and segmentation of WCLL Blankets. Considering that Power Plant Conceptual Studies launched in 2001 were questioning the WCLL banana–shape of modules, it appeared more appropriate to delay this activity until the achievement of the Power Plant Conceptual Studies (PPCS). This would have allowed to perform the analysis on the basis of an updated and relevant module segmentation.

However due to the re-organization of the liquid breeder concept towards He cooling, it has been agreed to perform this activity in the frame of the new Helium Cooled Lithium Lead (HCLL) project. In particular, a HCLL blanket for the DEMOnstration reactor has been assessed, coping with the task initial objective to optimise the amount of steel in the blanket.

2003 ACTIVITIES

2002 activities were focused on the mechanical optimization of the steel structure of a first version of the HCLL blanket concept (based on the Helium Cooled Pebble Bed DEMO 95 configuration). For completion of this activity, 2003 activities aimed at evaluating and optimizing the steel structure of an alternative HCLL blanket option.

Accurate drawings have been produced; thermal, thermo-mechanical analyses have been carried in order to evaluate the behaviour of the module principal components.

DESIGN DESCRIPTION

The generic blanket module consists of a steel box reinforced by vertical radial and toroidal Stiffening/Cooling Plates (S/CPs) forming rectangular ducts where the breeder lithium lead is flowing in. The box is closed in the top and in the bottom by cooled cover plates integrating the coolant and the lithium lead manifold system. Behind the BP are placed the main inlet/outlet headers for lithium lead and He. The steel box is constituted by a cooled plate bent to form the First Wall (FW, the region in the front of plasma) and the Side Walls (SWs). It is cooled by radial/toroidal/radial cooling channels having a rectangular cross section.

The S/CPs are 9 mm thick and are cooled by 4 mm side square channels having a 6 mm pitch. In order to withstand the pressurisation of the box in case of accident, the distance between two SPs should not be lower than 20 cm. Furthermore, in order to effectively evacuate the power deposited in the BZ (having an exponential distribution in the radial direction), the toroidal/poloidal S/CPs SPs are closer in the region nearest the plasma and sparser in the rear of the module.

A RAFM (Reduced Activation Ferritic Martensitic) steel, the Eurofer 97, is used as structural material, the eutectic lithium lead as breeder/neutron multiplier and high temperature He as coolant.

He outlet temperature has been fixed to 500°C, in order to achieve high thermodynamic efficiency, compatibly with limits on maximum steel temperature (~550°C). On the other hand, 300°C, has been fixed as inlet He temperature due to embrittlement of the eurofer when irradiated at lower temperature [1].

Only one He circuit is foreseen to cool both the FW and the BZ in order to limit the manifolds required space.

From a hemi-cylindrical collector placed behind the BP the cooling gas feeds the steel box channels and the bottom cover plate channels, then it is collected before feeding the SPs manifolds. From these manifolds, arranged on the bottom of the module, He is distributed in the SPs channels, in which it flows vertically towards the top manifolds symmetric the bottom ones and passes in the outlet collector.

The He flow distribution in the various SPs channels would need to be accurately assessed and special diaphragm systems could be necessary in order to homogenize the pressure drops and avoid preferential He pathway.

The liquid metal is fed from the rear at the top of the module; it passes in radial collectors placed under the cover, then it flows in the vertical rectangular ducts in counter current with the He. It is collected at the bottom in radial collectors and then leaves the module from the rear.

ANALYSES

A global thermal-hydraulic assessment has been performed to evaluate the He mass flow and the inlet/outlet He in the various module components (FW, cups, BZ). Finite Elements (FE) analyses have been carried out with the CASTEM code [2] in order to assess the thermo-mechanical behaviour of the blanket module in its current section.

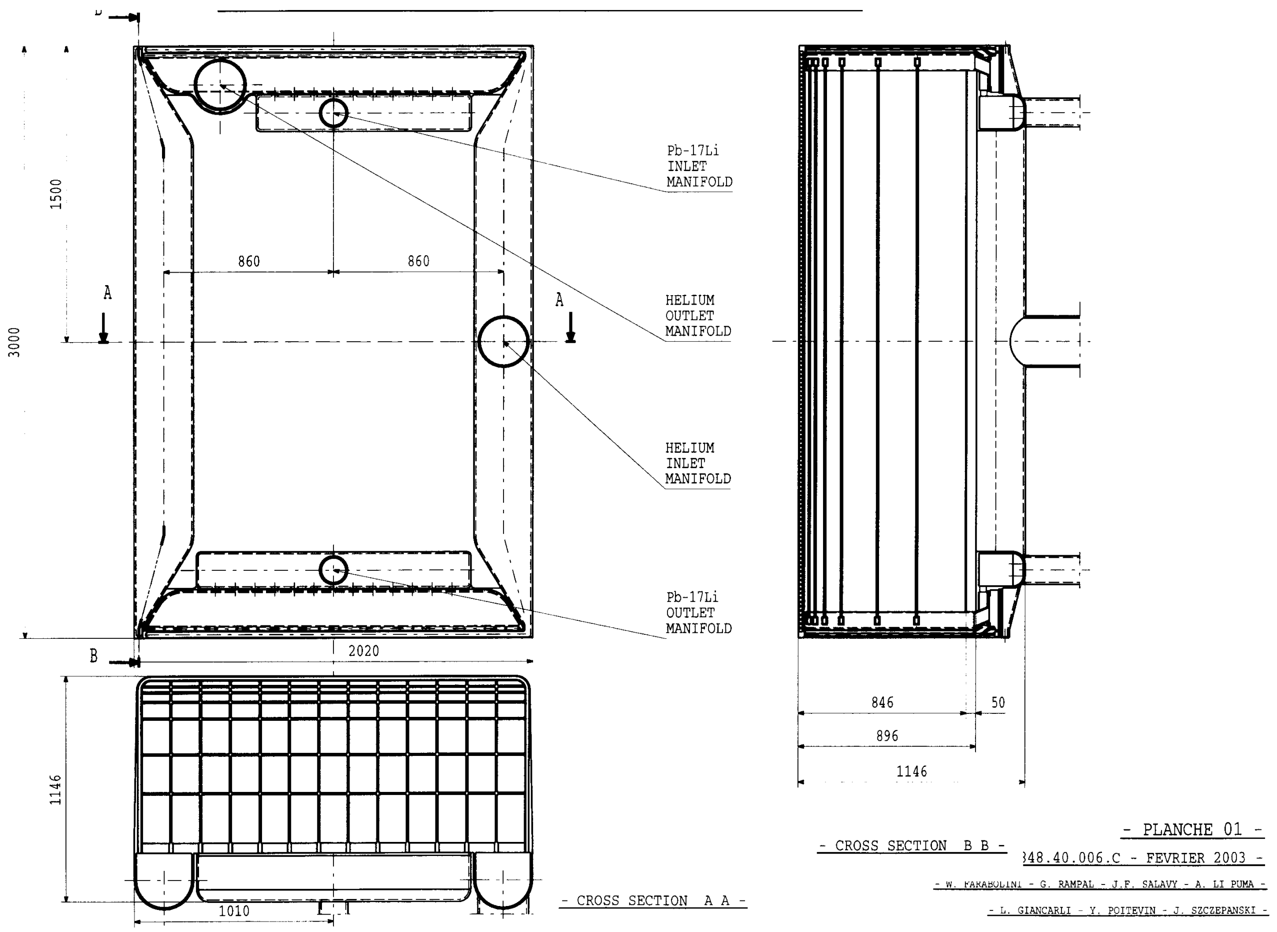

Figure 1

Furthermore, two FE models have been developed in order to investigate those aspects, which could represent a main concern.

THERMAL –HYDRAULIC ASSESSMENT

A module of $2 \times 3 \times 0.9$ (t × p ×r) m^3 has been considered for this assessment. Assuming a heat flux on the FW of 2.5 MWm^{-2} and a density power deposition from preliminary neutronic analyses, the total power on the module is 18 MW. Table 1 summarizes He parameters.

Table 1 : He thermal-hydraulic parameters

	v_{ave} (ms^{-1})	T_{in} (°C)	T_{out} (°C)
FW 133 chan. of 16 mm side	80	300	348
BZ 3156 chan. of 4 mm side	52	348	500

HORIZONTAL MIDDLE SLICE

A local FE model representative of the current horizontal section of the module has been developed. The model (shown in figure 2) represents a radial-toroidal slice of an half FW cooling channel poloidal height and comprised between two radial S/CPs. FW and CP channels are described with their suitable geometry.

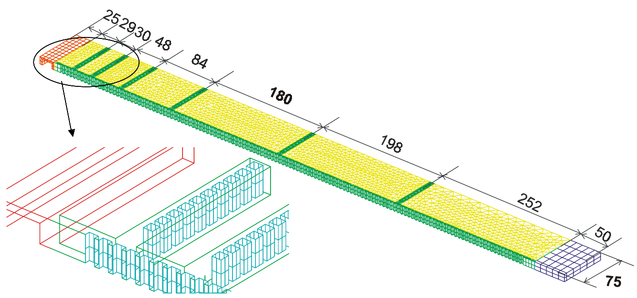

Figure 2 : Horizontal middle slice, FE model

Thermal-hydraulic boundary conditions from table 1 have been applied to evaluate thermal field at the top, in the middle and at the bottom of the module. Maximum, average and minimum temperatures in the various regions are summarized in the table 2 at the top and at the bottom of the module.

Table 2 : Min, ave and max temperatures in the various blanket regions

	T (°C) min/ave/max					
	He inlet (bottom)			He outlet (top)		
FW steel	369	420	502	373	428	502
S/CP steel	351	368	435	448	518	549
Interface	352		435	447		549
LiPb	352	437	633	447	576	751

The steel maximum temperature stays under 550°C, which has been fixed as upper limit for the Eurofer. However, the interface between the steel and the liquid metal reaches 549°C, well higher than the 480°C, which had been fixed as limit in the previous Water Cooled Lithium Lead concept. Due to high coolant exit temperature, this limit could not be respected in the HCLL concept, thus R&D activity has been is actually launched in order to assess the behaviour of this interface at temperatures up to 550°C.

Another concern could be the liquid metal average temperature at the exit of the module, which could make complex the design of the Li-Pb ancillary circuits.

3D POLOIDAL SLICE

The thermal analyses described in the previous section outlined the severe poloidal thermal gradient along the S/CPs, constrained at the top and at the bottom by thick covers. Global thermo-mechanical analyses of the box have been performed to verify the acceptability of the consequent thermal stresses.

At this scope, an FE model representative of the region between two toroidal CP along the entire poloidal length of the module has been developed. The channels in the cooling plates have not been represented, an equivalent thickness has been assumed for the plates.
Thermal fields obtained in the previous thermal analyses at different poloidal heights have been opportunely projected on the 3D model. Top and bottom covers temperatures have been fixed to 470°C and 530°C respectively, as evaluated from analyses on the collectors and covers (next section). The thermal field is displayed in the figure 3.

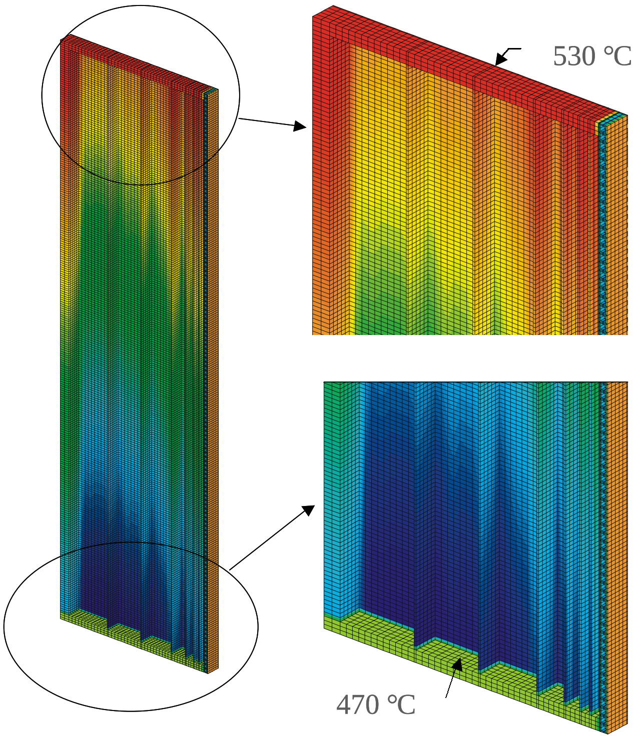

Figure 3 : Thermal field on the 3D poloidal slice, with the details of the top and bottom covers

Deformations due to this thermal load are displayed in figure 4 with an amplification factor of 500. Maximum obtained displacement are ~6 mm in the radial direction, 15 mm in the poloidal one and 0.4 mm in the toroidal one (which means ~10 mm for the whole module).

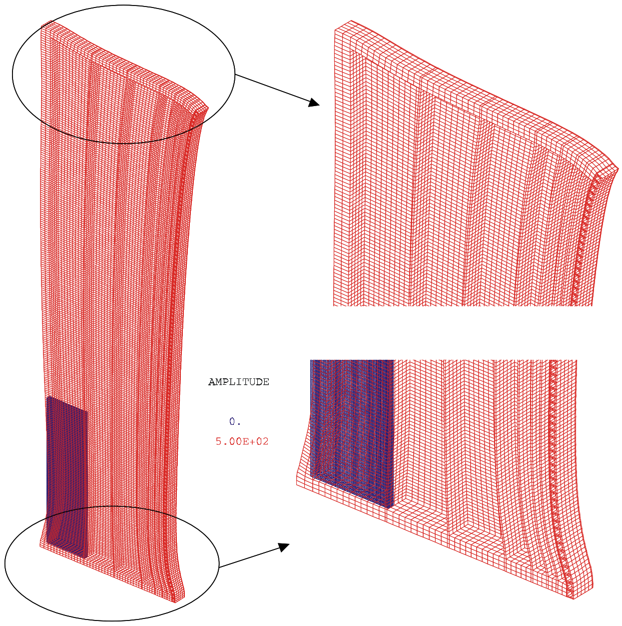

Figure 4 : 3D poloidal slice deformed model, amplification factor of 500

Maximum stresses are located at the bottom of the module. A part from a peak located in the radial SP, obtained Von Mises stresses stay under the $3S_m$ allowable value which limits the secondary stresses.

The peak is, however, very localised and located in a region in which the temperature field is not well described. It should not be, thus, a main concern.

COVERS AND He COLLECTORS

The thermal and thermo-mechanical response of the complex top and bottom collector regions has been assessed using the FE model shown in figure 5.

The model represents the region comprised between two parallel radial-poloidal S/CPs (one half for symmetry) and in particular, the radial slice between the first and the second toroidal-poloidal CPs (one half). It is this region, in fact, which is expected to be the more critical from the thermal point of view because of higher heat power density.

Two cases have been examined, representative of the top and bottom regions.

The assumed thermal-hydraulic boundary conditions for the two cases are summarized in table 3.

Since the channels in the toroidal-poloidal cooling plate have not been represented in the model, an equivalent convection coefficient has been assumed on its flat surface.

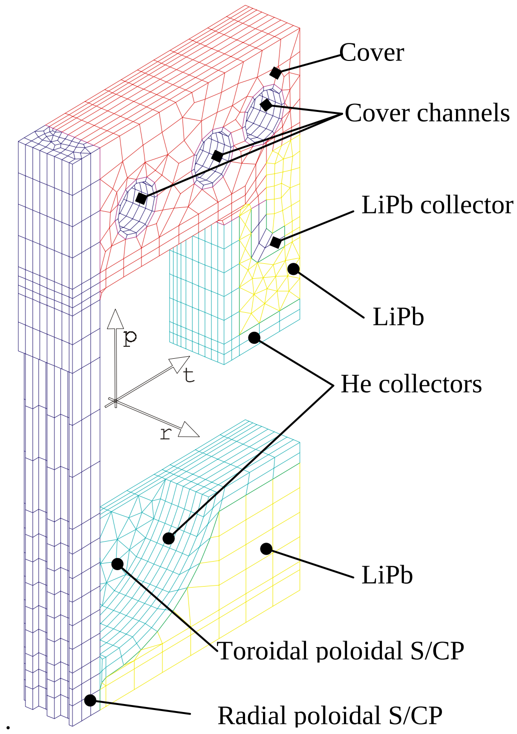

Figure 5 : FE model of the collector region

Table 3 : Thermal-hydraulic boundary conditions assumed for the collector region analysis (h in $Wm^{-2}°C^{-1}$, T in °C)

	v (ms^{-1})	Top		Bottom	
		T	h	T	h
Cover channels	10	350	1300	350	1300
Rad He collectors	50	350	3400	500	3000
Tor. He collectors	50	350	5200	500	4500
Rad/pol S/CP chan.	52	350	6020	500	5240
Tor/pol S/CP			5308		4620

Maximum steel temperature values, as expected, are located in the outlet collector region. The steel reaches 627°C in the LiPb collector and 590°C in the cover; the liquid metal reaches 634°C.

Performed analyses show that the collector region would need some modifications (optimization of the channels location in the cover, reduction of the LiPb collector thickness); no further work has been performed, however, in that sense because the alternative option based on radial orientation for the breeder zone has been preferred as reference solution for a DEMO blanket module [3].

Thermo-mechanical analyses have been also carried out on the same models, neglecting the LiPb. Thermal fields previously obtained have been applied as load to obtain secondary stresses. Primary stresses due to internal coolant pressure have furthermore been evaluated.

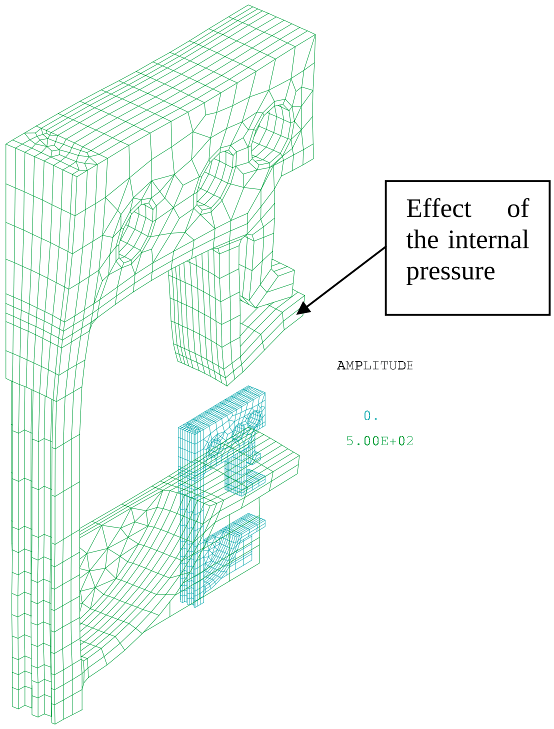

Figure 6 : Thermo-mechanical analyses of the outlet collector, deformed model amplified of a factor of 500

Figure 6 shows the deformed model (amplified of a factor of 500) of the outlet collector region when loaded with both thermal field and internal pressure. This region results the most critical due to the higher thermal dilatations.

The effect of the He internal pressure can be seen, in particular, on the toroidal collector.
Both primary and secondary stresses are everywhere under the assumed limits at corresponding temperatures.

CONCLUSION

In the frame of the EU Helium Cooled Lithium Lead project, the design of an alternative HCLL blanket concept for DEMOnstration is proposed coping with the task initial objective to optimise the amount of steel in the blanket.

Global thermal-hydraulic evaluations have been performed on the blanket module, together with FE thermal and thermo-mechanical analyses on models representatives of the various parts of the module.

In particular, the thermo-mechanical behaviour of the current section of the module has been assessed. A FE model describing an entire poloidal slice has been developed to assure that poloidal thermal gradient on the stiffening plates do not lead to excessive stresses. Finally, the collector region has been analysed.

Performed analyses showed that the proposed design is able to withstand the specification loads without exceed assumed limits in its current region.

However, some modification should be needed in the collector-cover regions, where the steel overreaches temperature limits. Further assessment in that sense has not been pursued because a blanket design based on radial orientation for the breeder zone has been preferred as reference solution for a DEMO blanket module, in order to ensure design convergence with the EU HCPB blanket concept.

REFERENCES

[1] F. Tavassoli, “DEMO Interim Structural Design Criteria, DISDC, Appendix A Material design limit data, A3.S18E Eurofer Steel, Intermi TTMS 5.2 & 5.5 Reports, CEA/ Report, CEA/DEN/SAC/DMN, D0-155-21/06/02, Revision 2, June 2003.

[2] CASTEM2000 finite element code, CEA Saclay, DEN/DM2S, 2002 version.

[3] A. Li Puma and al., Helium Cooled Lithium Lead blanket module for DEMO: designs and analyses, CEA Report, SERMA/LCA/RT/03-3297, September 2003.

REPORTS

Y. Poitevin et al., Helium Cooled Lithium Lead Blanket: preliminary study of an alternative design, CEA Report, DM2S/SERMA/LCA/RT/04-3390, March 2004.

TASK LEADER

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The inner plate contributes to guarantee the integrity of the module box in case of internal pressurisation (in faulted conditions). Inner and outer plates, furthermore, are stiffened by radial plates in order to withstand the He internal pressure (8 MPa in operating conditions).

All plates are drilled to allow the crossing of the feeding/manifolding pipes and of the stiffening plates as well. A feasibility fabrication study has been performed and a manufacturing sequence has been envisaged.

Although no killing points have been pointed out by this assessment, it appears that this manufacturing relies on assembling of several parts by welding, which could affect the reliability. Alternative fabrication techniques and assembling sequences should be investigated with the help of expert and industry.

He and LiPb Flow schemes

He at 300°C/500°C inlet/outlet temperature and 8 MPa pressure has been chosen as coolant. One circuit is used for cooling both the box and the breeder zone.

Being the FW the more solicited component both in terms of load and in terms of neutron irradiation, it has been chosen to be cooled with high velocity low temperature He.

The SPs are cooled in parallel with the FW and the covers, both in series with the BUs.

This allows to separate the SPs structural function from the CPs heat removal one, furthermore reducing the SPs steel temperature and then improving their mechanical performances.

The collecting of the He and its redistribution in the various components (FW, SPs, BU) are realized in the BP.

The liquid metal is fed from the back in the top of the module and recovered from the back, in the bottom.

It circulates radially-poloidally between the CPs of a unit cell and it passes from one cell to the one below through appropriate opening placed alternatively in the front and in the back of the horizontal grids.

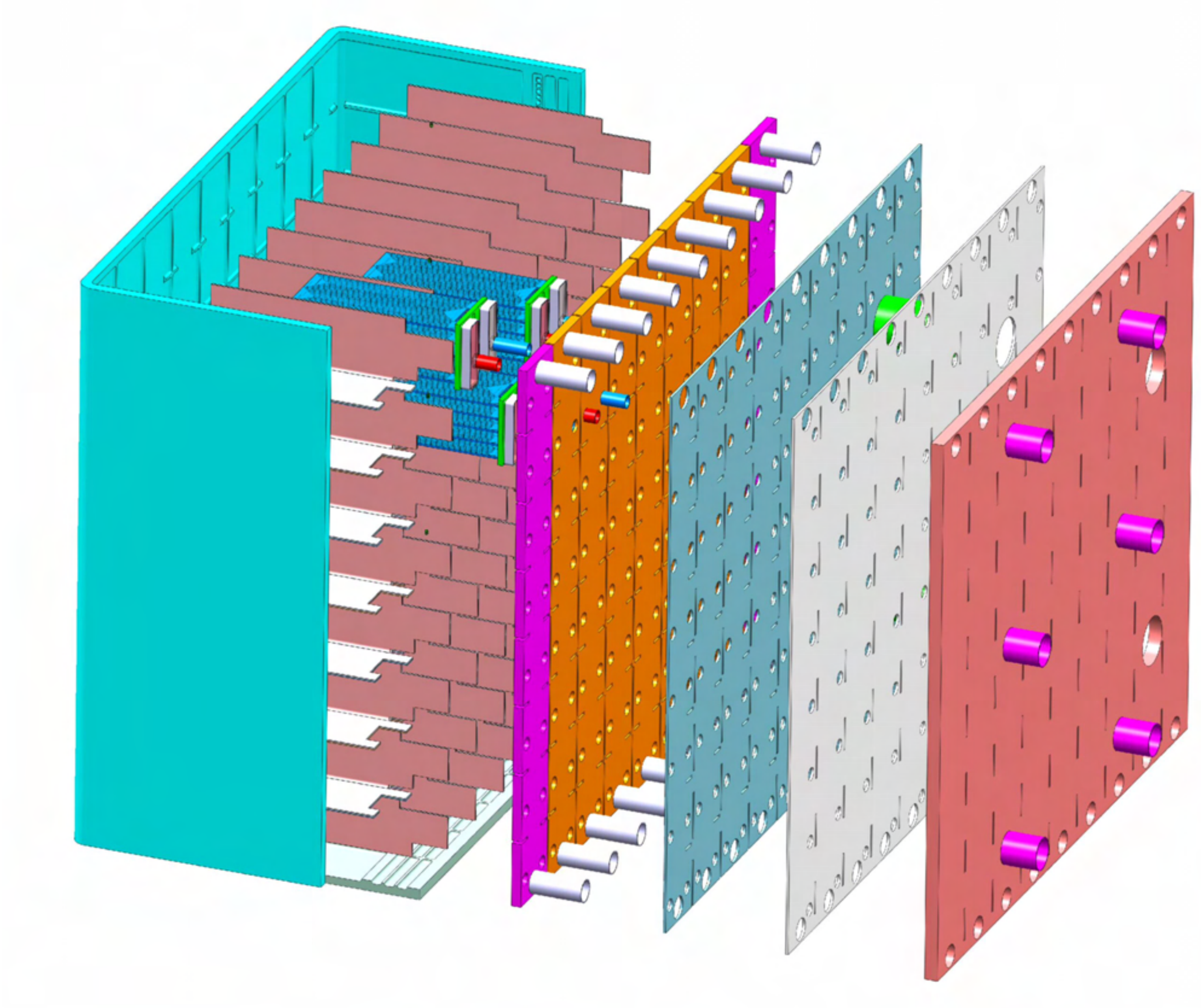

Figure 1 : Exploded view of the generic HCLL blanket module (from the rear) with external connexions

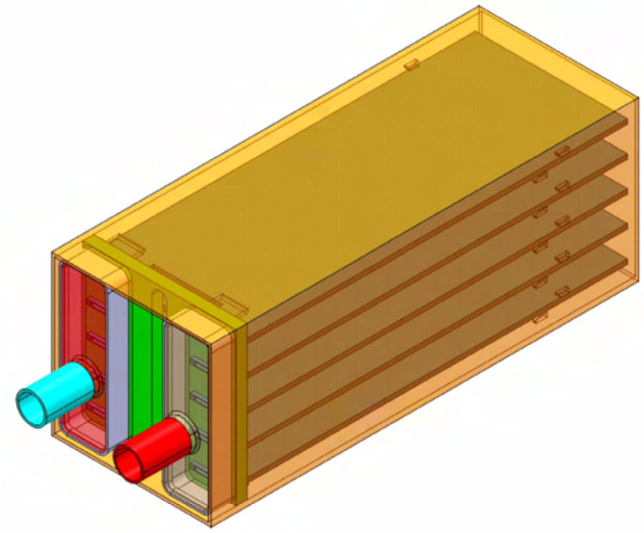

Figure 2 : 3D views of the generic BU (from the rear) inserted in the cell formed by the SPs (in transparency)

Analyses and results

The procedure and the criteria used for the analyses described thereafter relies on the ISDC [5, 6], DISDC [7] design rules.

Thermal hydraulic and thermal-mechanical calculations showed that the concept is able to recover the deposited heat with a quite good thermo-dynamic efficacy. Actually, although the temperature values reached in the FW steel are high (~560°C), the module is able to resist to the mechanical and thermo-mechanical loads in normal operating conditions for the foreseen life-time. A main concern could be the interface liquid metal/steel temperature, which reaches 544 °C, higher than the limit fixed for the WCLL blanket concept. However, an experimental campaign is ongoing to assess the behaviour of this interface until 550°C.

Mechanical analyses showed that, in case of a LOCA accident, the module could withstand the entire coolant pressure. Furthermore, the deformation should be compatible with the gaps foreseen between two adjacent modules, so guaranteeing the removal possibility. However, the return in service of the module after the accident could not be guaranteed.

The pressure drops in the module have been estimated at 0.34 MPa, leading to a pumping power of around 6 % of the gross electrical power. This value is acceptable if compared with the 10 % assumed as limit, however it does not take into account the pressure drops in the manifolding system, neither the one in the He ancillary circuit.

Testing goals, strategy and programme in ITER [8]

One of the key objectives of ITER is to provide the environment necessary for testing reactor-relevant breeding blanket concepts. This requires development of test modules based on DEMO blanket features and operating conditions, but modified to allow the achievement of the different specific test objectives in spite of the differences in operating conditions between DEMO and ITER (pulse, heat flux, neutron wall loading).

In the framework of the general objectives common to all breeding blanket to be tested in ITER, the main test objectives specific for the HCLL blanket program can be summarized as follows:

- Validation of the structural integrity of TBM's under integrated acting of thermal, mechanical, and electromagnetic loads. Estimation of the possibility to extrapolate experimental results to DEMO conditions.

- Study of PbLi flow and assessment of the MHD effects in a relevant magnetic field.

- Validation of accuracy to calculate tritium generation rate for general tokamak configuration and the DEMO HCLL blanket designs.

- Validation of thermal calculations results, and neutronic models, including nuclear libraries used in ITER and DEMO analyses especially for the prediction of tritium generation rates, nuclear heat deposition, neutron multiplication and shielding efficiency.

- Study of tritium recovery process efficiency (from PbLi and, whether required, from He coolant system), temperature dependence of residual tritium inventory in the blanket, and T-permeation towards the main He-coolant stream.

- Validation of irradiation effects studied in fission reactor spectrum with the aim to check out the impact of the neutron spectrum at least for low fluence irradiation of EUROFER structures, steel-steel welds and joints.

- Demonstration of HCLL DEMO blanket ability to generate high temperature heat suitable for electricity generation.

The HCLL blanket testing strategy in ITER is organized around the following lines:

- HCLL TBMs shall be inserted since day-1 of ITER.

- HCLL TBMs shall include relevant DEMO technologies, provided they have an impact on the concerned test.

- A progressive TBM qualification and testing program adapted to the different ITER operation phases.

- Four HCLL TBM types are envisaged today to cover the first 10 years.

In order to take the maximum benefit from all the ITER phases and related operating conditions it is foreseen to test 4 types of HCLL TBM.

The objectives and features of each one are defined in a way that they will provide a progressive qualification of the HCLL blanket line up to an integral demonstration in the final TBM (TBM-Integral):

- EM-TBM: Electromagnetic TBM.
- NT-TBM: Neutronic TBM.
- TT-TBM: Thermomechanics & Tritium Control TBM.
- IN-TBM: Integral TBM.

On the basis of these testing goals and strategy, a preliminary proposal of a testing program in ITER for the first 10 years of operations has been worded, taking into account ITER time-schedule. An overview of this programme is given in the table 3.

The design and testing programme of the first TBMs to be inserted in ITER is the objective of some activities launched within the 2004 technology programme, while this deliverable is more devoted to define the TBM design to be envisaged for a fully integrated test programme after few years of ITER operation and under D-T plasma. An outline of this TBM-Integral is described thereafter.

HCLL In-TBM for ITER

The Integral TBM should integrate all relevant DEMO technologies and features. It should be a prototypical portion of a HCLL DEMO blanket module (DEMO look-like), in the sense that it should represent, to the largest extent possible, all synergistic effects acting in a blanket module of a power reactor, like e.g., cooling the first wall and other structures, local heat generation and transport, and tritium generation and release.

The low neutron dose in ITER-FEAT requires to compromise, where necessary, the look-alike design in favour of the act-alike in order to achieve reactor relevant operating conditions understood mainly in terms of temperature level and temperature distribution prevailing in the heaviest loaded part of a projected power reactor blanket, i.e., in the outboard at reactor mid-plane in thermal equilibrium. This implies that coolant parameters like inlet/outlet temperature and pressure are within ranges suitable for an efficient power conversion. Neutron induced material degradation cannot be tested adequately in ITER.

TBM-In Design

The HCLL TBM will occupy a half of a test port. In principle, since the HCLL blanket is characterized by certain modularity, the orientation of its relevant TBM should not be a main concern. However, on the basis of some considerations on the MHD effects, a TBM having a vertical orientation is the preferred option.

The HCLL-TBM-In consists of a steel box of 1832 (poloidal) × 585 (radial) × 626 (toroidal) mm^3 overall dimensions. 24 unit cells take place inside the TBM having a poloidal-toroidal section of 214 × 180 mm^2, thus the SPs and the CPs channel cross section have been slightly modified compared to those of DEMO.

Stiffening beams are foreseen between the 1^{st} and the 4^{th} back plate to make the collector able to resist to the internal He pressure.

Analyses and results

Thermal-hydraulic evaluations and thermal FE analyses both stationary and transient have been performed to investigate the TBM-In behaviour under the ITER loading conditions. The response of the TBM (in terms of He mass flow and temperatures in the various circuits and of steel temperature) has been assessed both with regard to the design peak values (Heat Flux (HF) = 0.25 MWm^{-2} and Neutron Wall Loading (NWL) = 0.78 MWm^{-2}) and to typical values during the testing of the TBM (HF = 0.1 MWm^{-2} and NWL = 0.78 MWm^{-2}) [9].

The same He inlet/outlet temperatures as in DEMO have been assumed. Thermal-hydraulic showed that 4-pass FW channels are needed to guarantees suitable He velocity without changing the channels cross section.

Table 1 summarizes the coolant thermal-hydraulic parameters in the TBM, comparing them with the DEMO ones. Differently from the DEMO blanket module, in which the power is recovered approximately in the same measure (50 % and 50 %) in the passage through the FW//SPs and through the CPs, in the TBM the 75 % of the power is recovered by the He during its passage in the FW // SPs. This is because of the ratio between the HF and the NWL, which is higher than the DEMO one (0.25 / 0.78, instead of 0.5 / 2.4).

Table 2 compares significant stationary temperatures obtained in the TBM with those obtained for the DEMO blanket module. In the table, two sets of values are given for the CPs, the liquid metal and the interface, the first one relevant to the region between the horizontal SP and the adjacent CP, the second one relevant to the region between two CPs. With regard to the peak load conditions, results show the TBM representativeness in terms of thermal behaviour. Furthermore the interface LiPb / steel temperature, which is the key factor in the tritium permeation from the liquid metal towards the coolant, is comparable to the DEMO one, evincing that relevant information could be obtained on this phenomenon by the TBM testing. The liquid metal temperature reaches 550 °C in the region between two cooling plates, instead of 660 °C obtained for DEMO. However, it seems that this temperature, although could influence the diffusion and solubility of the T in the liquid metal, should not greatly affect the permeation towards the He. With regard to typical load conditions, while the FW maximum temperature decreases compared to the previous case, the maximum CPs and interface temperature are practically unchanged, because they depend essentially from the He max temperature and from the NWL.

Transient analyses carried out considering an ITER pulse with a duty cycle of 400 s / 1800 s showed that stationary conditions would be reached in the TBM front regions after some tens of seconds (60 in the FW).

Table 1 : DEMO and TBM HCLL : He thermal-hydraulic parameters

		FW			SPs			CPs		
	He mass flow	v	Tin	Tout	v	Tin	Tout	v	Tin	Tout
DEMO	~12	85	300	400	22	300	451	35	410	500
TBM Peak	0.9	25	300	451	7	300	447	10	450	500
TBM Typical	0.7	20	300	430	5.5	300	446	8	433	500

Table 2 : DEMO and TBM HCLL : temperatures in various regions

	Tmin / Tave / Tmax (°C)					
	FW	hSP	vSP	CP	LP	Interface
DEMO	420/475/563	443/467/515	453/472/537	415/464/512 459/495/544	418/472/535 416/505/660	418/ /515 416/ /544
TBM peak	475/507/567	450/466/511	450/474/507	450/476/515 452/485/532	450/474/519 452/494/560	450/ /515 452/ /532
TBM typical	452/478/513	444/463/506	442/465/507	438/469/509 438/479/527	439/469/512 439/488/527	439/ /509 439/ /527

Table 3 : Test description for the HCLL TBMs in ITER

#	Test Description	Test Requirement	Min. Duration
1	installation of TBM, leak tests, remote handling tests	Prior to day 1 Start operation: day 1	1 month
2	resistance against EM forces		Whole BPP
3	functionality, safety, thermal kinetics of ancillary circuits	vacuum in plasma chamber	3 months
4	heat extraction from FW, verification of H-H surface heat flux data	during plasma pulses	10 days (10 pulses)
5	MHD pressure drop as a function of LiPb flow-rate	during plasma pulses	4 weeks
6	H/D permeation from Pb-17Li into coolant, evaluation of D inventory	vacuum in plasma chamber	6 months
7	calibration of T source (n, ? flux) test of extractor w/ D	D-T plasma (no long pulses required)	10 pulses
8	deuterium permeation into FW coolant, verification of D-D surface heat flux data	D-D plasma	3 months
9	T inventory (experimental conditions TBD)	D-T plasma	3 months
10	temperature fields for code validation	D-T plasma	10 pulses
11	stress distribution for code validation	D-T plasma (pulse length > 100 s)	10 pulses
12	T permeation into coolant: • no LiPb circulation, no extraction	D-T plasma with nominal pulse length (400 s) and dwell time (1400 s)	30 consecut. pulses min.
13	• with LiPb circulation, no extraction	D-T plasma with nominal pulse length (400 s) and dwell time (1400 s)	3 months
14	• with LiPb circulation, with extraction	D-T plasma with nominal pulse length (400 s) and dwell time (1400 s)	6 months
15	Reproducibility of preceding tests, reliability	D-T plasma with nominal pulse length (400 s) and dwell time (1400 s)	as long as possible

TW2-TTBC-001-D01

Task Title: HELIUM COOLED LITHIUM LEAD TBM design, integration and analysis - Blanket system design and analysis - Integration and testing in ITER

INTRODUCTION

In 2002, under the constraint of reduction of financial commitments, EU has endorsed the decision [1] to concentrate the work on blanket modules for testing in ITER on a single coolant, Helium. Up to that time, two different coolants were envisaged for the EU Breeding Blankets: i) pressurized water for the Water Cooled Lithium Lead (WCLL) concept [2] and ii) pressurized He for the HCPB concept (Helium-Cooled pebble-Bed) [3]. In this frame, the general objective of the EU Task TW2-TTBC-001-D01 is to develop and optimize (with regard to tritium breeding, heat removal and shielding capability) a Helium Cooled Lithium Lead (HCLL) breeding blanket concept for DEMO and its corresponding Test Blanket Module (TBM) to be tested in ITER.

2003 ACTIVITIES

2003 activities have been devoted to complete the development of the HCLL blanket concept for DEMO and to define the outline design of a DEMO look-like test blanket module (TBM) for ITER.

An assessment has been furthermore performed in order to identify the testing goals, strategy and programme in ITER and preliminarily define the required number and types of mock-ups.

HCLL DEMO BLANKET MODULE

Design rationales and options

The HCLL blanket concept relies on the use of Reduced Activation Ferritic Martensitic (RAFM) steel, the Eurofer [4], as structural material, eutectic lithium lead as breeder material, neutron multiplier and tritium carrier and high temperature helium as coolant. The following main points have been taken into account to define the HCLL blanket module design:

- Similar manufacturing technology for HCLL/HCPB concepts.
- Modularity.
- Manifolds on the rear of the module (radial arrangement of the breeder unit cells).
- Blanket Module able to withstand the He pressure (8 MPa).
- Separation of stiffening and cooling functions (breeder cooling and box stiffening).
- Due to limited margin between the coolant temperature and the steel limit temperature, all steel structures have to be actively cooled.

Design Description

The generic DEMO blanket module consists of a steel box of 2 (poloidal) $\times$ 1 (radial) $\times$ 2 (toroidal) m^3 overall dimensions.

The box is formed by an U-shaped First Wall / Side Walls (FW/SW) cooled by He circulating in toroidal/radial/ toroidal channels and closed by two cooled caps (upper and bottom). It is closed, in the rear, by a Back Plate (BP) acting also as He collector and distribution system.

It is stiffened by poloidal-radial and toroidal radial stiffening plates (SPs) forming square radial cells of ~200 mm side. Each SP is cooled by He flowing in 4 channels forming three U-turns in a way that He enters and exits from the rear. He inlet and outlet legs are collected in two chambers opening on the relative stages of the BP collector. Vertical SPs are welded to the FW all along their poloidal length, whilst horizontal SPs are alternatively connected either to the FW or to the BP in order to allow the LiPb flowing between one cell and the one below. In each cell takes place a cooling Breeder Unit (BU) consisting of 5 parallel horizontal cooling plates (CPs), connected to a BU back plate ensuring the insert rigidity. The CPs are not welded to the SPs in order to avoid additional thermal stresses. In this way, the aim is hit that the stiffening of the box and the cooling of the Breeder Zone (BZ) are assured by two independent components. 8 square cross-section channels are machined in each CP to form, like in the stiffening ones, three U turns. In each CP, inlet legs of channels, and the outlet legs as well, are collected in two cooling chambers. Inlet and outlet chambers are then collected in two He unit collectors placed in the rear of the BU back plate and connected to the BP Collector through inlet and outlet pipes.

The BP collector region is made up of two thick outer plates (30 and 40 mm thick) having structural functions and two intermediate thinner plates (8 mm thick) for the flow separation.

CONCLUSION

This report describes the activities carried out in the EU Task TW2-TTBC-001-D01, in the frame of the European assessment on suitable breeding blankets for a DEMOnstration reactor and their corresponding Test Blanket Modules to be tested in the experimental reactor ITER.

The design of a HCLL blanket module for DEMO has been developed and optimized with regard to its multiple objectives. Thermal, thermal-hydraulic and mechanical calculations have been carried out showing that it is able to recover the deposited heat with a quite good thermo-dynamic efficiency without exceed thermal neither mechanical limits both in normal and accidental conditions.

Goals and strategy of the TBM testing in ITER have been defined and a test programme has been proposed according to the ITER time-schedule and operating conditions.

A DEMO-like HCLL TBM has been designed to be tested in ITER. The TBM has been investigated with regard to its thermal and thermo-mechanical behaviour, tritium production and permeation behaviour proving that, with some adaptations, it is able to achieve reactor relevant operating conditions for an extended period of time, in spite of the operating conditions of ITER which are quite different from those of DEMO.

REFERENCES

[1] EFDA Preparatory document for the STAC 2002, EFDA-STAC-T(02)-1/3.2.

[2] L. Giancarli, M. A. Fütterer, Water-Cooled Pb-17Li DEMO Blanket Line, Status Report on the Related EU Activities, CEA report DMT 95/505.

[3] Hermsmeyer et al., Improved Helium Cooled Pebble Bed Blanket, FZK Report, FZKA 6399, December 195.

[4] F. Tavassoli, DEMO Interim Structural Design Criteria, DISDC, Appendix A Material design limit data, A3.S18E Eurofer steel, Draft, Interim TTMS 5.2 & 5.5 Reports, CEA/DEN/SAC/DMN/ D0-155-21/06/02, Revision 2, June 2003.

[5] ITER Structural Design Criteria for in-vessel components (SDC-IC) G74 MA 8 01-05-28 W0.2.

[6] ITER Structural Design Criteria for in-vessel components (SDC-IC), Appendix B, guidelines for analysis, in-vessel components, G74 MA 8 01-05-28 W0.2.

[7] F. Tavassoli, DEMO Interim Structural Design Criteria, DISDC, Appendix A Material design limit data, A.GEN: General, Interim SM 5.1 Report, CEA/CEREM/DECM, September 1999.

[8] Y. Poitevin et al., ITER Test Blanket Working Group, Working Document for the Working-SubGroup #2 on LiPb-LAFS-He Blankets, Version 0, February 2004.

[9] Report from the TBWG for the Period of Extension of the EDA, April 2001.

REPORTS

A. Li Puma, Y. Poitevin et al. “Helium Cooled Lithium Lead blanket module for DEMO, Designs and analyses, CEA Report, DM2S/SERMA/LCA/RT/03-3297, September 2003.

A. Li Puma, Y. Poitevin, W. Farabolini, G. Rampal, J. Szczepanski, “Helium Cooled Lithium Lead Test Blanket Module for ITER, outline design and manufacturing sequence”, CEA Report, DM2S/SERMA/LCA/RT/03-3350, to be published (March, 2004).

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TW2-TTBC-002-D03

Task Title : BLANKET MANUFACTURING TECHNIQUES
Thermomechanical tests on HCLL blanket mock-ups

INTRODUCTION

The aim of this study is to validate the manufacturing of HCLL blanket mocks-up, made in the frame of the action TW2-TTBC-002-D01&D02, by means of thermomechanical loads, representative of blanket running conditions.

The principal program steps are:

- Design of the He cooling loop of DIADEMO.
- Design of the PbLi test section.
- Manufacturing.
- Thermomechanical tests.
- Endurance tests.

2003 ACTIVITIES

The conceptual design of the He loop have been carried out until the phase of tendering for manufacturing.

DESIGN DATA

The conceptual design of the He loop has been made taking into account that one cooling plate of the HCLL blanket concept was the DIADEMO test mock-up. The operating conditions were summarized in table 1

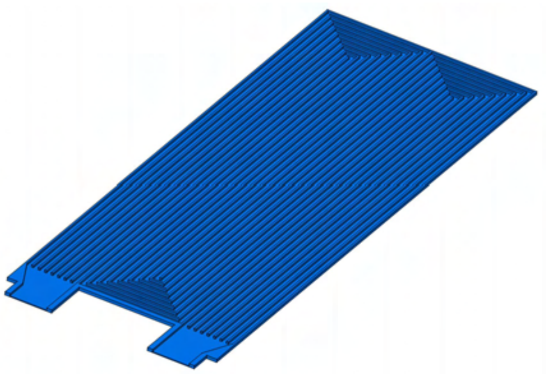

Figure 1 : ½ cooling plate

Table 1 : Operating conditions of the DIADEMO mock-up

Flow [$g.s^{-1}$]	Inlet/outlet Temperatures [°C]	Operating Pressure [bar]	Pressure loss [mbar]
30,4	410/500	80	840

FLOWSHEET OF HE LOOP

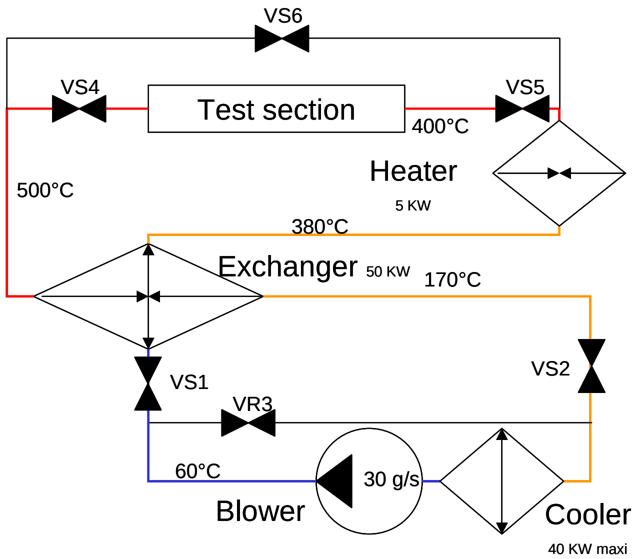

Figure 2 : DIADEMO HCLL loop

This loop is an isobaric device with an operating pressure of 80 bar (figure 2). A blower allows an He flow rate of 30 g/s, and the test section is fed by helium at a temperature between 300 and 400°C. After the test section the He temperature can be higher than 500°C. To optimize the power balance of the loop, an heat exchanger of 50 KW will be used. The inlet He temperature of the blower has to be lower than 60°C. So a cooler was designed to evacuate 40 kW maximum, by mean of the glycol/water loop of DIADEMO.

The principal characteristics of the different components are:

Heater: Its electrical power is about 5 kW. It is made with external heating elements implemented on the tube loop. The structure temperature of tube will be maintained lower than 550°C.

Exchanger: It is a counter flow exchanger. Its length is about 10 m, and it would be a bi-tube coil or like the cooler.

Cooler: It is a counter flow cooler, made by 2 concentric tubes (figure 3). In the internal tube flows the Helium and in the annular gap flows the glycol/water witch comes from the DIADEMO cooling loop. The thermal exchange power is limited at 40 kW.

Pump: It allows an He flow rate of 30 g/s (figure 4). The inlet He temperature has to be lower than 60°C.

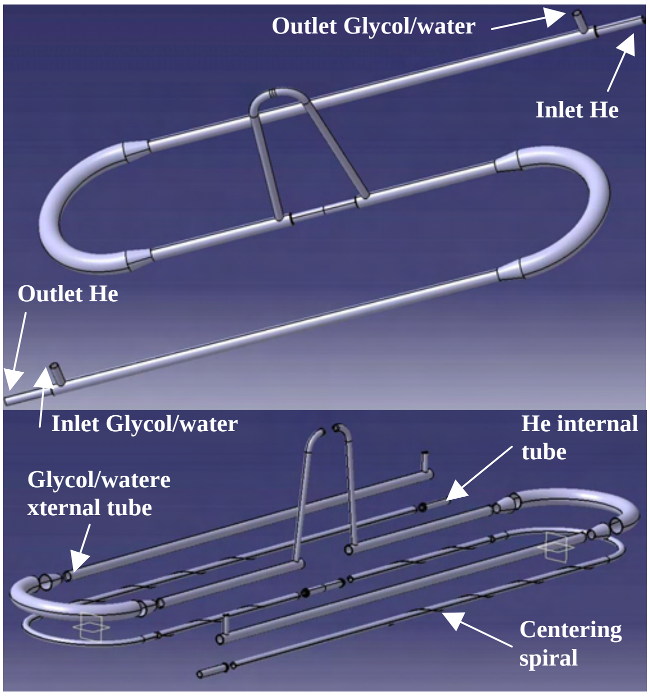

Figure 3 : Cooler design made on CATIA

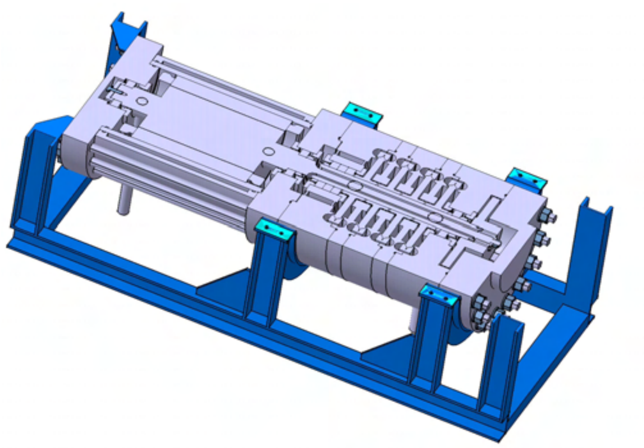

Figure 4 : Scheme of Pump

CONCLUSIONS

After, this conceptual design phase, a call for tender for "detailed design and manufacturing" will be launched at the beginning of 2004. Now we are waiting for a more precise definition of the "HCLL cooling plate" mock-up to be tested in DIADEMO HCLL to start the final design of the PbLi test section.

REPORTS AND PUBLICATIONS

[1] R. TALAND, L. CACHON, "APS de la boucle He de DIADEMO HCLL, NT DTN/STPA/LTCG 04-009.

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TW2-TTBC-005-D02

Task Title: HELIUM COOLED LITHIUM LEAD TBM system detailed safety and licensing

INTRODUCTION

Several concepts of tritium breeding blanket are envisaged for future D-T power plants, depending on type of breeder (lithium-lead or pebble bed) and type of coolant (water or helium or self-cooled). Based on these concepts, some Test Blanket Modules (TBM) will be integrated into ITER for validation, with purpose of transposition to future plants regarding: tritium breeding and recovery, energy multiplication and recovery, reduction of nuclear fluence in the structures of vacuum-vessel (V-V).

This task concerns the study of TBM based on Helium-Cooled Lithium-Lead (HCLL) concept. The general objective is to assess the impact of HCLL module with connected systems within ITER environment from safety viewpoint. The final aim is to prepare the licensing documents of HCLL-TBM.

The safety analysis of HCLL TBM during its life-phases (D02) is shared in subtasks as follows:

- Occupational radiation exposure in operating conditions (operation, maintenance, repair…).
- Wastes characterization and management.
- Radiological releases in normal conditions (effluent, tritium management).
- Identification of list of reference events and scenarios, and their classification.
- Safety classification of equipment.
- Analyses of accidental scenarios and their in-site, off-site consequences (radiological releases…).

The ongoing task D02 was planned for 2003-2004. However the three latter subtasks above have connection with a complementary task TW2-TTBC-005-D01 (TBM accidental safety study). In particular, D02 deliverable has to define a limited number of accidental scenarios as input data for detailed calculations to be performed in the frame of D01 task. So, among the D02 subtasks, those needed to initiate D01, were first considered in 2003.

However, conceptual and operational data of ITER HCLL version were not available in 2003. So the data of DEMO HCLL module were temporarily used when available second half of 2003. Additionally, conceptual design of the circuits connected at the back of HCLL module were presumed similar to that of the HCPB concept regarding the helium coolant circuit, and similar to that of the WCLL concept regarding the lithium-lead breeder circuit.

These loops and auxiliaries assure various functions required by the two separated fluids (inventory, pressure, detritiation, chemical composition, heat removal…).

The HCLL module consists of:

- a first wall (FW) which is internally cooled by a toroïdal He flow through single direction channels;
- horizontal and vertical stiffening plates (SPs) internally cooled by He through parallel double-U-shaped channels;
- radial cells delimited by SPs, containing Li-Pb flowing between cells, each breeder unit (BU) is equipped with parallel cooling plates (CPs) fixed to a unit backplate;
- three levels of He collectors at the rear of module: inlet-plenum of He spread out, intermediate-collector from FW and SPs outlet towards CPs inlet, outlet-collector from CPs return.

2003 ACTIVITIES

The first safety matters of HCLL concept and its integration into ITER were examined second-half of 2003 using temporarily the data of the DEMO HCLL version. A dedicated safety approach was built, starting from review of HCLL features concerning safety aspects, up to definition of lists of postulated events and scenarios retained as input data relevant for further safety analyses in 2004.

The main components of the dedicated safety approach built in 2003 are summarized here below.

MAIN LINES OF SAFETY APPROACH

The approach has to be suitable for safety analysis of integration of HCLL and auxiliaries into ITER plant in order to check the feasibility, i.e. HCLL integration should not significantly alter the reliability and safety of ITER. If not, a comeback to HCLL design and operation would be recommended (including connected systems). For this purpose, the first main lines of the approach were:

- Review and application of the existing ITER basic safety guidelines.
- Census of additional potential danger of ITER brought by HCLL integration and operation.
- Identification of involvements of the HCLL and connected systems regarding safety functions.

- Review of operating conditions of ITER including HCLL, in order to change the reference list of events according to a suitable method.

Safety influences of ITER/HCLL coupling were investigated following complementary orientations:

- Impact of initiating HCLL faults on itself (HCLL → HCLL).

- Impact of HCLL faults (either initiating or induced) on ITER (HCLL → ITER).

- Impact of initiating ITER faults on HCLL (ITER → HCLL).

- Impact of initiating ITER faults on itself, but involving a HCLL fault as aggravating (ITER + HCLL → ITER).

Studies following the former aim at preventing HCLL structural failure with possible outside consequences (2nd step). While the third step aim at identifying possible HCLL failure induced by ITER conditions that could act as aggravating in the latter step.

ADDITIONAL RISK TO BE MINIMIZED

For ITER without TBM, inventories of source terms and energies are known as well as the types of reference events leading to releases and the margins against the release guidelines. When introducing HCLL system, the level of ITER risk should be kept quite similar. So, several aspects of HCLL integration were considered:

- HCLL inventories of source terms and of potential energies.

- Modes of mobilization of source term and transport ways.

- Possible accidental scenarios that could lead to potential release larger than from reference ITER events.

Compared with ITER inventory, HCLL tritium production is low and probably shared between Li-Pb and He circuits. Moreover an efficient detritiation process is expected. So, HCLL tritium production is more a matter for occupational exposure than for offsite release. Nevertheless, the loops connected to the HCLL module represent potential ways of source transport for example via the He/water heat exchanger. Any possible bypass will be prevented by implementation of secondary barriers.

Otherwise the potential energies, associated to leaking He and/or Li-Pb, are considered following two aspects: first as initiator of barrier failure, secondly as mode of source transport. Compared with other TBM concepts, use of He coolant is limited to pressurization concerns, and Li-Pb use involves heat transfer and Li chemical reaction (with possible H_2 production) but the kinetics is expected to be slow for Li-Pb with water or air. Moreover such contacts presume double barrier failures. Lastly, no beryllium layer is foreseen at the inner side of FW of HCLL module, thereby hydrogen production by in-vessel water leak is rejected.

SAFETY FUNCTIONS AND DESIGN REQUIREMENTS

In the generic site safety report of ITER (GSSR) it is pointed out that safety functions will not be allocated to experimental components. While in the "test blanket" section, some TBM components (ex-vessel part, test blanket cells) are ranked in Safety Importance Class (SIC) as parts of confinement barriers. These assertions seem to conflict together unless an approach is built so as to render them consistent. An attempt to define such approach is proposed. First, some HCLL parts will be necessarily involved in safety functions because:

- A self-source term is produced inside the HCLL.

- Barriers are required either against self-source term or to prevent chemical interactions and bypass regarding mobilization of ITER-source term.

- HCLL cooling system is required to prevent barrier failures.

In addition to the TBM parts already contributing to confinement function (GSSR indication), some HCLL components assuring He mass flow (pressurization and volume flow) should be considered as safety components due to their role in accidental scenarios. It is then suggested that only TBM parts contributing directly for main safety functions will be in class 1 of SIC; other parts having safety support-functions will be in class 2.

Moreover, a progressive validation procedure is recommended. It means some HCLL parts contributing to safety functions should be experimentally validated (out- and in-ITER), prior to add other parts relating to HCLL purposes (energy multiplication, tritium breeding) but that could add up risks. Some concerns about cumulative loading during lifetime of structures and about radiation effects, to be taken into account in analyses of accidental behaviour, remains as open question but are at short time covered by pessimistic method of selection and classification of postulated events.

LIST OF POSTULATED EVENTS: METHOD AND LIST

Since the HCLL systems and their integration within ITER are not detailed yet, it's not worthwhile at short term to apply detailed method (e.g. FMEA) for identifying a list of events. So a simplified method is proposed based on: analysis per function and search of progressiveness in failure modes, in accordance with the simple but conservative Lines-of-Defense (LoD) method already used officially for some fission reactors. From application of the proposed approach, two complementary lists of events were set:

- List of specific HCLL events, some of which having only internal impact.

- Among the official list of ITER events (i.e. without TBM), those modified by HCLL integration.

The former is resumed in enclosed Table. Among the postulated events to be confirmed by further study, some ones were selected for first studies on HCLL structural behavior or on HCLL impact within ITER. Scenarios with loss of He mass flow (channels or global) seem to be of first concerns (poor He thermal inertia) regarding HCLL module behavior, to be studied in the frame of the complementary task D01.

Table : List of postulated HCLL events
(AOC: abnormal operating conditions, LOF: loss of flow, LOHS: loss of heat sink, LOSP: loss of supply power, LOC: loss of coolant or breeder)

Internal faults:	
- AOC	Loss of He pressure
	He overpressure
	He inventory change
	Fault of He purification system
	Li-Pb inventory change
	Fault of Li-Pb purification system
	Li-Pb flow change
	Heating circuit fault (Li-Pb freezing)
- LOF	He circulator trip
	He circulator seizure
	Li-Pb pump seizure
	Partial blockage of CP channel(s)
	Partial blockage of SP channel(s)
	Partial blockage of FW channel(s)
- LOHS	LOF of secondary circuit
	Loss of final heat sink (1 or 2)
- LOSP	Loss of one supply file
	Total LOSP
- LOC	Leak of cooling plate(s)
	Leak of stiffening plate(s)
	In-cells LOC of FW
	Rupture of HX tube(s) (1 or 2)
Faults impacting the 1st confinement:	
- LOC	In-vessel He leakage
	In-vessel Li-Pb leakage
	Small FW break (He+Li-Pb)
	Large FW break (He+Li-Pb)
Faults impacting the 2nd confinement:	
- LOC	In-cryostat Li-Pb leakage
	In cryostat He leakage
	In-cryostat (He+Li-Pb) leakage
	Ex-cryostat He leakage
	Leak of Li-Pb detritiation system
	Leak of Li-Pb purification system
	Leak of He detritiation system
	Leak of He purification system
	Leak of secondary coolant
	Leak of primary + secondary fluids

CONCLUSIONS

A first review of the TBM based on HCLL concept was done regarding the safety aspects. Besides existing ITER basic safety guidelines, complementary safety approach was proposed in order to initiate safety analyses of HCLL specificities and of ITER/HCLL coupling. A list of postulated events was established. Among the accidental scenarios to be studied at first, those involving He cooling faults are retained as relevant for HCLL design confirmation.

REPORTS AND PUBLICATIONS

P. Lo Pinto, "Safety analysis of the Helium-Cooled Lithium-Lead Test Blanket Module for ITER (HCLL-TBM)", CEA report DER/SESI/LESA NT DO 4.

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Task Title : HELIUM COMPONENTS TECHNOLOGY Available technology and proposition of tests

INTRODUCTION

In a future electricity generating fusion reactors, the blankets (BKTs) as well as the divertor (DV) could be cooled by gaseous helium.

In ITER BKTs and the DV will be cooled by water. Nevertheless, Test Blanket Modules (TBMs), implanted for experimental purposes, will be cooled by helium.

The temperature and pressure conditions are respectively in the ranges 300°C to 740°C, and 8 MPa to 10 MPa, quite equivalent to those found in Gas Cooled fission Reactors (GCRs).

The technological problems of GCRs He circuits had been listed in 2002, as well as solutions. A new approach, more specific to fusion reactors is done in 2003. Experiments, to get answers to some of them are prepared.

2003 ACTIVITIES

A short description of fusion reactor Blankets (BKTs) and Divertor (DV) helium cooling circuits has been done for Helium Cooled Lithium Lead (HCLL) reactors, by analogy with a former study on Helium Cooled Pebble Bed (HCPB) reactors, by S. Hermsmeyer (Karlsruhe) and G. Vieider (Studsvik).

The helium temperatures range from 300°C to 740°C, the highest value being at the outlet of the DV. The pressure is about 8.0 MPa in BKT circuits, and 10 MPa in DV circuits. These conditions are only slightly lower than in fission Gas Cooled Reactors (GCRs).

Technology problems relevant to fusion reactor Blankets (BKTs) and Divertor (DV) helium cooling circuits have been listed, component by component. The technological problems are listed in the two following tables.

Table 1 : Technological problems of DV and BKTs helium cooling circuits

Component	Technological problem
Steam generator	- High unit power
Inner Thermal Barriers (ITB), in DV cooling circuit hot gas pipes (740°C)	The following must be done for possible solutions : - Measure conductivity of ITB in He - Test mechanical behaviour with respect to : - differential expansions between ITB and pipe - vibrations (structural, flow induced, acoustic) - fast depressurisation - Test tribological behaviour (sliding, fretting ; 740°C) - Test behaviour with respect to erosion by high velocity He - Measure production of polluting chemical elements - Test resistance to chemical environment
Isolating components (e.g. valves) (300°C; 500°C, and possibly 740°C)	- Check the availability of large valves below 600°C (medium size are available) - Take part in developments by industrialists for use between 600°C and 740°C in He (easier than developments being done for 1000°C) - Test tribological behaviour (sliding, fretting ; 500°C ; 740°C ?) - Measure tightness between the upstream and the down stream - Measure tightness between the inside and the outside
Regulating components (300°C, 500°C)	- Check the availability of large valves below 500°C (medium size are available)
Circulator (compressor at 300°C, 500°C)	For high power circulators of PPCS, or more limited power circulator for ITER-TBM : - Make technical choices : - type of compressor - mobile element guiding & support - technology of tightness with outside environment (mainly around shaft) - mobile element guiding and support tests - tightness around shaft with outside environment - Test tribological behaviour of main bearings or catcher bearings (<300°C, <500°C) : - sliding - bearing - fretting

Filter on DV circuit	- Study & test great surface filters
Detection of impurities in He	- Make sure that specimen extracted and cooled for analysis are representative of concentrations in hottest places
Control of helium quality	- Specify concentration ranges to avoid - Chemical interactions leading to a change in the material properties - Production of radioactive isotopes, by activation - Demonstrate purification - injection systems
He leak detection and measurement	- Find needed sensitivities and time of response - Test existing techniques in representative conditions (Pressure drop, katharometer, mass spectrometer, and others)
Thermodynamic instrumentation	- Find proper means for introducing the thermocouples (TCs) through ITBs of DV hot duct - Assess the temperature error introduced by calorific irradiation of TCs, in permanent and transient conditions, and if necessary design shields
Any component with static seals	- Test seal leak tightness, in isothermal conditions - Test seal leak tightness with temperature gradients - Test tribological behaviour due to thermal expansion (<500°C, <740°C) : - sliding - fretting
Long term availability of He	- Helium is produced mainly from natural gas, which is being depleted. Find ways for extracting He from air at economically acceptable conditions.

Some of these problems are being adressed in Cadarache:

- tribology of sliding parts (tested on the He-TRIBOMETER). The difficulty comes from the small quantity of oxygen in pure helium, which increases friction efforts,
- leaktightness between static parts (tests on the HETIQ bench),
- behaviour of inner thermal barriers in static helium (tests on the HETHIMO bench).

In year 2003, the construction of the TRIBOMETER was finished, allowing experiments since the beginning of 2004. The HETIQ and HETIMO benches were designed, and they will be available by mid-2004. Provisional pluriannual test programs were written for each of the three addressed problems. The tribology program will include configurations in the 500°C – 800°C range (and up to 1000°C, for GCR's), with contact pressures between 2 to 20 MPa. All substrates are Haynes 230. The behaviour of the substrate without any coating will be tested up to 500°C. For temperatures up to 800°C, the coating by 75%Cr_3C_2 – 25%NiCr will be tested. For temperatures up to 100°C, the Ni-16%wtCr-6%wtAl0,5%wtY coating will be tested.

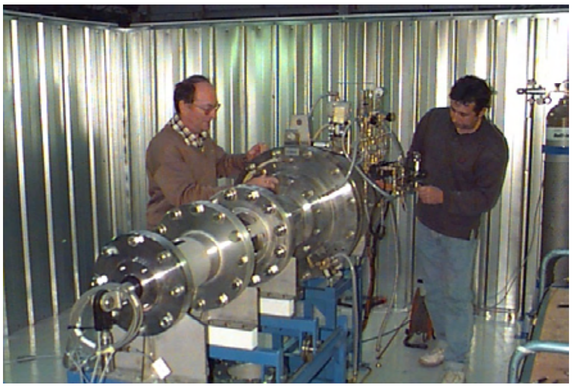

Figure 1

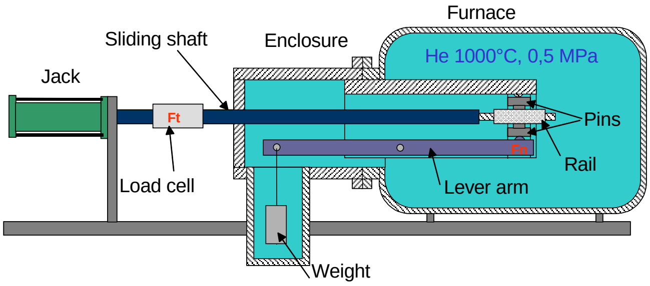

Figure 2 : TRIBOMETER

The leaktightness program covers three temperatures : 25°C, 500°C, 1000°C. The helium pressure will be up to 10 Mpa. The first tests will be at 500°C, 10 MPa, with a HELICOFLEX HN200 seal, or a "S" seal of the French SPG company.

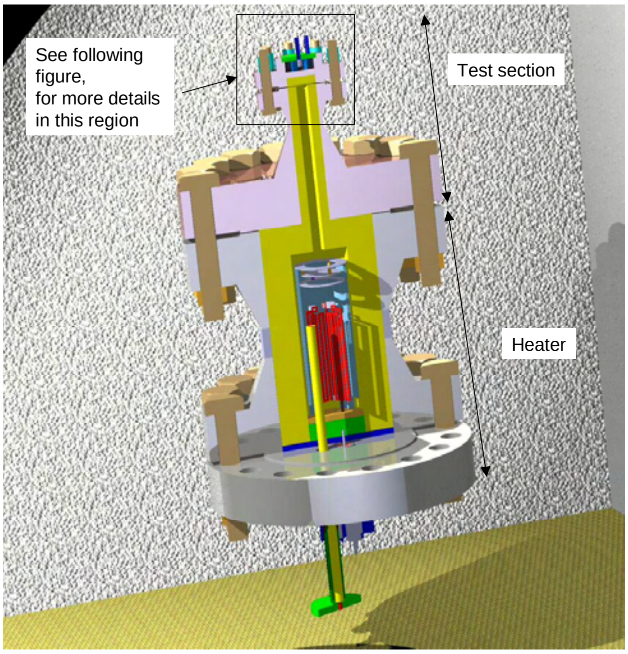

Figure 3 : HETIQ test section (1/2)

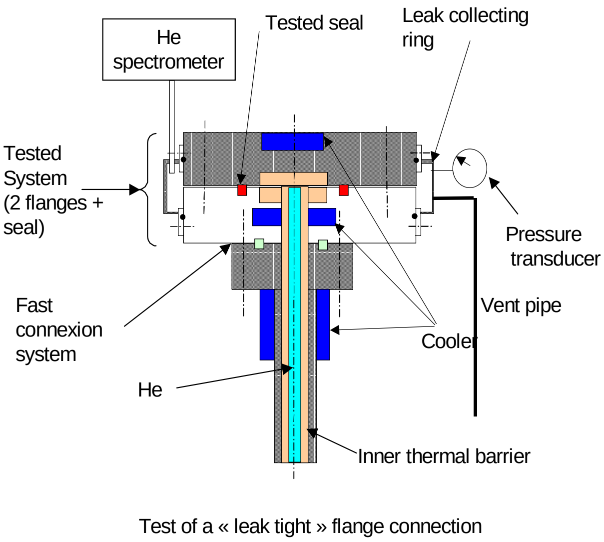

Figure 4 : HETIQ test section (2/2)

The inner thermal barriers program will allow tests in static conditions, thermal cycling, and pressure cycling (partial pressure losses at 2 MPa.s^{-1}). The objectives are to :

- Measure the conductivity of the barrier in helium.
- Test the mechanical behaviour, with respect mainly to differential expansion between ITB and pipe, and fast depressurisation.
- Test tribological behaviour (sliding, fretting).

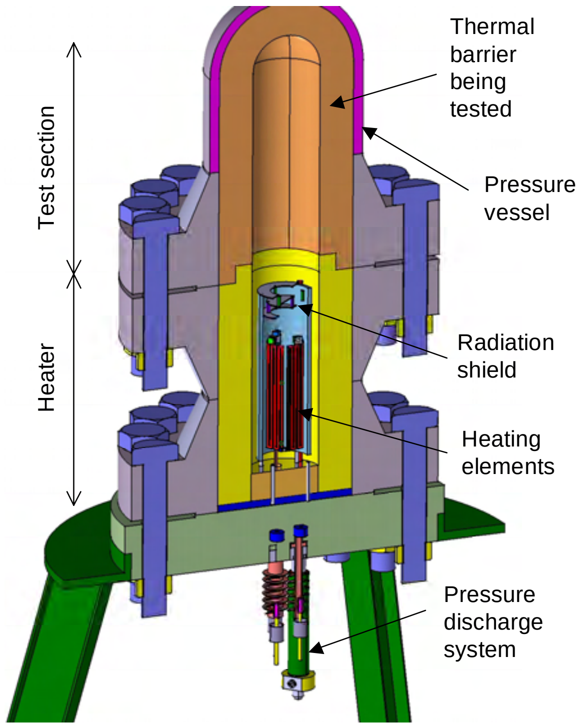

Figure 5 : HETHIMO test section

CONCLUSIONS

A number of potential technological problems have been indicated, some of them specifically due to the high temperature and pressures in the divertor circuit. The following ones are being adressed in Cadarache, and experiments will begin in 2004. These are the tribology of sliding parts, the leaktightness between static parts (tests on the HETIQ bench), and the behaviour of inner thermal barriers in static helium (tests on the HETHIMO bench).

It was also mentionned that, if necessary, a specific program could be settled about carbon fiber composites oxidation , with the existing thermobalance and OXYGRAPH facility, the same way as it is for Gas Cooled Reactors (GCR) isotropic graphite.

REPORTS AND PUBLICATIONS

[1] J.L. Berton et al., He technologies relevant to fusion blankets and divertor cooling, and proposition of specific tests, CEA report DER/STR/LCET 03/046, December 2003.

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