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The EC Conceptual Design Proposal of a Water-cooled Convertible Blanket for ITER

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ABSTRACT

For several years the EC laboratories have developed breeding blankets for DEMO. From this experience, it has been derived a proposal of tritium breeding blanket for the Extended Performance Phase (EPP) of ITER. The general basic ideas are the following: i) the switch from the shielding blanket used during the BPP to the breeding blanket for the EPP should not require segments replacement ("convertible" blanket); ii) its use should not have significant impact on the Basic Performance Phase (BPP); iii) design and used materials should assure good safety standards and acceptable public perception; iv) the blanket coolant should be compatible with the coolant required in the high heat-flux components (e.g. divertor, ect.); v) the required R&D should fit with the ITER time schedule; vi) the blanket should be able to withstand large power excursions and to accept long downtimes. The proposed design consists of a water-cooled liquid metal blanket, using the eutectic Pb-17Li during the EPP and a non-breeding Pb-alloy (Pb-18Mg or Pb-50Bi) during the BPP. Each segment is basically formed by a box containing the alloy, cooled by an array of poloidal hairpin-type cooling tubes and reinforced by toroidal and radial stiffeners. The coolant tubes are double-walled tubes allowing leak detections. The selected First Wall (FW) is a toroidally-drilled steel plate with brazed water-cooling U-tube. The structural material is austenic stainless steel (316L(N)) which limits the maximum acceptable neutron fluence to about 1 MWa/m^2. The advantages of using other structural materials requiring longer leadtimes, such as ferritic/martensitic steels, are also briefly discussed.

I. INTRODUCTION

In the last ten years, the European strategy for design and objectives of the fusion machine generation after JET has been to built a tokamak using a pure shielding blanket based on the use of low temperature/low pressure water and austenitic stainless steel for demonstrating plasma ignition in the first stage of operation and for testing DEMO breeder-blanket test-articles in the second stage (NET). During the ITER/CDA, beyond this Basic Performance Phase (BPP), the possibility of having an Enhanced Performance Phase (EPP) using a driver blanket (e.g., water-cooled ceramic blanket) after replacement of all shielding segments has been considered.

The use of a steel/water shielding blanket remains the EC reference option for the ITER/EDA. However, at the beginning of 1993, the ITER Joint Central Team (JCT) proposed a more advanced blanket, able to accomodate high wall loadings and to switch from shielding to breeder blanket without segments replacement ("convertible" blanket). These characteristics are obtained through the use of advanced materials, such as vanadium alloys.

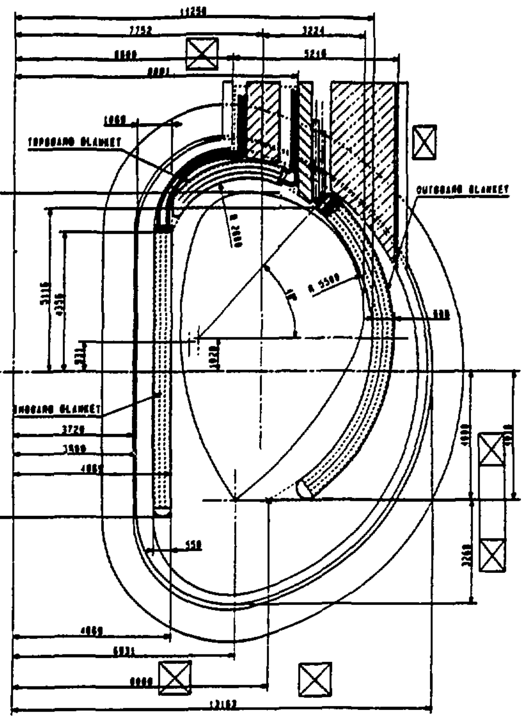

Fig. 1 : ITER layout considered in the present study

Within the EC, the choice of using advanced materials, requiring long leadtime, has been judged incompatible with the ITER timescale. However, it was recognized that the use a "convertible" blanket could be attractive. Therefore, an assessment of a "convertible" blanket option, using austenitic steel structures and water cooling, has been performed for a fusion power of 1,500 MW. It is based on the use of the eutectic Pb-17Li as breeder material during the EPP, and of another liquid Pb-alloy (Pb-50Bi or Pb-18Mg) during the BPP in order to work as "shielding blanket". Exploiting the liquid nature of the basic material, the transition shielding/breeder blanket can be performed by draining/refilling each segment, avoiding therefore segments replacement.

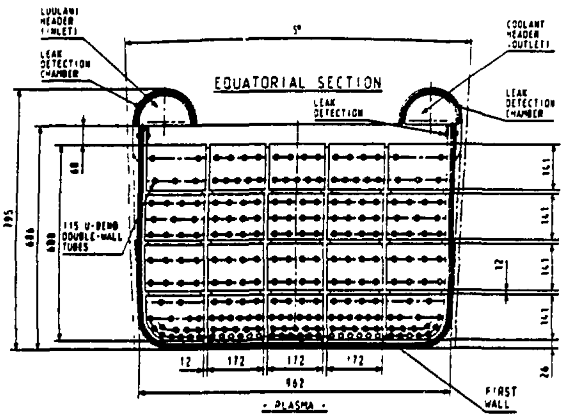

Fig. 2 : Midplane horizontal cross-section of an outboard segment

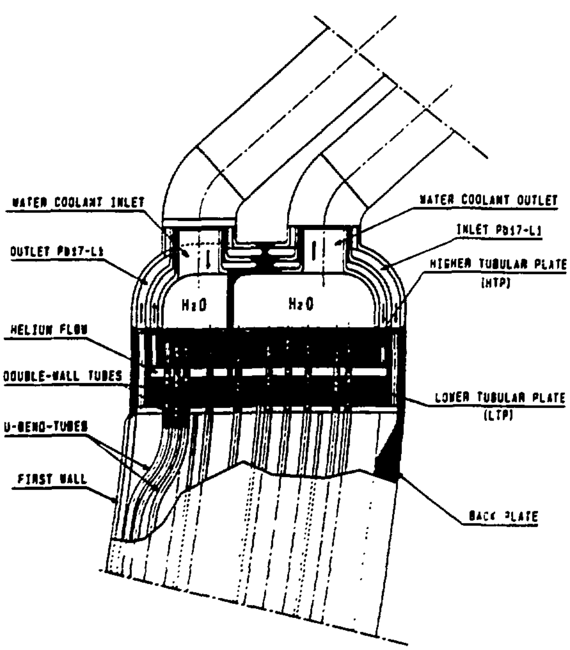

Fig. 3 : Outboard segment three-fluid headers

The work has been performed jointly by CEA and KfK with a participation of ENEA. A preliminar but quite complete report has been produced in order to assess the feasibility and attractiveness of this option and to list the required R&D in the next 5 year period [1]. The liquid metal can be cooled both with water and helium. The basic concept is independent of the coolant. However, several details depend on the coolant choice. Due to the limited room available, in the present paper only the conceptual design of the water-coolant version is presented, and in particular, only the analyses performed at CEA are described in details.

II. GENERAL DESIGN PRESENTATION

The assumed ITER layout is given in Fig.1. The blanket is formed by 72 outboard segments and 48 inboard and "topboard" segments. The selected operating parameters, together with the segment geometrical characteristics, are given in Table I. The present study was limited to the design of an outboard segment. The design and integration of divertor and limiters have not been addressed.

TABLE I : Main selected ITER specifications.

OPERATING PARAMETERS	
Fusion power (nominal)	1,500 MW
Neutron wall loading	1.0 MW/m^2
Power excursions (100 for 10 s)	2,500 MW
Maximal surface heat flux on FW (nominal)	0.25 MW/m^2
Operation time	
Basic Performance Phase	3,500 hours
Extended Performance Phase	1 year
Operating mode	pulsed
Impurity control	single-null divertor
First wall protection	2 mm of Be
Tritium Breeding	
BPP	none
EPP	as high as possible
First Wall temperature	> 200°C
Number of cycles	1-5 x 104
Pulse duration	about 1,000 s
GEOMETRICAL CHARACTERISTICS	
Plasma major/minor radius	775/280 cm
Plasma elongation	1.6
Outboard breeder-zone thickness	60 cm
Inboard breeder-zone thickness	40 cm
Number of inb/top/outboard segments	48/48/72

A. Blanket Materials

Because of the short time available for its development/qualification, the selected structural material is austenitic steel 316L(N). Its behaviour under irradiation suggests a maximum fluence of about 15 dpa and working temperature up to about 400°C. The maximum acceptable wall loading is dictated by the properties of this steel.

During the BPP, the selected liquid metal has to have physical properties close to those of Pb-17Li in order to allow the use of the same circuits. Pb-50Bi and Pb-18Mg are the most reasonnable choices. The first one, due to its low Melting Point (MP) -125°C-, allows the use of relatively low temperature/low pressure water which assures high reliabilty and safety standards during the BPP. The drawback of the relatively large polonium production due to the presence of a large quantity of bismuth has to be analyzed. The alternative is the use of Pb-18Mg (MP 250°C) which requires water-coolant characteristics close to that foreseen in the EPP.

B. Outboard Segment Layout

The segment design has been derived from the existing water-cooled and dual-coolant Pb-17Li conceptual design for DEMO [2] and from the following considerations: i) the segment structure should withstand the disruption forces; ii) the design should provide two barriers between coolant and both vacuum vessel and Pb-17Li; iii) the Pb-17Li should circulate slowly for tritium removal outside the vacuum vessel and, at the same time, for avoiding significant MHD-pressure drops.

It consists of a steel box, filled with poloidally-flowing (~3 mm/s) Pb-alloy. The liquid metal is cooled by an array of hairpin-type coolant tubes running in the poloidal direction. Such coolant tubes are double-walled tubes with a leak-detection gas (He or Ar) flowing in-between. The box is reinforced by toroidal and radial stiffening plates in order to withstand the disruption-induced loads. Fig.2 shows an horizontal cross-section of the box at the equatorial-plane level. Inlet and outlet for all fluids are located at the top of the segment. Fig.3 shows the header design which includes Pb-17Li, water, and leak-detection gas circuits.

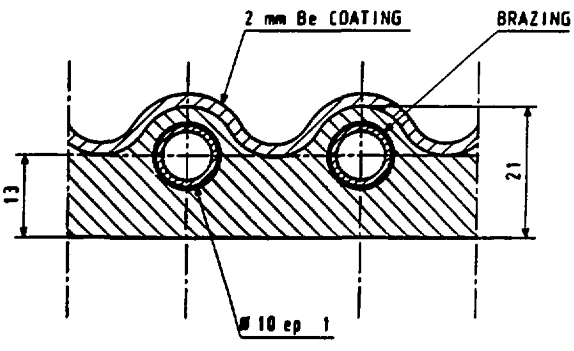

Fig. 4 : First wall cross-section (toroidal cooling)

The First Wall (FW) and the box side-walls are made in one piece and they are directly cooled by toroidally-running water tubes. The tubes are inserted in holes drilled in the plate. A FW cross-section is given in Fig. 4 where a 2mm Be-coating is also shown. The FW coolant-circuit header are at the back of the segment as shown in Fig.2. All welds are surrounded by a leak-detection chamber. One or two independent circuits can be envisaged. Fig.2 shows the case with one circuit.

III. THERMO-MECHANICAL ANALYSIS

All the mechanical and thermo-mechanical calculations have been performed with the CASTEM Codes System [3].

The use of 3 radial and 4 toroidal 8-mm-thick stiffeners enables the box to withstand an internal pressure of about 5 MPa. Considering that the use of double-walled cooling tubes drastically reduces the probability of having an accidental Pb-17Li/water interaction in comparison with single-wall ones, the requirement for the box to withstand larger pressure [4] may not be necessary (EPP water-pressure : 15.5 MPa).

For the BPP, the choice of the Pb-alloy has not yet been performed. However, for the thermo-mechanical analysis, considering that Pb-18Mg does not show significant difference compared to Pb-17Li, Pb-50Bi has been considered for the BPP because leads to a more favourable thermo-mechanical behaviour due to the lower water temperature and minimises LOCA consequences.

TABLE II : Design Point for the **Breeding Zone** during the EPP

COOLANT : **Light Water**	
pressure	15.5 MPa
inlet / avrg-max outlet temperatures	
at nominal power (P_n)	265/305-312°C
during power excursion (10 s at 1.7 P_n)	265/316-325°C
maximum velocity	7 m/s
LIQUID METAL : **Pb-17Li** (Breeder)	
inlet/outlet temperature (average)	305/265°C
average velocity in outboard segment boxes	2.5 mm/s
total outboard blanket flowrate	140 m³/h
STRUCTURAL MATERIAL : **316L(N) Steel**	
Pb-17Li/interface temperature :	
at nominal power (P_n)	< 375°C
during power excursion (10 s at 1.7 P_n)	< 390°C
COOLING TUBES : **316L(N) Steel**	
Poloidal wire-mesh bounded double-walled tubes	
U-tubes with leak detection	
internal/external outer tube diameter	13.9 / 16.9 mm
internal/external inner tube diameter	11.0 / 13.5 mm
Number of U-tubes in an outboard segment	125

For a fusion power of 1.500 MW, the estimated deposited power in an outboard segment is about 15 MW, of which about 3.6 MW in the segment envelope (FW and side walls). Starting from this integral data valid for both BPP and EPP, the thermo-mechanical analysis has been performed separately for the BPP and the EPP. The EPP showing the most severe thermal conditions, the coolant circuit dimensionning has, at first, been performed for the EPP, and then adapted for the BPP.

The two main conditions that have to be satisfied are: i) the maximum thermal stresses should comply with the standard design rules given by the French RCC-MR code; ii) the Pb-alloy/steel interface temperature should always be below 450°C because of corrosion problems (at least under nominal conditions).

A. Extended Performance Phase

The analysis has been performed with a 2D-geometry model (finite-element meshing) of the equatorial plane cross section, where the heat load is the highest. The radial distribution of the heat sources has been determined with a 1D-calculation, although if the equatorial-plane peaking factor has been determined with a 3D-calculation. The average power density in the FW steel and water is respectively about 11.5 and 10.5 W/cm^3, while is about 14 W/cm^3 in the first centimeter of Pb-17Li.

TABLE III : Design Point for the **First Wall** during the EPP

COOLANT : **Light Water**	
pressure	15.5 MPa
inlet / average outlet temperatures	
at nominal power (P_n)	265/295°C
during power excursion (10 s at 1.7 P_n)	265/320°C
maximum velocity	4 m/s
STRUCTURAL MATERIAL : **316L(N) Steel**	
protective coating	2 mm of Be
cooling-tube internal/external diameters	8 / 10 mm
tube channel/Be minimum distance	3 mm
FW overall thickness	21 mm
cooling tube channel pitch	
at the outboard segment midplane	26 mm
at the ouboard upper end	>27 mm
maximum temperature :	
at nominal power (P_n)	385°C
during power excursion (10 s at 1.7 P_n)	440°C
maximum thermal stresses	310 MPa
(Von Mises, generalized plane strain)	

The obtained design point for the EPP is given in Table II and Table III, respectively for Pb-17Li pool and FW. The tables also include the results related to the considered power excursions. From the results it appears that the FW is the most critical component because of the difficulty of accommodating the surface-heat load. The considered heat loads appear to be close to the maximum acceptable for 316L(N) steel. In these calculations, the water velocity in the FW has been limited to 5 m/s. If higher velocities are acceptable, significantly higher heat load could be accommodated.

B. Basic Performance Phase

The same type of analysis has been repeated for the BPP conditions in the case where Pb-50Bi is used. The heat sources are slightly higher than in the previous case (about 16 W/cm^3 in the FW steel, 11 W/cm^3 in the FW water, and 15 W/cm^3 in the first centimeter of Pb-alloy). However, due to the lower water temperature, the thermal conditions remain more favourable than during the EPP.

The BPP design point is given in Table IV.

TABLE IV : Design Point for an **Outboard Segment** during the BPP

FIRST WALL COOLANT : **Light Water**	
pressure	~3 MPa
average inlet / outlet temperatures	
at nominal power (P_n)	185/215°C
during power excursion (10 s at 1.7 P_n)	185/225°C
maximum velocity	4 m/s
LM-POOL COOLANT : **Light Water**	
pressure	~3 MPa
average inlet / outlet temperatures	
at nominal power (P_n)	155/195°C
during power excursion (10 s at 1.7 P_n)	155/210°C
maximum velocity	< 7 m/s
LIQUID METAL : **Pb-50Bi** (shielding, MP 125°C)	
inlet/outlet temperature (average)	195/155°C
STRUCTURAL MATERIAL : **316L(N) Steel**	
FW / blanket internal temperature :	
at nominal power (P_n)	< 330°C
during power excursion (10 s at 1.7 P_n)	< 378°C
maximum thermal stresses	390 MPa
(Von Mises, generalized plane strain)	

IV. BLANKET PERFORMANCES ASSESSMENT

The water-cooled Pb-alloy convertible blanket has shown satisfactory operating characteristics. The main results are listed below.

A. Tritium Breeding and S[illegible]'ding Efficiency

The Tritium Breeding Ratio (TBR) of the blanket has been estimated with a full 3D-geometry model. For the calculation, the outboard breeder zone thickness has been taken equal to 60 cm, the inboard and topboard ones equal to 40 cm. Without taking into account the presence of ports, the obtained TBR is about 1.17.

As far as the shielding characteristics are concerned no problems are expected. In fact, the presence of water in the blanket leads to a good shielding efficiency, especially during the EPP. Moreover, such an option is compatible with the presence of a steel/water shielding layer behind the liquid metal region, which guarantees a sufficient shield for the magnets even during the BPP.

B. Tritium Permeation towards the Coolant

One of the major concerns for water-cooled Pb-17Li blankets is the tritium permeation towards the water coolant. For DEMO, the development of tritium permeation barriers was launched several years ago [5]. Methods for producing these coatings on 316L(N), such as aluminisations, are already industrially available. Their behaviour under irradiation and in the expected blanket conditions has yet to be tested.

An estimation of the tritium permeation towards the coolant in the case where tritium barriers were not available has been performed using a 2D-geometry model. The same model and code used for the thermal analysis have been applied, exploiting the formal analogy between the heat transport equation and mass transport equation [6]. Double-walled tubes have been modelized as simple tubes, neglecting therefore the gas flow.

The results indicated that the tritium permeation towards the coolant is less than 2 g/d, if one assumes ten recirculation per day of the Pb-17Li inventory (v~2.5 mm/s) and a tritium extraction efficiency from Pb-17Li of 90%.

Assuming to apply existing water-detritiation techniques, the maximum tritium concentration level in the coolant can be kept below 5 Ci/kg. Reduction of such a value can be obtained by improving the water detritiation techniques or by using tritium permeation barriers on the coolant tubes.

More detailed calculations are however required before reaching a final conclusion. In particular, the technical feasibility of the T-extraction from Pb-17Li with 90% efficiency has to be proved. A significant reduction of such an efficiency would lead to a significant increase of the T-permeation.

C. Behaviour during Plasma Disruptions

Preliminary calculations have shown that, if all the segment FWs are electrically connected, the stiffened box is able to withstand the disruption-induced loads.

In the present study, no attempt has been made for designing such electrical connections.

D. Thermal Response during LOCA

Previous studies, performed for a "box-shaped" water-cooled Pb-17Li DEMO blanket had shown that if Pb-17Li pool and FW were cooled by two independent circuits, the maximum temperature remained within acceptable limits when a LOCA with delayed plasma shutdown happened on any one of the two circuits [7]. The extrapolation of those results to the present design indicates that no major problems have to be expected in the case of the reference option, where the FW itself is cooled by two indipendent circuits. The presence of two circuits has been foreseen in order to avoid a temperature difference between the two sides of the segment box.

IV. FABRICATION CONSIDERATIONS

A preliminary fabrication/mounting sequence analysis has been performed. The segment is divided in three sub-assemblies: i) a first one composed by the tube bundle and its spacing grids, the stiffeners, the external shell, the back plate and the two upper tubular plates; ii) a second one, composed by the three upper headers; iii) a third one, composed by the box bottom. After assembling separately the three sub-assemblies, they are welded together as the last operation.

Among the foreseen sequences, two operations appear more critical: i) the bending of the external shell of the box after insertion of the cooling tubes (double bending), and ii) the welds between the back of the stiffeners and the back plate. Discussions with industry are presently under way. The first problem could have an immediate solution if one accepts the box being formed by several straight parts, with the drawback of having horizontal double welds submitted to high neutron fluxes. The solution to the second point could be the use of latest laser-welding techniques.

VI. REQUIRED R&D

In order to prove the feasibility of the water-cooled convertible blanket using 316L(N) structural material, significant R&D has to be planned in addition to that performed in the last few years [5]. The most important areas of required R&D are: i) adaptation of the double-walled tube technology to 316L(N); ii) compatibility between Pb-50Bi (or Pb-18Mg) with steel; iii) tritium control including permeation barriers and T-extraction; iv) Pb-17Li purification from erosion and/or activation products; v) 316L(N) behaviour under specific fusion reactor conditions (e.g., low-cyle fatigue, high-energy neutron irradiation, etc.); vi) welds reliability and inspection; vii) blanket electrical connections and limiters.

It is expected that answers to all these areas can be given by the beginning of the ITER construction. The risk of non-success have however to be evaluated.

VII. COMPARISON WITH A STEEL/WATER SHIELDING BLANKET

The comparison between the option of a Convertible Blanket (CB) and the option of having for the BPP a more conventional steel/water Shielding Blanket (SB) and for the EPP a Driver Blanket (DB, e.g., water-cooled ceramic blanket) leads to the following considerations: i) during the BPP, the two blankets appear to be very similar in terms of fabricability, reliability and lifetime; ii) the CB can assure sufficient shielding, lower than the SB but equivalent to the DB; iii) DB can use more advanced materials but with the limits of using the same coolant as the SB; iv) main building construction costs increase with the increased water temperature and pressure (safety); v) the use of a liquid metal in the CB requires more complex facilities and increases the unavailability risks; vi) required R&D costs are larger for CB than for SB, but become similar if the DB R&D costs are added; vii) the risk of not succeeding on time for the breeding phase of the CB has to be weighted against the cost of hardware segments replacement.

VIII. USE OF FERRITIC/MARTENSITIC STEEL

Ferritic/martensitic steels could be used as a structural material in the water-cooled convertible blanket without any major change of the conceptual design described in the present paper. From the thermo-mechanical point of view the use of a ferritic/martensitic steel would allow to accommodate a much larger neutron wall loading, up to at least 2 MW/m^2, which corresponds in case of ITER to a fusion power of about 3,000 MW. A design with an indirectly cooled FW could be acceptable if the fusion power is limited to 1,500 MW.

Two major drawbacks have, however, to be considered: i) the significant increase of the DBTT for most steels of this family at neutron doses of few dpa makes difficult their use in a fusion reactor; ii) as a consequence, despite the significant R&D already performed on these steels (including behaviour under irradiations) the leadtime for reaching a steel composition with acceptable properties may be too long for the ITER timeschedule.

IX. CONCLUSIONS

The water-cooled Pb-alloy convertible blanket, using 316L(N) structural material appears to be a good candidate for ITER. The paper has described its major features.

Together with the significant advantage of avoiding segments replacement between the BPP and the EPP, it associates an expected reduction of the ITER availability during the BPP operations if compared to steel/water shielding blanket.

However, in comparison with other blanket proposal for ITER, it requires for the BPP version only limited improvement of present technologies. In particular, it can act as efficient shielding blanket without requiring new materials development.

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