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OVERVIEW OF EU ACTIVITIES ON DEMO LIQUID METAL BREEDER BLANKET

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The present paper gives an overview of both design and experimental activities within the European Union (EU) concerning the development of liquid metal breeder blankets for DEMO. After several years of studies on breeding blankets, two blanket concepts are presently considered, both using the eutectic Pb-17Li: the dual-coolant concept and the water-cooled concept. The analysis of such concepts has permitted to identify the experimental areas where further data are required. Tritium control and MHD-issues are, at present, the activities on which is devoted the greatest effort within the EU.

1. Introduction

The demonstration reactor, DEMO, can be defined as the unique step after ITER and should therefore contain the full technology of a fusion power plant. The main functions of the DEMO blanket using a D-T plasma are: i) the extraction of the heat produced by the plasma-neutron irradiation in a form suitable for electrical energy production; ii) the production of an amount of tritium sufficient for self-sustaining the fusion reactions; iii) the contribution to the shielding of the superconducting coils. These functions have to be ensured in a quasi-continuous way during the expected reactor life-time.

Since several years, many studies have been performed within the EU in order to identify the most attractive breeding-blanket technologies and to optimise the blanket conceptual design with respect to its functions. The preliminary assessment led to the selection of four lines of blankets, two of them using the liquid eutectic Pb-17Li as breeder material and the Pb-17Li itself or pressurised water as coolant, the other two using Li-ceramics as breeder and helium as coolant [1]. In 1989, a six-year programme has been started, the Test-Blanket Development Programme, with the objective of assessing and comparing the performances of these four blanket lines under agreed DEMO specifications, in view of the selection by mid-1995 of the two best options. The ultimate aim is to develop, beyond 94, test-articles representative of the selected blanket options, to be tested in ITER.

Liquid Pb-17Li blankets present the potential advantages over the solid breeder options of allowing tritium extraction outside of the blanket, of avoiding irradiation-induced degradation of the breeder/multiplier material, and of permitting an overall less complex design (e.g., neutron multiplication and tritium breeding are ensured by the same material). The present paper gives, at first, the description of the reference design for the two selected lines of Pb-17Li blankets including their main performances and their critical issues. Then the related R&D activities presently underway within the EU will be discussed. They concern all typical issues related to these types of blankets, such as water/liquid metal reactions, MHD phenomena, tritium extraction and control, steel corrosion by Pb-17Li, and Pb-17Li purification and composition control.

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2. Considered DEMO Specifications

Within the Test-Blanket Development Programme the design activities are seen mainly as the way of identifying the most promising blanket technologies for a future DEMO reactor and of defining the requirements in terms of R&D and of blanket or rating parameters. The definition of final specifications for DEMO and, consequently, the detailed blanket design will come out from the experience gained on previously operating fusion machines (e.g., ITER).

Therefore, since 1989, the set of parameters largely derived from NET and given in Table I has been selected [1] as a basis for the design work with the intention of keeping it unchanged until the end of the six-year programme in order to allow a consistent comparison and selection among the blanket lines.

The required performances for the blanket are: i) a Tritium Breeding Ratio (TBR) exceeding unity in presence of 10 ports (3m-high and with a full-outboard segment width) and of top- and bottom-divertor plates of given compositions with room for breeding zones behind them (double-null plasma); ii) net thermal efficiency exceeding 20%; iii) blanket lifetime of 20,000 h with full-power continuous operation; iv) possibility of segments removal after a single disruption (current quench time of 20 ms) but without recovering nominal operation; v) feeding-pipe access from the top side of the blanket segments, with the exception of the inboard segments for which a partial access from the bottom is tolerated.

As far as the structural material is concerned, because of the severe DEMO operating conditions (neutron irradiation up to 70 dpa, 250°C-500°C operating temperature range), the selected structural material for the present design activity is 1.4914-martensitic steel (MANET) which, compared to austenitic steel, presents the advantages of having an acceptable swelling in high-fluence, high-temperature conditions, a higher thermal conductivity and strength at high temperature, and at the same time has a satisfactory and well-established out-of-pile properties data base and a significant in-pile data base. On the other hand, the significant irradiation-induced shift in its ductile-to-brittle transition temperature (up to 250°C or more) requires an optimisation of its composition. The development of a suitable structural material for DEMO is the objective of an independent EU programme.

3. Lithium-Lead Blanket Reference Designs

The two EU Pb-17Li blanket reference designs, the dual-coolant and the single-box water-cooled concepts, are the results of a development process dating back to the early eighties. The dual-coolant concept is derived from the self-cooled one, where the Pb-17Li acts as a coolant for the whole blanket including the First Wall (FW), and whose original design is described by Malang et al. [2]. As far as the water-cooled blanket is concerned, several design versions have been developed in the past such as those described by Casini et al. [3] and by Giancarli et al. [4]. The single-box design has been retained as the reference concept mainly because of its resistance to disruption-induced loads.

The basic structure of a given blanket segment for both reference designs is very similar. It consists of a directly-cooled reinforced steel box acting as container for the flowing Pb-17Li (pool-type structure).

3.1. Dual Coolant Pb-17Li Blanket

In principle the most simple blanket design is achieved if the liquid metal is used not only as breeder material but as coolant as well. The design of such self-cooled blankets, however, has to overcome the following crucial issues:

- First wall cooling with liquid metal requires rather high velocities (> 1 m/s), leading to high magneto-hydrodynamic pressure drop.
- There is only a single wall between the liquid metal and the plasma. This implies the risk of a liquid metal spill into the plasma chamber if there are leaks. Such a spill could cause a rather long downtime of the machine.
- Malfunctions in the liquid metal circulation loop could make it difficult to avoid Pb-17Li freezing and to remove decay heat.

These issues were the incentive to develop a blanket concept with a self-cooled breeding zone but with a helium-cooled FW. The following sections give the details of this design.

3.1.1. Main Design Features [5]

The following criteria had been postulated for the design in order to achieve an optimum in safety and reliability:

a) real double containment of the liquid metal with a monitoring space between the two walls;

b) single walls allowed between helium and plasma but not single welds;

c) each weld is allowed to leak without requiring an immediate shutdown of the machine and a segment removal;

d) liquid metal ducts designed for the full gas pressure to avoid damage propagation in the case of gas leaks without fast pressure relief;

e) malfunction of one cooling loop must not lead to temperature rises in the blankets requiring segment exchange.

These criteria are met with the design shown in Figs. 1 and 2. The concept is characterised by a U-shaped FW with helium cooling channels in toroidal direction which, together with the helium manifolds, form a box which contains a grid of steel plates creating large poloidal ducts for the Pb-17Li flow. The box, shown in Fig. 1, is fabricated by diffusion welding and provides a real double containment of the liquid metal. A blanket segment is constructed by welding together sections of the U-shaped box fabricated with a poloidal height of approximately 1 meter (Fig.2). The design and technology of the welds between sections is an important detail of the concept. It is anticipated to use electron beam welding to obtain two independent welds with a monitoring gap in between. Monitoring a pressure in this intermediate gap allows to detect a leak in one of the welds at a time when the Pb-17Li is still contained by the second weld. Another benefit of the design is the position of the weld at the side walls of the rectangular helium channels, thus avoiding any weld between helium and the plasma chamber. Both principles, i.e., no welds between helium and plasma and redundant welds between Pb-17Li and plasma, are employed throughout the entire segment and result in improved safety and reliability of the blanket.

Helium cooling of the first wall is divided into two completely independent loops. The cooling channels are alternatively connected to one of the loops in order to minimise FW temperature increase in case of a Loss-Of-Coolant Accident (LOCA) in one of the circuits. The Pb-17Li enters the blanket at the top, flows downwards in the three rows of parallel channels at the rear side, turns around at the bottom by 180°, and flows upwards in the first row. This allows to adjust the velocity in each row accordingly to the local heat-generation rate. Compared to a full self-cooled concept, the liquid metal ducts are larger and the required liquid metal velocity is smaller since the FW heat flux has not to be removed by liquid metal cooling. Nevertheless, some electrical insulation is required between the load-carrying walls and the flowing liquid metal. This could be obtained either through the use of flow-channel inserts in which a thin ceramic layer is sandwiched between two steel sheets or by coating of the duct walls with an insulating layer. The second method has a higher potential if its feasibility can be proved and is, therefore, the proposed solution.

3.1.2. External Circuits and Components

External loops are required for heat and tritium extraction from the blanket. Both functions are combined in the proposed dual-coolant Pb-17Li concept where both heat and tritium are transported to the steam generators by circulating the Pb-17Li [6]. Double-walled tubes are used in the steam generators where the liquid metal flows around the outer tube, the water in the inner tube, and a secondary liquid metal (NaK) in the concentric gap between the tubes. While the heat flows from the Pb-17Li to the water through both walls and the NaK in between, the tritium permeates only through the outer wall and is dissolved in the NaK. The NaK is circulated at low velocity through the gap to a cold trap for tritium removal. Tritium permeation losses to the water are negligible because of the high tritium solubility of NaK, and the resulting very low tritium partial pressure. There are two cold traps arranged in parallel which are periodically drained and heated up for batchwise tritium recovery.

The total amount of NaK is small and the steam generators are designed in a way to avoid any failure propagation in case of water leak.

The heat removed from the FW by helium cooling amounts to roughly 25% of the total heat input into the blanket. This heat is transported to the same steam cycle connected with the liquid metal loops. The loop systems are completely independent and subdivided into a number of parallel loops in order to provide redundancy for the case that there is a malfunction in one of the loops. This redundancy results in a high availability of the blanket system and ensures a reliable afterheat removal.

3.1.3. Blanket Performances Assessment

The neutronic analysis is based on three-dimensional Monte Carlo calculations using the MCNP-code [7, 8]. The considered geometrical model consists of half a 22.5-degree torus sector with one inboard and one and a half outboard segments. Table II shows the Tritium Breeding Ratio (TBR) determined by following about 50,000 neutron histories [5].

The TBR in an actual reactor will be reduced due to the presence of openings in the blanket for devices such as those for plasma heating and remote handling. Following the DEMO specifications, it has been estimated that 10 horizontal ports in the outboard blanket reduce the TBR by about 6% to 1.15 [9].

A detailed calculation of the spatial power density distribution forms the basis of the thermal hydraulic blanket layout. For this purpose a coupled neutron-photon Monte Carlo calculation has been performed following about 150,000 neutron histories. The calculated total power (including an average surface-heat flux of 0.4 MW/m^2) is listed in Table III together with the main coolant parameters.

A MHD analysis [5], assuming a resistance of the insulating coating characterised by the product of resistivity and layer-thickness = 10^{-4} Ωm^2, determined a total pressure drop of 0.5 MPa in the blanket. The pressure drops in outboard and inboard segments are nearly identical since the blankets in the inboard region are split into upper and lower halves which out-balances the higher magnetic field strength there.

Scoping calculations showed that the dual-coolant concept is acceptable not only with insulating coatings but also with flow channel inserts too. A 0.5-mm-thick steel liner in contact with the flowing Pb-17Li would lead to a total MHD-pressure drop of roughly 3 MPa. This is a reasonably low value since the strong blanket segment can allow a liquid metal pressure up to 10 MPa.

The blanket design therefore offers large margins for additional loads caused, for instance, by plasma disruption or internal helium leaks. Even a LOCA leading to pressurisation of the Pb-17Li region up to the full helium pressure could be contained in the FW-box. Detailed thermal calculations have shown that a LOCA in one helium loop with a delayed plasma shutdown would lead to a temperature increase in the FW of less than 30°C.

3.2. Water-Cooled Pb-17Li Blanket

The use of water as a coolant for a Pb-17Li breeding blanket, which leads to the decoupling of the heat-removal and tritium removal functions of the blanket, permits to avoid MHD-issues because the Pb-17Li needs only to flow at a very low velocity (~ 5 mm/s). Moreover, the same coolant can be used for all reactor components (e.g., divertor, blanket, shield, vacuum vessel, etc.) and it can use for the heat-exchangers a mature nuclear technology (fission).

For this last reason, within EU all existing designs for this line of blanket make use of thermo-hydraulic data typical of a PWR which are: i) water pressure of 15.5 MPa, ii) maximum water velocity of 7 m/s, and iii) water outlet temperature of 325°C. The additional choice of the water inlet temperature of 265°C is dictated by the Pb-17Li melting temperature of 235°C. The recently-selected reference version, the so-called single-box version, follows the same basic characteristics.

3.2.1. Main Design Features [10]

The single-box design version adopts the following design options:

i) single walls allowed between plasma and liquid metal but not single welds;

ii) two barriers between cooling water and plasma, and between FW coolant and Pb-17Li (both barriers able to withstand the water-coolant pressure);

iii) double-containment of the Pb-17Li-pool cooling water towards the liquid metal, using Double-Walled Tubes (DWTs) with provision for detection of leaks through any of the two tubes, both able to withstand the water-pressure;

iv) each weld is allowed to leak without requiring an immediate shutdown of the machine for segment replacement;

v) malfunction of one cooling circuit, assuming a delayed plasma shutdown, should not induce temperature rises which could affect further segment operation.

Each blanket segment is formed by a single-steel box which acts as the liquid metal container. As for the design of the dual-coolant concept, radial and toroidal stiffeners are added in order to withstand the disruption-induced forces and, at the same time, both the Pb-17Li hydrostatic pressure in normal conditions and the full water-pressure under faulted conditions (Fig. 3). The design point for an outboard segment is given in Table IV.

The segment-box is directly cooled with an independent circuit. The cooling tubes are inserted and soft-brazed in toroidal channels which are previously drilled in a single steel plate ensuring two barriers between coolant and plasma chamber. This plate is then bent to form FW and side walls. The FW is machined to its final optimised undulated shape. The FW-coolant is collected in vertical headers located behind the back plate (Fig. 3) in order to have water inlet and outlet at the top of the segment (Fig. 4). All welds are placed in a double-wall system which provides leak detection by means of a low-pressure inert gas flow. Thermo-hydraulic considerations have led to the choice of a water inlet temperature of 300°C.

The cooling water in the Pb-17Li pool flows downwards in the front part of the segment and upwards in the rear part within U-shaped DWTs having both walls able to withstand the water pressure. As for the vertical collectors, the DWT are designed to permit a gas flows between the two walls in order to detect possible leaks and to permit to stop the reactor and to remove the faulted segment as soon as a maximum acceptable leak-rate has been reached. This feature reduces considerably the probability of tube rupture, although increases the blanket complexity with a consequent reduction of its reliability. The Pb-17Li flows in counterflow with water. The selected DWT type are the wire-mesh bonded DWT, developed for fast-breeder reactor steam-generators in order to limit the Na/water interaction [17]. The collector system for the three fluids (water, low-pressure inert gas and Pb-17Li) is placed at the top of the segment (Fig.4). The design, which envisages feeding pipes formed by three concentric tubes, ensures double containment of water until the outside of the vacuum vessel.

3.2.2. External Circuits and Components

Because of the choice of using PWR parameters for the water coolant, the layout and the main components of the cooling circuit can be very similar to the existing ones. However, because of the expected tritium permeation from Pb-17Li towards the water within the blanket, a water detritiation system has to be added to the cooling circuit (e.g. distillation columns). However, as it can be seen from the next section, the design of this component depends on the amount of tritium which actually permeates in the water. Such a value strongly depends on the availability of efficient permeation barriers and, also, on the development of DWT technology and on the possible exploitation of the presence of a flowing gas within the DWT gap.

The Pb-17Li flow rate is about 300 m^3/h for the whole reactor blanket. The main external components will be the tritium extractor and purification system (see following section). Preferred tritium extraction processes are gas/liquid metal extractors or permeators. The tritium extraction could be performed on line.

The inert gas circuit (He or Ne) for leak detection will be at low pressure and low flow rate. Leak detection could be performed by monitoring the variation of the gas pressure and/or of the gas composition. One independent detection system per segment is foreseen.

3.2.3. Blanket Performances Assessment

Three-dimensional 3D-geometry model calculations [10], performed with the MCNP Monte Carlo code and taking into account all heterogeneities and 3D-features such as exhaust-pump duct entrance, two divertor plates, and D-T neutron source distribution, with the exception of the horizontal ports, indicated a TBR-value of 1.16. The effect of the presence of 10 ports on TBR was estimated by Fischer [9] for the dual-coolant Pb-17 Li concept in a TBR reduction of about 6%. Using this figure for the water-cooled concept would lead to a TBR of 1.09 which can ensure tritium breeding self-sufficiency.

One criterion considered in the design has been that, in order to avoid significant corrosion, the Pb-17Li/steel interface temperature should remain below 480°C. After optimisation of the coolant-tube number and location in the Pb-17Li pool, such a condition has been met everywhere. The thermal stresses are below the acceptable limits for martensitic steel in all parts of the segment.

The thermal analysis performed under LOCA conditions with delayed plasma shutdown in one of the cooling circuits indicated that the temperature rise in the structure remains within acceptable limits. In case the LOCA concerns the Pb-17Li-pool cooling circuits the maximum temperature remains below 480°C, while when it concerns the segment-box cooling circuit the maximum reached temperature is about 520°C.

3.3. Main Critical Issues

One of the main objectives of the design activities was to identify at first the critical issues for each blanket line, and then to define the physical phenomena for which new experimental data were required and to propose R&D activities for those fields where new materials and/or techniques had to be developed.

The main critical issues for Pb-17Li blankets have been discussed in recent papers by Malang [11] and Proust [1]. Taking into account the evolution of the designs and the obtained experimental results, the relative weight of such issues has also evolved. Using the design work performed for the two EU reference Pb-17Li concepts as a basis, the main critical issues can be summarised as follows:

- MHD issues: in the dual-coolant concept, prohibitive pressure losses would occur due to the induced electric currents if the structural material is not electrically insulated from the Pb-17Li. The use of insulating layers or flow-channel inserts are possible solutions provided that their low conductivity can be ensured for the whole blanket lifetime.

- Tritium control: in the water-coolant concept, the Pb-17Li residence time is about two hours. Even considering a very low tritium concentration in the Pb-17Li (assuming the availability of an efficient on-line tritium extraction system), such residence time combined with the low tritium solubility in Pb-17Li leads to a significant tritium partial pressure (tens of Pascal) and consequent permeation to the water coolant. The tritium permeation rate, if no specific actions are taken on the cooling tubes, is of the order of 100 g/d (about 1/4 of the produced tritium in the blanket). The extraction of tritium from water is feasible (distillation) but cost and plant size considerations limit the rate to approx. 1 g/d. It is therefore required to reduce the tritium permeation rate by at least two orders of magnitude. Possible solutions are the use of tritium permeation barriers and/or exploitation of the leak-detection gap in the DWT [10]. Tritium inventories are of the order of few tens of grams with less than 20 grams in the Pb-17Li.

- Water/liquid metal interaction: in the water-cooled concept the use of DWTs with leak detection system considerably reduces the probability of water/Pb-17Li interactions. A possible event could be the guillotine break of a DWT, in case of a common-mode failure of the two tubes, within a segment box. Preliminary experimental results have indicated that this event would lead to an overpressure in the segment box equal to the water pressure. For this reasons the box has been designed to withstand such a pressure in faulted conditions. The increase of pressure in the box will also tend to stop the water leakage and consequently the reaction itself. In the dual-coolant concept water-liquid metal interaction could occur in the steam generator between steam and NaK (and eventually Pb-17Li). The minimisation of the quantity of NaK and Pb-17Li participating to the reaction (e.g., by partitioning the Pb-17Li inventory in many independent primary loops) is one of the design criteria for limiting the consequence of such an accident.

- Activation products in Pb-17Li: the presence of activated corrosion products and of radioactive transmutation-products due to Pb-neutron irradiation (e.g., ^{210}Po), could be a safety concern in case of accident. In-line purification methods are being developed for maintaining those products down to an acceptable level.

- Fabricability and reliability issues of blankets and ancillary equipment: these aspects are general issues for all kinds of breeding blankets [1]. However, some issues are specific to Pb-17Li blankets and some others are specific to the selected reference concepts. Among the latter, there are, for instance, the development of DWT and the manufactory of the segment-box in the water-cooled concept, and the development of diffusion welds and of the liquid metal steam generator in the dual-coolant concept.

4. Related Experimental Activity

A large experimental programme associated with Pb-17Li blankets has been running within the EU for several years [6, 12]. Many of the experimental activities foreseen in the programme are still underway and some are expected to continue for the next few years. They are focused on the main critical issues and on the main Pb-17Li data base uncertainties. In particular, they include work on MHD-issues, tritium extraction, insulating layer and permeation barrier development, water/liquid metal interaction, steel corrosion by Pb-17Li, purification and composition control of Pb-17Li. The following sections give a short summary of the main results obtained in all these fields.

4.1. Water/Pb-17Li Reactions

Experiments on the water/Pb-17Li interaction simulating a large break of a cooling tube occurring in a water-cooled Pb-17Li blanket were performed a few years ago in the BLAST facility [13]. They showed that the chemical interaction is limited by the formation of solid LiOH and Li_2O which insulates the melt from the water. The pressurisation of the container did not significantly exceed the actual water-injection pressure and the increase of temperature was limited to 100°C at a Pb-17Li-temperature of 350°C. Large scale experiments are required in the near future in order to confirm these results.

Experiments for investigating the evolution of water micro-leaks into Pb-17Li has been performed in the RELA-I facility [14]. The results of a 3-hour test indicated that no significant wastage was present. A 100-hour test is foreseen this year in order to obtain a more reliable extrapolation to the 20,000 hours of the DEMO-lifetime. It must be stressed, that in the reference design of the water coolant concept, because of the presence of DWT with leak detection, micro-leaks are no longer an issue.

4.2. Magneto-hydrodynamics Issues

4.2.1. General MHD Phenomena

For several years, the MHD program was primarily devoted to the understanding of the MHD issues related to the self-cooled Pb-17Li blanket where also the toroidal FW channels are cooled by Pb-17Li [11]. For this blanket concept, even in the case of insulated walls, pressure drop was a concern for: i) duct geometries with flow distribution or collection (poloidal-radial manifolds) and ii) multichannel effects for a system of U-bends (radial-toroidal-radial channels). Furthermore, the possibility of unfavourable velocity distributions in the FW coolant channels could result in unfavourable heat transfer and give rise to hot spots.

Different type of experimental investigations were performed in co-operation with the Latvian Academy of Sciences (Latvia) and the Argonne National Laboratory (USA) using

NaK, InGaSn, an Hg as liquid metals. The theoretical analyses were based on the Core Flow Approximation method [15]. Concerning the pressure drop issues, they showed that for all investigated geometries the pressure drops are significantly lower than those previously assessed, which increases the safety margin of this type of blankets. The experiments on velocity distributions in the toroidal U-bend showed that for blanket relevant conditions, the liquid metal will flow preferentially close to the FW, as predicted by the theory, which will result in favourable heat transfer characteristics. In the experiments, a pair of strong vortices was observed which should further enhance heat transfer.

4.2.2. Dual-Coolant Concept

The accumulated MHD-experience on MHD phenomena, together with specific experimental and theoretical work on insulated channels [16], is also very useful for the dual-coolant concept. The design is characterised by a simple flow-channel geometry, avoiding all MHD problems involved in cooling the FW with liquid metal. The Pb-17Li flow geometry consists essentially of: i) the inlet and outlet manifolds; ii) long ducts perpendicular to the magnetic field; and iii) the flow region at the blanket bottom where the flow from three poloidal channels is combined and reversed (180-degrees).

The pressure drop in these components depends on the kind of the used electrical insulation: Insulating Coatings (ICs) or Flow-Channel Inserts (FCIs).

The obtained results can be summarised as follows [16]:

- the pressure drop in the ducts perpendicular to the main magnetic field was predicted to be 0.5 MPa using ICs and 3.2 MPa using FCIs. In both cases, the contribution of geometry iii) is negligible compared to geometry ii), and the pressure drop in the inlet/outlet manifolds is expected to be small (order of 0.2 MPa).

- in case of use of FCIs, the total pressure drop is about 3.2 MPa which is considerably below the acceptable system pressure of 10 MPa. A broad basis of experimental data and theoretical analyses exists for this kind of insulation. It is sufficient for the blanket conceptual design phase.

- for direct ICs, which is the preferred design option, the pressure drop is reduced by a factor of about 50 compared to FCIs (assuming ICs without defects). For a detailed design, more theoretical and experimental work is required to better predict the velocity distribution in the blanket which has an impact on corrosion and to a lesser extent on heat transfer.

Another concern is the occurrence of insulation defects, which will increase pressure drop and might result in different flow rates in parallel ducts [21]. Although simple MHD methods exist to cope with coatings imperfections (e.g., by increasing intentionally the pressure drop in parallel channels by using electrical conducting parts) more MHD experiments with blanket-relevant materials will be required in the future.

4.2.3. Water-Cooled Concept

In this case, the Pb-17Li is circulated through the blanket at low velocity for tritium removal outside the vacuum vessel. If the liquid metal inventory is circulated ten times a day, the MHD pressure drop for uncoated tubes is about 0.2 MPa [17], which raises no problem from the pressure loss point of view. However, the consequences of the uneven velocity distribution which occurs have to be taken into account in the estimation of the tritium permeation rate towards the water-coolant. If tritium permeation barriers on the coolant tubes are used (i.e., Al_2O_3), they could also act as electrical insulators avoiding all concerns on the resulting velocity distribution.

4.3. Tritium Extraction Techniques

4.3.1. Dual-Coolant Concept

Tritium removal from Pb-17Li is performed by permeation into NaK and precipitation as tritide in a cold trap. For tritium recovery, the tritide is decomposed by heating-up the cold trap and pumping off the T_2-gas. Tritium permeation into the NaK is not considered as a critical issue because no oxide formations are expected at the interfaces steel/NaK and steel/Pb-17Li. Therefore, the investigations concentrated on the kinetics of tritium removal/recovery in/from cold traps. In the experiments, tritium was replaced by protium [18, 19].

For tritium removal, cylindrical cold traps with wire-mesh packing were used. Two types of experiments were performed: i) cold trap loading periods of about 5 hours and subsequent hydrogen recovery, and ii) long time loading (~100 h) and subsequent determination of the hydrogen distribution in the cold trap. For i), it has been proved that high cold trap efficiencies can be reached at blanket-relevant concentration range (~50 wppb). The precipitation process can be well described using a diffusion-dominated model for crystal growth. For ii), a good agreement has again been obtained between measured and predicted hydrogen distribution.

For tritium recovery, experiments were performed in a temperature range between 280°C and 400°C with complete cold traps or single wire-mesh packing. The recovery rate coefficient was described assuming first order reactions. No differences for the two types of experiments were found. At ~400°C, nearly 100% of the hydrogen is recovered in less than 3 h; two removal/recovery cycles per day appear feasible for a blanket tritium extraction system.

4.3.2. Water-Cooled Concept

Several techniques have been considered for the on-line tritium extraction from Pb-17Li, such as liquid/gas contactors or permeators. The objective is to optimise the design of the extractors in order to reach an extraction efficiency as high as 90%.

Liquid/gas contactors, such as plate columns or bubble columns, work by bringing a large Pb-17Li surface in contact with an inert purge-gas which desorbs dissolved tritium by Pb-17Li. The plate column extractor consists of a column of horizontal plates on which a relatively thin layer of tritium-enriched Pb-17Li flows by gravity motion. The inert gas flows in counter-current above the Pb-17Li. In a bubble-column extractor the Pb-17Li flows downwards in a column at the bottom of which gas is bubbled through a special nozzle. The purge-gas then has to be treated for the removal of gaseous tritium, e.g., by oxidation to H_2O and subsequent absorption in molecular sieves. The principle of permeators is to bring a highly-permeable metal membrane (Fe or V, eventually coated with Pd) in contact with Pb-17Li. The permeated tritium can then directly be pumped off in gaseous form. The relatively simplicity of this last method is paid for by the difficulties of requiring large contact surfaces and of maintaining the interface-wall free from oxides which reduce permeation.

At present, only few results on liquid/gas contactors are available. The tests have been performed in the MELODIE loops which use about 50 litres of H-enriched Pb-17Li, flowing in ducts with aluminised inside wall for reducing permeation [20]. So far, only plate column contactors have been tested. The maximum extractor efficiency was found to be less than 10% and to depend on the temperature (370-490°C) and on the H-partial pressure (100-2,000 Pa).

In fact, the tests were performed for understanding the basic phenomena occurring in the contactor and for measuring mass-transfer coefficients. Argon was used as purge gas, while He (doped with 1% of H) is expected to be used in the final process. Tests of optimised versions of plate-column contactors and of bubble-column contactors are already been planned in the next few months.

4.4. Insulating Layer Development

As discussed in section 4.2.1, the MHD-pressure drop can be significantly reduced by an electrical insulating layer placed between the flowing Pb-17Li and the metallic duct wall [21]. The candidate material for such a coating in contact with Pb-17Li is alumina. This ceramic has an high resistivity and is compatible with the liquid breeder up to 500°C. Methods are under development to form such a coating on steel structures. The most promising way is to deposit an Al-rich sub-layer between steel and alumina. This method has the advantage that cracks or other flaws in the insulator would lead to the exposure of aluminium to the liquid metal, resulting in the in-situ reparation of the insulating layer by diffusing oxygen from Pb-17Li into the aluminium. Such a self-healing mechanism is a prerequisite for the feasibility of insulating coatings because a direct contact between liquid metal and metallic wall would lead to an intolerable high MHD-pressure drop if occurring in more than one location in a duct.

Different methods for the formation of the Al-sub-layer and the alumina coating are under investigation. The most promising ones are either to dip the steel structure in a pool of aluminium or to pump aluminium through the component for a few minutes to coat the steel surface. The following step is the surface oxidation of the layer. There are number of tests under way to judge the adherence of this layer to the steel structure, and to evaluate its compatibility with Pb-17Li and the electrical resistance in contact with the liquid breeder [22].

The largest uncertainty in the feasibility of insulating coatings, however, is the degradation of the electrical resistance under irradiation. This degradation might be especially severe because of the presence of electric field during irradiation. Unfortunately, the Radiation-Induced Electrical Degradation (RIED) has not yet been investigated for relevant fluence or for the proposed coatings. The interaction between RIED, self repair, coating stoichiometry and mechanical properties (such as swelling and adherence) represent a complex set of variables that must be further explored. In particular, experimental tests aiming at the evaluation of the effects of thermal and mechanical cycling on these coatings are at present being performed in the LiFUS-2 loop [23].

4.5. Tritium Permeation Barrier Development

Efficiency, fabrication, compatibility with the other materials, irradiation behaviour, and lifetime are the criteria on which a given permeation barriers has to be judged. For obtaining useful data on these items, several experiments were performed in condition representative of those expected in a blanket segment [12].

The envisaged permeation barriers are deposition on the steel substrate of layers of materials presenting low tritium diffusivity and/or low surface recombination (^3H, $^3H_2$). The most promising materials are aluminium with an alumina-layer on its surface (as for insulating layer described in the previous section), and ceramic materials such as TiC, TiC/TiN double layer, and SiC [24]. The results obtained with out-of-pile tests for these layers deposited by Chemical Vapour Deposition (CVD) on AISI-316L indicated that significant reduction of the H-permeation can be obtained [25]. The maximum observed reduction was about one order of magnitude and the intrinsic permeability of TiC was estimated to be about a factor 6,000 smaller than the one of AISI-316L at 400°C. There are evidences that the permeation through CVD layers is limited by the flow through coating defects rather than through the barrier.

The in-pile LIBRETTO-3 experiment [26] for testing coatings in presence of Pb-17Li showed that pack cementation aluminisation (on AISI-316L substrate) was the presently-available best performing barrier. Such a coatings are produced by industry for corrosion protection applications. Permeation reductions of a factor 100 or more, as found in previous out-of-pile experiments [27] could not be confirmed, the preliminary interpretation of the experimental results leading to a reduction of a factor slightly larger than 10. It is still unclear if this fact is due to differences in aluminisation batches, irradiation effects and/or presence of Pb-17Li. In fact, the results of an out-of-pile experiment with such barriers in contact with Pb-17Li [28] are close to those obtained with CVD coated discs without Pb-17Li. However, recent out-of-pile experiments in presence of Pb-17Li have shown that the presence of a small number of defects (e.g., cracks) in the barrier could significantly reduce its efficiency. As in the case of insulating coatings, self-healing capability appears as a necessary requirement for a satisfactory permeation barrier.

The formation of a permeation barrier on DIN 1.4914 martensitic steel by plasma-spray aluminium [29] on the surface of the steel, followed by heat treatment of the sample to form Al_3Fe and Al_5Fe_2, has been studied. This method has shown a reduction of the permeation rate between two and three orders of magnitude. There was evidences that surface reactions were the rate governing process. The assessment of other available techniques for production of Al_2O_3- or Al-enriched coatings on 1.4914-steel using optimised heat-treatments for avoiding modification of the steel metallurgical structure is being performed in order to select the most promising technique for industrial development.

An experimental campaign aiming to study the effect of combined presence of thermal and mechanical stress, in a Pb-17Li environment, on the stability and resistance of this coatings has been performed. The behaviour of the pack-cementation-Al_2O_3 coatings during 1,500 cycles, under experimental conditions representative of the blanket operations, was satisfactory [23]. Small defects and areas showing Al_2O_3 detachment were present in specimens tested for 3,000 cycles. These results confirm a sufficient compatibility of the aluminised coatings with Pb-17Li previously determined in forced and quasi-stagnant conditions [22, 30, 31] at temperatures ranging from 450 to 500°C.

4.6. Pb-17Li/Steel Compatibility

The compatibility between Pb-17Li and structural material (bare or coated) is a common issue for all Pb-17Li blankets. However, because the corrosion phenomena depends on the temperature and on the velocity of the alloy, the experimental results have different impact on the two reference designs.

4.6.1. Corrosion Aspects in the Dual-Coolant Concept

In the dual-coolant concept all liquid metal channel walls are coated with an alumina layer. This layer is compatible with Pb-17Li, up to a temperature of 500°C at least. Steel corrosion by Pb-17Li is relevant only for the back-up system employing FCIs instead of ICs. Furthermore, the FW-box is cooled by helium with a maximum temperature of 350°C. Therefore, the maximum temperature at the interface between the blanket structure and the alloy is in general below the temperature of the alloy itself. Temperature calculations showed that this interface temperature does not exceed a value of 450°C. All corrosion experiments performed with martensitic steel in Pb-17Li at velocities up to 0.3 m/s showed that the corrosion rate at this temperature remains below 50 μm/y [33].

It has been shown that the corrosion in the system MANET/Pb-17Li is controlled by diffusion of wall material into the liquid metal [34]. Therefore a diffusion model has to be used to extrapolate the measured corrosion rates to the blanket conditions. Compared to the corrosion-experiment devices, the blanket is characterised by larger ducts, higher velocities but suppressed turbulence. Estimates showed that an interface temperature of 450°C is allowed in the blanket for both alumina coatings and FCI-steel liners. The corrosion rates in the feeding pipes and in the steam generator are even lower since the temperatures there are limited to the maximum alloy temperature of 425°C.

4.6.2. Corrosion Aspects in the Water-Cooled Concept

Pb-17Li corrosion of both austenitic and martensitic steel has been largely investigated under experimental conditions of water-cooled blankets [12]. Under the same conditions, corrosion data have been produced also for Al-based permeation barriers indicating their good compatibility with Pb-17Li [12, 35]. In the water-cooled concept the maximum Pb-17Li/martensitic steel interface temperature has been kept below 480°C, which in association with the low Pb-17Li velocity (~5 mm/s), leads to maximum corrosion rates for bare steel in the blanket below 10 μm/y [32]. In the feeding pipes, the Pb-17Li temperature is below 330°C, and, therefore, despite a larger velocity, the Pb-17Li corrosion is negligible.

The effects of the magnetic fields on the corrosion rate have also been investigated in the ALCESTE loop where a magnetic field of 1.4 T is applied perpendicularly to the Pb-17Li flow [36]. For a Pb-17Li mean velocity of 1 mm/s and hot/cold temperature of 400°C/475°C, and after 3,000 hours of test, the found corrosion rates for martensitic steel was greater by about 30% than the one obtained in the same experiment but without magnetic field.

4.6.3. Effect of Pb-17Li on the Mechanical Properties of Steels

A certain number of data are available on the influence of Pb-17Li on the mechanical properties of steels [12]. For AISI-316L steel, some tests carried out just above the melting point of the alloy showed that no Liquid Metal Embrittlement (LME) had occurred. Studies of the effects of constant and cycling loads (10 to 134 MPa) indicated that at these low stresses the corrosion rate is not affected by the applied loads. A satisfactory behaviour of this steel in flowing Pb-17Li was also noted in low-cycle fatigue experiment [37]. Specimens of DIN-1.4914 steel, tested under creep rupture and low-cycle fatigue conditions, showed a decrease of the time to rupture when in contact with Pb-17Li but a better low-cycle fatigue behaviour. No effects of stress corrosion cracking has been observed. Martensitic steels present LME only in simulated welded conditions, if the post-weld heat treatment is not carried out.

4.7. Pb-17Li Purification and Composition Control

Considering the large mass of liquid breeder material involved in a Pb-17Li blanket, the lithium burn-up is relatively small. The only concern could be an increase of the alloy melting temperature. Such composition changes can be easily compensated for by adding lithium or Li-enriched Pb-alloy in the external circuits.

Several techniques for on-line Li-content monitoring, such as electrical resistivity meters, electrochemical sensors, and plugging indicators, are under development and testing within the EU [12]. Preliminary tests on electrochemical sensors have shown a very high sensitivity to Li-content in the alloy, despite of a limited reliability due to cracking of the ceramic body of the sensors under thermal stress [38].

Purification systems are required to remove impurities and corrosion products in order to minimise the neutron-induced activation of the breeder material. Systems for removing corrosion products are being developed. The production of the α-emitter ^{210}Po is of a special concern. This impurity is formed by neutron irradiation of bismuth isotopes, which can be present as impurity in the original Pb-composition and/or build-up from neutron irradiation of lead. Recent experiments [39] have shown that the release of polonium is determined by the vapour pressure of an inter-metallic Po-Pb compound which is orders of magnitude lower than that of polonium. Therefore, the release of ^{210}Po is by far not as critical as it was expected a few years ago. Nevertheless, it is desirable to develop on-line Bi-removal techniques in order to limit the Bi-level to 1 ppm resulting in approximately 0.1 ppb of ^{210}Po, without needing to remove polonium itself.

5. Final Remarks and Conclusions

The European Union is engaged since 1989 in large design and R&D effort for producing by the end of 1994 a coherent data base enabling the EU blankets teams to select the two best breeding blanket-line candidates to be used in a DEMO reactor. The selected candidates will be further developed for permitting, in the short term, a large-scale blanket testing in ITER.

Pb-17Li blankets are attractive breeding blanket proposals because the liquid nature of the breeder material permits to avoid 14-MeV-neutron-induced damages which are quite difficult to characterise, and to perform tritium extraction outside of the vacuum vessel.

Of course, such blankets are not free from feasibility issues, such as the feasibility and reliability of insulating layers or tritium permeation barriers. In this respect, further design optimisation and R&D efforts are still required before being able to perform segment-tests in ITER and to ensure reliable operation under DEMO conditions.

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Table I : Selected DEMO specifications

Major radius [m]	6.3
Minor radius [m]	1.82
Aspect ratio	3.45
Plasma current [MA]	20
Fusion power [MW]	2,200
Average neutron wall loading [MW/m^2]	2.2
Average surface heat flux [MW/m^2]	0.4
Operating mode	continuous
Impurity control	double-null divertor
First wall protection	none
Number of TF coils	16
Toroidal magnetic field on axis [T]	6
Number of segments	32 inboard, 48 outboard
Blanket/shield thickness [m]	1.18 (inboard), 1.86 (outboard)

Table II : Neutron balance in dual-coolant Pb-17Li concept

Neutron multiplication, M	1.59 (0.1%)
Tritium breeding ratio, TBR	
outboard segment	0.82 (0.4%)
inboard segment	0.28 (1.2%)
divertor regions	0.12 (1.3%)
TBR for the whole blanket	1.22 (0.4%)

Table III : Thermal-hydraulic parameters for the dual-coolant Pb-17Li concept

Total power deposited in the blanket	2,050 MW
- extracted by Pb-17Li	1540 MW
- extracted by helium	510 MW
Pb-17Li coolant conditions	
- inlet temperature	275 °C
- outlet temperature	425 °C
Helium coolant conditions	
- system pressure	8 MPa
- inlet temperature	250 °C
- outlet temperature	350 °C
Maximum temperature of the structural material (MANET) during normal operation	530°C

Table IV : Design point for an outboard segment in the water-cooled Pb-17Li concept

Total deposited power	33 MW
- Pb-17Li pool	25 MW
- FW box	8 MW
Pb-17Li-pool / water coolant (15.5 MPa)	
- inlet temperature	265 °C
- outlet temperature	325 °C
- maximum velocity	5.9 m/s
- nb. DW poloidal U-tubes	205
FW / water coolant (15.5 MPa)	
- inlet temperature	300 °C
- outlet temperature	325 °C
- average velocity	5.0 m/s
- nb. toroidal tubes	378
Maximum temperature of the Pb-17Li/steel interface temperature	480°C

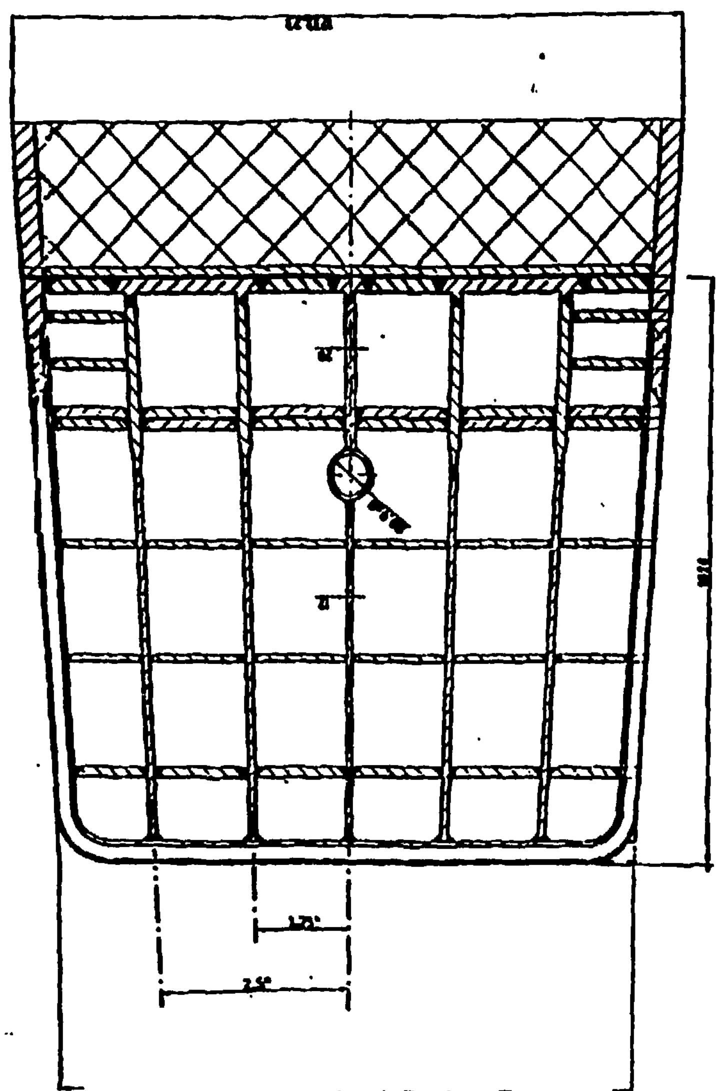

Fig. 1 : Reference Dual-Coolant Concept -
Outboard Segment Mid-plane Cross-section

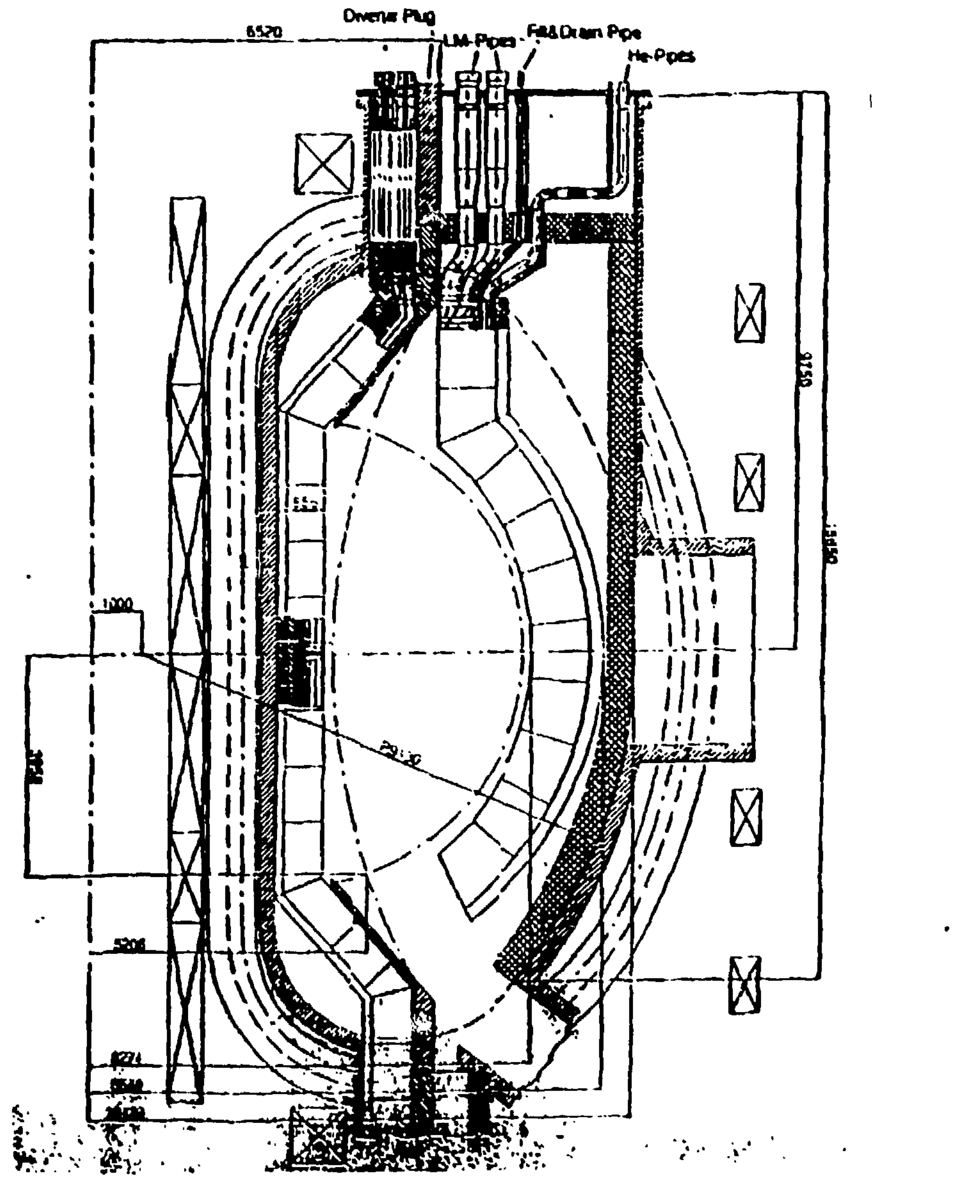

Fig. 2 : - Vertical Cross section of the Reference Dual Coolant Concept

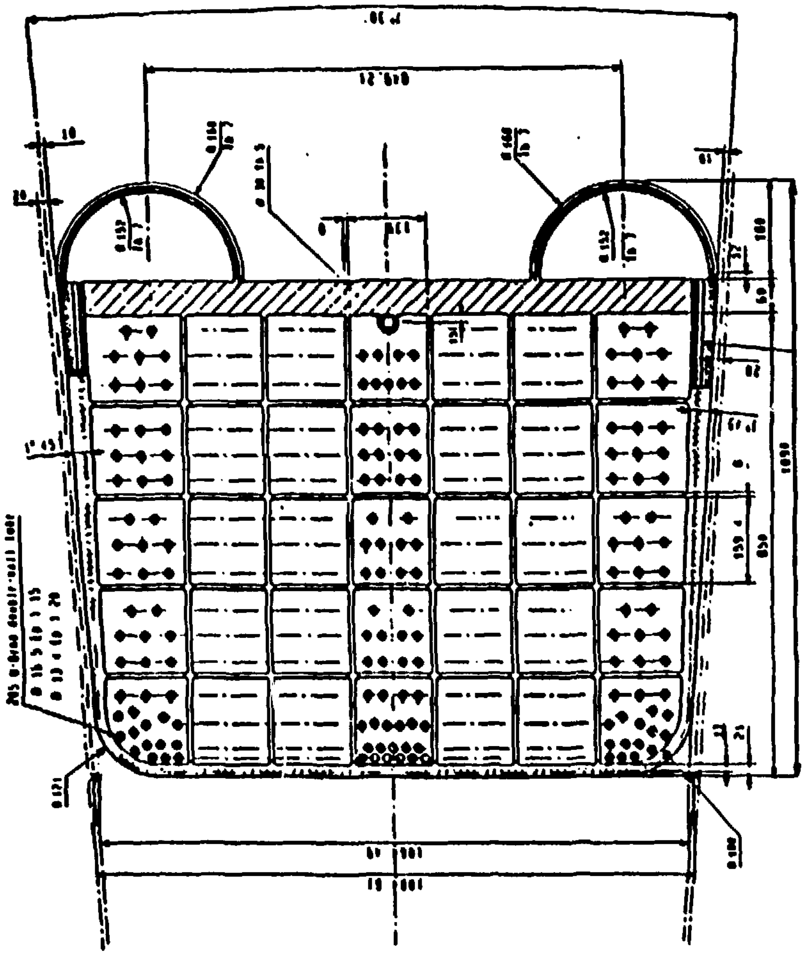

Fig. 3 : Reference Water-Cooled Concept - Outboard Segment Mid-plane Cross-section

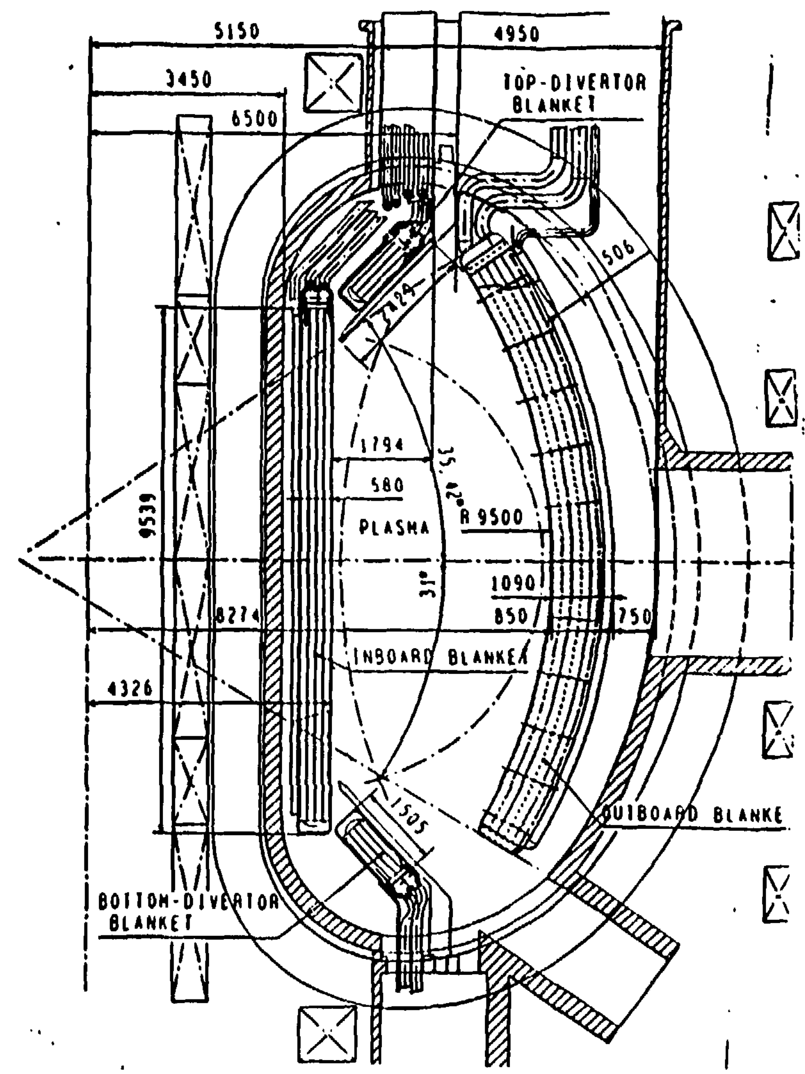

Fig. 4 : - Vertical Cross-section of the Reference Water-Cooled Concept