BREEDING BLANKET for DEMO

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This paper presents the main design features, their rationale, and the main critical issues for the development, of the four DEMO-relevant blanket concepts presently investigated within the framework of the European Test-Blanket Development Programme.

1. INTRODUCTION

DEMO, the fusion demonstration reactor considered in Europe as the overnext step after JET, should contain the full technology of a fusion power plant. In particular DEMO will rely on a "hot breeding blanket" to :

- *convert into heat the kinetic energy of the neutrons* created in the plasma chamber by the fusion reactions

$$D + T \Rightarrow {}^4He\ (3.5\ MeV) + n\ (14.1\ MeV)$$

  and to transfer this heat (80% of the fusion energy) to a coolant under pressure and temperature conditions appropriate for driving a thermodynamic cycle of fair efficiency
- *breed tritium* to replace that burnt in the plasma chamber
- *contribute to the radiation shielding* of the magnet coils

The European Community is engaged since 1989 in a Test-Blanket Development Programme, the purpose of which is :

- in the short term to perform, through design and experimental work, a comparative assessment of the most promising blanket concepts for a DEMO application, with a view to selecting by mid 1995 the two best ones for testing them in NET/ITER
- in the medium term to develop NET/ITER test-blankets, that is test-articles representative of the DEMO-relevant features of the selected blanket concepts

Four candidate blanket concepts for DEMO are being developed within the framework of this programme, primarily at CEA, KfK and ENEA, and, at a lower extent, at JRC/Ispra..

Two of these concepts rely on the use of lithium ceramics as breeder material, of beryllium as neutron multiplier, of helium as coolant, and of martensitic as structural material. They differ essentially by the architecture of their cooling system. One is of the "Breeder Inside Tube" (BIT) type (that is the breeder material is in form of pellets stacked inside tubes, with the helium coolant flowing outside the tubes) while the other one is of the "Breeder Outside Tube" (BOT) type (the coolant flows in tubes, with the breeder material in form of pebbles located outside).

The two other blanket concepts have in common the use of an eutectic of lithium and lead liquid above 235°C -Pb-17Li- as both breeder material (lithium) and neutron multiplier (lead), and of martensitic steel as structural material. One is "self-cooled" (that is the liquid eutectic also serves as coolant) while the other one is cooled by pressurized water.

This paper tentatively presents the main design features of these four candidate DEMO-blanket concepts, their rationale, and the main critical issues for their development.

2. CONSIDERED DEMO SPECIFICATIONS

These blanket concepts are all designed to tentatively meet a set of DEMO specifications (see below) adopted within the framework of the European DEMO-Blanket

Table 1. Considered DEMO Characteristics

major radius (m)	6.3
minor radius (m)	1.82
aspect ratio	3.45
plasma current (MA)	20
fusion power (MW)	2200
mean neutron wall load (MW/m^2)	2.2
surface heat flux (MW/m^2)	0.4 average
impurity control	divertor double null
operating mode	continuous
number of disruptions	1
first wall protection	no
number of TF coils	16
number of segments	48 outboard 32 inboard
inboard thickness blanket + shield (mm)	1176
outboard thickness blanket + shield (mm)	1856
possibility to locate blankets behind the divertor	yes
ports number and geometry (for neutronic calculations only)	10; 3.4 m heigh, full segment width

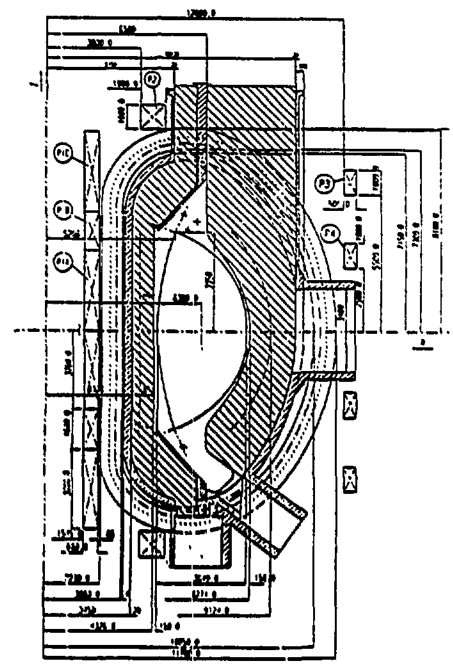


Figure 1. Considered DEMO Geometry

Development Programme for the sake of consistency of the concept comparison/selection studies.

2.1 Considered DEMO Characteristics

For reasons of simplification and convenience, DEMO is considered here as an upgraded version of the next step machine NET having the same dimensions with a higher power and neutron wall loading, and purposely modified to increase the blanket coverage ratio while remaining consistent with the maintenance procedure. This results in the DEMO characteristics summarized in Table 1 and Figure 1.

2.2 Considered DEMO-Blanket Requirements

The main requirements specified for the DEMO-blanket proper are:

- Tritium Breeding Ratio (TBR) exceeding unity in 3D neutronic calculations taking into account the DEMO geometrical characteristics including the 10 ports
- coolant conditions as required for electricity production with a net thermal efficiency exceeding 20%
- blanket segment lifetime exceeding 20,000 hours of full power operation (the fluence level corresponding to this lifetime at 2.2 MW/m^2 is given in Table 2)
- use of a structural material having a well established out-of-pile properties data base and known to behave satisfactorily under high fluence (fission) irradiation
- blanket segment resistance to a disruption with rapid reduction of the plasma current (20 MPa to zero in 20 ms) such that, afterwards, the blanket segments may be non operational and deformed but must still be removable through the vacuum vessel chimney

Table 2. Typical neutron fluence (averaged) immediately behind the first wall of the considered DEMO reactor

neutrons with energy > 1.0 MeV	$2.0\ 10^{22}$ n/cm^2
neutrons with energy > 0.1 MeV	$3.6\ 10^{22}$ n/cm^2
all neutrons	$6.2\ 10^{22}$ n/cm^2

3. MAIN DESIGN FEATURES OF THE FOUR CANDIDATE DEMO-BLANKET CONCEPTS

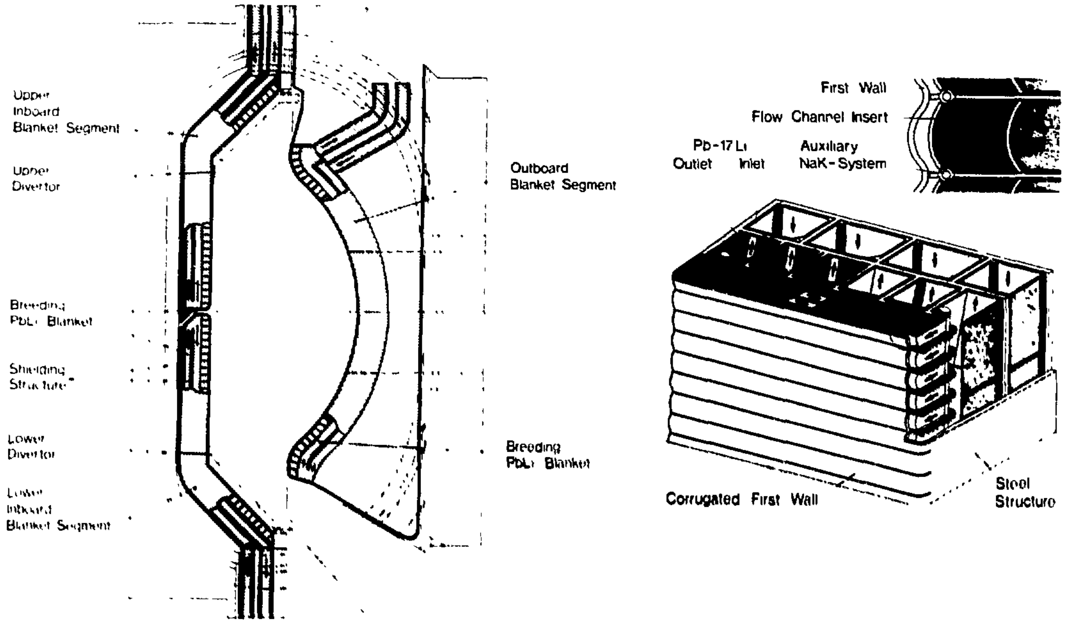
3.1 Introduction

Before presenting the European candidate DEMO-blanket concepts it is worth briefly discussing the basic consequences of the above requirements on blanket design.

Tritium is most-efficiently produced by the reaction

$$^6Li + n \Rightarrow T + {}^4He + 4.8\ MeV$$

and therefore in any blanket concept *tritium breeding is made by irradiating a lithium compound with the neutrons created by fusion reactions.* Attaining a TBR



Arrangement of the Blanket Segments **Cross-Section of a Blanket Segment**

Figure 2. EC self-cooled Pb-17Li blanket concept. DEMO vertical section showing blankets arrangement, and cross section of an outboard segment showing the Pb-17Li flow path.

exceeding unity proves to be difficult (ref. 1). Indeed the production of one tritium atom by the ^6Li (n,α) T reaction consumes one neutron, while its fusion with a deuterium generates only one neutron which exhibits a significant probability (30-35%) of not being available for tritium production (because of parasitic absorbtions in blanket structures, of streaming through the blanket openings, ...). These neutron losses must therefore be compensated for, usually by also *incorporating in the blanket, in addition to a ^6Li-rich compound, a material apt at multiplying neutrons by (n,2n) reactions, like beryllium or lead.*

Furthermore, because of the amounts of tritium to be produced (~100 kg/year), and for safety and starting load procurement reasons, *tritium must be recovered from the blanket and reprocessed on-line.*

Finally, the plant efficiency requirement restricts the type of usable coolant (and its minimum operating pressure and minimum inlet/outlet temperature range) to essentially water (15.5 MPa, 270/320°C), helium (5-8 MPa, 250/450°C) and liquid metal (250/400°C). With such a coolant temperature range, the specified neutron fluence level, combined with the geometrical impossibility to efficiently shield structural welds, *imposes the use of a martensitic steel* as structural material.

3.2 The EC Self-Cooled Pb-17Li Concept

This concept originates in the fundamental idea that the larger the number of different materials used in the blanket, the more complex is its design, and the more complex the design is, the less reliable it becomes. Therefore a relative design simplicity is obtained using a single material, Pb-17Li, an eutectic alloy of lithium and lead liquid above 235°C, to perform all blanket functions except mechanical integrity. Thus Pb-17Li, which is basically both a tritium breeder and a neutron multiplier, is also exploited in this concept for heat transport - (self)cooling- and tritium transport.

Practically, in this concept (ref. 2) the blanket segment is designed (fig. 2) as a thick-walled segment box directly containing the Pb-17Li and cooled by it (including its front wall which is the first wall of the plasma chamber).

This segment box, at the upper end of which liquid breeder inlet and outlet ducts are connected, is equipped

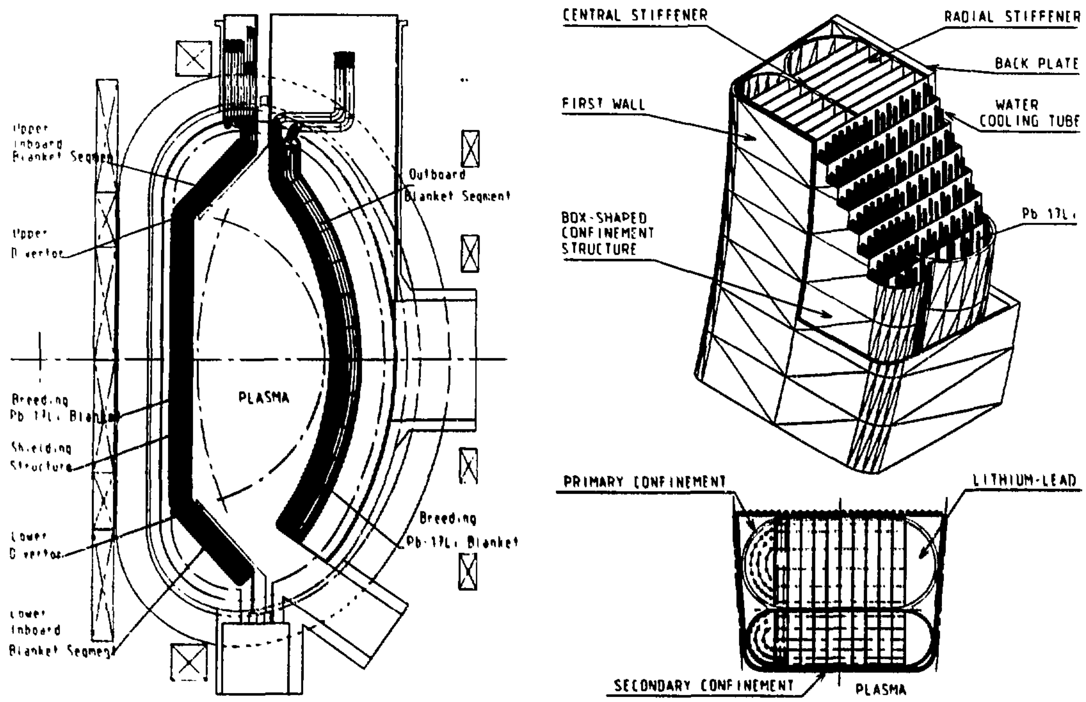


Figure 3. EC water-cooled Pb-17Li blanket concept. DEMO vertical section showing blankets arrangement, and cross sections of an outboard segment

with internal walls which play two roles. First they reinforce this large structure which has to withstand a Pb-17Li normal-operation pressure of 3 MPa or more (a pressurization level required because of the high magnetohydrodynamic -MHD- pressure losses undergone by the metallic eutectic flowing at high velocity orthogonally to the high-intensity toroidal magnetic field of the reactor inside electrically-conducting-walled ducts). Secondly they serve as flow separators. They oblige Pb-17Li to first flow poloidally downwards at the rear of the blanket, and then, after a U turn at the segment bottom end, upwards in the central zone. There the flow separators force Pb-17Li to switch from the poloidal low velocity (~0.5 m/s) flow to a toroidal high velocity (~2 m/s) flow (parallel to the main magnetic field) in the highly heat-loaded first wall region, and then back.

This cooling routing scheme, together with the electrically-insulating liners equipping the poloidal Pb-17Li channels, are required to limit the MHD pressure losses (which otherwise would be prohibitive), while maintaining good cooling conditions (high coolant velocity) for the first wall. To the same aim, on the inboard side where a higher magnetic field prevails, the blanket is splitted in two halves, with the lower half fed in Pb-17Li at its bottom end.

The segment box also integrates a NaK circuit embedded in its walls, the function of which is to preheat the structures and maintain Pb-17Li liquid during long shut-down periods. The heat transported by the liquid eutectic is transferred to the power cycle by means of a double-walled-tube steam generator, with NaK flowing in between the two walls of the tubes.

The tritium produced in the blanket within Pb-17Li is transported in a dissolved state by the eutectic to the steam generator where it is transferred to the NaK (from which it is recovered by cold traps) by permeation through the external wall of the steam tubes. Thus NaK is used both to recover tritium, and to prevent contamination of the steam circuit by trapping tritium and by detecting any leaking crack in one of the two walls of the steam tubes.

As already explained, the main advantage of this blanket concept is the potentially fair reliability of the blanket to be expected from this "relatively simple"

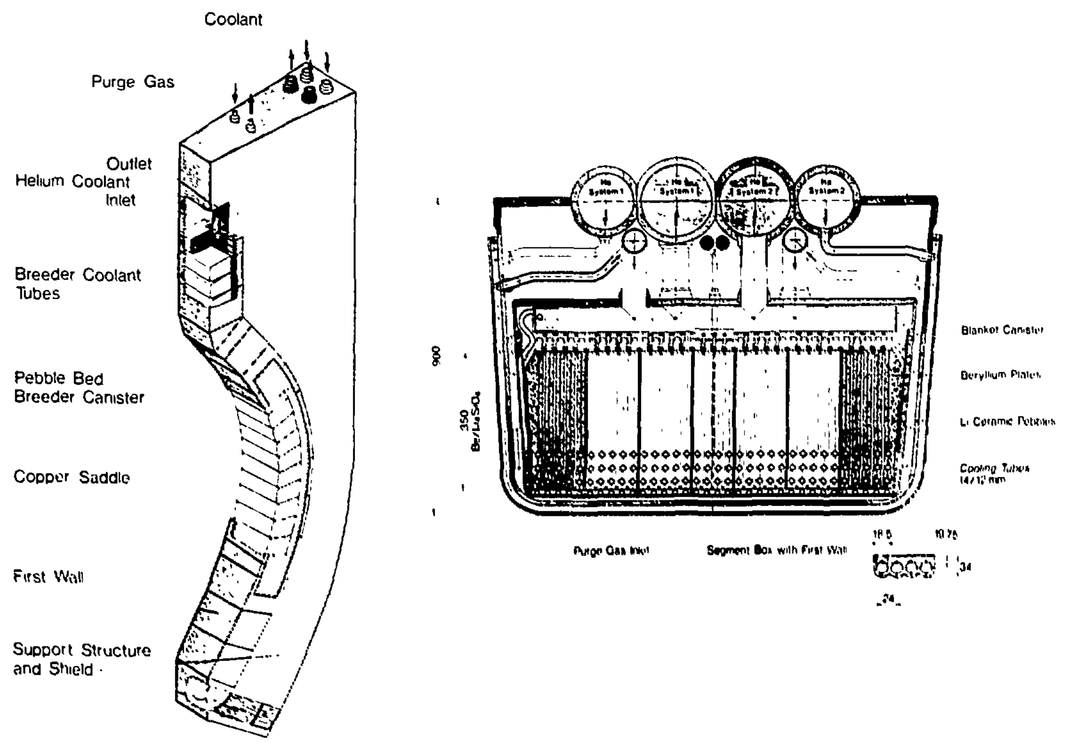


Figure 4. EC helium-cooled ceramic BOT blanket concept. Outboard segment isometric view and mid plane cross section

design. In return, the self-cooling option leads to a complex arrangement of the blanket external circuits and requires the use of a second type of cooling system of completely different technology (with water or helium coolant) for the most loaded components of the reactor (Pb-17Li cooling of the divertor plates being not envisageable). Much more important, it raises the crucial feasibility issue related to the complex and high-pressure-drop-inducing MHD effects, while the use of a chemically-reactive liquid metal as breeder material raises safety issues (see sections 4.2 and 4.3).

3.3 The EC Water-Cooled Pb-17Li Concept

This other Pb-17Li blanket concept based on pressurized-water cooling, was proposed in order to minimize if not avoid any MHD-related feasibility issue, to simplify the blanket external circuits (in particular, but not only, by using a single type of coolant for all components of the reactor), and to ensure double confinement of the liquid eutectic-breeder Pb-17Li vis-à-vis the plasma chamber (safety aspect), and all this without sacrifying too much blanket design simplicity and system reliability.

It is featured (ref. 3) by a monobloc architecture: Pb-17Li is contained inside a martensitic vessel of roughly parallelipipedic shape (fig. 3), which ensures its first confinement; it is cooled by pressurized water (15.5 MPa, 270/325°C inlet/outlet temperature) flowing in an array of poloidal U-shaped cooling tubes. Water is supplied and returned via two water boxes located at the upper end of the Pb-17Li containment vessel.

This vessel is accommodated inside a water-cooled segment box constituting for Pb-17Li a second confinement vis-à-vis the plasma chamber. It is reinforced by radial stiffeners so as to withstand the overpressure (15.5 MPa) resulting from the accidental failure of a cooling tube (the normal operating pressure of the Pb-17Li circuit is on the order of 1 MPa). It is divided into two regions by a toroidally-running steel sheet, the coolant first cooling the front region where the heat deposition is higher, before returning by the back region to the outlet water box.

This cooling scheme makes it possible to reduce the amount of water in the front region of the blanket where it is prejudicial to neutron multiplication (inelastic scattering on oxygen slowing down high energy neutrons, the only

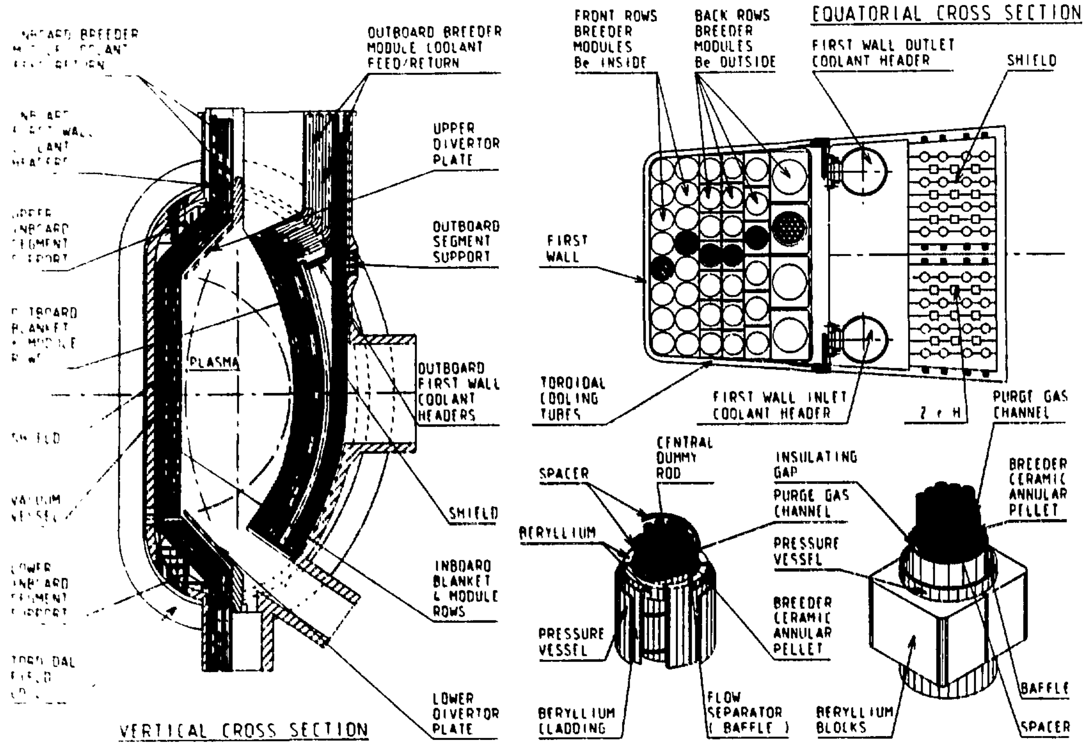


Figure 5. EC helium-cooled ceramic BIT blanket concept. Vertical and mid-plane blanket cross section and details of the front and back row breeder modules

one capable of inducing (n,2n) reactions on lead), while increasing it at the back where it promotes breeding captures in lithium (due to its slowing-down properties).

The function of on-line tritium transport to the extraction unit external to the vacuum vessel is ensured by the liquid breeder which is slowly circulated through the blanket. A recirculation rate on the order of ten times a day only is required. Thus the maximum Pb-17Li velocity in the blanket barely averages 1 cm/s, which leads to a marginal MHD pressure drop.

It is worth mentioning that, in parallel with the design solution here outlined, an alternative design, named "banana-shaped" design has also been worked out in the recent years in the EC. In this design the breeder and the related cooling tubes are contained in cylindrical units arranged poloidally around the plasma inside the segment boxes. The number of modules and their location result from an optimization which objective is to minimize the neutron leakage in the radial direction while achieving a good temperature distribution inside the blanket. On the outboard side these cylindrical modules have a banana shape whereas on the inboard they are straight, disposed along radial rows and extending also into the space available behind the divertor plates.

The main advantage of these blanket concepts is to lead to a design substantially less complex (and therefore potentially more reliable) than those associated with ceramic breeder concepts. It is more complex than the self-cooled one, but avoids the MHD-effects-related feasibily issue. This is however not obtained without raising specific feasibility issues, such as the control of tritium permeation to the coolant, the effect/mitigation of accidental Pb-17Li/water interactions inside the blanket, and the risk of a water spill from the water-cooled segment box into the vacuum vessel (see sections 4.2 and 4.3).

3.4 The EC Helium-Cooled Ceramic Breeder Concepts

Helium-cooled ceramic-breeder concepts are proposed in order to avoid the safety problems associated with the use of chemically-reactive coolants and liquid breeders, and to provide additional margins on the TBR by taking advantage of the need to use a more efficient neutron multiplier than lead: beryllium.

3.4.1 The ceramic BOT concept (ref. 4) relies on the use of breeder modules made-up of nearly rectangular self-supporting canisters mounted the ones over the others on the back plate of a toroidally helium-cooled segment box (fig. 4).

Each canister, whose steel walls are cooled by welded pressure-tubes, contains the neutron multipler material in the form of vertically-arranged beryllium plates separated by gaps. These gaps are filled with a bed of 0.35 to 0.6 mm in diameter lithium orthosilicate (Li_4SiO_4) pebbles through which, for tritium transport purposes, a low-pressure sweep-gas flows. The use of small orthosilicate pebbles avoids the problem of the ruptures caused by the thermal stresses to which ceramics, and in particular Li_4SiO_4, are subjected. Cooling of the Be/ceramics assembly is ensured by helium (8 MPa, 250/450°C inlet/outlet temperature) which, after a preheating through the segment box walls (including the first wall) and then through the canister walls, flows through cooling coils embedded in each Be plate. These cooling coils are connected alternatively to two sets of toroidally-running subheaders routing it to two sets of poloidal manifoldings, thus providing a fully redundant cooling system attractive from the safety point of view. The canisters are furthermore provided with radial stiffening plates so as to withstand the pressure build-up consecutive to the hypothetical failure of a cooling coil..

This general arrangement allows to achieve in the front region of the blanket, despite the use of a low density coolant, a high compactness beneficial to tritium breeding, while the mostly radial coolant routing scheme permits to obtain rather homogeneous temperatures in the canisters despite the large coolant inlet-outlet temperature difference typical of helium cooling and the steep radial power-density gradiants encountered in blankets.

The attractive safety features of this concept, typical of helium-cooled ceramic-breeder concepts, are obtained at the expense of a greater design complexity (compared with liquid-breeder concepts) and of specific feasibility problems all also typical of ceramic-breeder concepts, the most severe of which are likely the high temperature/high fluence behaviour of beryllium (swelling, embrittlement, tritium inventory) and the high lithium burn-up/high dpa behaviour of the ceramic breeder (see section 4.4)

3.4.2 The ceramic BIT concept, this other EC helium-cooled ceramic-breeder concept, is proposed among others as an attempt to minimize the problems associated with the use of a multipler material -beryllium- likely highly susceptible to swelling, and to avoid the uncertainties related to the behaviour of a ceramic breeder material used in form of small pebbles.

This concept (ref. 5) adopts a poloidal modular architecture (fig. 5). Poloidally-running breeder modules of two different designs are associated inside a segment box.

At the back of the blanket, they consist of a martensitic pressure-tube closed at its lower end (thus acting as a pressure vessel) and on the outer wall of which beryllium blocks are either brazed or mechanically attached. This pressure tube contains a bundle of breeder rods surrouded by a baffle. Each one of the rods is made-up of a steel tube containing a stack of annular pellets of sintered lithium aluminate ($LiAlO_2$) or zirconate (Li_2ZrO_3) through which the sweep gas (tritium carrier) flows. The modules are connected at their upper end to a coolant supply and return(coaxial) duct. The coolant, helium at 6 MPa, 250/530°C inlet/outlet temperature, after preheating through the segment box walls, first flows downwards in the annular space between pressure-tube and baffle, reverses direction at the bottom end of the module, and then flows upwards through the bundle.

This coolant routing scheme permits to maintain beryllium and the pressure-loaded structures at moderate temperature, while taking advantage of the heat deposited in these materials for thermally conditioning the ceramic breeder, thus promoting tritium release.

The breeder modules located in the front part of the blanket, that is closer to the plasma, are exposed to a much higher neutron flux. Their slightly different design (with in particular the beryllium blocks located inside the pressure tube in a helium atmosphere allowing to provide gaps for permitting its free expansion/swelling, but also rods of smaller diameter and a larger coolant volume fraction) provides a better accommodation of the higher power densities and fluence prevailing in this region of the blanket. Because this "Be-inside" module design leads to a relatively poor compactness of the blanket detrimental to the tritium breeding ratio, its use is restricted to the first two or three module rows of the blanket in the hope that they will induce enough neutron flux attenuation for permitting to use, behind them, modules more compact but of a more limited swelling accommodation capability.

4. MAIN CRITICAL ISSUES OF THE CANDIDATE DEMO BLANKET CONCEPTS

The conceptual design studies of the four candidate DEMO-blanket concepts described above, supported by a substantial experimental programme launched in the early eigthies (ref. 2, 4, 6, 7), have now reached a stage of realism where, *on the paper*, a Tritium Breeding Ratio (TBR) in excess of unity can be claimed for all of them (TBR ranging from 1.05 to 1.10, as calculated by Monte-Carlo 3D neutron/gamma transport codes like MCNP or TRIPOLI using neutron cross-sections libraries derived from the European Fusion File EFF-1 (ref. 2 to 5)).

These quite encouraging results should however be taken with great carefullness since they are obtained with concepts which all raise severe feasibility problems. The most crucial ones are tentatively presented in the following sections where an attempt is made to distinguish between those which are generic and those which are concept-specific.

4.1 Main Critical Issues Common to All Concepts

4.1.1 Behaviour of the structural material. As already explained in section 3.1, a martensitic steel has been adopted as structural material for all candidate DEMO-blanket concepts, essentially because of its acceptable swelling at high fluence and high temperature, although its higher thermal conductivity and higher strength at elevated temperature compared to austenitics also present advantages. Iindeed, under DEMO operating conditions - 70 displacements per atom (dpa), 250-500/550°C minimum-maximum temperature- titanium-stabilized austenitic steels would cause prohibitive weld/base metal differential swelling problems because of the geometrical impossibility to efficiently shield welds.

One of the most crucial problems associated with the use of martensitic steels as DEMO-blanket structural material is related to the substantial irradiation-induced shift in ductile-to-brittle transition-temperature (DBTT) inherent to such steels. Indeed, under irradiation their DBTT could reach 250°C or more, that is very close to, when not exceeding, the blanket coolant inlet temperature presently considered for all candidate designs. Since this inlet temperature is generally dictated by plant efficiency and hot spot limitation reasons, the solution of this problem is hoped to lie in an appropriate selection of the steel heat or, if pratical, on the implementation of periodic in-reactor blanket annealing procedures.

Another crucial issue is the behaviour of this kind of material under fusion neutron irradiation. Indeed, because of the much harder neutron spectrum resulting in particular in a larger production of gases per dpa (helium, hydrogen produced within steel by (n,α) and (n,p) reactions), this behaviour might differ from the one observed in fast breeder fission reactors (martensitic steel is now being used for the wrapper tubes).

Additional issues worth a special mention include the behaviour under thermal cycling (pulsed operating mode typical of tokamaks and never encountered in fission reactors), the irradiation-induced tritium trapping phenomena (which could cause safety concerns) and welding problems (caused by the "high" thermal conductivity of martensitics) which might be significant for blanket structures whose complex geometry requires a large number of welds.

4.1.2 Forces and stresses caused by plasma disruptions. Successful operation of tokamak power reactors requires the elimination of hard plasma disruptions or at least a drastic reduction in their degree and frequency. If the large plasma current decays in the order of milliseconds, huge eddy currents are induced in the first wall and blanket structure. These currents flow partly perpendicular to the magnetic field, resulting in large forces and torques, and, consequently, in high mechanical stresses in the first wall and blanket structure.

None of the European candidate DEMO-blanket designs appears to have the capability to withstand even a single major disruption as extrapolated to DEMO from the experience of present machines. Although it is obviously impossible presently to predict the intensity of the disruption that the future DEMO will undergo, but it is very likely that the design of its blanket will be significantly influenced by the requirement to withstand it.

In this respect, since the level of eddy currents depends on the electrical conductivity of the blanket segment, it is clear that larger currents and therefore higher mechanical loads will be induced in liquid metal breeder (Pb-17Li) blanket segments than in ceramic breeder ones. However it remains to be seen whether these larger forces are not compensated for by the generally higher mechanical strength of liquid breeder blanket structures, and to which extent the forces are reduced by the electrical insulation of the poloidal Pb-17Li channels tentatively implemented in the self-cooled concept.

Recent publications (ref. 11, 12) indicate however that, in case of the reference disruption scenario considered in the EC DEMO-blanket studies (20 MA to zero in 20 ms) this problem could have solutions. 3D calculations in transient assuming non-ferromagnetic blanket structures (assuming austenitic instead of martensitic steel structures) and 3D steady-state calculations assuming ferromagnetic structures, performed on the ouboard segments of the ceramic BOT and self-cooled Pb-17Li breeder concepts, lead to acceptable segment deformations (segments still removable) provided some parts of the segments are appropriately electrically insulated.

4.1.3 Reliability of blanket and ancillary equipment. Studies based on the quality standards presently achieved by the nuclear fission industry in the manufacturing of tubes, welds and so on indicate clearly that the availability of the blanket system is a crucial problem for all candidate concepts. This is especially true for concepts using a large number of small diameter tubes inside the blanket for heat extraction (that is all concepts except the self-cooled Pb-17Li one), resulting in a relatively high frequency of coolant leaks. Coolant leaks inside the blanket segment generally lead to a long down time of the machine (for replacement of the failed segment) because there is no practical way to provide full redundancy in the heat extraction system.

Some advantage in this respect can be claimed for the EC self-cooled Pb-17Li concept since full redundancy is easier to achieve with concepts placing the heat transfer surface outside the torus.

In the same way, the EC ceramic BIT concept, where the coolant flows outside breeder tubes inside a relatively small number of large modules, could also claim some reliability advantage if normal operation of the blanket could be continued despite the failure of a breeder tube, which is not yet clear (the operating pressure of the sweep gas flowing inside the breeder tubes was chosen close to the one of the coolant in this view). Also in the case of the alternative version of the BOT concept recently proposed (ref. 10, see end of section 4.4.1), operation with a failed coolant tube is possible in principle, as the segment box can sustain the full pressure of the coolant gas. In this case however, the sweep gas flow of the blanket segment in question should be routed to a small tritium recovery system working at high pressure, kept usually in service.

4.2 Main Critical Issues Common to Pb-17Li Breeder Concepts

One of the major disadvantages of Pb-17Li breeder concepts is the high chemical reactivity of the liquid lithium-lead alloy with water (and to a lesser extent with air) due to the presence of lithium. Chemical reactions inducing high temperatures and hydrogen liberation can potentially lead to the release of radioactive products (especially those contained in the liquid alloy) and are therefore a crucial safety issue for this class of concepts.

4.2.1 Chemical reactions between Pb-17Li, water and air. Liquid lithium reacts quite vigourously with water and air. The Li/H_2O reaction, which usually goes until there is no lithium left, can induce a large energy release and the liberation of large quantities of hydrogen. This phenomenon raises safety problems since:

- the energy release may lead to high temperatures endangering the integrity of components and thus causing the release of activated products
- the hydrogen release may lead to an explosive hydrogen-oxygen mixture if there is an air environment

For these reasons the use of liquid lithium in a fusion reactor can be envisaged only if there is no water cooled components adjacent. Thus, considering that water-cooling could be unavoidable for highly heat-loaded components such as divertors, the eutectic lithium-lead alloy Pb-17Li was developed in particular for its property of being a-priori much less reactive than pure lithium (because containing only 17 at.% of lithium).

Experimental tests simulating the real conditions of Pb-17Li/water interactions in case of an accident (large break of a cooling tube) have confirmed this a-priori: the chemical reaction proved to be limited; the solid reaction products (LiOH and Li_2O) partially insulating the melt against water energetic vapour explosions appear unlikely; the pressurization of the reactor vessel did not exceed in general the actual water injection pressure, and the maximum breeder temperature increase did not exceed 100°C at an initial temperature of 350°C.

In view of these results various design measures have been implemented in the two European candidate Pb-17Li breeder concepts to mitigate the consequences of Pb-17Li/water interactions:

- for the water-cooled concept, the Pb-17Li containers in the blanket have been designed to withstand the maximum water pressure under faulted conditions. Thus the water inflow is limited by the rising pressure in the liquid metal.
- for the self-cooled concept, the blanket-segment structures can withstand too the maximum water/steam pressure of the secondary circuit. Additionally double-walled tube steam generators (providing a double barrier between Pb-17Li and water/steam) are used, and the total Pb-17Li inventory is partitioned in a large number of completely separated primary loops (one per couple of blanket segments, or 40 in total) in order to limit the mass of reacting materials.

4.2.2 Activation Products in liquid Pb-17Li. Chemical reactions with liquid metal breeders are a critical safety issue since they can lead to the release of radioactive products, in particular those contained in the liquid metal. Sources of radioactivity in the liquid breeder include tritium, the corrosion products from the blanket structural material, and the activation products of Pb-17Li itself. This third source term, caused by neutron activation of lead and its inherent original or lead-transmuted impurities bismuth, polonium, thallium and mercury, has long been a special concern essentially because of the high radiotoxicity of ^{210}Po (a 138 d α-emitter with a limit of annual intake for the public of 10^{-4} Bq only) and of the high release potential of ^{203}Hg (a 47 d β-γ-emitter). Recent evaluations using improved activation codes and libraries and taking into account the fraction of radiotoxic species released in case of accident, as infered from simulation experiments (and not crudely the total radiotoxic inventory of the eutectic alloy) however show that the early doses produced by ^{210}Po and ^{203}Hg should be lower than those produced by tritium, which themselves are expected to be lower than the recommended limit for the public (0.1 Sv).

Nevertheless it will be essential for the credibility of Pb-17Li breeder concepts to minimize the radioactive inventory in the eutectic alloy by developing and implementing very efficient methods for on-line Pb-17Li purification vis-à-vis activated corrosion products, and for on-line tritium extraction from Pb-17Li.

4.3 Main Critical Issues Specific of the Considered Pb-17Li Concept

MHD effects and control of the tritium permeation to the coolant are the main critical issues specific of the self-cooled and water-cooled Pb-17Li concepts respectively.

4.3.1 MHD effects (self-cooled concept). As already explained, the design of a self-cooled liquid-metal-breeder blanket is dominated by magneto-hydrodynamic (MHD) considerations. Liquid metal flow in uninsulated ducts perpendicular to the strong magnetic field in a tokamak is accompanied with a very high MHD pressure drop which causes a severe feasibility problem because it may result in mechanical stresses beyond the available limits of the structural material. Furthermore, the magnetic field also influences the flow partitioning in parallel channels (the blanket front-region toroidal chanels) and cause velocity profiles in channels completely different from "normal" viscous flow (velocity profiles in cooling ducts influence the heat transfer from the wall to the bulk flow which is in any case degraded by the suppression of turbulence by the magnetic field).

All three problems -pressure drop, flow partitioning, and heat transfer reduction- are crucial issues for the European self-cooled Pb-17Li concept. A number of research programs have been launched to address them. Significant progress has been made during the last years in modeling the flow in single ducts. The pressure drop in single ducts caused by two-dimensional flowing currents is now well understood and can be described with theoretical models but remains a feasibility issue for electrically-uninsulated ducts. Solutions for insulating ducts (flow-channel inserts, ceramic coatings) are being developed but their behaviour and effectiveness under irradiation still remains to be proven experimentally. Simplified theoretical models and preliminary experimental results have been recently produced regarding the multi-channel problem but more detailed models and experiments are required to predict the pressure drop and the flow partitioning in blankets employing the flow concept with first wall cooling in the toroidal direction. This work is going on.

Despite this progress, it is clear that the level of understanding and modeling of MHD phenomena is still far from the one required to produce an engineering design of a self-cooled blanket.

These MHD problems plus the desire to increase safety and availability have lead the designers of the self-cooled concept to recently propose an alternative concept: the "dual coolant" concept (ref. 9). This concept, which relies on toroidal helium-cooling for the first wall and therefore restricts Pb-17Li self-cooling to the blanket segment internals (through large poloidal channels) indeed avoids MHD problems of flow partitioning, minimize heat transfer problems, and promises high safety and availability features due to a leak-tolerant design. The blanket segment itself is even more simple than the purely self-cooled one but the required two types of cooling systems with completely different technology increases the operating complexity unless helium-cooling be used for other components too.

4.3.2 Tritium permeation to the blanket coolant (water-cooled concept). In order to avoid the feasibility problems associated with MHD phenomena, the Pb-17Li flow velocity inside the blanket has been chosen minimal for this concept, that is about ~5 mm/s leading to a liquid breeder residence time in the blanket of 2 hours. Even with the very low tritium concentrations in Pb-17Li at blanket inlet potentially available with the efficient tritium extraction methods under development, such a residence time, combined with the low solubility of tritium in this alloy, creates a substantial tritium partial pressure in the blanket. If no specific measure were implemented, and with cooling tubes exempt of oxide layer, it would result a tritium permeation rate towards the water coolant (through the steel cooling tubes) on the order of 100 g/day. Considering the restrictions imposed for safety reasons on the maximum tritium concentration in the coolant, the recovery of only one hundredth of this amount appears feasible from the viewpoints of coolant detritiation plant size and cost. Reduction of the tritium permeation to the coolant (by a least two orders of magnitude) is therefore a feasibility issue for the European water-cooled Pb-17Li concept.

Reduction of tritium permeation to the water due to the natural formation of an oxide layer at the tube/coolant interface during operation can be expected from fission reactor experience, in particular that of CANDU steam generators. However there are doubts about the resistance of this layer under fusion reactor conditions (irradiation, thermal cycling) and anyway the observed reduction factor does not meet the requirement. Therefore tube coatings acting as permeation barriers are being developed.

The three main types of coating presently under investigation are aluminides, titanium carbide and ternary oxides. Aluminide coatings, widely used in the aerospace industry for anti-corrosion purposes, are known to quite well withstand thermal cycling, and have been shown to be highly compatible with Pb-17Li and, more important, to reduce permeation by three to four orders of magnitude in an out-of-pile gas environment. However the efficiency of such coatings under irradiation and (moderate) thermal cycling, in presence of Pb-17Li (aluminides) and over the DEMO-blanket lifetime (20,000 full power hours) still remains a major uncertainty.

In this respect, some positive hints are given by the recent and very preliminary interpretation of short-duration irradiation tests of Pb-17Li-316 steel capsules equipped with aluminide coatings, and by the ex-Soviet-Union known to have developed hydrogen permeation barriers efficient under long term irradiation for their ZrH-moderated nuclear space-power-reactors.

4.4 Main Critical Issues for the Helium-Cooled Ceramic Breeder Concepts

The two European candidate ceramic breeder concepts are much closer to each other than the Pb-17Li breeder concepts, so that they basically suffer the same crucial feasibility problems. Only the extent of these problems varies from one concept to the other depending on the considered issue, as discussed below.

4.4.1 High temperature irradiation behaviour of beryllium. A fairly extensive data base on the properties of irradiated beryllium was generated during the 1950's and later years. However, these data were essentially obtained in programmes carried out in support of its use as a reflector and moderator in *low-temperature-cooled (<100°C)* material-test reactors. So that in fact very little is known of the irradiation behaviour of beryllium in the operating temperature range typical of helium-cooled ceramic-breeder DEMO-blankets (250-550°C+). Furthermore; these experiments exhibit a great variability in their results, indicating that the properties are dependent on the material purity and processing parameters as well as the irradiation and test conditions. They are therefore even less relevant since they were made on the material available at that time, that is on hot pressed blocks of relatively impure Be which had low ductility and anisotropic properties, while modern improved fabrication methods provide now more ductile and isotropic materials. Nevertheless, indications can be found in this data base according to which the high temperature high fluence swelling, irradiation-induced embrittlement and tritium retention of beryllium could prove to be major feasibility issues for this class of concepts.

♦ *high temperature high fluence swelling.* At the end-of-life of a DEMO-blanket, the total helium production within beryllium, essentially by $^{9}Be(n,2n)2\alpha$ reactions, corresponds to about 15,000-20,000 appm in the most irradiated blanket region. Low temperature (<100°C) long term irradiation followed by high temperature annealing tests, as well as high temperature short term (<1500 appm He) irradiation tests have shown the high susceptibility of hot-pressed relatively impure beryllium to helium-driven swelling. When tentatively extrapolated up to DEMO end-of-life helium concentrations, the results of these tests lead to swelling rates which can exceed 10% in volume at temperatures above 500°C. Such swelling rates would raise feasibility problems for any blanket design concept. Thus, the ceramic BOT blanket, which relies on Be plates brazed on steel cooling-tubes for heat transfer, would fail at a swelling rate of a few percent only. Even the ceramic BIT blanket design, although it was specifically designed to leave Be pieces free to expand in the most irradiated (front) zones of the blanket, could likely not accommodate Be swelling rates much in excess of 10 vol.%.

♦ *irradiation-induced embrittlement.* Past experiments have shown that hot-pressed beryllium was severely embrittled at all temperatures up to 700°C following short term irradiation (1/10th the DEMO-blanket fluence) at low temperature (<100°C). If irradiation-induced embrittlement were excessive, Be cracking may occur in the blanket, producing a decrease in the thermal conductivity of the Be pieces and resulting in enhanced helium-driven swelling, and, possibly, disintegration of the material.

Be embrittlement is thus a crucial issue for the ceramic BOT design which relies on the use of radial Be plates on which steel cooling tubes are brazed (indeed these plates, which are stressed by Be/steel differential swelling, ensure an essential heat transfer function). This is also a crucial issue for the ceramic BIT concept, although to a lesser extent since here Be ensures no other functions than neutron multiplication and is left free to expand so as to minimize its mechanical load.

♦ *tritium retention in beryllium.* Over the lifetime of a ceramic-breeder DEMO-blanket, about 2.8 MCi of tritium (2.8 kg or 9 days of production of the blanket) are produced within beryllium, essentially by $^{9}Be(n,T)^{6}He(^{6}Li)$ reactions. Two recent experiments seem to indicate that, if Be is operated at temperatures below 600°C (which might prove mandatory for avoiding prohibitive swelling), a large fraction of this tritium could remain trapped within Be. This trapped tritium, which would thus constitute the main contributor to the total tritium inventory of the blanket, could raise serious safety problems since it would be mobilized as soon as, during a cooling accident, the Be temperature will exceed 600°C.

Until the results of tests under DEMO-relevant conditions are available, the irradiation behaviour of beryllium will therefore remain a major feasibility issue for the European candidate ceramic-breeder concepts. Hopefully Be produced by modern, improved fabrication methods is expected to be less susceptible to irradiation-induced embrittlement and swelling than the material up to now irradiation-tested. For instance irradiation results at moderate neutron fluence and temperature have shown that the "modern" beryllium becomes less brittle under irradiation.

If the acuity of these problems were nevertheless confirmed, they might be solved, at least in theory, by using Be in form of a pebble bed instead of massive blocks. Indeed the relative "fluidity" expected of the bed offers the prospect to more easily accommodate large swelling rates and to limit the consequences of the fracturation of a low ductility beryllium. Furthermore the small dimensions of the Be pebbles eliminate the problems of Be differential-swelling- and differential-thermal-expansion-caused stresses and should facilitate the release of gases (helium and tritium) and therefore reduce both swelling and tritium retention. The pebble bed option does not however exhibit advantages only: the lower conductivity of pebble beds compared to bulk materials either leads to higher peak Be operating temperatures at BOL and therefore to a higher swelling rate, or requires a significantly larger number of cooling tubes to maintain the the same peak temperature.

These uncertainties on the irradiation behaviour of beryllium have thus led the designers of the ceramic BOT concept to recently propose an alternative concept (ref 10) where Be is used not in form of plates brazed on cooling tubes, but in form of small pebbles mixed with Li_4SiO_4 particles to form an homogeneous bed surrounding an array of cooling tubes. The bed BOL temperature proves to be higher than in case of plates (680°C against 600°C). It is however expected that by volume swelling, the Be pebbles will deform, increasing their contact surface and consequently the bed thermal conductivity, thus leading to a decrease of the bed temperature checking the swelling. Thermal experiments with mixed beds of ceramics/aluminium pebbles and preliminary calculations indicate this effect could be very strong (temperature reduction from 680°C to 500°C in the region of maximum neutron-fluence/ Be swelling) if the ductility and/or thermal-creep rate of Be were high enough.

4.4.2 Behaviour of ceramic breeders at high burn-up/ high dpa. The data base on the out-of-pile and low-fluence-irradiation behaviour of the ceramic breeders developed within the framework of the European Test-Blanket Programme ($LiAlO_2$, Li_2ZrO_3, Li_4SiO_4) has been substantially enlarged over the last years. The properties of the materials now available appear to meet design specifications for beginning-of-life DEMO-blanket operating conditions in terms of tritium release, thermal conductivity, tensile strength, ... The extent of the irradiation-induced degradation of these properties at End-Of-Life (EOL) fluences however remains a major uncertainty. Degradation of the tritium release performances is a special concern because these performances have been shown to be quite sensitive to the ceramic microstructure and phase (stoechiometry), two characteristics likely to undergo substantial modifications over the blanket lifetime due to the high lithium burn-up and high dpa levels achieved at EOL (25 to 40 dpa and 10 to 25% burn-up respectively). In this respect, it is clear that the candidate ceramic breeder of the BIT concept, $LiAlO_2$ and Li_2ZrO_3, because of their relatively low lithium content, will have to withstand a significantly higher Li burn-up than the one of the BOT concept, Li_4SiO_4. However it remains to be seen whether this higher burn-up is not compensated for by the higher chemical stability of these two materials.

4.4.3 Tritium control. The helium-heated Steam Generators (SG) of power plants require a tubing operated at relatively high temperature (250/450-500°C) and exhibiting a large heat exchange area. Consequently they are highly permeable to tritium. Thus typical permeability figures turn around 30 kCi per GWth per day and per $Pa^{1/2}$ of tritium gas (for a coolant containing pure helium plus T_2 only and inconel tubes exempt of oxide layer). Meeting the safety requirements on the maximum tritium losses to the steam circuit (likely on the order of 10-20 Ci per day) therefore appears a feasibility problem for any helium-cooled blanket concept. The solution of this problem lies in an efficient control of the chemistry of the coolant and of the sources of tritium contamination of the coolant.

♦ *control of the coolant chemistry.* It is clear that the sources of tritium contamination of the coolant cannot be limited to the level of 10-20 Ci/d where tritium extraction from the coolant would not be necessary. The economics of tritium extraction will require to operate at a substantial tritiated-species concentration in the coolant (say at least a few tenth of ppm). Such a concentration level, even in a SG equipped with efficient permeation barriers, would lead to acceptable losses only if most of the tritium is present in form of non-permeating tritiated water. This could be achieved by tritium oxidation (either by passing the coolant through an oxidizing bed, or by adding a small amount of oxygen or water to the coolant) associated with a preoxidation of the SG tube external surface (for limiting tritiated-water reduction).

♦ the *control of coolant contamination sources*, that is essentially of tritium permeation through the first wall and through the sweep-gas-circuit envelop, should aim at releasing the requirements on the coolant chemistry control by reducing the amount of tritium to be recovered from the coolant, may necessitate the use of permeation barriers. The control of the latter source, which is clearly of much higher magnitude for the ceramic BIT concept because of its extended sweep-gas circuit, may also be achieved by using a sweep-gas containing a small amount of water vapour in order to convert all the tritium-gas released by the breeder into non-permeating tritiated water.

5. SUMMARY AND CONCLUSIONS

The European Community is engaged since 1989 in a Test-Blanket Development Programme, the purpose of which, for the present 6-year period, is to perform, through design and experimental work, a comparative assessment of the most promising candidate blanket concepts for a DEMO application. The objective is to select by mid 1995 the two best ones for testing them in NET/ITER.

Four candidate DEMO-blanket concepts are being investigated within this framework: two liquid metal breeder (Pb-17Li) concepts, one self-cooled, the other one water-cooled, and two helium-cooled ceramic breeder concepts. The conceptual design studies of these four candidate concepts, supported by a substantial experimental programme launched in the early eigthies, have now reached a stage of realism where, *on the paper*, a Tritium Breeding Ratio in excess of unity can be claimed for all of them.

These quite encouraging results should however be taken with great carefullness since they are obtained with *concepts which all raise severe feasibility problems.*

The ongoing programme tentatively addresses the most crucial of these problems to an extent which, due to the restricted budget available, should barely permit to perform within 3 years a well-grounded selection of the concepts to be further developed up to in-ITER testing. To progress in the demonstration of the feasibility of achieving tritium self-sufficiency in the fusion power plant will require much more important technological efforts.

Over the past 40 years, fusion research has been focused on plasma physics. Progress has been such that breakeven and ignition are now within hand reach. In the future the emphasis will therefore have to be put more and more on addressing the thechnological problems associated with obtaining electric power from controlled thermonuclear fusion.

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