FR8503544

COMMISSARIAT A L'ENERGIE ATOMIQUE

CENTRE D'ETUDES NUCLEAIRES DE SACLAY
Service de Documentation
F91191 GIF SUR YVETTE CEDEX

CEA-CONF -- 7914

L5

STATUS OF FUSION REACTOR BLANKET EVALUATION STUDIES IN FRANCE

CARRE, F.; CHEVEREAU, G.; GERVAISE, F.; PROUST, E.

CEA, CEN Saclay, IRDI, DEMT

Communication présentée à : 6. Topical meeting on the technology of fusion reactor

San Francisco, CA (USA) 3-7 Mar 1985

STATUS OF FUSION REACTOR BLANKET EVALUATION STUDIES IN FRANCE

F. CARRE[1], G. CHEVEREAU[2], F. GERVAISE[1] and E. PROUST[1]
Centre d'Etudes Nucléaires de Saclay
Département des Etudes Mécaniques et Thermiques
[1]Service d'Etudes des Réacteurs et de Mathématiques Appliquées
[2]Novatome/Framatome
91191 GIF-sur-YVETTE CEDEX (FRANCE)
(16-6) 908.54.04

ABSTRACT

In the frame of recent CEA studies aiming at the evaluation and at the comparison of various candidate blanket concepts in moderate power conditions ($P_n \sim 2$ MW/m^2), the present work examines the neutronic and thermomechanical performances of a water cooled $Li_{17}Pb_{83}$ tubular blanket and those of a helium cooled canister blanket taking advantage of the excellent breeding capability of composite Beryllium / $LiAlO_2$ (85/15%) breeder elements. The purpose of the following discussion is to justify the impetus for these reference concepts and to summarize the state of their evaluation studies updated by the continuous assimilation of calculations and experiments in progress.

I. INTRODUCTION

Within the framework of the Blanket Technology Study Group formed in 1981 for two years by Euratom, a joint study of Tokamak Reactor Blanket Evaluation[1] was conducted by the Commissariat à l'Energie Atomique (CEA) in collaboration with the european laboratories of Jülich (FRG), Mol (B), Petten (NL) and the Joint Research Centre of Ispra (I). The objectives of this programme were the identification of key technological limitations for various blanket design concepts, establishment of a basis for assessment and comparison of the design features of each concept, and the recommendation of a few reference solutions. The approach used involves a screening review and tentative classification of previously proposed blanket designs, the discussion of the respective merits of each considered solution according to the choice and respective arrangement of the breeder/coolant couple (breeder in or out of tube) and to the general orientation of the cooling lines within the blanket (poloïdal, toroïdal or radial). A consistent set of very few reference blanket concepts is finally proposed as a basis for the further engineering investigations and as guidelines for the recommendations of specific and prioritary actions for the European Fusion Technology programme.

The purpose of the following discussion is to justify the impetus for the reference concepts and to confirm their attractive breeding potential, thermomechanical performances and tritium cycle, prior to recommend their adaptation to the Next European Torus (NET).

II. SCREENING REVIEW AND TENTATIVE CLASSIFICATION OF BLANKET CONCEPTS

A. Design Parameters

A neutron wall load of 2 MW/m^2 and a surface heat flux to the first wall of 0.2 MW/m^2 are assumed as reference design parameters. These rather conservative values, derived from the characteristics of a 1200 MW power reactor (table 1) proposed in 1981 by the Department of Controlled Fusion of Fontenay aux Roses, are dictated by the consistency with the aim of a 10 years lifetime and of an integrated first wall load not exceeding 20 MWy/m^2.

TOK α^1	$R = 10$ m $\quad B_o = 7$ T $\quad \beta = 2\%$ $a = 3$ m $\quad B_\theta^o = 1$ T
300 MWth LIMITER (10% of FW AREA)	$P_\alpha = 0.5$ MW/m^2
300 MWth FIRST WALL	
2400 MWth NEUTRONS	$P_n = 2.0$ MW/m^2
600 MWth ENERGY MULTIPLICATION	$P_\gamma = 0.5$ MW/m^2
3600 MWth=3000 (FUSION POWER)+600	$P_W = 3.0$ MW/m^2

Table 1
TOK α^1 Commercial Reactor General Characteristics

In the absence of any satisfactory material for a power reactor, 316 stainless steel is selected. Acceptable operating conditions, consequently require the temperature not to exceed 450 and 550 °C for swelling reasons, respectively in the front region and within the bulk of the blanket, and the primary stress not to exceed the yield stress of 316 SS at 500 °C (130 MPa if annealed and 400 MPa if 20% cold worked and not welded). An appropriate blanket design and coolant routing scheme is necessary to make these requirements compatible with the use of pressurized water or helium in working conditions relevant to power production : respectively 15 MPa and [280;320]°C or 5 to 8 MPa and [250;500]°C.

B. Methodology for a consistent analysis of Blanket Concepts

The blanket screening review begins with the consideration of a general matrix of options (table 2), that includes a wide range of possible candidate breeders, coolants, breeder-coolant arrangements and coolant directions.

LIQUID: LITHIUM, $Li_{17}Pb_{83}$; SOLID: CERAMICS, ALLOYS	x	LITHIUM, H_2O–D_2O, [illegible], HELIUM	x	RADIAL, POLOIDAL, TOROIDAL	x	BREEDER OUT OF TUBES, BREEDER IN TUBES	x	INTEGRATED, SEPARATED
BREEDER x COOLANT x FLOW PATH x ARRANGEMENT x FIRST WALL								

Table 2
Matrix of Various Considered Blanket Design Options

Engineering reflexions and considerations are developed in the Fusion Reactor Comparative Evaluation Study[1], about the candidate breeder/coolant associations and arrangements and about the implications of either orientation of the coolant lines upon the size and the complexity of the manifolding sections. The tentative classification of blanket concepts according to the above design options makes it possible to derive some correlations between the selected options and the blanket performances and hence to identify the limiting factors to be further investigated.

The screewing review converges on the proposal to concentrate the future studies on a few reference concepts listed in table 3, which exhibit promising features with respect to safety, breeding potential, tritium recovery and tractable mechanical designs.

Safety recommendations lead to a priori exclude any breader subject to violent reactions with the various coolants used in the reactor. Therefore, considering lithium as breeder, precludes the presence of water, not only within the blanket but also in the first wall and in the limiter/divertor, what constitutes a drastic technological challenge, given the high thermal load experienced by these front components and the questionable suitability of alternative coolants. Likewise, helium is the exclusively recommended coolant for Li_2O blankets. The potential for an effective full Tritium regeneration in relevant power blanket conditions, and for the sustenance of the reactor fuel cycle is a criteria of paramount importance. Anticipating the degradation of the breeding performances associated with a realistic coverage ratio of 0.8/0.85, leads to only take into consideration blanket solutions featured by local breeding ratios in excess of 1.3. Both complementary approaches brought by improved neutronic predictions (heterogeneous 3D Monte Carlo calculations and sensitivity analysis[4]) and by a more accurate assessment of the breeding performances[10] required by the Tritium cycle, are necessary to evaluate blanket solutions featured by a breeding potential comparable to the minimum required threshold.

[illegible]	[illegible]	COOLANT	FIRST WALL	[illegible]	[illegible]	[illegible]
[illegible]	LITHIUM	LITHIUM	[illegible]	[illegible]		
		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	$Li_{17}Pb_{83}$	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	Li_2O, $LiAlO_2$, Li_2SiO_3	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
					[illegible]	
		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]			[illegible]	
	[illegible]	[illegible]				

Table 3. Possible Arrangements of the Candidate Breeder-Coolant Couples

C. Reference Blanket Concepts

The performances of the selected reference blanket solutions are compared in table 4 according to 15 evaluations criteria of importance for the concept viability and extrapobability to reactor relevant conditions.

BREEDER/COOLANT BLANKET ARRANGEMENT COOLING DIRECTION	LITHIUM/HELIUM, TOROIDAL/POLOIDAL	$Li_{17}Pb_{83}/H_2O$	CERAMICS HELIUM BREEDER PIN BUNDLES, RADIAL	TOROIDAL	CERAMICS/H_2O BREEDER BLOCK, TOROIDAL
INDEPENDENT FIRST WALL	HELIUM COOLED	WATER COOLED	HELIUM COOLED		WATER COOLED
BREEDING POTENTIAL	(BLANKET THICKNESS)	(WATER + PORT EFFECTS)	?	-	!
COMPATIBILITY WITH TEMPERATURE AND STRESS DESIGN CRITERIA	?	-	-	?	
COMPLEXITY OF COOLANT DUCTING SIZE OF THE MANIFOLDS	!	!	?	-	?
CONSEQUENCE OF PRESSURE PIPE FAILURE		?			?
DIMENSIONAL, GEOMETRICAL, AND MECHANICAL INTEGRITY	- CORROSION	- CORROSION	- CANNED BREEDER PINS		? UNCLAD BREEDER
UNCERTAINTY OF TRITIUM RECOVERY	!	!	-	-	?
TRITIUM PERMEATION TO THE COOLANT AND POTENTIAL TO MINIMIZE THE TRITIUM RELEASE	!	?	? DEPENDS OF THE NATURE OF THE COLLECTED TRITIATED SPECIES	-	-
EXTRAPOLABILITY TO HIGHER WALL LOADINGS	-	-	-	?	
ADAPTATION TO A VARIABLE POWER LEVEL	!	!	!	-	?
THERMAL EFFICIENCY PUMPING POWER	?		-	?	
BLANKET WEIGHT	!	?	-	-	-
COMPATIBILITY WITH WATER COOLED LIMITER/DIVERTOR	?	!	-	-	!
SPECIFIC CRITICAL ISSUES	SPECIAL REQUIREMENTS OF THE LIQUID METALS; CORROSION; DETRIMENTAL EFFECTS ON BREEDING; BREEDER CIRCULATION; SAFETY		SPECIAL REQUIREMENTS OF THE SOLID BREEDERS (AT WINDOW); SEPARATE PURGE AND COOLANT CIRCUITS; NEED FOR A NEUTRON MULTIPLIER		

Table 4
Comparison of Reference Blanket Concepts According to Major Evaluation Criteria : satisfactory (!), fair (-), questionable (?)

1. Liquid Blankets. Among the attractives feature of the liquid blankets is the confidence that the tritium recovery is achievable, provided the breeder be efficiently circulated ; the extraction is indeed performed in a separate processing unit, that can be optimized independently of the blanket operating conditions.

The proposed design for liquid blankets (helium cooled lithium or pressurized water cooled $Li_{17}Pb_{83}$) consists of rows of tubular breeder containers wound along the plasma chamber along a half poloïdal contour. The coolant circulates in pressure tubes organized in bundles inside each container and the density of cooling pipes is adapted to the power generation of each breeder row. Compatibility considerations with water cooled limiter/divertor lead to concentrate the investigations on the water/$Li_{17}Pb_{83}$ version.

2. Solid Blankets. The reputed poor breeding capability of helium cooled solid blankets, associated with a low breeder proportion and a high structure content, is significantly improved by the use of composite Beryllium/lithiated ceramics breeder elements (TBR > 1.4). This justifies a reniewal of interest in these solutions, that exhibit in comparison with water cooled concepts, a better adaptation to the use of canned breeder rods and a lesser sensitivity of the tritium recovery to the blanket power generation. Any cooling direction may a priori be envisaged and the selection of the most suitable option (radial canisters or in series cooled toroïdal breeder rows), results from tradeoffs between the amplitude of the requisite breeder temperature range, the maximum allowable pumping power and peak temperature in the front region, and the tolerable complexity of the manifolding scheme. The investigations in this area concentrated so far, on the conceptual study of a canister blanket[7,8], which combines the attractive thermal performances of helium cooling in the radial direction, which minimizes the breeder temperature gradient along a cooling channel, with the promising breeding capability of composite Beryllium/$LiAlO_2$ (85/15%) breeder elements[4].

The advantage of water cooled over helium cooled solid blankets with respect to extrapolability to higher thermal loads and release of the pumping power requirements, is a major incentive to consider them also among the reference solutions. However, the viability of such concepts severely depends on critical issues, that still need to be thoroughly addressed, such as the difficult adpatation to actively cooled canned breeder elements with the 15 MPa pressure contained with an acceptable structure proportion or such as the questionable and little reliable adaptation of the requisite breeder coolant temperature difference (∿ 200 °C) at variable blanket power levels. Most of the concerns associated with the questionable breeder temperature adaptation for a satisfactory tritium recovery, would be alleviated by the development of a solid breeder able to efficiently release the generated tritium at a temperature as low as 300 °C. The engineering implications of these requirements were examined on preliminary designs[6], which tentatively considered internally cooled tube breeder elements arranged in toroïdal assemblies housed in actively cooled tubular containers. Essential data to assess the feasibility of the requisite breeder operating conditions for an efficient tritium recovery, are expected from the solid breeder experimental program conducted in France[15].

III. SELF COOLED BLANKET CONCEPTS

The Fusion Reactor Comparative Evaluation Study[1] investigates the possibility to circulate the liquid blanket (or part of it) to simultaneously remove the heat and the Tritium and tentatively addresses the question, whether or not the MHD pressure drop and the associated stresses on the cooling pipes can be kept tolerable in a realistic blanket design.

Even though the lithium blankets are known for their attractive breeding potential when they are thick enough for the neutronic reactions on ^7Li to efficiently contribute to the Tritium production[4], they inherently suffer from major objections independent of the MHD effects induced by the circulation. These concerns mainly relate to potentially violent reactions of lithium with the air (fire) and with water, which is unavoidably needed to cool the highly loaded limiter and first wall structures. Given its poor heat transfer property, helium is not a suitable coolant for the limiter/divertor, but may be considered for the first wall if the required 5 to 10% pumping power fraction to keep the wall peak temperature below 450 °C can be tolerated[1]. The viability of the lithium blankets therefore drastically depends on the demontration, either that the liquid metals can realistically be used in place of water to cool the limiter/divertor or that multiple confinement barriers are capable to absolutely protect penetrations of pressurized water from reacting with the surrounding tritiated lithium.

The study concentrates on the feasibility evaluation of concepts based on the lithium circulation within steel ducts, as the present experience about the realization, the performances and the lifetime of low conducting composite wall solutions, to reduce the MHD pressure drop, are considered by far unsufficient to be promoted for a realistic blanket design.

The investigation of various possible cooling directions with lithium working temperatures of [250;500]°C, concludes that MHD considerations (pressure drop and associated stresses in conducting pipe wall) are not so critical, that they cannot absolutely be managed by the use of an appropriate design . the coolant routing scheme, not only within the blanket but also along the feed and return sections between the blanket and the magnetic field outer boundary. However, all investigated solutions, that are able to limit the total pressure drop to 10 MPa (consistent with a hoop stress limited to 100 MPa within ducts featured by $2t/D \sim 0.1$) require an extremely complicated arrangement of the cooling tubes.

In addition, all concepts require very large coolant and manifold cross sections (typically 10% of the blanket lateral surface), that have to be carried across the magnetic configuration, and hence imply a prohibitive volume of unproductive and tritiated lithium outside the breeding blanket (typically 20% of the blanket content).

Even more crucial than the pressure drop considerations is the prediction of the velocity profile in large ducts, subject to changes in cross sections and in direction within a time and space dependent magnetic field. The local development of flow blockage or sharp velocity peaks along the tube walls may prevent the efficient heat and tritium removal from some blanket regions.

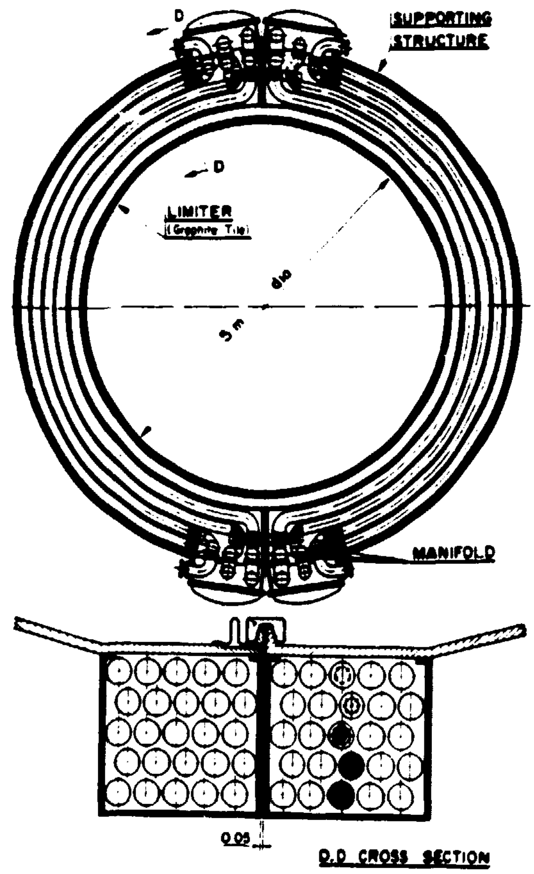
Given the number of identified problems about the use of lithium as coolant and the fact that some are unavoidable and independent of the questionableness of the assumed model for the pressure drop calculations, the self cooled lithium blankets are not presently considered as reference solution.

IV. WATER COOLED $Li_{17}Pb_{83}$ BLANKET CONCEPT[2]

Water cooled $Li_{17}Pb_{83}$ blankets exhibit the interesting characteristics of combining both advantages of a liquid breeder and of water as coolant, provided that an acceptable breeder coolant chemical compatibility be demonstrated. The use of water as coolant provides the lowest possible temperature range (350-450 °C) compatible with the power production and thus alleviates part of the swelling problems observed on stainless steel irradiated at higher temperature. In spite of these interesting features, the feasibility of such a blanket solution strongly depends on the existence of acceptable technological trade offs to accommodate the drawbacks of the $Li_{17}Pb_{83}$/water association : high density and corrosive properties of the alloy, steep power and irradiation gradients, fair breeding capability and probable tritium permeation problems.

A. Blanket Architecture

The proposed design (Figure 1) consists of 5 rows of tubular breeder containers wound around the plasma chamber along a half poloïdal contour (about 10 m long and 12 to 14 cm in diameter). When arranged into a triangular array with a 15 cm pitch, the blanket contains a 30% vacuum proportion (tube interspace). The coolant circulates with a typical velocity of 3.5 m/s in about 1 cm in diameter and 0.75 mm thick pressure tubes organized in bundles inside each container. A constant heat flux of 10 W/cm^2 is exchanged through the tube walls and the coolant density is adjusted to the needs of the heat removal by varying the number of cooling pipes in each row according to figure 2. The table 5 indicates the coolant distribution and the associated breeder and structure proportions derived from the heat generation calculation[3] of a D_2O cooled blanket with 30% enriched $Li_{17}Pb_{83}$ in ^6Li.



Figures 1-2. Schematic Vertical and Horizontal Cross Sectional Views of the Tubular $Li_{17}Pb_{83}$ Blanket Concept

ROW NUMBER	HEATING RATE (W/cm^3)	POWER GENERATED IN A SINGLE TUBE (kW)	NUMBER OF COOLING TUBES	VOLUMETRIC COMPOSITION		
				WATER	STRUCTURE	BREEDER
1	14.5	1955	40	23%	14%	64%
2	5.	681	27	8%	10%	82%
3	2.	293	19	4%	9%	87%
4	1.	130	9	2%	7.2%	91%
5	0.5	67.5	5	1.2%	7%	91.8%
AVERAGE	4.0	627	20	8%	9.5%	92.5%

Table 5. Characteristics of the Breeder Rows

It is essential to note that more than 60% of the blanket power output is generated within the first row and that consequently more than 60% of the total water content circulates within the first breeder row. The detrimental consequences of this situation upon the breeding performances and the tritium permeation to the coolant will be later analysed.

The first wall consists of an independent water cooled panel set in front of the breeder rows. No attempt was made to integrate the first wall into the first breeder row for improving the breeding performances, since this solution would further increase the thermal load of the first row and would remove the ultimate protection against a possible contamination of the plasma chamber by $Li_{17}Pb_{83}$.

B. Impetus for the Design

1. Choice of Pressurized Water as Coolant. The choice of light or eventually heavy water (for neutronic reasons) makes the power production compatible with a satisfactory low temperature range (350-450 °C) within the steel structure. This intermediate temperature field is most desirable to limit the corrosion and permeation problems, that are anticipated with the eutectic alloy. In addition, it alleviates the swelling problems and the stresses associated with the steep irradiation gradients (differential swelling) of the first wall region. Additional advantages of water relate to the low pumping power fraction (0.3%) and to a low coolant proportion (about 8%), that permits visualizing compact blanket solutions with higher breeder proportions than 80%.

The main concern however, about the use of pressurized water are the consequences of a cooling pipe failure and the risk of in series breeder container failure. However, the study of the dynamic response of such tubular modules subject to the sudden rupture of a pipe pressurized at 8 MPa (NET operating conditions), indicates that a thickness to radius ratio of 5% is sufficient to prevent the failure of the vessel[12].

2. Poloïdal Breeder Rows. The poloïdal cooling direction provides a satisfactory adaptation to the toroïdal geometry with uniform heated length (as opposed to the toroïdal direction) and easy access to the manifolds located at the top and the bottom of the vessel. The direct consequences are a satisfactory modularity and standardization and a handy maintenance of the heat exchanger (plugging of the failed coolant tubes may even be envisaged). The poloïdal direction also provides the largest single pass heated length and consequently minimizes the coolant cross section associated with a given coolant proportion within the blanket : the manifold cross section associated with a 3.5 m/s water velocity does not exceed 5.5x2 m^2 (top+bottom), i.e. less than 1% of the torus lateral surface. Additional arguments relate to the absence of natural convection, that occurs within horizontal (toroïdal) heat exchangers and to the advantages of the quasi vertical direction to meet the requirements of liquid metals : need for a free surface with an inert gas pressure, need for a draining device.

The blanket segmentation into rows of breeder is expected to limit the weight of each independent container for maintenance operations, to limit the consequences of a breeder container failure and also to accommodate the steep radial heat generation and irradiation gradients associated with $Li_{17}Pb_{83}$. The breeder tubes are maintained by actively water cooled grids, that leave them free to react separately to the various stresses (weight, thermal, differential swelling, creep).

C. Analysis of Blanket Performances

1. Breeding Performances[3,4]. One dimensional neutronic calculations[3] have been carried out with the blanket composition of Table 1 and a 2 cm thick first wall composed of 1 cm graphite and 1 cm of an homogenized steel and water mixture in the respective proportions of 70 and 30%. A local Tritium Breeding Ratio (TBR) value of 1.28 is obtained with a 30% enriched in ^6Li breeder and with the use of either light or heavy water throughout the entire blanket thickness (85 cm). The combined use of D_2O in the first wall and first breeder row and of an increased H_2O content (7.5%) in the four last rows leads to an improved TBR value of 1.40 (1.44 with a 60% enrichment in ^6Li).

Detailed Monte Carlo calculations of poloïdal tube arrays[4], emphasize a degradation of the predicted breeding ratio as the representation of the actual heterogeneous geometry gains in accuracy : inter-tube streaming effects and internal tube heterogeneity effects respectively decrease the breeding performances by 3% and 2%. A further reduction by 2% is expected from the use of recent CEA measurements of Lead cross sections[4] in place of the ENDFB IV data.

Given the fact that the γ heating on Lead is an important contribution to the total power generation in the blanket front region, the power distribution exhibits a very steep exponential decrease across the blanket thickness (about 20.exp(-6x) W/cm^3 (according to Table 6), with x being the distance to the first wall expressed in meters). Keeping a uniform temperature rise of the coolant flowing with the same velocity in all cooling tubes, requires adjusting the amount of water proportionally to the steep decreasing power generation ; the increased moderator content of the blanket front region reduces the potential (n,2n) reaction rate on Lead and thus negatively reacts upon the breeding performances. Low power blanket designs proposed for INTOR or NET[11], which can afford cooling the first row with a water content of 8% only, do not face this deleterious effect, which is typical of the DEMO blanket conditions.

2. Thermal Performances. In the conditions of Table 5 (3.5 m/s water velocity and uniform 10 W/cm^2 exchanged flux through the cooling tubes), a satisfactory heat transfer coefficient h=3. W/cm^2/°C is achieved. In the first wall region, the total breeder-coolant temperature rise is kept as low as 30 °C with a 20 W/cm^3 power generation and with a 20 °C contribution within the breeder itself. Due to the high water proportion and to the low exchanged flux, the $Li_{17}Pb_{83}$ maximum temperature does not exceed 350 °C, which is well below the critical corrosion threshold of 400 °C identified by preliminary experimental investigations conducted in France[13].

If temperatures in excess of 400 °C are envisageable, it is interesting to minimize the water content in the first wall region for neutronic purpose. This means operating with the highest exchanged flux compatible with the pressure + thermal stress equal to the design limit of 100 MPa. Elementary mechanical analysis shows, that the optimum thickness to radius ratio for this purpose is 0.3 for a 15 MPa water pressure. This corresponds to both pressure and thermal stresses equal to 50 MPa. The maximum allowable flux on a 1 cm diameter (and 0.15 cm thick) tube becomes 30 W/cm^2 and the heat removal now requires a water velocity of 7.5 m/s (h=5.7 W/cm^2/°C). The water content of the first wall region is reduced to about 14% and the maximum breeder-coolant temperature difference increases up to 90 °C with a 60 °C contribution within the breeder itself. Compared to the situation of Table 5, operating at the upper stress and breeder temperature limits, improves the tritium breeding ratio by 3%.

The confirmation by the corrosion experiments in progress[13], of a breeder temperature restricted to 430 °C for compatibility with a stainless steel structure in quasi static conditions[9], would ascertain the limited extrapolability of the present design to higher wall loadings, as an increasing water proportion in the front row progressively degrades the neutronic performances. It is noteworthy that the viability of the concept in relevant power reactor condition depends on the release of the corrosion limitation by the use of improved structural materials (Ferritic steels, Refractory alloys ?).

3. Tritium Recovery and Blanket Inventory[5]. Like the exponentially decreasing heat deposition within the successive breeder rows (Table 5), the tritium production is very inhomogeneous, so that about 46% of the tritium is bred in the first row (28% and 14% in the second and third row respectively). The breeder circulation must consequently be adjusted, to obtain a uniform tritium concentration within the breeder extracted from the various rows to be processed in the tritium recovery unit ; this velocity regulation also minimizes the permeation to the coolant since it prevents a preferential tritium build up in the first row, which contains 60 % of the total breeder-coolant interface.

The daily circulation of the whole blanket corresponds to average and peak (first row) velocities of 0.15 and 0.4 mm/s, which appear quite reasonable with respect to steel corrosion by $Li_{17}Pb_{83}$.

The tritium inventory and the permeation flux to coolant have been calculated for the first wall and the various blanket components[5], accounting for the trapping effect within the irradiation defects with the assumptions of a daily circulation of the blanket and of a quasi total extraction of the tritium within the processing unit ; the steady state values expressed in Table 6 are obtained in a few ten days (10 days for the first wall and 50 days for the blanket), with a trapping energy E_T=0.85 eV and a non graphite protected first wall characterized by a sticking factor α=5.10^{-3}.

The above assumptions yield a total torus inventory of about 2.5 kg with 80% trapped in the structures. It is important to note, that about 1/3 of the daily tritium production (450 g/d) will permeate to the coolant, due to the low tritium solubility in $Li_{17}Pb_{83}$ and in spite of the low temperature of the water cooled structures. The comparison of the increase of the blanket inventory brought by the use of permeation barriers, with the equilibrium inventory within a tritium recovery unit from water[5] is a clear incentive to limit as much as possible the Tritium permeation to the first wall and blanket coolants. Furthermore, both inventory and permeation can be reduced by accelerating the breeder circulation up to the upper limit compatible with acceptable corrosion rate and breeder inventory within the processing unit.

		FIRST WALL	BLANKET					
		Steel structure	$Li_{17}Pb_{83}$ breeder	Cooling tubes	Breeder tubes	Steel structures	Tube interspace	Total blanket
Volume (m^3)		0.3	475	16	30	50	330	
Temperature (°C)		350	335	310	330	450	330	
Tritium inventory (g)	Mobile	85	180	75	50	70	85	390
	Trapped	370		195	700	700		1075
Permeation to coolant (g/d)		25		310				310

Table 6. Tritium Inventory within the First Wall and the Blanket[5]

V. HELIUM COOLED CERAMICS/BERYLLIUM BLANKET[6,7,8]

The main advantages of helium compared to water as solid blanket coolant are its chemical inertness and the adequacy of a medium pressure (5 to 8 MPa instead of 15 MPa) in most power systems ; this mitigates the consequences of a pressure pipe failure but also limits the amount of structural material to contain the pressure in blanket modules and hence enables visualizing concepts using canned breeder elements and potentially compatible with an acceptable breeding capability. As an offset to poorer heat transfer characteristics than water, helium offers the unique advantage to be consistent with breeder elements contained in an actively cooled cladding, which ensures the geometrical and mechanical integrity of the assembly, which are essential to control the breeder temperature and the tritium recovery over the plant lifetime.

In spite of these interesting features, the feasibility of helium cooled solid blankets strongly depends on acceptable technological trade-offs to accommodate the requirements of an in situ tritium recovery (constraints upon the operating conditions, need for a tritium purge circuit) and to compensate for the disadvantages inherent to helium as coolant. Radial and multipass toroïdal cooling schemes are both identified as possible options[1] to adapt the large inlet-to-outlet temperature rise to the technological limits of the structural materials and to maintaining the breeder working conditions within the temperature range recommended for a satisfactory tritium release[9]. Moreover, the low helium density implies a high coolant volumetric flow rate and results in a significant pumping power and in large coolant cross sections, which respectively degrade the net cycle efficiency already bounded by the helium upper temperature limitation, and detrimentally react upon the breeder filling factor and the size of the manifolds ; short heated lengths typical of radial canisters and toroïdal breeder rows provide some answers to these preoccupations.

Finally, most helium cooled solid blankets exhibit a poor breeding capability ($\sim$ 1.06 local TBR for UWMAK II[14]) in consequence of the above limitations, and the development of such a viable blanket design has been essentially made possible by the derivation from recent CEA neutronic studies[4] of an optimized composite Beryllium/ceramics breeder arrangement capable of excellent neutronic performances.

A. Blanket Architecture

The generic module of the proposed blanket (figure 3) consists of a 20 cm in diameter cylindrical canister equipped with individual coolant and purge circuits and with a separate and actively cooled first wall. The breeding cell contains a 19 pins hexagonal bundle composed of an alternate piling of Beryllium and γ $LiAlO_2$ hollow pellets externally clad and equipped with wire wrap spacers ; the optimization of the breeder composition leads to the respective proportions of 15% γ $LiAlO_2$ (60% enriched in ^6Li) and 85% of Beryllium[4], which acts simultaneously as moderator and neutron multiplier. The excellent thermal conductivity of Beryllium minimizes the radial temperature gradient in the breeder material and makes it possible to consider breeder rods larger than 3 cm in external diameter and hence to minimize the proportion of cladding material and the associated parasitic captures.

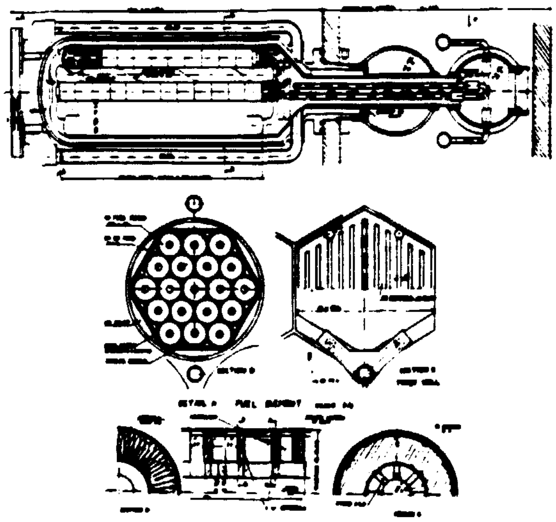


Figure 3. Schematic Layout of a Generic Blanket Canister with Detailed Cross Sections of the Breeder and of the Separate First Wall

A low pressure purge gas enters the central cavity (1 cm in diameter) of each breeder pin, sweeps the grooved surface of the $LiAlO_2$ pellets and is finally collected in large grooves machined in the Beryllium pellets outer surface. The cold helium (250 °C) is directed towards the most heated structures of the front region, through the gap between the hexagonal assembly and the cylindrical cell, thus providing an efficient cooling of the canister outer wall (0.5 cm) ; the coolant then enters the breeding zone at 300 °C and flows backwards through the breeder assembly. A possible improvement of the breeder/coolant exchange coefficient can be expected from the partial obturation of the bundle subchannels by additional structure or Beryllium pins (figure 3).

The generic breeding section of the canister exhibits the following composition : 14% of structure, 27.5% of helium (Inlet, Outlet and Purge), 41.9% of $Be/LiAlO_2$, 13.4% of intercell vacuum and 3.2% of Al_2O_3 as thermal insulator between the helium in and out flows. Individual ducting of the coolant and tritium purge circuits of all cells arranged in close triangular array, leads to a very imbricat d and complicated design of the poloïdal headers, which collect both lines in the blanket lower region, within a separately pumped manifolding chamber.

Pressure drop and pumping power considerations require large pipe diameters (10 to 15 cm), which result in the need for a blanket extra thickness of 30 to 40 cm tc house the manifolds and makes the advisability of the radial cooling direction questionable for any fusion device such as NET, restricted to a blanket thickness less than 1 meter[8].

Erosion, disruption, replacement scheme considerations and reduction of the heat deposition within the nose of the cell, tell in favour of a first wall in the form of a separate hexagonal protective tile set in front of the cell and actively cooled (12 channels shown on figure 3) in parallel or in series with the canister.

The canisters are individually mounted on a supporting structure which leaves them free to react separately to the various stresses (weight, thermal, differential swelling, creep) ; the independent pumping of the plasma and manifolding chambers makes the mounting structure act as a boundary between the primary and secondary vacuum and allows for a non absolute tightness of the canister attachment to the wall[8].

B. Analysis of Blanket Performances

1. Breeding performances[4]. The 3D-Monte Carlo calculation of a generic canister equipped with a 1.3 cm thick first wall (0.3 cm steel + 1 cm graphite) and a 70 cm thick breeder region (85% Be + 15% $LiAlO_2$, 60% ^6Li) yields a tritium breeding ratio of 1.52 which confirms the significant improvement brought by the considered composite breeder composition to a blanket concept usually characterized by poor breeding performances (1.06 local TBR for UWMAK II[14]). Additional calculations show a negligible streaming through the interspace of 90 cm long canisters and a negligible sensitivity (< 1%) to the pellet radius and thickness (no significant shielding effects in ceramic pellets thinner than 0.75 cm). The strong sensitivity of the breeding capability to the structure content is illustrated by the decrease of TBR by 0.75% with each additional 1% of steel, and by 1.4% with each additional mm of the first wall thickness. The satisfactory performances calculated for 1 m thick canisters allow for the replacement of Beryllium by any alternate moderator, in the outer region of the blanket where the multiplication reaction rate becomes negligible. A significant reduction of the allowable breeder thickness results in a drastic drop of the breeding performances owing to the modest breeder filling factor of 45% inherent to the highly segmented blanket concept. It is finally noteworthy that the promising breeding properties result from the breeder composition and that comparable performances could be obteined as well, with blanket designs using other cooling directions or water as coolent, provided the structure content and the breeder proportion be kept acceptable.

2. Thermal Performances. Under a 2 MW/m^2 neutron load, each canister experiences a total heat deposition of 80 kW in the bulk of the cylindrical cell and 14.5 kW in the separate first wall. The restriction to 450 °C of the maximum first wall temperature requires to cool the front tile in series with the breeding cell with the cold helium entering at 250 °C. Meeting the aimed helium working conditions of [250;500]°C assigns the helium mass flow rate to 7.315 x 10^{-2} kg/s, which leads to a pressure drop and a temperature rise across the first wall of 0.02 MPa and 40 °C respectively.

The removal of the 16 kW deposited in the 0.45 cm thick walls and in the nose of the cylindrical cell leads to another helium temperature rise of about 40 °C so that the coolant enters the breeding zone at 335 °C close to the lower boundary of the requisite temperature window for most breeders[9]. The canister outer wall temperature increases from 300 °C in the blanket outer region, to a peak value of 600 °C at the center of the hemispherical section, what is beyond the acceptable working conditions of steel in such a highly irradiated zone and stresses the necessity to efficiently enhance the heat transfer in this critical part of the design. These concerns result in the lack of incentive to consider alternate concepts with integrated first walls, which would be desirable from the neutronic point of view but would be submitted to the additional surface heat load from the plasma (20 W/cm^2 in the present moderate conditions). Elementary mechanical analysis indicates a hoop stress of 120 MPa and a maximum thermal stress of 50 MPa in the canister wall ; these values are consistent with the use of annealed 316 SS, characterized by a yield stress of about 200 MPa at 550 °C.

The table 7 below summarizes the breeder operating conditions throughout the canister for various power levels. A reasonable fit with the calculated volumetric heating rate at nominal power is given by $\omega_o \exp(-\mu x)$ with $\omega_o \sim$ 22.5 W/cm^3 and $\mu \sim 5\ m^{-1}$.

DEPTH INTO THE BREEDER	COOLANT TEMPERATURE	100% Pn Pn=2 MW/m²			50% Pn			25% Pn		
		CLADDING	BREEDER		CLADDING	BREEDER		CLADDING	BREEDER	
x (cm)	T_{He} (°C)	T_c (°C)	T_{out} (°C)	T_{in} (°C)	T_c (°C)	T_{out} (°C)	T_{in} (°C)	T_c (°C)	T_{out} (°C)	T_{in} (°C)
0	336	[illegible]	501	515	482	477	479	465	450	453
10	404	491	502	511	479	485	489	489	472	474
30	469	500	504	506	496	498	499	492	493	496
50	492	503	504	505	501	502	502	500	500	501
70	500	506	506	506	503	504	506	503	503	503
AVERAGE BREEDER T(°C)		505			496			490		
BREEDER ΔT(°C)		[illegible] ± 7			[illegible] ± 18			478 ± 2[illegible]		

Table 7
Breeder Temperature Distribution Throughout a Canister Operating at Various Power Levels

The above results prove the exceptional thermal performances of the proposed design, which combines the advantages of the outwards helium cooling which minimizes the temperature gradient along a cooling channel, and the additional benefit of the excellent thermal conductivity of Beryllium mixed to the breeder, which flattens the radial temperature distribution in 3 cm in diameter breeder rods. It is remarkable that the average breeder temperature sticks closely to the coolant outlet temperature (500 °C) which is about the center of the prescribed temperature windows for most ceramic breeders[9] and exhibits only a 20 °C drift between nominal and 25% of nominal power. The natural adaptation of the concept to ceramic breeders assigned to very strict and narrow working conditions is then largely independent of the power level and provides a valuable margin for any variation in the thermal and heat transfer characteristics over the blanket lifetime. Elementary mechanical analysis indicates a hoop stress of about 200 MPa and a maximum thermal stress of 10 MPa in the cladding ; these values are about consistent with the use of 20% cold worked 316 SS, characterized by a yield stress of about 500 MPa at 500 °C.

The pressure drop in 12 cm in diameter and 12 meters long poloïdal headers used to feed each of the 50 cells connected in parallel with a mass flow rate of 7.315×10^{-2} kg/s is estimated at 2 x 0.01 MPa. A total pressure drop of 0.04 MPa along the cooling lines (canister + headers) requires a pumping power of 0.7% the removed thermal power. This reasonable but not negligible value is the sum of about equal contributions from the headers and the radial blanket, yet characterized by the shortest cooling channel ; this remark *stresses the difficulty to consider for a* power reactor, any toroïdal and a fortiori poloïdal blanket arrangement compatible with a pumping power restricted to 3% of the thermal power.

Pumping power limitations prescribe a maximum neutron wall load of 4.5 MW/m^2 for the considered blanket concept with the present coolant working conditions ([250;500]°C, 6 MPa). The improvement of the heat transfer through the acceleration of the coolant, partially compensates for the increase of the heat deposition and makes it possible to keep comparable temperature working conditions in the various parts of the blanket. Considering higher wall loads would require increasing the helium exit temperature beyond 500 °C or the pressure beyond 6 MPa. Limitations in both directions arise from the necessity to develop advanced alloys for enhanced irradiation and temperature conditions and from the pressure limit beyond which the use of canned breeder elements contained in pressurized modules becomes questionable.

C. Tritium Recovery and Blanket Inventory

The Tritium inventory and the permeation to the primary coolant have been estimated for a supposed 1000 m^3 blanket of a 1200 MWe power reactor. The trapping effect within irradiation defects (1% in atomic proportion) is tentatively accounted for and a total partial pressure of 0.1 Pa in tritiated species is assumed, leading to a volumetric flow rate of about 60 m^3/s in the purge circuit. The table 8 summarizes the results obtained with the supposed recovery of 100% pure T_2 ; the scarce available data do not permit to evaluate the inventory associated with a possible adsorption of T_2O/HTO, which may represent the major contribution to the breeder inventory.

		FIRST WALL	BLANKET (1000 m^3)				
		SEPARATE SS PANEL	$LiAlO_2$ BREEDER	SUPPORTING SS STRUCTURE	INTERNAL SS STRUCTURE	BERYLLIUM	TOTAL BLANKET
VOLUME (m^3)		10	70	30	140	350	
TEMPERATURE (°C)		350< <450	~ 500	~ 500	300< <400	~ 500	
TRITIUM INVENTORY (g)	MOBILE	85	· 1	5.	15.	1.5	21.5
	TRAPPED	370	9	30.	80.	200	310 - [illegible]
PERMEATION TO COOLANT (g/d)		35			190		190

Table 8
Tritium Inventory within Various Components of a 1000 m^3 $Be/LiAlO_2$ Canister Blanket

If all the tritium is recovered as T_2, the above results indicate a total torus inventory of about 1 kg (plus the possible adsorbed inventory) and a very important permeation from the purge circuit to the main coolant loop, on account of the cladding significant surface (50 000 m^2), reduced thickness (0.05 cm) and high temperature (500 °C). The recovery of almost 50% of the daily bred tritium from the 6 MPa helium circuit is clearly not acceptable on account of the excessive inventory of the processing plant able to extract such a high *tritium flow rate at the low concentration* (a few 0.1 Pa) required to restrict the tritium releases to 50–100 Ci/day, with the anticipated helium leaks from the primary circuit (0.025 to 0.05% per day).

In this extreme situation, the viability of the concept depends on the capability to reduce the total inventory of the blanket system to an acceptable level, through the use of permeation barriers or of an oxidising atmosphere in the purge stream.

The assumption of respective proportions of 90% T_2O/HTO and 10% T_2 in the tritiated species collected by the purge stream, brings the permeation flux down to 60 g/d but possibly increases the adsorbed inventory. This results in better balanced contributions from the blanket and the first wall and in comparable inventories in the processing plants associated with both helium circuits. The still high daily tritium leakage to the primary coolant calculated with 90% of the tritium oxidized in the purge circuit, proves the permeation to be a critical issue of the helium cooled breeder in tube blankets, featured by a large thermal exchange surface at about 500 °C. This justifies a posteriori the design options of a separate purge circuit with low partial pressures of tritiated species (∿ 0.1 Pa) and sweep gas, and stresses the potential benefit to be expected from permeation barriers or from a more complete tritium oxidation in the purge circuit, if this is proved feasible and not likely to excessively increase the adsorbed inventory at the breeder surface.

VI. ASSESSMENT OF THE REQUIRED BREEDING POTENTIAL

Previous studies[10] developed a synthetic model for the Tritium circulation between the breeding blanket, the plasma chamber, a storage unit and various reprocessing plants to recover the Tritium from the breeder, from the primary coolant and from the plasma exhaust. This simplified simulation is hereafter applied to assess the minimum breeding performances required for each considered blanket concept to effectively sustain the fuel cycle of the associated fusion reactor, under given assumptions about the equilibrium inventory in the various Tritium units and about the permeability of the primary circuit. The minimum required breeding potential for a doubling time of 10 years is evaluated in the table 9 below, with the optimistic assumption that no other tritium losses occur than the radioactive decay. These values are compared with the best today available estimate of the breeding capability (heterogeneous 3D Monte Carlo calculations), accessible to both optimized blanket concepts : water cooled $Li_{17}Pb_{83}$ with D_2O in the front region and helium cooled $Be/LiAlO_2$ (85/15%) canister. The uncertainty about the neutronic calculations associated with the defective knowledge of the nuclear data (mainly ^7Li $(n,n'\alpha T)$, $^6Li((n,\alpha)$, $\sigma_{tot},\sigma_{el})$, $Pb((n,2n),\sigma_{inel})$, $Be(n,2n))$ is evaluated with the help of sensitivity analysis techniques such as sensitivity profiles and covariance matrices[4].

	[illegible] $Li_{17}Pb_{83}$ [illegible]		[illegible] $Be/LiAlO_2$ [illegible]	
	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	0.25 day	2.15 kg (2)	0.25 day	2.15 kg
[illegible]	0.5 day (1) " 7 days	3.15 kg	15 days (P_{T_2} ∿ 0.115 Pa P_{T_2O} ∿ 0. Pa)	0.5 kg
[illegible]		(3)	100 days (4)	10. kg
[illegible]		6.8 kg		1. kg
[illegible]	99.3	10.1 kg	99.2 (5)	17.05 kg
[illegible]	1.0[illegible] + ε		1.0[illegible] + ε	
[illegible]	1.30 ± 0.08		1.[illegible] ± 0.08	

(1) Complete Tritium extraction from a daily circulated breeder
(2) Assumed fractional burn-up of 5%
(3) Permeation barriers are assumed fo fully inhibit the permeation to the primary coolant
(4) Assumed permeability of the primary circuit *of 300 g/d$\sqrt{Pa}$*
(5) Assumed total allowable tritium leakage of 100 Ci/d

Table 9
Comparison of Required and Calculated Breeding Potential for Both Considered Blanket Concepts

Assuming that realistic values for the effective coverage ratio range from 0.80 to 0.85, makes it possible to derive for each proposed concept, the available margin to compensate for additional breeding requirements associated with effects neglected in the table 9. This margin respectively amounts to 2.9-9.3% and to 15.2-22.4% for the proposed liquid and solid blankets ; it may be considered as an offset to the possible penalty brought by the recovery of an increased proportion of the bred Tritium from the primary coolant (if the control of the Tritium permeation proves to be difficult) or by the non radioactive Tritium losses inherent to the reprocessing operations (and particularly those associated with the plasma exhaust reprocessing, which are amplified by the number of recyclings required by the low fractional burn-up). A more in depth comparison of the breeding potential of both concepts is beyond the scope of the present study and should account for detrimental effects to be anticipated as the designs gain in realism ; such effects could arise from the design of the first wall, from the incentive to deal with a single coolant (H_2O or D_2O), from the necessity to increase the water proportion in the front region of the $Li_{17}Pb_{83}$ blanket, and from the need for an increased steel proportion, particularly detrimental for the Be/Ceramics blankets.

VII. ENGINEERING STUDIES FOR THE NEXT EUROPEAN TORUS (NET)

A. Adaptation of Reference Blanket Concepts to the NET Boundary Conditions

The figure 4 illustrates a possible adaptation of the CEA water cooled $Li_{17}Pb_{83}$ blanket to the NET boundary conditions. The number of breeder rows is respectively decreased to 2 and 3 within the 0.35 m thick inboard and 0.65 m thick outboard blankets. Such a moderate neutron wall loading as 1.3 MW/m^2 makes it possible to decrease the water proportion to 7% in the first row and still respect the temperature limitation dictated by corrosion. Furthermore, a slight decrease of the coolant working temperature ([270;300]°C) permits to limit the water pressure to 10 MPa and thus to reduce the structure proportion. Unidimensional neutronic calculations predict a global breeding ratio of 1.18 with a full coverage, with light water as coolant and with a 30% enrichment of $Li_{17}Pb_{83}$ in ^6Li.

A possible adaptation to NET of the helium cooled canister blanket is shown on figure 5. The design involves a total number of 16 000 generic cells arranged with a variable pitch ranging from 0.15 to 0.20 m. The poloïdal headers of both helium circuits are housed within an actively pumped manifolding chamber which acts as intermediate vacuum boundary and limits the consequence of a pipe failure. Typical helium working conditions such as 6 MPa and [250;400]°C, are expected to provide quasi iso-

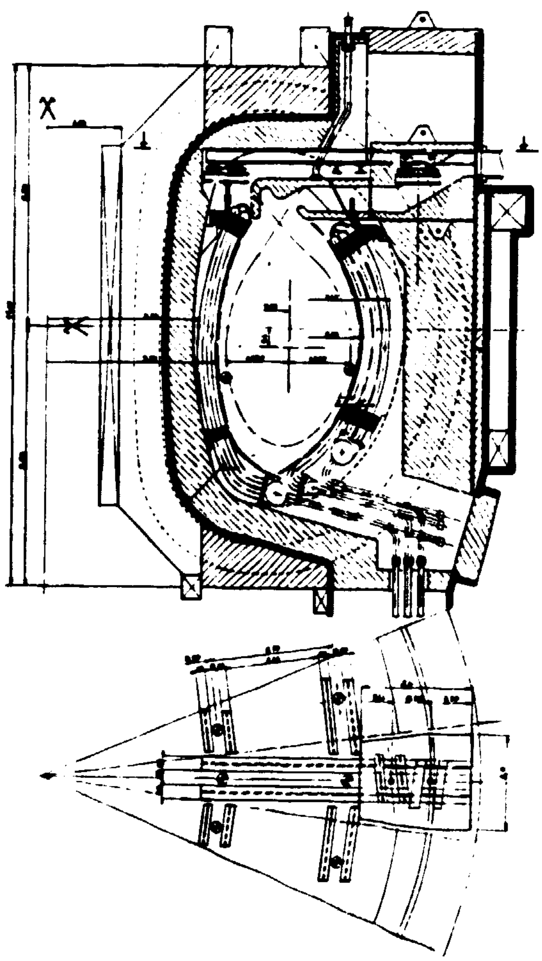


Figure 4. Adaptation to NET of the Proposed Water Cooled $Li_{17}Pb_{83}$ Blanket Concept and Principle of the Segment Handling Scheme

thermal breeder operating conditions about 420 °C, suitable for the tritium recovery from most ceramic breeders. Considering for NET the use of a canister blanket on the outer side of the torus only, where enough space is available to realistically house radial blanket concepts, leads to a minimum breeding potential of 1.08, with a full outboard coverage and a 0.5 m thick $Be/LiAlO_2$ breeding zone. Recently developed alternative designs based on in series cooled toroïdal breeder rows, exhibit a better adaptability to blankets of restricted thickness, and rise beyond 1.25 the achievable breeding potential of NET with a full coverage.

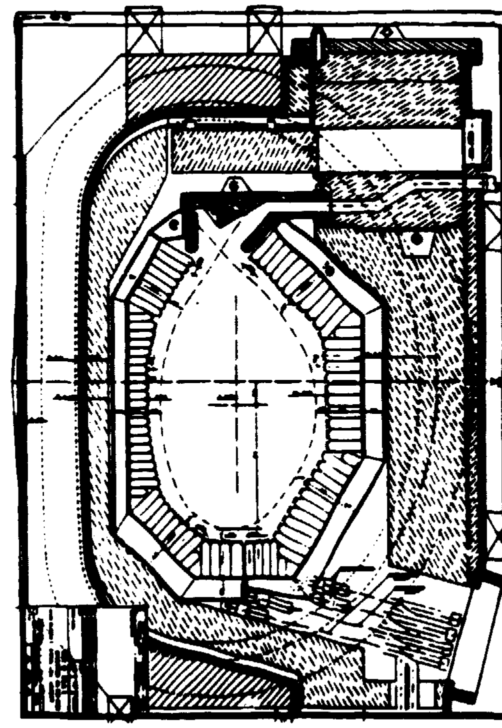


Figure 5. Adaptation to NET of the Proposed Helium Cooled $Be/LiAlO_2$ Canister Blanket

B. NET Blanket Dismounting and Handling Scheme

Other engineering studies specifically devoted to NET relate to the developement of a maintenance scheme exclusively based on vertical and horizontal displacements to dismount and to handle alternatively inboard and outboard blanket segments through the 8 top access doors of the NET II-B version (divertor located at the top). The combination of permanent and removable tracks, enables a rolling handling device to take up position above each of the 6 accessible blanket segments from a single access port, whilst the coolant supply lines concentrated at the bottom of the plasma chamber are remotely disconnected from underneath through special openings. Even though somehow complex because of the numerous requisite steps, this handling scheme exhibits both advantages of relying on already developed techniques and of enabling the removal of larger blanket segments than authorized by purely radial translations, under the constraint of given poloïdal magnet number (16) and size.

VIII. CONCLUSION

The present work, performed in the frame of the European Fusion Technology Programme, examines the potential interest of two blanket concepts recommended among others, as guidelines for future engineering studies by the initial Fusion Reactor Blanket Comparative Evolution Study[1]. The proposed $Li_{17}Pb_{83}$ blanket combines the advantages inherent to the liquid breeders, with that of water cooling ; the suggested design in the form of internally cooled poloïdal breeder rows is intended to accommodate the specific requirements of the liquid metals and those of the steep heat deposition within $Li_{17}Pb_{83}$ power blankets. Conservative assumptions about the temperature limit for corrosion prove a minimum breeding potential of 1.3 with a full coverage to be accessible with such moderate power conditions as $Pn \sim 2$ MW/m^2. The major identified critical issues for this design relate to the blanket weight, to the necessity to efficiently inhibit the permeation to the coolant and to the consequences of a 15 MPa pressurized cooling pipe failure. The proposed solid blanket combines the attractive thermal performances of helium cooling in the radial direction, with the promising breeding capability of composite Be/$LiAlO_2$ breeder elements. The considered canister design provides quasi isothermal breeder working conditions (500±20 °C) and thus brings a valuable margin to offset any change in thermal and heat transfer characteristics, that could remove part of the blanket from the requisite temperature range of a satisfactory Tritium recovery. Breeding performances in excess of 1.45 with a full coverage and a typical pumping power fraction of 1% are proved to be compatible with such a moderate wall loading as 2 MW/m^2. The major identified critical issues for this design relate to the size and complexity of the poloïdal manifolds, to the great number of tubes and connections, which questions the system reliability, and to the significant Tritium permeation from the purge circuit to the primary coolant. Both proposed blanket designs are in process of adaptation to the NET European Torus specific operating and geometrical conditions.

Future engineering studies will include the assessment of the potential benefits to be expected from alternate concepts such as helium cooled lobular[9] or multipasses toroïdal blankets or from water cooled systems. Essential information for the guidance of these studies is expected from the soon available results of the European $Li_{17}Pb_{83}$ and breeder ceramics experimental programs.

IX. REFERENCES

1. F. CARRE, E. PROUST, J. REMOLEUR, A. ROCABOY and Z. TILLIETTE, "Fusion Reactor Blanket Comparative Study," EMT/SERMA/BP/84/584, CEA (1984).

2. F. CARRÉ, Z. TILLIETTE, J. REMOLEUR and E. PROUST, "Conceptual Study of a Lithium Lead Eutectic Blanket for a Power Reactor," Proc. 5th Topical Meeting on the Technology of Fusion Energy (1983).

3. F. BARRÉ, F. GERVAISE and L. GIANCARLI, "Fusion Reactor Blanket Neutronic Studies in France," Proc. 5th Topical Meeting on the Technology of Fusion Energy (1983).

4. F. GERVAISE and L. GIANCARLI, "Progress in Neutronic Analysis of Fusion Reactor Blanket," Proc. 13th Symposium on Fusion Technology (1984).

5. E. PROUST, "Analysis of the Tritium Inventory and Permeation in a $Li_{17}Pb_{83}$ Blanket Proposed for INTOR and Extrapolation to a Power Reactor," Proc. 13th Symposium on Fusion Technology (1984).

6. G. CHEVEREAU, "Solid Breeder Blanket Concepts," Proc. Workshop on Fusion Blanket Technology (Erice, 1983).

7. F. CARRÉ, G. CHEVEREAU, F. GERVAISE and E. PROUST, "Conceptual Study of a Helium Cooled Ceramics/Beryllium Blanket for a Power Reactor," Proc. 13th Symposium on Fusion Technology (1984).

8. G. CHEVEREAU, "Adaptation to NET of a Beryllium Canister Blanket Concept," Proc. 13th Symposium on Fusion Technology (1984).

9. M. ABDOU, C. BAKER & al., "Blanket Comparison and Selection Study," ANL/FPP-84-1, Argonne National Laboratory (1984).

10. F. CARRÉ, E. PROUST and A. ROCABOY, "Analysis of the Tritium Requirements for a Power Reactor," Proc. 5th Topical Meeting on the Technology of Fusion Energy (1983).

11. G. VIEIDER, W. DAENNER, B. HAFERKAMP, "Design Concepts for NET First Wall and Blankets," Proc. 13th Symposium on Fusion Technology (1984).

12. H. Th. KLIPPEL, "Dynamic Response of INTOR/NET Blankets after Coolant Tube Rupture," Proc. 13th Symposium on Fusion Technology (1984).

13. T. FLAMENT & al., "Corrosion of Structural Materials on Flowing $Li_{17}Pb_{83}$," Progress Report on Task B6 of the European Fusion Technology Programme, CEA/12718/PF, Fontenay-aux-Roses.

14. R.W. CONN, G.L. KULCINSKI and C.W. MAYNARD, "The UWMAK-II Study : a Conceptual Design of a Helium Cooled, Solid Breeder, Tokamak Fusion Reactor System," Nuclear Engineering and Design 39, 5-44 (1976).

15. B. RASNEUR, "Tritium Breeding Material : γ $LiAlO_2$," Proc. 6th Topical Meeting on the Technology of Fusion Energy (1985).