FR8602348

2

COMMISSARIAT A L'ENERGIE ATOMIQUE

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

CEA-CONF -- 8511

L5

COMPOSITE BERYLLIUM-CERAMICS BREEDER PIN ELEMENTS FOR A GAS COOLED SOLID BLANKET

CARRE, F.; CHEVREAU, G.; GERVAISE, F.; PROUST, E.
CEA CEN Saclay, 91-Gif-sur-Yvette (France). IRDI, DEMT

Communication présentée à : Technical committee and workshop on fusion reactor design and technology
Yalta (USSR)
25 May - 6 Jun 1986

COMPOSITE BERYLLIUM/CERAMICS BREEDER PIN ELEMENTS FOR A GAS COOLED SOLID BLANKET

F. CARRÉ, G. CHEVEREAU, F. GERVAISE and E. PROUST
Centre d'Etudes Nucléaires de Saclay
IRDI-DEDR-DEMT-SERMA
91191 GIF-sur-YVETTE Cedex (FRANCE)

1/ LIMITATIONS OF CONVENTIONAL HELIUM COOLED PIN BLANKETS

The main advantages of helium compared to water as coolant for solid blankets, are its chemical inertness and the adequacy of a moderate pressure (5 to 8 MPa instead of 15 MPa) for satisfactory heat transfer properties above 300 °C. This attractive feature not only mitigates the consequences of a pressure pipe failure but also makes it possible to contain the pressure in blanket modules, rather than in pressure tubes, with a reasonable amount of structural material ; helium cooling consequently permits visualizing blanket concepts using canned breeder pins and still compatible with acceptable breeding properties. The actively cooled cladding ensures the geometrical and mechanical integrity of each breeder pin and thus permits an efficient control of the breeder temperature which is essential to minimize uncertainties in tritium recovery over the plant lifetime. Moreover, cooling of the blanket module walls is effected by the same coolant as that of the breeder, and does not require any complicated and little reliable array of tubes brazed on the structure.

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 the 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 to adapt the large inlet-to-outlet helium temperature rise to the technological limits of the structural materials and to maintain the breeder working conditions within the temperature range recommended for a satisfactory tritium release. Moreover, the low helium density implies a high coolant volumetric flow rate and results in large coolant cross sections to mitigate the pumping power requirements ; this, combined with the use of pins, results in increased helium and structure volume fractions and hence in a limited breeder filling factor, that detrimentally reacts upon the breeding performances. Short heated lengths typical of radial canisters and of toroïdal breeder rows, partly relax the pumping power needs and the incentive to larger helium cross sections, but make the ducting of the coolant lines to an increased number of blanket modules more difficult.

In consequence of the above limitations, most helium cooled solid blankets exhibit a poor breeding capability (TBR ∿ 1.06 for UWMAK II [1], TBR ∿ 1.04 for $LiAlO_2$/Be/Ferritic steel concept considered in the BCSS [2]) ; the development of attractive helium cooled blankets based on breeder pin assemblies [3,4,5] has been essentially made possible by the derivation from recent CEA neutronic studies of an optimized composite beryllium/ceramics breeder arrangement, capable of excellent neutronic performances [8,9].

2/ DESCRIPTION OF THE PROPOSED COMPOSITE Be/Ceramics BREEDER ELEMENTS

Two blanket concepts were successively developed by the CEA, with the orientation of the cooling lines optimized to maintain the breeder working within an as narrow as possible temperature range ; this was intended to allow some margin to accommodate variations of the power generation along the poloïdal contour and possible temperature drifts caused by blanket power swings and by changes in thermal properties and heat transfer characteristics under irradiation.

Even though characterized by different module arrangements (radial canister [3,4] and in series cooled tubular rows [5]), both blanket concepts adopt a breeding zone made of hexagonal pin bundles, composed of an alternate stack of beryllium and γ $LiAlO_2$ hollow pellets externally cladded and equipped with wire wrap spacers (Figures 1 and 4) ; the optimization of the bredder composition leads to respective volume fractions of 20 % γ $LiAlO_2$ (60 % enriched in ^6Li) and 80 % of beryllium, which acts simultaneously as moderator and as neutron multiplier.

The excellent thermal conductivity of beryllium minimizes the radial temperature gradient ΔT in the breeder material and makes it possible to consider breeder rods larger than 3 cm in external diameter and hence to minimize the volume fraction of cladding material and the associated parasitic captures ; usual blanket conditions (ΔT < 200 °C, Pn ∿ 2 MW/m^2) would restrict to about 1.5 cm the upper limit for the diameter of pure ceramics rods placed in front position.

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.

Fully heterogeneous 3D Monte Carlo calculations proved both blanket concepts developed by the CEA, to be capable of breeding performances in excess of 1.5 with a 70 cm thick breeder zone [8,9,10] ; this is achieved in spite of such a moderate $Be/LiAlO_2$ filling factor as 42 % and with such high void and structure proportion as 41-46 % and 12-14 % respectively.

The canister blanket design developed for a DEMO relevant blanket (Figure 1), was proved capable of assuming quasi isothermal breeder working conditions with a moderately demanding pumping power fraction (1 %), when exposed to a typical neutron wall loading of 2 MW/m^2. Prospects for the extrapolation to heat loads of 5 MW/m^2 were found encouraging [3]. However the adaptation of this blanket concept to the geometry of NET is unfavourable as the available space for the blanket in the near term device is excessively restricted in the radial direction (35 cm inboard + 65 cm outboard) : the poloïdal helium headers take up a significant volume of the outboard side and the inboard side cannot actually be covered with radial cells.

In order to permit the test of the proposed breeder assembly in NET, an alternative design has been developed, in the form of a toroïdal arrangement of in series cooled tubular modules. The proposed cooling scheme, illustrated on Figure 5, is intended to partly keep some of the advantages of the radial systems, while cooling first the most heated structures of the front region with helium at the inlet temperature and combining the coolant temperature rise with the decreasing power deposition in the bulk of the blanket, to keep the breeder working in narrow temperature conditions.

The next sections of this paper are devoted to the description of the proposed toroïdal blanket layout for NET, to the analysis of its main performances and to a final discussion about the merits of the considered composite Be/Ceramics breeder elements.

3/ ADAPTATION TO NET OF A HELIUM COOLED TOROÏDAL BLANKET CONCEPT, BASED ON Be/Ceramics BREEDER ELEMENTS

3.1. Review of design constraints and technical options

The NET III B version of near term device [6] were adopted as boundary conditions for this study ; the associated design constraints are hereafter briefly summarized and the main technical options for the concept are shortly justified.

Design feature	Comment
NET parameters Major chamber radius : 6.49 m Plasma radium : 1.64 m Plasma elongation : 1.6 β toroïdal total : 4.8 % Neutron wall load (average) : 1.5 MW/m^{-2} Number of TF coils : 16 Outb. blanket thickness : 0.65 m Inb. blanket thickness : 0.35 m	The larger size of NET III compared with NET II (6.49 m versus 5.18 m in major radius) makes it better adapted to the integration of a toroïdal blanket concept, under the constraint of a segmentation into an identical number of sectors. A lesser volume fraction is wasted in the non breeding module end sections and hence an improved breeder filling factor is achieved.
Blanket segmentation 24 outboard sectors and 24 inboard sectors handled through 8 main ports	Blanket and divertor handling procedure was found feasible through the machine top and consistant with the NET integration study conducted in 1984 [7]
Divertor position At the machine top	Consistant with the alternative configuration NET II B [7]
Structural material : SS 316 L	NET reference material
Breeder materials $LiAlO_2$, Li_4SiO_4, Li_2SiO_3, (Li_2O)	Ceramics developed within the framework of the European programme
Coolant : Helium Pressure : 6 MPa Temperature : 250 to 500 °C	The moderate pressure leads to low structure volume fraction. The coolant temperatures are compatible with the breeder and structural material requirements
Tritium extraction : low pressure Helium purge (about 0.2 MPa), separate from the pressurized coolant	Minimization of the amount of tritiated gas to be processed and minimization of the tritium inventory in the processing line
Tritium permeation control Oxidizing atomosphere on the coolant side	The oxidation of the coolant atmosphere is intended to restrict to 100 Ci/d, the tritium losses to the environment. Oxidizing the atmosphere of the purge gas is likely to raise compatibility problems between the breeder ceramics and the metallic beryllium

3.2. Blanket layout

The general layout of the blanket sectors is illustrated on Figure 2 : it consists of a number of in series cooled tubular breeder rows arranged in the toroïdal direction.

The inboard blanket contains 1 row of small diameter tubes (70 mm) and 1 row of larger diameter modules (200 mm), that are 1 to 1.5 meter long. The minimum requirement of 2 breeder rows is justified by neutron streaming considerations ; the difference by a factor of 3 in the module size of both rows is intended to accommodate the steep decrease of the power deposition, as the distance from the plasma increases.

Figure 3 shows the cross-section of a front breeder row. Mixing of the ceramic breeder and of metallic beryllium in respective volume fractions of about 15 and 85 %, is achieved by stacking alternately pellets of both kinds, as indicated in section 2. The purge helium flows in grooves machined in both sides of the ceramic pellets ; it is distributed by 4 outer grooves and collected in the central duct. The pressurized coolant flows between 2 concentric tubes of SS 316 L, held in place by fins or by helical spacing wires ; it is collected at both ends by a triple connection to the headers feeding the back rows.

Figure 4 illustrates the cross section of the 2nd, 3rd and 4th rows : each tubular modules of 200 mm in diameter contains a bundle of 7 cladded composite breeder pins, made of lithiated ceramic pellets alternating with beryllium pellets. As the heat deposition decreases in the bulk of the blanket, the pellet thickness, that control the temperature gradients in the ceramics, increases. Bare rods of beryllium (20 mm in diameter) may be inserted in the breeder bundle to improve the filling of the tubular module. The pressurized coolant flows in the gaps between the container and the 13 rods. Helical wire spacers wound around the breeder pins improve the heat transfer and accommodate the thermal expansion of the bundle.

Figure 5 is an equatorial cross section of the outboard blanket. It illustrates the principle of the helium coolant and purge manifolding :

- coolant and purge gas enter the tubular modules at opposite ends, sweep the breeder pins externally and internally in a countercurrent flowpattern and are finally collected at both opposite ends and ducted towards the next breeder row,
- 3 adjacent tubes of the front row are connected in series with a single tube of each other breeder rows,
- the main headers, that run in the poloïdal direction, are integrated into the back shielding and supporting structure of the blanket,
- the first wall is assembled with the back structure so as to contain the breeder modules in a actively pumped closed box that acts as a secondary vacuum chamber.

3.3. Thermohydraulic performances

Thermohydraulic calculations aim at assessing the average temperature on the axis of each module (assumed axisymmetrical) in steady state operation. Heat transfer calculations are carried out at each end of the 4 breeder rows of the outboard blanket ; poloïdal variations of the wall loading about the average value (1.5 MW/m^2) are neglected.

The separate first wall considered in thermohydraulic and neutronic calculations is assumed cooled by helium in rectangular channels running in the poloïdal direction. The surface heat load is taken equal to 15 W/cm^2 and the bulk heat load equal to 11.3 W/cm^3 in average, over the total wall thickness of 1.5 cm (SS 316 L). A sacrificial layer of 1.5 cm of graphite is assumed, with an average bulk heat load of 12 W/cm^3.

Thermal conductivities of 1 W/cm/°C and 0.02 W/cm/°C are used in the calculations, respectively for the beryllium (400-500 °C) and the lithium ceramics (500-600 °C). The solid-solid gap heat transfer coefficient (Ceramics/Be, Be/SS, Ceramics/SS) is assumed equal to 0.5 $W/cm^2/°C$, as typical values observed at the UO_2/clad interface of fission fuel pins range from 0.5 to 1.1 $W/cm^2/°C$. Nusselt and Colburn laws are used to calculate the convective heat transfer coefficient h :

$$h = \frac{Nu.k}{D} \quad \text{with} \quad Nu = 0.023\ Re^{0.8}\ Pr^{0.4} \quad (Re > 2500)$$

The main results of these calculations are summarized in table 1 and the extreme working temperatures of each breeder row are indicated in Figure 6.

Section	FW		1^{rst} R	2^{nd} R	3^{rd} R	4^{th} R
	C					
P_S W/cm^2	15					
P_V W/cm^3	12	11.3	7.4	4	1.65	0.6
He T_{In} °C		240	250	325	440	484
He T_{Out} °C		435	325	440	484	500
He V m/s		100	83	20	22.5	23.5
Q kg/s		5.4	22	22	22	22
Trans. C^t W/cm^2 °C		0.63	0.5	0.13	0.13	0.13
T_{max} SS °C		550	378	459	496	505
T_{max} Be °C			436	517	510	510
T_{min} Cer. °C			358	394	465	493
T_{max} Cer. °C			619	593	537	520

TABLE 1

Estimation of coolant and breeder temperatures at both ends of the successive tubular rows - Comparison with the recommended temperature range for an efficient tritium recovery from the main breeder candidates, is indicated on Figure 6.

The assumed coolant working temperatures and routing scheme, effectively meet the requisite temperature window [2] for an efficient tritium release from breeder candidates such as $LiAlO_2$ and Li_4SiO_4. Other ceramics such as Li_2SiO_3 and Li_2O would partly work at a temperature below the window threshold ; this would affect about 15 % of the first row content. Means to rise the minimum breeder temperature up to 410 °C (lower bound of the tempereature window for Li_2SiO_3 and Li_2O), include increasing the coolant inlet temperature from 250 to 280 °C or cooling the first wall in series with the breeder rows, which would effect a preheating of the coolant by about 30 °C above the inlet temperature.

The beryllium temperature is moderate and stays below the swelling threshold predicted by the BEESTON model [11] as a function of the fast neutron fluence and of the subsequent helium production. According to this model based on irradiation experiments at low temperature only, swelling would occur beyond 400 °C in the first row, beyond 500 °C in the second row and beyond 600 °C in both rear rows ; the anticipated beryllium volumetric swelling should then be less than 0.5 %.

The helium velocity is increased to 150 m/s in the poloïdal headers, to keep their size reasonable ; however, it is purposely kept moderate in the blanket to avoid flow induced vibrations and excessive pumping power fractions ($\sim$ 1 %).

3.4. Neutronic performances

The tritium breeding ratio afforded by the current section of this blanket, was estimated with a 1D-cylindrical representation of both inboard and outboard blanket segments simultaneously ; $LiAlO_2$, 60 % enriched in ^6Li is selected for the calculation. In these conditions, the estimated breeding capability with full coverage, amounts to 1.21 with respective contributions of 0.95 for the outboard and 0.26 for the inboard segments. Optimizing the shielding design, is expected to somewhat relax the constraint upon the thickness of the breeding zone and to thus improve the breeding performances by a few per cents.

The assessment of the effectively achievable coverage ratio is also of great importance for the estimation of the actual breeding capability. A first estimate of the coverage ratio can be derived from the necessary fraction of the toroïdal length needed for piping connections between the modules of the first row, as illustrated on Figure 5.

The table 2 below, summarizes the degradation of the coverage ratio attributable to the module ducting scheme, for various NET geometries (NET II or NET III) and for various segmentations into toric sectors (24, 32 or 48). Other sources of neutron leakage, such as pumping or heating ports or divertor slots have also to be taken into account.

Geometry		Inboard Blanket	Outboard Blanket	Total Blanket
NET	Segments			
III	24	0.83	0.9	0.88
III	32	0.78	0.86	0.83
III	48	0.66	0.79	0.76
II	32	0.76	0.83	0.81
II	48	0.63	0.75	0.71

TABLE 2

Degradation of blanket coverage ratio as a function of the NET geometry and of the torus segmentation.

The toroïdal blanket arrangement appears to be greatly impaired by the segmentation into 48 sectors, such as that considered by recent NET integration studies. The blanket configurations considered for a satisfactory integration and for a tractable handling scheme, rank in the following order, with respect to their adaptation to the proposed toroïdal blanket concept :

1. NET III A (32 segments)
2. NET II B2 (32 segments)
3. NET II A (32 segments)
4. NET III A (48 segments)

Figure 7 illustrates the second configuration, with poloïdal rows of tubular modules replacing the proposed toroïdal blanket arrangement in the inboard, with a view to decreasing the loss of coverage fraction attributable to the modules end connections.

4/ ATTRACTIVE FEATURES OF THE COMPOSITE Be/Ceramics BREEDER ELEMENTS

4.1. Neutronic properties of beryllium

The exceptionally low energy threshold of the (n,2n) reaction on beryllium (∿ 2 MeV), makes the multiplication accessible not only to 14 MeV neutrons but also to neutrons having already experienced a few collisions. This is especially important for breeder ceramics as inelastic scattering on oxygen results in slowing down most 14 MeV neutrons below the energy threshold of the multiplication reaction of all alternative multiplier materials (∿ 7 MeV for Lead and Zirconium). Moreover neutronic studies proved the substitution of BeO for metallic beryllium to totally spoil the attractive breeding performances of the proposed breeder assembly [10].

Beryllium is also the only neutron multiplier, that also acts as an efficient moderator. The massive use of beryllium in the blanket therefore minimizes the distance covered by the incident 14 MeV neutron until it be slowed down enough to be captured ; this assures a minimum neutron leakage for a given blanket thickness and breeder filling factor.

Neutronic sensitivity studies [7] proved the reemitted neutrons by beryllium to be efficiently captured by iron. The attractive breeding performance of the proposed breeder element is subject to rapid degradation as the volume fraction of steel increases (- 0.75 % on TBR with each additional 1 % of blanket structure and - 1.4 % on TBR with each additional mm of structure in the First Wall [9]. Minimization of parasitic captures in Steel then results in the incentive to gather both multiplier and breeder materials inside the same cladding.

The massive use of beryllium in the blanket also enhances the Fusion energy multiplication factor beyond 1.4.

4.2. Thermal properties of beryllium

The relatively high melting temperature of beryllium ($\sim$ 1280 °C) and the large difference of enthalpy at the melting point ($\Delta H \sim 1300$ kJ/kg), are well adapted to most solid blanket working conditions and totally relax the cooling difficulties associated with change of phase for lead to be avoided.

The excellent thermal conductivity of beryllium ($\sim$ 1 W/m/°C), significantly improves the effective thermal conductivity of the composite Be/Ceramics breeder element and consequently greatly reduces the temperature gradient associated with a given thermal load. No longer the ceramics pellet radius controls the temperature gradient across the rod, as the heat deposited in $\gamma\ LiAlO_2$ is axially conducted to the adjacent beryllium pellet, that acts as a heat sink and flattens the temperature profile. Adjusting the breeder working temperature to the requisite temperature window for adequate tritium release, is therefore easier for composite Be/Ceramics rods than for plain ceramics pins ; the proposed breeder composition consequently permits larger pin diameters and hence reduced structuve volume fraction.

Beryllium is also featured by the highest specific heat among the solids (2.25 kJ/kg/°C) ; the massive use of beryllium in solid blankets minimizes temperature swings and thermal cycling of the ceramics, in case of pulsed operation and significantly increases the thermal inertia of the breeder elements. The later consideration is of particular importance for the safety of gas cooled systems subject to loss of coolant accidents.

4.3. Respective volume fractions of beryllium and ceramics

Neutronic optimization studies of Be/Ceramics breeder elements prove the breeding capability to be maximum for respective volume fractions of 80-85 % of metallic beryllium and 20-15 % of lithiated ceramics (60 % enriched in ^6Li).

However, reduction of the beryllium inventory can be considered with limited impact upon the breeding capability :

- as 80 % of the neutron multiplication reactions occur in a 35 cm thick region in the front of the blanket, the replacement of beryllium (that mainly acts as a moderator beyond this point) by any alternative moderator materials can be envisaged with little degradation of the neutronic performances,
- if the depletion in ^{6}Li and the associated damages are shown to endanger the ceramics integrity, a slight decrease of the beryllium content (e.g. from 80 to 70 %) and a subsequent increase of the ceramics inventory (e.g. from 20 to 30 %), may be expected to partly slow down the burn up integration rate, that is accelerated by a factor of 5 to 6 in the proposed blanket concept, compared with others designs based on separate neutron multiplier and breeder materials.

The tritium breeding ratio exhibits only a slow degradation with the depletion in ^{6}Li (- 0.4 %/MWy/m^{2} with average and peak depletion rates of (- 3.0 %/MWy/m^{2} and - 6.9 %/MWy/m^{2} [11]) respectively).

5/ CRUCIAL ISSUES OF COMPOSITE Be/Ceramics BREEDER ELEMENTS

The main technical issues raised by this blanket relate :

- to the mechanical integrity of the beryllium exposed to the considered temperature and fast neutron fluence conditions,
- to the chemical compatibility of beryllium with the lithiated ceramics, with the cladding material and with the purge gas atmosphere,
- to the possible trapping of the tritium in the irradiated beryllium.

The experimental investigation of these difficulties is a requisite for the thorough investigation of the considered blanket concept and for the assessment of its viability.

5.1. Radiation induced swelling of beryllium

Of crucial importance for the integrity of the proposed breeder element and hence for the recoverability of the regenerated tritium, is the possibility to overcome the anticipated swelling and degradation of mechanical properties of beryllium under irradiation. This can be achieved either through a sufficient design flexibility or through the development of beryllium metallic alloys of improved performances.

According to section 4, only moderate swelling of beryllium is anticipated in the NET operating conditions ; however the predictions of the swelling model should be checked against additional experiments run in proper thermal and fluence conditions. According to [13], the efficiency of porosity in accommodating helium induced swelling in beryllium is uncertain.

Differential swelling and thermal dilatations between the SS 316 cladding and beryllium can be adapted :

- axially by elastic washers made of a copper beryllium alloy or of inconel 718 ; such devices could be used in the first row where the temperatures are low enough and the swelling is the most important,
- radially by plastic deformation of the cladding after closing the interface gap. Mechanical tests are needed to assess the limits of such a deformation.

5.2. Chemical compatibility of metallic beryllium with the blanket environment

The viability of composite Be/Ceramics breeder elements, as illustrated in Figures 1 and 3, is subject to the demonstration of satisfactory behaviour of canned beryllium in the Fusion blanket environment. This includes the demonstration of acceptable chemical compatibility :

- with the breeder, and verification that either the oxidation of metallic beryllium can be inhibited or that the kinetics is slow enough to be tolerated ; of importance is also the assurance that no low temperature melting alloys (such as Be Al Si eutectic) is likely to form,
- with the structure, and inhibition of possible corrosion of austenitic stainless steel caused by the affinity of beryllium for nickel,
- with the atmosphere of the purge gas, containing moisture (T_2O, HTO), hydrogen (H_2,T_2) and possibly O_2 addings. Previous studies [12] already demonstrated the benefit of 0.4 % Ca addings for resistance to oxidation. The possible formation of beryllium hydride (BeH_2 that decomposes at 125 °C) needs to be assessed.

Some of the improvements that could be implemented to overcome the above technical difficulties, include :

- improving the properties of beryllium by adding alloying elements such as Ca, Mg, Ba, Cr, La or Zr,
- using coatings [14] or corrosion protective surface treatments (chromate coating, Berylcote D, Alodin, Iridine, Fluoride coating (0.2 to 1 µm)), or anodized films of Cr, Fe or Mo (2 to 100 µm),
- replacing conventional breeder candidates by $Li_2Be_2O_3$, proved capable of satisfactory breeding performances [10],
- considering ferritic steel as cladding material, to relax corrosion problems associated with the high Nickel content of austenitic steels.

5.3. Tritium trapping in beryllium

Uncertainties in tritium release under irradiation, must also be proved not likely to cause the accumulation of excessive tritium inventory, that would offset the benefit of tailored temperature conditions for enhancement of the tritium release and would raise potential safety concerns.

The applications to the proposed blanket concept, of models of tritium trapping derived from retention and thermal release experiments of deuterium implanted in beryllium [15], yields a total inventory of 75 g at saturation. Moreover, this estimation is expected to be conservative, as the operating temperature of beryllium in the blanket is well above both experimentally identified temperature thresholds, for the successive stages of deuterium thermal release (150 and 400 °C).

The possible formation of beryllium hydride (BeT_2, likely to decompose between 125 and 250 °C as BeH_2) and the consequence upon the blanket inventory and the beryllium integrity need to be assessed.

5.4. Beryllium ressource limitation

Beryllium ressource limitation, was shown by the BCSS [2], not to be so drastic as to exclude the use of beryllium as reference neutron multiplier material for the first and second generations of Fusion reactor service ($\sim$ 1800 and 3000 GWe-y, respectively).

6/ ALTERNATIVE BLANKET CONCEPT

In case the technical issues of beryllium in the fusion blanket environment could not satisfactorily be resolved, an alternative blanket design illustrated on Figure 8, could be considered with probable performance degradation.

The arrangement of the tubular modules in the toroïdal direction are basically kept. However, the breeder pin bundles is redesigned so as to use metallic beryllium and lithiated ceramics in separate pins of the breeder bundle ; the number of pins of each kind is adjusted to keep the optimum volume fraction of 85/15 %. The breeder rods are made of canned hollow pellets, designed with the same sweeping gas routine scheme as in the reference design. The metallic beryllium rods can be either bare or protected by a thin cladding (0.3 mm), depending on the expected corrosion by the impurities of the coolant.

The thermal and breeding performances of this alternative design are less favourable than those of the reference concept due to the degradation of the effective thermal conductivity when the beryllium is not associated with the ceramics and to the parasitic captures in the cladding material, of the reemitted neutrons by the beryllium.

The comparison in table 3 of the material content in the current section of the reference and alternavie concepts shows that the SS volume fraction is not increased and then suggests that the anticipated TBR degradation may be moderate.

	Material	Reference Design	Alternative Design	
			Bare Be	Cladded Be
1st row	SS	12	9.1	11.2
	Be	34	40	38
	Li ceramic	8	5.7	5.7
	P. He	4	1	1
	C. He	9.5	11.7	11.7
	Void	32.6	32.6	32.6
2nd, 3rd, 4th row	SS	13.8	9.2	10.6
	Be	43	44.5	43.1
	Li ceramic	7.8	8.5	8.5
	P. He	2.4	1	1
	C. He	15.3	19.3	19.3
	Void	17.6	17.6	17.6

TABLE 3

Comparison of Reference versus Alternative Blanket Design, with respect to the material content.

7/ PROSPECTS OF BERYLLIUM/CERAMICS BREEDER ELEMENTS

Even though they may not be credible as illustrated on Figure 2, Beryllium/Ceramics breeder elements exhibit definite neutronic and thermal advantages that reactivated in Europe the interest for helium cooled pin blankets, affected so far by breeding performances ranging grom poor to fair. Even though particularly attractive for this type of blanket suffering from a large structure content and from a low breeder filling factor, the inherent advantages of the considered original Breeder/Multiplier assembly can also be extended to any type of solid blanket, irrespectively of the cooling options, provided the structure and the breeder contents be kept acceptable.

As an illustration to the performances of blanket concepts based on composite beryllium/ceramics breeder pin bundles, a tubular blanket is proposed for NET, that consists of in series helium cooled toroïdal breeder rows. The toroïdal orientation of the modules permits a satisfactory integration in the NET blanket segments, with an acceptable loss of coverage (attributable to the module ends), provided the number of toric segments do not exceed 32.

In series cooling of the successive rows not only permits to adjust the length of the cooling path, so as to keep the pumping power fraction reasonable ($\sim$ 1 %) and the manifolding scheme tractable, but also affords partly keeping some of the attractive thermal performances of the radial systems : cooling first the most heated structures of the front region with helium at the inlet temperature and combining the coolant temperature rise with the decreasing power deposition in the bulk of the blanket, to keep the breeder working in controlled temperature conditions for the tritium release.

In spite of the low breeder filling factor and of the large amount of structural material, typical of helium cooled pin blankets, the use of composite beryllium/ceramic elements leads to breeding performances consistent with the objectives of NET (TBR > 0.8 with blanket restricted to 0.65 + 0.35 m) and offers encouraging prospects when extrapolated to DEMO conditions (TBR $\sim$ 1.5 with full coverage and 0.70 thick breeding zone).

Addressing the technical issues of this peculiar breeder element in the European breeder experimental programme is a requisite for the thorough investigation of this blanket concept and for the assessment of its viability.

In case the technical issues of beryllium in the Fusion environment could not be satisfactorily resolved, an alternative design with separate beryllium and ceramic pins could be considered with probable performance degradation.

8/ REFERENCES

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

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

[3] 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).

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

[5] G. CHEVEREAU and L. BARAER, "A Helium Cooled Toroïdal Blanket for NET with Li Ceramic Breeder and Beryllium Multiplier",
CEA-DEMT-SERMA Report Nr 85/718 T.

[6] R. TOSCHI, "Approach to the Definition of NET",
Proc. 13th Symposium on Fusion Technology (1984).

[7] INTOR - Phase II^{A}, Second Part, CRITICAL ISSUES - European Contributions to the INTOR - Phase II^{A} Workshop (Volume I).

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

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

[10] F. GERVAISE, "Etudes neutroniques de couvertures de réacteur à fusion. Calculs complémentaires sur la solution Be/$LiAlO_2$ à refroidissement radial en hélium",
CEA-DEMT-SERMA Report Nr 85/1667 T.

[11] L. GIANCARLI, "Atomic Burn Up Effects on Tritium Regeneration",
UKAEA Presentation at the NET-Blanket Engineering Meeting on Solid Breeder Blankets (Karlsruhe 26-28 Nov. 1985).

[12] G. CHEVEREAU, "Revue bibliographique des propriétés du beryllium en vue de son utilisation dans la couverture de réacteurs à Fusion",
CEA-DEMT-SERMA Report Nr 84/1639 T.

[13] T.J. Mc CARVILLE, "Technical Issues for Be use in Fusion Blanket Applications",
UCID 20319, Jan. 85.

[14] D.R. FLOYD and J.N. LOWE, "Beryllium Science and Technology",
Volume 2, p. 428.

[15] W.R. WAMPLER, "Retention and Thermal Release of Deuterium Implanted in Be",
3rd Meeting on Fusion Reactor Materials, Sept. 1983.

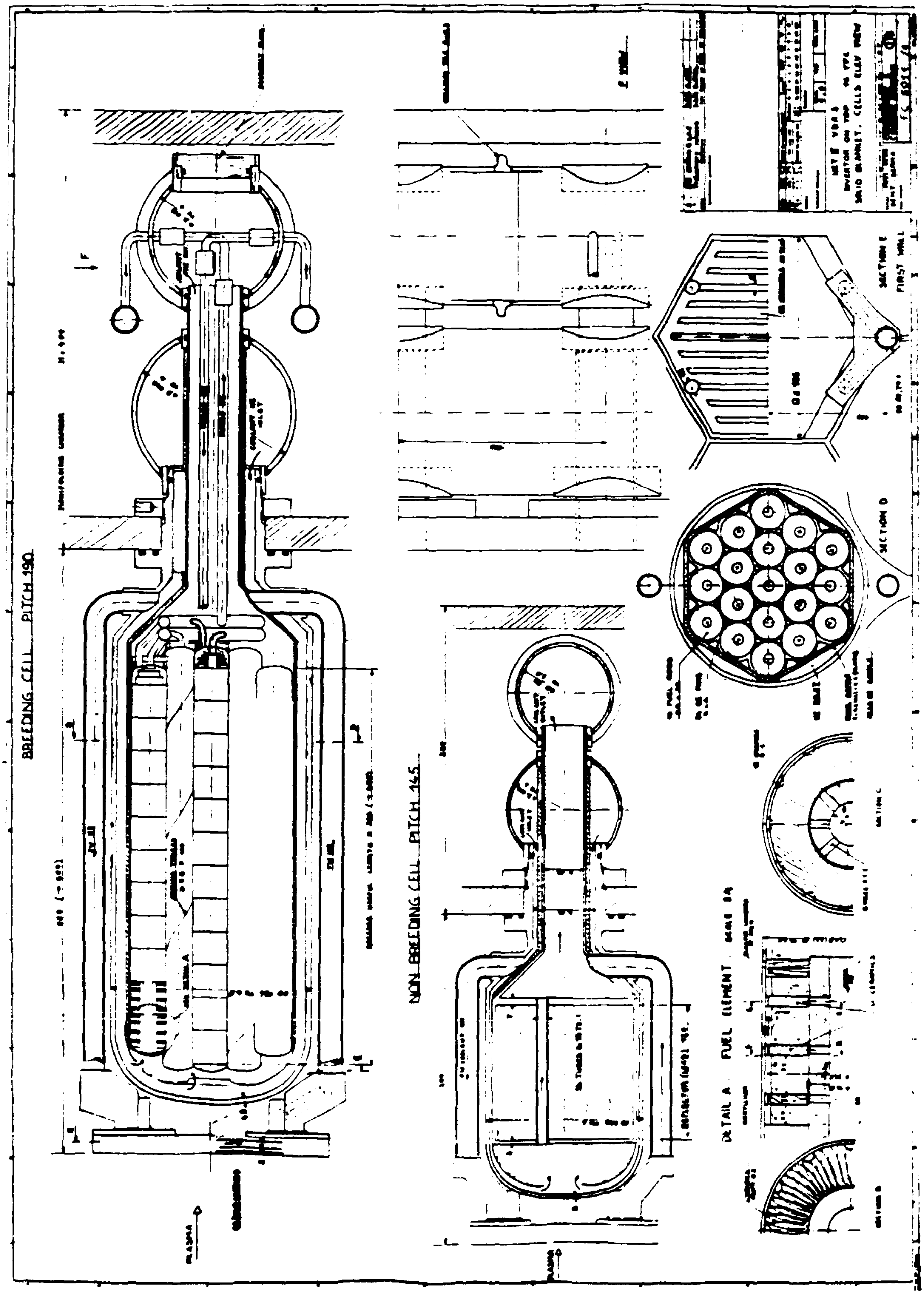


Figure 1 : Radial Canister Blanket Concept Adapted to NET II Boundary Conditions. Cells Elevation View.

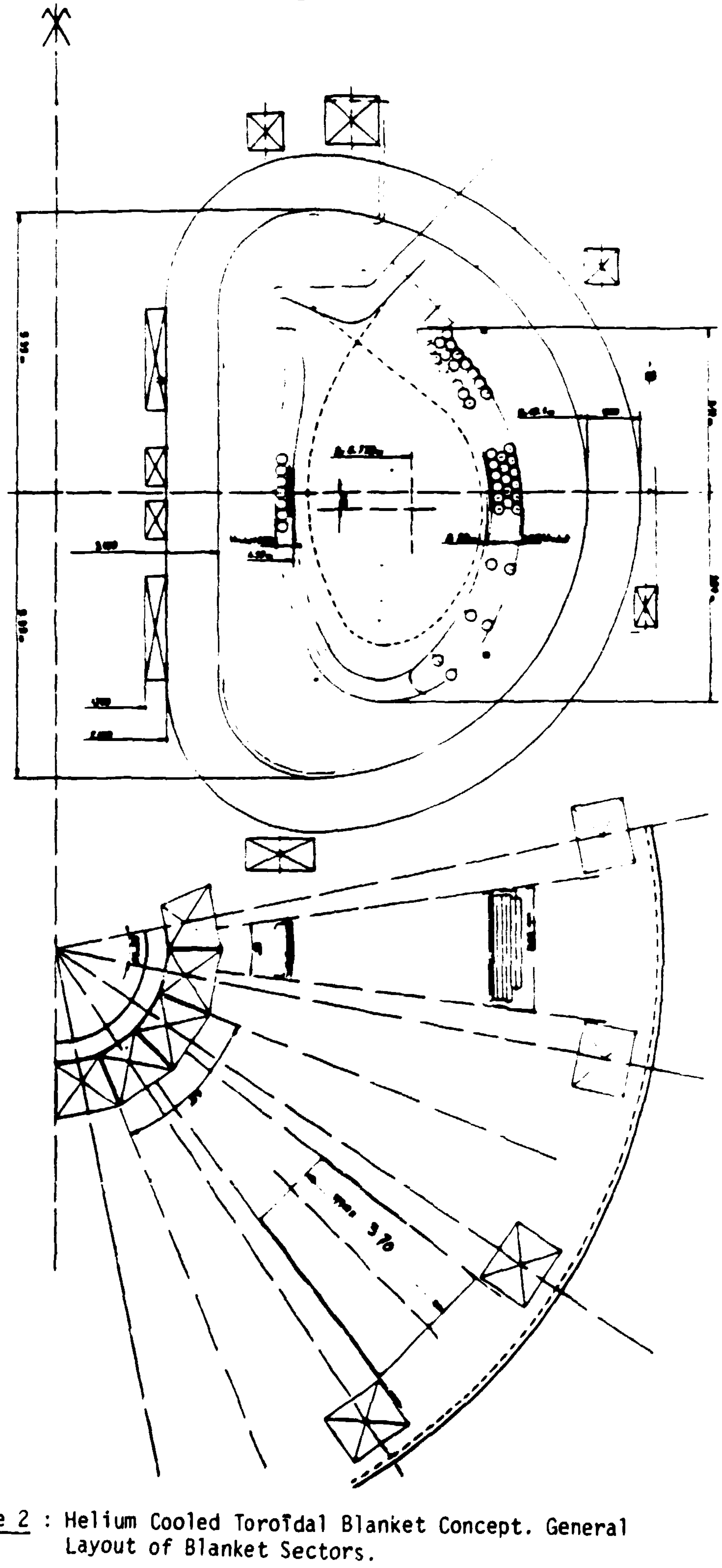
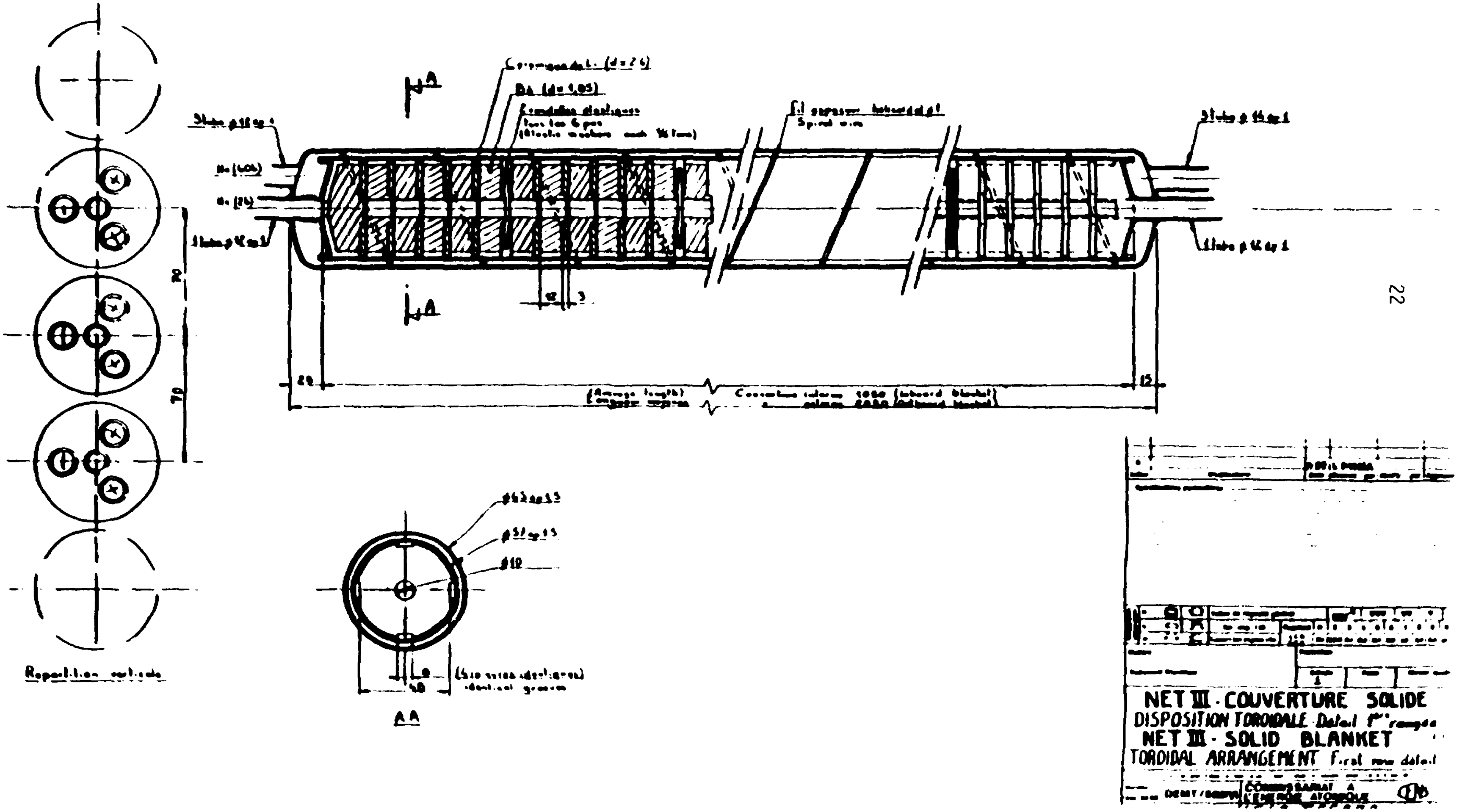


Figure 2 : Helium Cooled Toroïdal Blanket Concept. General Layout of Blanket Sectors.

Figure 3 : Helium Cooled Toroïdal Blanket Concept - Cutaway View of the Front Breeder Row.



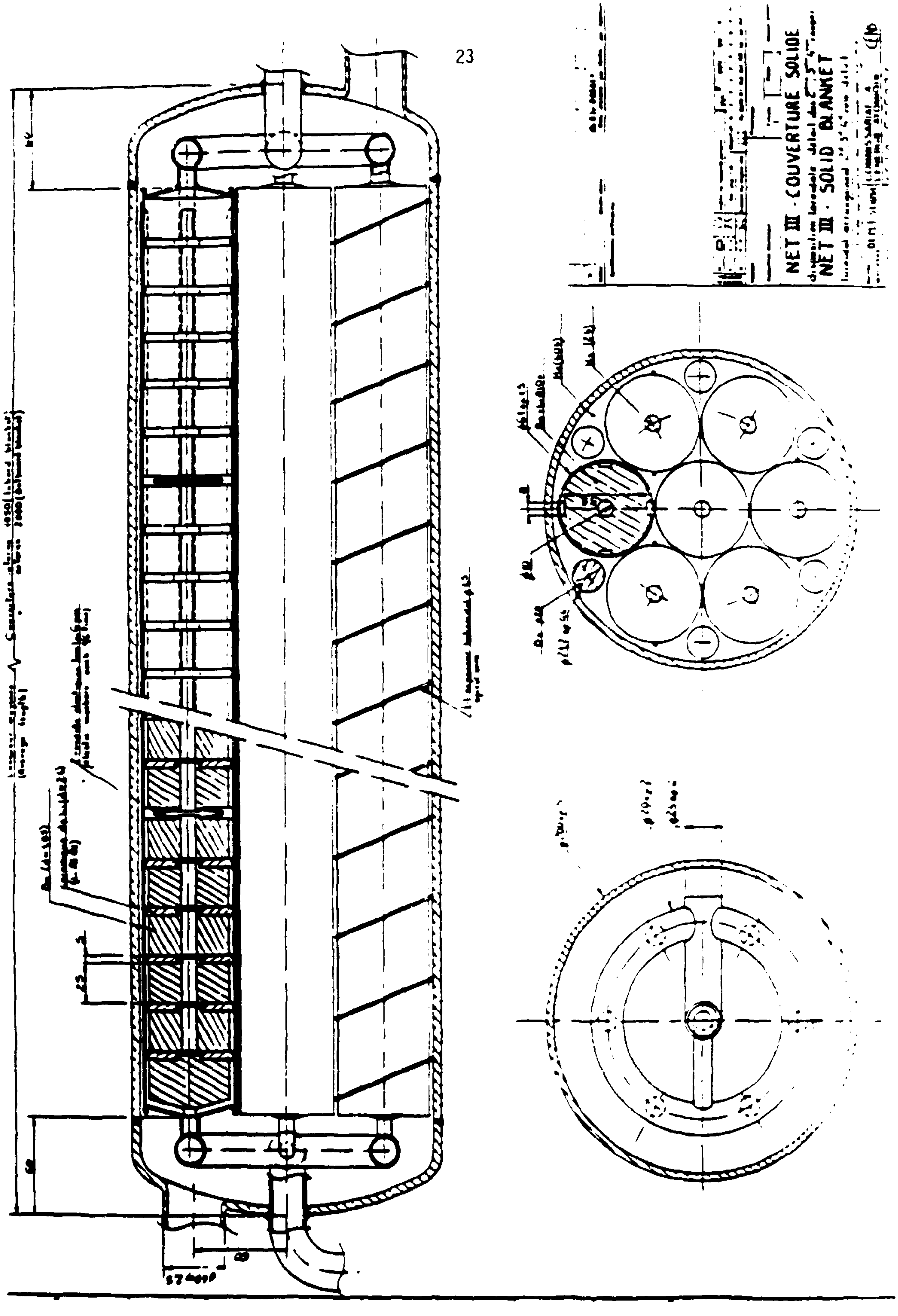


Figure 4 : Helium Cooled Toroïdal Blanket Concept - Cutaway View of a Tubular Module Typical of the 2nd, 3rd and 4th Breeder Rows.

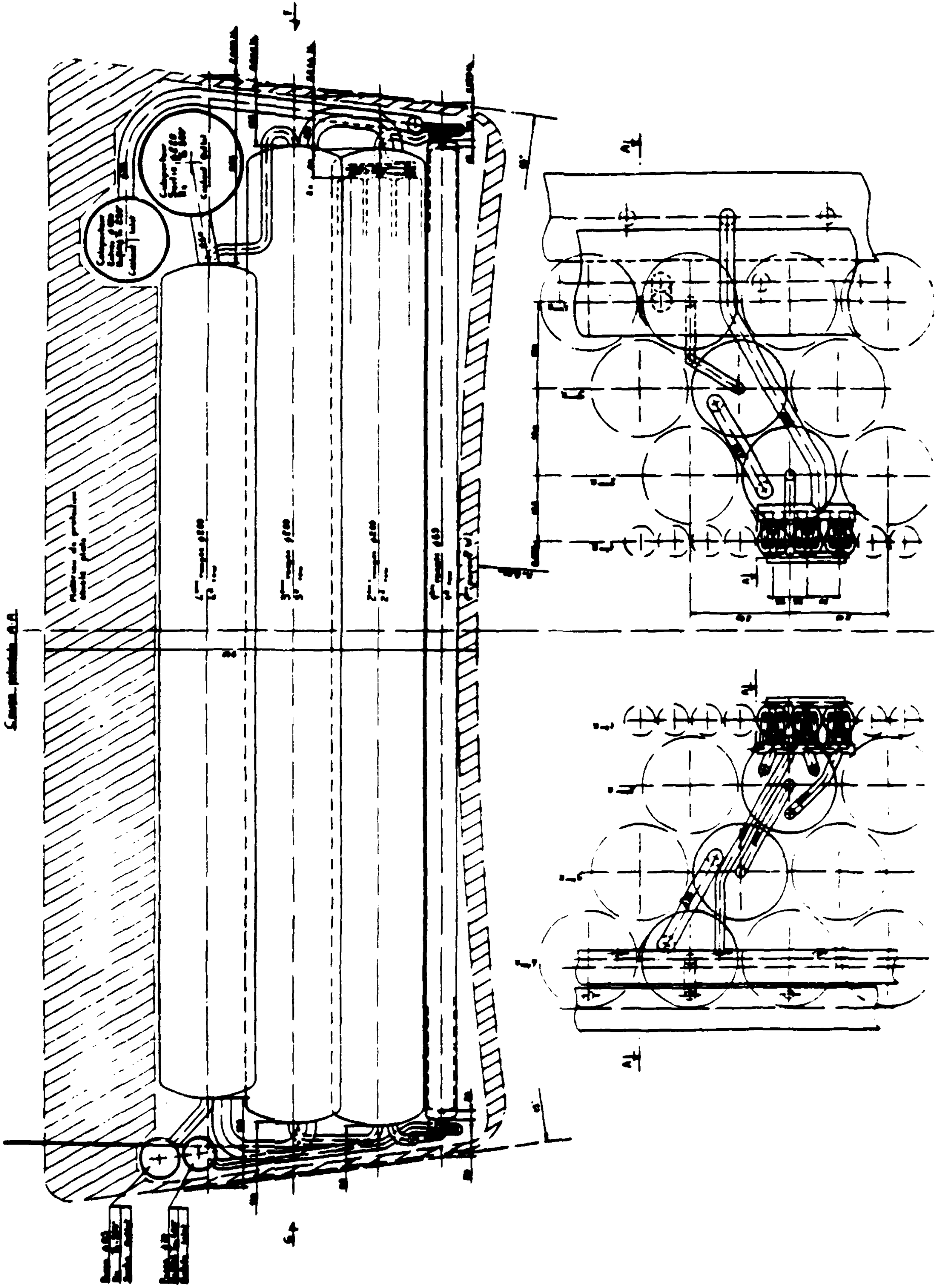


Figure 5 : Helium Cooled Toroïdal Blanket Concept - Cutaway View of the Outboard Blanket in the Equatorial Plan : Principle of Helium Manifolding.

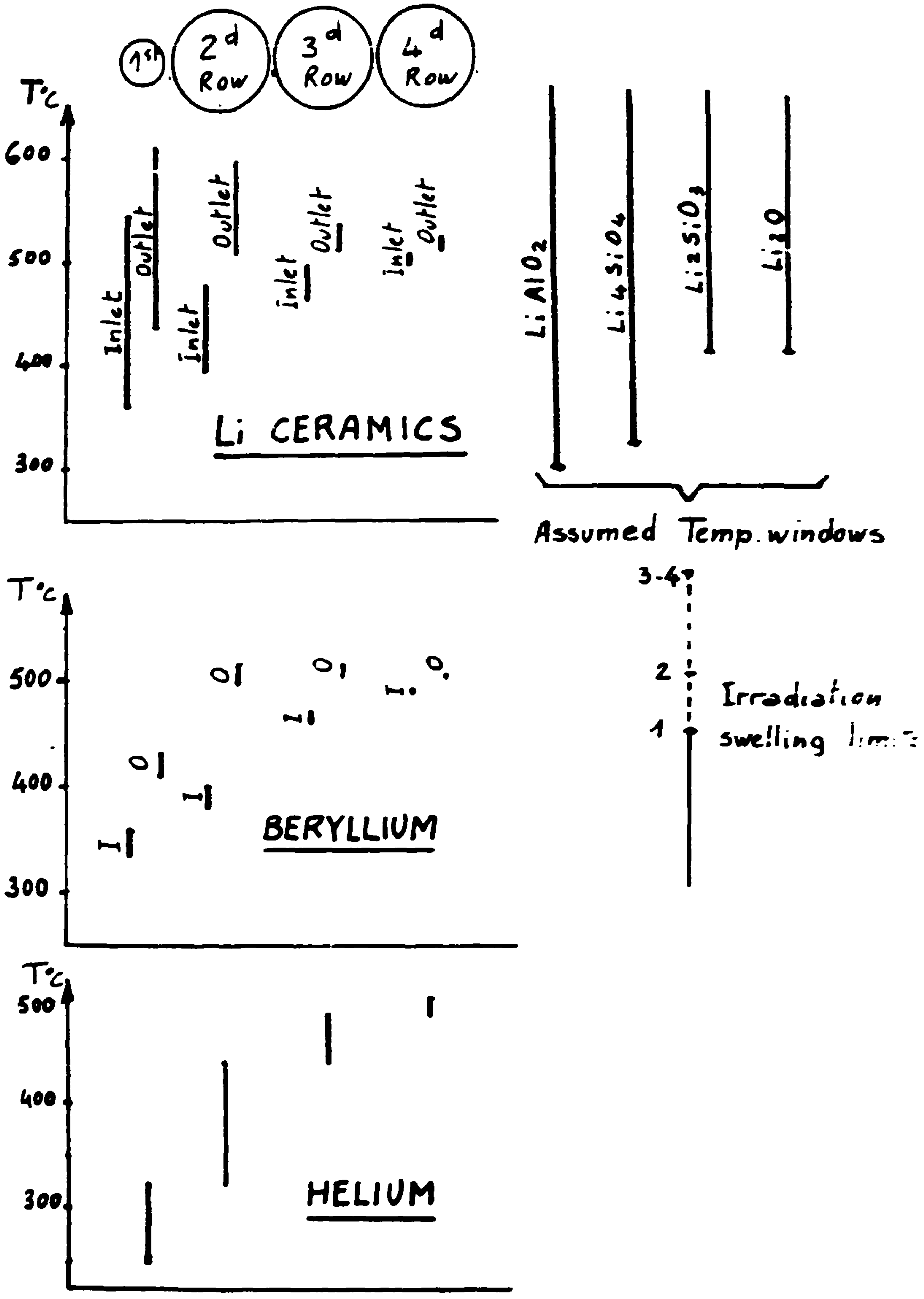


Figure 6 : Helium Cooled Toroidal Blanket Concept - Working Temperature Ranges.

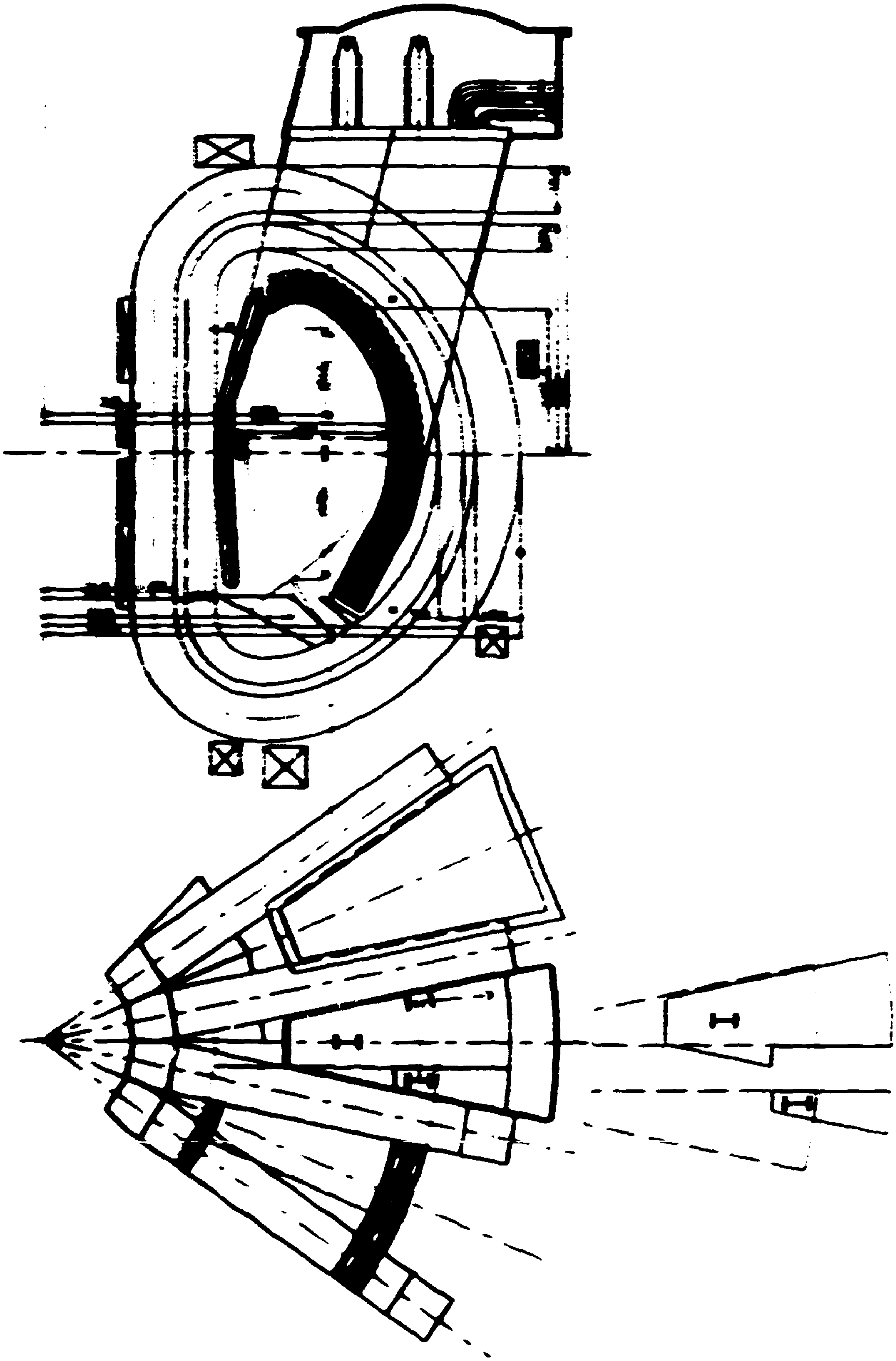


Figure 7 : Helium Cooled Toroïdal Blanket Concept - Second Configuration with Improved Coverage Factor Poloïdal Rows of Tubular Modules Replace the Toroïdal Arrangement in the Inboard Side.

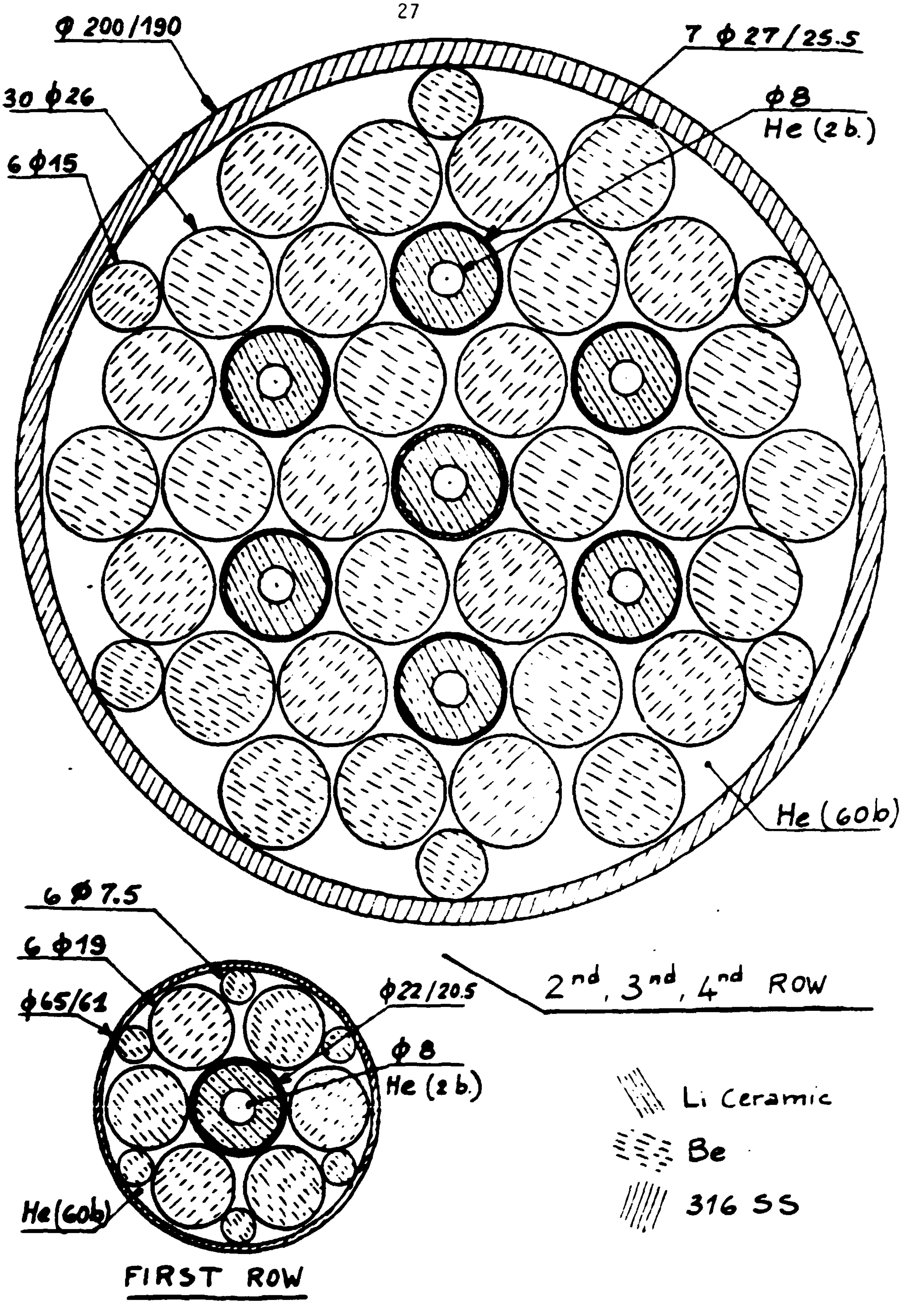


Figure 8 : Alternative Helium Cooled Toroïdal Blanket Concept. Cutaway View of Typical Tubular Modules Shawing Bundles of Separate Beryllium and Ceramic Pins.