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Status Report KfK Contribution to the Development of

DEMO-relevant Test Blankets for NET/ITER

Part 1: Self-cooled Liquid Metal Breeder Blanket

Volume 1: Summary

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Abstract

A self-cooled liquid metal breeder blanket for a fusion DEMO-reactor and the status of the development programme is described as a part of the European development programme of DEMO relevant test blankets for NET/ITER. Volume 1 (KfK 4907) contains a summary, Volume 2 (KfK 4908) a more detailed version of the report.

Both volumes contain sections on previous studies on self-cooled liquid metal breeder blankets, the reference blanket design for a DEMO-reactor, a typical test blanket design including the ancillary loop system and the building requirements for NET/ITER together with the present status of the associated R&D-programme in the fields of neutronics, magnetohydrodynamics, tritium removal and recovery, liquid metal compatibility and purification, ancillary loop system, safety and reliability.

An outlook is given regarding the required R&D-programme for the self-cooled liquid metal breeder blanket prior to tests in NET/ITER and the relevant test programme to be performed in NET/ITER.

This work has been performed in the framework of the Nuclear Fusion Project of the Kernforschungszentrum Karlsruhe and is supported by the European Communities within the European Fusion Technology Program.

Zusammenfassung

KfK-Beitrag zur Entwicklung von DEMO-relevanten Testblankets für NET/ITER

Ein selbstgekühltes Flüssigmetall-Brutblanket für einen DEMO Fusionsreaktor und der Stand der Entwicklungsarbeiten, als Teil des Europäischen Entwicklungsprogramms für ein DEMO-relevantes Testblanket für NET/ITER werden beschrieben. Band 1 (KfK 4907) enthält die Zusammenfassung und Band 2 (KfK 4708) den detaillierten Bericht.

In den beiden Berichten werden bisher durchgeführte Untersuchungen für selbstgekühlte Flüssigmetallbrutblankets beschrieben. Es werden der Referenzentwurf für das DEMO-Reaktorblanket und ein tpyischer Entwurf für ein Testblanket in NET/ITER mit den dazugehörigen externen Kreisläufen und einem Komponentenaufstellungsplan vorgestellt. Der augenblickliche Stand der Forschungs- und Entwicklungsarbeiten bezüglich: Neutronenrechnungen, Magnetohydrodynamik (MHD), Tritiumgewinnung, Beständigkeit im Flüssigmetall, Flüssigmetallreinigung sowie Sicherheit und Zuverlässigkeit der Kreisläufe wird aufgezeigt.

Es wird ein Ausblick gegeben auf die noch vor dem NET/ITER Test notwendigen F+E-Arbeiten für das selbstgekühlte Flüssigmetallblanket und das zugehörige Testprogramm, das in NET/ITER durchgeführt werden soll.

Die vorliegende Arbeit wurde im Rahmen des Projekts Kernfusion des Kernforschungszentrums Karlsruhe durchgeführt und ist ein von den Europäischen Gemeinschaften geförderter Beitrag im Rahmen des Fusionstechnologieprogramms.

Volume 1

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1. Previous Studies on Self-cooled Liquid Metal Blanket Concepts and Reasons for the Selected Design

Liquid metal breeder materials have a number of inherent advantages [1] over solid breeder materials such as the possiblity to remove the tritium from the breeder material outside the blanket, a practical immunity to irradiation damage and a high thermal conductivity. These features facilitate a relatively simple blanket design. Therefore, liquid metal breeders are selected in a number of reactor- and blanket studies.

The family of liquid-metal breeder materials is comprised of pure lithium, the Pb-17 Li eutectic alloy, and low melting point ternary Li-Pb-X alloys. Two major design classes for liquid metal blankets are under consideration: Separately-cooled blankets use stagnant or slowly flowing liquid-metal breeder with helium, water, or another liquid metal as coolant. In self-cooled blanket concepts, the liquid-metal breeder serves as coolant as well and is circulated for heat extraction. In this report, the main technical aspects of self-cooled designs will be examined.

From the beginning of fusion research lithium has been used in blanket concepts as breeder and coolant. However, the Pb-17Li eutectic alloy as proposed in the WITAMIR-study [2] is an interesting alternative. Compared to pure lithium, the main advantage of Pb-17 Li is its much lower chemical reactivity with air and water which allows for example even water cooling of the liquid breeder. In the case of self-cooled Pb-17Li blankets, water-cooled components such as divertors or limiters adjacent to the blanket segments are acceptable and simpler steam generator designs are possible in contrast to lithium loops.

A self-cooled Pb-17Li blanket concept similar to the WITAMIR-design has been proposed in the MARS-study [3]. Both machines are of the mirror type with a small surface heat flux to the first wall and a considerably reduced magnetic field strength compared to the one encountered in tokamaks. These features of the mirror machine allow for a much simpler blanket design without excessively high magnetohydrodynamic (MHD) pressure drop in comparison to a self-cooled blanket concept for tokamaks.
Self-cooled liquid metal breeder blanket concepts have been proposed in the Blanket Comparison and Selection Study, BCSS [4]. The design, shown in Fig. 1-1 is very similar for the lithium blanket and the Pb-17Li blanket. Breeding blankets are arranged at the outboard and the inboard region of the torus. In the first wall, the coolant flows in toroidal channels. The self-cooled blankets are the leading

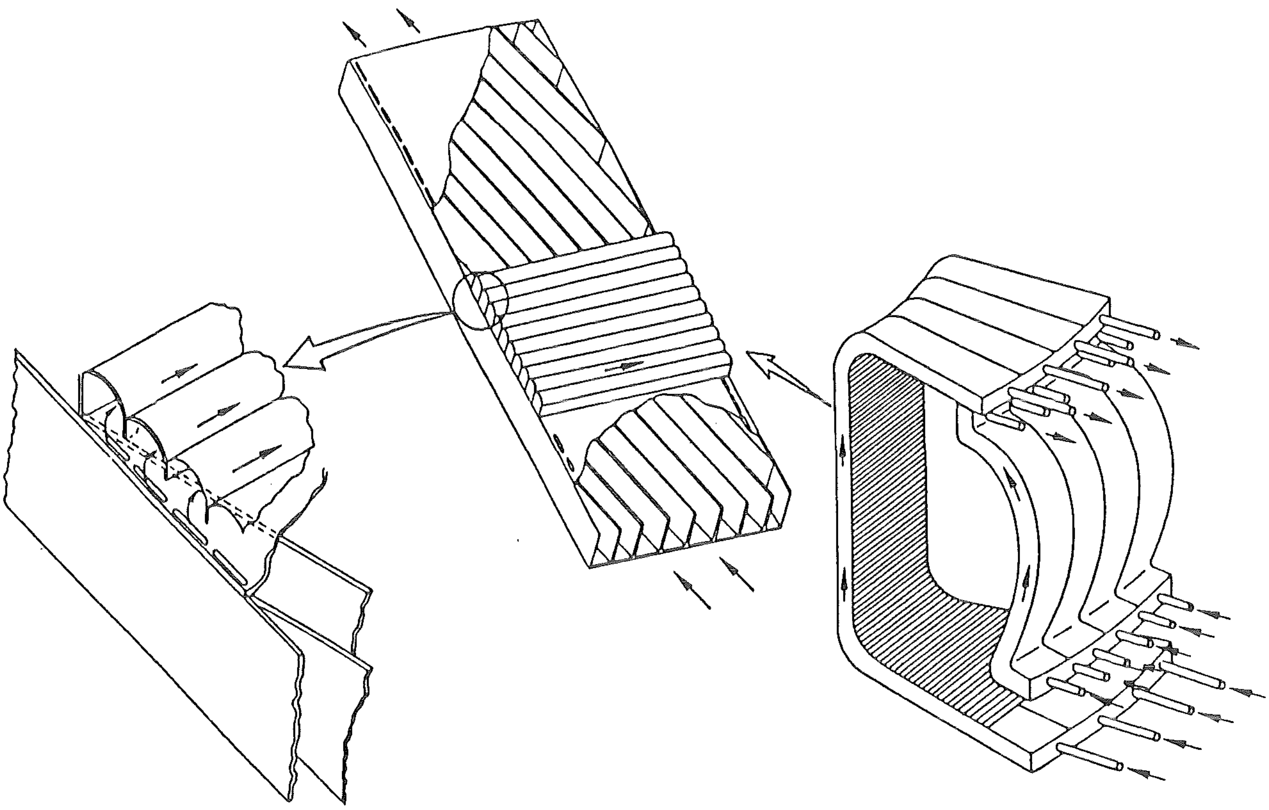


Fig. 1-1 Self-cooled liquid metal breeder blanket (BCSS)

concepts in the BCSS but the concept using Pb-17Li has been ranked lower for tokamak reactors in the BCSS compared to the concepts using lithium. In the European blanket development program, however, lithium has been excluded because of safety reasons.

A design of a self-cooled Pb-17Li blanket for the Next European Torus (NET) is described in Ref. [5] and shown in Fig. 1-2. The design is based on the same flow concept as considered in the BCSS but has been adjusted to the boundary conditions of NET. This concept served as a basis for a first design of a DEMO-relevant self-cooled blanket concept investigated at KfK [6,7]. It is based on the use of a 300 mm thick layer of beryllium in the front region of the blanket segment as shown in Fig. 1-3. This material serves as a neutron multiplier, providing tritium self-sufficiency without breeding blankets at the inboard region of the torus. Water-cooled steel reflectors are arranged at the inboard region. Restricting breeding blankets to the outboard region is especially advantageous for self-cooled liquid metal blankets because at the inboard region the magnetic field strength is roughly 50% higher and the space is more limited compared to the outboard region. Both differences lead to a MHD pressure drop in the inboard blanket segment higher by at least a factor of two compared to the outboard segments. The use of beryllium in the outboard segments avoids the need for inboard breeding but causes other problems. A different way to arrive at feasible concepts of self-cooled blankets has been enabled by the new specification of a DEMO reactor as released by the Test Blanket Advisory Group (TAG) in March 1990. This new geometry allows to split the inboard segments into upper and lower halves with separate coolant supply from the top (upper half) and bottom (lower half) leading to acceptable MHD pressure drop. This solution has been selected as reference concept and is described in the next section.

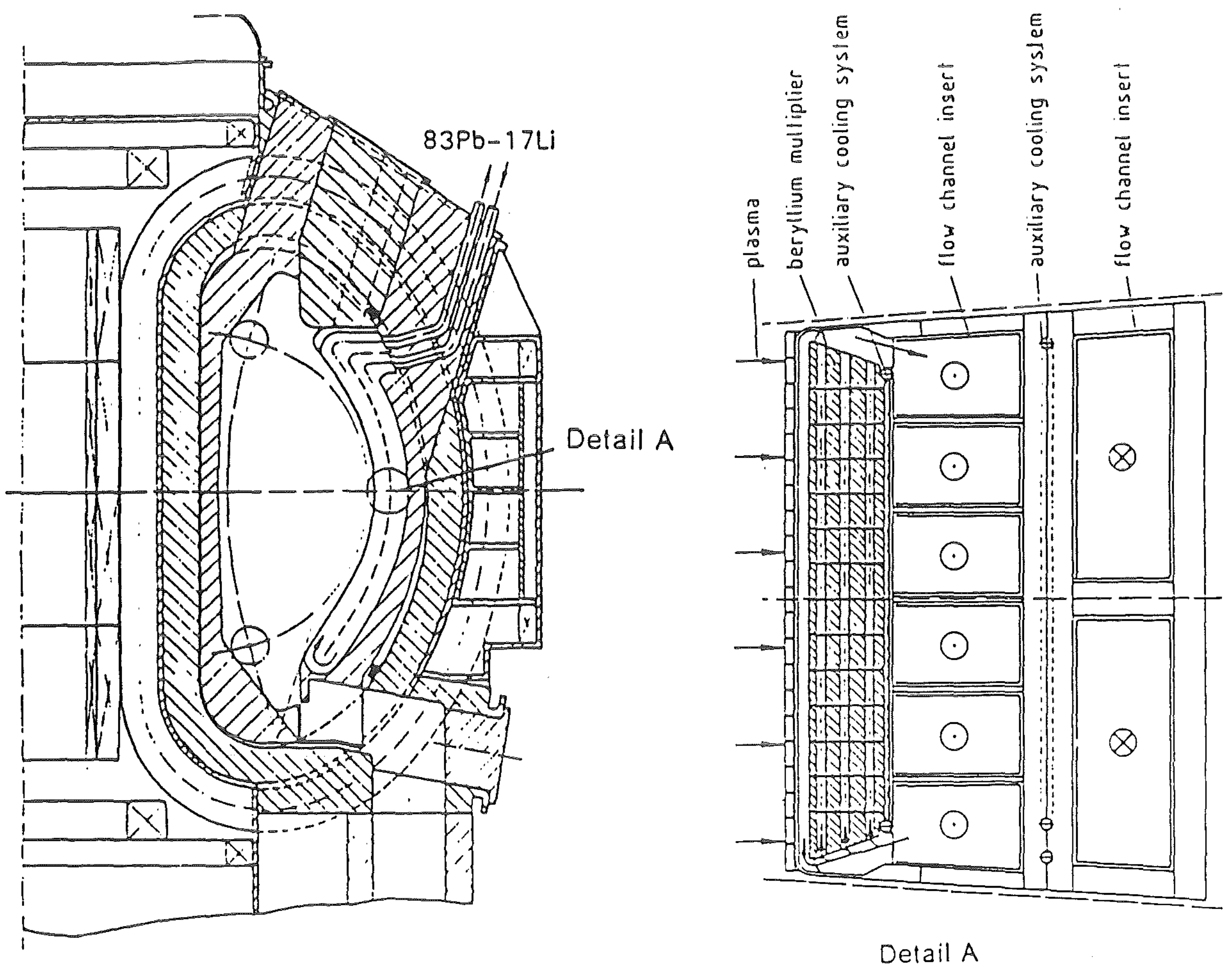


Fig. 1-2 Self-cooled liquid metal blanket for NET

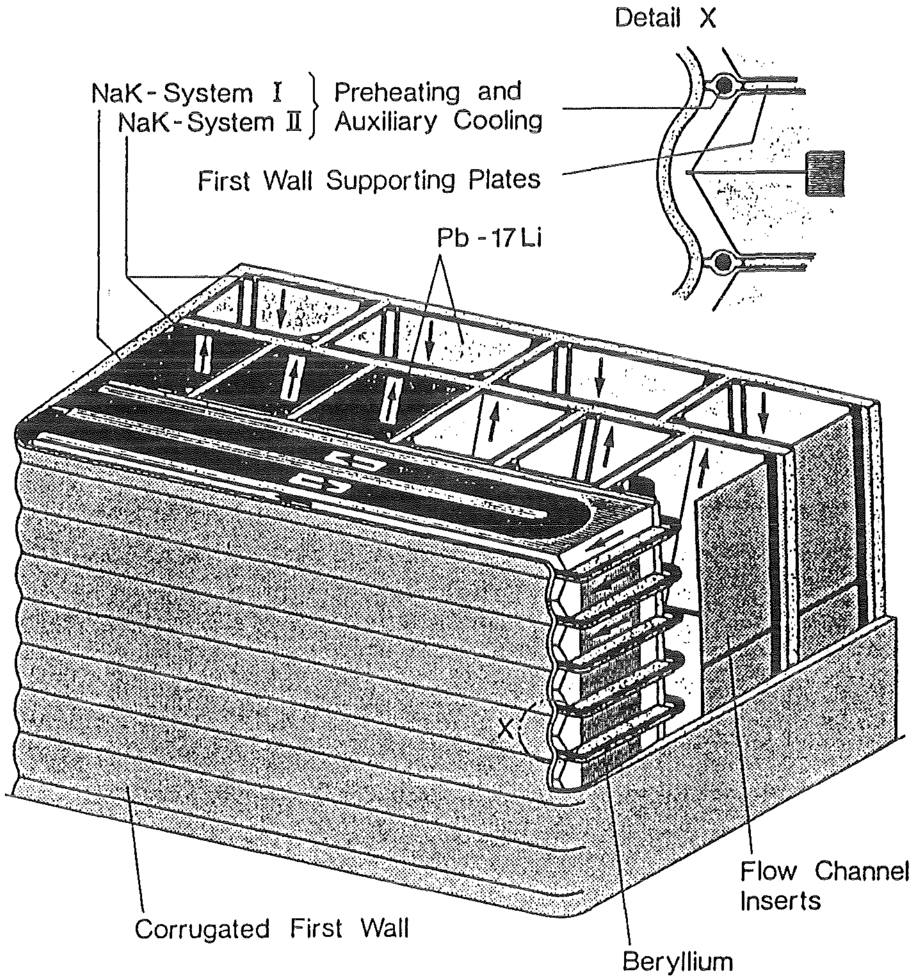


Fig. 1-3 Cross section of a self-cooled Pb-17Li blanket segment for DEMO

2. Blanket Design for a DEMO-Reactor

The main objective of the design work on DEMO-relevant blanket concepts is to provide the basis for a meaningful selection of a small number of concepts to be tested in NET/ITER. This implies on one hand that no detailed design is necessary but the key issues have to be identified and investigated. On the other hand, it is not sufficient to study the design of blanket segments only but the complete system including external loops required for heat- and tritium extraction and the main safety issues have to be investigated.

2.1 Blanket Segment

2.1.1 Design Description

The principle arrangement of the blanket segments is identical to the one proposed for ITER [8]. There are 48 outboard segments, with both coolant inlet and outlet tubes attached at the top end of the segments.
At the inboard side the torus is divided into 32 segments. The inboard segmets are split into upper and lower halves. At the upper half the coolant inlet and outlet tubes are connected to the top end of the segment, at the lower half to the bottom end. This arrangement can be seen in Fig. 2.1-1 which shows a vertical cross section of the torus.

The arrangement of inlet and outlet tubes at the same end of a segment requires a double pass of the coolant through the segment. The flow path is identical for the outboard segments and the upper half of the inboard segments. Coolant enters the blanket at the top end, flows downward in the rear channels, turns 180 deg at the bottom end and flows upward. This flow perpendicular to the magnetic field does not allow velocities high enough for sufficient first wall cooling. Therefore, the total coolant flow in upward direction is diverted into the relatively small toroidal channels (parallel to the main magnetic field) of the blanket front region. This coolant diversion is achieved by a slight inclination of the walls in the return ducts. The liquid metal flows with relatively high velocity in the toroidal channels between the plasma facing first wall and the second wall. Passing two meander shaped cooling channels in toroidal direction, the coolant is heated up further before it flows back in the return channels to the exit at the top end of the blanket.

This flow path used in the front part of the blanket can be seen in Fig. 2.1-2 which shows a cross section of an outboard segment.

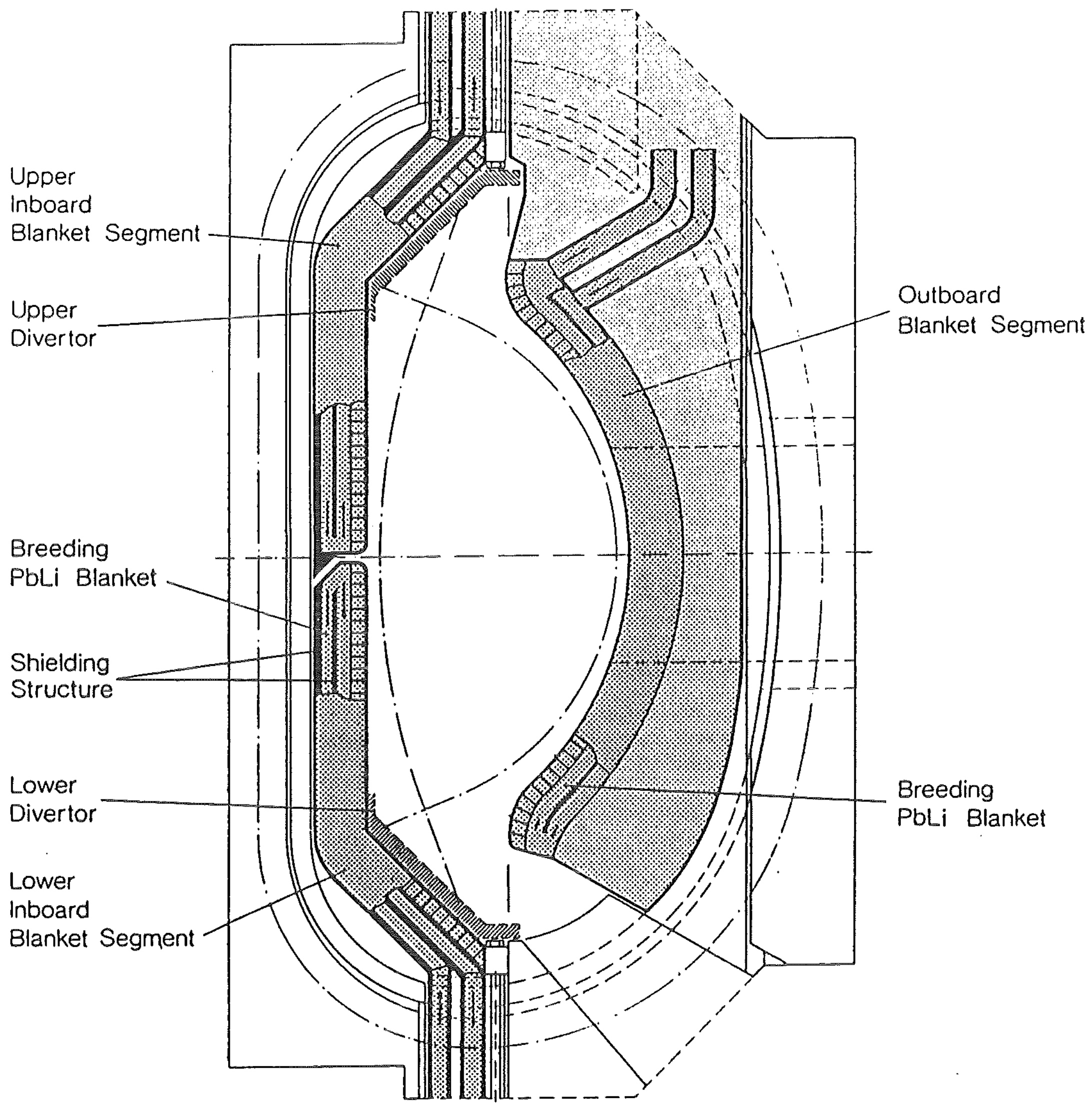


Fig. 2.1-1 Arrangement of self-cooled breeding blankets in DEMO

Compared to a single toroidal channel for first wall cooling the arrangement of meander-shaped channels leads to a mechanically stiff region in the front of the segment and to lower liquid metal temperatures in the first wall cooling channel because this liquid metal is coming from a zone characterized by a low power density due to the steep gradient in volumetric heat generation.

The first wall separating the toroidal cooling channels from the plasma chamber is considered to be the most critical part of the blanket segment. It is thermally loaded by a rather high heat flux caused by radiation and a particle flux at the plasma facing surface. To avoid excessive thermal stresses, the wall thickness must

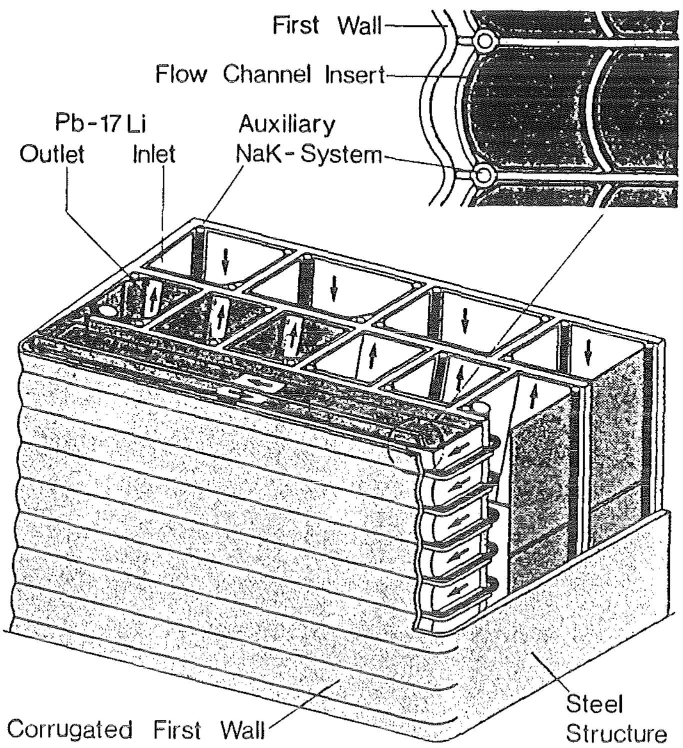


Fig. 2.1-2 Cross section of a self-cooled blanket segment

not exceed 6 mm. The stiffness of the first wall is increased by giving it a corrugated shape with webs welded in between to support it.

In the blanket segment, the coolant flows with velocities between 0.5 m/s (in the inlet channels perpendicular to the magnetic field) and 2 m/s (along the first wall, parallel to the magnetic field). In spite of this optimum flow and velocity distribution, failure to decouple electrically the load-carrying walls from the liquid metal would result in excessively high MHD pressure drops due to the voltage induced by the magnetic field.

In the present blanket design, all coolant channels except the first wall cooling channel are equipped with flow channel inserts (FCIs) described in Section 4.3. and shown in Fig. 4.3-4. Mechanical stresses in the FCI become insignificant when a longitudinal slot or holes are made for pressure equalization between the inner flow region and the outer gap. The voltage induced in the liquid metal flowing through the FCI is short-circuited only over the inner liner. Even the longitudinal slot does not give the voltage an additional short-circuit path. The FCI covers the total surface of a flow channel or only parts of it. The thickness of the inner liner can be reduced to a minimum value necessary for sufficient corrosion resistance

($\simeq$0.5 mm) in order to minimize the MHD pressure drop. The flow in the toroidal channels is basically in parallel to the main magnetic field. This direction is usually not accompanied by a large MHD pressure drop. Nevertheless, FCI's are used in nearly all channels in order to avoid excessively large currents flowing perpendicular to the plates which would cause the multi-channel effect as explained in Section 4.2.

Under normal conditions, the breeder material is kept liquid and the afterheat is removed by circulating it in the primary loop, providing cooling or heating outside of the blanket segment. If for some reasons the breeder material in the segment solidifies or if afterheat removal by forced or natural convection is not possible (for example during blanket exchange), an indepenent auxiliary heat transport system to be housed in the segment is necessary. NaK serves as the heat transfer medium. The NaK system allows for the afterheat removal during or subsequent to emptying the blanket segment.

2.1.2 Neutronics

The main neutronic task is to assess the real tritium breeding ratio, the shielding performance and to provide the power density distribution in the three-dimensional geometrical configuration of the DEMO reactor; this can be achieved by means of Monte Carlo transport calculations.

For this purpose a three-dimensional torus sector model of the Demo reactor has been set up. The DEMO reactor is composed of 16 torus sectors (22.5^0), each consisting of two inboard and three outboard segments. Due to its toroidal symmetry it is sufficient to model a torus sector of 11.25^0 and applying reflective boundary conditions (Fig. 2.1-3). Their poloidal arrangement can be taken from Fig. 2.1-4.

The Monte Carlo code MCNP [9] has been used to set up the torus sector model and for performing all neutronic calculations presented here. The nuclear data used in these calculations originate from the European Fusion File EFF-1 and have been processed into a MCNP working library at PSI Würenlingen [10].

In the Monte Carlo calculation, the history of a neutron is followed from its birth to its death by making use of its stochastic behaviour. In this context, the spatial distribution of the plasma source density is described by a corresponding probability distribution for the 14 MeV source neutrons (for details see Ref. [11]). A large

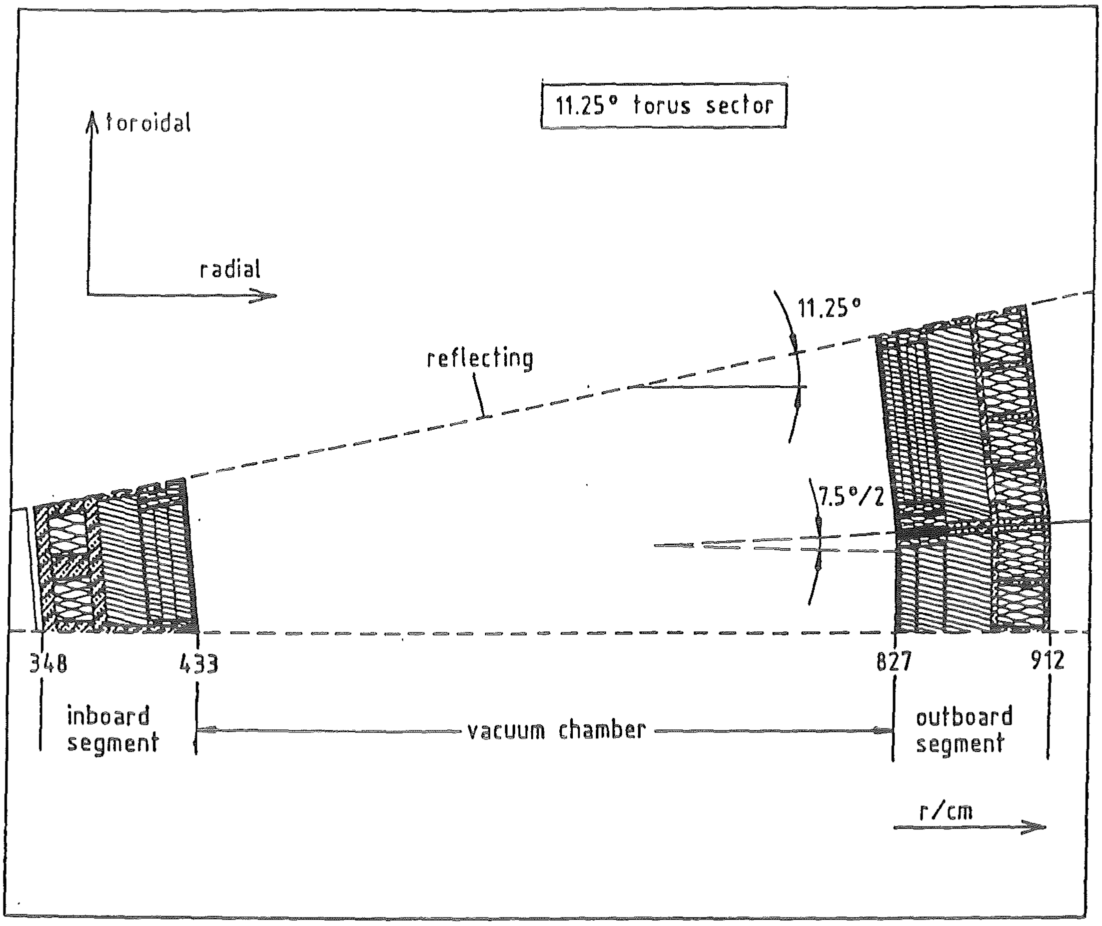


Fig. 2.1-3 Radial-toroidal cross section of the 11.25° torus sector model

number of neutron histories have to be followed in order to assure a sufficient statistical accuracy.

Tritium breeding ratio

For the calculation of the tritium breeding ratio (TBR) about 50 000 neutron histories have been followed. Table 2.1-1 shows the neutron balance and the associated statistical errors obtained in these calculations. The rather high global breeding ratio of TBR = 1.165 obviously is due to the utilization of the divertor region for breeding. The breeding ratio would drop to TBR = 1.057 ± 0.3% in case without divertor breeding.

The tritium breeding in the actual reactor will be reduced due to the inclusion of blanket ports for plasma heating, remote handling, pellet injection, diagnostics, etc. The DEMO reactor design provides horizontal ports in a total of 10 outboard blanket segments. To assess their impact on the breeding performance a torus sector model of $4 \times 11.25^0 = 45^0$ with 4 inboard and 6 outboard segments has been constructed [12]. It represents the whole reactor with 32 inboard and 48 outboard seg-

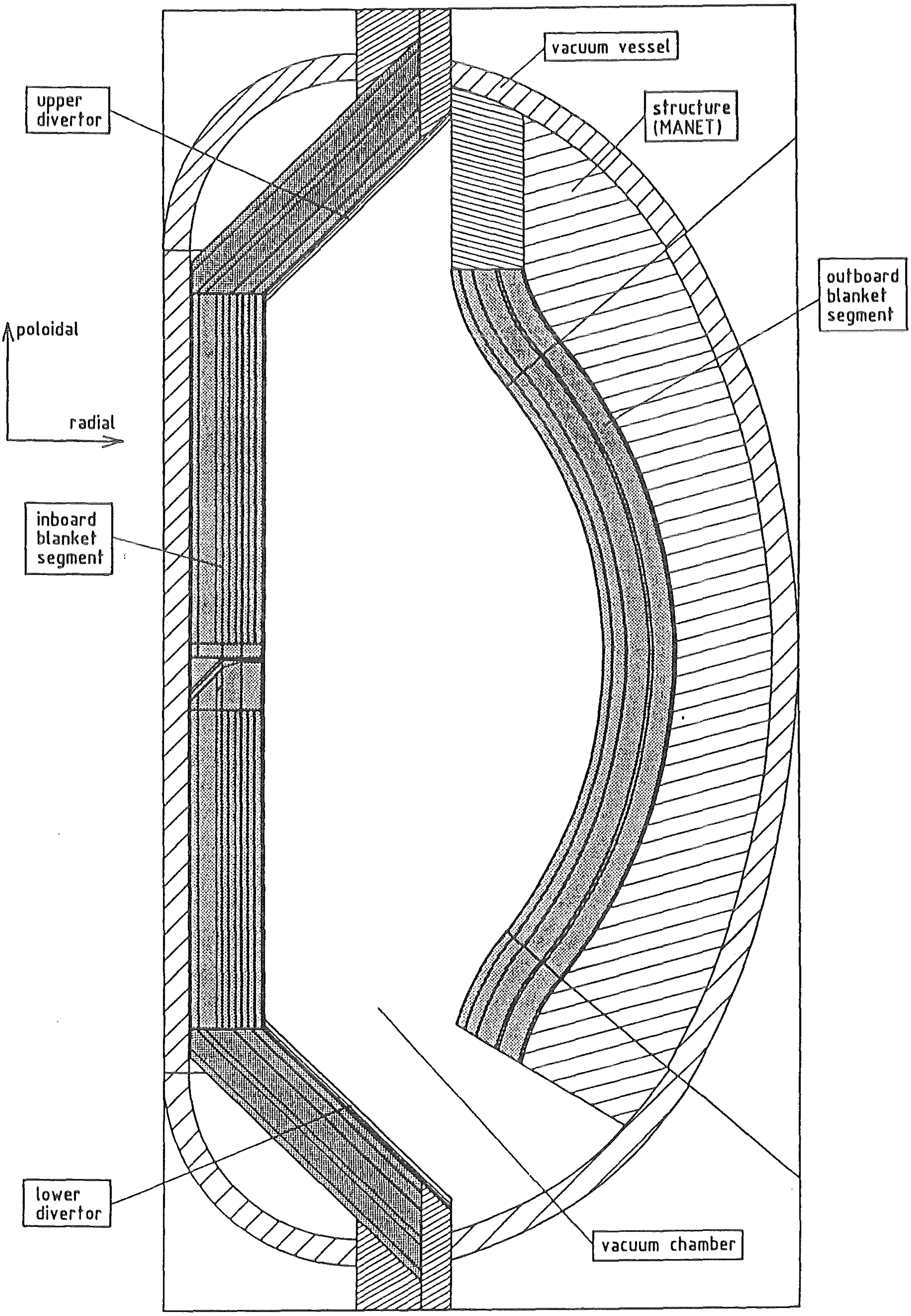


Fig. 2.1-4 Radial-toroidal cross section of the torus model

ments by applying reflective boundary conditions. In this model 8, 12 and 16 horizontal ports of the DEMO reactor are simulated. Each port covers an area of 340 cm height times the full segment width in the centre of an outboard segment. For 10 ports a TBR loss of 0.06 has been interpolated. Thus it can be deduced that the global TBR of the self-cooled liquid metal blanket in the actual configuration of the DEMO reactor with a total of 10 ports would decrease from TBR = 1.165 to about TBR = 1.10. On the other hand it should be mentioned that the in- and outlet liquid metal flow channels of the outboard segments are not taken into account in the TBR-calculations; actually this would provide a TBR-increase of about 0.05.

neutron multiplication M	1.620 ± 0.1 %
tritium breeding ratio TBR	
outboard segment	0.766 ± 0.4 %
inboard segment	0.285 ± 0.7 %
divertor region	0.113 ± 1.2 %
total TBR	1.165 ± 0.3 %

Table 2.1-1: Neutron balance for the liquid metal blanket in the DEMO reactor configuration

Power production

For the thermal-hydraulic blanket layout a detailed evaluation of the spatial power density distribution is needed. This is achieved by performing coupled neutron-photon Monte Carlo transport calculations with up to 150 000 source neutrons. In these calculations the material zones of the torus sector model are appropriately segmented in radial and poloidal direction. Based on a fusion power of 2216 MW of DEMO the calculated power production of single blanket segments and the complete reactor, equipped only with self-cooled liquid metal blanket segments, is given in Table 2.1-2. 80% of the power is produced in the front part of the blanket segments, where the meander flow scheme is applied (about 1/3 of the total blanket thickness, see Fig. 2.1-2), whereas the back part of the blanket with the liquid metal inlet channels (with again about 1/3 of the blanket thickness) produces only 4% of the power.

inboard segment (11.25^0)	
central part	13.5
divertor breeding region	3.74
total inboard	17.2
outboard segment (7.5^0)	25.7
total power production	1783

Table 2.1-2: Power production [MW] in single blanket segments and in the complete reactor based on a fusion power of 2216 MW

Due to the inherent nuclear properties of Pb-17Li the radial and poloidal profiles of the power density are comparatively flat. The maximum power density of Pb-17Li is not more than about 23 W/cm^3 (outboard first wall cooling channel at torus mid-plane).

Shielding

Radiation shielding is most crucial at the inboard side of the DEMO reactor: although the total thickness of the blanket/shield system amounts to 115 cm, the thickness of the vacuum vessel, acting as major shielding component, is only 30 cm, whereas the total blanket thickness is 85 cm. The shielding performance of a breeding blanket in general is poor (with the exception of water cooled blankets).

Therefore, it is questionable, if the shielding requirements can be met in the actual configuration of the DEMO reactor. To clarify this question, appropriate shielding calculations have been performed in the 11.25^0 torus sector model of the DEMO reactor. The MCNP-code has been applied for performing the shielding calculations making use of the importance sampling technique. About 150 000 neutron histories have been followed in the shielding calculations to assure a sufficient statistical accuracy in the region of the TF-coil. For the vacuum vessel the following layout has been used: a steel layer (SS-316) of 5 cm thickness, a layer of borated water (40 g H_2 BO_3 per liter of water) with a thickness of 2 cm, a steel layer of 7 cm, again a layer of borated water (5 cm) and a steel layer of 11 cm.

	Radiation design limits	Reference design	Optimized vacuum vessel	"ZrH-option" (0.2 ZrH / 0.8 Pb-17Li)	
				reference	inboard blanket divided
Epoxy radiation dose [rad]	$\sim 5 \cdot 10^{9}$				
torus mid-plane		$1.02 \cdot 10^{10}$	$3.61 \cdot 10^{9}$	$3.34 \cdot 10^{9}$	$9.55 \cdot 10^{9}$
poloidal average		$5.67 \cdot 10^{9}$	$2.26 \cdot 10^{9}$	$1.88 \cdot 10^{9}$	$2.63 \cdot 10^{9}$
Fast neutron fluence [cm^{-2}]	$\sim 10^{18}$				
torus mid-plane		$2.73 \cdot 10^{18}$	$8.78 \cdot 10^{17}$	$1.37 \cdot 10^{18}$	$2.86 \cdot 10^{18}$
poloidal average		$1.50 \cdot 10^{18}$	$5.92 \cdot 10^{17}$	$7.24 \cdot 10^{17}$	$8.97 \cdot 10^{17}$
Maximum power density [mW/cm^3]	~ 1.0				
torus mid-plane		2.52	0.891	0.822	2.35
poloidal average		1.40	0.557	0.464	0.650
Total power production in the TF-coil [kW]	~ 20	18.90	12.25	6.67	8.48

Table 2.1-3: Radiation loads on the inboard TF-coil for various design options (20 000 h integral operation time)

The results of the shielding calculations for the reference design and for some improved shielding options are listed in Tab. 2.1-3. One of the improved options is to insert 20% (vol) of ZrH in the outer liquid metal ducts in order to moderate the neutrons. It can be seen that the radiation design limits for the DEMO reactor (2216 MW fusion power and 20 000 h integral operation time) can be met in case of neglecting the division of the inboard blanket segment. The inclusion of the blanket division increases the radiation loads at the torus mid-plane by about a factor 3 and the radiation design limits are exceeded by a factor 2 in this case. This factor can be eliminated easily by increasing the ZrH-fraction in the inflow liquid metal channels to about 40%, or, by optimizing the vacuum vessel for shielding, or, simply by increasing the thickness of the vacuum vessel by about 5 cm.

2.1.3 MHD Analysis

The design of the liquid metal cooled blankets is dominated by MHD considerations. Liquid metal flow within the strong magnetic field of a tokamak is accompanied by a high MHD presssure drop which can be a feasibility issue because it may result in mechanical stresses beyond the allowable limits of the structural material. Additionally, the magnetic field may also influence the velocity in the cooling channels and, herewith, the heat transfer.

The MHD pressure drop of the liquid metal flowing in the poloidal and radial ducts is calculated using mainly simple analytical models based on the well established slug flow model for liquid metal channel flow [13].

In this model a constant velocity over the cross section of the cooling channels is assumed, an assumption which is also taken for the thermal-mechanical analysis in this report.

This model can be applied mainly for straight or slightly bended channels which are perpendicular to the magneticl field. Channel flow in bends in plains perpendicular to the magnetic field are treated in a similar way using an equivalent mean length.

The pressure drop in expansions and contractions is calculated using simple analytical correlations, which are based on experiments and numerical calculations.

The flow of the liquid metal in and out of the magnetic field is calculated numerically using the Core Flow Solution [14]. The radial to toroidal bend, this means the change of the flow direction from the radial feeding to the toroidal front channels as well as the meander shaped flow are also calculated with simple expressions based on experiments [15]. Additionally, the meander shaped flow was numerically analysed with a generalized version of the Core Flow Solution.

There are no models or experimental results available to determine the pressure drop resulting from the MHD interaction of the parallel flow in the first wall channels. Therefore, for a first assessment, the pressure drop caused by multichannel effects has been estimated using an electrical network method.

Table 2.1-4 shows a summary of the pressure drops calculated for the different regions. It should be mentioned, that the values of category ① (pressure drop in ducts perpendicular to the magnetic field) contain already a factor of 1.2 in order to account for the increased pressure drop in regions of developing flow. The total pressure drop in an outboard segment amounts to 3.4 MPa, in an inboard segment to 3.1 MPa. Mechanical stresses in the blanket structure caused by these pressure drops and the static pressure are described in the following section.

Table 2.1-4 MHD Pressure Losses

Category		Δp [MPa]	
		Outboard Segment	Inboard Segment
①	Pressure drop in ducts perpendicular to the magnetic field	1.744	1.379
②	Pressure drop in bends between flow directions perpendicular and parallel to the magnetic field	0.012	0.025
③	Pressure drop in the transition zone between poloidal distribution channels and radial ducts	0.641	0.509
④	Pressure drop in the meander flow region	0.220	0.230
⑤	Pressure drop caused by multi channel effects	0.78	1.0

2.1.4 Thermal-mechanical Analysis

The thermal-hydraulic analysis receives input from a neutronic analysis. The spatial distribution of the power density in the liquid metal (Pb-17Li) and the steel structure (MANET) has been determined by means of a three-dimensional Monte Carlo calculation (cf. Section 2.1.2). The volumetric heat generation in a single blanket segment caused by the neutron flux results in a heat input of about 17.2 MW and 25.7 MW for the inboard and outboard blanket segment, respectively. Taking into account a conservatively high mean value for the surface heat flux of 0.4 MW/m^2, the total heat input amounts to 20.1 MW and 29.2 MW for the inboard and outboard blanket segment, respectively. The total coolant mass flow rate is determined by the allowable temperature rise of the liquid-metal flow between the blanket inlet and outlet. This temperature rise is limited by the allowable maximum temperature at the coolant-to-wall interfaces as dictated by corrosion considerations. A mean exit bulk temperature of 400°C has been selected to keep the maximum temperature at the interface of coolant and martensitic steel wall below 470°C with respect to corrosion of the structural material. The blanket inlet temperature is governed by the melting temperature, which has a value of 235°C for the Pb-17Li eutectic alloy. Therefore, the inlet temperature has been set at 275°C, a temperature which is well above the ductile-brittle-transition temperature for irradiated martensitic steel. This results in an overall temperature rise of 125 K and a total mass flow rate of 850 kg/s and 1236 kg/s through the inboard and outboard blanket segment, respectively. Since the inboard segment is divided into an upper and a lower half of equal size, the mass flow rate through each half is 425 kg/s. There is one more constraint which determines the thermal-hydraulic design, i.e. the first wall temperature limit of 550°C is based on strength of material data.

Steady-state temperature calculations have been carried out for the front zone with meander-shaped coolant channels at the torus midplane (Fig. 2.1-5). Because of the higher thermal load of the outboard blanket it is sufficient to calculate this part only. The computations have been performed by using the finite-element code ABAQUS [17]. For the computation of the temperature field, a maximum surface heat flux of 0.5 MW/m^2 has been assumed, which is considered as a peak value of DEMO. The radial distribution of the material dependent power density has been provided by neutronic calculations. The heat generation in the steel structure decreases with radial distance from the first wall. In the first wall, the volumetric heat generation amounts to 20 W/cm^3. The power density in the coolant itself takes a maximum value of 23 W/cm^3 in the first wall cooling channel.

Figure 2.1-6 shows the temperature field at the exit side of the first wall channel. It can be seen that there is a boundary layer near the first wall where the tempera-

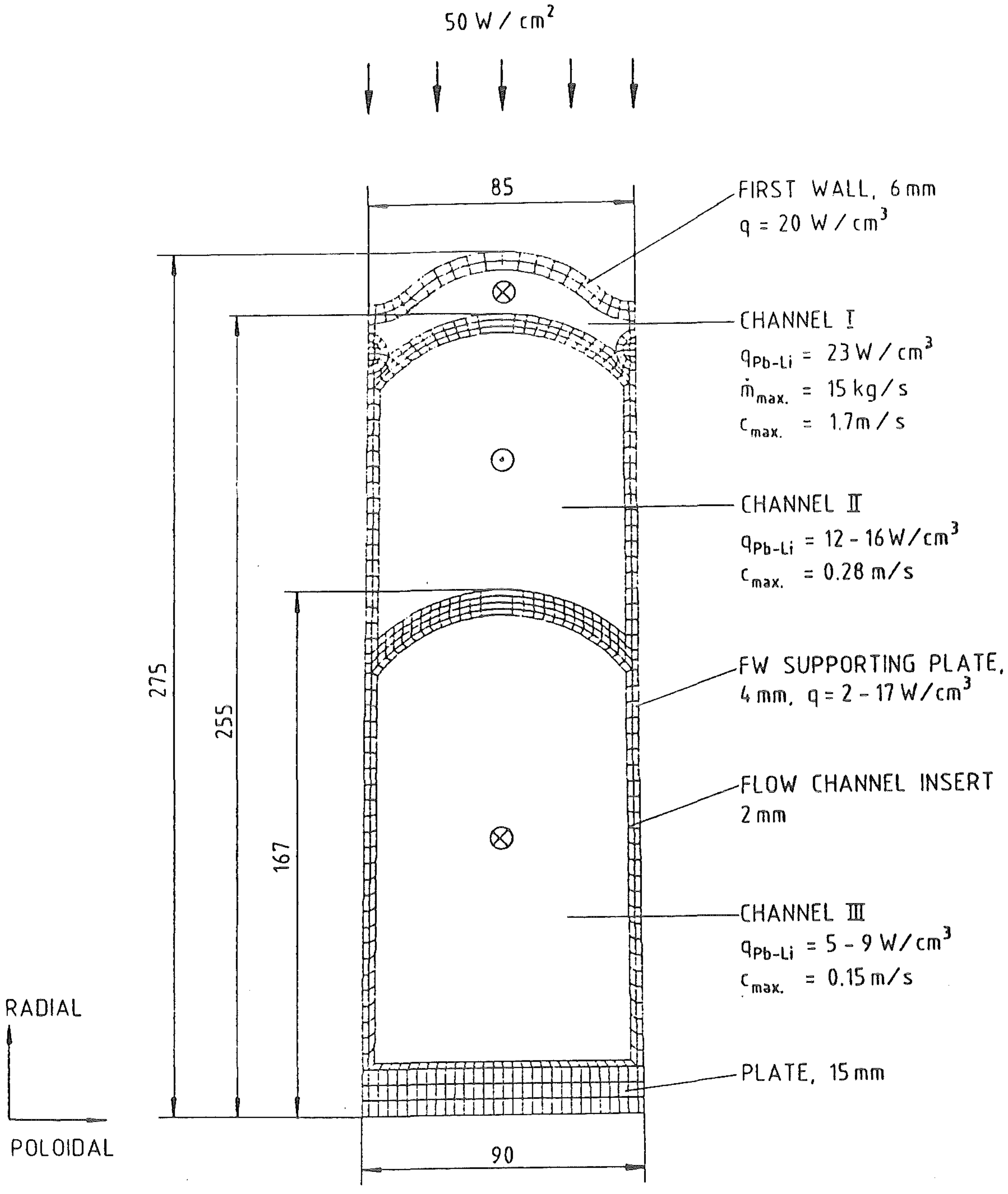


Fig. 2.1-5 Cross section of the front region with meander-shaped coolant channels

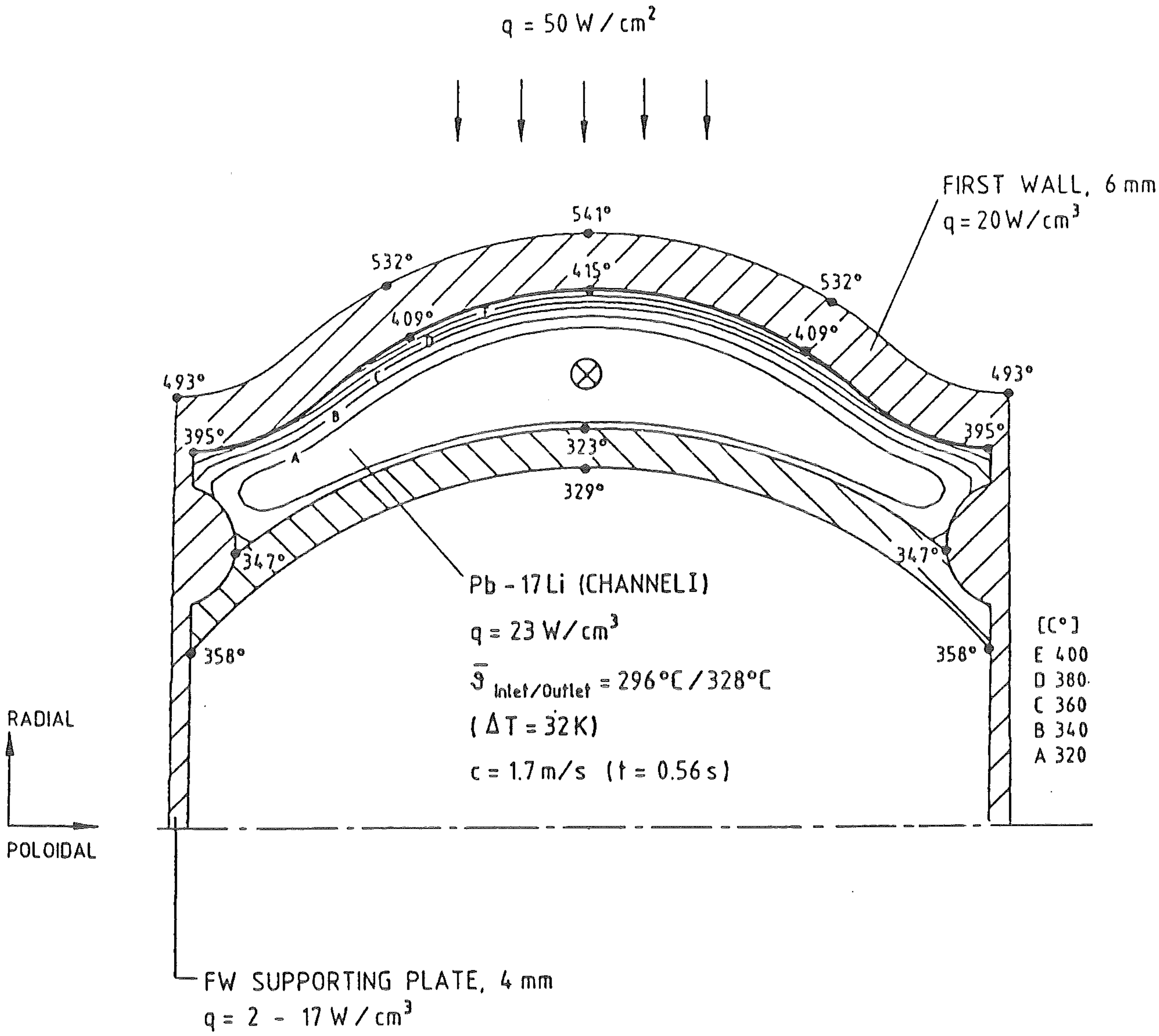


Fig. 2.1-6 Temperature distribution of the first wall cooling channel.

ture rises sharply primarily due to the effect of the surface heat flux. The maximum temperature at the first wall-to-coolant interface is 415 °C. Accross the first wall of 6 mm thickness a large temperature difference of $\Delta T = 126$ K occurs which results in a maximum first wall temperature of 541°C at the plasma facing surface.

The stress analysis which has been previously carried out for the first version of the self-cooled DEMO-blanket [18] (Fig. 1-3, see also Section 2.2) with respect to the ASME Boiler and Pressure vessel code [19] has shown that a considerable amount of safety margin exists for the stresses. The limit for the internal pressure loading has been found to be 6.5 MPa.

Compared with the results for that version using a beryllium multiplier, the maximum first wall temperature is only a little higher. With the unchanged temperature difference in the first wall the secondary stress should not be higher, and there are considerable reserves in strength. Because of taking away the beryllium the maximal temperature of steel structure is at 420 °C much lower than for the previous version. Therefore, the limit on the internal pressure in the first wall cooling channel at the torus midplane will be at least 6.5 MPa, the value determined for the first design. The maximum liquid metal pressure at this location is composed of 1 MPa static pressure and less than 1.5 MPa MHD pressure drop between torus midplane and coolant outlet. Therefore, the safety margin of at least 4 MPa for additional loads, i.e. caused by plasma disruptions, is very large.

2.2 Alternative Blanket Designs

Self-cooled liquid metal blankets are especially difficult to design for the inboard region of the torus. In this region the strength of the magnetic field is roughly 50% higher than in the outboard region and the space available for the blanket is more limited. Therefore, the resulting MHD pressure drop in the inboard blanket segment is the critical issue. One possibility to avoid this problem is to use a thick layer of beryllium in the front part of the outboard blanket. The neutron multiplication in this material increases tritium breeding so much that tritium self-sufficiency can be achieved without the use of breeding blankets at the inboard side of the torus. This design is described in [7]. There are water-cooled steel reflectors arranged at the inboard region. The flow principle in the outboard segments is identical to the one employed in the reference design with first wall cooling in toroidal direction and meander-shaped channels in the front region. The beryllium plates with a thickness of roughly 300 mm are fabricated in two halves, both are canned with a 0.5 mm thick steel sheet after oxidizing the entire beryllium surface. Beryllium oxide is an excellent electrical insulator, therefore, it decouples the liner from the beryllium plates. This insulation is necessary to avoid excessively large currents flowing perpendicular to the plates which would cause the multi-channel problem as explained in Section 4.2.

A neutronics analysis has shown [20] that a 300 mm thick beryllium multiplier in the outboard blanket together with a steel reflector at the inboard side leads to roughly the same tritium breeding ratio as breeding blankets without beryllium at both the outboard and inboard side. The alternative for self-cooled Pb-17Li blankets is therefore either to find solutions for the MHD-problems encountered in designing inboard systems (for example split the inboard segment into two halves, see reference design) or to use beryllium in the order of 200 tons for a DEMO-reactor (alternative design).

A completely different approach is to combine helium-cooling of the first wall with self-cooling of the breeding zone of a Pb-17Li blanket. A segment cross-section of such a dual-coolant concept is shown in Fig. 2.2-1. This concept is characterized by

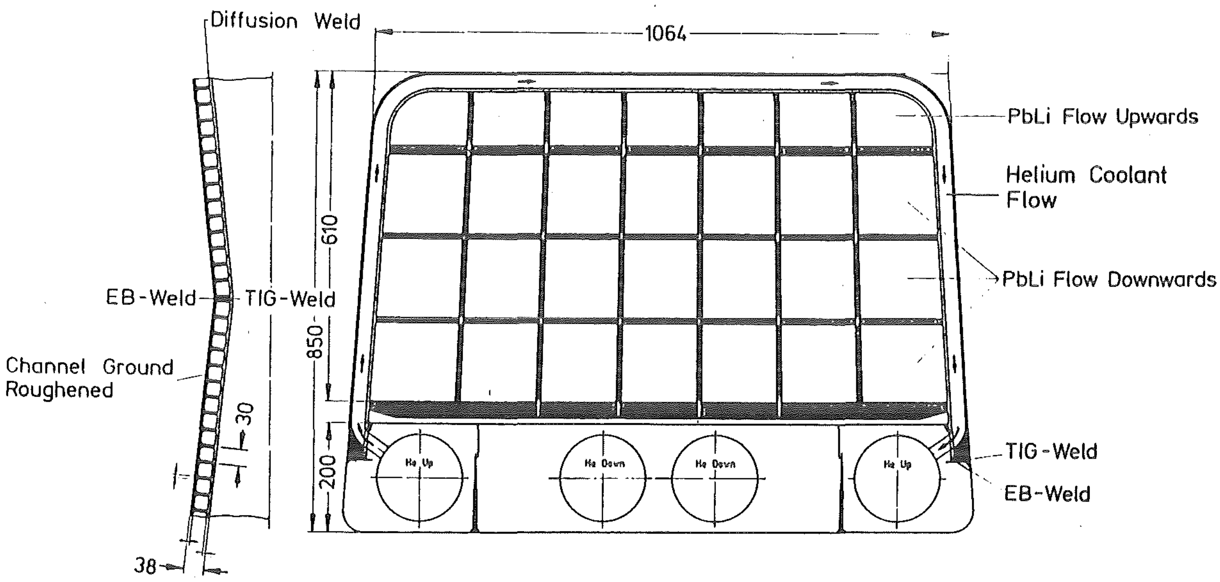


Fig. 2.2-1 Self-cooled liquid metal breeder blanket with helium-cooled first wall.

a stiff first wall box with rectangular cooling channels in toroidal direction. Connected to this box is a grid of steel plates forming large ducts for liquid metal cooling in poloidal direction. This is a very novel design not yet analysed in details. The main advantages are a real double containment of the liquid metal, a more simple geometry and much less problems with liquid metal cooling. Most of the MHD-problems in self-cooled liquid metal blankets are caused by the surface heat flux to the first wall requiring relatively high liquid metal velocities. This is avoided here by using gas-cooling which results in much lower temperatures at the steel/liquid metal interface and decisively lower liquid metal velocities. First estimates indicate that this concept is feasible even without splitting the inboard segment into two halves and that the breeding ratio will be roughly the same as for the reference design. The disadvantage is the need for a second coolant with separate ancillary systems. On the other hand, there is no auxiliary cooling system required for after heat removal since this is possible either with the helium cooling of the first wall or with the liquid metal cooling of the breeding zone.

2.3 Ancillary Loop System

The ancillary loops are required for the heat and tritium removal from the blanket segments. 80 loop systems are proposed for DEMO, 48 for the outboard blankets and 32 for the inboard blankets. The principle operation of an ancillary loop system for the selfcooled PbLi blanket is as follows:

The PbLi is circulated by a pump through the blanket, the heat exchanger (steam generator) and a purification system. compare Fig. 2.3-1. Fig. 2.3-2 shows the selected steam generator design. This heat exchanger has two functions: first the cooling of the PbLi in order to produce steam for electrical power production and, secondly, the removal of the bred tritium from the PbLi using NaK as a secondary fluid. The tubes in the heat exchanger are double-walled with water (steam) in the inner tube, PbLi surrounding the outer tubes and NaK in the gaps between both walls. While the heat flows from the PbLi through both walls and the NaK gap to the water, the tritium permeates only the through outer wall and is dissolved in

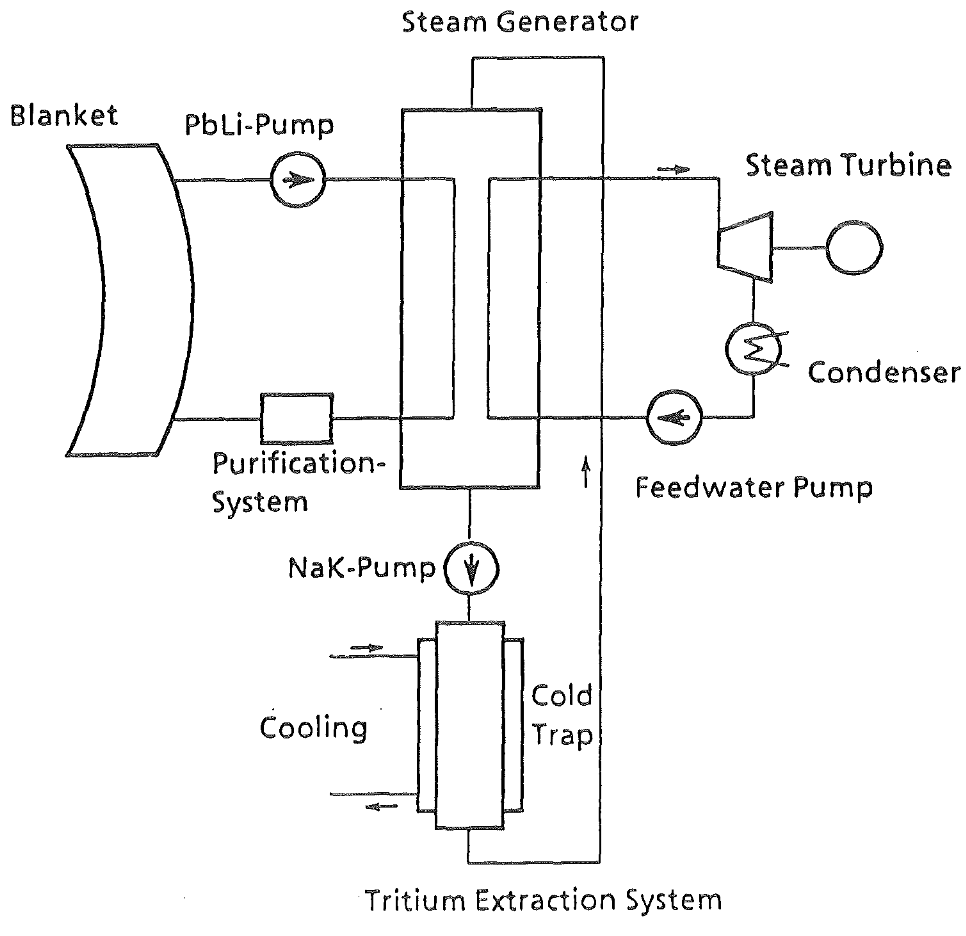


Fig. 2.3-1 Systems for heat and tritium extraction from self-cooled Pb-17Li blankets

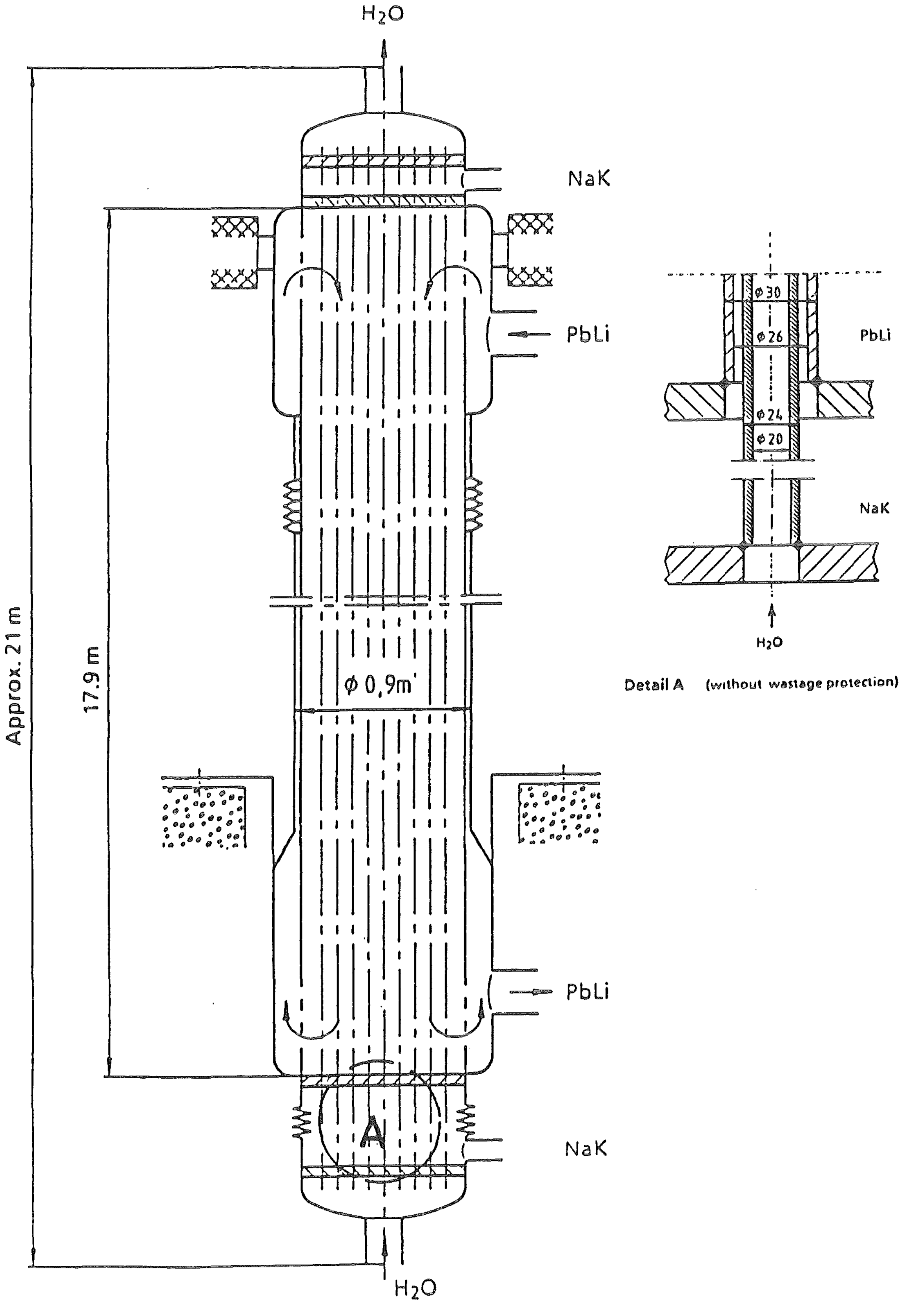


Fig. 2.3-2 Straight tube double-wall heat exchanger (steam generator) for DEMO.

the NaK. The NaK is circulated with a low velocity by means of a magnetic pump from the heat exchanger through an economiser to one of the cold traps, compare also Fig. 2.4-1, for details see Section 2.4.

The purification system has the function of removing the corrosion products, oxides and other contaminants from Pb-17Li in order to reduce the radioactive contamination of the loop components by activated corrosion products or even to prevent plugging of the pipes.

2.4 Tritium Removal and Recovery

The requirements on the blanket tritium removal and recovery system are a low tritium inventory in the total blanket system and an acceptable tritium loss through the steam generator into the water loop. The latter requirement is the crucial one for a Pb-17Li blanket due to the low tritium solubility of Pb-17Li. In the present design a tritium loss of 20 Ci/d for all reactor blankets is assumed.

The selected technique includes the following steps [21]:

- tritium permeation into the NaK-filled gap of the double-walled heat exchanger
- tritium removal from the NaK by precipitation as potassium tritide in a cold trap
- tritium recovery by thermal decomposition of the tritide and pumping off the tritium gas.

Figure 2.4-1 shows schematically the flow sheet for one blanket segment : two cold traps are operated in parallel: one for tritium removal by circulating the tritium dissolved in the NaK to the cold trap; the other for tritium recovery. For this purpose the cold trap is decoupled from the circulation loop, drained from NaK, heated up to temperatures of about 380°C and the released tritium gas is pumped off and stored in a getter bed.

The critical design value is the tritium pressure in the NaK-filled gap which must be so small that the tritium loss does not exceed the value given above. This tritium pressure depends on the permeation barrier which preferentially occurs at the water side of the steam generator. The following numbers are valid for a permeation barrier factor of B = 100 (which means that the permeation rate is decreased by a factor of 100 compared to ideal permeation conditions). Factors between 10 and 100 are reported for natural oxide layers at the water side but taking into ac-

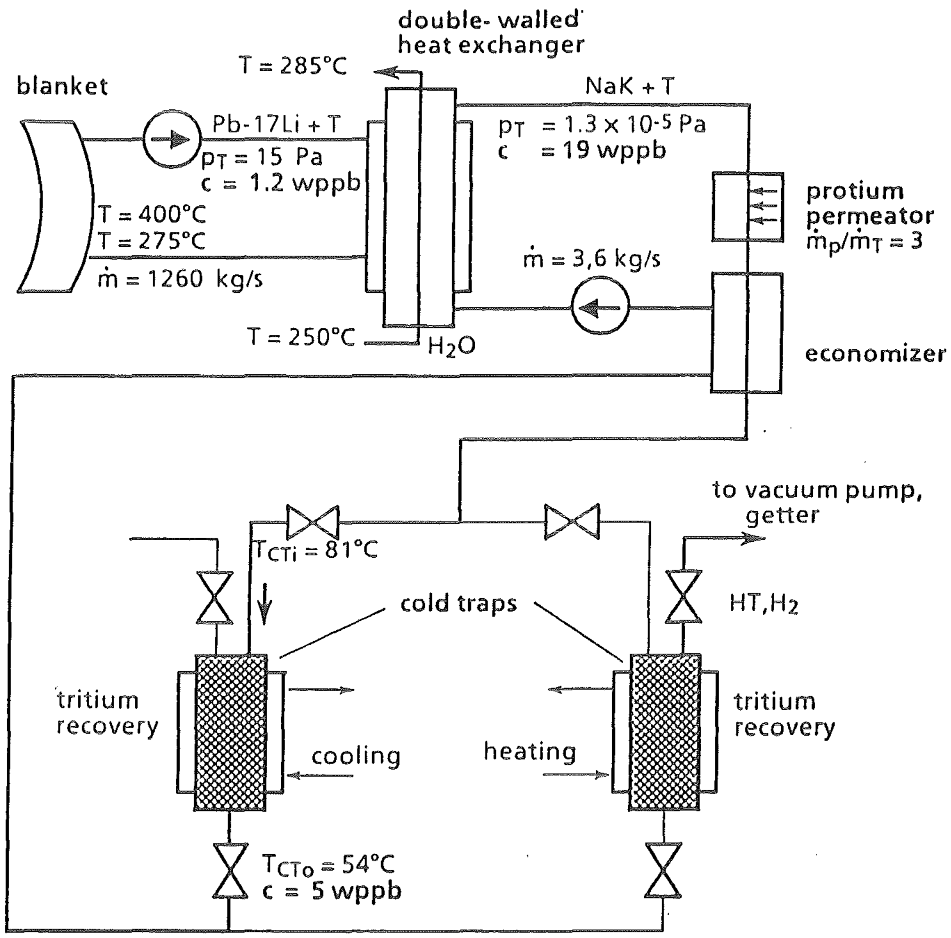


Fig. 2.4-1 Tritium flow sheet for a DEMO blanket segment

count the extensive work on the development of permeation barriers, compare e.g. [22], a value of B = 100 appears to be feasible for the time when the DEMO reactor will be built. In Table 2.4-1, which shows the values of the tritium removal system for one outboard segment, B is considered as a parameter and results for other values are also presented.

2.5 Reliability and Safety

Reliability

This investigation has been limited to the blanket cooling system. The reliability of the blanket segments will be investigated later. A study based on the quality standard achieved by the nuclear fission industry in manufactoring of tubes, welds and so on [23] indicates clearly that the main source of unavailability of the cooling system is leaks in the steam generator tubing. Such leaks would lead to at least 30% unavailability of the system if there is no redundancy. Redundancy of

Table 2.4-1 Tritium Recovery System of a DEMO Blanket Segment

permeation barrier factor B(1)	50	100	500	1000
tritium production rate $\dot{m}_T$(g/d)	5.05	5.05	5.05	5.05
mass flow rate in $\dot{m}_{NaK}$(kg/s)	14.0	4.0	1.0	0.5
tritium concentrations				
steam generator C_{SG}(wppb)	6.1	12.1	60.1	121
cold trap inlet C_{CETi}(wppb)	8.2	19.4	89.9	182
cold trap outlet C_{CTo}(wppb)	4.0	4.8	31.5	60.5
isotopic swamping IS = $(mol_p + mol_T)/mol_T$	10	10	1	1
cold trap inlet temp. T_{CTi}(°C)	64	81	66	79
cold trap outlet temp. T_{CTo}(°C)	<51	<54	<47	<58
cold trap dimensions				
volume V_{CT}(m^3)	4.11	1.17	0.29	0.14
height h_{CT}(m)	1.5	1.5	1.5	1.5
diameter D_{CT}(m)	1.0	1.0	0.5	0.4
number n(1)	3	1	1	1
tritium inventory				
in steel J_s(g)	0.10	0.10	0.10	0.10
in PbLi J_{PbLi}(g)	0.26	0.26	0.26	0.26
in NaK J_{NaK}(g)	0.05	0.06	0.27	0.53
in cold traps J_{CT}(g)	2.5	2.5	2.5	2.5

steam generators and pumps can be achieved by connecting neighboring loops in a way that a failed component can be disconnected from the blanket segment and the load can be shifted to the remaining loops by operating of valves.

Linked together are the loops of the three outboard segments and the two loops of the inboard segments in one torus sector. More combinations of loops are not allowed due to requirements of electrical insulation. By this measure the availability of the machine can be increased by one order of magnitude.

A functional analysis of the blanket system indicated the main safety issues. With scoping studies the removal of afterheat by natural convection, possible NaK-water reactions, and the generation of polonium by neutron irradiation have been investigated.

Afterheat removal

For the investigation of afterheat removal by natural convection it has been assumed that immediatly after a pump outage the plasma stops burning but the toroidal magnetic field further exists and that the flow velocity of the PbLi-flow is zero. It has been further assumed that the heat sink is still intact. The elevation difference H between the vertical middle point of the blanket and of the steam generator has been varied in the calculation as a parameter. Fig. 2.5-1 shows the afterheat production as a function of time and the calculated blanket outlet temperature and the coolant velocity for the case of H = 5 m. The diagram shows that for H equal or greater 5 m natural convection is sufficient to prevent blanket overheating.

NaK-Water reactions

If NaK comes into contact with water in case of a leakage in the inner steam generator tube walls it will react violently. In the double-walled steam generator the zone where this reaction can occur is very small in the region of the narrow gaps. Therefore, no damage propagation is expected. In the region of the NaK collectors at both ends of the gap the outer tube is prolongated forming a protection sleeve for the adjacent pipes in order to prevent wastage by NaK water reaction. The

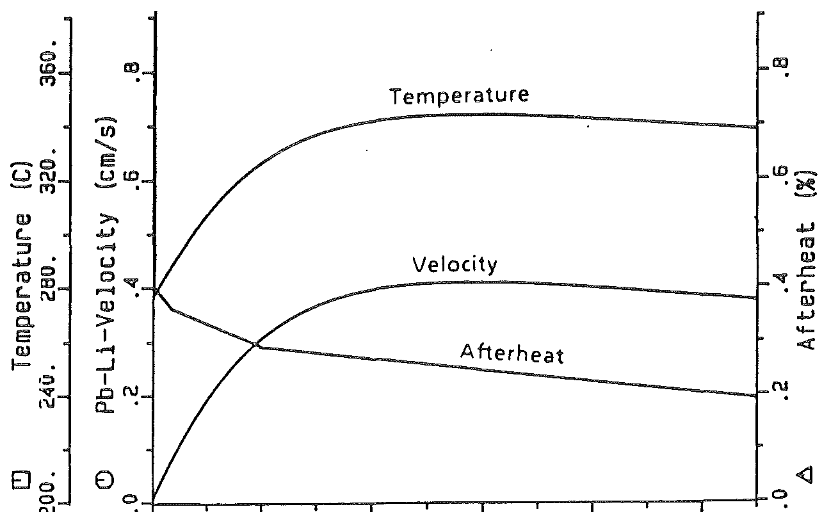


Fig. 2.5-1 Afterheat removal by natural convection

NaK system inclusive the cold traps must be designed to withstand a short reaction shock wave of 100 bar.

Polonium production

The polonium production in the blanket segment has been determined by neutronic calculations. Po-210 is an α-emitter with a half-life of 138 days. It is produced by neutron capture in Bi-209 which is both an original constituent in lead and a transmutation of lead.

A three-dimensional Monte Carlo transport calculation with the MCNP-code [9] is performed to obtain the spatial and energetic neutron flux distribution. For this purpose a 7.5°/2 torus sector model is used with a simplified layout of the self-cooled liquid metal blanket. The neutron spectra are calculated in the 100 energy groups of the GAM-II-structure. In order to obtain a sufficient statistical accuracy for the calculated neutron spectra about 20000 neutron histories are followed.

It has been assumed for these calculations that the liquid metal breeder is not moved during the total irradiation time of a blanket segment which is specified to 20000 h. In the real case the liquid metal in the blanket segments is circulated in the primary loops, leading to both an increased liquid metal inventory and a complete mixing of breeder material from front and back regions. Both effects are not considered here and would lead to a reduced polonium production rate.

The total amount of Po-210 generated in the blanket segments containing $5 \cdot 10^6$ kg Pb-17Li is listed in Tab. 2.6-1. The table shows that the initial bismuth content is of nearly no influence on the total production of polonium as long as this value is not higher than 10 ppm. Estimates have shown [45] that the accidental release of Po-210 to the environment must be limited to values below 0.01 g. This implies, that for a Po-210 inventory of 100 g the allowable release fraction would be 10^{-4}. There are indications that realistic release fractions are even lower than this value but more experimental work is necessary to confirm this. Much lower polonium production could be achieved if it is possible to extract bismuth on-line. If, for example a concentration of 10 ppm can be maintained, the production of polonium can be reduced by a factor of 10 compared to a case where the bismuth concentration is allowed to rise.

Tab. 2.6-1 Polonium production during the blanket life time

	initial Bi-content [appm]				Bi-content [appm] maintained by on-line purification			
	0	1	10	100	0.01	0.1	1	10
specific Po-210 activity [Bq/kgPb]	$2.9 \cdot 10^9$	$3.0 \cdot 10^9$	$3.1 \cdot 10^9$	$6.4 \cdot 10^9$	$0.3 \cdot 10^6$	$0.3 \cdot 10^7$	$0.3 \cdot 10^8$	$0.3 \cdot 10^9$
Po-210 inventory [g] in $5 \cdot 10^6$ kg Pb-17Li	87	89	92	190	0.009	0.09	0.9	9

3. Test Object Design for NET/ITER

One of the major objectives of NET/ITER is to test blanket concepts relevant for a DEMO-reactor [24]. NET/ITER offers the unique possibility to test simultaneously all aspects of such blanket concepts in the real geometrical configuration, with the real magnetic field, and with an incident neutron flux having the real neutron spectrum and spatial distribution. The main differences to DEMO are the lower wall load (1.0 MW/m^2 instead of 2.2 MW/m^2) and the shorter burn time. The lower power is the main reason why fully integrated tests with look-alike test objects are not really suitable to investigate all issues of a blanket concept simultaneously. Therefore, it has been suggested in the FINESSE-study [25] to use act-alike test modules instead of look-alike test modules in order to obtain maximum benefit from testing at reduced device parameters.

There is a large number of issues involved in a blanket concept. The most critical ones for self-cooled liquid metal blankets are:

- MHD pressure drop and flow distribution
- electrical insulation of the flow channels either by flow channel inserts or by direct insulation
- potential chemical reactions between the liquid metals (Pb-17Li, NaK) and air, water, or concrete
- response of blanket segments to plasma disruptions.

All these issues either require tests in NET/ITER or have to be taken into account in designing test objects, ancillary loops and interfaces to the basic machine.

3.1 Test Module Design

Different test objectives require dedicated test modules. The basic machine will be designed for a frequent replacement of test modules. For this purpose a number of blanket test ports are allocated at the equatorial midplane, each of them roughly 3 m high and 1 m wide. In order to cope with the very limited testing time, it is anticipated to divide a test port into up to four sub-ports for a number of tests. Figure 3.1-1 shows such a quarter-sized test module which is designed for exchanging it completely independent of the neighbouring sub-modules. The module shown in this figure is located behind a first wall provided by the basic machine. This is anticipated for all tests during the physics phase of NET/ITER operation and for the

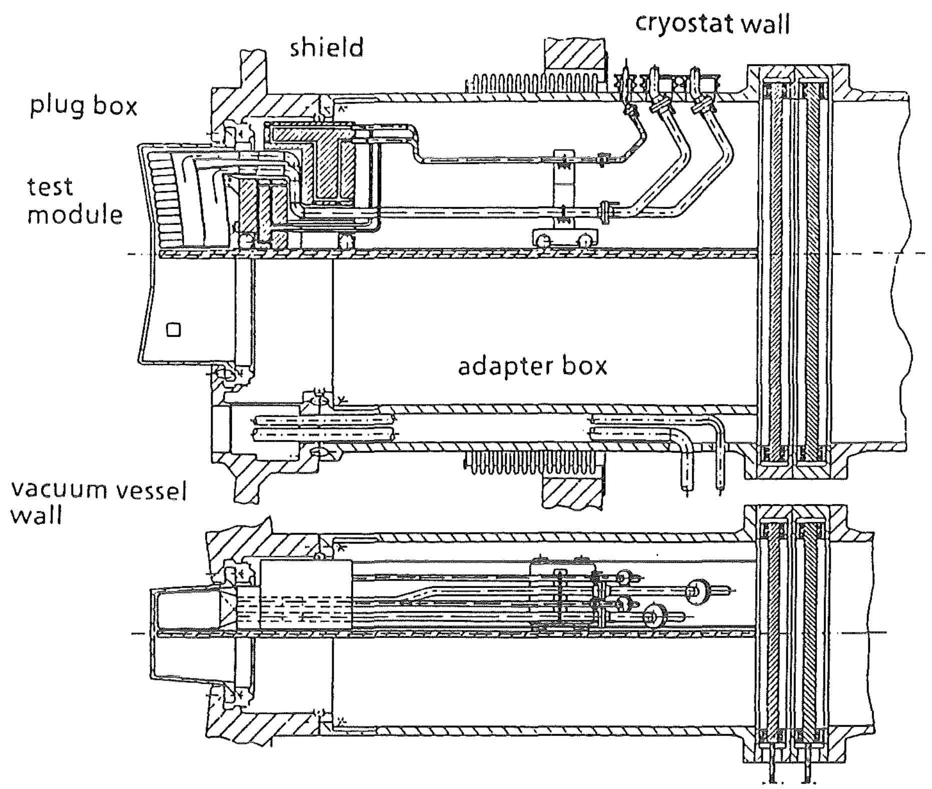


Fig. 3.1-1 Test module for the self-cooled Pb-17Li concept

first part of the technology phase respectively in order to limit the risk to the machine caused by blanket tests. Exposure of the test modules to the plasma required for final tests of a blanket concept will be allowed only after extensive testing of the concept behind the first wall.

3.2 Ancillary Loop System

The goal of the NET/ITER ancillary loop system is the cooling of the test module and the removal and recovery of the bred tritium. Fig. 3.2-1 shows the schematic of the NET/ITER-ancillary loop system, Fig. 3.2-2 the arrangement of the main components.

Because NET/ITER will be the first fusion plant with a burning plasma the operational risk of the ancillary loops should be as low as possible. Conventional and well approved components are, therefore, chosen.

Two heat exchangers of the KNK type with an intermediate NaK loop are the main components. These components with a long time operation experience promise a good reliability of the loop system.

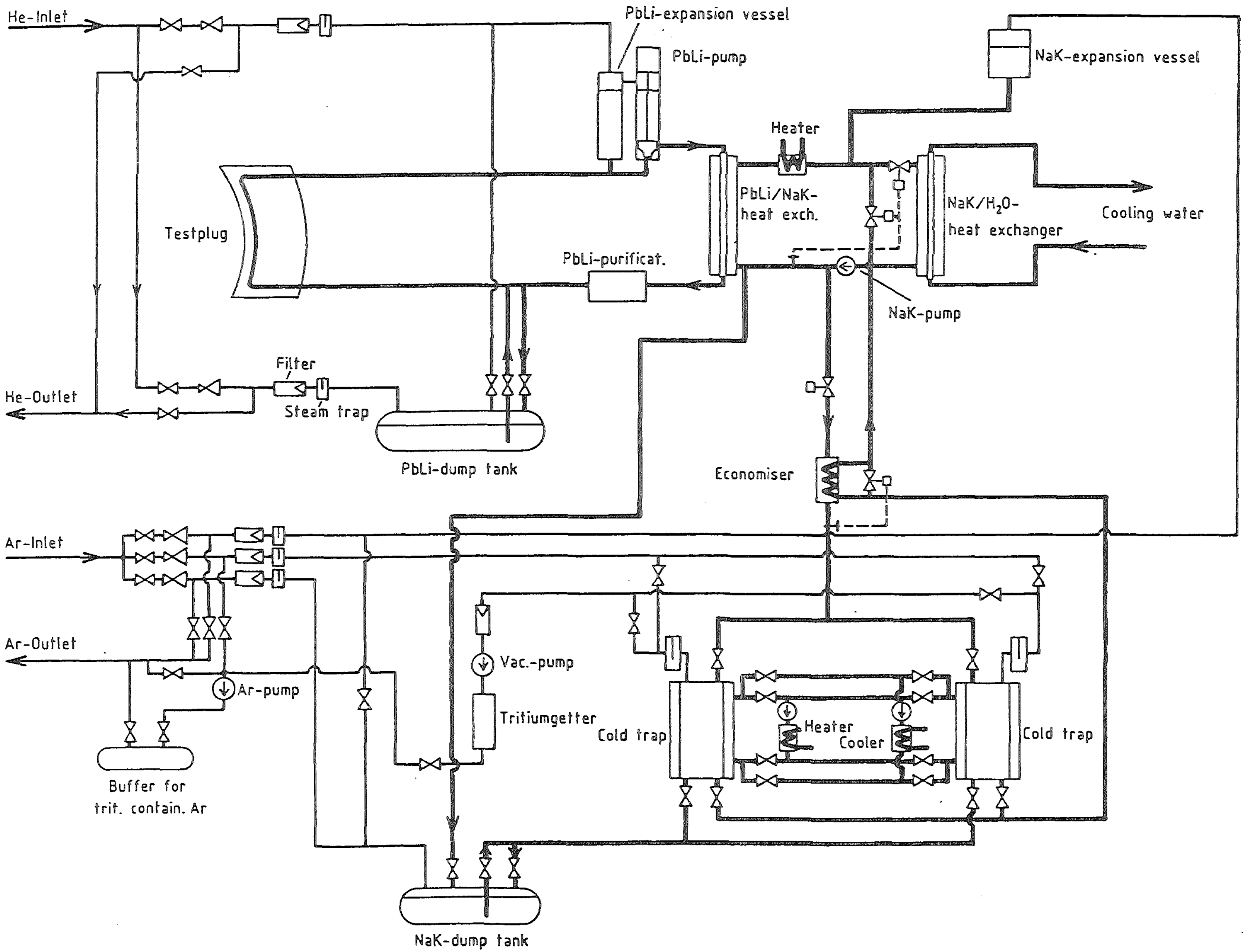


Fig. 3.2-1 Ancillary loop system for testing self-cooled Pb-17Li blankets in NET/ITER

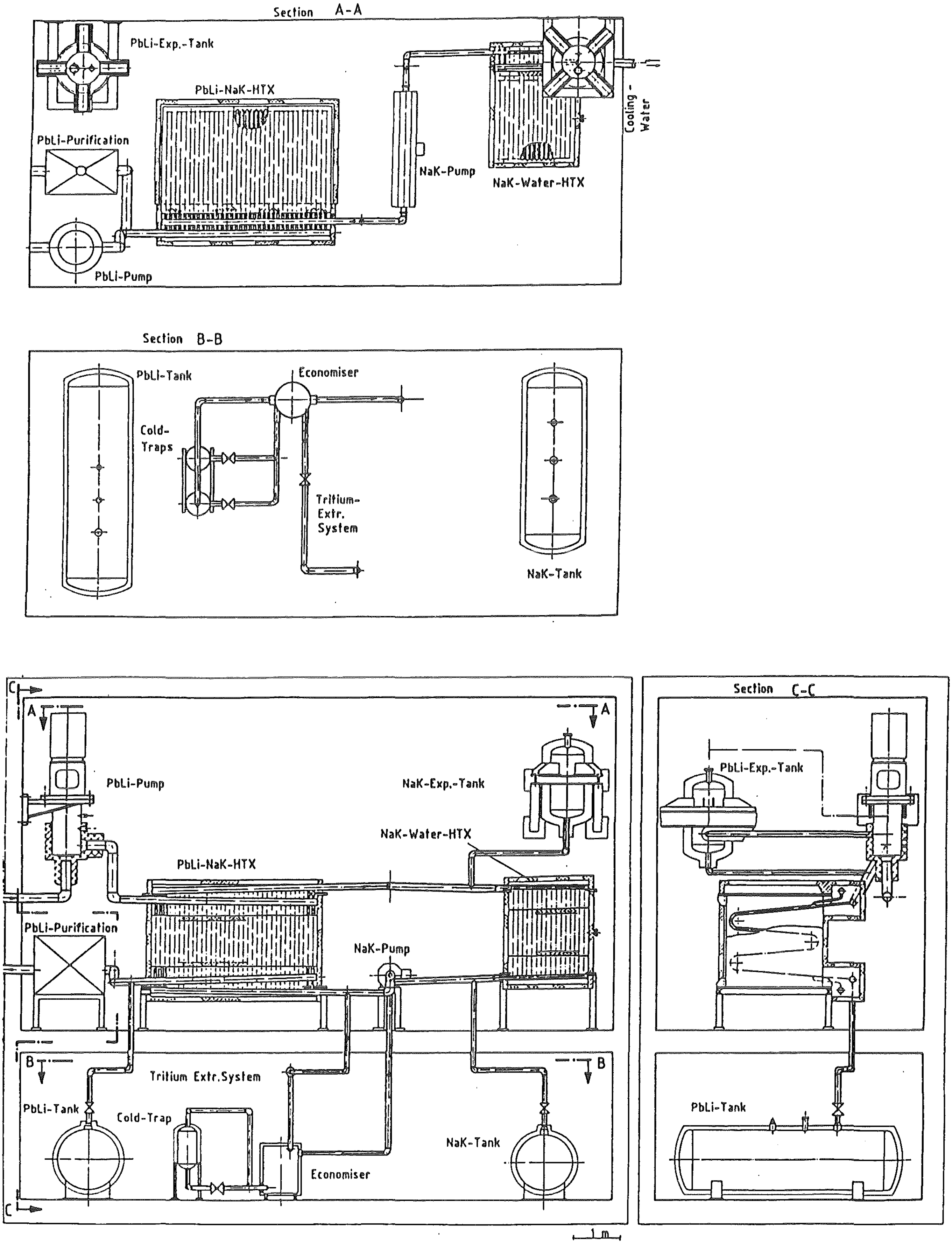


Fig. 3.2-2 Ancillary loops for self-cooled Pb-Li blanket test module

The liquid PbLi is circulated with a one-stage free-surface circulation pump through the test plug, the first heat exchanger and the purification system. The heat and the tritium are transferred through the heat exchanger from the PbLi into the NaK of the intermediate loop. This loop is necessary here in contrast to DEMO due to the division of the heat exchanger in two components. An electromagnetic pump is used for the NaK flow circulation. The second heat exchanger is transferring the heat to the site cooling water system without steam production.

The tritium removal and recovery system for NET/ITER principly works as described for DEMO (Section 2.4). This system is located in a bypass of the intermediate loop. The mass flow rate to this system is only 0.15 kg/s which is about 0.2% of the intermediate loop NaK flow.

3.3 Tritium Build-up in the NET/ITER Test Object System

Before equilibrium measurements can be performed in respect to tritium transport, inventory and removal efficiency, the fusion reactor has to operate for a certain time period to build up the required tritium concentrations in the NET/ITER test object and the ancillary system. Steady-state conditions are reached approximately after 100 hrs if the cold traps are in operation from the beginning. A final tritium inventory in the liquid metals and structural materials of about 1.8 g is obtained. The additional tritium inventory in the cold traps during the testing period is about 0.5 g (assuming two recovery cycles per day).

One method to reduce the operational time of the reactor is to start cold trap operation only when the steady-state concentration in the secondary loop is reached. With this, the required time period then becomes about 20 hrs.

Another method to reduce reactor operation time would be to add tritium to the liquid metal loops (essentially to the NaK-loop) by permeation through a special permeator unit. This could be done by heating electrically the circuits without reactor operation at all. Then, reactor operation would be mainly required only for the testing period.

3.4 Building, Heat Sink and Handling Requirements of the Blanket Test Programme

Blanket testing will be performed through horizontal access ports around the torus. There will be a separate heat extraction system for each test module. This implies that, if for example a test port is shared by four sub-modules, space has to be provided for a number of independent heat- and tritium extraction systems. It has been estimated [24] that the space requirement for four sub-modules amounts to 300 $m^2 \times 11$ m. This space has to be provided as close as possible to the test ports in order to minimize the length of liquid metal pipes between test module, primary heat exchanger, and secondary heat exchanger. Short connecting pipes are necessary to minimize the liquid metal inventory for safety reasons and to reduce the time required to achieve tritium equilibrium conditions. Space allocation close to the test port is of less importance for gas- or water-cooled blanket test modules.

An important issue is the tritium permeation rate to the cooling water in the secondary heat exchanger. The present estimate is, that this permeation loss will be limited to a value below 20 Ci/day for a 5 MW test module. This value may be tolerable since a closed water loop is required in any case. Tritium recovered during regeneration of the cold traps is of high purity. It can be transferred to storage getters without any additional interface system.

The handling equipment has to be designed for a frequent exchange of test modules having a weight of up to 10 tons. Test port design and handling equipment should allow for replacement of one sub-module without interfering with the neighbouring sub-modules. Double doors between test port and transfer flask as shown in Fig. 3.1.2 will facilitate the module exchange without opening the test port to the building hall. This is especially important for modules exposed to the plasma.

4. Status of the R&D Programme

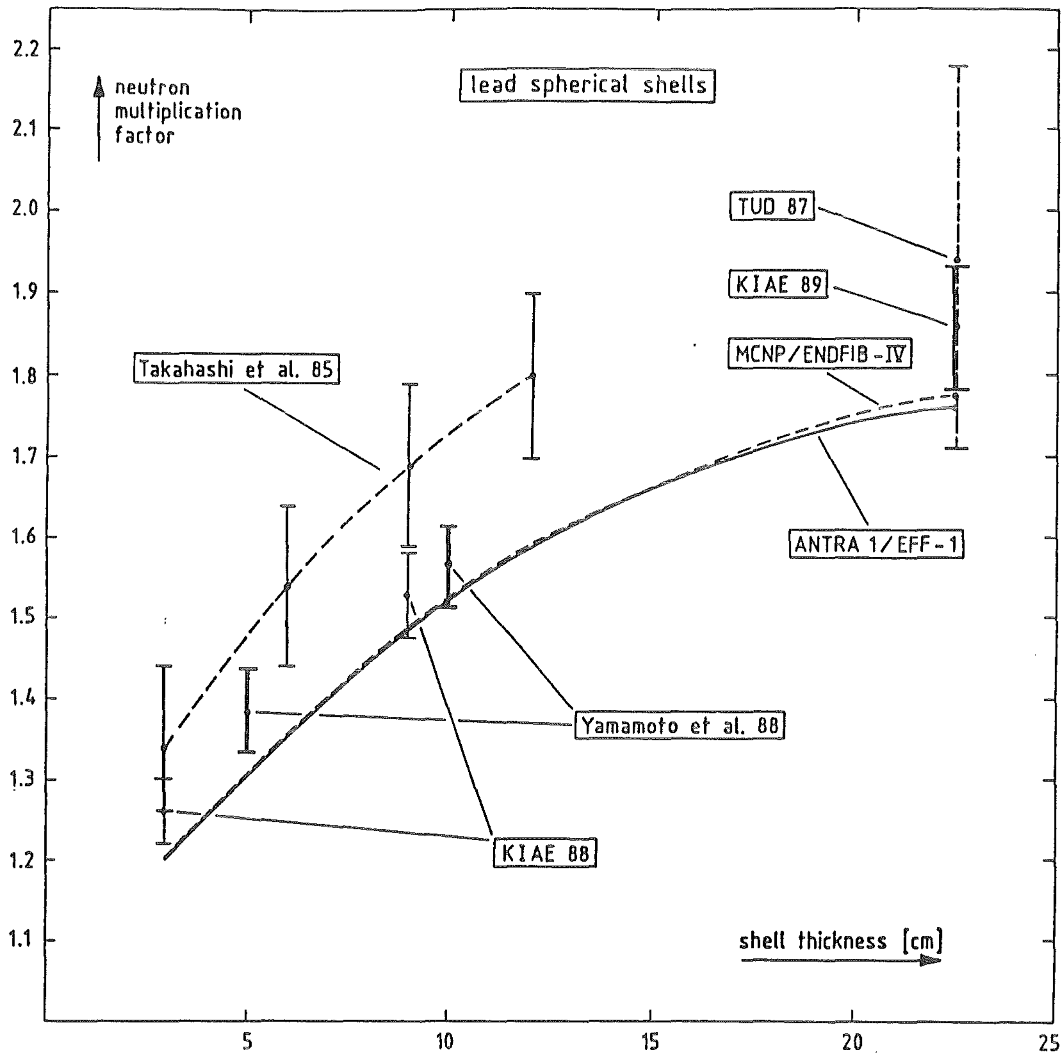
The chapters "Blanket Design for a DEMO-Reactor" and "Test Object Design for NET/ITER" concentrated on selected design aspects. These aspects, however, are based on an extensive R+D effort to provide the theoretical and experimental base. Additionally the scope of the R+D is much more comprehensive and allows alternatives which are also described in the sections 4.1 to 4.7 giving the state of the art.

4.1 Neutronics: Methods and Data

The neutronic calculations for the blanket layout are performed with the Monte Carlo transport code MCNP and nuclear data from the European Fusion File EFF-1 (see Section 2.1.2). In principle there are no restrictions imposed on the quality of the Monte Carlo transport procedure: the accuracy of a specific calculational quantity depends on the number of events contributing to this quantity and, of course, on the involved nuclear cross-section data. Therefore, the accuracy of the calculated quantities (reaction rates, flux densities, power densities etc.) mainly is limited by the accuracy of the applied nuclear cross-section data.

With regard to tritium breeding in the Pb-17Li liquid metal blanket, the most important nuclear interaction processes are the 6Li (n, α)t- and the Pb (n, 2n)-reaction. The nuclear cross-section of the 6Li (n, α)t-reaction is rather well known and the (n, 2n)-cross-section of Pb, therefore, is the main source of data uncertainties with respect to the uncertainty of the TBR-calculation in the Pb-17Li blanket. Considerable effort has been spent on improving the knowledge of the neutron interaction processes in lead: integral 14 MeV neutron multiplication experiments in spherical geometry have been performed and have been analysed by using various computational approaches and data evaluations. The lead spherical shell experiment performed at the Technical University of Dresden (TUD) [26] has been the subject of an extensive benchmark task organized by the IAEA. As a result of these activities it can be stated that satisfactory agreement between experiments and calculations has been achieved for the neutron multiplication factors of lead.

Discrepancies between calculated and measured neutron multiplications factors found in the past, mainly could be traced back to large uncertainties in the integral experiments (see Fig. 4.1-1). Therefore, the Pb (n, 2n) cross-section is well established now at 14 MeV neutron energy; there, however, are serious doubts about



4.1-1 Neutron multiplication factors of lead spherical shells with a central 14 MeV neutron source

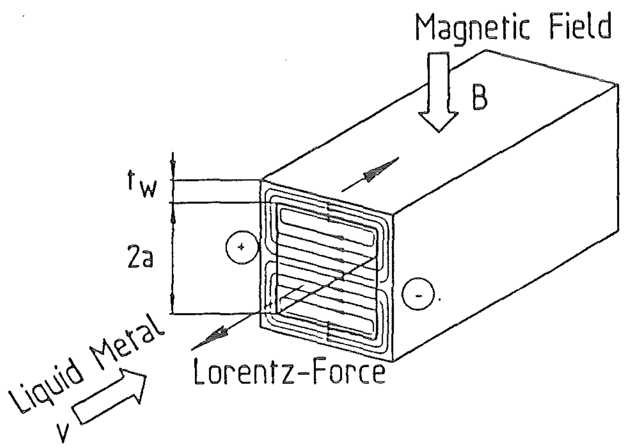
the shape of its excitation function, i.e. the energy-dependent (n, 2n)-cross-section between the reaction threshold and 14 MeV. Actually this has a significant impact on the neutron multiplication power of a liquid metal Pb-17Li blanket, as the neutron spectrum in the blanket is degraded. Therefore, new (n, 2n)-measurements between 7 and 14 MeV are strongly recommended.

The analysis of 14 MeV neutron multiplication experiments requires the proper description of the transport phenomena in the multiplying assembly. In case of lead this necessitates the use of double-differential cross-sections (DDX-data). The most appropriate way to use DDX-data in SN-transport calculations is to avoid the Legendre series expansion of the scattering kernel. This can be achieved by using explicitely angular dependent scattering matrices along with an appropriate numerical integration scheme in the transport calculation. At KfK such a computational procedure has been developed, both for one- and two-dimensional transport problems [27], and has been validated in the framework of the lead spherical shell benchmark analyses [28]. It has been shown that the newly developed transport programmes are appropriate computational tools for benchmarking methods and data describing the 14 MeV neutron transport. Furthermore, it has been shown that there is a need for using DDX-data in the transport calculations and the Monte Carlo code MCNP, therefore, should be enabled to handle DDX-data, as it is the main computational tool for the neutronic blanket layout.

4.2 Magnetohydrodynamics in Self-cooled Liquid Metal Blankets

The design of liquid metal cooled blankets is domianted by MHD-considerations. The magnetic field causes large pressure drops, influences the velocity profiles and is the reason for multi-channel effects if the liquid metal flows in parallel electrically connected ducts. Usually heat transfer is degraded by a suppression of turbulence [29]. Electrical insulation of the duct walls may be required to reduce MHD pressure drop and the multi-channel effects.

Some of the issues are well understood and can be described by theoretical models. An example for this class is the pressure drop in single ducts caused by two-dimensionally flowing currents. A typical case is a conducting straight duct, shown in Fig. 4.2-1.



$$\Delta p = L \cdot B^2 \cdot v \cdot \sigma_w \cdot \frac{t_w}{a}$$

$$\Delta p \sim t_w$$

mechanical stresses in cylindrical vessels

$$s \sim 1/t_w$$

Fig. 4.2-1 Pressure drop in a conducting straight duct

Liquid metal flows in this duct perpendicular to the magnetic field which induces a potential difference across the channel. The result is an electrical current flowing two-dimensionally in the fluid and in the duct walls which causes so-called Lorentz-Forces. The relationship listed in the figure indicates that the pressure drop required to overcome these forces is proportional to the wall thickness. The mechanical stresses in a pressure vessel are approximately proportional to the reciprocal value of the wall thickness. The concequence is that stresses caused by MHD pressure drop cannot be reduced by increasing the wall thickness. This is a feasibility issue of self-cooled blankets.

The pressure drop, however, is not the only critical issue of liquid metal flow in a magnetic field. In more complicated geometries, for example in bends, there are three-dimensional currents influencing the velocity field and causing additional pressure drop.

These currents flow mainly inside the liquid metal and cannot be avoided by insulating the duct walls. The resulting velocity profiles can cause cooling problems in the region of bends especially if they are located near the first wall. Figure 4.2-2 shows schematically the electric current and velocity profile in a bend.

On the other hand, it is possible to design liquid metal ducts in such a way to obtain a positive influence of the magnetic field on the velocity profiles (so-called flow tailoring) by creating high velocities near the first wall [30]. The investiga-

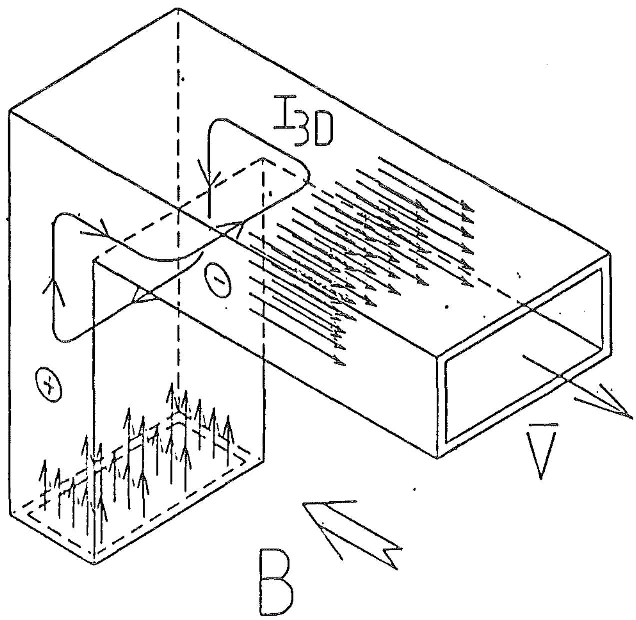


Fig. 4.2-2 Electrical current and velocity profiles in a bend

tion of the velocity field - both experimental and theoretical - is a very ambitious task but the results are essential for all liquid metal cooled blankets.

Significant progress has been made during the last years in modeling the flow in single ducts. There are two methods under development. One of them is called "Full Solution" because nonlinear inertial effects as well as viscosity forces are included. However, even for large computers, the application of the full solution is restricted to low Hartmann numbers and simple geometries. An alternative method is the "Core Flow Solution" first developed by ANL and the University of Illinois allowing numerical predictions in the relevant range of application.This second method is very promising for all cases where inertial and viscous effects can be neglected because it allows to model nearly all blanket relevant duct geometries and boundary conditions.

There are a number of experimental MHD-programs in progress investigating all kinds of duct geometries and boundary conditions providing the basis for a verification of computer codes.

The largest uncertainty exists in respect to the interaction between the flow in parallel electrically connected channels. Here, currents are not restricted to flow inside each channel, but so-called leakage currents may flow across the duct walls into the adjacent channels. One special case of this multi-channel effect is the BCCS design of a self-cooled blanket which has been analysed previously [32].

The region of the radial-toroidal-radial bends was modeled and a large current was predicted which flows perpendicular to the toroidal channels upwards at one side of the blanket and downwards at the other one as shown in Fig. 4.2-3. This large current loop is closed by the toroidal channels and causes a large pressure drop and an unequal partitioning of the total flow rate into the parallel channels.

If there is no electrical insulation between the parallel channels, the pressure drop caused by this effect will become exceedingly high and the flow distribution will cause cooling problems in some channels. Insulation in the region of the bends can improve the situation, but this issue is extremely difficult to investigate by theoretical models. Tests to investigate this issue are underway in the MEKKA-test facility at KfK and in the Latvian Academy of Sciences in Riga.

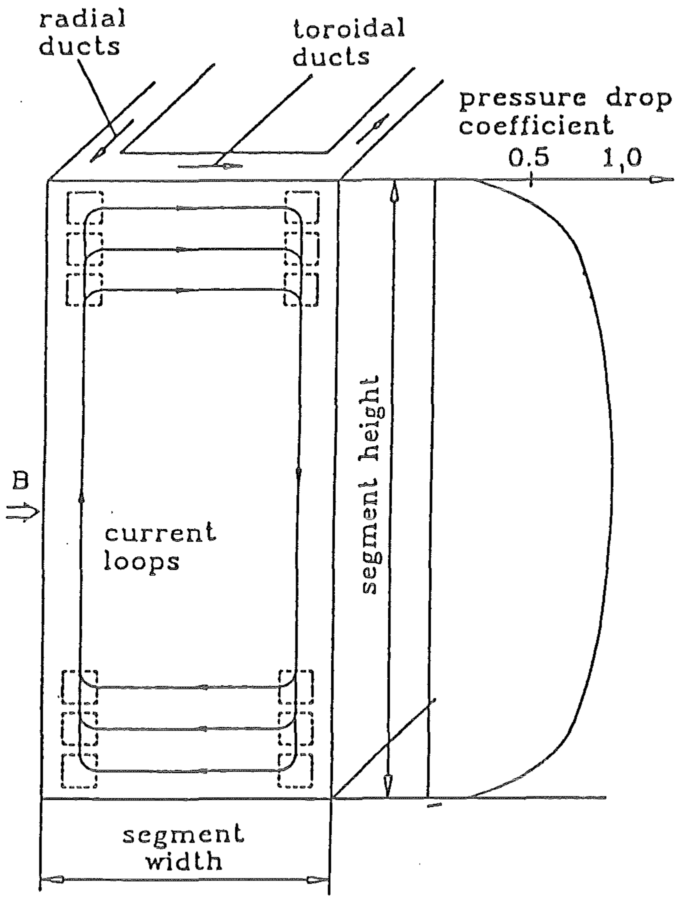


Fig. 4.2-3 Leakage currents and pressure drop distribution in a blanket segment employing a radial-toroidal-radial flow concept

Another open question is the possible improvement of heat transfer by turbulence. There is experimental evidence, that a special kind of turbulence ("two-dimensional turbulence") can occur under certain conditions. The mechanism of this turbulence and the degree of heat transfer improvement are not well understood at present. It remains to be seen, if the improvement will be large enough to enable a blanket design which is characterized by much simpler cooling channel geometries.

A crucial issue for all self-cooled blanket concepts is still the electrical insulation between the flowing liquid metal and the load-carrying walls.

Different methods of insulation are indicated in Fig. 4.2-4. The best solution would be a layer in direct contact with the liquid metal but this is a material problem, which is investigated in a new task.

The contact between insulator and the liquid metal breeder is avoided by the second and the third method indicated in the figure. In the laminated wall concept a steel liner is supported by the load carrying wall via a ceramic insulator. In the so-called flow channel inserts the ceramic layer is sandwiched between two steel

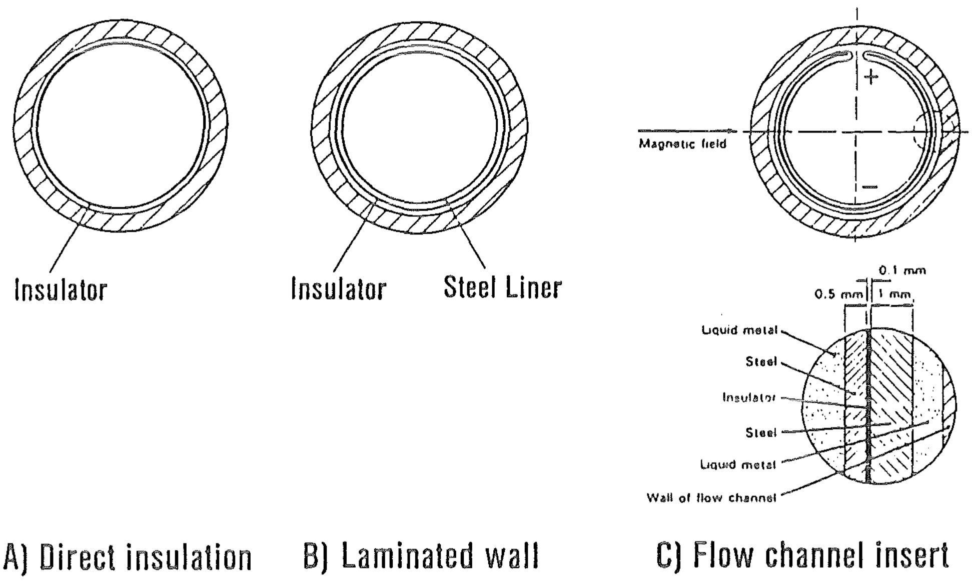


Fig. 4.2-4 Electrical insulation of duct walls

sheets which are welded at all edges. By the arrangement of these flow channel inserts, the load carrying walls are electrically decoupled from the voltage induced in the flowing liquid metal.

Progress has been made in fabricating flow channel inserts, but the behaviour of these inserts during thermal cycling and under irradiation is still a feasibility issue. The same statement is even more true for insulating coatings. There are promising candidate materials and methods available, but it remains to be seen if the problems can be solved.

4.3 Tritium Extraction and Recovery

Permeation through the tube walls of the steam generator

The favourized tritium removal and recovery technique is based on tritium permeation from Pb-17Li through the ferritic wall of the double-walled heat exchanger into the NaK-filled gap, tritium removal from NaK by cold trapping as tritide and tritium recovery by thermal decomposition of the tritide.

Surface layers, especially oxide layers can drastically reduce permeation. However, corrosion experiments of steel in Pb-17Li have shown that initial oxide layers are dissolved. In the NaK system, the chemical activity of NaK towards oxygen is also higher than that of the ferritic wall material and oxide layer cannot build up and will be dissolved if they exist initially. Therefore, surface layers are not likely to occur on both sides of the heat exchanger wall. An engineering assessment shows that the tritium concentration difference across the liquid metal boundary layers is also small compared to the bulk concentration.

Therefore, no serious difficulties are expected in the permeation process which require specific investigations in an early stage.

Tritium removal by cold trapping

Cold trapping is based on the fact that the saturation concentration c_{sat} decreases with decreasing temperature. When a liquid metal is cooled down in the cold trap, hydride crystals start to form below the corresponding saturation temperature. Ideally, the liquid metal leaves the cold trap with the concentration c_{osat}, corresponding to the saturation concentration for the cold trap outlet temperature T_o. In practice, this equilibrium is not reached due to nonideal mass transfer and the difference between the actual cold trap outlet concentration and the non-equilibrium value, c_o-c_{osat}, is of main importance.

Presently, a heterogeneous nucleation mechanism is assumed which means that the hydride nuclei are generated on solid surfaces in the cold trap (mainly on the large surface of the cold trap inserts). Then, these crystals grow and precipitation rates are governed by growth kinetics.

The question is if the nucleation or the growth process is dominating for the fusion blanket cold trap. Because of operation at very low concentrations, cold trap volumes can become quite large assuming mass transfer coefficients used for the cold trap design for conventional applications. These cold traps are designed such to ensure a large loading capacity (long-term operation without plugging). Premature plugging is no critical problem of the present cold trap due to frequent cold trap regeneration. If it proves that mass transfer is diffusion controlled, then, the mass transfer coefficients can be increased considerably by increasing the NaK velocity through the cold trap, which - in total - results in a decreased cold trap volume.

To determine the mass transfer characteristics two kinds of experiments are performed

a) the cold trap outlet concentration c_0 is measured as a function of time during hydrogen loading at constant operating parameters (mass flow rate, cold trap inlet concentration, inlet and outlet temperatures).

b) after termination of an experiment whith a large hydrogen loading period the axial distribution of the deposited hydrogen in the cold trap is determined.

First experiments have been performed in the WAWIK-facility. Protium is used to simulate tritium. The test facility is shown schematically in Fig. 4.3-1. Figure 4.3-2 shows the experimental cold trap which consists of mainly a vertical pipe filled with up to 9 wire mesh packings which is cooled counter currently by air.

First results [33] on cold trap efficiencies as a function of loading time are shown in Fig. 4.3-3. For comparison, the result from a sodium cold trap [34] is also shown. Both experiments were performed at similar hydrogen concentrations; the specific wire mesh surface of the Na cold trap, however, was about one third of that used in the present experiment.

At the beginning of hydrogen removal only mass transfer due to nucleation can occur and the efficiency is lowest. The mass transfer by crystal growth starts and becomes finally the dominating process. For the Na cold trap the nucleation process seems to dominate during the first 20 hours whereas the crystal growth process is dominant after this period of time. For the NaK cold trap the efficiencies are considerably higher, even shortly after starting the hydrogen loading. This indicates that nucleation is much faster in NaK than in Na. The final efficiency is considerably higher for the present NaK cold trap. For the same specific wire mesh surface as used in the Na cold trap a final efficiency of 85% is estimated. Corresponding experiments will be performed in the near future.

First measurements of the axial hydrogen distribution in the cold trap are shown in Fig. 4.3-4 together with the results of a two-dimensional (2d) calculation. In this calculation only mass transfer due to crystal growth was considered.

For comparison with the 1d model, 2d calculations of the temperature, velocity, concentration and precipitation distributions in the cold trap were performed [35]. The numerical results show that natural convection has an significant effect in the wire mesh packings. Further experimental and theoretical work is required in order to develop a reliable mass transfer model.

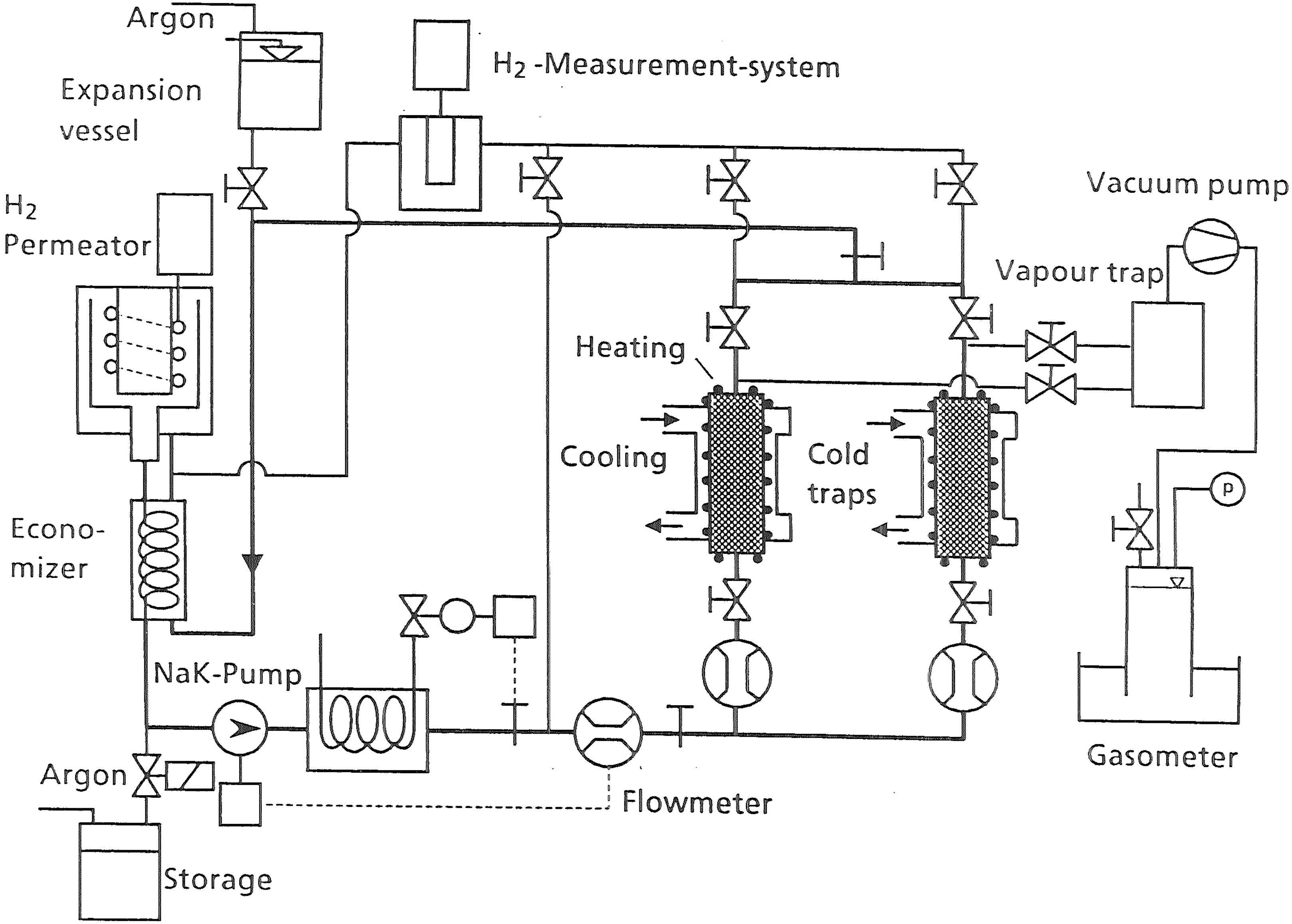


Fig. 4.3-1 WAWIK-test facility for hydrogen removal and recovery

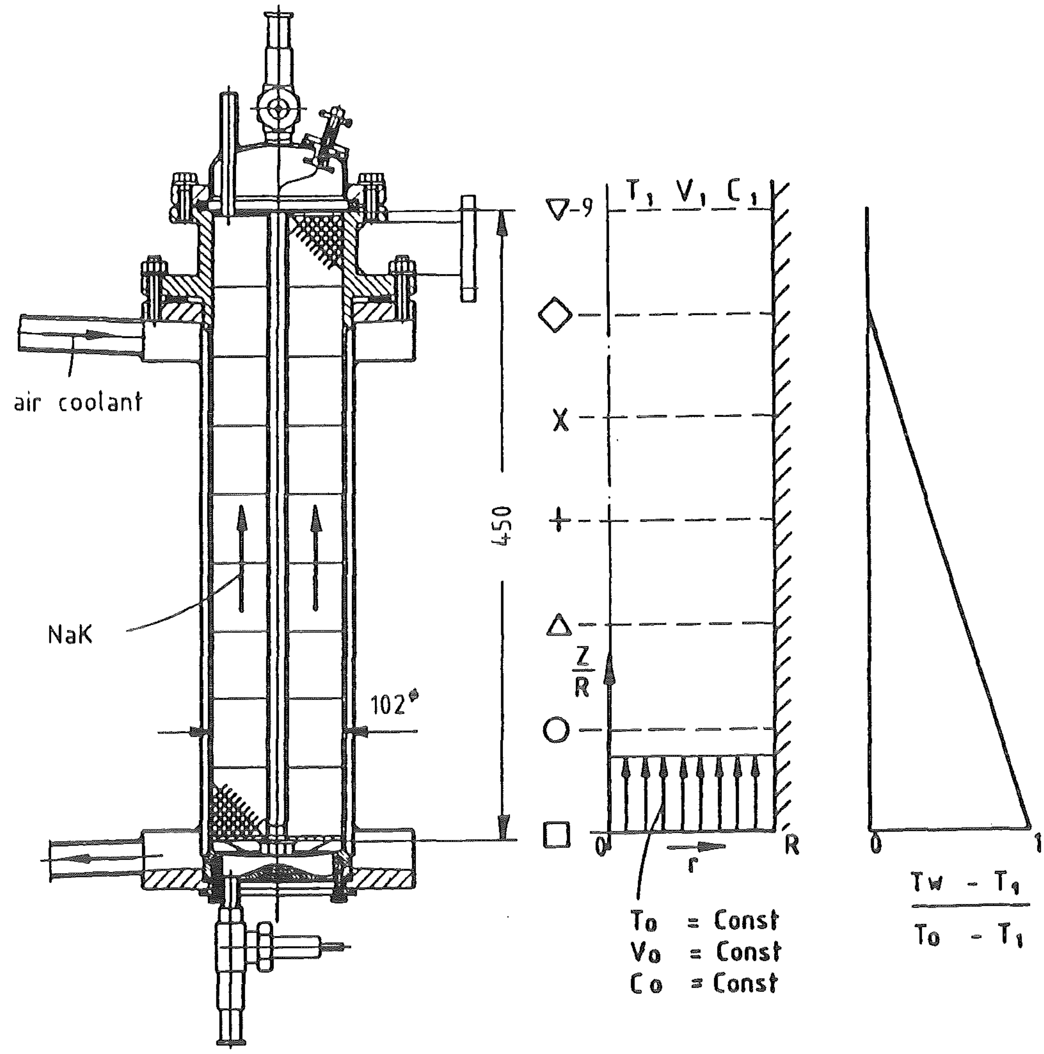


Fig. 4.3-2 Experimental cold trap

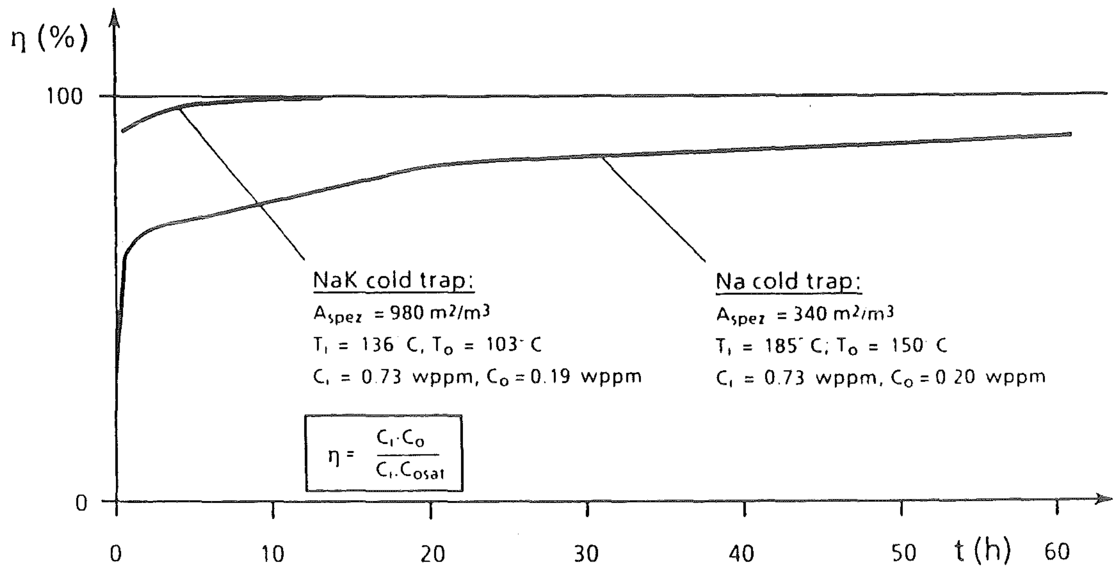


Fig. 4.3-3 Cold trap efficiency as a function of hydrogen loading time

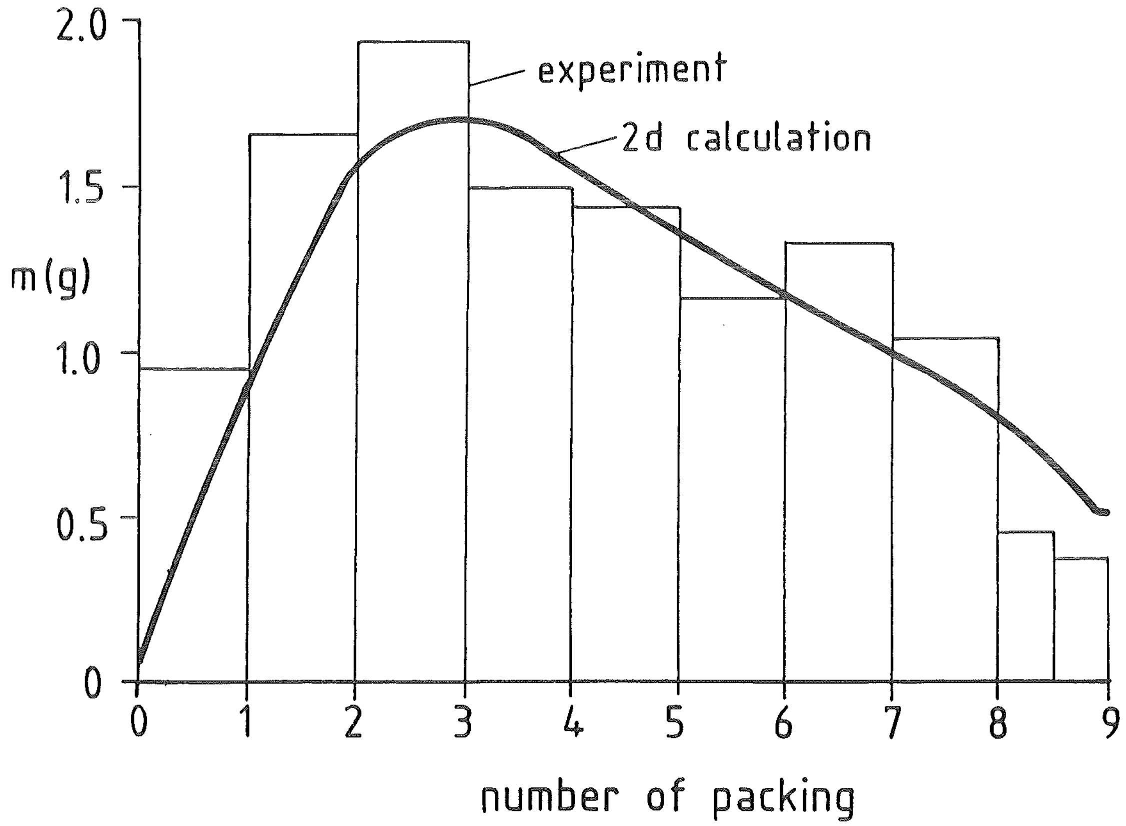


Fig. 4.3-4 Axial hydrogen distribution in a cold trap

Tritium recovery from cold traps

In the experiments again protium was used to simulate tritium. First experiments to investigate hydrogen release by thermal decomposition were performed with commercial fine NaH-powder. In a next step, NaH-crystals were generated in a Na filled vessel where hydrogen was absorbed at the top of the Na interface and hydride crystals precipitated at the cooled bottom of the vessel. This experimental set-up was then also used with NaK instead of Na. The release coefficient for KH crystals is significantly larger than for NaH-crystals as long as release by hydrogen bubble formation prevails. However, hydrogen release without hydrogen bubble formation in the NaK pool occurs at low release rates. Then, the release coefficients become considerably lower because diffusion in the liquid pool and desorption mechanisms at the free surface become rate determining. The practical consequence from these experiments is to drain the cold trap from NaK before heating up the cold trap.

Figure 4.3-5 contains new results where either the hydrogen from the single cold trap packings was determined or the hydrogen content from a complete cold trap was released [36]. In the former case the hydrogen loading was quite large whereas in the latter case the cold trap were only loaded for some hours which is characteristic for the anticipated application. The release coefficients are slightly higher at lower temperatures than for the crystallizer experiments. No significant differences is observed for high and low loading.

For a release temperature of 400°C, 99% of the tritium would be released in about three hours which is a very favourable result. Two regeneration cycles per day could be feasible; the tritium inventory in the two cold traps (batch operation) would be about 0.5 tritium productions per day.

Requirements for processing tritium recovered from the self-cooled Pb-17 Li blanket: The Blanket Interface

The aim of the blanket tritium recovery system (BTRS), in our case the cold trap system, is to recover the tritium bred in the blanket. However, not only gaseous tritium may leave the BTRS but, depending on the blanket concept, also protium, tritium carrier gas and various impurities. An additional tritium processing system (TPS) may be required to obtain pure T_2 gas. The boundary between these two processing systems was called Blanket Interface.

The specific features of the BTRS for the self-cooled Pb-17Li blanket are:

- No carrier gas is required for tritium recovery (small volume flow rates have to be processed).
- Tritium is recovered in gaseous form (HT) (no processing of HTO required).
- The blanket tritium recovery system (BTRS) is disconnected from the neutron field (no loading of the BTRS with radionuclides besides tritium).
- The BTRS is disconnected from the Pb-17Li loop (much lower corrosion in Na or NaK which results in very low impurity levels).

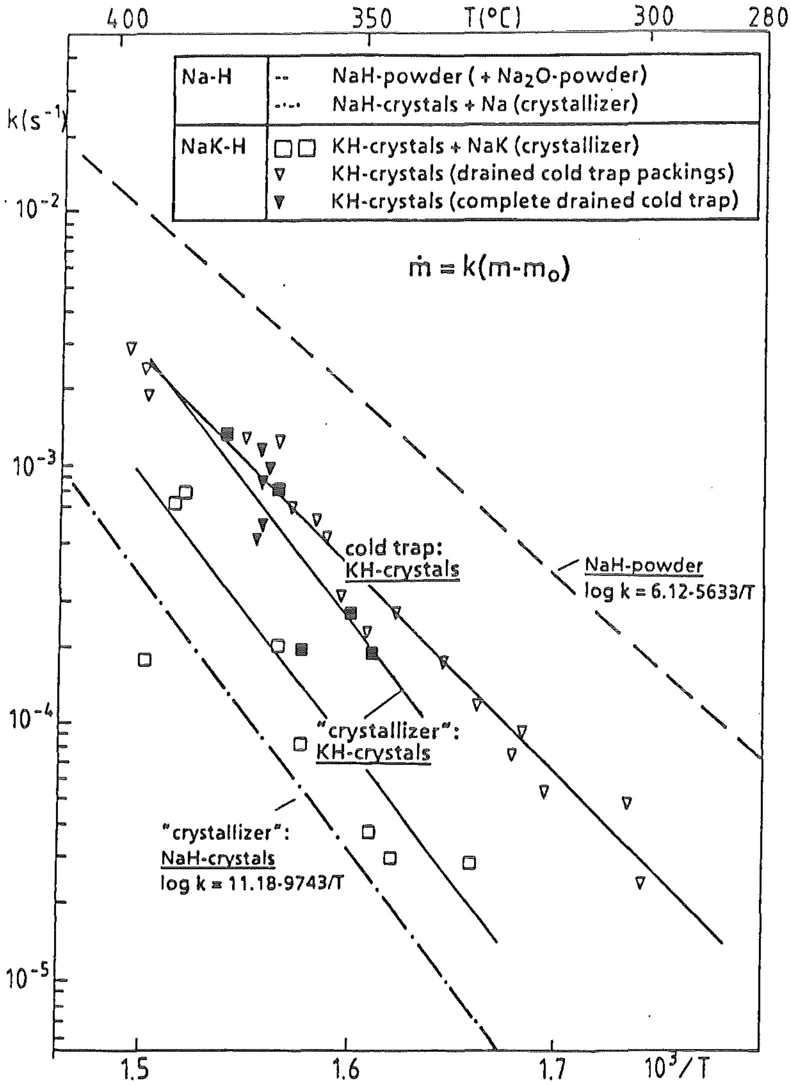


Fig. 4.3-5 Hydrogen release coefficients for NaK cold trap

An estimation of impurity source terms in the intermediate loop showed that besides protium and tritium no significant amounts of other species leave the BTRS. This flow can then be processed in the system required for fuel clean up without any significant modifications.

4.4 Pb-17Li Compatibility and Purification

The compatibility of the structural materials with the liquid metal breeder is a major issue for self-cooled blankets because it determines the maximum allowable temperatures. Corrosion in the blanket segment leads to the transport of radioactive corrosion products and, as a consequence, to the activation of the loop system. Additionally, deposition of corrosion products in cooler parts of the primary loop can also cause plugging.

Compatibility

The corrosion of the martensitic steel X18 CrMoVNb 12 1 (MANET) in flowing liquid Pb-17Li was studied at 823 and 773 K. The test section and the other high-temperature components of the PICOLO loop [37] were fabricated of ferritic steel. The low-temperature components were of austenitic steel, since an electromagnetic pump and flow meter and a magnetic trap for particles were placed in this part. The corrosion effects were determined by means of weighing the specimens and measuring their diameters before and after exposure in PICOLO. Additionally, metallographic studies were performed in order to evaluate the corrosion effects on the bulk of the material.

The measurements of changes of the weight and diameters of the specimens resulted in fairly well agreeing results. All specimens indicated that the steady state corrosion followed a period of initial corrosion. The rates of diameter changes were much slower in this initial phase, they did not exceed one tenth of the steady state reaction rates [38, 39]. The losses of diameter are shown in Fig.4.4-1 as a function of time for the two testing temperatures.

The steady state corrosion rates are calculated as $r_1 = 0.13$ [mm/a] at 773 K and $r_2 = 0.91$ [mm/a] at 823 K. The temperature dependence of the rate constants can be expressed by an Arrhenius law. The rate constants r_n of the linear corrosion equation

$$\Delta R = r_n \cdot t$$

are related to the hydraulic parameters of the test section of the loop. It can be calculated on the basis of the relation $r_n = \beta(x_s - x_0)$, in which β is the mass transfer coefficient and x_s the saturation concentration of the material in Pb-17Li, x_0 its concentration in the bulk flow of liquid metal. x_0 is small compared to x_s and can be neglected [39].

The mass transfer coefficient is related to hydraulic parameters according to the equation $\beta = (Sh \cdot D)/d$ with Sh as Sherwood Number, D as diffusion coefficient (of iron in Pb-17Li) and d as hydraulic diameter of the test section. For developed turbulent flow, Sh can be calculated from the simplified equation

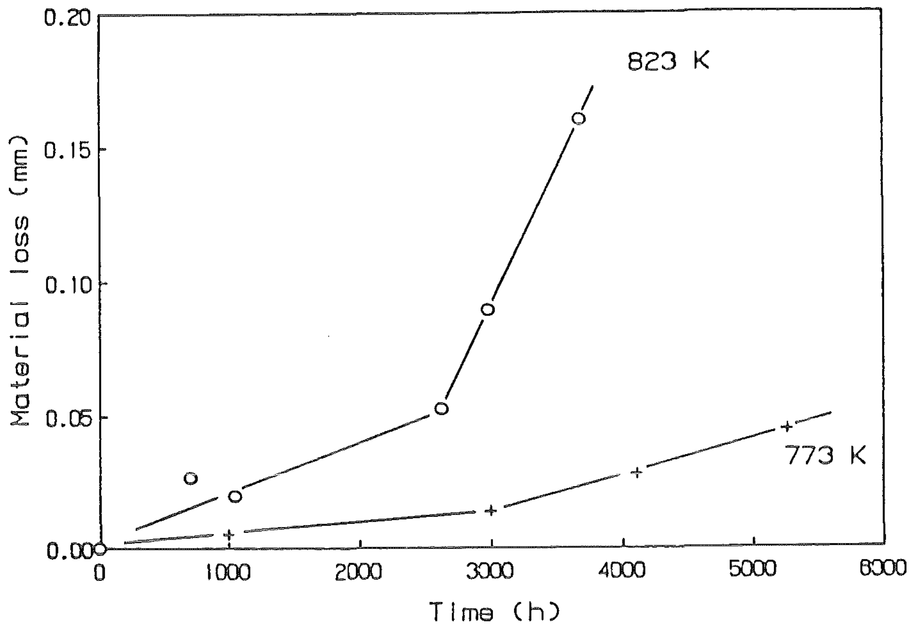


Fig. 4.4-1 Corrosion rates as a function of time

$Sh = 0.037 \cdot Re^{0.75} \cdot Sc^{0.42}$. 1.3. Re is the Reynolds Number of the test section for the given flow velocity u, $Re = (u \cdot d)/\upsilon$ (υ is the kinematic viscosity of the liquid metal). The value of Sc is received from $Sc = \upsilon/D$. The rate constants r_n are calculated based on the data taken from tables for the liquid metal and calculated for the dimensions of the loop and the parameters of tests. We receive $r_1 = 0.37$ [mm/a] at 773 k and $r_2 = 1.47$ [mm/a] at 823, in good agreement with the measured values.

The agreement of measured corrosion rates with values calculated on the basis of the hydraulic model indicates that the corrosion mechanism is a dissolution of the solid metal in the liquid metal, and a precipitation of the dissolved material in the low temperature branch of the loop.

The chemical activity of oxygen in Pb-17Li is favourable for the chemical stability of several oxides which may be used as electrical insulating layers. While Al_2O_3 is not stable against liquid lithium and reacts thoroughly at temperatures above 350°C, it should be compatible with Pb-17Li at moderately elevated temperatures. The verification of this assumption by means of experimental studies is still open.

The influence of the eutectic alloy Pb-17Li on the mechanical properties of the MANET steel was studied in stagnant medium. The creep-rupture behaviour at 500°C did not show a significant influence of the environment.

There was no apparent influence of the liquid alloy in tests of short duration, as was seen in some low-cycle fatigue and tensile tests. The tensile tests clearly demonstrated that there was no tendency for a liquid metal embrittlement in the system MANET steel - Pb-17Li at temperatures in the range 250 to 300°C [40].

Purification

The development of purification methods for Pb-17Li has just started. Three kinds of impurities are of special concern.

Impurities undergoing mass transfer will be mainly studied in the facility TRITEX. First results were obtained also from thermal convection loops. It was found that particles from different parts of the loop had different compositions. Most amazing was an intermetallic phase between Mn and Ni, inspite of low Mn concentration in the Pb-17Li. Particles of this were found in the coldest area of the cold spot. The results were not published so far, but reported during an international workshop at Nottingham in 1990.

Radioactive impurities occur often in very low concentrations, some even as tracers. Po-210 is always in tracer concentrations. In sodium cooled reactors such impurities show a different behavior than expected from larger concentrations [41, 42]. The investigation of the behavior of Po-210 is under way [43]. It was found that Po-210 is enriched in oxide layers, but it is not clear so far if Po-210 can be separated completely from the Pb-17Li by oxidation. Currently, a study of the volatility of Po-210 from the molten eutectic has started.

The third kind of impurities are interstitial elements of the steels. Will there be a transport of carbon and other elements in Pb-17Li as known from sodium systems, and will such elements have an impact on blanket operation?

So far, in all observations never any influence was seen of oxide layers on results, neither on deuterium behavior nor in compatibility tests. Like in sodium systems methan was seen in the TRITEX covergas at higher temperatures. Probably some residue lubricant had reacted with Pb-17Li. Oxides from thermal convection loops contained some elemental carbon.

4.5 Ancillary Loop System, Components

Steam generator design

The first concept of a PbLi-heated steam generator for DEMO was based on a straight double-wall-tube design with NaK flowing slowly inside the gap between the two concentric tubes. In order to provide a more profound basis for that design, alternative steam generator designs were investigated. They can roughly be divided into the following categories according to their tube designs

- Bajonet tubes
- Tubes with a 90° elbow (J Tubes)
- Tubes with a 180° elbow (U-tubes)
- Helical tubes.

Among these variations only the helical-tube design seems to be a promising alternative solution to the straight-tube design. In particular, it offers a smaller construction height and the possibility to identify faulty outer tubes, which is a problem for the straight-tube steam generator.

However, mainly due to the very cost-intensive fabrication of the helical-tube design, the straight-tube double-wall steam generator is considered as the reference version and the helical-tube design as a realistic alternative.

Steam generator accident study (NaK-H_2O-reactions)

If NaK comes into contact with water/steam it will react under formation of hydroxides, oxides and gaseous hydrogen at a high temperature level. Consequences of such a NaK-H_2O-reaction mainly depend on the leak rate and on the geometrical conditions in the leak area.

In case of small leaks (H_2O-leak rate some g/s) and a sufficient free volume of NaK a so-called reaction flame will be formed. Due to the high temperatures and the high velocities of hydroxide-particles, neighboured tubes may be damaged in relatively short times by an effect called wastage; the accident is escalating. To minimise the escalation of a wastage, several design precautions can be taken, such as: protection sleeves around the inner tubes in the NaK collector area, reliable detection systems, adequate design of the liquid metal system.

The proposed steam generator design excludes the possibility of a simultaneously multi-tube rupture during a NaK-H_2O reaction due to the double-wall concept and additional wastage protection tubes in the NaK collector area. For the case of a sudden guillotine fracture of one tube pressure load calculations have been performed with the ROLAST code.

A simplified flow scheme and the pressure development at different points of the NaK system is shown in Fig. 4.5-1. The rapid pressure increase in the reaction zone (outlet of the gaps at the wastage protection plate) is mainly due to the small volume of the NaK collector area. This high pressure transient provokes pressure pulses with maximum values at about 100 bar (heat exchanger and cold trap). Because the NaK system is designed for this pressure and no depressurization of the NaK system via bursting discs has taken into account, the overall pressure reaches the water side pressure of 70 bar after about 3 s and the water leak flow is stopped.

Pipe concept

The main pipes for the DEMO blanket cooling system have a diameter of 0.4 m. Due to the large specific weight of PbLi, special attention has to be paid on the pipe support concepts. Taking into account static and dynamic loads, a maximum distance of 6 m between supports of horizontal piping was calculated. The calculations also showed that most of the pipe clamps for static load dissipation (hangers) may be combined with dynamic load dissipation (shock absorbers or pendulum supports). Fixed points, hangers or plain bearings were investigated and found to be feasible as supports.

4.6 Safety and Reliability

With respect to safety the potential hazards resulting from the use of liquid metals seem to be of most concern. These materials imply the possibility of chemical reactions if leaks occur. Application of the general safety objectives:

- Minimizing of facility damage, this means avoiding of failure propagation, and
- Minimizing of radioactivity mobilization

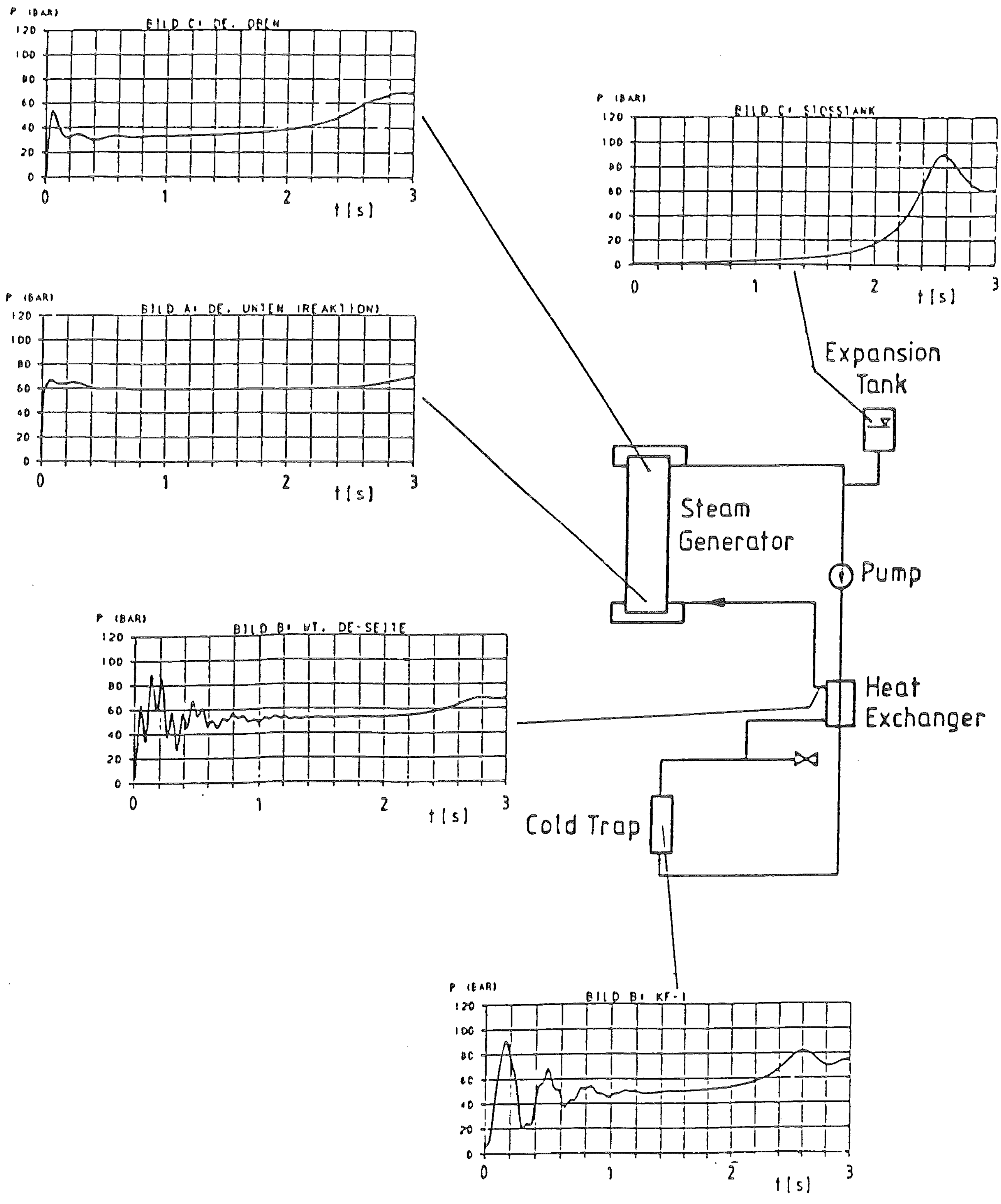


Fig. 4.5-1 Pressure transients at different loop locations

leads to the demand of minimizing possible reactions by reduction of reactive inventories and separating the reaction partners. To avoid release of aerosols into the environment confinement buildings and filtering systems are recommended.

Three different chemical reactions have to be considered for both Pb-17Li and NaK: Liquid metal/gas, liquid metal/water, and liquid metal/concrete reactions. The reaction severity depends on different variables, e.g. the temperature and pressure of the reactancts, and the contact mode. Table 4.6-1 gives a qualitative picture about the behaviour of the different reactants.

The previous investigations show that there is a close connection between design and reliability. Reliability assessment becomes, and is to see, as a permanent and accompanying part of the design from the early conceptional phase until the operating of the complete equipment.

The reliability assessment is of increasing importance in the early conception and design phase. The recognition that by early and adequate conceptional and design precautions the risk of bad developments can be reduced, is based on the experience with the first results. The problem of missing or insufficient operating experience and, therefore, the lack of reliable data must be passed by an appropriate safe design, as demonstrated on the example of the blanket cooling system. This experience suggests, especially for the relatively complicated blanket components, a partition or reduction to the elementary components as there are welds, screws, tubes, plates etc., and the consideration of this parts separately in view of reliability. Then, the individual reliability results will be combined to the complete blanket element, a procedure as practised in electrical or electronical equipment, where operating experience for the whole component is not available. The quality of the results is very much dependent on the degree of detailability of the lay-out.

Table 4.6-1: Chemical reactions

	Pb-17Li	NaK
Gas (air, N_2, CO_2)	No significant temperature increase in air or nitrogen. Rapid temperature increase in CO_2. Here mobilization of radioactive aerosols and hydrogen seems possible. More recent investigations [44] indicate that the peak reaction rate (g-Li/min · cm^2) of Pb-17Li with air is higher than that of Pb-17Li with steam. This means that Pb-17Li may react more violently and liberate more heat in an air environment than in a steam environment.	No reaction with nitrogen and the noble gases. Reaction with carbon dioxide producing highly caustic oxide and poisonous carbon monoxide.
Water (steam, liquid)	Reaction with steam significant, producing some aerosols. Hence, mobilization of activation products possible. Reaction with liquid appears to result in lower temperatures, however, in higher pressurization than in the case of steam. Injection mode seems to be of most concern. The potential for Pb-17Li/Water reactions to mobilize large amounts of radioactivity within the containment is largely unknown. Since the self-cooled blanket does not contain water this question is of concern within the vacuum vessel only for a simultaneous leak of a blanket seqment and a leak in another in-vessel component being water-cooled. However, failure in the steam generators with possible liquid metal/water reactions need special attention. In principle, efforts to remove routinely activation products, especially the more volatile isotopes mercury, polonium incl. bismuth, and thallium, should be enhanced (see chapter 4.5 and 5.4).	Violent exothermic reaction with water. If an oxigen atmosphere is available a secondary explosive reaction is possible between the liberated hydrogen and the oxygen.
Concrete	Mild reaction, with hydrogen production and mobilization of radioactive aerosols. Concrete protection by a steel liner is standard practice.	NaK spills onto concrete may react with water or chemicals in the concrete. To prevent reactions and protect the surface steel liners are normally used.

4.7 Electromagnetic Forces

The existing code CARIDDI which computes the eddy currents and electromagnetic forces in toroidal structures has been improved. By introduction of the extended memory option the number of degrees of freedom which can be considered could be increased significantly. Thus modelling of complex, asymmetric structures with internal walls and electrically conducting fluids is possible. For such large problems also a reduction of the computing time could be achieved by optimization of the data processing. Right now, the improved capability of CARIDDI is being demonstrated by application of the code to a rather realistic blanket model. It is state-of-the-art that the forces calculated by CARIDDI are used as input to carry out structural dynamics analyses. The feedback of the structural deformations on the elctromagnetic behavior and thus on the resulting forces is neglected.

5. Required R&D Programme prior to Test in NET/ITER

This status report has been prepared at the end of the first three year R+D period of the European DEMO-relevant blanket program. A second three year period will now follow up to the year 1995 at which date a selection among the European liquid metal blanket alternatives will take place. The selected blanket will be designed in detail and large scale experiments are envisaged. Consequently the coming R+D program described here is limited to key problems and small scale experiments. Sections 5.1 to 5.7 summarize the R+D tasks foreseen. The blanket design will be modified if required by the results.

5.1 Magnetohydrodynamics

Experimental and theoretical MHD work is required to support the selfcooled Pb-17Li blanket concept for a Fusion DEMO plant. Key problems of the present design with the poloidal-toroidal flow concept which will be investigated first are the pressure drop and the flow distribution in poloidal-toroidal single and multichannel bends:

- the single channel elbow experiment
  (jointly performed with ANL)
- the multichannel U-bend experiment.

If the above experiments together with the analysis do not allow to extrapolate to the flow structure and the pressure drop of the meander shaped front channels a meander channel experiment is planned to be conducted jointly with the Latvian Inst. of Physics in Riga.

5.2 Electrical Insulation in the Flow Channels

Prior to application of flow channel inserts in a blanket, the following properties and the operational performance must be investigated:

- mechanical bonding, thermoshock-resistance, electrical insulation, corrosion behaviour (out-of-pile)
- mechanical bonding after irradiation up to 10 (70) dpa, electrical insulation under combined load of irradiation and electrical potential (in-pile tests)

- behaviour under MHD-conditions, filling and draining of fluid in channels with FCI, stability under combined load of temperature, pressure, neutron irradiation.

5.3 Tritium Extraction and Recovery

For the development of the reference concept based on tritium permeation through heat exchanger tube walls into NaK and cold trapping of this liquid metal, the following R&D tasks have to be continued until 1993:

- Theoretical and experimental investigation of hydride precipitation kinetics in cold traps
- Regeneration of cold traps by thermal decomposition
- Conceptual design of a fusion relevant cold trap system
- Integral permeation experiment in the system Pb-17Li/ferritic steel wall/NaK (decision end 1992)

5.4 Compatibility, Liquid Metal Purification

Compatiblity

Two major testing series have to be performed in the corrosion test facility PICOLO in the next future. One of them is concerned with the MANET steel. Some additional corrosion tests of this material at different parameters (testing temperature, flow velocity, Reynolds number, etc.) should generate data on the basis of a broad field of parameters in order to verify the relation between the corrosion model and the measured corrosion rates. This program will also be used to get more information on the deposition behavior of corrosion products in the cold branch of PICOLO.

The second one is related to the development of an insulating layer on the structural material which should be stable in contact with liquid Pb-17Li alloy. Alumina layers formed on 316 L(N) steel surfaces after alitization of the material will be the first insulators to be tested. In a following step the alitization of MANET steel will be developed and the stability of alumina layers formed on the iron aluminide layers on MANET will be tested. The program includes the development of a method to measure the resistivity of such layers on specimens which are inserted in Pb-17Li molten alloy.

Purification

The development of purification methods for Pb-17Li has just started. Three kinds of impurities are of special concern.

Impurities undergoing mass transfer will be mainly studied in the pumped loop facility TRITEX. Here different types of cold and magnetic traps will be used. There is also an access in TRITEX to the liquid metal/cover gas interface. Here, the possibility to remove some impurities (Bi) with oxides will be investigated. Radioactive impurities are often in very low concentrations, some even as tracers (Po-210 is always in tracer concentrations). Corresponding experiments will be performed in TRITEX loop and thermal convection loops.

5.5 Ancillary Loop Systems, Components

Different systems are proposed for blanket tests in NET/ITER and for the operation of a DEMO reactor respectively.

The system anticipated for NET/ITER is based on designs and technologies used already in liquid metal cooled fission reactors and requires therefore no large development programme. This is especially true for the issue of liquid metal/water reactions in the heat exchanger which has been investigated extensively. NaK/water reaction, however, has to be investigated for the case of the double walled steam generator in DEMO. Theoretical evaluations indicated that a failure propagation in this heat exchanger is excluded by design measures. For confirmation, however, some experimental investigation of the NaK/water reaction in the concentric gap of the double walled tubes are required.

5.6 Safety and Reliability

R&D work has to be done in different areas: (1) The investigation of the possible interaction of Pb-17Li with different reactants, (2) the chemistry of polonium, and (3) the analysis of accident sequences in order to assess their consequences.

The work for the first area can be started immediately and will be performed with a high priority and with a very wide scope, because it must be ensured that the operation of a loop with such coolants can be licensed. Beside the chemical reactions of the different species the release of e.g. polonium from liquid Pb-17Li into different atmospheres must be investigated. A phase diagram for such three component systems is not available but will be needed.

More detailed analysis of accident sequences requires a fixed design of the test module and its environment in the NET/ITER machine. Areas of special interest are the investigation of a loss of coolant accident inside and outside of the vacuum vessel with possible thermal and radiological consequences, and a loss of site power.

Operational safety and reliability need special attention. Hence prior to tests in NET/ITER it is recommended to test the module and its loop system out of pile under real thermal conditions and continuous operation. The final design should be based on a reliability assessment taking credit from experimental data from liquid metal fission reactor technology and fusion component failure rate data.

5.7 Electromagnetic Forces

Electrical currents and electromagnetic forces in a blanket segment during plasma disruptions can be calculated with the CARIDDI code.

New models, however, have to be developed to account for the fluid structure interaction in the determination of mechanical stresses in the blanket structure during plasma disruptions.

6. Test Programme in NET/ITER

Test strategy

Tests in NET/ITER are the last step in the development programme of breeding blankets prior to the use in a DEMO-power reactor. Keeping this goal in mind it is desirable to perform an extensive blanket test programme in NET/ITER covering both multiple test modules and long term performance tests of some modules or better segments. However, testing space as well as total fluence obtainable in NET/ITER are very limited.

The following approach is proposed [24]:

- Make extensive use of all testing possibilities outside of NET/ITER.
- Perform as many tests as possible during the physics phase.
- Divide the test port into parts in order to conduct parallel tests during the first years of the technology phase.
- Perform sequential tests with full size modules during the second half of the technology phase.
- Test segments of one or two blanket designs towards the end of the technology phase.

Technical issues and objectives of tests

The technical issues of self-cooled liquid metal blanket concepts are:

- Tritium self-sufficiency
- MHD effects
- Heat transfer
- Materials interactions (e.g. corrosion)
- Structural response in a fusion environment
- Tritium recovery and control
- Components and system interactions.

For liquid metal cooled designs, the issue which tends to dominate most considerations is MHD. Potentially, MHD effects could influence fluid flow, heat transfer, corrosion rates, and stresses (pressure) in the blankets. Therefore, this issue has an important role in the test program. Other issues, although important, do not play as crucial a role in the NET/ITER programme. Individual issues which can be studied without a fusion environment, could be investigated separately, outside of NET/ITER. On the other hand, an in-depth investigation of the structural response would require a neutron fluence much higher than the one achievable in NET/ITER. These considerations indicate that the objectives of the test program have to be chosen carefully in order to eliminate tests which can either be performed earlier, better, and cheaper in a non-fusion environment or would produce results of questionable value due to the low fluence in NET/ITER.

Keeping these limitation in mind, the tests cover the following range of objectives

- Tests of the predictive capabilities of engineering codes
- Tests of the engineering performance of particular concepts
- Tests of the engineering reliability of the concepts.

The specific tests are related to the key issues identified earlier. The tests to be conducted in NET/ITER are MHD tests, combined MHD/thermalhydraulic tests, short term and extended term performance tests, and post-test examinations.

Impact of reduced NET/ITER operating time on the blanket test programme

There is a strong interaction between the design of NET/ITER and the maximum operating time of the machine. The important issue is the tritium supply. A limited amount of tritium can be obtained from external sources allowing a total burn time of approximately 2000 to 3000 hours. Longer operating times require tritium breeding blankets in NET/ITER. So called "driver blankets" not only add to the complexity and costs of the machine but also can reduce the reliability of the operation because integrated testing of those blankets require the operation of NET/ITER. To avoid these problems, it is highly desireable to limit the total burn time required for blanket tests to a value below 3000 hours. Estimates have indicated, that such a limited time would allow for a minimum test programme only, excluding all extended performance tests. Only with the help of extensive material irradiations programmes and highly sophistacated computer codes an extrapolation to DEMO can be made.

Mandatory for a meaningful blanket test programme is a relatively high availability of the machine because the average time required for a useful single test is in the order of a few days. A mean time between two unscheduled outages lower than this value would require too many attempts for a successful single test, "wasting" too much tritium. The conclusions of these considerations are:

- Blanket tests require a relatively high reliability of NET/ITER.
- Without driver blankets no extended performance tests are possible.
- Reduced operating time of NET/ITER places a high burden on computer codes necessary for the extrapolation to DEMO-conditions.

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