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Summary

The report describes design studies on a water cooled in-vessel shield blanket for NET and its modification into an aqueous lithium salt blanket. The shield blankets are exchangable against breeding blankets and fulfill their shielding and heat removal functions. Emphasis is on simplicity and reliability.

The water cooled shield is a large steel container in the shape of the blanket segment which is filled by water and containes a grid structure of poloidally arranged steel plates. The water flows several times in poloidal direction through the channels formed by the steel plates and is thereby heated up from 40°C to 70°C.

When water is replaced by an aqueous lithium salt solution the shield can be converted into a tritium breeding blanket without any design modification or in-vessel component replacement. When compared with other concepts this blanket has the advantage that the solution can replace water cooling also in the divertor and in segments dedicated to plasma heating and diagnostics, what increases the coverage considerably.

Extensive three-dimensional neutronics calculations were done which, together with literature studies on candidate materials, corrosion, and tritium recovery led to a first assessment of the concept. There is an indication that no major corrosion problems are to be expected in the low temperature region envisaged. Tritium recovery capital costs were estimated to be in the 20 MECU to 50 MECU range and tritium breeding ratio is comparable to the best breeding blanket concepts.

Tritium production can be increased when beryllium plates are placed in the front part of the containers. The plates should be cooled by the lithium salt solution flowing in poloidal slits. In this way an effective tritium breeding ratio is obtained which by far exceeds that of all other concepts.

Potential und Probleme eines Blankets mit wässriger Lithium-Salz-Lösung für NET

Zusammenfassung

Es wurde eine Entwurfsstudie für ein wassergekühltes NET-Abschirmblanket durchgeführt, das durch Zugabe von Lithium-Salz in ein Tritium-Brutblanket verwandelt werden kann. Die Abschirmblanket-Segmente sollen an Positionen von Brutblankets eingesetzt werden können, um deren Funktion in Bezug auf Abschirmung und Wärmeabfuhr zu übernehmen. Einfachheit und Zuverlässigkeit waren die Hauptkriterien des Entwurfes. Die Segmente bestehen aus einem großen gebogenen Wasserkanister, der genau in das Gehäuse mit Erster Wand paßt, ein Gitter von poloidal angeordneten Versteifungsplatten bildet Strömungskanäle für das Kühlwasser. Das Wasser strömt mehrmals in poloidaler Richtung durch solche Kanäle und wird dabei von 40° auf 70°C aufgeheizt.

Ersetzt man das Wasser durch eine wässrige Lithiumsalzlösung, so kann man ohne jede Umbauarbeit das Abschirmblanket in ein Brutblanket verwandeln. Ein besonderer Vorteil dieses Blanket besteht darin, daß auch in den für Plasmaheizung und Diagnostik vorgesehenen Segmenten das Kühlwasser durch die Lithiumsalzlösung ersetzt und zum Brüten verwendet werden kann. Umfangreiche Neutronikrechnungen in dreidimensionaler Geometrie zusammen mit einer Literaturstudie zur Korrosion und Tritiumabtrennung führten zu einer ersten Beurteilung des Konzeptes.

Bei der niedrigen Temperatur werden keine ernsthaften Korrosionsprobleme erwartet. Die Kapitalkosten für die Tritiumextraktionsanlage wurden auf 20 MECU bis 50 MECU abgeschätzt. Die Tritiumbrutrate in NET ist mit der der besten Blanketkonzepte vergleichbar.

Die Tritiumbrutrate kann noch erhöht werden, wenn man Berylliumplatten an der Vorderseite der Wasserkanister anbringt. Die Platten werden durch die Lithiumsalzlösung gekühlt. Auf diese Weise wird eine Tritiumbrutrate erreicht, die die Werte aller anderen Konzepte deutlich übertrifft.

Content

1. Introduction

In spite of an extensive pre-NET testing program all of the blanket concepts, deemed to exhibit DEMO-prototypic features, will need a detailed in-NET testing prior to their insertion in a large number of segments. Thus, it is now the generally accepted blanket insertion strategy of NET to start the NET operation with just a few test blankets (one for each basic concept) and to fill the other positions with dummy segments. These dummy elements have to replace the blanket in its shielding and heat removal functions but without tritium breeding. It is important that they are simple enough to be operated without special in-NET testing.

In this situation (D,T)-operation is possible only by external tritium supply and the maximum plant availability may be limited by the availability of tritium. The presently assumed figure is an external tritium supply of 1.5 kg/a /1/. With 0.153 g tritium consumed per MWd of (D,T)-Fusion and 600 MW average fusion power this would allow 16.3 days of full power operation per year or an availability of 4.5%. Here tritium losses, tritium radioacitve decay, build-up of process inventory etc. are neglected.

With the stockpiling of tritium prior to a significant (D,T)-operation the availability could be increased for a certain period of time. Because of the radioactive decay of tritium this is an expensive procedure of limited potential. To buy 1.5 kg tritium one year earlier than needed leads to a loss of about 16 MECU because of radioactive decay (basis: 2 ECU/Curie Tritium). Stockpiling is absolutely not suited for a strategy where the tritium demand is increasing with time, as would be expected in the course of blanket testing.

16 full power days per year may be more than enough for an early testing period, but it is too limited for extensive testing. Thus, one should look for a "driver blanket" which could contribute to the supply of tritium to adequately perform the testing program.

In case the dummy elements are water cooled a possibility for extra tritium supply consists in replacing the water by an aqueous lithium salt solution with tritium extraction from the solution. This leads to a so-called self-cooled blanket /2/. Tritium production can be increased when beryllium plates are placed in the front part of the elements. These plates are cooled by the lithium salt solution, flowing in poloidal slits. In the following the potential of such concepts will be checked.

2. The in-Vessel Shielding Blanket Design

At KfK both, a helium cooled and a water cooled version of in-vessel shielding segments are studied. In the following only the water cooled version will be described. It will be denoted as "dummy segment". Design basis is the NET-DN configuration with 48 segments. It is assumed that a water cooled closed segment box exists. The box consists of a strong (30 mm) back plate, radially arranged side plates and a first wall with poloidally resp.toroidally running coolant channels and graphite protection. The cooling system of the box is independent of that of the blanket or dummy blanket. The box design is not subject of this study.

The dummy blanket basically consits of a closed steel container with the same outer dimensions as those of a normal blanket. Poloidally running stiffening plates inside guarantee a rigid structure which can withstand the electromagnetic forces. The plates form coolant channels for the water which fills the complete dummy segment and flows in poloidal direction.

Fig. 1 shows a torus midplane cross section of an outboard segment. Water flows downward in the front channels, turns at the segment bottom to the upward direction into the next row, goes down again and upwards in the last row of channels. In series cooling is chosen since parallel flow would lead to a very low flow velocity with possibly laminar flow and stagnant flow regions.

There is a main cooling system with two pumps and coolers in parallel which is connected with the dummy blanket by a 150 mm tube for supply and outlet. A separate emergency cooling system serves for the decay heat removal. Decay heat power is about 100 kW/segment. A tiny water tube brings the emergency coolant directly to the spray system on top of the front channels. The outlet coolant flow passes the emergency condenser/cooler because in case of a container rupture some steam has to be removed.

Table 1 summarizes the main data for an outboard dummy blanket module.

It is a low temperature, low pressure system with a very homogeneous temperature distribution. No large stresses or cyclic loads are to be expected. The conditions are similar to those of low power research reactors and similar systems for water purification, ion exchange, and heat removal are needed. Fabricability of the components is being checked now and may lead to modifications.

Table 1: Main data of a water cooled outboard dummy module.

power release	6 MW
coolant flow	48 l/s
coolant inlet temperature	40°C
coolant outlet temperature	70°C
system pressure	0.3 MPa
total pressure drop	0.01 MPa
coolant flow velocity in front channel	1.5 m/s
max. steel temperature	90°-105° C
flow velocity in supply tubes	3 m/s.

The whole container is rigidly mounted on the back plate of the box. Because of the small temperature differences and temperature variations thermal expansion effects will not cause problems.

The way the supply lines are arranged in the top part of the blanket is indicated in Fig. 2.

Shielding effectiveness was calculated in one dimensional cylindrical geometry: The torus axis coincides with the axis of the infinite cylinder: Inboard and outboard shield and blanket regions are represented by concentric annular zones, the radii of which are equal to the radii in the torus midplane of the real geometry. Source intensity is chosen to deliver an average 14 MeV neutron flux at the first wall of 1 MW/m^2. It was shown elsewhere that this representation is a very good approximation to the flux and power density conditions in the torus midplane /4/. Calculations were done with the KfK-version of the one dimensional transport code ONETRAN /5/ in S_8-approximation. The data library is based on VITAMIN-C and contains 25 neutron and 21 gamma energy groups in P_3-approximation.

The 65 cm thick permanent shield was described as a steel structure cooled by borated water. It is followed by a 10 cm steel wall of the superconducting magnet.

There are several radiation limits which should not be exceeded. It turned out, however, that the most restrictive one is the dose rate in the epoxy of the superconducting magnets, with the limit resulting primarily from the mechanical strength and not from the insulating properties /6/. Different limits are used by different authors.

Table 2 shows some neutron and gamma fluence and dose limits together with the values calculated for the shielding blanket described above.

Depending on the limits used, the margin for the NET shield varies between a factor of two and twenty. With the permanent shield being too much idealized no attempt was made to reduce the shield thickness and this margin.

The calculations referred to the outboard blanket. On the inboard side there is only room for a 25 cm thick in-vessel shielding module. This 10 cm reduction in thickness would lead to an increase of the dose at the superconducting magnet by a factor of 2.6. On the other hand the inboard wall load is lower by a factor of 1.73. Thus, the dose values will be higher than those given in Table 2 by a factor of 1.5. This is still well below the limits.

Alternative to the container design an arrangement of poloidally running tubes filled with flowing water also was considered. The tube arrangement for the outboard blanket is shown at Fig.3 as a cross section in the torus midplane.

Shielding effectiveness is less good than for the container design and an assessment of neutron streaming effects is difficult. Tubes with a stepwise change in diameter could be considered to improve the filling factor. But the container concept is preferred and was taken as the reference solution.

Tabele 2 Comparison of Dose Limits at Superconducting Magnets with Doses obtained at NET in 3 MWy / m² with the H_2O Cooled In-Vessel Shield Blanket.

Quantity	Unit	quoted limits				H_2O cooled NET shield
		Ref. 6	Ref. 1	Ref. 7	Ref. 8	
$\Phi_n \cdot t$	n / cm²		$5 \cdot 10^{17}$	($0.9 \cdot 10^{17}$)		$4.2 \cdot 10^{16}$
$\Phi_\gamma \cdot t$	γ / cm²		($4 \cdot 10^{17}$)	($1.2 \cdot 10^{17}$)		$2.5 \cdot 10^{16}$
$\Phi_n \cdot t$	Gy		$3 \cdot 10^{6}$	$9 \cdot 10^{5}$		$4.2 \cdot 10^{5}$
$\Phi_\gamma \cdot t$	Gy		$2 \cdot 10^{6}$	$6 \cdot 10^{5}$		$1.25 \cdot 10^{5}$
total	Gy	10^{6}	$5 \cdot 10^{6}$	$1.5 \cdot 10^{6}$	10^{7}	$5.45 \cdot 10^{5}$

in brackets: Quantities converted with [6] :

10^{6} Gy ≙ 10^{17} n/cm² } for Epoxy
10^{6} Gy ≙ $2 \cdot 10^{17}$ γ/cm² }

3. Modification of the Shield Module into a Tritium Producing Blanket

3.1 Selection of the lithium compound, material problems

When water in the shielding module and first wall box is replaced by an aqueous lithium salt solution tritium can be produced. As shown in Fig.4 (three-dimensional calculation) the tritium breeding ratio increases with the ^6Li concentration. Thus, compounds of high solubility are asked for. Other selection criteria are low corrosivity, low neutron absorption cross section, good chemical stability in the radiation field, and good separability from H_2O in the tritium extraction process. In case that electrolysis is chosen as the first step in this process, immediate use of the solution as electrolyte is also an important criterion.

Table 3 shows the solubility and lithium concentration at 20°C for some lithium compounds of adequate chemical stability. A similar but more comprehensive list is given in /7/.

Table 3: Saturation lithium content of some aqueous solutions at 20°C

compound	Molecular Weight	Weight% of compound in solution at 20°C	density of solution at 20°C g/cm³	gLi/cm³ sol.
LiCl	42,39	45,0	1,295	0,095
LiBr	86,85	61,5	1,713	0,084
LiJ	133,84	62,3	1,778	0,057
LiOH	23,95	11,3	1,135	0,037
$LiNO_3$	68,94	42,0	1,302	0,055
Li_2SO_4	109,94	25,65	1,234	0,034
$Li_2Cr_2O_7 \cdot 2H_2O$	265,90	65,17	1,595	0,054

($\rho = 2{,}34 g/cm^3$)

For comparison the lithium concentration in the Li_4SiO_4 pebble bed is $\rho_{Li} = 0.29$ g^{Li}/cm^3 and $\rho_{Li} = 0.064$ g^{Li}/cm^3 in Pb-17 Li .

The halogenes exhibit the highest solubility but are rather aggressive with a tendency to pitting corrosion. In addition the halogens have rather high neutron absorption cross sections.

Li_2SO_4, $LiNO_3$, and LiOH seem to behave better. LiOH has the advantage not to form acid anions. A disadvantage may be that some LiOT is formed which is separated from the water in the tritium extraction process thereby reducing the tritium recovery effectiveness (see chapter 4).

With respect to corrosion the situation is not very clear. An extensive literature survey has been done by Bogaerts, Embrechts, and Waeben together with some scoping measurements /7/.

$LiNO_3$ which has a higher solubility than Li_2SO_4 and LiOH seems also to offer advantages from the corrosivity point of view /7,9/. In /9/ it is stated that "stainless steel is perfectly suited for use with nitrate solutions at any temperature and concentration". In /10/ however, the statements are more reluctant with respect to nitrate and preference is given to Li_2SO_4 for which it is said that stainless steel is completely stable at all concentrations and temperatures of Li_2SO_4 solutions. In /7/ pitting corrosion is mentioned as a problem specific to Li_2SO_4.

Although there is an indication that no major corrosion problems are to be expected in the low temperature region envisaged here (40°C-70°C), more measurements at relevant conditions and engineering tests are urgently needed before a realistic assessment and material selection can be made. This should include the impact of welds and possible improvements by inhibitors..

Radiolysis is a problem which also exists for the water cooled shield blanket without lithium and will be similar to the situation in present light water reactors. The α-particle and triton from the $^6Li(n,\alpha)T$ reaction have ionization track lengths which are similar to those of recoils from neutron scattering on oxygen so that no new phenomena are to be expected. The rate of radiolysis will be high, however, and has to be investigated.

3.2 Neutronic Calculations

Neutronics calculations formed the central part of the study. They indicate the potential of a lithium salt blanket to improve the NET availability by increasing the tritium supply. Somewhat arbitrarily LiOH with 60% ^{6}Li-enrichment was taken with 127 g LiOH/l. This is close to the saturation density at 20°C. As can be seen from Fig. 4 we are in a rather flat part of the TBR vs. ^{6}Li-concentration curve so that the other compounds at the same or higher ^{6}Li concentration will give slightly higher values for the TBR.

The basic calculations for the reference case were performed with the Monte Carlo Code MCNP in three dimensional geometry /11/. A very close approximation to the real NET-DN configuration was used. Fig.5 shows a poloidal and Fig.6 the toroidal cut of the outboard side in the geometrical representation for the MCNP-calculations. The side walls of the boxes covered by copper plates with coolant tubes on their inner sides are modelled as well as the outer walls and the internal structure of the LiOH solution containers. The blanket container thickness is 35 cm. An inboard blanket was included with water cooled divertor sections on the top and the bottom parts. The inboard blanket thickness is 25 cm.

Full coverage with the reference segments is assumed what allows a high toroidal symmetry so that only half of one 7.5 degree segment has to be considered.

The overall geometrical dimensions of the box surrounding the container were taken from the ceramic breeder canister blanket concept /3/. The water cooled first wall structure used in the calculation is shown in Fig.7. It was represented by the mixtures given in Table 4.

Not only the water in the dummy elements but also in the first wall and the divertor was replaced by the LiOH solution.

The neutron source density distribution is given by /12/

$$s(a) = \left[1-(a/A)^2 \right]^4$$

Table 4 First wall representation in the 3d-calculations.

composition	vol. fraction (%)		thickness mm
graphite	C	100	15
coolant and connecting region	C SS H_2O void	47.3 28.7 18.7 5.3	27
steel plate	SS	100	10
total			52

SS = stainless steel 316L

where A = 135 cm is the plasma minor radius and a is a parameter with $0 \leq a \leq A$. This parameter is constant on closed surfaces the R and Z coordinates of which are given by /12/:

$$R = R_0 + a\cos(t + \delta \sin t) + e\,(1-(a/A)^2)$$

$$Z = E \cdot a \sin t$$

$$0 \leq t < 2\pi$$

R_0 = plasma major radius = 518 cm

E = elongation = 2.18

e = excentricity = 16.2 cm

δ = triangularity = 0.65

The values were transmitted by Daenner /13/.

In Fig. 8 the contours of a = const. curves are given together with the source intensities for the contours. The source is strongly concentrated in the central region.

A continuous neutron energy representation was used in the MCNP calculations with the nuclear data essentially based on the ENDFB IV file. Only for beryllium a more recent Los Alamos evaluation was used /14/.

The calculations were made for the reference configuration just described as well as for arrangements were the tritium breeding is improved by placing beryllium plates in the front part of the container just behind the wall.

In addition to the three dimensional calculations, calculations also were done in one-dimensional cylindrical geometry as described above in chapter 2. These calculations were mostly also done with MCNP in order to avoid differences coming from a different nuclear data base. Because the Kerma-factors of the MCNP library are rather outdated and partly very unreliable, S_N-calculations with the ONETRAN code were done for shielding and power density applications. All earlier calculations were done with ONETRAN.

The most important results, namely the tritium breeding ratio for the reference case and variations which include beryllium multipliers are summarized in Table 5. Included are data for other blanket concepts. Cases 1 to 5 are directly comparable, they were calculated with the same geometrical model. Only case 5 has inboard no blanket but a steel reflector. It would be rather difficult for this blanket concept to have an inboard blanket in the NET-DN geometry with integrated divertors.

Case 6 was taken from /16/. It is not really a three dimensional calculation because it ignores the side walls of the boxes. If these were taken into account the 3d-TBR value would decrease by about 5% and would fall below that of the aqueous LiOH solution without beryllium.

The results of one-dimensional calculations are included for illustration. The higher TBR values of the 1d-calculations partly result from the neglection of the sidewalls but mainly from the neutron losses in the top and bottom part which only are accounted for in the 3d-calculations, but not in 1d, which corresponds to 100% torus coverage by the blankets.

The ratio of 3d to 1d values is design dependent. It decreases when the reflectivity of the blanket to neutrons increases i.e. by addition of beryllium or lead. Then more neutrons are lost through the ports at the top and the bottom.

Table 5 One- and threedimensional Monte Carlo calculations of the tritium breeding ratio (TBR) of different blanket concepts.

No.	blanket concept	Be (cm) outb./inb.	TBR 3d	TBR 1d	$\frac{TBR_{3d}}{TBR_{1d}}$
1	aqueous	0/0	0.73	0.82	0.878
2	LiOH	10/10	0.92	1.10	0.836
3	solution	15/10	0.98	1.19	0.824
4	solid breeder blanket /3/		0.95	1.27	0.748
5	Pb-17 Li self cooled /15/		0.83	1.06	0.783
6	Pb-17 Li water cooled /16/		0.75	0.91	0.817

Before checking the impact of the tritium breeding on NET availability a few results from parameter variations will be given. In Ref. 2, heavy water was proposed instead of H_2O in order to utilize the neutron multiplication effects of deuterium. Table 6 shows what such a replacement would give for the NET conditions.

In spite of the slight increase in neutron multiplication there is no gain in the TBR when changing to D_2O, especially when neutrons are already multiplied by beryllium. The reason for this behaviour can be traced back to the worse slowing down properties of heavy water which are particularly serious for a thin blanket with a significant steel structure.

Important effects result from the graphite tiles and the LiOH in the first wall. This is seen from Table 7. The 1d calculation exaggerates the effects by nearly a factor of two, however /4/.

Table 6: The influence of heavy water on the tritium breeding ratio.

beryllium thickness (cm)		1d-TBR with		neutron multiplication	
outboard	inboard	D_2O	H_2O	D_2O	H_2O
0	0	0.84	0.82	1.18	1.10
15	10	1.02	1.06	1.61	1.57

As a consequence of the short migration length of neutrons in water tritium production is concentrated in the first 10 centimeters of the blanket and the tritium breeding ratio does not increase much with the blanket thickness. This is illustrated in Fig. 9 which shows the radial dependence of the tritium production rate in the aqueous LiOH solution.

Table 7: Impact of graphite tiles and LiOH in the first wall on the TBR (1d calculations)

Graphite tiles	LiOH in 1. Wall	TBR
yes	yes	1.10
yes	no	0.95
no	yes	1.25

Addition of a lithium salt to the coolant water has also an influence on the steel activation. More neutrons are absorbed in the coolant what means less neutrons are captured in steel which leads to reduced activation. This effect was estimated

from the heat generation in steel. A reduction by a factor of two was found to be a typical value.

The addition of a neutron absorber to the coolant in principle also improves the shielding effectiveness. But with boron being dissolved in the water coolant of the permanent shield anyway, the additional effect of lithium in the blanket was negligible.

3.3 Improvement of Availability

It was shown by the 3d calculations of Sect. 3.2 that even when all positions are occupied by tritium breeding segments of high breeding performance a tritium breeding ratio of 1.0 can not or may just be achieved in NET. Thus, for the really expected coverage tritium production is not self sufficient but only adds to the external tritium supply by purchase and will improve the availability once it is limited by the available amount of tritium.

In order not to obscure the essential point by too many details in the following only a very simple relation will be used which refers to an equilibrium situation and ignores tritium losses, radioactive decay, and stockpiling of tritium. In this case the following tritium belance equation holds:

$$\text{consumption} = \text{purchase} + \text{production} \qquad (1)$$

using the definition

$$\text{tritium breeding ratio} = \text{TBR} = \frac{\text{production}}{\text{consumption}} \qquad (2)$$

one gets:

$$\text{consumption} = \frac{\text{purchase}}{1\text{-TBR}} \qquad (3)$$

When plant availability is limited by the tritium supply the maximum availability is given by

$$a_{max} = \frac{\text{possible consumption per year}}{\text{consumption per full power year}} \qquad (4)$$

According to Ref. (1) the maximum possible purchase of tritium is 1.5 kg per year. The tritium consumption by the (D,T)-reactions is 0.153 g tritium per MWd. This gives for 600 MW fusion power (NET-DN reference solution) in the full power year

$$33.5 \text{ kg T per F PY.}$$

Thus, with tritium purchase only and without breeding one gets the maximum availability:

$$a_{max} = \frac{1.5}{33.5} = 4.48\%$$

what corresponds to 16.3 full power days per year or 1.1 hours per day on the average.

There will be a long period of NET operation where this is more than enough, but once NET is running well and could be applied for blanket testing, 16 days per year are not satisfactory. For the ceramic breeder blanket, for instance, it may take more than two weeks before tritium release reaches its equilibrium. The final goal of NET was set to 25% availability. With tritium breeding the maximum availability is given by

$$a_{max} = \frac{0.045}{1\text{-TBR}} \qquad (5)$$

Two dimensional neutronics calculations in (r,θ)-geometry have shown that the tritium breeding ratio varies linearly with coverage /4/ (see Fig.10), so that

$$\text{TBR} = \text{TBR}_{100} \cdot \text{coverage} \qquad (6)$$

with TBR_{100} = TBR for 100% coverage

$$\text{coverage} = \frac{48\text{-(numb. of non breeding segemnts)}}{48}$$

For TBR_{100} the value from the 3d calculation has to be used.

According to Ref.1 (page 254) 5 ports or 15 segments will be used for heating and diagnostics and are not available for breeding blanket introduction. In addition

four segments are occupied by test blankets which can not be assumed to deliver tritium. Thus, the maximum coverage c_{max} by driver blankets is

$$c_{max} = \frac{29}{48} = 0.604$$

That means one segment less used for heating or diagnostics has the same effect as an increase in local tritium breeding ratio of 3%. It may be easier in practice to save a few segments than to improve the driver blanket towards higher breeder performance.

All segments, also those used for plasma heating and diagnostics, need coolant and water coolant is generally the simplest solution. The lithium salt blanket has the advantage that also in these segments water can be replaced by the aqueous lithium salt solution and practically no loss or a much smaller loss in coverage occurs.

Table 8 shows the effective tritium breeding ratio TBR and the maximum availability a_{max} as calculated with Eq. (5) and Eq. (6) when different blanket concepts are used as driver. It has not yet been investigated how well the diagnostics and plasma heating segments can be utilized for breeding with an aqueous lithium salt solution.

Therefore two cases were considered:

a) breeding is as good as in an normal in-vessel shielding module (optimistic, cases 4 and 5, Table 8)

b) breeding is only half as effective as in a normal shielding module what is equivalent to the situation that only 7 segments are lost instead of 15. (cases 6 and 7, more realistic).

Table 8 illustrates that the poor breeding ratio of the lithium salt blanket is compensated by its better coverage even without beryllium. With beryllium it is clearly superior to any other concept and seems the only way to arrive at the required availability of NET with an external tritium supply of 1.5 kg per year.

Table 8 Maximum availability a_{max} for different driver blanket concepts.

No.	blanket concept	number of non-breeding segments heat.	tests	TBR c = 100%	TBR	a_{max} %
1	solid breeder /3/	15	4	0.95	0.57	10.5
2	Pb-17 Li self cooled /15/	15	4	0.83	0.50	9.0
3	Pb-17 Li H_20 cooled /16/	15	4	0.75	0.45	8.2
4	Li OH without Be	0	4	0.73	0.67	13.6
5	Li OH with Be	0	4	0.98	0.898	44.1
6	Li OH without Be	7	4	0.73	0.56	10.3
7	Li OH with Be	7	4	0.98	0.755	18.4

It must be emphazised however, that the figures of Table 8 are upper bounds since losses and radioactive decay of tritium are ignored. If it takes one year, for example, between tritium production in the blanket and the availability of the material as plasma fuel the tritium breeding ratios have to be multiplied by 0.945 to account for tritium radioactive decay. For case 7 it would mean 15.7% availability instead of 18.4%.

In principle, the aqueous lithium salt solution blanket has the advantage that once NET is running well it can be switched from pure shielding to tritium production without complicated and time consuming exchange of segments. It is important to note, however, that tritium is not immediately available when the lithium salt solution is filled in and tritium production starts. First the tritium has fully to be used for building up the hold-up in the tritium recovery system, especially in the isotopic separation plant. Only when the cascades are in equilibrium, product can be extracted. Let us denote the tritium inventory in the tritium recovery system by I and the delay time between start of tritium production and start of tritium extraction by t_I) then we have

tritium production rate x $t_D = I$

and with tritium production rate = 1.5·TBR /kg/a/

$$t_D\,/a/ = \frac{I/kg/}{1{,}5 \cdot TBR}$$

With TBR $\approx$ 0.66 one gets

$$t_D/years/ \approx I\,/kg/$$

as a simple rule of thumb approximation.

For the tritium recovery system discussed in the next section one gets with a crude estimate a delay time of a few months.

4. Tritium Recovery

The fact that the tritium produced is immediately dissolved in large amounts of water and has to be extracted from it by isotope separation is a disadvantage of the lithium salt self-cooled blanket concept.

Tritium extraction from water is not a principle problem, on the contrary it is a technique for which a large scale long term experience exists. The problems are cost and safety when the amounts of tritium are very large.

The process has to be selected on the basis of the following criteria:

- minimum capital cost
- minimum operating costs
- low tritium inventory
- low tritium concentration in H_2O

It is evident that these are conflicting requirements. For a given tritium production rate the amount of water which has to be processed varies inversely with the allowed maximum tritium concentration in it, and the processing costs will always increase with the throughput /18/. CANDU reactors and the HFR Grenoble operate with D_2O in the reactor vessel containing about 20 Ci/l tritium.

For NET generally a value of 1 Ci/l is assumed. It is based on good experience with CANDU-reactor tritium processing at this level /17/.

When the aqueous lithium salt blanket tritium processing plant would use a feed concentration of 1 Ci/l several units had to be operated in parallel. Then the losses are presumably not much smaller than with a higher tritium concentration and a smaller number of units. Thus, a feed water concentration of 10 Ci/l was tentatively assumed.

The amount of water in the blanket and primary circuit is about 200 m^3. With 10 Ci/l this corresponds to a tritium inventory of 2 MCi or 200 g tritium.

Based on the operating experience of the Karlsruhe research reactor FR2 tightness of the circuits will be good enough to keep the losses - extrapolated to NET dimensions - in the order of 0.4 ml/d. Maintenance losses come in the range of 50 l/year what corresponds to an average of 1.4 Ci/d for the primary and secondary circuit of the aqueous lithium salt blanket.

The amount of tritium produced in equilibrium can be estimated from Eq. (1) and Eq. (3) of Sect. 3.3.

$$\text{tritium production} = \frac{\text{purchase}}{\text{1-TBR}} - \text{purchase}$$

taking a purchase of 1.5 kg tritium per year and TBR = 0.70 one gets

$$\text{production} = 3.5 \text{ kg T per year}$$

or on the average

$$9.6 \text{ g per day}$$

what means roughly

$$10^5 \text{ CiT/d} \quad \text{or} \quad 10^4 \text{ l/d} \quad \text{with} \quad 10 \text{ Ci/l}.$$

When operating at full power, 64.3 gT/d are produced so that a buffer storage is needed to smooth out the tritium production rate in order to arrive at 10g/d.

Up to now no detailed analysis of the process options and the process costs were done. The following estimates are entirely based on a draft version of a report of

S.K.Sood and O,K.Kveton /17/ which refers to data quite close to what would be needed here.

For reasons of simplicity direct electrolysis followed by cryogenic distillation for isotope separation is tentatively selected as the tritiated water processing system although it may not be the ultimate choice.

With high probability the Li OH, $LiNO_3$, or Li_2SO_4 of the solution can directly be used as electrolyte for the electrolysis. If this is not the case one could start with a distillation step which would only marginally increase the overall tritium extraction costs.

In the following we assume that this step is not needed what is fairly shure for LiOH.

In case of LiOH isotopic exchange between LiOH and HTO will occur. The fraction of tritium which is bound as LiOT will approximately be the same as the ratio of H in LiOH to total H-content in the solution. For 127g LiOH per liter this is 4.7%.

A schematic diagram of the process is shown in Fig.11 for the case of LiOH. It would be similar for the other Li-compounds, but without the 0.5 g tritium per day recycled in the form of LiOT.

Lithium burn-up has to be compensated by the addition of LiOH what is a very small effect in NET.

The total primary loop inventory is roughly $2 \cdot 10^5$ kg solution 5% of which is extracted per day for tritium recovery. With a tritium production rate of 10 g per day this leads to a tritium inventory in the primary loop of 200 g. When NET is running at full power for 10 days or more the tritium inventory would go up to 1270g.

According to /17/ the electrolysis step leads to a tritium pre-enrichtment by about a factor of ten. This would give at the final stage of a batchwise processing 100 Ci/l, which is beyond the 70 Ci/l limit quoted in /17/. But one can easily run the process in a way that the limit is kept at the expense of a slightly lower pre-enrichment.

Cost estimates for tritium extraction from water by different processes are given in /17/. For direct electrolysis followed by cryogenic distillation estimates are

made for a feedwater flow of 11000 kg/d and both, 1 Ci/kg and 4 Ci/kg feed concentration. Capital costs are the same for both cases namely

electroysis	5.0 MECU
cryogenic distillation	17.0 MECU
	22.0 MECU

In our case the flow is with 10000 kg/d very similar but the tritium concentration is 10 Ci/kg what may increase the capital cost.

In the INTOR Phase Two A Report a capital cost formula is given for combined electrolysis and catalytic exchange followed by cryogenic distillation /18/. It reads:

$$c = 15 \left[\frac{\text{tritium mass flow Ci/d}}{(\text{tritium concentration Ci/l}) \cdot 1000} \right]^{0.55}$$

with

c = capital cost in Million US $ (1982)

This leads for 10^5 Ci/d and 10 Ci/l

to c = 53 M $ and with 1$ = 1 ECU

to c = 53 MECU

Power costs can be estimated to about 2 MECU per year with supposedly a similar figure for cost of personnel.

5. Conclusions

One of the most important decisions to be taken in the NET technology development concerns the use of a water cooled first wall. If this is acceptable from the safety point of view - at least for low temperature, low pressure water - than water cooled in - vessel shield modules are a logic consequence. They combine excellent shielding performance with simple technology, low temperature, small temperature gradients and low stresses.

When the modules are optimized for simplicity and adequate shielding performance they contain a large volume fraction of water. Then they are very well suited as self cooled blanket modules with an aqueous lithium salt solution without any design modification.

This blanket has a tritium breeding ratio comparable to other blanket concepts. When the lithium salt solution is also used as coolant instead of water in all plasma facing components and in the segments dedicated to plasma heating and diagnostics the breeding performance is superior to all other blanket concepts.

The decision to use this option can be taken rather late in the program, because it has practically no influence on the in-vessel components or components in the main building. But it would require a special tritium processing plant somewhere.

Tritium breeding can further be improved by placing beryllium plates in the front part of the blanket container. This requires a different design of the container and has to be foreseen from the beginning. Otherwise an exchange of blanket segments would be necessary.

Even with the highest performance driver blanket, 25% availability can possibly not be achieved in NET in an equilibrium condition. One year operation with 25% availability is possible by some stockpiling of tritium in the years before. Based on external tritium supply only the NET objectives can not be met; this would also be very questionable with a (Pb-17 Li) water cooled driver.

All the scoping studies done so far for the aqueous lithium salt blanket conclude the concept feasibility at least for low temperatures /2,7,19/. For final assurance however, more R and D work is needed. This includes

- extensive corrosion measurements, first of scoping nature and then under NET relevant conditions and selection of the best candidate material,
- analysis and design work for the tritium processing plant,
- examination of safety issues and design measures against accidents,
- coolant system design of the plasma heating and diagnostic segments for optimum tritium breeding with an aqueous lithium salt solution,

- investigation of segment electrical insulation in spite of the conductive coolant.

- investigations on radiolysis

The design work and fabricability tests of the in-vessel shield segments have to be done anyway.

The required extra R. and D. effort is modest when compared with other blanket concepts and may be justified also when the concept has no potential for use in a fusion power reactor.

When the lithium salt blanket is used as a tritium producer for NET the other blanket concepts should not be considered as alternatives in this function. But then, for them the aspect of prototypicality is strengthened, that means the concepts should have the potential to be developed for fusion power reactor application with tests in NET as an important step.

This leads to the question whether the aqueous lithium salt blanket also has the potential to be developed for use in fusion power reactors.

Some corrosion tests at elevated temperature look promising /7,20/. With more investigations on inhibitors and/or appropriate steel compositions it may be possible to solve the corrosion problem. But the design has to be in the form of pressure tubes what significantly degrades the breeding potential (more steel, less good coverage, less efficient beryllium positioning). It is questionalbe whether adequate tritium breeding is feasible. Conclusions from calculations in idealized geometry may be misleading.

References

1. Next European Torus, Status Report 1985, NET Report 51, CEC Brussels 1985

2. D. Steiner et al. "A Self-Cooled Heavy Water Breeding Blanket" Proc. 11th. Symp. on Fusion Engineering, Austin Texas 1985, 539.

3. M. Dalle Donne et al. "Pebble Bed Canister: A Ceramic Breeder Blanket with Helium Cooling for NET", 14. SOFT Avignon 8.-12. Sept. 1986

4. U. Fischer: "Multidimensional Neutronics Analysis of the "Canister Blanket" for NET", KfK 4255, in press

5. R.T. Hill: "ONETRAN, a Discrete Ordinates Finite Element Code for the Solution of the One-dimensional Multigroup Transport Equation" LA-5990-MS (1975)

6. W. Maurer "Neutron and Gamma Irradiation Effects on Organic Insulating Materials for Fusion Magnets", KfK 3974 (1985)

7. F. Bogaerts, J. Embrechts, R. Waeben, "Application of the Aqueous Self-Cooled Blanket Concept to a Tritium Producing Shielding Blanket for NET" University of Leuven Jan. 1987.

8. J. Schultz "Recommendation of Neutron Irradiation Limits on the Toroidal Field Magnets of ETF" MIT letter 1980

9. ABC der Stahlkorrosion, Mannesmann Verkaufsgemeinschaft Düsseldorf

10. E. Rabald, H. Bretschneider DECHEMA-Werkstoff-Tabelle, DECHEMA Frankfurt 1959

11. MCNP A General Monte Carlo Code for Neutron and Photon Transport LA 7396 Los Alamos 1981

12. K.A. Verschuur: "FURNACE, A Toroidal Geometry Neutronic Program System", Report ECN-162 (1984)

13. W. Dänner NET-Team, private communication

14. P.G. Young, L. Stewart: "Evaluated Data for n + ^{9}Be-Reactions" LA-7932-MS (1979)

15. S. Malang, V. Casal, K. Rust, K. Arheidt, U. Fischer, K. Link "Liquid Metal Cooled Blanket Concept for NET" 14. SOFT, Avignon 8.-12. Sept. 1986

16. W. Dänner, M. Rieger, K.A. Verschuur "Progress in Design and Analysis of the NET Water Cooled Liquid Breeder Blanket" 14. SOFT, Avignon 8.-12. Sept. 1986

17. P.J.Dinner, M.Chazalon, S.K. Sood, O.K. Kveton, "Tritium System Concepts for the Next European Torus Project", Fusion Techn. Vol. 8 2228 (1985)

18. INTOR, International Tokamak Reactor Phase Two A Part I IAEA, Vienna 1983 (p. 349)

19. D. Steiner et al. "Applications of the Aqueous Self-Cooled Blanket Concept" Fus. Techn. 10 Part 2A, 641 (1986)

20. M.C.Bloom, W.A.Fraser, M.Krulfeld "Corrosion of Steel in Concentrated Lithium Hydroxide Solution at 316C". Corrosion 18, 401 (1962)

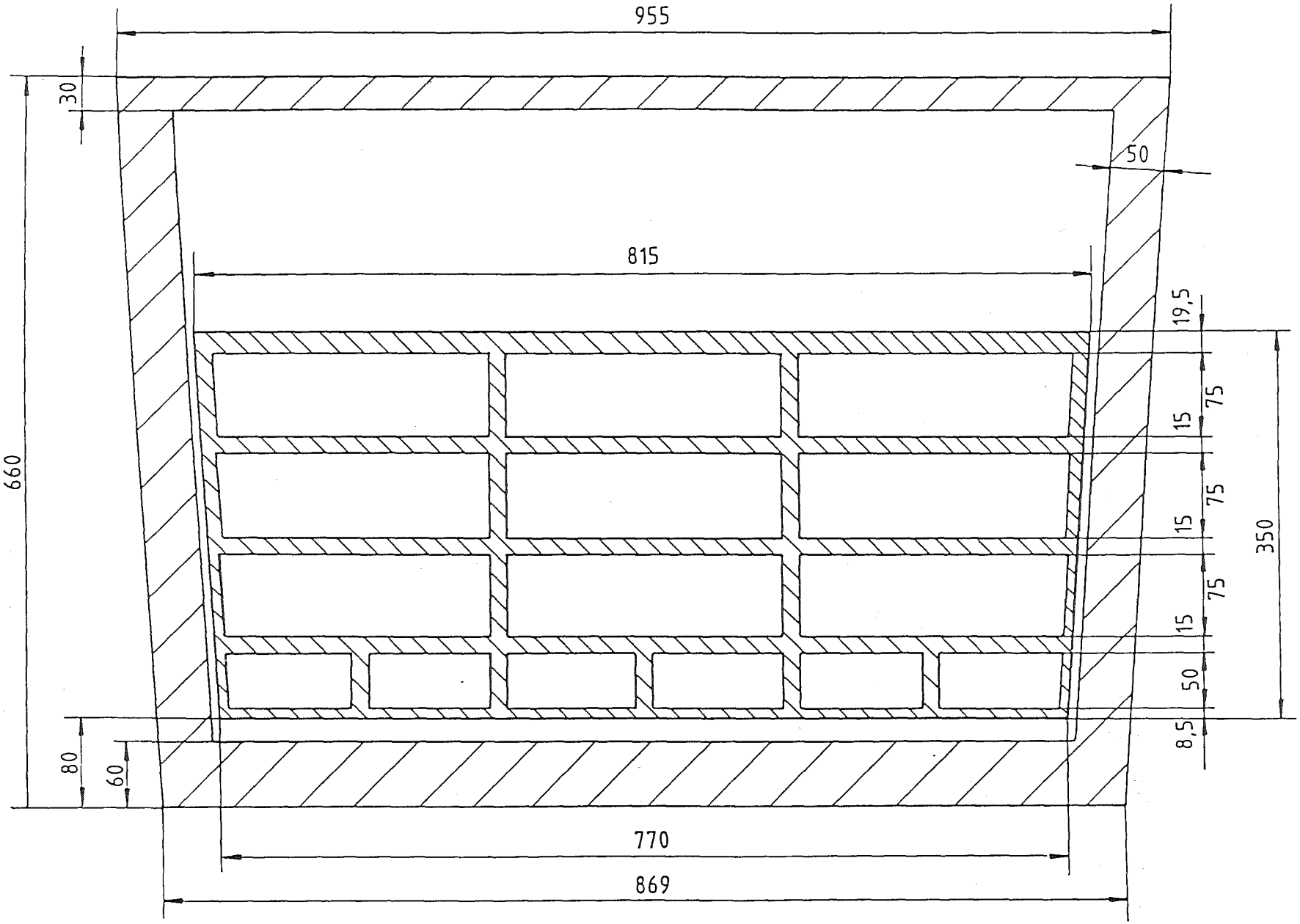


Fig. 1 Toroidal midplane cross section of the water cooled dummy blanket

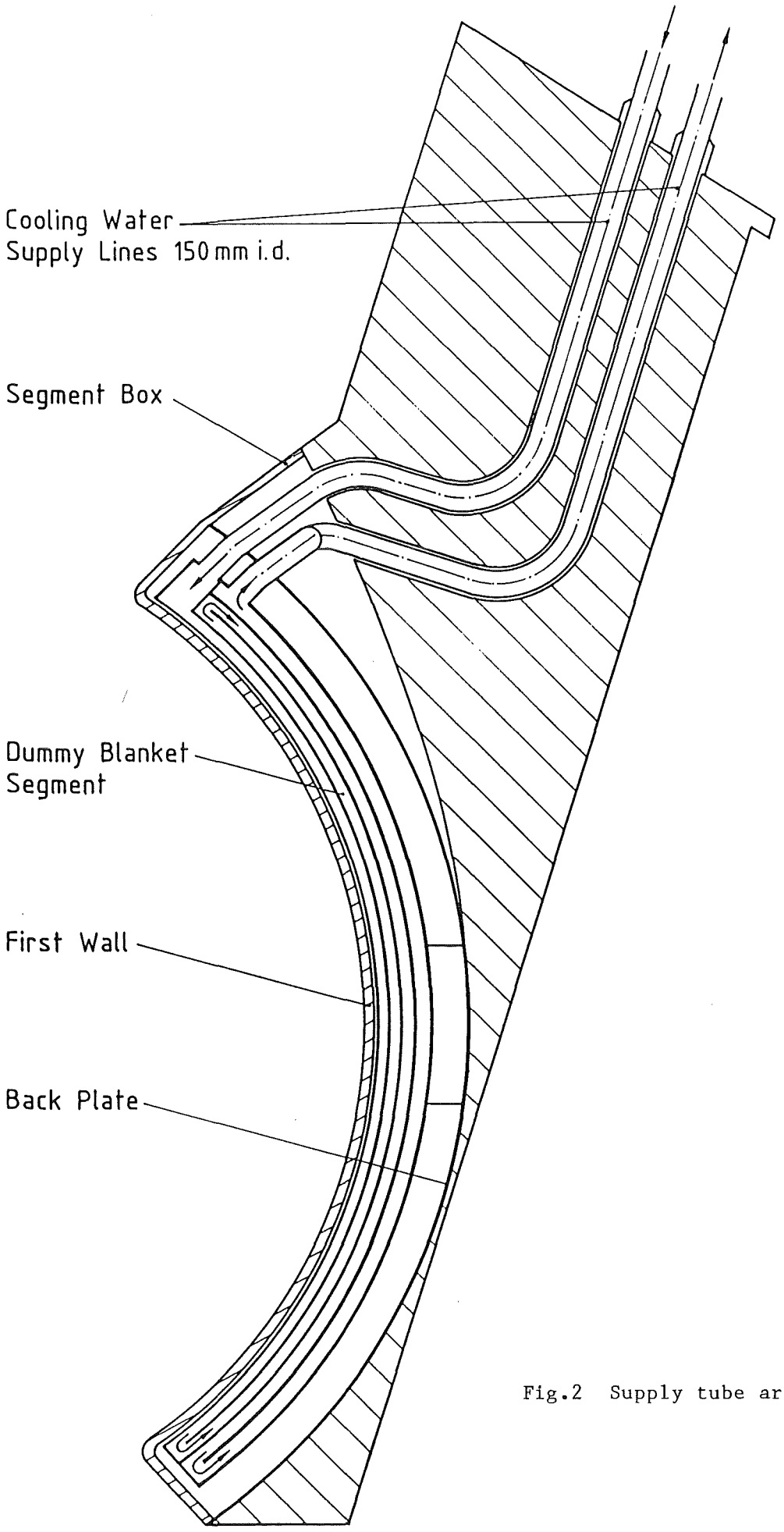


Fig.2 Supply tube arrangement

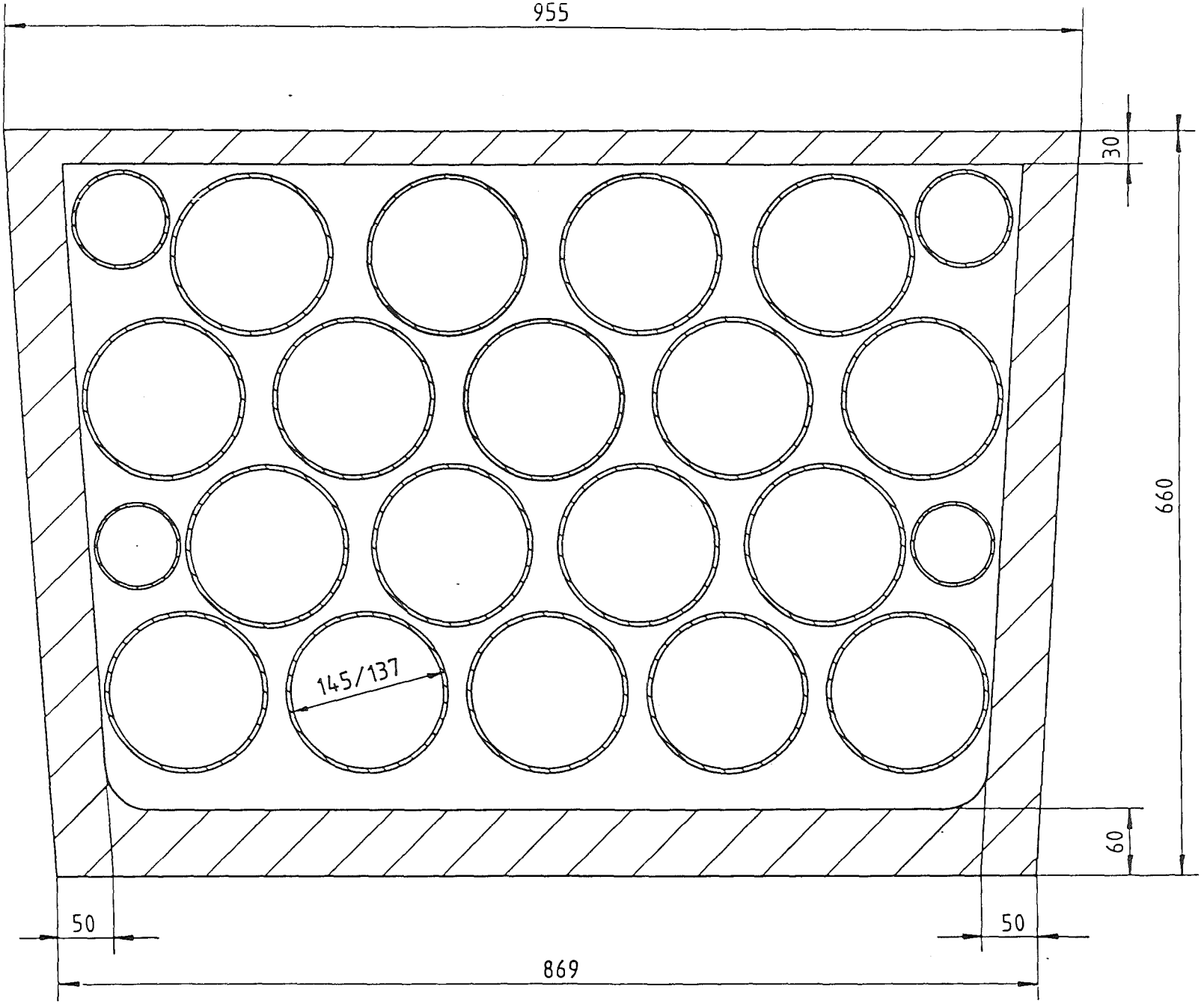
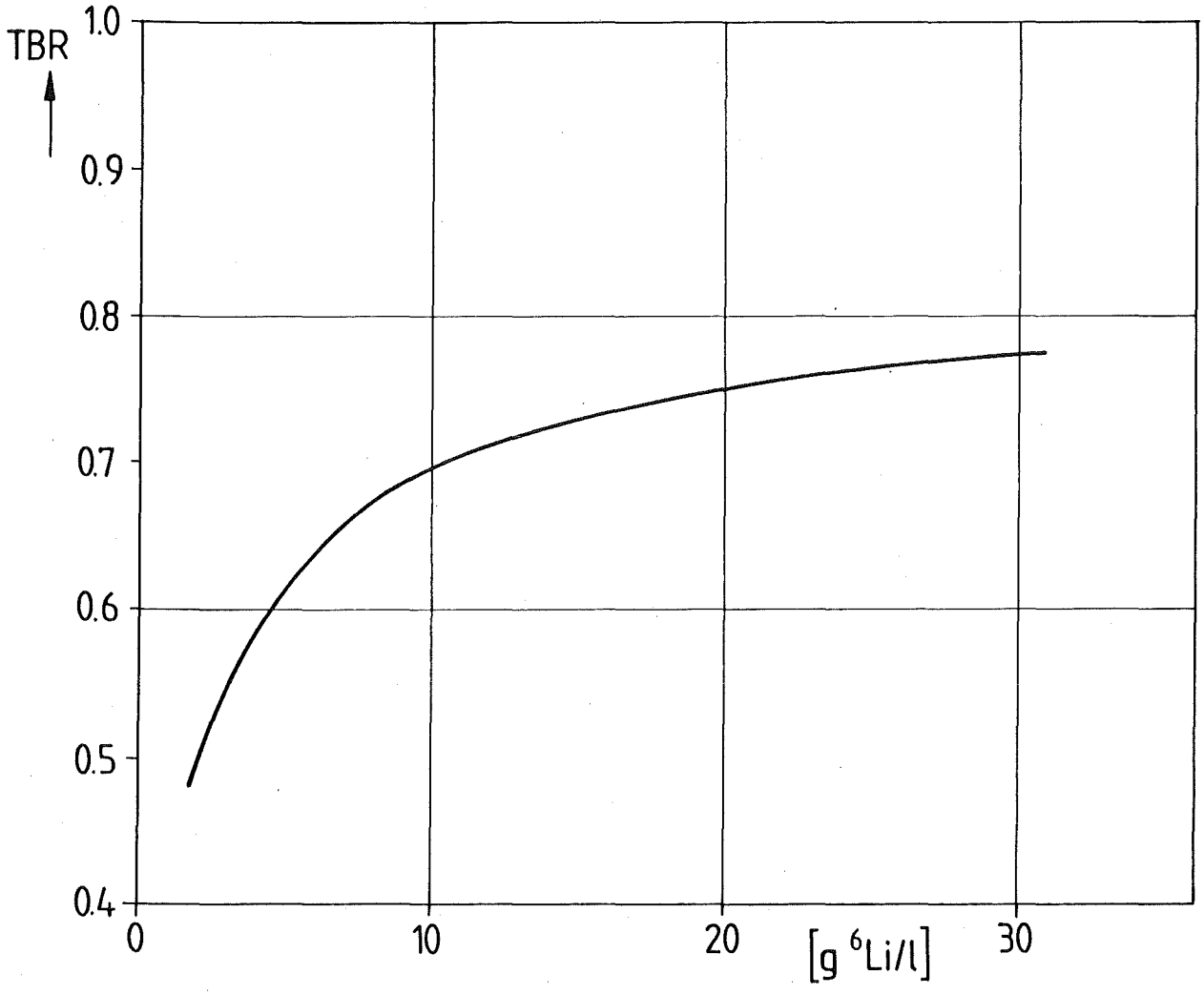


Fig. 3: Toroidal midplane cross section of the water cooled dummy blanket, concept with tubes



LiOH	7.5	30	60	90
$LiNO_3$	7.5	30	60	
Li_2SO_4	7.5	30	60	90

Fig. 4: Tritium breeding ratio (TBR) vs. ^{6}Li-concentration in the aqueous lithium salt blanket. Lower scales: ^{6}Li-enrichment (%) to achieve the ^{6}Li-concentration in 20°C-saturated solutions

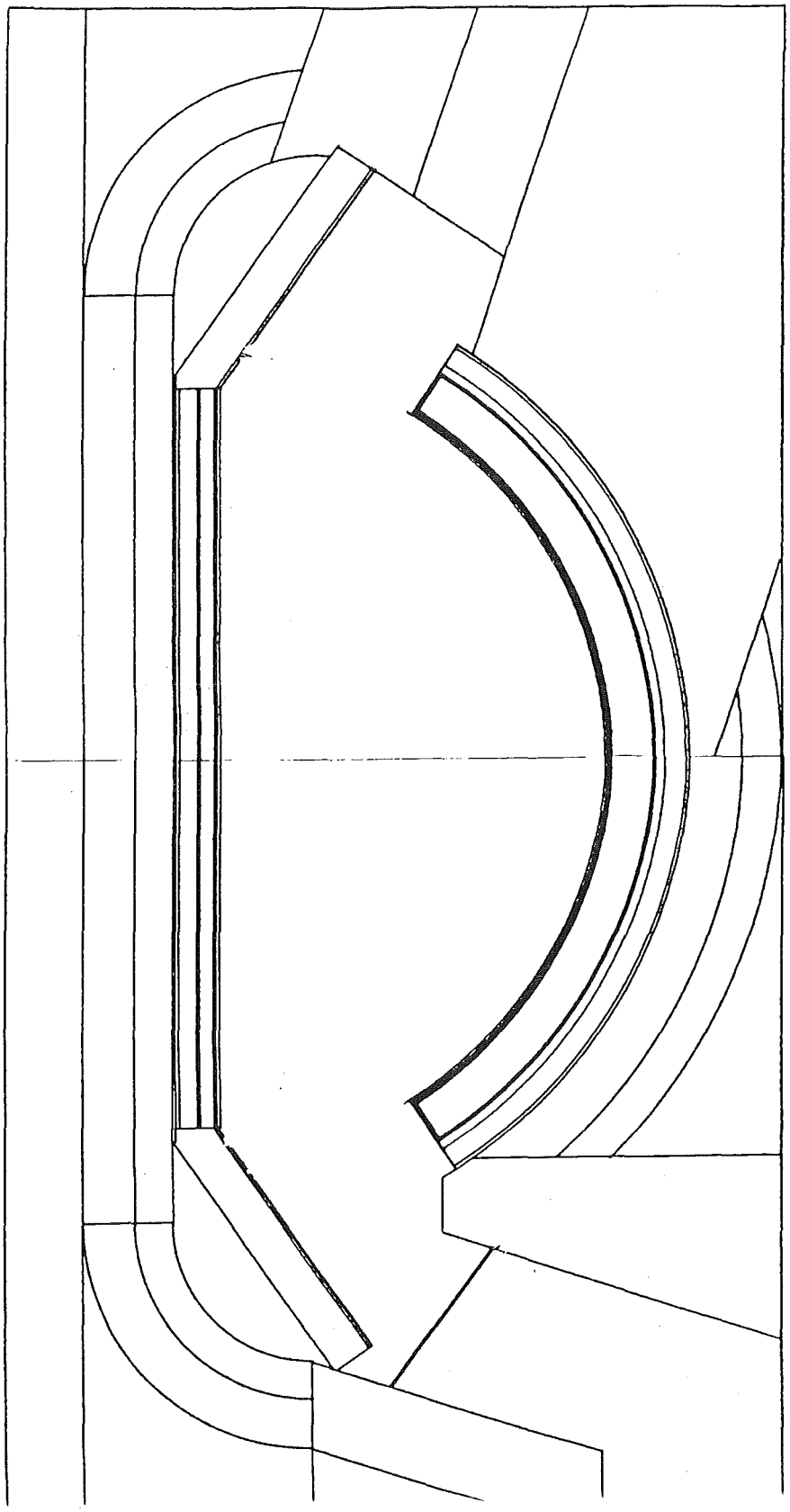


Fig.5 Poloidal cut of the geometrical representation of the lithium salt solution blanket in the MCNP calculation

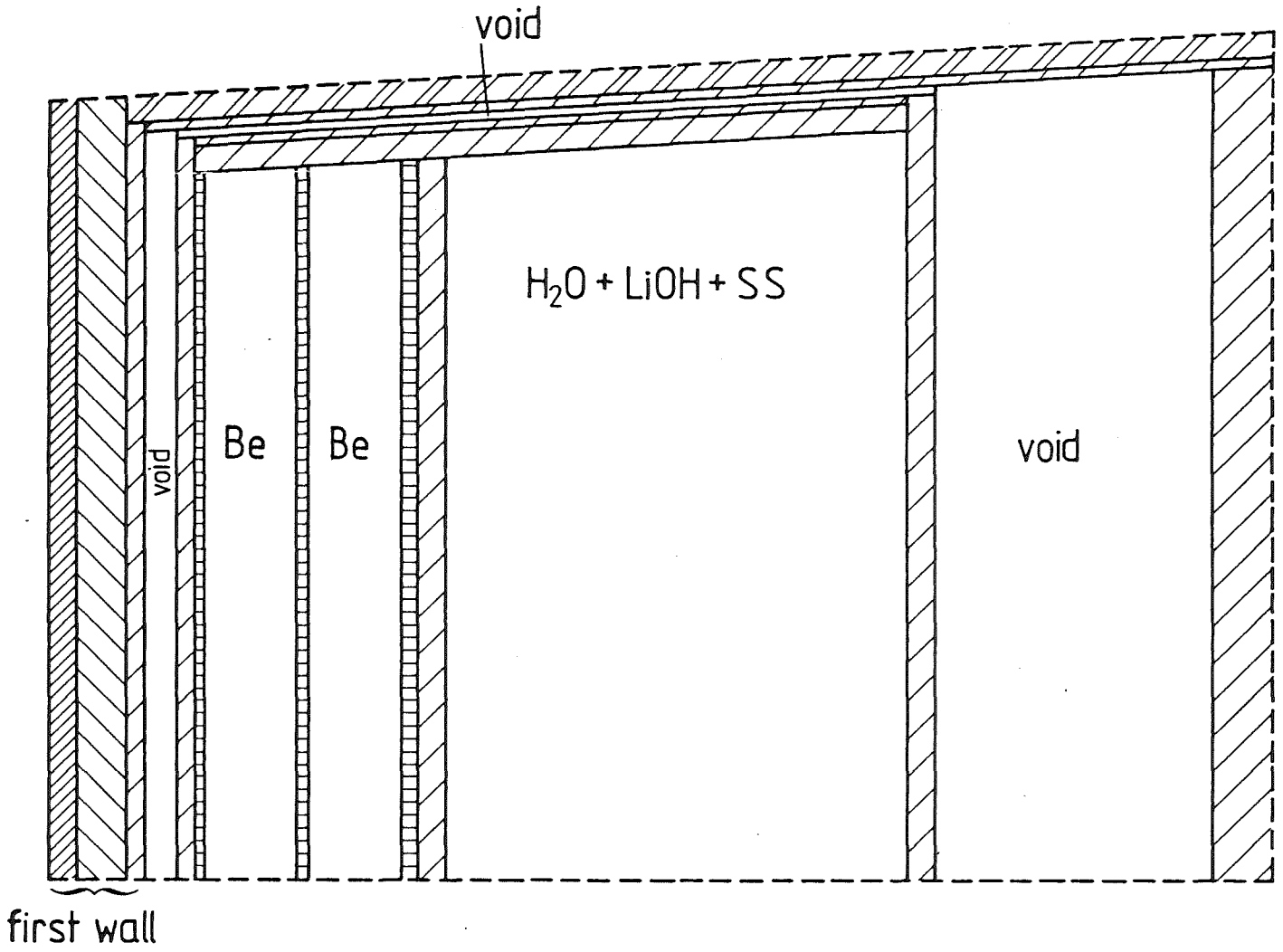


Fig.6 Toroidal midplane cross section of the outboard blanket with beryllium plates (half of a 7.5° sector)
3d MCNP-calculation

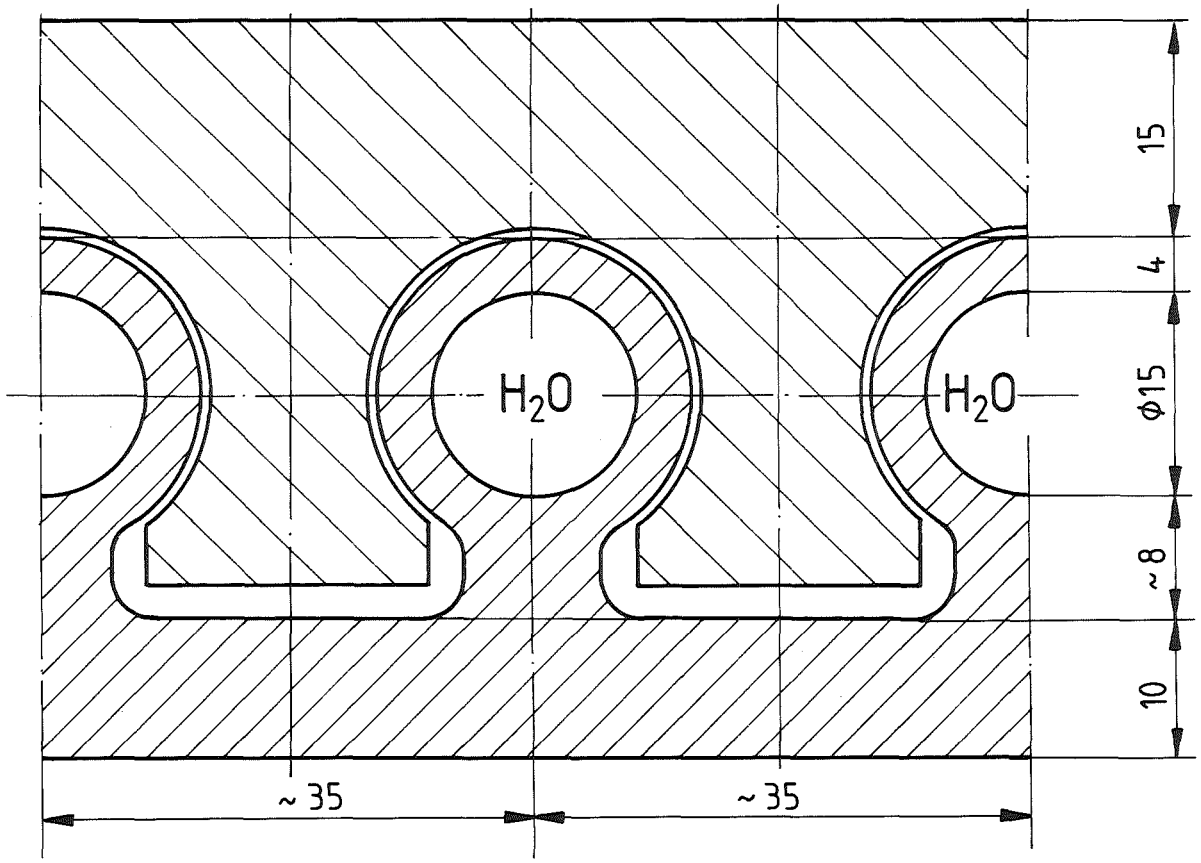


Fig. 7 water cooled first wall with ratiation cooled graphite tiles

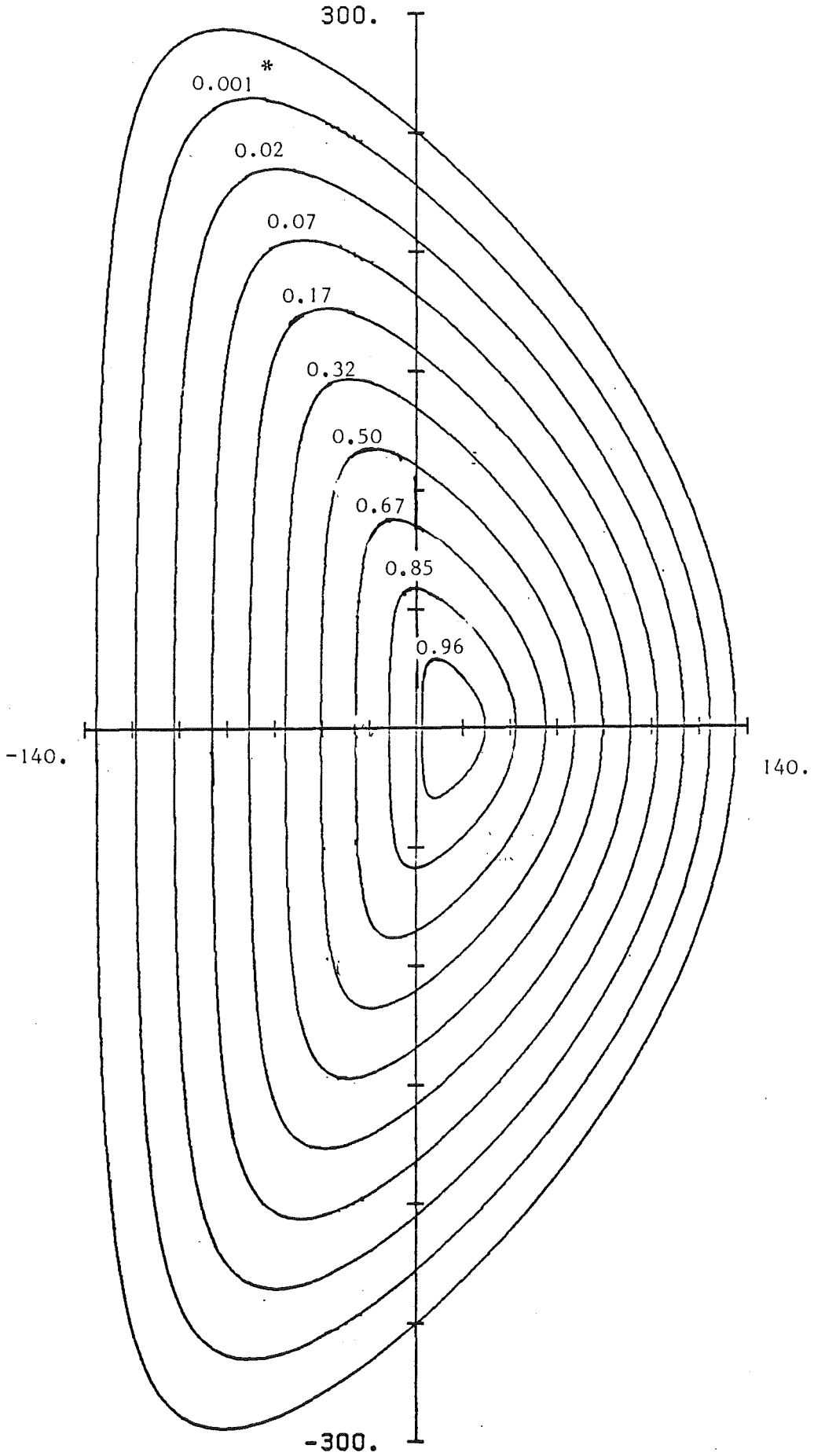


Fig. 8 Poloidal cut, surfaces of equal neutron source intensity

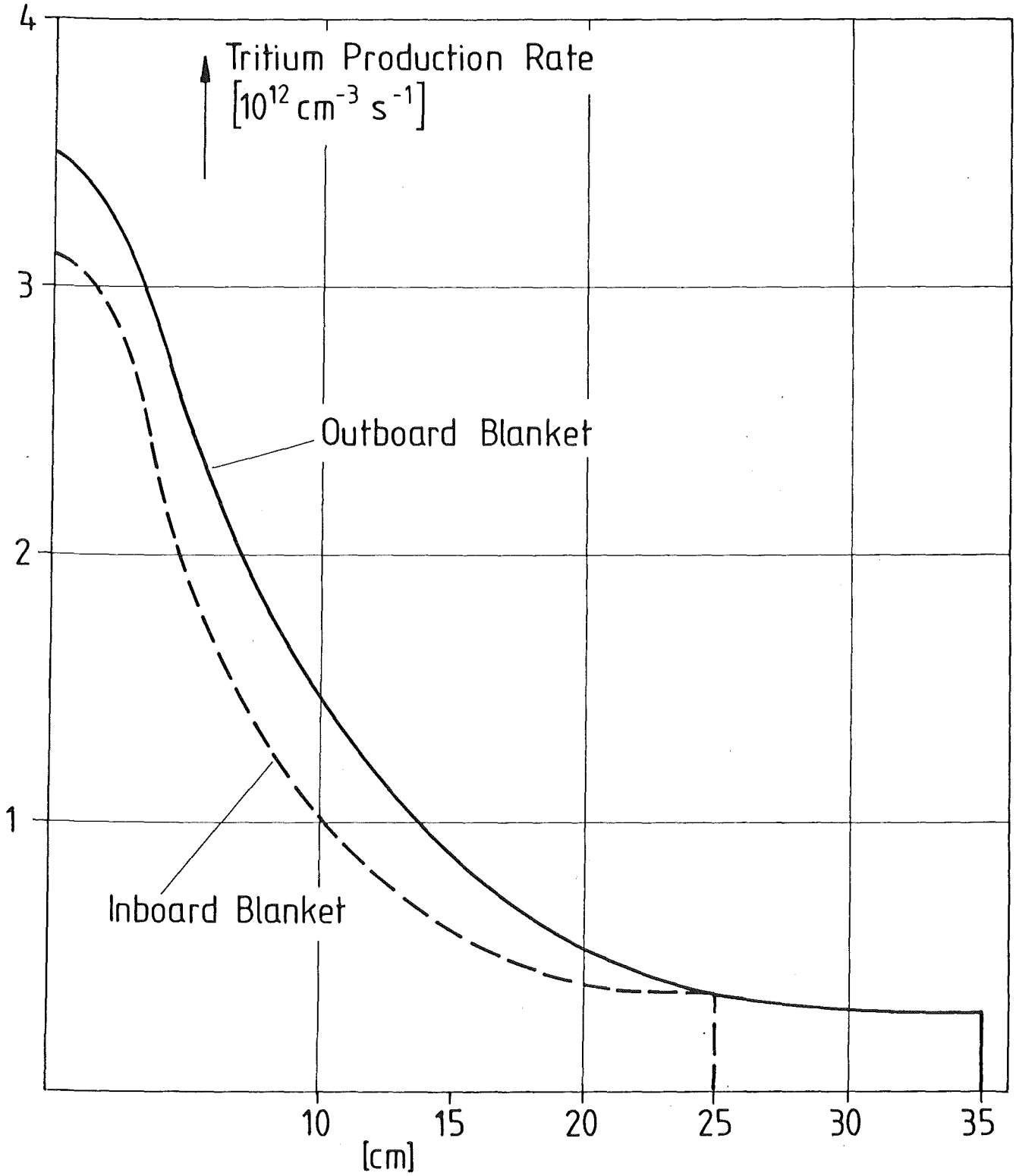


Fig. 9 Radial distribution of the tritium production rate in the LiOH-solution (outboard) 3d-MCNP calculation

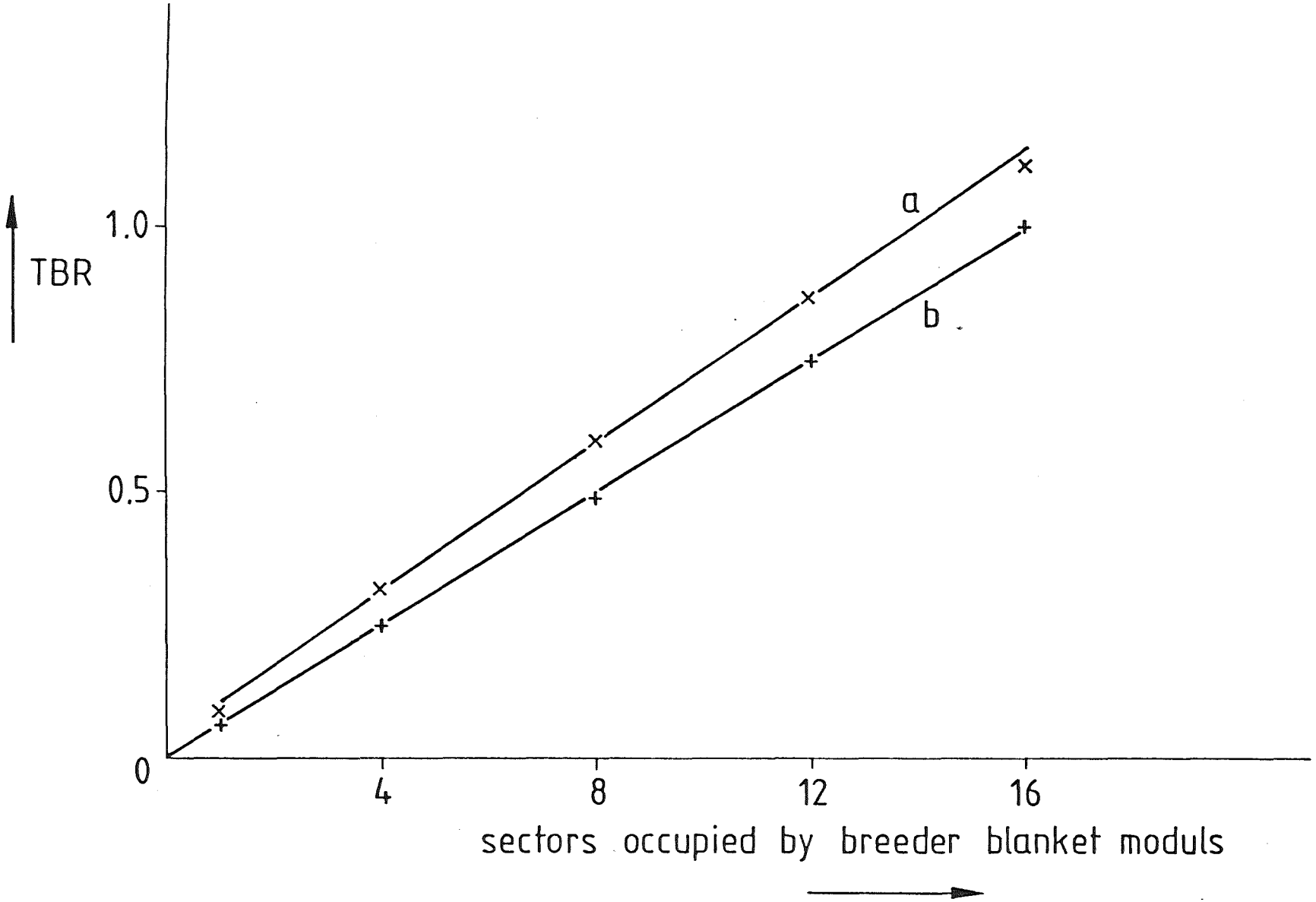


Fig.10 Tritium Breeding Ratio versus number of outboard sectors occupied by breeding blankets. Rest of outboard sectors and inboard:
a) helium cooled stainless steel shield
b) 10 vol % H_2O cooled SS.
Ceramic breeder blanket, canister concept

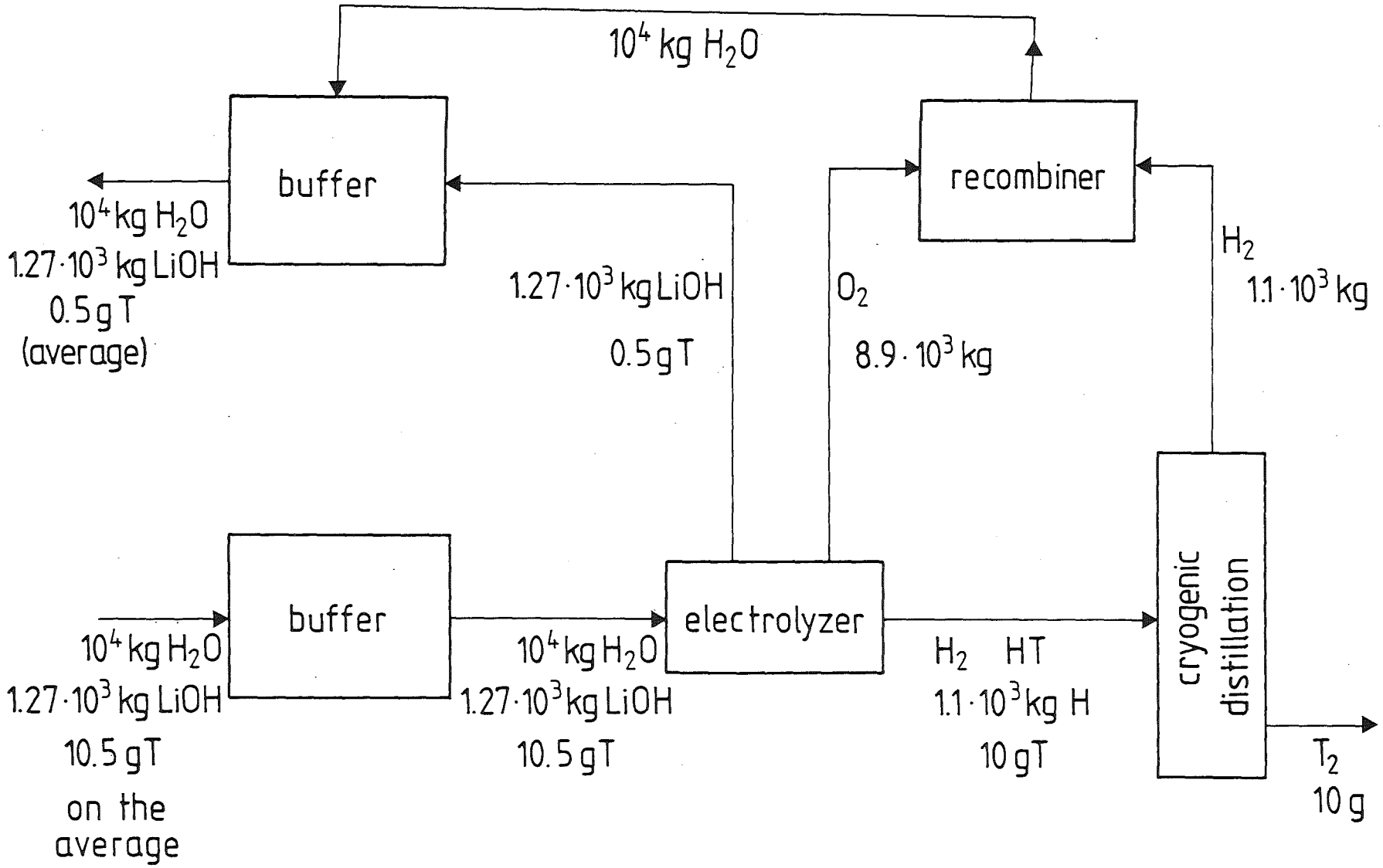


Fig. 11 Tritium Recovery LiOH Blanket 10 Ci/l daily throughput