

Tritium Economy and Tritium Safety  
of Fusion Power Plants.

W. Dänner

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**MAX-PLANCK-INSTITUT FÜR PLASMAPHYSIK**  
**GARCHING BEI MÜNCHEN**



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The second part deals with the application of diffusion forced tritium removal methods. Two different solutions are investigated: tritium extraction from the secondary circuit and inside a bypass of the main string of the primary loop of a fusion reactor power conversion system. Six possible types of thermal power cycles are investigated.

Special effort was devoted to the bypass solution, which offers some advantages in the way of system independence, flexibility, and safety. A method of optimizing such a device is described. In order to demonstrate this procedure, a number of numerical calculations were performed, the results of which are presented in a series of tables.

The most essential of these results are the tritium concentration, the initial inventory, the doubling time, the lithium inventory, and the tritium extraction rate. Besides these economic aspects, the safety aspects of the tritium removal system were also investigated. The tritium risk potential were evaluated.

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Abstract

Tritium economy and tritium safety are key problems in fusion reactor development. Both problem areas are strongly connected with the reactor design and the kind of thermal power conversion system applied. Of special importance is the method of tritium recovery, which additionally determines the tritium concentration level in the liquid metal.

In the first part of the report the inter-relationships between the quantities governing questions of economy are outlined.

The second part deals with the application of diffusion forced tritium removal methods. Two different solutions are investigated: tritium extraction from the secondary circuit and inside a bypass of the main string of the primary loop of a fusion reactor power conversion system. Six possible types of thermal power cycles are investigated.

Special effort was devoted to the bypass solution, which offers some advantages in the way of system independence, flexibility, and safety. A method of optimizing such a device is described. In order to demonstrate this procedure, a number of numerical calculations were performed, the results of which are presented in a series of tables.

The most essential of these results are the tritium concentration, the initial inventory, the doubling time, the lithium inventory, and the permeation surface area requirements. Besides these economic quantities, some safety parameters allowing conclusions about the contamination of the secondary circuit and the tritium risk potential were evaluated.



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## 1. Introduction

Tritium economy is one of the most crucial problems in fusion reactor development. To cover this problem as a whole, it is not sufficient to calculate the nuclear characteristics of the tritium breeding blanket, e.g. the tritium breeding ratio. This information has to be supplemented by detailed knowledge, especially about the tritium inventory of a fusion reactor.

As initial inventory, this is a decisive cost factor, at least for a first generation of fusion reactors, which is closely connected with the question of availability. As total inventory, it is the basis for estimating the tritium safety both during reactor operation and in the event of accidents. The initial inventory and breeding ratio are responsible, finally, for the size of the doubling time, which is an important economic quantity for any long-term fusion reactor strategy.

## 2. Present Knowledge

The evaluation of the tritium breeding capability of fusion reactor blankets has meanwhile reached a stage where nobody doubts that breeding ratios above 1.0 are possible. A lot of papers [1 - 8] are devoted to this problem. These present values of up to about 1.7, with allowance for a rather realistic amount of structure material within the blanket. However, up to now preferentially niobium and to a less extent molybdenum have been considered for this purpose. In the near future it might be useful to have equivalent data for the use of vanadium or stainless steels, too. Furthermore, it should be noted that all these calculations are concerned with an idealized blanket, i.e. a completely compact one. There is great



probability, however, that there will be a lot of penetrations within the blanket providing space for the heating, refuelling, and vacuum equipment, which surely will reduce the theoretical breeding ratio by a few per cent.

As opposed to the situation in the field of neutronics calculations, very few attempts have been made to estimate the tritium inventory of a fusion reactor. This is because for such evaluations it is necessary to make a statement about the kind of tritium recovery method. The development of consistent ideas on this problem, however, is just at the beginning. To our knowledge, only one author has treated this topic. A.P. FRAAS [9, 10] based his investigation on a tritium removal process involving diffusion of the tritium from the primary lithium to a secondary potassium cycle inside the potassium steam boiler. At the low temperature end of the potassium power cycle tritium should concentrate and be drawn off either as a noncondensable or chemically bonded as potassium tritide or lithium tritide in cold traps. By applying these methods it is said to be possible to keep the tritium inventory down to 10 kg or, as stated in the latter paper, even to about 2 kg.

In our opinion this tritium removal system has two disadvantages:

- 1) It is strongly fixed to the overall fusion reactor concept proposed by the author and therefore not applicable to other cooling systems, e.g. a direct helium power cycle or a steam power cycle avoiding the potassium topping cycle.
- 2) Even if the potassium cycle is used, it could be, for safety reasons, a disadvantage to have the secondary coolant cycle contaminated by tritium.

### 3. Scope of the Present Investigation

The present paper deals with the problem of tritium economy, being essentially based on estimates of the tritium inventory. For tritium removal, too, a diffusion process is assumed. Beside the method proposed by FRAAS a further tritium recovery system is investigated characterized by a fluidized bed on the secondary side of the diffusion path. This system was at first considered by WEICHSELGARTNER [11,12]. This method has the advantage that it can be applied to a number of different power cycles. An estimate was made for six such possible cycles. The results of these evaluations do not only yield the tritium inventories but they also allow conclusions concerning safety problems.

The work presented here is a continuation and a summary of that reported in [29].

### 4. Inventory, Breeding Ratio, and Doubling Time

The relationship between inventory  $I$ , breeding ratio  $B$  and doubling time  $t_2$  was first established by STEINER[3]. The doubling time defined in this paper was related to the total tritium inventory, which, in our opinion, is not a good choice. It is our conviction that a better definition would be afforded by relating it to the initial inventory  $I_0$  necessary for the start-up of a new reactor. Indeed, there is not much difference if the breeding ratio  $B$  is only slightly above unity. But to achieve short doubling times,  $B$  must exceed 1.0 by a few per cent and the same percentage is valid for the difference between the total and initial inventories.

In his numerical example STEINER used for the total inventory the figures established by FRAAS [9]. He emphasizes, however, that there is urgent need to estimate the tritium holdup in detail.



A similar derivation of the relation between the three quantities  $I$ ,  $B$  and  $t_2$  was done by VOGELSANG [14] taking into account the radioactive decay of tritium. As a measure of the tritium holdup he introduced a mean residence time of tritium in the blanket. In this case, too, we doubt that the choice of this quantity is a good one because it is too far away from technological features.

We recommend using the mean tritium concentration in lithium  $C_{Li}$  as the free parameter instead of the mean residence time because this quantity is a more realistic one with regard to the diffusion mechanism. According to fig. 1 we define the following quantities:

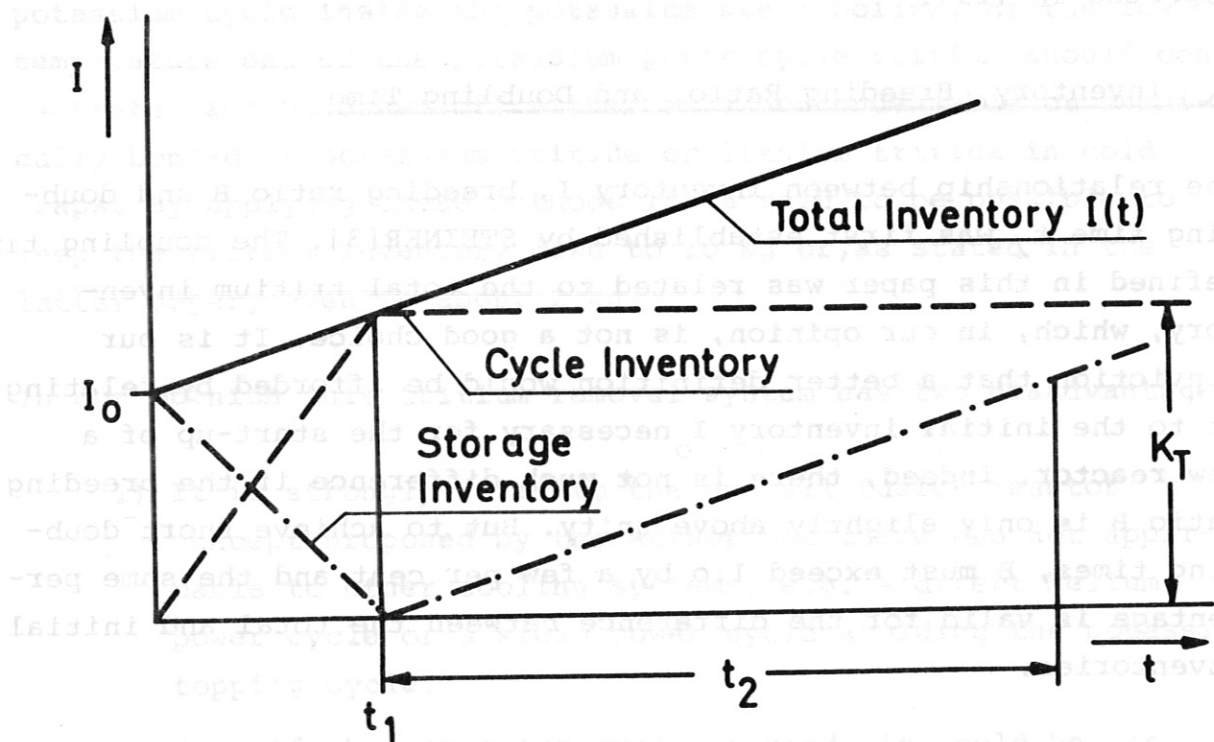


Fig1: Definition of Tritium Inventory and Doubling Time

At reactor start-up an initial inventory  $I_0$  must be available inside a suitable storage. During the first days of operation the reactor is supplied by this storage until it is exhausted at the time  $t_1$ . During this time the tritium concentration in the lithium cycle rises because of the permanent tritium production inside the blanket. At the time  $t_1$  the concentration has reached a level high enough to force the diffusion process. From this moment the reactor is supplied by the tritium recovery system, while the concentration inside the lithium cycle remains constant. The tritium inventory at this stage will be called the tritium capacity  $K_T$ . If the breeding ratio exceeds  $B = 1.0$ , an excess inventory is bred which will be fed to the storage. After the doubling time  $t_2$  this excess inventory will just have reached the value of the initial inventory  $I_0$ .

The essential quantity in this consideration is the tritium capacity  $K_T$ , which can be expressed by the following equation:

$$K_T = C_{Li} \cdot G_{Li} + C_s \cdot G_s \quad (1)$$

In this equation  $C_{Li}$  means the tritium concentration inside the Lithium and  $G_{Li}$  the lithium weight inside the whole primary lithium cycle. Since the structure materials in contact with the liquid metal will also be loaded with tritium, the corresponding inventory must be added, this being given by the mean tritium concentration in the structure  $C_s$  and its weight  $G_s$ .

With the tritium capacity thus defined the initial inventory  $I_0$  becomes

$$I_0 = \frac{K_T}{B} \quad (2)$$

With the tritium consumption rate of the reactor  $V_T$  the total inventory  $I$  can be calculated as a function of time  $t$ :

$$I(t) = I_0 + V_T(B-1) \cdot t \quad (3)$$



The total inventory just equals  $2 I_0$  when  $t$  has reached  $t_2$ .  
From this condition the doubling time  $t_2$  can be evaluated:

$$t_2 = \frac{I_0}{V_T (B-1)} \quad (4)$$

The time  $t_1$  when the concentration  $C_{Li}$  has reached its optimum value can be expressed either by the initial inventory  $I_0$  and the consumption rate  $V_T$  or by the tritium capacity  $K_T$  and the production rate  $P_T$ :

$$t_1 = \frac{I_0}{V_T} = \frac{K_T}{P_T} = \frac{K_T}{B \cdot V_T} \quad (5)$$

## 5. Estimate of the Tritium Capacity

As indicated in the preceding section, the tritium capacity of the entire lithium bearing system is the quantity of main interest in this investigation. It consists of one part joined to the lithium and another joined to the structural material. The latter one can be neglected in this estimate. It is of great importance, however, for the mechanical properties of structural materials in view of hydrogen embrittlement.

Neglecting the second part of equ. (1) the tritium capacity can be written

$$K_T = C_{Li} \cdot (G_{LB} + G_{LC}) \quad (6)$$

$G_{LB}$  being the lithium weight inside the blanket and  $G_{LC}$  that inside the external circuit.

### 5.1 Lithium Weight Inside the Blanket

The lithium weight inside the blanket depends exclusively on the reactor design, which includes the type of reactor, the blanket concept, and the mean wall loading. Fig. 2 shows the lithium weight  $G_{LB}$  as a function of the mean wall loading for a 5000 MWth Tokamak-type reactor with an aspect ratio of  $A = 4.0$  and a ratio of plasma and first wall diameter of  $y = 0.8$ . It is valid for a lithium temperature of about 500 C.

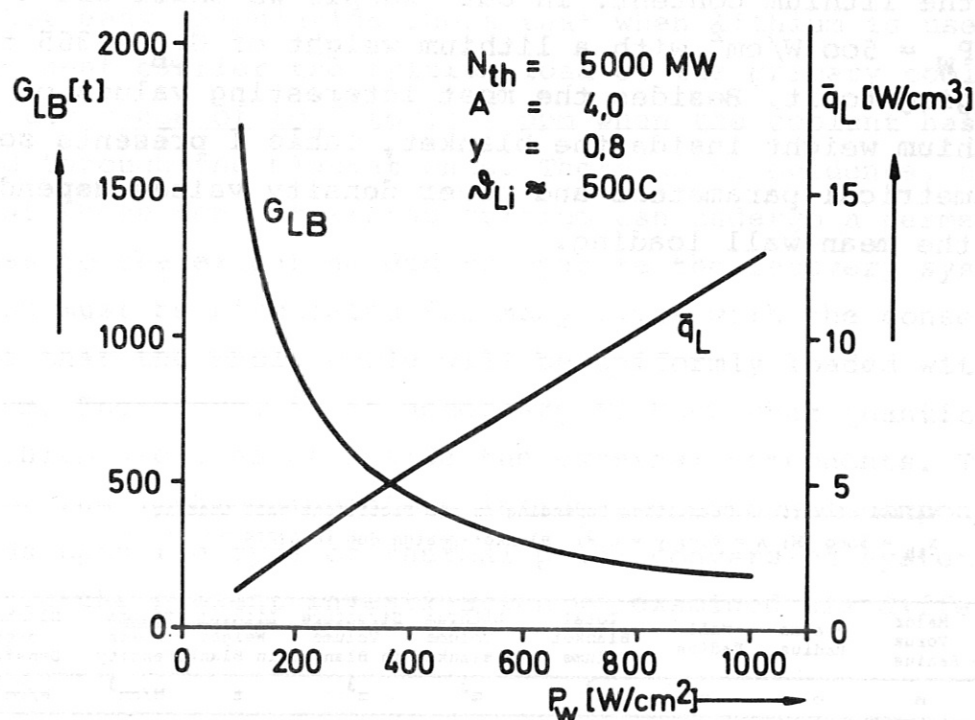


Fig.2: Lithium Weight Inside the Blanket  $G_{LB}$  and Mean Power Density Inside the Lithium Versus Wall Loading  $P_W$

The blanket system was assumed to be built up of three zones.

- Zone I: 60 cm thick, 94% lithium  
6% structural material
- zone II: 30 cm thick, 100% graphite
- zone III: 10 cm thick, 94% lithium  
6% structural material

This blanket could be an approximate mixture of the "Standard Model" laid down at the International Working Session on Fusion Reactor Technology [15] for use in neutronic calculations and of the engineering study performed at Culham [16].

From Fig. 2 it can be seen that for high wall loadings the lithium weight is in the range of that assumed by FRAAS [9,10]. The latest tendency of reducing the wall loading to values of about  $300 \text{ W/cm}^2$ , however, results in a remarkable increase



of the lithium content. In our example we shall use a value of  $P_W = 500 \text{ W/cm}^2$  with a lithium weight of  $G_{LB} = 365 \text{ t}$  belonging to it. Besides the most interesting values of the lithium weight inside the blanket, Table I presents some geometrical parameters and power density values dependent on the mean wall loading.

Table I: Values of Various Quantities Depending on the Fictitious Wall Loading  
 $N_{th} = 5000 \text{ MW}$ ;  $A = 4.0$ ;  $y = 0.8$ ; Blanket-Design due to STEINER

Wall Loading	Major Torus Radius	Plasma Radius	Wall Radius	Total Blanket Volume	Lithium Volume in Blanket	Structure Volume in Blanket	Lithium Weight in Blanket	Plasma Power Density	Blanket Power Density	Lithium Power Density
$\text{W/cm}^2$	m	m	m	$\text{m}^3$	$\text{m}^3$	$\text{m}^3$	t	$\text{W/cm}^3$	$\text{W/cm}^3$	$\text{W/cm}^3$
100	20.13	5.03	6.29	5400	3500	223	1700	0.39	0.93	1.43
200	14.24	5.56	4.45	2780	1790	114	870	1.10	1.80	2.79
300	11.62	2.91	3.63	1900	1220	78	590	2.02	2.64	4.11
400	10.07	2.52	3.15	1450	930	59	450	3.11	3.45	5.40
500	9.00	2.25	2.81	1180	750	48	370	4.37	4.25	6.67
600	8.22	2.05	2.57	1000	630	40	310	5.76	5.02	7.91
700	7.61	1.90	2.38	870	550	35	270	7.25	5.78	9.13
800	7.12	1.78	2.22	770	480	31	240	8.83	6.53	10.33
900	6.71	1.68	2.10	690	430	28	210	10.51	7.27	11.52
1000	6.37	1.59	1.99	630	390	25	190	12.36	7.99	12.69

## 5.2 Lithium Weight Inside the External Primary Circuit

When FRAAS made his estimate of the tritium inventory he obviously considered only the lithium weight inside the blanket. He did not take into account that huge quantities of lithium are also present in the external part of the primary circuit, these being preferentially concentrated inside the heat exchangers.

A rather easy calculation shows that when lithium is used as the heat carrier the tritium load of the primary coolant is in the range of  $10^{-3}$  to  $10^{-4}$  ppm when the coolant has passed through the blanket once. There is no evidence, however, that at these concentrations tritium can undergo a permeation process to the extent needed to operate the recovery system. Lithium must be circulated for many times with the consequence that the whole cycle will be uniformly loaded with tritium. Therefore, it is necessary to know what quantities of lithium are present inside the external components. The size of the lithium inventory inside the external components depends upon the type of thermal power conversion system used. In the present investigation we examined six different types:

#### 5.2.1 The System $\text{Li} \rightarrow \text{Li} - \text{H}_2\text{O}$

A scheme of this system is shown in Fig. 3. This power conversion system is equivalent to that proposed in the Na2 study of the Nuclear Research Centre

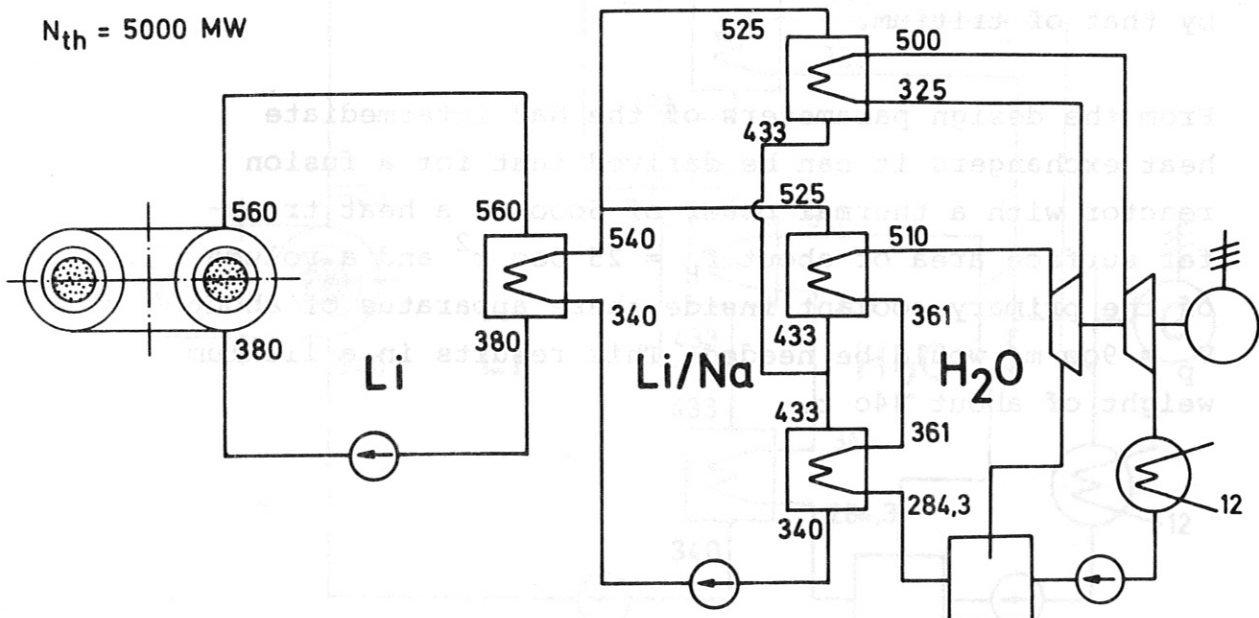


Fig. 3: Heat Flow Diagram of the  $\text{Li} \rightarrow \text{Li} - \text{H}_2\text{O}$  Power Conversion System



Karlsruhe [17], and has to be considered as a precursor of the SNR-300 to be built in the near future. In this scheme the fast breeder reactor has been replaced by a fusion reactor with a thermal power of 5000 MW and lithium has to be used as primary coolant instead of sodium.

The heat produced inside the blanket is carried by the lithium to an intermediate heat exchanger where it is transferred to a secondary liquid metal loop. The working fluid of this loop can be either lithium or sodium. It distributes the power to the components of the tertiary steam cycle.

Within the fast breeder reactor philosophy the intermediate sodium loop acts as a safety barrier in order to prevent activated sodium from circulating outside the safety vessel and to protect the reactor against consequences arising from a possible sodium-water reaction inside a steam generator in the event of leakage. The same philosophy could be applied to a fusion power plant if the sodium activity is replaced by that of tritium.

From the design parameters of the Na2 intermediate heat exchangers it can be derived that for a fusion reactor with a thermal power of 5000 MW a heat transfer surface area of about  $F_H = 25\,500\text{ m}^2$  and a volume of the primary coolant inside these apparatus of about  $V_p = 900\text{ m}^3$  would be needed. This results in a lithium weight of about 440 t.

### 5.2.2 The System $\text{Li} \rightarrow \text{H}_2\text{O}$

The second type of power conversion system is similar to the first one, but dispenses with the intermediate liquid metal loop. The heat exchangers of the steam cycle are directly heated by the primary coolant (see fig. 4). This version could be solution for the more distant future if the safety problems connected with the liquid metal- water-reaction can be solved by a corresponding layout of the steam generators. The fact of tritium activity and diffusion should not play as big a role as the sodium activity. In this case the specific data of the steam generators and superheaters presented in [17] can be a guide for equivalent figures. Extrapolating these data to the power level of the fusion reactor yields a heat transfer surface area of about  $F_H = 41\,000\text{ m}^2$  and a primary coolant volume of about  $V_P = 1060\text{ m}^3$ . If the same temperature level as in the former case is assumed the external lithium weight becomes about  $G_{LC} = 520\text{ t}$ .

$N_{th} = 5000\text{ MW}$

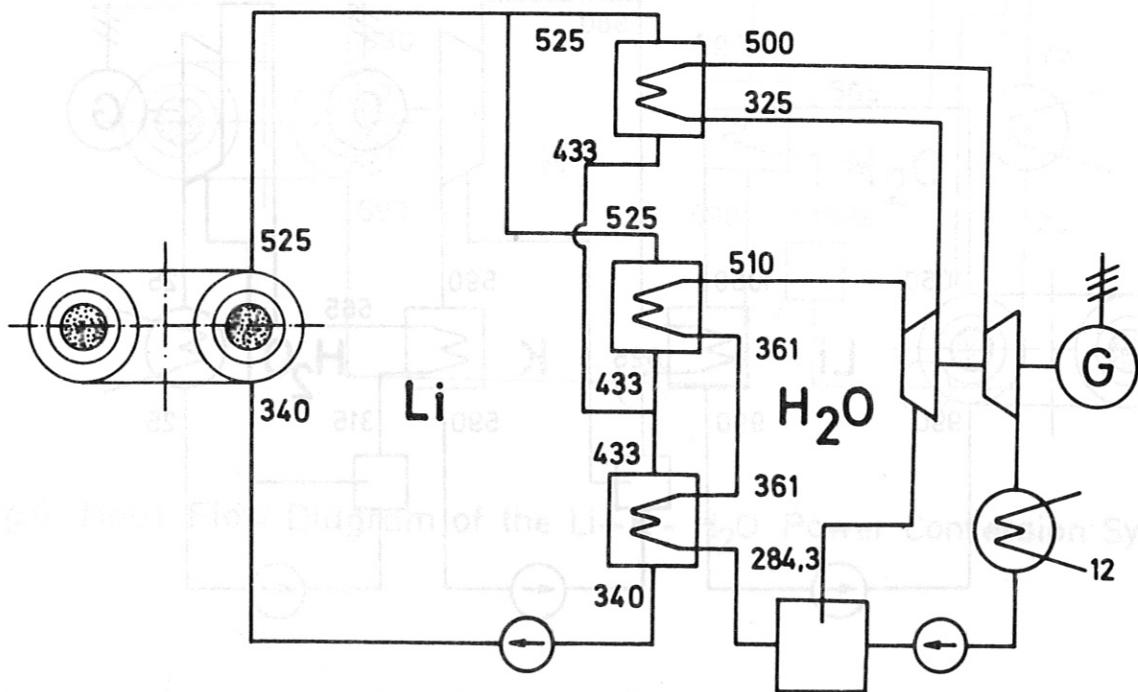


Fig.4: Heat Flow Diagram of the  $\text{Li} \rightarrow \text{H}_2\text{O}$  Power Conversion System

### 5.2.3 The System $\text{Li} \rightarrow \text{K} - \text{H}_2\text{O}$

Besides the low-temperature cycles described above, the high temperature binary potassium-steam cycle is of predominant interest because of its high thermal efficiency and hence low waste heat production. The scheme shown in fig. 5 is that proposed by FRAAS [18], being likewise the basis of his considerations on tritium recovery systems presented in [9, 10].

These references stipulate a surface area for the lithium-heated potassium boiler of  $F_H = 7000 \text{ m}^2$ . Assuming the same specific values for the ratio of primary coolant volume and heating surface area as in the case of a lithium-lithium intermediate heat exchanger results in a primary coolant volume of about  $V_p = 250 \text{ m}^3$  and, owing to the higher temperature level, a lithium weight of  $G_{\text{LC}} = 110 \text{ t}$ .

$N_{\text{th}} = 5000 \text{ MW}$

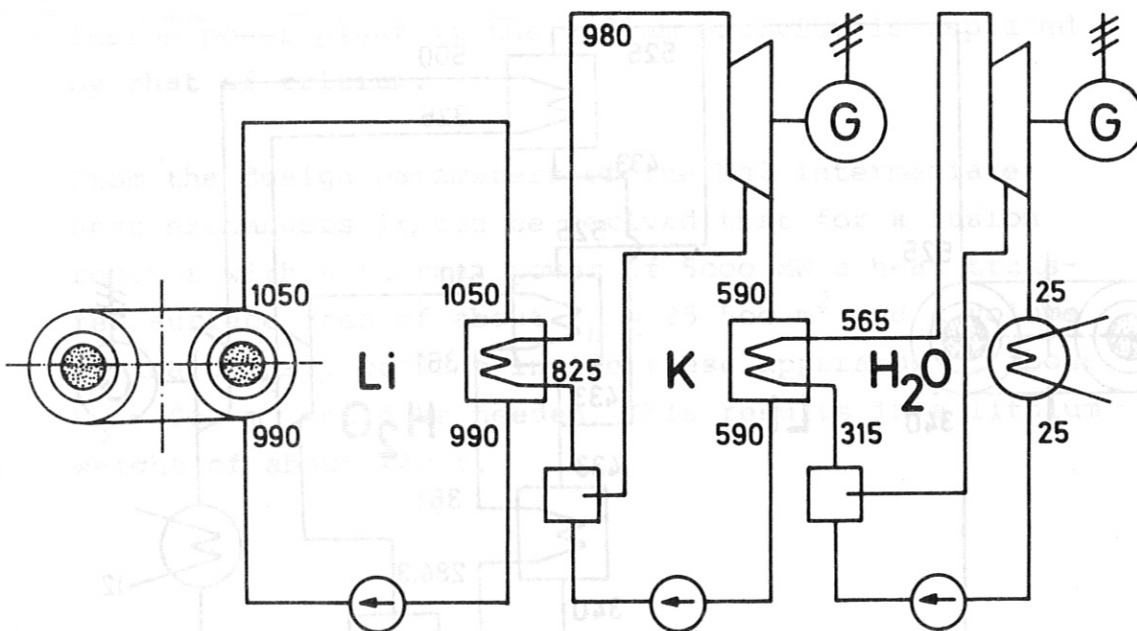


Fig.5: Heat Flow Diagram of the  $\text{Li} \rightarrow \text{K} - \text{H}_2\text{O}$  Power Conversion System



#### 5.2.4 The System $\text{Li} \leftarrow \text{K} - \text{H}_2\text{O}$

A binary potassium-steam thermal power cycle (fig. 6) also seems possible by using the blanket itself as the potassium boiler. Proposals of this kind were presented by WERNER [8, 19], MITCHELL [15] and FRAAS [30].

As far as the tritium capacity is concerned, this would be an ideal solution because no external lithium inventory is necessary. The heating surface of the internal potassium boiler could probably be the same as in the case of an external boiler.

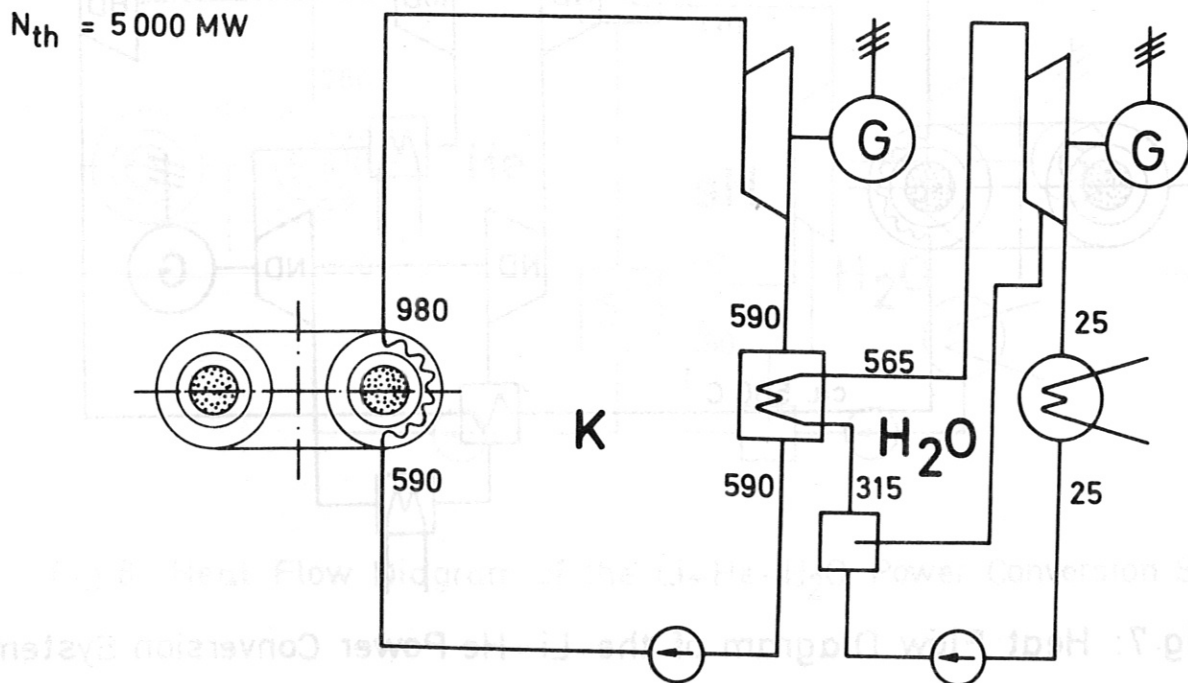
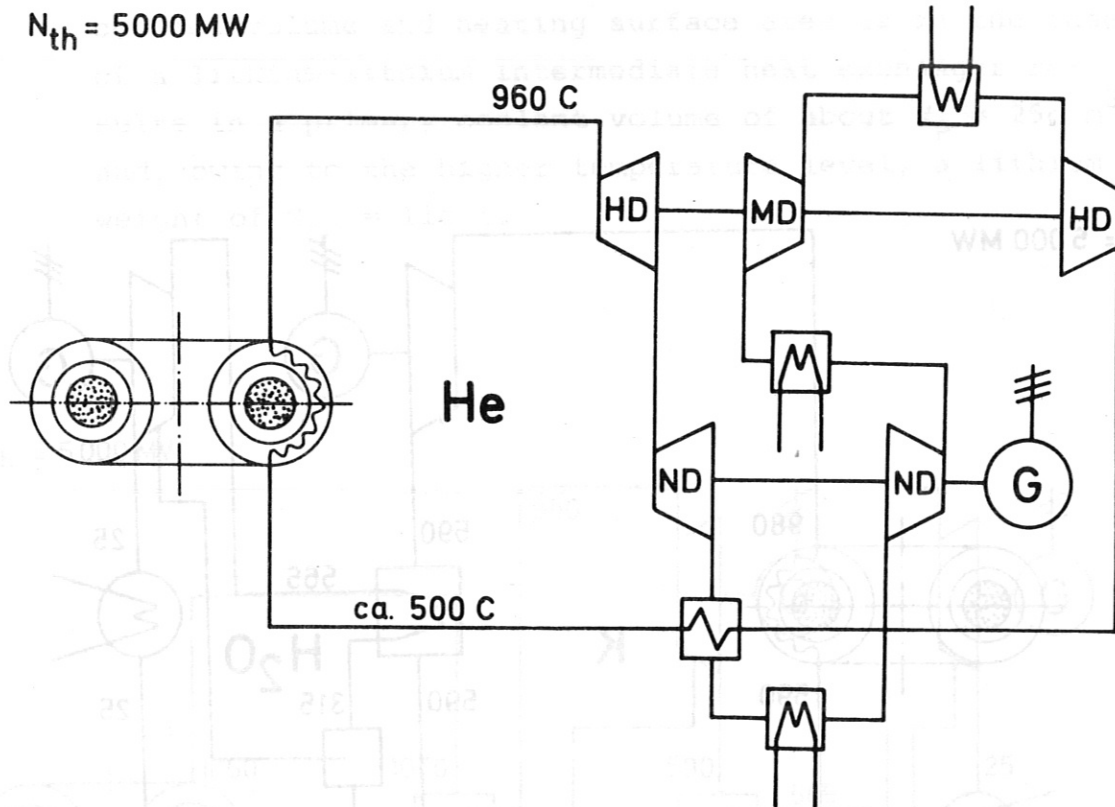


Fig.6: Heat Flow Diagram of the  $\text{Li} \leftarrow \text{K} - \text{H}_2\text{O}$  Power Conversion System

### 5.2.5 The System Li $\leftarrow$ He



**Fig.7: Heat Flow Diagram of the Li-He Power Conversion System**

### 5.2.6 The System $\text{Li} \leftarrow \text{He} - \text{H}_2\text{O}$

An intermediate step in the direction of the direct helium cycle might be the system used today in high-temperature-reactor technology [23]. Here helium is used only as the heat transfer agent from the heat source to a conventional steam power system (see fig.8). Projecting this application to a fusion power plant so far yields the same conclusions as in the case of the direct cycle.

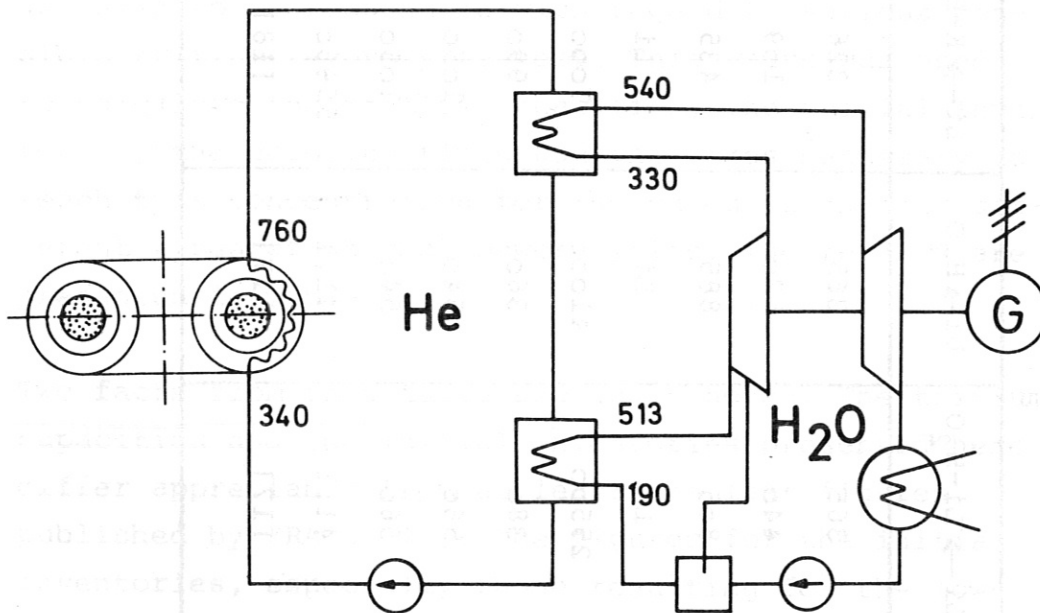


Fig.8: Heat Flow Diagram of the  $\text{Li} \leftarrow \text{He} - \text{H}_2\text{O}$  Power Conversion System

### 5.2.7 Summary

The data given for the six different power conversion systems are once more summarized in Table II together with some other characteristic parameters, e.g. the coolant inlet and outlet temperature  $\vartheta_{ci}$  and  $\vartheta_{co}$ , the maximum lithium temperature expected  $\vartheta_{Li \max}$  and the net efficiency  $\eta$  of the power plant.



Table II: Characteristic Data of Different Power Conversion Systems

$N_{th} = 5000 \text{ MW}$ ;  $A = 4.0$ ;  $y = 0.8$ ;  $P_w = 500 \text{ W/cm}^2$

Power Conversion System	$\text{Li} \rightarrow \text{Li-H}_2\text{O}$	$\text{Li} \rightarrow \text{H}_2\text{O}$	$\text{Li} \rightarrow \text{K-H}_2\text{O}$	$\text{Li} \leftarrow \text{K-H}_2\text{O}$	$\text{Li} \leftarrow \text{He}$	$\text{Li} \leftarrow \text{He-H}_2\text{O}$
Lithium Weight in Blanket [t]	365	365	326	313	306	320
External Lithium Weight [t]	440	520	109	-	-	-
Total Lithium Weight [t]	805	885	435	313	306	320
Coolant	Li	Li	Li	K	He	He
Heat Transfer Surface Area [ $\text{m}^2$ ]	25500	41000	7000	7000	1600	1600
Coolant Inlet Temperature [C]	380	360	990	990	518	340
Coolant Outlet Temperature [C]	560	540	1050	1050	950	760
Max. Lithium Temperature [C]	560	540	1050	1200	1300	1100
Net Efficiency [%]	41.1	41.1	55÷60	55÷60	46	39
Reference	[17]	[17]	[18]		[21]	[23]

From this table it can be concluded that the high-temperature power conversion systems would yield the lowest tritium inventories if the lithium content were the only quantity responsible for it. The low-temperature cycles would only be competitive if the tritium concentration could be kept to less than half of that in case of high-temperature cycles.

This section concludes with a numerical example. During the discussions about the Memorandum on technological problems of fusion reactors [24] a value of 60 ppm for the tritium concentration in lithium  $C_{Li}$  was said to be reasonable with regard to various possible tritium recovery systems. This value was used to calculate the tritium capacities, the initial inventories, the doubling times and the times necessary to reach this concentration for the cases of the six different types of thermal power cycles. The results are presented in Table III.

Two facts from this table should be noted. The tritium capacities and the initial inventories presented here differ appreciably from at least the last figures published by FRAAS [10]. The figures for the initial inventories, especially those resulting for the low-temperature power cycles, do indeed raise the question whether tritium is available in sufficient quantity to start a fusion reactor. There is not much evidence of marked reduction of the external lithium weight unless heat exchangers of much more compact design can be developed. As for the internal lithium weight, there is more likelihood of having a larger amount of lithium because the fictitious wall loading  $P_W$  will more probably be below  $500 \text{ W/cm}^2$  than above this value. Therefore, it should be more worthwhile to have a look at the optimum concentration.

Table III: Some Quantities Concerning Tritium Economy for a Tritium

Concentration Level of  $C_{Li} = 60$  ppm.

$N_{th} = 5000$  MW;  $P_W = 500$  W/cm<sup>2</sup>;  $B = 1.2$

Power Conversion System	Li→Li-H <sub>2</sub> O	Li→H <sub>2</sub> O	Li→K-H <sub>2</sub> O	Li→K-H <sub>2</sub> O	Li→He	Li→He-H <sub>2</sub> O
Total Lithium Weight [t]	805	885	435	313	306	320
Tritium Capacity [kg]	48.3	53.1	26.1	18.8	18.4	19.2
Tritium Initial Inventory [kg]	40.3	44.3	21.8	15.7	15.3	16.0
Time $t_1$ to Reach Equilibrium [d]	67	74	36	26	26	27
Doubling Time $t_2$ [d]	336	369	182	131	128	133

## 5.3 Tritium Concentration in Lithium

### 5.3.1 The Permeation Process

In order to calculate the tritium concentration in lithium, we refer to a number of proposals [9, 10, 11, 12] recommending removal of the tritium from the lithium by a diffusion process. The driving force of this process is the difference of the tritium partial pressures on both sides of the permeation wall. The equation governing this process is

$$\varphi = \frac{C}{X} \cdot (\sqrt{P_1} - \sqrt{P_2}) \cdot e^{-\frac{Q}{RT}} \cdot 10^{-5} \quad (7)$$

as presented by, for example, WEICHSELGARTNER [12].

In this equation  $\varphi$  means the volume flow rate of the permeating gas,  $X$  is the thickness of the wall,  $P_1$  and  $P_2$  are, respectively, the upstream and downstream partial pressures of the gas.  $C$  and  $Q$  are, respectively, the permeation constant and the activation energy of diffusion, which are both characteristic properties of the gas-metal system considered.  $R$  stands for the universal gas constant and  $T$  for the absolute temperature.

Fig. 9 shows the specific hydrogen permeation rates for various wall materials of interest in this investigation as dependent on the wall temperature. This picture is valid for the case where the wall thickness is set at 1 mm, and the partial pressures on the upstream and downstream sides are just 760 and 0 torr respectively. If this picture is applied to tritium instead of hydrogen, the figures have to be divided by  $\sqrt{3}$ , with due allowance for the different atomic weight of the two substances [9, 10]. Fig. 9 is an extract of the equivalent presentations by WEBB [25] and FRAAS [9].



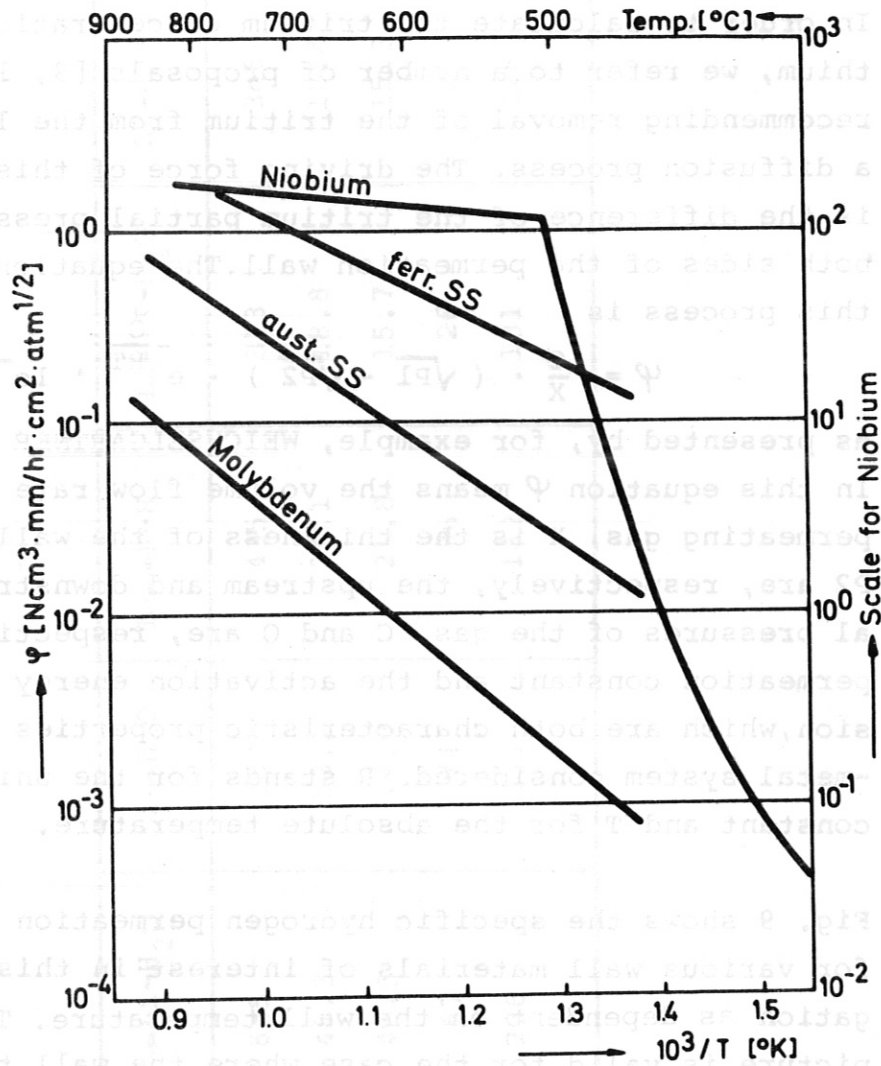


Fig.9: Specific Hydrogen Permeation Rates [25]

### 5.3.2 The Tritium Partial Pressure in the System Li - T

In order to make use of equ. (7) in calculating the permeation rate, it is necessary to know the tritium partial pressures on each side of the permeation wall.

As far as the downstream side is concerned, it is assumed in the following investigation that the value of  $P_2$  is  $P_2 = 0$ . This is indeed a simplification of the problem. On the other hand, it makes the diffusion problem independent of the kind of agent on the downstream side which takes up the permeated tritium and will be sufficient for the purpose of comparison. Furthermore, this simplification will yield results which can be interpreted as minimum values of, for example, the size of the permeation wall area needed.

At the primary side of the diffusion path the tritium partial pressure depends upon the properties of the lithium-tritium system. Obviously, the partial pressure  $P_1$  is a function of the temperature  $\vartheta$  and the tritium concentration  $C_{Li}$ .

Fig. 10 shows this dependence, which can be expressed by the equation

$$P_1 = 2.81 \cdot C_{Li}^2 \cdot 10^{-\frac{8540}{\vartheta+273}} \quad (8)$$

This picture was obtained by using the dependence on concentration given by FRAAS [9]. The influence of the temperature was gained by extrapolating the data presented by MESSER [26].

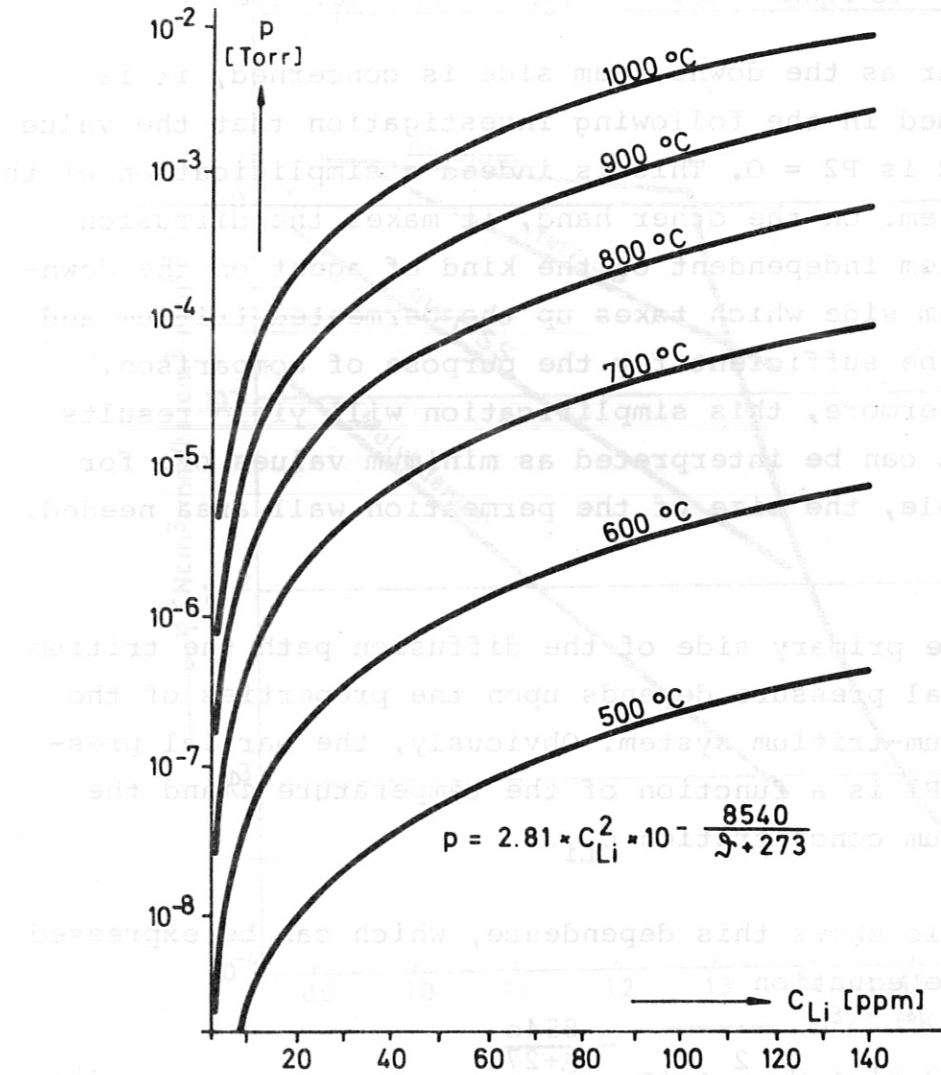


Fig. 10: Tritium Partial Pressure in System Li - LiT

If  $P_1$  is plotted versus  $1/T$ , a constant slope governed by the size of the reaction enthalpy becomes evident. This is what van't Hoff's law states. By applying this relationship and assuming that the reaction enthalpy does not vary with temperature, it is possible to extend the range of concentration down to values representative of fusion reactor parameters. The results gained by this procedure are in rather good agreement with the publication of VELECKIS and VAN DEVENTER [27].

### 5.3.3 Partial Pressure and Permeation Rate

Equation (7) indicates that the permeation rate varies as the square root of the partial pressure. This classical Sievert's law, which is also valid for the dependence of the solubility, has been proved by a lot of diffusion experiments performed in a range of partial pressures exceeding  $P_1 = 10$  Torr. Latest experiments performed by, for example, WEBB [25] lead to the conclusion, however, that beneath this value there exists a linear dependence upon  $P_1$ . An exact determination of the position of this transition region is not yet available. In accordance with FRAAS [10] we think that this transition value is dependent on the gas-metal system investigated.

Since there is a lack of data sufficiently confirmed by experimental work, this investigation makes use of the assumption that at partial pressures exceeding  $P_1 = 10$  Torr Sievert's law will be applicable, whereas at lower values of  $P_1$  a linear dependence starting from the value corresponding to 10 Torr will be valid.



#### 5.3.4 Summary

Using equ. (8) of section 5.3.2, it is possible to calculate the tritium partial pressure  $P_1$  in the system Li - T as a function of temperature  $\vartheta$  and tritium concentration  $C_{Li}$ . This result can be used to evaluate the specific permeation rate  $\varphi$  taking into account the different dependences on the size of the pressure indicated in section 5.3.3. As a consequence a relationship between the tritium concentration  $C_{Li}$  and the size of the permeation wall area can be derived.

When this relationship is applied to the tritium removal system of the fusion reactor, one boundary condition must be satisfied in every case: In removing the tritium by means of a diffusion process the permeation rate  $\varphi$  must equal the tritium production rate inside the blanket.

$$\varphi = P_T \quad (9)$$

In the case of the 5000 MWth fusion reactor this quantity has to be  $P_T = 30 \text{ g/h}$ . This value results from the assumption that the total thermal power of the reactor arises from an energy release of 22.4 MeV per fusion reaction, which includes exothermic reactions inside the blanket. The second assumption used is a tritium breeding ratio of  $B = 1.2$ .

## 6. Results and Discussions

### 6.1 Investigation Procedure

A number of calculations were made on the basis of the foregoing sections. In every case the relationship between the tritium concentration in lithium  $C_{Li}$  and the permeation wall area  $F_p$  was evaluated with allowance for the fact that the permeation rate has to equal the tritium production rate inside the blanket.

A first series of calculations was concerned with the proposals which allow the tritium to enter the secondary circuit of the power conversion system. Special effort was devoted to the problems of material choice. The conclusions drawn from these results were taken into account in a second series of calculations. Here the characteristics of a tritium removal system in which the admission of tritium to the secondary circuit should be prevented was investigated.

In all cases the initial inventory  $I_0$ , the time  $t_1$  necessary to reach the equilibrium concentration, and the doubling time  $t_2$  were evaluated as the quantities representative of the tritium economy. In order to get a feeling for tritium safety problems due to diffusion processes, two loss rates  $L_{Te}$  and  $L_{Ti}$  were evaluated.  $L_{Te}$  characterizes the "external tritium loss rate", which is equivalent to the tritium loss from the primary into the secondary cooling circuit. The "internal tritium loss rate"  $L_{Ti}$  is a more or less fictitious quantity the importance of which should be seen not as much in the absolute values presented but in the relative ones by comparing them with the equivalent values of the different cooling systems. It was assumed that each of the lithium containing zones I and III of the blanket is radially limited by a 5 mm thick structural wall. All the

tritium escaping from the two zones by diffusion through these walls was further assumed to be able to reach the reactor containment.

The two loss rates were used to calculate two further quantities  $g_e$  and  $g_i$  which stand for the amount of tritium accumulating within a period of 20 years of continuous reactor operation, neglecting radioactive decay.

Finally, a cost factor  $K$ , whose meaning will be explained in section 6.3.2, is presented.

## 6.2 Tritium Diffusion into Secondary Circuit

### 6.2.1 General Remarks

If tritium diffusion into the secondary circuit of the power conversion system is allowed, the heat transfer surface  $F_H$  acts as the tritium permeation wall. The surface area and wall thickness are fixed by the thermohydraulic layout of the blanket cooling system and by the mechanical properties of the materials used. Consequently, the tritium concentration and hence the tritium capacity are ruled by heat transfer problems and there is only limited space to meet the specific tritium removal requirements.

The method of tritium extraction from the secondary circuit is suitable only in the case of high-temperature thermal power cycles, which means the use of either potassium or helium as the heat transfer agent in the secondary loop. Applying this method to, for instance, the  $\text{Li} \rightarrow \text{H}_2\text{O}$  system would result in tritium contamination of the secondary steam cycle. To remove the tritium from the water would require an isotope separation process, which obviously is not the most economical solution of the tritium recovery problem. If an intermediate liquid metal loop were used, there is no

convincing argument for withdrawing the tritium from the secondary circuit because it could be removed from the primary one in the same way. Therefore, in investigating this scheme the power conversion systems defined in 5.2.1 and 5.2.2 have been excluded.

#### 6.2.2 Results for the Basic Model

Like the heat transfer areas  $F_H$ , the wall thicknesses are also different for the different power conversion systems. With niobium as the wall material, a thickness of 1 mm seems to be sufficient according to FRAAS [9,10,18] in the case of a potassium secondary cycle. In the case of helium cooling a thickness of 2 mm is necessary [21] to withstand the high coolant pressure.

Table IV summarizes the most essential results for the case where only the heat transfer and permeation wall surfaces are made of niobium, the remainder of the blanket structure, however, being made of molybdenum.

It can be seen that the temperature level is of predominant influence upon the tritium concentration. Especially the lithium temperature is significant since it is decisive for the size of the tritium partial pressure. In the case of indirect cooling ( $\text{Li} \leftarrow \text{K} - \text{H}_2\text{O}$ ,  $\text{Li} \leftarrow \text{He}$ ) where the lithium temperature is fairly above the coolant temperature this advantage with regard to the tritium concentration becomes obvious. This advantage is increased by the lower lithium inventory, which leads to extremely low tritium inventories.

It should be noted that, except for the last column of Table IV, all concentrations  $C_{\text{Li}}$  evaluated are lower than 60 ppm as was previously assumed. The size of the



Table IV: Results for Tritium Removal from Secondary Circuit

Power Conversion System	Li→K-H <sub>2</sub> O	Li←K-H <sub>2</sub> O	Li←He	Li←He-H <sub>2</sub> O
Total Lithium Weight [t]	435	313	306	320
Blanket Structure Material	Mo	Mo	Mo	Mo
Heat Transfer Surface Material	Nb	Nb	Nb	Nb
Heat Transfer Surface Area [m <sup>2</sup> ]	7000	7000	1600	1600
Thickness of Heat Transfer Walls [mm]	1	1	2	2
Tritium Concentration in Lithium [ppm]	41.5	40.5	31.4	85.0
Tritium Capacity [kg]	18.3	13.2	9.4	27.2
Tritium Initial Inventory [kg]	15.3	11.0	7.8	22.7
Time t <sub>1</sub> to Reach Equilibrium [d]	26	18	13	38
Doubling Time t <sub>2</sub> [d]	128	92	65	189
External Loss Rate [g/h]	30	30	30	30
Internal Loss Rate [g/h]	9 · 10 <sup>-3</sup>	9 · 10 <sup>-3</sup>	5.5 · 10 <sup>-2</sup>	1.5 · 10 <sup>-2</sup>
20-Years Accumulated Internal Losses [kg]	1.6	1.5	9.5	2.6
Costs [10 <sup>6</sup> DM]	216	173	129	249

initial inventory  $I_0$  is well within the range of preliminary estimates. The doubling times  $t_2$  based upon a breeding ratio of  $B = 1.2$  are also within a range appropriate to an accelerating fusion power installation. The external loss rates are, by definition, identical with the tritium production rate inside the blanket. In all four cases the internal loss rates yield values within the same order of magnitude. Only the direct helium cycle is somewhat more disadvantageous by about a factor of 5.

### 6.2.3 Influence of Material Choice

A second series of calculations was run in order to provide information about the influence of material choice based on the model in which the tritium is removed from the secondary circuit. Also investigated were two cases involving the only use of either niobium or molybdenum for constructing both the heat transfer surface and the remaining blanket structure. In the case where molybdenum was assumed for the heat transfer surface the wall thickness was reduced to 0.6 mm according to FÖRSTER [21]. The most essential results are reported in Table V.

If niobium were used for constructing all the structural components inside the blanket, there would be an advantage with regard to the tritium concentration and inventory. This advantage is appreciable in the case of helium cooling, while it is nearly insignificant in the case of potassium secondary cycles. The internal loss rate, however, can be expected to rise some orders of magnitude. As far as the helium cooling system is concerned this loss rate could even exceed the permeation rate into the helium cycle because of the

Table V: Results for Tritium Removal from Secondary Circuit  
Effect of Material Choice

Power Conversion System	Blanket Structure Material	Heat Transfer Surface Material	Tritium Concentr. ppm	Initial Inventory kg	Internal Loss Rate g/h	External Loss Rate g/h	Effective Breeding
Li → K-H <sub>2</sub> O	Nb	Nb	39.0	17.0	4.0	26	1.04
	Mo	Nb	41.5	18.0	0.009	30	1.20
	Mo	Mo	990	430	5.15	24.85	0.99
Li ← K-H <sub>2</sub> O	Nb	Nb	38	12.5	3.8	26.2	1.05
	Mo	Nb	40.5	13.2	0.009	30	1.20
	Mo	Mo	835	275	3.7	26.3	1.05
Li ← He	Nb	Nb	21.1	6.5	16.5	13.5	0.54
	Mo	Nb	31.4	9.4	0.054	30	1.20
	Mo	Mo	371	114	8	22	0.88
Li ← He-H <sub>2</sub> O	Nb	Nb	57.0	18.5	16.5	13.5	0.54
	Mo	Nb	85.0	27.2	0.015	30	1.20
	Mo	Mo	1940	620	8	22	0.88

extremely low heat transfer surface requirements. If these internal losses had to be considered as non-recoverable, this fact would strongly affect the tritium breeding ratio of the blanket. The last column of Table V lists the "effective breeding ratio" obtained with respect to these losses. Particularly in the case of helium cooled blankets the figures presented demonstrate the extent of this problem.

Quite a similar situation arises if molybdenum is exclusively used. Because of the extremely low specific permeation rates additionally huge concentrations and initial inventories would be needed.

#### 6.2.4 Discussion

One conclusion to be noted from this chapter is that the choice of the blanket structure material should also be made with the tritium diffusion problem in mind. If tritium is to be drawn off from the secondary circuit, it is most probable that the material for the heat transfer surface will be other than that which will be used for the remainder of the blanket structure. Otherwise the tritium losses from the blanket make a second tritium recovery system necessary unless a significant decrease in the effective tritium breeding ratio can be tolerated.

As mentioned above, the figures concerning the internal tritium losses should, however, not be taken as absolute amounts. The assumptions involved in this consideration are not sufficiently well founded. There is a fair chance of a marked reduction of these losses owing to the certainly non-zero tritium partial pressure beyond the 5-mm walls of the blanket. On the other hand,



there is also a chance of an increase of these losses owing to the much more increased permeation surface in the case of applying a modular blanket. Regardless of these incertainties, it should be possible in every case to remove these losses in a controlled manner. This is especially valid if the vacuum sealing of the reactor vessel is positioned in the shielding region. This seems to be advantageous already with regard to temperature and radiation load.

Another point should be noted with respect to the figures evaluated for the tritium concentration and inventory. As in the case of the internal losses, zero tritium partial pressure was also assumed on the secondary side of the diffusion path in the case of permeation into the secondary circuit. If this is not so - and it surely will be - the values of the tritium concentration and all other quantities depending on it will rise. If the size of the heat transfer surface is absolutely fixed by thermohydraulic considerations, there will be no possibility of reducing these figures by an oversizing of the surface area. This seems to be a weak point of this concept.

### 6.3 Tritium Extraction in Bypass

In order to keep the tritium concentration and the initial inventory low, an alternate possibility was investigated: the entire tritium removal equipment was arranged in a bypass to the primary lithium loop. Obviously, this is a necessity for power conversion systems using a heat transfer agent other than lithium inside the blanket if extraction from the secondary cooling cycle is to be avoided. The two main advantages of this solution are the following:

- There is no spreading of tritium to other components of the power plant except the primary circuit of the reactor. In view of the high leak tightness of liquid metal systems already operating this is an essential safety factor with regard to tritium contamination.
- There is great flexibility in the layout of the tritium removal equipment not limited by thermohydraulic requirements.

#### 6.3.1 Tritium Removal System

The arrangement of such a bypass tritium removal system is shown in fig. 11. The lithium loaded with tritium is drawn off from the primary loop or, in the case of applying an indirect cooling method, directly from the blanket and fed to a regenerative heat exchanger. Inside this apparatus and in a subsequent auxiliary heater it is heated to a temperature  $\vartheta_s$  appropriate to an effective diffusion process. This process takes place in a special separator which is designed like a heat exchanger.

The lithium enters a vessel in which a lot of thin-walled tubes made of niobium or any other material characterized by a very high tritium permeability are installed. Inside these tubes the tritium permeating through the tube walls is chemically bonded by the solid phase of a fluidized bed, e.g. titanium powder. Helium can be used as carrier gas. The solid particles are separated from the gas in a cyclone. Chemically reconditioned and unloaded of tritium, the powder is refed to the carrier gas.

With a decreased concentration level the hot lithium then again enters the regenerative heat exchanger to transfer its heat to the "cold" stream before being fed back to the main string of the primary cooling loop.

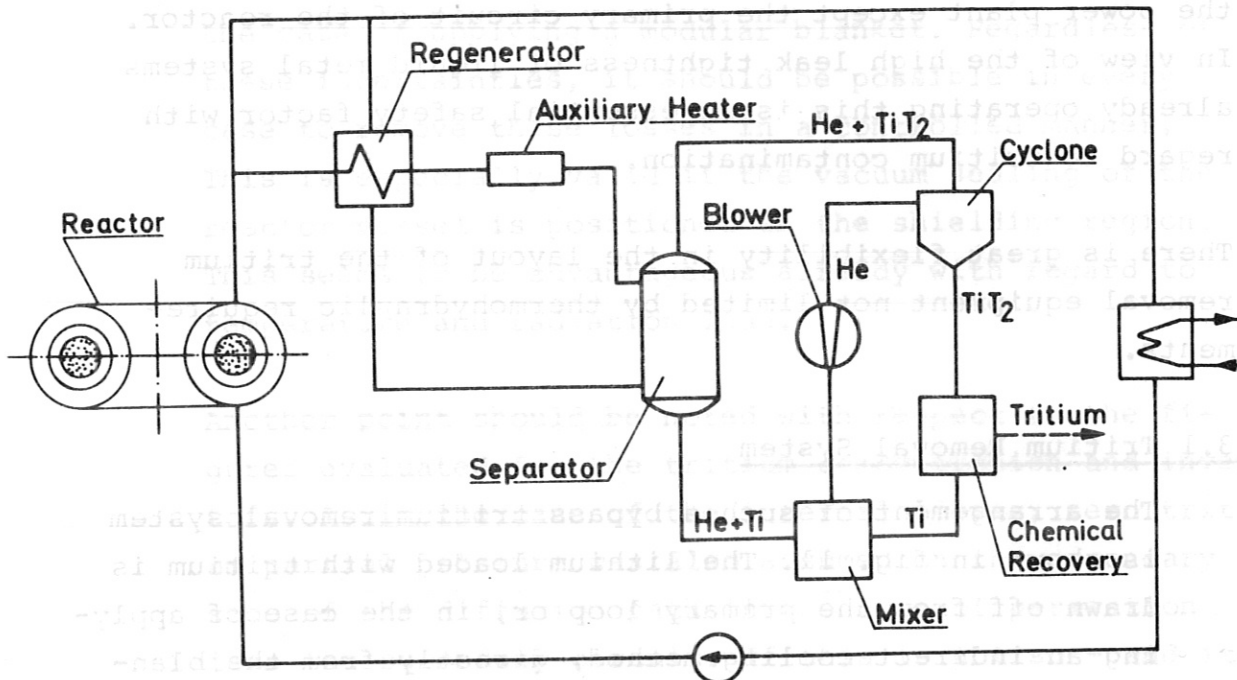


Fig. 11: Bypass Tritium Removal System

As is easy to see, this tritium removal system offers two essential advantages as regards flexibility in the layout. There is no restriction in the choice of permeation area. Additionally, the process temperature  $\vartheta_s$  provides a further free parameter.

Compared with these advantages the disadvantages should be of less importance. The entire bypass, especially the separator device, necessitates an increase of the total lithium weight inside the primary circuit in connection with an increased tritium inventory. The auxiliary heating system requires additional power to provide the temperature difference within the regenerative heat exchanger and to compensate heat losses within the

whole bypass equipment. The most significant heat losses have to be expected inside the separator. Therefore, it would be advisable to keep the fluidized bed at a temperature near the process temperature  $\vartheta_s$ . The pressure of the carrier gas should be governed by the lithium pressure level inside the separator to permit the application of thin-walled tubes, thus reducing the necessary permeation area and hence the lithium and tritium inventories.

As far as the auxiliary heater is concerned, the power requirements are modest. In order to achieve efficient tritium diffusion, the concentration gradient along the separator tubes should not be too large. A preliminary estimate shows that the mass flow rate of lithium in the bypass should not fall below 0.1 - 1% of the total mass flow rate in the primary loop. Since the temperature difference inside the regenerative heat exchanger will surely be lower than the lithium temperature rise inside the blanket, the heating power in this case will be lower than 0.1% of the reactor thermal power.

#### 6.3.2 Optimization Procedure

The higher flexibility of the proposed system means that a new criterion for the optimum layout has to be introduced. Such a criterion can be supplied by cost estimates. Increasing the permeation surface area entails increasing material costs and additionally increasing lithium and hence tritium inventories owing to the amounts present in the bypass equipment. On the other hand, an increase in the permeation area causes a decrease in the concentration necessary to force the diffusion process and hence a decrease in the initial inventory and the costs for it.

The costs K can be roughly expressed by the equation

$$K = \alpha + \frac{\beta}{C_{Li}^2} + \frac{\gamma}{C_{Li}} + \delta C_{Li} \quad (9)$$

with the coefficients

$$\alpha = \sigma_{Li} [G_{LB} + G_{LC}] \quad (10)$$

$$\beta = 2 \cdot 10^6 \frac{E_R \cdot s}{f(\varphi_s)} [s \cdot \sigma_{Nb} \cdot \rho_{Nb} + v' \cdot \sigma_{Li} \cdot \rho_{Li}] \quad (11)$$

$$\gamma = \frac{2 \sigma_T V_T \cdot s \cdot v' \cdot \rho_{Li}}{f(\varphi_s)} \quad (12)$$

$$\delta = \frac{10^{-6} \cdot \sigma_T [G_{LB} + G_{LC}] \cdot V_T}{E_T} \quad (13)$$

In these equations  $G_{LB}$  and  $G_{LC}$  denote the lithium weights inside the blanket and the external part of the primary circuit respectively,  $V_T$  is the tritium consumption rate of the reactor,  $E_T$  the tritium extraction rate in the separator, which equals the tritium production rate  $P_T$  if tritium losses are neglected.  $s$  is the thickness of the separator tubes,  $v'$  the specific lithium volume of the separator related to the unit surface area of the permeation wall.  $\rho_{Nb}$  and  $\rho_{Li}$  are the densities of niobium and lithium respectively:  $\sigma_{Nb}$ ,  $\sigma_{Li}$ , and  $\sigma_T$  are the specific costs of niobium, lithium, and tritium per unit weight.

The four terms of equ. (9) represent the costs for

1. the lithium inventory in the main string of the primary loop,
2. the structure material of the separator and the additional lithium inside the bypass,
3. the additional tritium inventory, and
4. the basic tritium inventory.



This estimation is, indeed, not complete; it is the intention of this paper only to recommend a model to perform an optimum layout of the tritium removal device. This model is open to further refinement.

If the cost coefficients  $\alpha$ ,  $\beta$ ,  $\gamma$ , and  $\delta$  are kept constant, it is possible to find a cost minimum by differentiating equ. (9) with respect to  $C_{Li}$ :

$$\frac{\partial K}{\partial C_{Li}} = 0 \quad (14)$$

The optimum concentration  $C_{opt}$  thereby results from the equation

$$C_{opt}^3 = \frac{2\beta}{\delta} + \frac{\gamma}{\delta} \cdot C_{opt} \quad (15)$$

### 6.3.3 Assumptions for Model Calculations

In evaluating equ. (15) for the different power conversion systems the following assumptions were made:

- a) The densities of lithium and niobium were taken to be  $\rho_{Li} = 0.5 \text{ g/cm}^3$  and  $\rho_{Nb} = 8.55 \text{ g/cm}^3$ .
- b) Three quantities depend upon the reactor design.  
For a total thermal power of 5000 MW, a fictitious wall loading of  $500 \text{ W/cm}^2$ , and a tritium breeding ratio in the blanket of 1.2, the corresponding values can be derived for the
 

tritium consumption rate	$V_T = 25 \text{ g/h}$
tritium extraction rate	$E_T = 30 \text{ g/h}$
lithium inventory in the blanket	$G_{LB} = 3.65 \cdot 10^8 \text{ g}$

To achieve these values, a zero tritium loss rate and a blanket construction as defined in section 5.1 were assumed.

- c) Two quantities depending on the separator layout were kept fixed:

$$\begin{aligned} \text{thickness of separator tubes} \quad s &= 0.1 \text{ cm} \\ \text{specific lithium volume} \quad v' &= 3.6 \text{ cm}^3/\text{cm}^2 \end{aligned}$$

The thickness of the separator tubes was chosen arbitrarily, whereas the specific lithium volume was derived from the equivalent data for the intermediate heat exchanger of the liquid metal fast breeder reactor study Na-2 [17].

- d) The specific material costs for tritium, niobium, and lithium are the most uncertain quantities in this investigation. The production costs for tritium and lithium have been presented by WERNER [8]. Although these figures represent the cost situation of the year 1969, they have been used here in the awareness that this calculation is only intended as an example. The costs for niobium were quoted by relevant industrial companies. It must be mentioned, however, that the figure presented here relates to the pure material costs for niobium tube material of special dimensions and no further production processes were taken into account. The following figures were used:

$$\begin{aligned} \text{Tritium} \quad \sigma_T &= 8.10^3 \text{ DM/g} \\ \text{Lithium} \quad \sigma_{Li} &= 0.03 \text{ DM/g} \\ \text{Niobium} \quad \sigma_{Nb} &= 0.6 \text{ DM/g} \end{aligned}$$

Beside these fixed quantities two further quantities are involved in the following calculations. The first is the lithium inventory inside the external part of the primary loop,  $G_{LC}$ . The corresponding values were taken from Table II.

The second quantity is the process temperature  $\vartheta_s$ , which enters the calculation in implicate form:

$$f(\vartheta_s) = \Phi(\vartheta_s) \cdot 10^{-\frac{8540}{\vartheta_s + 273}} \quad (16)$$

This function contains the temperature dependence of the specific permeation rate  $\Phi(\vartheta_s)$  according to fig. 9 and the temperature dependence of the tritium partial pressure for a fixed concentration according to fig. 10.

#### 6.3.4 Results and Discussions

The results gained by evaluating equation (15) for the different types of thermal power conversion systems are presented in the following. From the optimum concentration resulting from this evaluation some further interesting quantities are derived.

Four process temperatures  $\vartheta_s$  were considered: 1000, 1100, 1200, and 1300 C. It will certainly depend on the mean lithium temperature inside the blanket whether the choice of one or another temperature  $\vartheta_s$  will be realistic. The figures are given for comparison only.

Also listed for comparison are the equivalent data already presented in section 6.2.2 (Table IV). They are given in the first columns of the following tables. The advantages and disadvantages of the bypass removal system are thereby obvious.

#### Tritium Concentration in Lithium

The first step of the calculation is to evaluate equ. (15) in order to gain the tritium concentration corresponding to the cost minimum. The results are summarized in Table VI. It is obvious that with increasing process

Table VI:  
Results for Tritium Removal from Bypass  
Tritium Concentration in Lithium [ppm]

	-	1000	1100	1200	1300
Li → Li-H <sub>2</sub> O		35.3	22.9	15.8	
Li → H <sub>2</sub> O		33.9	22.1	15.3	
Li → K-H <sub>2</sub> O	41.5	39.6	28.2	19.5	
Li ← K-H <sub>2</sub> O	40.5	44.0	31.3	21.6	
Li ← He	31.4	48.3	31.3	21.6	15.6
Li ← He-H <sub>2</sub> O	85.0		31.3	21.6	15.6

**Table VII:** Results for Tritium Removal from Bypass  
Total Lithium Inventory [t]

	-	1000	1100	1200	1300
Li → Li-H <sub>2</sub> O		1020	957	918	
Li → H <sub>2</sub> O		1110	1043	1000	
Li → K-H <sub>2</sub> O	435	552	525	501	
Li ← K-H <sub>2</sub> O	326	421	399	380	
Li ← He	306	401	375	357	345
Li ← He-H <sub>2</sub> O	320		392	373	361





**Table IX:** Results for Tritium Removal from Bypass  
Doubling Time [d]

	-	1000	1100	1200	1300
$\text{Li} \rightarrow \text{Li-H}_2\text{O}$		250	150	100	
$\text{Li} \rightarrow \text{H}_2\text{O}$		260	160	106	
$\text{Li} \rightarrow \text{K-H}_2\text{O}$	128	150	105	70	
$\text{Li} \leftarrow \text{K-H}_2\text{O}$	92	130	85	55	
$\text{Li} \leftarrow \text{He}$	65	135	80	55	38
$\text{Li} \leftarrow \text{He-H}_2\text{O}$	189		85	55	40

Table X: Results for Tritium Removal from Bypass  
Permeation Area Requirements [m<sup>2</sup>]

Li → Li-H <sub>2</sub> O			1000	1100	1200	1300
Li → H <sub>2</sub> O			12250	8700	6500	
Li → K-H <sub>2</sub> O	7000		13180	9380	6930	
Li ← K-H <sub>2</sub> O	7000		7500	5750	4250	
Li ← He	1600		6060	4690	3470	
Li ← He-H <sub>2</sub> O	1600		6490	4690	3470	2650
						2650

Table XI: Results for Tritium Removal from Bypass  
Costs [ $10^6$ ]DM

	-	1000	1100	1200	1300
Li $\rightarrow$ Li-H <sub>2</sub> O		388	274	213	
Li $\rightarrow$ H <sub>2</sub> O		377	259	194	
Li $\rightarrow$ K-H <sub>2</sub> O	216	264	207	166	
Li $\leftarrow$ K-H <sub>2</sub> O	173	230	183	147	
Li $\leftarrow$ He	129	221	161	127	107
Li $\leftarrow$ He-H <sub>2</sub> O	249		164	129	108

Table XII: Results for Tritium Removal from Bypass  
Risk Potential Due to Internal Losses

	-	1000	1100	1200	1300
Li → Li-H <sub>2</sub> O		4.3	1.8	0.85	
Li → H <sub>2</sub> O		3.95	1.65	0.8	
Li → K-H <sub>2</sub> O	8 · 10 <sup>6</sup>	7 · 10 <sup>6</sup>	4 · 10 <sup>6</sup>	2 · 10 <sup>6</sup>	
Li → K-H <sub>2</sub> O	8 · 10 <sup>6</sup>	9 · 10 <sup>6</sup>	5 · 10 <sup>6</sup>	2 · 10 <sup>6</sup>	
Li → He	5 · 10 <sup>7</sup>	1 · 10 <sup>8</sup>	5 · 10 <sup>7</sup>	2 · 10 <sup>7</sup>	1 · 10 <sup>7</sup>
Li → He-H <sub>2</sub> O	1 · 10 <sup>7</sup>		2 · 10 <sup>6</sup>	9 · 10 <sup>5</sup>	5 · 10 <sup>5</sup>

Core 1000

Results for Tritium Removal from Bypass

Table XI:



Table XIII:  
Results for Tritium Removal from Bypass  
Tritium Amount Accumulated in Secondary  
Circuit after 20 Years of Reactor Operation [g]

	-	1000	1100	1200	1300
$\text{Li} \rightarrow \text{Li-H}_2\text{O}$		$9 \cdot 10^{-3}$	$4 \cdot 10^{-3}$	$2 \cdot 10^{-3}$	
$\text{Li} \rightarrow \text{H}_2\text{O}$		$3 \cdot 10^{-3}$	$2 \cdot 10^{-3}$	$7 \cdot 10^{-4}$	
$\text{Li} \rightarrow \text{K-H}_2\text{O}$		$7 \cdot 10^3$	$4 \cdot 10^3$	$2 \cdot 10^3$	
$\text{Li} \leftarrow \text{K-H}_2\text{O}$		$1 \cdot 10^4$	$6 \cdot 10^3$	$3 \cdot 10^3$	
$\text{Li} \leftarrow \text{He}$		$7 \cdot 10^4$	$3 \cdot 10^4$	$1 \cdot 10^4$	$7 \cdot 10^3$
$\text{Li} \leftarrow \text{He-H}_2\text{O}$			$1 \cdot 10^3$	$5 \cdot 10^2$	$2 \cdot 10^2$

temperature the necessary concentration decreases. A look at the high-temperature thermal power cycles shows that the optimum values of the bypass solution do not deviate too much from those valid for the secondary loop extraction. A slight increase of the temperature level, however, already yields a marked gain in concentration requirements. In the case of low-temperature thermal power cycles a significant lithium temperature rise in the bypass system is needed to achieve figures comparable with the high-temperature processes.

#### Lithium Inventory

Table VII lists the total lithium inventory, which is the sum of the quantities of lithium present inside the blanket, the main string, and the bypass of the external primary loop. These inventories also decrease with increasing temperature  $\vartheta_s$ , the contributions of the additional bypass amounts being between 10 and 35%.

#### Tritium Initial Inventory and Doubling Time

The tritium initial inventories are summarized in Table VIII. With regard to the same temperature levels in 1.) the bypass solution and 2.) the secondary loop extraction there is no appreciable difference because the decrease in concentration is nearly compensated by the increase in the lithium inventory. Increasing the process temperature  $\vartheta_s$ , however, offers the possibility of a significant reduction of the initial inventory and hence the doubling time. By choosing a very high process temperature even in the case of low-temperature power cycles it is possible to keep the initial inventory down to values discussed today.

The doubling times presented in Table IX are worth discussing in every case. Indeed, an early generation of

fusion power plants will need short doubling times. But even the highest value listed in Table IX is small if compared with the doubling time of the electric energy requirements.

#### Permeation Area Requirements

As can be seen from Table X, the permeation area requirements are rather significant. If potassium topping cycles are applied, these requirements are of about the same order as those for the heat transfer equipment. In the case of helium cooling they even exceed the heat transfer area as well, if a marked temperature rise inside the bypass is achieved. The situation will be quite different if low-temperature thermal power cycles are considered. In relation to the total thermal output of the reactor the equivalent figures are, indeed, higher, but in comparison with the heat transfer surface area necessary to withdraw the heat from the liquid lithium they will only be about 30 - 50%.

#### Cost Figures

Table XI contains a summary of the cost figures resulting for the optimum point of equ. (9). In order to permit a comparison with the method of tritium extraction from the secondary cycle, the estimated material costs of the heat transfer surface, the blanket structural material and the secondary coolant have been added. For the heat transfer surface material and for the blanket structure material stainless steel was assumed in the case of low-temperature thermal power cycles, molybdenum in the case of high-temperature thermal power cycles, in order to avoid tritium contamination of the environment. Where tritium extraction from the secondary loop

is considered the material choice indicated in section 6.2 was applied: niobium for the heat transfer surface, molybdenum for the remainder of the blanket structure.

The specific costs not yet defined were taken to be:

Heat transfer surface:

molybdenum tubes  $\sigma_{Mo} = 0.27 \text{ DM/g}$

stainless steel tubes  $\sigma_{SS} = 0.02 \text{ DM/g}$

Blanket structure:

molybdenum sheet  $\sigma_{Mo} = 0.09 \text{ DM/g}$

stainless steel sheet  $\sigma_{SS} = 0.007 \text{ DM/g}$

Potassium  $\sigma_K = 0.015 \text{ DM/g}$

From Table XI it can be concluded that the bypass solution in the case of high-temperature thermal power cycles should not be more expensive than the method of secondary loop extraction if the process temperature is chosen to be slightly above the coolant outlet temperature of the blanket. From this point of view low-temperature thermal power cycles are competitive provided the process temperature is chosen to be of the same order as in the case of high-temperature cycles.

#### Tritium Risk Potential

Finally, a comparison of the tritium risk potential based on the "internal tritium losses" already defined is presented. As indicated in section 6.1, it shall be assumed that all tritium penetrating the 5 mm thick blanket structural walls will be able to escape from the reactor body and will subsequently reach the reactor containment. This containment shall be completely leak

tight. As a consequence, the tritium will accumulate here and continuously increase the level of radioactive radiation. It was now calculated which value the activity concentration inside the containment will have reached after 20 years of continuous operation in relation to the maximum permissible concentration of tritium in the air. According to [28] the MPC of tritium in air is  $2 \cdot 10^{-6}$   $\mu\text{Ci}/\text{cm}^3$ . For the containment dimensions a cylinder 50 m in both diameter and height was assumed.

Table XII lists the values for the risk potential thus defined. Most apparent is the huge difference between the low and the high-temperature thermal power cycles. The figures calculated for the  $\text{Li} \rightarrow \text{H}_2\text{O}$  cycle with and without intermediate liquid metal loop indicate that owing to the internal tritium losses no special safety precautions have to be provided. Current sealing techniques may be suitable to meet this problem unless the vacuum technology applied to very large vessels necessitates new methods.

In the case of high-temperature thermal power cycles the risk potential is orders of magnitude higher in spite of the use of molybdenum as an excellent diffusion barrier. There are two reasons for this:

- 1.) The high lithium temperature, which is responsible for a high tritium partial pressure, and
- 2.) the higher specific permeation rate of molybdenum in contrast to austenitic stainless steels at those temperatures used here as a basis.



The worst case is that of the direct helium cycle. But the remaining high-temperature processes also require special attention in solving tritium safety problems. One consequence prescribed by the knowledge already noted in section 6.2.4 is that the vacuum sealing should be positioned outside the blanket region, thus providing a first "tritium containment". It should thereby be possible to remove the tritium diffusing from the blanket together with the unburnt fuel of the plasma. If this could be done, no special safeguards would be needed. A decision about this problem, however, requires further investigation in the scope of extended system studies.

#### External Tritium Losses

A similar situation becomes obvious regarding the external losses, i.e. the tritium losses to the secondary circuit. The amounts of tritium lost to the secondary system during 20 years of reactor operation are listed in Table XIII. The figures presented here make it clear that, if potassium or helium cycles are used, special cleaning systems for removing the tritium from the coolants are needed. As far as low-temperature thermal power cycles are concerned, such systems would possibly be unnecessary. The figure belonging to, for example, the  $\text{Li} \rightarrow \text{Li} - \text{H}_2\text{O}$  process at a separation temperature  $\theta_s = 1000^\circ\text{C}$  indicates that after 20 years of reactor operation the activity concentration in the secondary liquid metal loop exceeds the maximum permissible concentration in water by a factor of 2. It can be expected that the water inventory of the last steam cycle in both of the low-temperature systems exceeds that of the liquid metal inventory of the intermediated loop by far more than a factor of 2. From this fact it can be concluded that the steam cycle in the event of low-temperature systems being applied is not more contaminated by tritium than permitted by safety standards.

## 7. Conclusions

Discussion of the amount of tritium necessary to start a fusion reactor is a very complicated matter. This problem cannot be solved by simple estimate. In every case the aim should be to keep the tritium inventory as low as possible. This demand arises predominantly from safety considerations. For the same reason the tritium should be kept concentrated as much as possible, and not spread across a variety of reactor components and systems. Only in this way does effective control seem practicable.

To determine the necessary tritium inventory, one must have some idea of the size of the tritium bearing system, on the one hand, and of the method of tritium removal, on the other. The size of the tritium bearing systems mainly depends on the reactor design, especially the size of the wall loading, and on the kind of thermal power conversion system applied. The tritium removal method determines the necessary tritium concentration.

Today the most complete information on tritium recovery systems offer processes based on the diffusion method. However, in this case, too, there is a lack of knowledge on various points, such as the dissociation behaviour of liquid metal tritides and the dependence of permeation rates on the partial pressure in very low pressure ranges.

Although this knowledge is not yet complete, it is possible to compare various ideas as long as they are based upon the same principle and the same suppositions. This was tried in this report with regard to the tritium removal from the secondary circuit and in a bypass of the primary loop.

In both cases the problems of material choice and the selection of the thermal conversion system became apparent. The importance of these questions is strongly connected with safety factors.

It has also been shown that both methods are rather equivalent as far as effectiveness and expense are concerned. The bypass solution, however, seems to have some essential advantages:

- a) It is independent of the thermal power conversion system applied.
- b) It offers great flexibility in layout, irrespective of heat transfer requirements.
- c) It permits concentration of the tritium and hence diminishes safety problems.

The comparison made in this report is, however, characterized by a certain arbitrariness due to rather inaccurate cost estimates. This, of course, does not affect the method, but the results. A more realistic basis for performing more exact investigations in this direction can only be provided by more refined system studies. Studies of this kind should not stop short at the boundary of the reactor body, but should also include the external equipment of a fusion reactor.

The results gained in this investigation show that external components of the reactor system can produce feedback to the layout of internal components. Furthermore, the safety of a fusion reactor as one of the most essential arguments for fusion development does not only depend on the design of the reactor, but is also strongly influenced by its external systems.

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