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TRITIUM TECHNOLOGY REVIEW*

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TRITIUM TECHNOLOGY REVIEW

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ABSTRACT

The methods used in recovering, containing and controlling tritium in fusion reactor systems are the subject of the paper. Safety, cost, and breeding needs for tritium control are outlined.

I. INTRODUCTION

Tritium will be present in deuterium-tritium reactors in several forms: as energetic particles in the plasma; as T_2 or T_2O in the process streams and in the blanket; and as a radioactive impurity present in materials throughout the fusion plant. Tritium losses from the plant arise from gaseous emissions, liquid and solid wastes, and tritium decay. Major goals in fusion reactor design are to control tritium losses and to minimize tritium inventories. If these goals are achieved then: 1) the tritium breeding ratio is minimized; 2) the environmental and worker hazard is minimized; and 3) capital costs are minimized.

There are five major methods which are available to control tritium losses: 1) using material with low permeability in series, as composites or as coatings; 2) using isotopic swamping; 3) converting the tritium gas, T_2 , to a non-permeable form, T_2O ; 4) recovering the tritium lost; and 5) reducing the tritium partial pressure and the inventory at the source.

The goal of this paper is to address the issue of tritium control and its complement tritium recovery as they relate to tritium technology. For comparison purposes, a reference reactor producing 4000 MW fusion power at 75% availability is assumed.

II. TRITIUM SYSTEMS

Within a fusion reactor, one has an interconnected set of systems. At the outer boundary of the plasma are the first wall, the limiter or divertor, halo scrapers, direct

convertors, and neutral beam dumps. Materials used here see a combination of tritium implantation and diffusion at high heat fluxes. The coolants associated with these systems require a tritium recovery system to reduce the tritium losses. Next is the blanket system (breeder, coolant, heat exchanger). The subsystems included are a tritium removal and impurity removal system for the coolant and a tritium control system for the heat exchanger/steam generator. Interfaced with the reactor is the system which fuels the reactor and purifies its exhaust. A prototype of this system has been described.¹ Last is the atmospheric tritium recovery system which removes tritium from the plant's air. The issues which affect its efficiency and cost have been summarized.^{2,3}

The areas in which major tritium losses occur are at the boundary of the plasma and in the blanket system, areas with high temperature and large surface areas. These areas will be emphasized in the following discussions.

III. BREEDING NEEDS

The breeding ratio required in a commercial fusion reactor depends on three factors: 1) the tritium lost due to decay, processing losses, releases; 2) the doubling time chosen to provide fuel for a new reactor; and 3) the tritium inventory required for a new plant. Each of these factors are highly dependent on the design of the fusion plant.

The tritium losses due to decay depend on the base inventory in a fusion plant. This inventory is the sum of the tritium in storage (reserve for system failures), the tritium in all processing systems and in the reactor, the tritium in the blanket itself, and the excess tritium being stored for a new reactor. For example, in a 4000 MW plant, the total inventory is ~7 kg if the following assumptions are made. In reserve storage there is two days of burn (~1.1 kg), in all processing systems and waste areas there is ~0.9 kg, in the blanket

there is <1.5 kg (~6 wppm), and in excess storage ~3.5 kg (the sum of the first three). The yearly decay loss is ~0.4 kg.

The processing losses are dependent on three factors: 1) the overall efficiency of a given processing system; 2) the amount of material processed (this is inversely proportional to the fractional burn for the fueling system); and 3) the tritium wastes and/or releases not recovered. The overall efficiency of the process system can be improved by using a number of highly efficient systems in series. If an efficiency of 90% is assumed for each component, the overall efficiency for three component systems is 99.9% or a loss of 0.1%. For a 4000 MW fusion reactor, a processing loss of 0.1% corresponds to 20.8, 10.4, 7.1, and 5.6 g/d if the fractional burns are .05, .10, .15 and .19 and if fueler efficiencies are ~50%. At 75% availability this is 5.7, 2.8, 1.9, and 1.5 kg/yr of tritium respectively. An example of a processing loss is permeation into the coolant. At a rate of 1 g/d, ~0.3 kg/yr is lost by this route. If a recovery system (capital cost <\$50 M-Section VI) is installed, this loss is eliminated.

The excess tritium produced for breeding ratios of 1.03, 1.05, 1.07, 1.10, 1.15, and 1.20 is 4.4, 7.1, 10.1, 14.5, 21.9 and 29.3 kg/yr. As an approximation, the total tritium loss each year was assumed to be equal to the decay plus 0.1% processing loss. The doubling time needed versus breeding ratio is shown in Fig. 1 as a function of R, the ratio of the processing loss (%) to the fractional burn. If six years is the desired doubling time, then at an R of 0.5, a breeding ratio of 1.04 is adequate. This R ratio is achieved with a reactor design with a burn of 0.02 and processing losses of 0.01%/d, or a burn of 0.2 and 0.1%/d losses.

IV. SAFETY REQUIREMENTS

The radiation guidelines and limits for both workers and the general public are summarized in Table 1. The worker policy of DOE is to keep exposure as low as reasonably achievable (ALARA).

This policy limits onsite workers to less than 1 rem/y (0.5 mrem/h). Based on working 8 h/d and 40 h/wk, the occupational exposure is 0.5 mrem/h from all radiation sources. Exposure of an unprotected finger to water at 1 Ci/L results in a dose of 0.5 mrem/min exposed. The dose for hand exposure is 4 mrem/min exposed.⁶

If there is a 500 m exclusion zone around the plant and if all releases occur from a 100 m stack, the regulatory limit from all pathways is 200 Ci/d for all types of radiation. An upper limit for tritium releases would then

Fig. 1. Doubling time vs. breeding ratio as a function of fractional burn (PB) and daily tritium losses ($L = Z$) where $R = L/PB$.

be 100 Ci/d or ~4 g/yr from all sources. (This does not include tritium contained in wastes which are correctly packaged.) The guideline limit would be ~10-20 Ci/d from the steam generator and any gaseous waste streams.

V. TRITIUM CONTROL PARAMETERS

The choice of material used in a given application depends on its function. For some blankets, permeation is used to recover the tritium produced. Thus, a material with high permeability is desired. Contrary, in a limiter, loss of tritium after implantation is undesirable, so materials with low diffusivities are used.

A. Permeation

The permeability of hydrogen isotopes in metals and alloys in a non-irradiated environment varies by more than ten orders of magnitude, depending on the temperature chosen. Palladium is at the high end, with a room temperature permeability of $4E-10$ mol/m-s $Pa^{1/2}$, and gold is the lowest at $1E-27$ under similar conditions. Silicon, germanium, and ceramics have permeabilities, extrapolated to room temperature, of less than $1E-40$, all with very high activation energies. For more than a dozen rare metals, there is no data. There is little data on ceramics, because of their low permeability, and, consequent difficulty in measurement. Table 2 from a recent review⁷ gives selected permeabilities for materials that don't readily form hydrides.

Hydrogen permeation through materials is affected by the surface condition. The standard permeation equation represents the steady state "diffusion controlled" condition which requires that all surface processes leading to solution, and their reverse, be much more rapid than the permeation flux.

TABLE 1. U.S. POLICY ON RADIATION DOSES

Operating Conditions	Probability of Occurrence (p/yr)	On-Site Dose		Off-Site Dose	
		Guideline	Limit	Guideline	Limit
Normal operation (uncontrolled area)	$p = 1$	≤ 0.1 R/y	≤ 0.5 R/y	≤ 10 mR/y ^a	≤ 50 mR/y ^b
Moderate frequency (controlled area)	$10^{-1} < p < 1$	≤ 1 R/y	≤ 5 R/y ^c	≤ 100 mR/y ^a	≤ 0.5 R/y
Infrequent incidents	$10^{-3} < p < 10^{-1}$	≤ 5 R	≤ 25	≤ 1 R	≤ 5 R
Limiting faults	$10^{-7} < p < 10^{-3}$	≤ 25 R	≤ 25	≤ 5 R	≤ 25 R

^a Per individual.^b EPA⁴ (Environmental Protection Agency) has proposed guidelines of 10 mrem/y. The NRC⁵ has a limit of 5 mrem/y (whole body).^c This is yearly limit. The quarter limit is 3 Rem.TABLE 2. TRITIUM PERMEABILITY VALUES FOR METALS⁷

Metal	ϕ_0^a (mol/m ² s Pa ^{1/2})	E_ϕ (K)	Temperature Range (K)	Pressure	Ref. Gas
Aluminum	5.8(-5) ^b	14,800	420-520	20 kPa	T ₂
Beryllium	5.8(-14)	2,220	670-1170	Outgassing	T ₂
Cobalt					
(ε)	5.1(-9)	6,850	< 670	0.5 MPa	D ₂
(α)	3.1(-8)	7,750	> 670	0.5 MPa	D ₂
Copper	4.9(-7)	9,320	470 - 700	0.13-130 kPa	H ₂
Germanium	7(-6)	24,000	1040 - 1200	100 kPa	H ₂
Gold	2.5(-6)	14,800	500 - 900	100 - 500 kPa	D ₂
Iron ^c	2.4(-8)	4,200	> 375	< 10 MPa	H ₂
Lead ^d	-	-	-	-	-
Molybdenum	1.3(-7)	9,710	500 - 1700	-	H ₂
Nickel	1.54(-7)	6,500	300 - 650	0.1 - 30 MPa	T ₂
Niobium	3.6(-9)	- 4,270	> 700	< 10 kPa	H ₂
Palladium	1.3(-7)	1,885	300 - 709	0.04 - 7 Pa	H ₂
Platinum	6.9(-8)	8,500	540 - 900	70 kPa	H ₂
Silicon	8(-6)	27,000	1240 - 1485	100 kPa	H ₂
Tantalum	3.3(-9)	- 2,430	> 650	< 10 kPa	H ₂
Tungsten	4.5(-6)	17,000	1100 - 2400	10 - 100 kPa	H ₂
Vanadium	2.3(-9)	- 2,990	> 600	< 10 kPa	H ₂

^a $\phi_0(T_2)$ adjusted with $\sqrt{m_{H_2}/m_{D_2}} / \sqrt{m_{T_2}}$ mass ratios.^b $5.8(-5) = 5.8 \times 10^{-5}$.^c The value for T₂ of Louthan et al. (1975) seems high.^d Data unreliable.

B. Implantation

For materials near the plasma, implantation of tritium at the near surface produces a tritium concentration potential. A review article⁸ summarized recent developments, theoretical and experimental, on implantation.

Baskes⁹ has derived a general expression for k_r , a phenomenological rate constant for molecular recombination. It is:

$$k_r = k_r^0 \exp((E_g - E_D)/kT), \quad (1)$$

where

$$k_r^0 = (8/MT)^{1/2} (\alpha c_1/S_0^2), \quad (2)$$

E_D is the activation energy for diffusion, $c_1 = 2.64 \times 10^{25} (K \text{ AMU})^{1/2} \text{ cm}^{-2} \text{ s}^{-1}$, M is the molecular weight of the gas molecules, S_0 is the solubility preexponential, and α is a phenomenological factor which characterizes the surface "cleanliness". (If $\alpha \sim 1$ the surface is considered "clean"; at 10^{-4} it is heavily contaminated. Table 3 lists selected values¹² of diffusivity and solubility; the data for non-metals is based on questionable data.

Good agreement is found between experiment and theory for many metals, e.g.,^{9,20-21}, but very little is known for non-metals. A recent study by Causey et al.²² has shown the effects of surface contamination of k_r for stainless steel. Surfaces under high energy H bombardment ($> 100 \text{ eV}$) are observed to increase their cleanliness and also the k_r value.

Tritium migration rates and inventories for two limiter materials²³ are presented in Table 4. With lithium as a coolant, vanadium might be chosen as a limiter material since the tritium could be removed in the same processing system as used for blanket tritium recovery. With water as a coolant, the use of PCA as a limiter material results in high permeation rates and costly cleanup ($\sim \$50 \text{ M}$). The permeation rate for PCA as a function of α is shown in Fig. 2.

The effects of neutron and gamma irradiation are the most critical unknowns in modeling tritium permeation in fusion reactors. Gamma irradiation effects on hydrogen transport were reviewed by Causey,²⁴ however the database is too limited to make any definitive statements on possible gamma effects.

Four hydrogen transport experiments conducted on neutron irradiated materials have mixed results. Miyake²⁵ and Katsuta²⁶ reported small changes in hydrogen transport in low fluence neutron-irradiated titanium, molybdenum, and platinum. Dobrozensky and co-workers²⁷ reported small, irreproducible changes in hydrogen permeation measured

Fig. 2. Tritium permeation through PCA into the coolant as a function of time (4000 MW plant).²³

dynamically in a fission reactor core with and without reactor power. Quick²⁸ observed a 100-fold decrease in hydrogen solubility of 310 stainless steel following a fast neutron fluence of 10^{17} n/cm^2 at 100°C .

C. Barriers

High recycling coatings on the plasma surface, as well as low permeability coatings on the coolant surface have been proposed as barriers. This subject was reviewed by Kerst and Swansiger.⁸ While reductions of 20 orders of magnitude are predicted for downstream barriers such as silicon and Al_2O_3 , (Fig. 3) the observed reductions in experiments ($< 10^3$) are less impressive because of problems with coverage, adherence and integrity. In a fusion reactor, a permeation barrier will be in an extremely harsh environment that includes thermal cycling and neutron bombardment. The reduction factor achieved may be much less than 10^5 . To be successful, a coating must be elastic to follow the expansion of the metal, i.e. it must be integral with the metal.

TABLE 3. SELECTED VALUES OF DIFFUSIVITY AND SOLUBILITY

Metal	D_0 (cm^2/s)	E_D (eV)	S_0 ($\text{H}/\text{cm}^3 \sqrt{\text{atm}}$)	E_S (eV)	Ref.
Be	$3(-7)^a$	0.19	7.5(17)	0.0	10
Cu	1.1(-2)	0.40	1.3(20)	0.37	10
Fe	7.8(-4)	0.08	2(20)	0.28	11
Nb	5(-4)	0.11	5.1(19)	-0.37	12,13
Mo	4.8(-3)	0.39	6.8(20)	0.54	14,15
Stainless	2(-3)	0.54	7.7(19)	0.11	10
Ta	5(-4)	0.16	4.8(19)	-0.36	10
V	5.2(-4)	0.08	4.4(19)	-0.34	10
W	4.1(-3)	0.39	6.9(20)	0.98	10
Zr	4.2(-3)	0.41	6.5(19)	-0.63	16,17
Non-metals					
B	---	0.8	---	---	10
BeO	---	2-2.5	---	---	10
C	3.3(2)	4.3	9(15)	-1.45	10
SiC	1.58	3.2	4.2(14)	-1.6	10
TiC	1.3(-12)	0.13	---	---	18
(H saturated)					

^a $3(-7) = 3 \times 10^{-7}$.

TABLE 4. TRITIUM PERMEATION AND INVENTORY FOR PCA AND VANADIUM^a

Tritium flux Area (m^2) Material	$\left(\frac{\#}{\text{cm}^2 \cdot \text{sec}}\right)$	1.28(16)		1.5(15)	
		V	PCA	V	PCA
Wall thickness (mm)		6	6	2	3
Front surface temperature ($^{\circ}\text{C}$)		600	595	500	450
Back surface temperature ($^{\circ}\text{C}$)		400	330	500	450
Front surface recombination (α_f)		1	0.5	1	0.5
Back surface recombination (α_b)		1	5(-5)	1	5(-5)
Tritium permeation (g/d)		168	0.13	162	0.19
Tritium inventory (g)		5.6	18.7	0.65	3.7

^a For 4000 MW reactor, ref. 23.

^b $1.28(16) = 1.28 \times 10^{-16}$.

Fig. 3. Permeability of hydrogen through various candidate permeation barrier materials (Ref. 7).

In limiter blade and halo scraper design, a combination of two or more materials have been used to reduce the amount of tritium transported to the coolant. Systems considered are Be/Cu, C/Cu, and W/Cu.

D. Isotopic Swamping

The effect of excess hydrogen on the permeation of tritium has been examined.²⁹⁻³⁰ The permeation rate is reduced since the square root pressure dependence on P_{T_2} becomes a linear dependence on P_{HT} i.e.

$$P_{T_2}^{1/2} \text{ vs. } P_{HT}/P_{H_2}^{1/2}. \quad (3)$$

In a recent study²³ the effect of isotopic swamping on blanket recovery systems was assessed. It was found that the purge gas rate for tritium recovery from blanket systems can be reduced since the gas being recovered is HT instead of T_2 . The mole fraction x_H is $\gg x_T$, therefore P_{HT} is larger than P_{T_2} at the same tritium concentration.

$$x_T^2 \text{ vs. } x_T x_H \quad (4)$$

In addition, the excess hydrogen enhances the surface recombination rate which facilitates the release of tritium from the blanket material. The calculated effect of the use of isotopic swamping in the 17Li83Pb system is

shown in Table 5. It was found in an in-situ solid breeder tritium recovery experiment that adding hydrogen to the helium purge gas reduced permeation.³¹

E. Oxidation

The conversion of tritium to the oxidized form, T_2O , is a useful technique if the species formed is later collected so as not to become a worker or public hazard, and if the conversion is greater than 99.9% effective. From the thermodynamics of the reaction

one would assume that the tritium is completely oxidized.³² However, the gas phase homogeneous kinetics³² and the gas phase heterogeneous kinetics are both slow.³³ In addition, water in contact with a reducing surface (stainless steel) is reduced to some extent³⁴ (see Fig. 4) i.e.

However a model has been proposed which predicts in-situ formation of T_2O ³⁵ on the back surface of a material through which tritium atoms are permeating.

Before this model can be accepted, experimental studies are needed on the kinetic significance of the proposed surface reaction.

VI. COOLANT CLEANUP

Several processes were investigated by INTOR³⁶ participants to separate tritium from water coolant. Of these processes the four that appear most appropriate are: Vapour Phase Catalytic Exchange/Cryogenic Distillation (VPCE/CD), Direct Electrolysis/Cryogenic Distillation (DEL/CD), Combined Electrolysis Catalytic Exchange/Cryogenic Distillation (CECE/CD), and Water Distillation/Electrolysis/Cryogenic Distillation (WD/EL/CD). All four processes use low-temperature (cryogenic) hydrogen distillation for the final isotopic tritium enrichment but differ in the front end process used to transfer the tritium from the water to the elemental form. A fifth system, LPCE/CD, would be comparable in cost to the first three for light water systems.

Of the first four, DEL/CD, CECE/CD and WD/EL/CD have comparable capital and operating costs for light water based coolant systems. The VPCE/CD process would have significantly higher capital and operating costs for light water but would compare favorably for a heavy water based coolant.

TABLE 5. THE EFFECT OF ISOTOPIC SWAMPING ON $^{17}\text{Li}^{183}\text{Pb}$ BLANKET TRITIUM RECOVERY SYSTEM

	No H_2	0.01 kPa H_2	0.001 kPa H_2
H_2 pressure (KPa)	0	1.(-2) ^a	1.(-3)
Outlet P_{T_2}/P_{HT} (kPa)	6.5(-7)	9.3(-5)	2.8(-5)
	(T_2)	(HT)	(HT)
Purge gas flow rate (L/s)	1.(7)	1.45(5)	4.85(5)
Max. P_{T_2}/P_{HT} (kPa)	1.3(-10)	1.45(-6)	1.(-7)
allowed at SG ^b			
Mass H_2 (kg/d)	0	42.5	14
System efficiency (%)	99.98	98.5	98.5

^a 1(-2) = 1×10^{-2} .

^b SG = steam generator.

Fig. 4. Hydrogen permeation of two sample capsules of austenitic steel with different capsule contents, plotted as a function of temperature. Curve A is for pure hydrogen, curve B is for hydrogen and water (ratio 1:1), curve C is for water only (Ref. 34).

The CECE/CD technology has the advantage of detritiating water to a concentration that is acceptable for unrestricted release to the environment. This feature becomes important when decommissioning a coolant loop and its $\sim 10^5$ liters of water. A large CECE/CD system would also be capable of detritiating water from the building atmospheric tritium recovery systems and the primary fuel cycle purification system because it has the capability of accepting feeds at different tritium concentrations.

The process flowrate necessary to detritiate the water coolant depends on the tritium permeation rate and the allowable coolant concentration. In fact, the volume processed is equal to the permeation rate divided by the coolant tritium concentration and the system detritiation efficiency:

$$(L/d) = (C/d) / [C/L] (0.0-1.0) \quad (9)$$

If the detritiation efficiency is assumed to be one (valid for CECE/CD) then the process flowrate can easily be calculated for the permeation rates and coolant concentration of interest.

The capital cost (\$M-(82)) for CECE/CD is

$$C = \$15 \left[\frac{(C/d)}{[C/L] \times 1000 (L/d)} \right]^{0.55} \quad (10)$$

Limited work has been done on clean up systems for other coolants. Helium clean up systems would include the oxidation of tritium followed by flow through molecular sieve for moisture absorption. Later regeneration of the sieves is needed for tritium recovery. The critical issue is the rate of oxidation and the allowable tritium concentration in the

coolant. If the oxidation rate is slow, tritium may have to be removed in the form T_2 , which is much more difficult. If safety considerations require a low tritium concentration in helium coolant, then the percentage of the coolant which has to be processed becomes large. A further constraint is that at low water partial pressures, the molecular sieve becomes less efficient. Unfortunately, a detailed system design has not been performed.

The only other non-breeding coolant currently considered is the nitrate salt.²³ Very little work has been done in this area. It is known that nitrate salt is a good oxidizing material and therefore, tritium will be oxidized. It is assumed that T_2O will be recovered by an evaporation process.

VII. TRITIUM RECOVERY FROM BLANKETS

A schematic of blanket tritium systems for both a self-cooled and a separate cooled system is shown in Fig. 5. For each system, up to three separate tritium recovery subsystems may be required for a given design. For some self-cooled systems, an intermediate coolant loop is also required. Because the efficiency of a tritium recovery process can be affected by impurities, impurity removal systems are important subsystems in the blanket tritium recovery system. The protium addition system (HI) was discussed under isotopic swamping.

A. Liquid Breeders

The tritium problems associated with the two liquid metal breeder candidates are different. The tritium solubility in lithium is very high. Therefore, the tritium recovery system is designed to recover tritium at a concentration with an acceptable inventory. The tritium partial pressure is low so that containment is not a key problem in normal operation. For ^{17}Li - ^{83}Pb breeder, the solubility is very low. Therefore, the tritium recovery system will have to be designed to reduce the tritium partial pressure to acceptable levels to reduce undesired permeation. The tritium inventory is small so that it is not an important issue.

The tritium recovery systems for liquid lithium are of three types: 1) permeation to a secondary loop where the tritium is collected using a flowing gas or a sodium cold trap; 2) the use of yttrium getters; and 3) the use of molten salt extraction. All three systems have been demonstrated at the laboratory scale but face different problems when scaled to reactor level systems.

There has been no experimental verification of tritium recovery in ^{17}Li - ^{83}Pb . Due to the high tritium partial pressure, it is reasonable to assume that tritium will be

A. SELF COOLED

B. SEPARATE COOLANT

II, I2, I3: IMPURITY REMOVAL SYSTEMS

TR1, TR2, TR3: TRITIUM RECOVERY SYSTEMS

HI: PROTIUM ADDITION SYSTEM

Fig. 5. Schematic of blanket tritium recovery systems.

recovered from the vapor. A gas sparging system³⁷ and vacuum degassing of a stream³⁸ of droplets have been considered. To avoid the economic limit on the pumping rate ($\sim 10^6$ liter/sec) and to lessen the diffusion resistance, the tritium partial pressure can not be reduced lower than 10^{-4} torr. At such a high tritium partial pressure, a tritium diffusion barrier will be needed to reduce the permeation by a factor of 500 at the steam generator. To accomplish this, a double-walled steam generator is used.³⁹ All piping and valves require secondary containment. The tritium permeation calculation assumes a helium purge stream in the gap contains 1 torr oxygen partial pressure. The tritium is assumed to oxidize upon collision with the oxygen.

Oxidation kinetics in the gas phase can be slow.²³ If the kinetic rate of oxidation is a true limit, the effectiveness of the double-walled steam generator in providing tritium diffusion resistance may be reduced. To achieve an acceptable tritium permeation rate with a reasonably sized tritium recovery system, the concept of isotopic swamping is used. Since tritium permeates as the T form, hydrogen is added to suppress the formation of T. Isotopic swamping may explain some experimental observations.^{31,40,41}

B. Solid Breeders

Recent in-pile experiments indicate that a significant fraction of the tritium generated in a solid breeder is released as T_2 .^{31,42} The amount was 99% for γ -LiAlO₂, and up to 20% for Li₂O. In addition, the addition of hydrogen to the helium purge stream used for tritium recovery appeared to enhance tritium release. Contrary, the addition of oxygen actually impeded tritium release. As mentioned in the isotopic swamping section, tritium permeation was reduced when hydrogen was present. In Fig. 6, the expected behavior of a γ -LiAlO₂ blanket system²³ is shown. The time needed to attain 99% of the tritium release rate is 2 months; to attain the steady state inventory (>1.5 kg) takes 4 years.

Fig. 6. Tritium response to an instantaneous reactor startup for the LiAlO₂/H₂O/Be/HT-9 design with $T_{min} = 350^\circ\text{C}$ and $T_{max} = 1000^\circ\text{C}$. Assumed grain diameter is 0.2 μm . $\tau_{67.4}$ = time to reach 67.4% of the steady state value. τ_{99} = time to reach 99% of the steady state value (Ref. 23).

One of the primary problems in the design of a solid breeder blanket tritium recovery system is the tritium permeation rate into the coolant. For the HT-9 structural material, Figs. 7, 8, and 9 show the limiting tritium partial pressure, the purge gas volumetric flow rate, and the purge gas pressure drop as a function of the allowable permeation rate and the permeation resistance of modified HT-9. Using these figures, and the cost of a tritiated water coolant system, one can design the most cost effective blanket.

C. Comparison of Blanket Tritium Recovery Systems

In a recent study,²³ an attempt was made to compare the systems needed to achieve tritium recovery from several blanket systems in a

Fig. 7. Limiting tritium partial pressure in the purge channel plotted as a function of maximum tritium permeation rate ($\dot{P}_A$) into the primary coolant loop (γ -LiAlO₂/H₂O/HT9/Be blanket).

cost effective manner which met radiation guidelines. The tritium transport rates in various sections of the system are found in Table 6. The tritium inventories in the system are found in Table 7. For the lithium self-cooled system, an intermediate sodium loop is used (Example A in Fig. 5). No intermediate loop is present in the 17Li83Pb system, but protium swamping is used. The costs for the systems are ~\$100 M.⁴³

VIII. CONCLUDING REMARKS

The major need in tritium control in fusion reactors is the development and demonstration of cost-effective methods of reducing the tritium partial pressure. To date, most conceptual reactor designs have assumed that either a barrier will stop permeated tritium or that oxidation is effective in controlling it. Neither of these methods has been demonstrated. An area in which little work has been done is the development of ceramic heat exchangers. An appropriate ceramic is needed in which tritium permeation rates are reduced orders of magnitude while heat transfer is optimized. Within the reactor, the effect of

TABLE 6. TRITIUM MASS FLOW RATES IN THE FIRST WALL/BLANKET FOR SOME REPRESENTATIVE SYSTEMS

	First Wall to Breeder g/d	First Wall to Coolant g/d	Breeder T ₂ Pressure Pa	H ₂ Added in Purge g/d	E H ₂ Pressure Maximum Pa	T to Coolant from Be g/d	Purge to Coolant g/d	Coolant T ₂ Pressure Pa	Leakage to He Curie/d	Leakage to HX Curie/d
Li/Li HT-9	.2	---	8.6(-9) ^a	---	8.6(-9)	---	---	---	200	2
LiPb/LiPb V	162	---	1.3(-4)	4.2(4)	.3	---	---	---	---	10
Li/Li V	162	---	8.6(-9)	---	8.6(-9)	---	---	---	200	2
Li ₂ O/He HT-9	---	0.2	0.78	5.6(4)	238	---	0.65	6.4(-4)	---	20
LiAlO ₂ He/HT-9 Be	---	0.2	0.84	5.7(4)	256	6	2.31	1.9(-3)	---	43
Li/He HT-9	---	0.2	1.1(-7)	---	1.1(-7)	---	---	7.8(-7)	---	17
FLiBE/He ^b HT-9/Be	---	0.2	2.6(3)	---	2.6(3)	6	83	.14	---	191
LiAlO ₂ /H ₂ O HT-9/Be	---	0.2	2.2	3.7(3) 10 Pa(H ₂ O) ^c	42	---	0.01	6(-6)	---	2

^a 8.6(-9) = 8.6×10^{-9} .^b A W coating is required in the blanket and an Al coating is required in the steam generator. System was designed at lower thermal power and scaled to the higher thermal power for both tokamak and TMR designs.^c 10 Pa of hydrogen over pressure in the primary coolant was assumed and T/E ratio in the gas phase was assumed to be equal to T/E ratio in water.

TABLE 7. TRITIUM INVENTORIES IN THE STRUCTURE AND BLANKET FOR SOME REPRESENTATIVE SYSTEMS

	First Wall Inventory g	Blanket Structure Inventory g	Structure Maximum H Concentration wppm	Purge Inventory g	Coolant Inventory g	Breeding Material Inventory g
Li/Li HT-9	3.7	2(-2) ^a	0.3	---	---	336
LiPb/LiPb V	0.65	5	10	0.02	---	0.1
Li/Li V	0.65	0.47	0.3	---	---	336
Li ₂ O/He HT-9	3.7	1.0	0.6	0.7	3.4(-5)	131
LiAlO ₂ He/HT-9 Be	3.7	0.8	0.6	0.7	1.3(-4)	24
Li/He HT-9	3.7	7(-2)	0.3	---	6(-7)	233
FLiBE/He HT-9/Be	3.7	139	1.5	---	---	0.4
LiAlO ₂ /H ₂ O HT-9/Be	3.7	9.3	0.5	0.03	53	1500

^a 2(-2) = 2×10^{-2} .

Fig. 8. Limiting volumetric gas flow rate in the purge channel plotted as a function of maximum tritium permeation rate ($\dot{m}_t$) into the primary coolant loop (γ -LiAlO₂/H₂O/HT9/Be blanket).

neutrons on material tritium diffusivity and solubility needs to be determined.

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Fig. 9. Purge channel pressure drop (Δp) plotted as a function of maximum tritium permeation rate ($\dot{m}_t$) into the primary coolant loop (γ -LiAlO₂/H₂O/HT9/Be blanket).

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