AN ANALYSIS OF TRITIUM DISTRIBUTION AND LEAKAGE CHARACTERISTICS
FOR TWO FUSION REACTOR REFERENCE DESIGNS

by Victor A. Maroni

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AN ANALYSIS OF TRITIUM DISTRIBUTION AND LEAKAGE CHARACTERISTICS FOR TWO FUSION REACTOR REFERENCE DESIGNS

Victor A. Maroni
Chemical Engineering Division
Argonne National Laboratory
Argonne, Illinois

Summary Results of analyses of the tritium distribution and leakage characteristics for two current fusion power reactor designs—the Princeton Reference Model Tokamak and the LASL/ANL Reference Theta Pinch Reactor—are presented and compared. The philosophy of the model used in these analyses is discussed, and the computer code, developed to carry out the calculations, is described. Consideration is given to the practical aspects of those design features which have the greatest influence on tritium inventories and leakage rates; i.e. (1) the blanket and fuel processing systems, (2) the protective atmospheres, and (3) the application of some special materials as tritium barriers. The results of these analyses have permitted several interesting comparisons to be made between a molten salt-stainless steel reactor design (the Princeton Tokamak model) and a liquid lithium-refractory metal reactor design (the LASL/ANL Theta-Pinch model).

Introduction

A study has been initiated to evaluate the distribution and leakage profiles for tritium in two current fusion power reactor designs—the Princeton Reference Model Tokamak¹ and the Reference Theta Pinch Reactor.² These profiles are determined by the technical features of each design and are expected to have an important bearing on the operational characteristics of the reference models. The motivation for investigating the location and quantities of tritium in currently conceived D-T-fueled fusion reactors can be simply stated as follows: Prototypical fusion power plants will require tritium inventories in excess of 10^7 curies and it is likely that the permissible level for tritium release to the environment from such plants will be limited to about 3000 Ci/yr/1000 MWt. (This corresponds to a daily loss rate of less than 1 ppm of the total inventory.) Based on anticipated standards, tritium releases in excess of this figure may begin to constitute an objectionable radiologic hazard to the environment. In addition to the environmental considerations there are other potentially significant consequences of the tritium in fusion reactors and they too must be assessed. These include hold-up of tritium in the structure, accelerated corrosion and loss of mechanical integrity due to hydriding, and assorted maintenance problems.

The principal objective with regard to control and handling of tritium is to minimize the total inventory and loss rate of tritium in every sector of the plant. For the most part, the minimization of both the tritium inventory and the loss rate can be accomplished by (i) processing the blanket so as to achieve the lowest possible concentration of tritium, (ii) employing a contiguous system of barriers (both metal and ceramic) that are impervious to the hydrogen isotopes in any chemical form, and (iii) operating a processable gaseous purge flow system around and throughout the plant on a continuous basis.

Important Properties That Affect Tritium Distribution and Leakage

The important properties of the hydrogen isotopes that will determine their distribution and leakage

characteristics in currently conceived fusion reactors are their solubilities in the materials with which they come in contact, their diffusion rates through the working fluids and structural components, and the stable chemical forms in which they exist. In this section we review briefly the formalisms which permit us to express these properties in terms of parameters that we expect to control.

In order to make a reasonable estimate of the tritium held-up (dissolved) in the structure and fluids we must have expressions for the solubility of tritium in the materials of construction. For the case of dilute solutions of gaseous hydrogen (H_2, D_2, T_2) in metals, the relationship between solubility and pressure is given by Sieverts' Law

$$C = K_s \sqrt{P} \quad (1)$$

where C is the concentration, P is the H_2 partial pressure, and K_s is the Sieverts' constant.³ It can further be shown that K_s is exponentially dependent on temperature, hence the general expression relating concentration to H_2 pressure and temperature is given by

$$C = K_s^0 \sqrt{P} \exp(Q_s/kT) \quad (2)$$

in which K_s^0 is a constant, Q_s is the activation energy for solution of hydrogen in the metal, k is the Boltzmann factor on a molar basis ($k = 1.987$ cal/mole) and T is the absolute temperature. Values of K_s^0 and Q_s for materials of interest to the subject models are given in Table I.

The solubilities of the important tritium bearing species in a molten salt blanket fluid (HT, T_2, TF , etc.) are directly related to their partial pressure in accord with Henry's Law and can be expressed as follows

$$C = K_H P \quad (3)$$

where C and P are the concentration and pressure of a given species in the blanket, and K_H is the Henry's Law constant for that species. Since K_H is an exponential function of temperature, the general expression which relates concentration to temperature and pressure is given by Eq. 4.

$$C = K_H^0 P \exp(Q_H/kT) \quad (4)$$

Malinauskas and Richardson¹⁰ have measured the solubility of H_2 and D_2 in molten Li_2BeF_4 at pressures in the range of 1.0 to 2.1 atm and for temperatures between 500 and 700°C. Values of K_H^0 and Q_H for H_2 and D_2 determined from their data are given in Table II, together with estimated values of these parameters for T_2 solutions in Li_2BeF_4 . The solubilities of HF and DF in lithium fluoroberyllates have been determined by Field and Shaffer.¹² Values of K_H^0 and Q_H for HF and DF and estimated values of these parameters for TF are also given in Table II. An assumption inherent in the calculations which follow is that the Henry's Law constants determined from the data in Table II are applicable at pressures that are 10^4 to 10^6 times smaller than the pressures used by the experimenters.

The rates of migration of H, D , and T through structural interfaces are determined from available permeation data for the materials of construction.

TABLE I. Parameters for the Sieverts' Law Constants, K_s , of H_2 , D_2 , and T_2 in Reference Design Materials

$$K_s = K_s^0 \exp (Q_s / RT)$$

Material	K_s^0 (wppm ^a /Torr ^{1/2})			Q_s (cal/mole)			Reference
	H_2	D_2	T_2	H_2	D_2	T_2	
PE-16	0.4	0.7	1.1	-1,680	-1,750	-1,830	(b)
Incoloy 800	0.4	0.7	1.1	-1,680	-1,750	-1,830	(b)
Copper	13.6	(27.2) ^c	(40.8) ^c	-13,911	(-13,911) ^c	(-13,911) ^c	6
Lithium	5.9	8.3	10.7	12,910	12,910	12,910	7
Sodium	7.2	14.4	21.6	-550	-550	-550	9
Niobium	0.35	0.49	0.6	8,430	8,430	8,430	8

^awppm = weight parts per million H,D, or T

^bCalculated by using the data of Ref. 4 for 347-SS and assuming the values of K_s^0 and Q_s for PE-16 and Incoloy 800 would be essentially the same as for 347-SS. (Values of K_s for Incoloy 800 and PE-16 are not expected to be significantly different from those for 347-SS based on previous studies³ of the effect of composition on solubility of hydrogen in Fe-Cr-Ni alloys.)

^cApproximated by assuming the same parameters that are given for H_2 in Ref. 6.

Table II. Parameters for the Henry's Law Constants, K_H , of H_2 , D_2 , T_2 , HF, DF, and TF in Li_2BeF_4

$$K_H = K_H^0 \exp (Q_H / RT)$$

Species	K_H^0 (wppm ^a /Torr)	Q_H (cal/mole)
H_2 (b)	3.92×10^{-3}	-7,692
D_2 (b)	4.35×10^{-3}	-6,578
T_2 (c)	(4.97×10^{-3})	(-6,036)
HF (d)	9.14×10^{-5}	5,981
DF (d)	12.0×10^{-5}	5,432
TF (c)	(14.8×10^{-5})	(6,642)

^awppm = weight parts per million H,D, or T

^bObtained by least squares refinement of the data from Ref. 10

^cEstimated using the method in Ref. 11

^dObtained from Table IV of Ref. 12

(Webb^{13a} and Stickney^{13b} have reviewed the principals of the permeation process for hydrogen isotopes.) The expression which relates permeation to the operating parameters is given by Eq. 5.

$$\phi = \left(\frac{A}{X} \right) (\sqrt{P_H} - \sqrt{P_L}) B \exp (-Q_p / RT) \quad (5)$$

where ϕ is the flux of hydrogen through the interface, A and X are the area and thickness of the interface,

Table III. Parameters for the Permeability, ϕ , of Protium^a Through the Reference Design Construction Materials

$$\phi = B \exp (-Q_p / RT)$$

Material	B (g cm/cm ² day Torr ^{1/2})	Q_p (cal/mole)	Reference
PE-16	0.65×10^{-2}	16,100	(b)
Incoloy 800	0.65×10^{-2}	16,100	(b)
Copper	3.34×10^{-11}	20,000	15

^aValues of ϕ for D_2 and T_2 were obtained by multiplying the values for H_2 by the $\sqrt{2}$ and the $\sqrt{3}$ respectively.

^bThe values of ϕ for hydrogen in PE-16 and Incoloy 800 are not expected to be significantly different from those for 304-SS based on previous studies¹⁶ of the effect of composition on permeability of hydrogen in Fe-Cr-Ni Alloys.

P_H and P_L are the hydrogen pressures on each side of the interface, ($P_H > P_L$), B and Q_p are parameters related to the material under consideration, and R and T are the Boltzmann factor and absolute temperature. Values of B and Q_p for the permeation of protium through materials¹⁵ used in the reference design models are listed in Table III. In most cases only limited data are available for deuterium and tritium permeation through these materials, hence we have (of necessity) estimated the deuterium and tritium permeation rates

by using the values of B and C_1 for protium and multiplying the result obtained via Eq. 8 by the square root of the mass of the hydrogen isotope of interest.

The properties above, particularly the permeation rates for the hydrogen isotopes, are strongly affected by the nature of the hydrogen bearing species at various interfaces. Several species, notably H_2 , D_2 , T_2 , HF , DF , and TF , have already been identified as being important in typical blanket systems. The extent to which other species, such as HD , HT , and DT , are formed is, of course, fixed by the concentrations of H_2 , D_2 , and T_2 . In the case of the reference design models considered herein, a number of other hydrogen bearing species will also be important. For instance, the circulating helium purge flow system and the helium coolant loop of the PPL-Tokamak design will contain some amount of molecular oxygen, O_2 . Because of the conditions that will exist with regard to temperature and oxidizing potential in these circuits, most of the molecular hydrogen entering them will react with the oxygen present to form water. (Johnson and Watson have considered ways of using this oxidation to advantage in creating barriers to tritium migration.)

For the purposes of our calculational model we will assume that the following equilibria are the only ones of any consequence.

The relationships between the partial pressures, p , of the species involved in these equilibria and the equilibrium constants, k_1 , k_2 , and k_3 are given by Eqs. 6 through 8.

$$k_1 = \frac{p_{HT}^2}{p_{H_2} \cdot p_{T_2}} \quad (6)$$

$$k_2 = \frac{p_{H_2O}}{p_{H_2} \cdot p_{O_2}^{1/2}} \quad (7)$$

$$k_3 = \frac{p_{HTO} \cdot p_{H_2}}{p_{H_2O} \cdot p_{HT}} \quad (8)$$

Existing literature data¹⁶⁻¹⁸ on the temperature dependence of k_1 , k_2 , and k_3 were analyzed by a least squares method to give the best fit to Eq. 9.

$$\ln(k_i) = A_i + B_i \ln(T) + C_i/T \quad (9)$$

The values for the fitting parameters (the A_i 's, B_i 's and C_i 's) obtained in this analysis are given in Table IV.

Calculations of the constants k_1 , k_2 , and k_3 over the operating temperature range of the reference design models $\leq 650^\circ\text{C}$, have revealed several interesting features. For example, by rearranging Eqs. 7 and 8 to give the relationships in Eqs. 10 and 11

$$p_{O_2}^{1/2} \cdot k_2 = \frac{p_{H_2O}}{p_{H_2}} \quad (10)$$

[†]Since B is in weight units, ϕ for H_2 must be multiplied by $2/\sqrt{2} = \sqrt{2}$ for D_2 and $3/\sqrt{3} = \sqrt{3}$ for T_2 .

$$p_{O_2}^{1/2} \cdot k_2 \cdot k_3 = \frac{p_{HTO}}{p_{HT}} \quad (11)$$

it is possible to calculate the relative amounts of water and molecular hydrogen as a function of the oxygen partial pressure. At temperatures between 200 and 700°C the value of k_2 is greater than $10^{12} \text{ Torr}^{-1/2}$ and k_3 has a unitless value in the range from 1.0 to 5.0. If an O_2 partial pressure of 10^{-2} Torr in the helium stream is acceptable from a materials point of view, then according to Eqs. 10 and 11 the predominant hydrogen bearing species in the helium will be water (H_2O , HTO , or T_2O) by a factor of more than 10^{11} . The only limitation imposed on this process is the extent to which the kinetics for water formation can be made favorable. To this end it should be possible to incorporate reaction initiators (e.g., spark discharge, hot filament, catalytic surface) in the helium circuits to

Table IV. Parameters for the Equilibrium Constants k_1 , k_2 , and k_3 in Eqs. 6 through 8
 $\ln(k_i) = A_i + B_i \ln(T) + C_i/T$

Equilibrium Constant	Fitting Parameters		
	A_i	B_i	C_i
k_1^a (unitless)	1.2787	0.0265	-141.34
k_2^b ($\text{Torr}^{-1/2}$)	-3.7675	0	30,428
k_3^c (unitless)	-2.4292	0.2920	774.82

^a Obtained by least squares fitting of the data of Ref. 16

^b Obtained by least squares fitting of the data of Ref. 17

^c Reference 18

assist the kinetics.

The net result of these water-producing reactions is that the partial pressure of the molecular forms of hydrogen in the helium carrying zones may be reduced to a value many orders of magnitude lower than in adjacent systems.¹⁹ Therefore, the primary helium coolant loop can act as a buffer region which inhibits the migration of protium (from the steam generators) to the blanket, and of tritium (from the blanket) to the steam system. The helium purge flow system will act in a similar way to prevent migration of tritium beyond the immediate containment shell of the equipment under purge. Since, this purge will undoubtedly operate at a lower temperature than the primary coolant helium the formation of water will be even more favorable thermodynamically. In order to take advantage of the hydrogen-trapping capability of low levels of oxygen in the helium circuits, a processing scheme has been devised^{1,19} to (i) remove and collect the HTO and H_2O , (ii) recover and separate the hydrogen isotopes, and (iii) recycle the tritium to the fueling system.

^{*} Observe that it is not necessary to consider the reaction

because the concentration of T_2O is fixed by Eqs. 6 through 8.

Philosophy and Description of the Computational Model

In our study of the hydrogen isotope distribution and migration characteristics of the reference design models we have treated ^{1,2} those primary reactor systems which come in contact with the most significant quantities of tritium, i.e., the primary fuel loop, the blanket, and the heat transfer and energy conversion circuits. The hydrogen isotope inventories and migration rates for these subsystems can be determined on an individual basis and then summed to give an estimate of the overall hydrogen isotope profile for the entire plant.

For the purposes of analytical simplicity each subsystem can be divided into N isothermal zones in such a way that the temperature drop, ΔT , across each zone is relatively small. (A ΔT of about 20°C was used in most of the calculations.) The fractional area of a particular subsystem in each temperature zone is established by the temperature profile over the entire subsystem. In the calculational model the subsystems are considered to fall into one of two categories--isobaric or interfacial. The isobaric systems are those throughout which the chemical activities and hence the partial pressures of the hydrogen bearing species are constant. The interfacial subsystems are those which support a hydrogen concentration gradient and consequently a net flow of hydrogen through the interface.

In all cases we assume that the partial pressure drop across the interface is greater than one hundred fold (i.e. $P_H/P_L > 100$). We have also assumed that the amount of dissolved hydrogen, deuterium, or tritium in the structural material of an interfacial subsystem is approximately one-half that calculated by Eq. 1.

In order to calculate the amount of dissolved hydrogen, $S(I)$, in the i^{th} temperature zone we use Eq. 12

$$S(I) = [0.5\sqrt{P_H} + \sqrt{P_L}] K_s^0 [A F(I) X(I) d] \exp(Q_s/RT(I)) \quad (12)$$

where

A = the total interface area,

$F(I)$ = the fraction of the total interface area in the i^{th} zone,

$X(I)$ = the thickness of the interface in the i^{th} zone,

d = density of the interface material,

$T(I)$ = the absolute temperature of the i^{th} zone, and K_s^0 and Q_s are as given in Table I.

The rate of hydrogen permeation, $PERM(I)$, through the interface in the i^{th} temperature zone is determined using Eq. 13

$$PERM(I) = \frac{A F(I)}{X(I)} (\sqrt{P_H} - \sqrt{P_L}) B \exp(-Q_p/RT(I)) \quad (13)$$

where the parameters on the right hand side are the same as those defined for Eqs. 5 and 12 and in Table III. A computer program has been written which calculates the $S(I)$'s and $PERM(I)$'s and then sums them over the N temperature zones to give the total amount of dissolved hydrogen, C , and the permeation rate Φ , for the subsystem being investigated.

[†]The average concentration in the interfacial material, C_{AVG} , is approximately given by

$$C_{AVG} = 0.5(\sqrt{P_H} + \sqrt{P_L}) K_s^0$$

$$C = \sum_{I=1}^N S(I) \quad (14)$$

$$\Phi = \sum_{I=1}^N PERM(I) \quad (15)$$

Application of the Computational Model to the Reference Designs

For the blanket and heat transfer systems of both reference designs we have assumed that only protium and tritium are present in significant amounts. The predominant hydrogen bearing species in the blanket are assumed to be H_2 and T_2 and in the case of the PPL-Tokamak HF and TF as well. Tritium is of course bred in the blanket. The important sources of protium to the blanket are considered to be (n,p) reactions and back permeation from the steam system. Because of the isotopic swamping effects which should occur in the steam system only H_2 , H_2O and HTO are considered to be important species there. As we showed above, the predominant hydrogen bearing species in the helium streams (coolant or purge) are H_2O and HTO with trace amounts of H_2 , HT , and T_2 . The primary fuel loops of both designs are composed almost exclusively of D_2 , DT , and T_2 , with very little H_2 , HT , or HD , because the steady state protium level of the fuel stream is of necessity very low on a relative basis^{1,2}.

As was mentioned previously the hydrogen isotope partial pressure drop across any interfacial barrier is over two orders of magnitude (i.e. $P_H/P_L \geq 10^2$). The driving forces which maintain these pressure drops are as follows: As a result of the fueling and burning characteristics of the reference designs the partial pressures of deuterium and tritium species on the plasma side of the vacuum wall will stay in the range 10^{-3} to 10^{-5} Torr, while the partial pressures of protium bearing species will be considerably lower. In the case of the PPL-Tokamak design the blanket fluid ($LiBeF_4$) will be processed by a gas sparging method¹ so that the partial pressure of the tritium bearing species T_2 and HT is $< 10^{-2}$ Torr. According to Grimes and Canfor²¹ this leads to the requirement that the processor also maintain the HF and TF pressures in the blanket at levels < 1 Torr. As a result of Eq. 11 the partial pressures of HT and T_2 in the helium coolant of the PPL-Tokamak¹ could be reduced to levels $< 10^{-12}$ Torr by maintaining the O_2 partial pressure in the helium at 10^{-2} Torr and removing the H_2O and HTO at a fast enough rate so that their partial pressures in the helium do not exceed 10^{-2} Torr.

For the LASL/AVL 6-pinch design, the molten salt contactor system² used to process the lithium blanket is expected to reduce the partial pressure of tritium-bearing species to $< 10^{-9}$ Torr. In this latter design no provision has been made for tritium removal from the sodium secondary coolant. As a result the chemical activity of tritium-bearing species in the sodium would be expected to reach but not exceed the partial pressure levels of those species in the lithium blanket.

For all of the subsystems included in the primary loop processing system it is assumed that the partial pressures of HT and T_2 just outside the primary containment of each subsystem are maintained at levels $< 10^{-15}$ Torr by the helium purge flow system in accord with Eqs. 10 and 11.

Results of the Distribution and Leakage Calculations

It is not possible to provide herein a detailed discussion of the results of all calculations performed for the two reference design studies; hence only the most significant tritium leakage pathways will be considered. The first round of calculations for each design showed quite conclusively that the heat transfer system (HTS) represented the major tritium loss route¹. The principal components of the HTS, i.e. the heat exchange walls, are the only subsystems of a fusion power plant that are not amenable to disengagement from the environment using a relatively impermeable ceramic barrier, a purge jacket, or a combination of both. Since there is continuous leakage of water from the steam system to the environment, the tritium loss rate to the environment via the steam system will eventually equal the rate of tritium leakage into the steam during normal operation of the plant. This will be true regardless of the rate of loss of steam (or water), and it follows that the tritium inventory in the water will be inversely proportional to this fluid loss-rate.

The results of mapping calculations on the LiBeF_3 /helium/steam HTS for the PPL-Tokamak design and the lithium/sodium/steam HTS for the LASL/ANL Theta-Pinch design are summarized in Table V and in Figures 1 and 2. (Additional details of the calculations are given in the reference design reports^{1,2}.)

Table V. Summary of Heat Exchanger Characteristics for the Two Reference Design Models

	PPL-Tokamak	LASL/ANL Theta-Pinch
Primary Heat Exchanger	LiBeF_3/He	Li/Na
Area (cm^2)	2.2×10^8	3.9×10^8
Thickness (cm)	0.29	0.2
Temperature Range ($^\circ\text{C}$)	360-663	410-541
Secondary Heat Exchanger	He/Steam	Na/Steam
Area (cm^2)	8.4×10^8	4×10^8
Thickness ^a (cm)	0.38-0.47	0.2
Temperature Range ($^\circ\text{C}$)	320-595	349-521
Tritium Losses to Steam (Ci/day) ^b		
With Copper Layer ^c	~ 1	~ 6
Without Copper Layer ^c	~ 6	~ 30
Total Thermal Power	5300 MWt	12,000 MWt

^aWithout copper layer

^bOne Curie of tritium $\approx 10^{-4}$ gram of tritium

^cThickness of copper layer = 0.25 cm.

Calculations on both designs were carried out with the option of using either ordinary Incoloy-800 or a laminated metal composite as the material of construction for the secondary (steam) heat exchanger.

[†]This was not surprising since numerous earlier studies had reached the same conclusion (e.g. see references 3, 13b, 19, and 20).

The composite was assumed to consist of an 0.25 cm thick layer of copper sandwiched between two layers of Incoloy-800. The purpose of using the copper is to provide a limiter to the tritium flow through the secondary (steam) heat exchanger wall. The copper must of course be tightly clad to the Incoloy-800 at both interfaces so that there is minimal impedance to heat transfer and to prevent streaming of tritium across the interface surfaces in the event that the copper layer develops holes. Copper is not necessarily the best material to use in this application (i.e. materials like tungsten, molybdenum, and gold are less permeable to hydrogen) but there is demonstrated experience²¹ with stainless steel/copper composites in terms of bend formation, mechanical integrity, and temperature stability. Furthermore, results of calculations performed throughout the design studies have indicated that copper used in this way could provide upwards of a six-fold decrease in tritium permeation without having a significant effect on heat transfer.

The tritium loss rates to the steam as summarized in Table V for the four cases considered are all under the guidelines of ~ 3 Ci/day/1000 MWt advanced by Steiner and Fraas² for a prototypical fusion power plant. It is important to note that firm limits for tritium releases from fusion power plants remain to be established. In the absence of such limits it is our opinion that the guideline used by Steiner and Fraas² is a reasonable one upon which to base design goals for the current generation of fusion power plant models.

The quantities of tritium dissolved in the fluids and structural components which are in direct contact with or interface with the blanket are in general insignificantly small (parts per billion or less) considering the large volumes anticipated for these media. There is, however, one exception to this generalization. Because liquid lithium has such a tremendous capacity to dissolve hydrogen (relative to most other materials), the lithium blanket of the LASL/ANL Theta Pinch design will contain upwards of half a kilogram of dissolved tritium. This is true even at a steady state tritium activity corresponding to 10^{-10} Torr, assuming that this incredibly low activity can be maintained in a large system which is subjected to continuous tritium insertion. The prospects for reaching and maintaining this activity level are discussed in greater detail in reference 2.

Conclusions

Based on the results of the analyses discussed above and of more detailed studies described in the reference design reports^{1,2} for the subject fusion power plant models, a number of instructive observations can be made.

1. The principal tritium leakage pathway to the environment for currently conceived D-T fueled fusion power plants appears to be the heat transfer circuit.
2. Trapping and subsequent recovery of tritium using inert gas purge streams that contain oxygen or comparable oxidizing materials appear to be practical if the kinetics of formation of the oxidized tritium species can be made favorable.
3. The amounts of tritium dissolved in the structural materials that interface with the blanket fluids and the heat transfer systems should present no significant problems from the standpoint of tritium economy, materials failure (due to hydriding), and safety.

FIGURE 1

TRITIUM MAP OF THE HEAT EXCHANGER NETWORK
FOR THE PPL TOKAMAK DESIGN*

FIGURE 2

TRITIUM MAP OF THE HEAT EXCHANGER NETWORK
FOR THE LASL/ANL θ -PINCH

- Bonded metal composites may have some practical applications as tritium permeation barriers.
- High pressure helium and molten salt heat transfer fluids appear to have advantages over liquid metals with regard to tritium recovery and control of tritium migration.
- The most serious tritium handling problem associated with liquid lithium blanketed fusion reactors will be the development of a highly efficient process for recovering tritium from the lithium.

A family of potentially significant effects that could result from isotopic swapping phenomena as described by Hickman²² have not been considered in the calculations performed thus far. The next generation of tritium modeling studies to be carried out as part of the design of a Fusion Engineering Radiation Facility based on the Theta-Pinch concept will attempt to address the possible consequences of isotopic swapping and related effects in greater detail.

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