Tritium Handling, Breeding, and Containment in Two Conceptual Fusion Reactor Designs: UWMAK-II and UWMAK-III

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Tritium Handling, Breeding, and Containment in Two Conceptual Fusion Reactor Designs: UWMAK-II and UWMAK-III

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Abstract

Tritium is an essential component of near term controlled thermonuclear reactor systems. Since tritium is not likely to be available on a large scale at a modest cost, fusion reactor designs must incorporate blanket systems which will be capable of breeding tritium. Because of the radiological activity and capability of assimilation into living tissues, tritium release to the environment must be strictly controlled. The University of Wisconsin has completed three conceptual designs of fusion reactors, UWMAK-I, UWMAK-II, and UWMAK-III. This report discusses the tritium systems for UWMAK-II, a reactor design with a helium cooled solid breeder blanket, and UWMAK-III, a reactor design with a high-temperature liquid breeder blanket. Tritium systems for fueling and recycling, breeding and recovery, and plant containment and control are discussed.

1.0 Introduction

Near-term fusion engineering technologies for the production of useful power will utilize the deuterium-tritium fusion reaction. The natural abundance of deuterium and the existence of a separation technology ensure a sufficient supply of this isotope for the fueling of fusion reactors. On the other hand, it is clear that any practical reactor will have to breed at least as much tritium as it consumes, because the natural abundance of tritium is negligible and the cost of external breeding is high. For the purpose of tritium breeding the reactor blanket contains lithium in some form. The reactor design must include, therefore, in addition to the handling of deuterium and tritium in the plasma fueling and exhaust system, a system for the recovery of the bred tritium. Also, because of the radioactivity of tritium (β^-, $t_{1/2} = 12.3$ yrs), all systems from the fuel injection to the power generating turbines must be designed to minimize the loss of the tritium.

In the Tokamak reactors considered here, the plasma fueling and exhaust systems use very similar tritium recovery procedures. However, owing to the great difference in the operating temperatures of the two reactors, the blanket designs, power cycles and associated tritium recovery systems are not similar (Table 1.0).

Thermodynamic and diffusion rate data on tritium are very limited. For this reason, it was often necessary to utilize relevant information on hydrogen or deuterium and extrapolate over a large temperature range in order to obtain the data required for these studies.

The permeation of tritium through materials is an important component of the studies. Normally one would expect the permeation of tritium through a wall to follow a square root dependence [1] on the tritium pressure. Recent experiments [5,6] indicate however that at low pressures (<1 torr) the dependence is close to first order. These results seem to indicate that inefficient surface desorption is the rate limiting step in the permeation process. If the surfaces are clean and a mechanism is provided for removal of tritium from the downstream surface the square-root dependence of the permeation is probably the best model to use. In other cases, the square-root dependence represents a worst case analysis. In this paper we have consistently used the square-root dependence in the permeation calculations.

2.0 Tritium in the Fuel and Exhaust Systems

A schematic diagram of the fuel supply and exhaust system, Figure 1 indicates that the following subsystems are required: (a) storage of tritium and deuterium, (b) pumping and metering of gaseous tritium and deuterium, (c) insertion of fuel, (d) exhaustion of unburned fuel and plasma chamber products (e) purification of fuel and (f) separation of hydrogen isotopes as necessary.

The required flow rate of fuel supplied and exhausted from the plasma is determined by plasma engineering considerations. In order to produce 5000 MW(th) of power, nearly 0.7kg of tritium and 0.5kg of deuterium are consumed per day. The total fuel supplied to and exhausted from the plasma chamber is much larger, however, because of the small burn up per particle confinement time in the plasma. The fractional burn up in UWMAK-II was calculated to be 4.85%, while the more detailed space dependent codes used for plasma physics calculations in UWMAK-III resulted in a calculated burn up of only 0.8%. As a result, the flow rates are 15 and 74.4kg/day of T_2 and 10 and 49.6kg/day for D_2 in UWMAK-II and UWMAK-III, respectively. Because of the limited supply and high value of tritium, this feed rate for tritium could be sustained for only a limited time. It is necessary, therefore, to exhaust the unburned fuel from the plasma, recycle it and return the tritium to the feed stream as rapidly as possible. Each subsystem must be designed, therefore, to minimize the equilibrium inventory of tritium. Each subsystem will be briefly described and estimates made of its total tritium inventory, Table 2.0.

2.10 Storage

Four types of deuterium-tritium storage systems [7] are feasible, namely: gas storage in pressure tanks, liquid storage in cryogenically-cooled containers, liquid storage as D_2O and T_2O, and solid storage in compounds such as a metal deuteride and tritide. Most industrial experience with hydrogen storage has been in a metal pressure tank. For tritium storage, an austenitic stainless steel tank would be used and the contents kept near atmospheric pressure so that the mechanical strain and tritium embrittlement of the container are minimized and the long-term integrity of the tank assured. The storage of liquid tritium occupies less volume than a low pressure gas and is compatible with the isotopic separation technique but suffers from the costs for refrigeration. Liquid storage as T_2O occupies less volume, and requires a reduction step, but the main concern with this method of storage is related to biological risks if a leak should occur. The seriousness of this potential problem needs further study review. Storage as a metal tritide, such as uranium tritide has been used [8,9] but requires more development as a process technique. A six hour supply of fuel without recycle has been assumed to be the storage inventory in UWMAK-II and UWMAK-III.

2.20 Pumping Requirements

2.21 Fuel Handling.

Independent of the fueling system involved, be it gas blanket, neutral beam injector or solid pellet injection, the pumping and metering of gaseous fuel to the fuel injection systems at near atmospheric pressure requires technological development because of the high flow rates. The pumps must not contaminate the gases with pump fluids and the pumps must be designed so that tritium does not contact any of the normal organic lubricants of a mechanical pump; consequently, presently developed bellows pumps or dry compressors probably will be used but their flow rates will be increased. Serious exhaust pumping and high tritium inventory problems are anticipated because the low burn efficiencies require a high fuel through put.

2.22 Particle Collection

In both UWMAK-II and UWMAK-III, the unburned fuel and product helium ions are directed to divertor plates where their energy is imparted to heat transfer fluids. In UWMAK-II these very energetic particles transfer their kinetic energy to a film of liquid lithium flowing over the divertor plates. At a lithium flow rate of 4.94×10^6 kg/hr a temperature of 325°C will be maintained. The liquid lithium, in fact, serves as a pump for the D and T particles and results in an accumulation of D and T at a rate of 1×10^{-6} atom fraction during each pass of the lithium through the divertor. A tritium extraction system must be provided in this lithium system to recover the tritium. Information on this tritium recovery system will be presented later along with other tritium separation processes.

In UWMAK-III the divertor plates are TZM sandwiches containing 0.1mm layer of immobile lithium for thermal contact. These plates are in turn cooled by liquid sodium pumped at a rate sufficient to maintain a temperature at the surface of the plate no higher than 800°C. The outlet temperature of the sodium coolant is 600°C. The thermolyzed particles are then collected by cryoadsorption pumps.

The exhaustion of the unburned fuel from the plasma chamber represents a formidable task because of the high flow rates and the required low residual pressure of 10^{-5} torr. Many types of vacuum pumps have been suggested for this service, but the cryoadsorption pumps appear to have the best characteristics. The cryoadsorption pumps operate as a batch process, however, so that two sets of pumps are needed, one set on-stream while the second set is being regenerated. The time during which the pumps are on-stream depends upon the minimum time (90 min) required for regeneration, and the design parameters of the pumping system and the purification system which follow. Some trade off must be made. Thus a design which minimizes the on stream time will result in a minimum tritium inventory in the pumping system but maximize the energy requirements because of a large number of heating and cooling cycles per unit time. In UWMAK-II the on-stream period of 4 hours results in a tritium inventory of 83.3g whereas in UWMAK-III the on-stream period of 5 hours results in a cryopump tritium inventory of 15.5kg.

2.30 Purification

The cryopumps will initially contain in addition to the D_2, T_2, DT, and HT molecules, the following impurities: He, H_2, HD, Ar, N_2, O_2 and hydrocarbons. These impurities must be removed before the fuel is returned to the plasma feed system. Possible purification techniques are (a) adsorption followed by desorption of the hydrogen isotopes on an active metal, (b) diffusion of the hydrogen isotopes through palladium tubes and (c) chromatographic stripping. Technique (a) is a slow, batch process, and (b) requires that the gas temperature and pressure be increased and a large amount of palladium tubing be utilized. For these reasons, the chromatographic stripping by the use of molecular sieves cooled to -131°C is utilized [10] for the removal of all impurities except helium.

The off-gas from the purifier will now contain the molecular species D_2, DT, T_2, HD, HT plus some helium, which is produced in the fusion reaction at the rate of 1.0kg/day. Since the maximum desirable level of He in the fuel is about 5 atom per cent, the maximum permissable levels in one day's fueling are 2kg in UWMAK-II and 10kg in UWMAK-III. One kilogram per day of helium is separated in the fuel preparation step, because the D-T mixture has to be liquified to prepare the frozen fuel pellets and the helium gas is simply pumped off. Essentially all the fueling in UWMAK-II and UWMAK-III is done by injection of solid frozen DT pellets. In addition, very small amounts of D_2 and T_2 are used in UWMAK-II for neutral beam injection. Here the D_2 is supplied from an external source and the T_2 required for neutral beam injection is supplied from the blanket recovery system. UWMAK-III does not use neutral beam injection.

In addition, about 1g. of protium is produced per day from nuclear reactions in the structural first wall. The maximum desirable level is about 2kg of protium in one day's fueling for UWMAK-II or UWMAK-III. To maintain an acceptable protium level in the fuel, 0.1% of the recycled fuel stream from the purifier is passed to a cryogenic distillation unit [11,12,13] and the HD is removed.

3.0 Tritium Breeding and Recovery

3.10 Breeding Material in UWMAK-II

The choice of a breeder material for UWMAK-II was influenced by some of the problems found in designing a stainless steel reactor cooled with a liquid lithium breeder such as UWMAK-I [14]. In particular, tritium recovery from liquid lithium at very low concentrations is difficult owing to the high stability of the lithium tritide solution; consequently a relatively high inventory of tritium, 8.7kg, existed in the lithium blanket. Corrosion of 316 stainless steel by lithium is anticipated at 500°, and the MHD pressure drop associated with pumping conducting fluids across magnetic flux lines requires a judicious design of the fluid flow channels or the development and utilization of non-conducting tubing. For these reasons and others such as the fire hazard, alternate breeder materials were considered for UWMAK-II. The use of 316 stainless steel as the structural material virtually eliminated molten salt as a coolant; so the choice narrowed to a solid breeder with helium as a coolant.

Among the solid breeder compounds, lithium compounds of aluminum oxide and silicon dioxide appeared to be the most practical to prepare reasonably pure and free of water on a large scale. The orthosilicate Li_4SiO_4 has a high lithium atom density, but melts by a peritectic reaction at 1255°C, with the rapid vaporization of the lithium oxide at this temperature [15]. Also, as the lithium in the orthosilicate transforms in the neutron nuclear reaction, a eutectic liquid between the ortho and metasilicate, Li_2SiO_3, will form at 1024°C; consequently the useful temperature limit of the orthosilicate appears to be <1000°C.

The Li_2O-Al_2O_3 phase diagram [16] shows only two compounds $LiAlO_2$ and $LiAl_5O_8$. The melting point of the $LiAlO_2$ is reported to lie between 1610-1700°C, and a eutectic liquid is reported at ∿1670°C between the compounds Li_2AlO_2 and $LiAl_5O_8$ which will form as the lithium is transmuted by the neutron nuclear reaction. The appearance of this eutectic and the volatilization of Li_2O limits the usefulness of this compound to <1400°C and the possibility of sintering [17] probably reduces the practical temperature by another 100°C.

In addition to thermal stability, the breeding material must show a reasonably low tritium inventory. Regardless of which compound is used, a steady-state will be approached in which the rate of tritium release equals the generation rate. Therefore, the kinetic and equilibrium factors which affect this steady-state condition must be considered. The solubility of tritium in the compound is determined by the thermodynamic equilibrium constant and the tritium partial pressure in the extraction system. On the other hand, the rate of approach to the steady-state condition is dependent upon kinetic considerations, such as diffusion of tritium in the compound. Both diffusion and solubility of tritium in $LiAlO_2$ will be considered.

Diffusion data for tritium bred in $LiAlO_2$ does not exist. In fact, little data on diffusion of hydrogen species in oxides has been reported; see Figure 2. for existing data [18-21]. Measurements of the diffusion of hydrogen through similar material, Al_2O_3, indicate diffusion values at 900°C, for instance, from 10^{-7} to $5 \times 10^{-9}\ cm^2/sec$ [18,19]. Preliminary diffusion values estimated [22] from the release rate of tritium from neutron irradiated $LiAlO_2$ powders are reported to be of the same magnitude. This indicated low diffusion rate for tritium in $LiAlO_2$ makes it necessary to keep the tritium diffusion path lengths very short in order to minimize the tritium inventory in the breeder. For instance, for an average breeder the temperature of 900°C the tritium inventory **I can be calculated for the** release of a solute from a sphere by the relationship [23]

$$I = \frac{1}{15} \frac{T^\circ S^2}{D}$$

where T° = tritium generation rate = 10^{-2}g/sec

S = radius of the sphere

D = diffusion coefficient $\approx 10^{-8}\text{cm}^2$/sec.

Under these conditions, the inventory of tritium in the blanket would be 670g for a sphere with a radius of 1mm.

Because the temperature in the tubes containing $LiAlO_2$ is not uniform, this effect on the tritium inventory must be considered. The radial temperature profile in these tubes is approximately parabolic from 600°C at the surface to a maximum in the center of 1097°C. For an optimum breeding ratio, $LiAlO_2$ should be 90% of theoretical density. The fabrication of $LiAlO_2$ breeder material of 90% theoretical density with short diffusion path lengths requires considerable development. In order to achieve 90% packing density, three sizes of spheres might be used whose diameters are in the ratio 45:6:1. A mixture of spherical particles of diameters 300μm (65%), 40μm (24%), and 7μm (11%) was used for the design. Based upon an intermediate value for the diffusion of hydrogen in Al_2O_3 (Fig. 2) being similar to the diffusion of tritium in $LiAlO_2$ and an assumed radial temperature profile, a tritium inventory of ∿750g is calculated due to slow diffusion in these spheres. The performance of a breeder composed of such a mixture of spheres is difficult to evaluate because sintering of this ceramic will become significantly rapid above 1300°C which is nearly the predicted temperature near the center-line of each breeder capsule; consequently, the spheres in this hotter region would fuse together and the tritium diffusion path length will increase. It should be noted that the sintering occurs at the temperature where the diffusion coefficient is highest. Thus, if sintering is not extensive, the tritium inventory should not be exceptionally increased.

The equilibrium solubility of tritium in the ceramic breeder is difficult to estimate because the solubility of hydrogen has been measured over a wide range of temperature and pressure for only one oxide, ZnO [24]. The observed solubility relationship was found, at the low hydrogen mole fractions and low hydrogen pressures, to follow a one-half power dependence as might be expected for the dissociation of a diatomic gas. Based upon this relationship and a residual hydrogen gas pressure of $<10^{-4}$ torr, the hydrogen solubility in $LiAlO_2$ was estimated at 900°C to be 7grams for the total reactor. Because of the chemical differences, the solubility of tritium in $LiAlO_2$ should be less than that of hydrogen in ZnO [25]. Therefore, of the total inventory of tritium in $LiAlO_2$ the solubility is much less important than the retention resulting from the slow diffusion of tritium.

The chemical species of tritium released from irradiated $LiAlO_2$ must be carefully reexamined because preliminary reports present conflicting information. In one report [26], the tritium was recovered mainly as tritiated water; however, this material was not chemically pure because a large amount of carbon dioxide was also released. For another material of apparently higher purity [27], a larger fraction of tritium was recovered as molecular tritium than tritiated water. On the basis of the nuclear reaction, and subsequent dihydration reaction

$$\begin{aligned} {}^{1}_{0}n + {}^{6}_{3}LiAlO_2 \rightarrow {}& {}^{3}_{1}TAlO_2 + {}^{4}_{2}He \\ & \qquad\downarrow \\ & 1/2\ T_2O + 1/2\ Al_2O_3 \end{aligned}$$

it would appear that T_2O should be the reaction in order to maintain the charge balance. In any event, the tritium extraction system must be capable of processing both molecular tritium and tritiated water.

3.11 Cooling of the Blanket and the Extraction of Tritium

The utilization of the finely powdered ceramic, $LiAlO_2$, has some major impacts upon the designs of the blanket, the heat transfer system, and the tritium extraction system. A schematic flow diagram of these systems is shown in Fig. 3. The ceramic breeder would need to be enclosed in stainless steel tubes approximately 3.5cm diameter with a wall thickness of 0.75mm and a length of 30cm. The ceramic breeder must be contained in the blanket because the pressure differential of the helium coolant across a packed bed of this fine powder would be excessive. These tubes would be closed at the end facing the plasma and connected by small tubes, 0.9cm diameter, to a gas plenum attached to a tritium extraction system external to the reactor.

The tritium or tritiated water released by the breeder material could be removed either in a stream of slowly flowing helium or by evacuation of the tubes with cryogenically-cooled adsorption pumps. The residual tritium partial pressure inside the breeder tubes has been estimated to be approximately 10^{-4}torr. The external surfaces of the tubes containing the breeder material are cooled by the flow of the helium coolant system. Because of the large surface area of these tubes ($3.3 \times 10^{8} cm^{2}$) thin walls (0.75mm) and high average wall temperatures, ∿600°C, nearly 48g of tritium will permeate through these tube walls into the helium coolant. To avoid the loss of tritium by permeation through the intermediate heat exchangers to the steam power system and finally to the environment, a tritium extraction system is placed in the helium coolant stream. An oxygen partial pressure of 10^{-2}torr is maintained in the helium so that with the aid of a catalyst, the tritium is rapidly oxidized to T_2O. The T_2O partial pressure is maintained at 10^{-3}torr by diverting a small side stream of the helium (0.7%) through a molecular sieve dissicant maintained at 95°C. Under these circumstances, the tritium partial pressure in equilibrium with T_2O is reduced to nearly 10^{-14}torr. The low oxygen partial pressure should not be deleterious to the stainless steel reactor system [28].

An intermediate system has been inserted between helium of the breeder loop and the steam cycle of the generator loop for safety reasons and thermal storage. Liquid sodium is used in this intermediate heat transfer system. The partial pressure of tritium in the sodium should never be greater than it is in the helium, 10^{-14}torr; however, for several reasons such as leakage, corrosion, etc., the tritium partial pressure in the helium may rise causing a corresponding rise in the tritium partial pressure in the sodium which would in turn increase the permeation of tritium to the steam generator. In order to limit the concentration of tritium in the sodium, a tritium extraction system is inserted into the sodium system. This tritium extraction utilizes the high affinity of the active metal yttrium for hydrogen; consequently, the chemical equilibrium [29], $Y + T_2 = YT_2$ is predicted to maintain a tritium partial pressure of only 1.5×10^{-11}torr at 280°C; this is equivalent to a tritium concentration in the sodium of only about 5×10^{-5}ppm. At this concentration, only 1.2 Ci/day of tritium is predicted to permeate into the steam. Tritium in the steam system is considered to be lost to the environment.

An additional hydrogen isotope permeation reaction must be considered, also in a steam power plant system, namely the diffusion of hydrogen from the steam into the sodium [30]. The hydrogen in the steam exists because of the thermal decomposition of water. At the temperature of the steam (500°C) the equilibrium constant favors the formation of water; however, the steam pressure in the superheater is high, 3690 psia, so that the hydrogen partial pressure may be as large as 5×10^{-5}torr. The independent movement of hydrogen isotopes through a metal membrane has been confirmed; consequently, hydrogen will permeate through the heat exchanger from the steam to the sodium at the same time that tritium is diffusing in the opposite direction. On the basis of the hydrogen partial pressure, the leak rate of H_2 from the steam to the sodium is calculated to be 3.18×10^{18}atoms/sec $\left(S^{T}_{Na}\right)$. The leak rate of tritium from the helium into the sodium is only 1.8×10^{14}atoms/sec $\left(S^{H}_{Na}\right)$. The yttrium getter in the sodium probably will not significantly distinguish between hydrogen and tritium; consequently the total concentration of hydrogen isotopes in the sodium will be 5×10^{-5}ppm and the fraction of this total which is tritium will be in the ratio S^{T}_{Na}/S^{H}_{Na}. The partial pressure of tritium will be correspondingly reduced so that the permeation of tritium into the steam system may be as low as 5×10^{-5} Ci/day.

3.20 Breeding Material in UWMAK-III

UWMAK-III is a conceptual fusion reactor designed to operate at high temperatures in which the first wall is the molybdenum alloy TZM, and the coolant, liquid lithium, is also the breeder. The operating temperature will be 980°C. Because a stationary solid breeder would develop much higher internal temperatures (about 1500°C maximum) and sintering would occur, liquid lithium was chosen.

3.21 Tritium Pathways

The breeding blanket in UWMAK-III comprises only the outer portion of the torus. The inner portion of the torus is cooled by helium which flows directly to a helium turbine. The outer blanket coolant, Li, flows to an intermediate heat exchanger, transferring its thermal energy to a sodium loop. The intermediate sodium is then passed to a heat exchanger which transfers its thermal energy to a helium gas stream. The helium then drives a turbine. It is necessary to control the tritium in all the coolant fluids shown in Figure 4.

The largest amount of tritium is handled in the fueling and reprocessing cycle. The temperatures of the piping and containment vessels for this system, except within the torus, are very low and the tritium loss is minimal as long as the many joints and valves perform satisfactorily. However, the outer blanket power cycle operates at a very high temperature. Here the tritium bred in the lithium diffuses through the heat exchanger walls to the sodium and then to the helium. The pressure of tritium over the sodium is maintained at 1×10^{-12}atm, which will ultimately become the pressure of tritium in the helium and in the building. Losses to the building environment at this point should be low (<1 Curie/day). In turn the tritium in the lithium must be maintained at a concentration such that diffusion into the sodium does not result in a tritium loss rate that exceeds the sodium cleanup system capacity. The tritium concentration in the lithium must also be low enough to prevent substantial leakage through the piping and valves.

We have designed a system which meets the above requirements. This system operates with a steady state tritium inventory in the lithium of 1kg (2.08ppm) and a leak rate into the sodium of 1256 Curies/day. The sodium cleanup system will process this amount of tritium and maintain a tritium pressure over the sodium of 7.6×10^{-10} torr.

3.22 Recovery of Tritium from Heat Transfer Fluids

The primary energy deposition and tritium production site is in the outer liquid lithium blanket [31]. Here the neutrons react with lithium to produce tritium and helium. The total mass of lithium in the blanket is 4.8×10^{5}kg which at 980°C has a density of 0.44g/cc and thus occupies a volume of 964m^3. An additional reservoir [32] of sufficient volume to contain all the liquid lithium is available in the event of a plant shutdown. This reservoir has a wall thickness of 1.43cm and is heated to ensure that the lithium remains liquid. Tritium diffusion through the walls is tolerably low under these conditions.

The liquid lithium is circulated through the system at a rate of 7.12×10^{6}kg/hr. Five per cent of the lithium is diverted to a niobium "window" [33] for the extraction of tritium. Niobium is particularly well suited for this purpose as it has a high permeability to hydrogen at these temperatures [34] and allows us to reduce the tritium concentration of the lithium to a level compatible with the subsequent tritium losses to the environment. The niobium "window" is of heat exchanger design [31] with liquid lithium flowing through niobium tubes of wall thickness 0.38mm and helium containing 10^{-2}torr oxygen flowing past the tubes at a flow rate of $4.1\text{x}10^{3}\ell$/hr. The total surface area will be $1.47\text{x}10^{7}\text{cm}^{2}$. Calculations have shown that diffusion of tritium from the bulk of the lithium to the wall surface will not be the rate determining step in the permeation process. The oxygen in the helium should rapidly react with the tritium atoms on the catalytic metal surface to yield T_2O. The five per cent bypass of lithium into the "window" system reduces the total tritium concentration from 2.08ppm to 2.04ppm and removes 0.666kg of tritium in a 24 hour period.

The fabrication of the desired window will require considerable development. The oxygen level in the niobium must be less than 100ppm or the lithium will erode the surface and preferentially attack grain boundaries [35]. Also carbon transport from the TZM will occur, [36] causing embrittlement of the niobium. In addition, at 980°C niobium is subject to attack by oxygen and therefore some protective barrier is required. Palladium is the barrier of choice because it has a high hydrogen permeability [33,34]. A palladium thickness of 0.001cm deposited by electrodeposition [35,37] will give a coherent layer of suitable durability. This will require a total of 170kg of palladium for the calculated surface area of the window.

Even though palladium oxide is thermodynamically unstable and the oxygen concentration is low, some transport of palladium oxide from the window surface will occur. However, this is not expected to be a serious problem as extrapolating preliminary data [38] indicates a lifetime for the palladium layer of 28 years; essentially the same as the plant lifetime. It is also conceivable that counter diffusion of tritium through the niobium metal may inhibit this process. The mutual solubility of the two metals is quite high [39] and it is expected that the palladium will diffuse into the niobium, though there are no firm data for the rates of interdiffusion. Therefore, we propose to introduce an intermediate yttrium layer to minimize the rates of diffusion so that a practical "window" lifetime can be achieved. Yttrium was chosen because it is among the few metals which do not form solid solutions with niobium [39]. The required yttrium layer thickness is estimated to be 2500Å. This will require a total of 2.0kg of this element for the calculated surface area of the window.

The permeability of yttrium to hydrogen is unknown, although it is expected to be similar to zirconium. We estimate the permeability of the 2500Å yttrium layer to be the same as the 0.38mm niobium wall. This estimate was used to calculate the required "window" surface area. Assuming that a "window" can be successfully constructed to these specifications we would expect the lifetime to be 10 years and, therefore, replacement would be required twice during the lifetime of the plant. The design includes additional reserve "window" modules (10% of the capacity) so that a tube failure would not require a plant shutdown but only a switch to a reserve "window" module.

The T_2O recovered from the "window" will be collected on Linde 5A molecular sieves at 95°C. One bed will contain 460kg of dessicant with a capacity of 2.5kg of T_2O. The gaseous T_2O will be regenerated by heating the bed to 200°C. This will then pass over uranium [40,41] metal or an alternate active metal, to reduce the T_2O and T_2 under conditions in which the metal hydride will not form. Again a dual system will be employed so while one reduction bed is operating, the spent bed is being regenerated. To process 2.5kg of T_2O, 13.5kg of uranium will be required.

Other methods for the extraction of tritium from lithium, such as yttrium beds, molten salt extraction, liquid alloy getters and fractional distillation, were considered. The yttrium bed [42,43] will not reduce the tritium level in the lithium to the desired concentration under the temperature and tritium pressures of this design. The extraction of tritium from liquid lithium via liquid metal alloy getters [44] is attractive, although more data on the mutual solubilities of lithium and alloy are required. Similarly, although the extraction of tritium with a molten potassium chloride-lithium chloride mixture [45,46] has shown potential, the utility of this technique will depend upon the development of a suitable method for the recovery of the tritium from the salt mixture. Finally, vapor pressure measurements of deuterium species Li_2D, LiD and D_2 over liquid lithium [47] at 973°C suggests that a fractional distillation technique might be a useful route to separation of hydrogen isotopes from the bulk lithium since Li_2D is reported to become the important gas phase species at elevated temperatures. However, the comparable tritium system species has not been observed at this point.

The liquid lithium-sodium intermediate heat exchangers act as a buffer between the tritium bearing lithium and the helium which will drive the turbines. These heat exchangers consist of TZM tubes with a wall thickness of 0.89mm and a total area of $5.75x10^7cm^2$. Under the conditions of the system the tritium concentration on both sides of the heat exchanger will be $7.6x10^{-10}$torr. The tritium concentration will be reduced by reaction with yttrium metal on the "cold" leg of the heat exchanger. The "cold" sodium will initially pass through a regenerative heat exchange [31] to reduce its temperature to 200°C and then to the yttrium beds containing 0.020kg of 0.10mm yttrium foil. It is calculated that the tritium concentration in the sodium will be reduced by 10% by reaction with yttrium. The exiting sodium will again pass through the regenerative heat exchanger to return the temperature to 590°C before it rejoins the main sodium loop. The yttrium beds are also operated in tandem so that one set is extracting while the other set is being regenerated. Each set contains a total of 0.020kg of yttrium foil. In the regeneration process the yttrium sheets are heated to above 500°C to thermally decompose the yttrium hydride formed. The composition of the yttrium hydride ranges from a low of $YT_{0.8}$ to a maximum of $YT_{1.8}$, so the capacity of the yttrium bed is one mole of tritium per mole of yttrium. However, there will always be a constant tritium inventory in these beds amounting to 0.8 mole of tritium per mole of yttrium. This gives us a tritium inventory in the yttrium beds of 1.0×10^{-4} kg. As the tritum decays to ^{3}He with subsequent formation of bubbles [8] which degrade the mechanical strength of the foil,the lifetime of the beds is estimated to be 5 years.

4.0 Plant Containment and Tritium Control

A review of the plant designs indicates that during operation an appreciable quantity of tritium will be present in the plasma fuel, exhaust system, the breeder blanket, and extraction system. Because tritium is by far the most mobile radioactive constituent in the fusion reactor power plant, it is necessary to design and operate high integrity containment systems so that the tritium release to the environment will be minimal.

The tritium handling systems are composed of a large number of individual components and connecting devices. Tritium permeation and leakage from this multitude of components is inevitable. Based upon the concept of multiple containment systems, nearly all of the tritium leakage can be confined within the reactor building. These containment systems will be described briefly. The tritium which may permeate into the reactor coolant system and exit through the steam condensate has been previously discussed. A very detailed study of tritium containing systems in CTR designs has been carried out by Kabele and Johnson [48].

4.10 Primary Containment System

The primary containment system consists of the multitude of components and pipes which house the tritium-bearing materials. These components will generally be fabricated from materials which have a low permeability for tritium. Connections and joints are preferentially made by welding so that seals are avoided. Stainless steel bellows valves are used whenever possible so that organic gaskets and packings are avoided.

4.20 Secondary Containment System

Certain components of the primary containment system will contain large inventories of tritium or have a high probability for leakage. Each of these components may be contained in large enclosed boxes, similar to laboratory gloveboxes [49-51], from which the tritium will be continuously removed. One type of system which has been useful for this application is the use of a helium atmosphere which continuously circulates through these boxes and subsequently through a bed of molecular sieves cooled to liquid nitrogen temperatures, -196°C. Such a system removes air; hydrogen (including tritium) and water vapor to levels <1ppm (vol.) in the helium. In addition to govebox systems, the high temperatue lithium in UWMAK-III is contained in double-walled insulated piping. A slowly flowing stream of inert gas is processed to remove the tritium that permeates into the space between the pipes.

4.30 Tertiary Containment System

The main building containing the fusion reactor will serve as the tertiary containment system. The building may have a volume of approximately $2.5\times10^{5}m^{3}$ and have concrete walls up to two meters thick for neutron shielding. The interior walls have a steel liner which may be required as a tritium seal, to prevent lithium or sodium reactions with concrete. Because the walls of this building are at ambient temperature, diffusion of tritium will be very slow; consequently the most likely path for tritium escape will be by leakage. In order to prevent leakage into adjacent structures, the reactor building will be operated at a reduced pressure with a regulated quantity of the building atmosphere exhausted to a stack. The effluents to the stack will be scrubbed for tritium removal before the gases are exhausted to the environment.

The principal system for tritium effluent removal from the gaseous streams will consist of catalytic oxidation to form tritiated water and the adsorption of the water on molecular sieve dessicant beds [49-51]. By use of this technique decontamination factors as high as $3.3x10^6$ have been obtained for total tritium levels of 1ppm (vol.). Several of these systems may be used to process the gaseous effluents from air-locks and certain processes which may be tritium contaminated. During routine operation, clean-up of the entire building atmosphere will not be required.

In the event of an accidental release of tritium to the building atmosphere, the exhaust to the stack will be closed and an emergency containment system, detritiation process will be activated. This process utilizes the catalytic oxidation of dilute tritium atmospheres, 0.5ppm (vol.), at a relatively low temperature, 177°C, and adsorption onto a dessicant. Multiple recirculation of the building atmosphere permits the use of a small flow-rate system, compared to the large volume of the containment structure. The decontamination factor can be increased by isotopic dilution in which hydrogen (protium) is introduced with the tritium before the catalytic reactor and the dessicant is pre-saturated with H_2O.

Liquid effluents, consisting chiefly of dilute tritiated water will accumulate from various locations in the plant. These liquids will be collected in holding tanks, monitored and discharged if below RCG concentrations. If discharge of this dilute water is not acceptable, techniques are being developed for the solidification of tritiated water in concrete or polymeric materials [52]. Research studies are being initiated also which may make it economically feasible to remove tritium from water.

Solid wastes contaminated with tritium will accumulate during the reactors operation. Economical and environmental considerations may make it advantageous to decontaminate these items on-site for repair and recycle.

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Table 1.0

Blanket and Power Cycle Characteristics

	UWMAK-II [2,3]	UWMAK-III [4]
Inlet Temperature (°C)	450	630
Outlet Temperature (°C)	650	980
Material of Construction	316 SS	TZM
Breeder	$LiAlO_2$ (solid)	Li (liquid)
Neutron Multiplier	Be	none
Primary Coolant	He	Li (liquid)
Intermediate Coolant	Na	Na
Power Coolant	steam	He

Table 2.0

Steady State Inventory of Tritium (g)

	UWMAK-II	UWMAK-III
Power Cycle		
Breeder Material	760	1000
T-Recycle	350	666
Intermediate Heat Transfer Fluids	0.2	0.36
Divertor		
Coolant	8 (Li)	0.6 (Na)
Recovery (Y Beds)	3625	7.5
Cryopumps	41.7	7750
Reprocessing	41.7	7750
Reserve Storage (6 hrs Fueling)	3750	18,600
Reprocessing and Purification	—	900
	8577	36,675

Figure Captions

Fig. 1. The components of the fueling cycle are indicated. Fuel is injected into the plasma as gas, pellets, or neutral beams. Then the diverter recovers the unburned D&T, which is pumped and processed for refueling.

Fig. 2. The available data for diffusivity D of hydrogen species in oxide ceramics is shown.

Fig. 3. Tritium recovery systems in the UWMAK-II power cycle and breeder blanket are shown. The tritium extraction bed recovers most of the bred tritium from the blanket. Residual tritium is recovered from the helium and sodium coolants. Less than one curie per day leaks through the steam generator.

Fig. 4. The tritium systems in UWMAK-III are shown. Tritium bred in the lithium blanket is recovered with a Nb window. Small amounts of tritium are recovered from the intermediate sodium loop, the sodium divertor coolant, and the inner blanket He coolant.

PLASMA FUEL AND EXHAUST SYSTEM

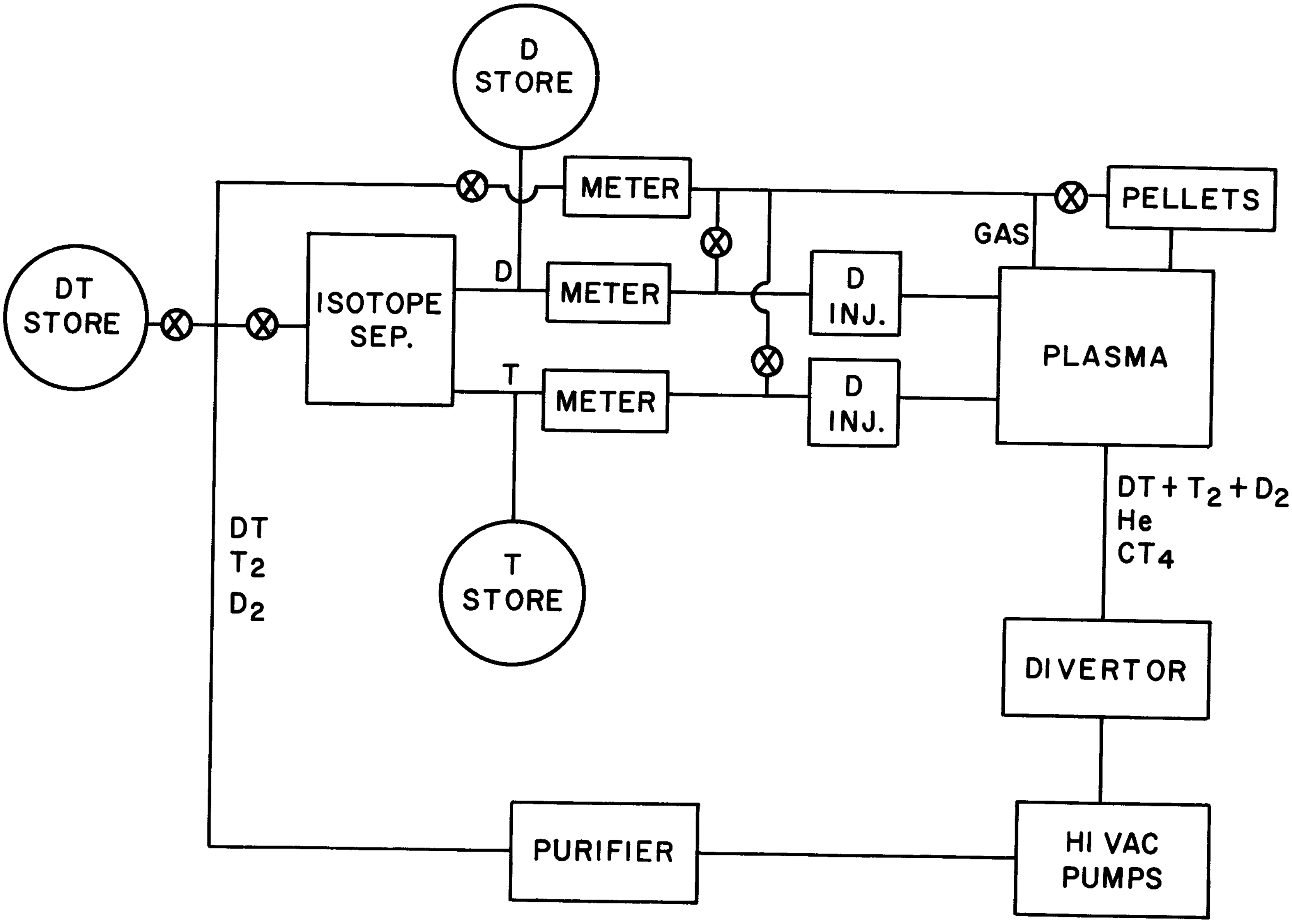

FIGURE 1

DIFFUSION OF TRITIUM OR HYDROGEN
IN CERAMIC MATERIALS

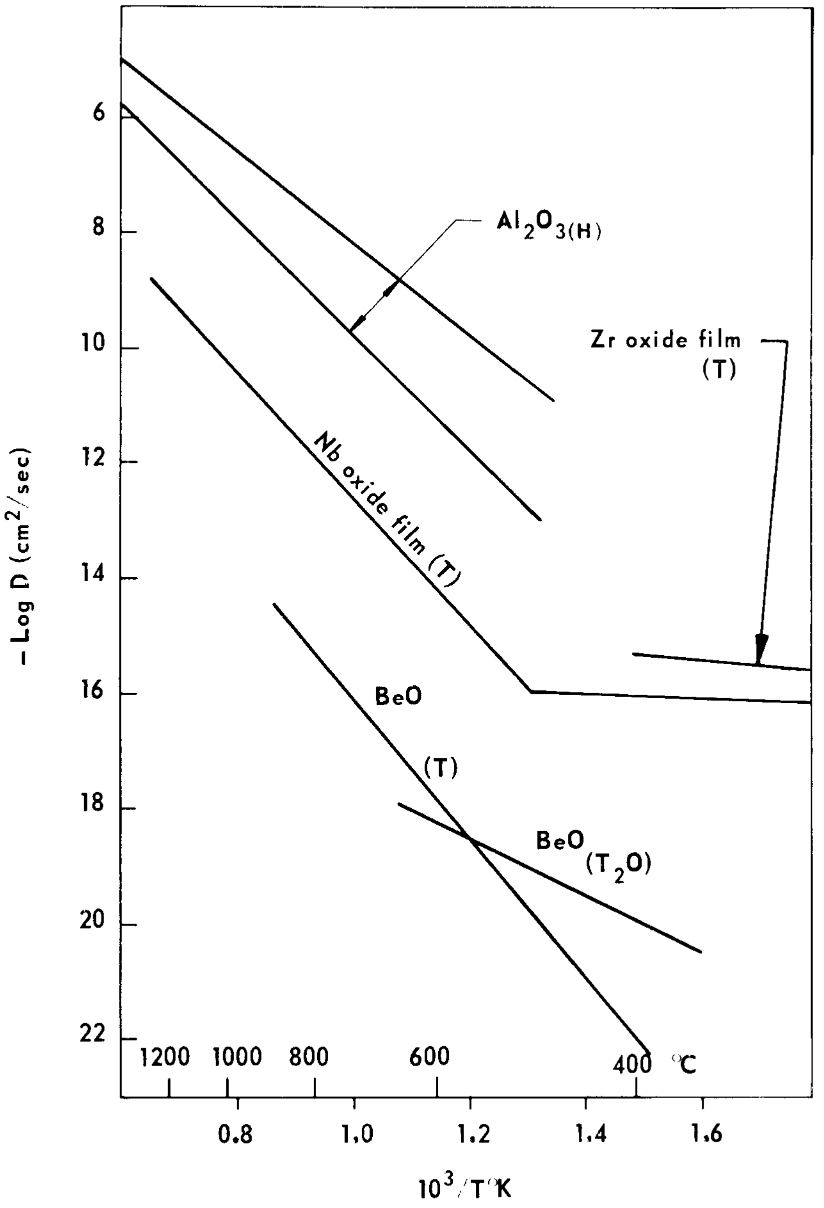

FIGURE 2

COOLANT FLOWS FOR UWMAK-II

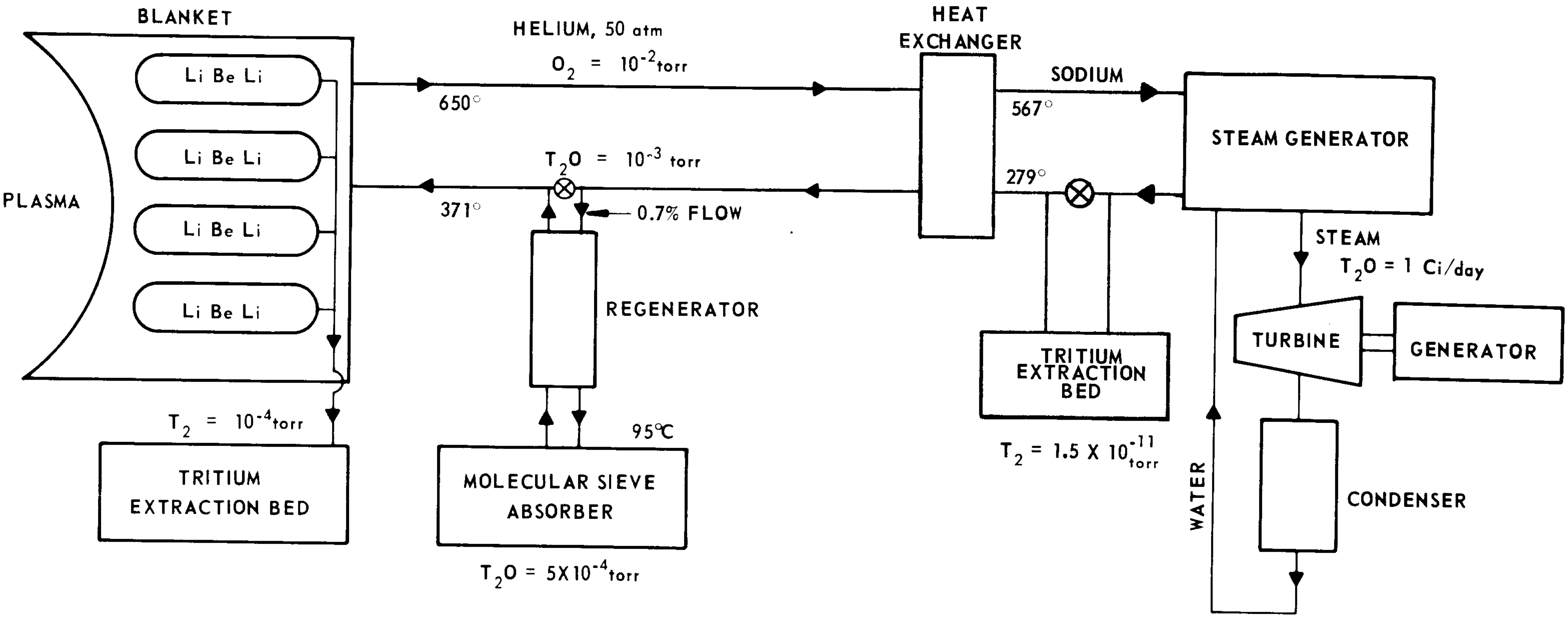

FIGURE 3

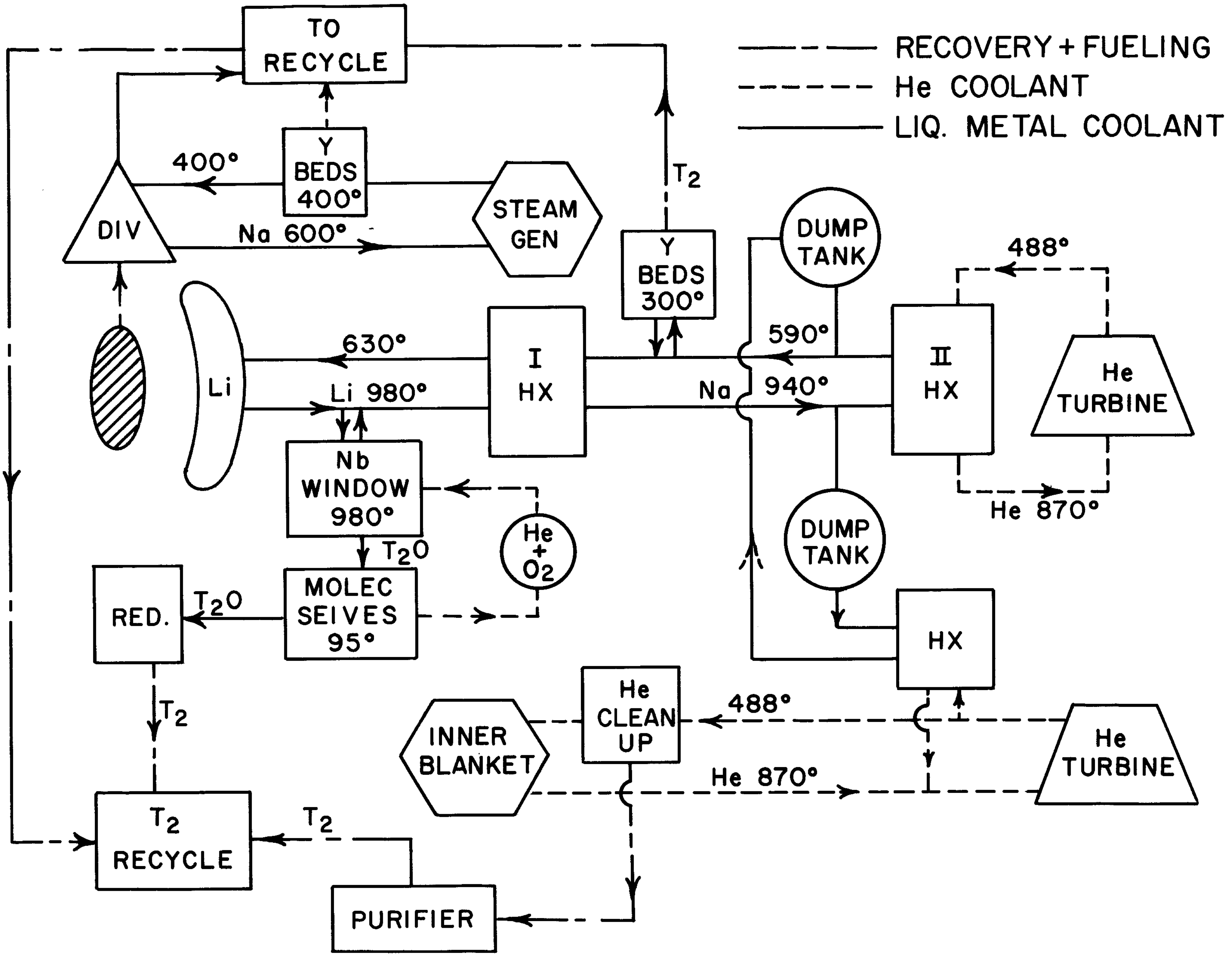
TRITIUM IN UWMAK-III SYSTEM

FIGURE 4