

6. TRITIUM SYSTEM

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6. TRITIUM SYSTEMS

6.1. INTRODUCTION

The design of the ARIES-I tritium system is focused on reducing the tritium inventory and simplifying the design. However, because of the material selection and system configuration, some new problems have been uncovered during this study and they are documented here. Further work, both theoretical and experimental, will be required to address these problems. Because of its small size, the cost of the tritium system will be of secondary importance.

The largest tritium unit in a DT fusion reactor is the plasma-exhaust processing system. The throughput of the ARIES-I plasma exhaust is reasonably small because of the high tritium burn fraction (19.3%). A batch operation unit, such as the molecular sieve or cryogenic pump, usually has large tritium inventory. Therefore, we use a Pd diffuser for separating hydrogen isotopes from the waste, and we use a turbo-molecular pump for pumping the exhaust. Another unit that has large tritium inventory is the cryogenic distillation (CD) unit. In a conventional fuel-exhaust process, a CD unit is required to separate the D and T. In the ARIES-I study, we use the CD only to separate the H from D and T while keeping the D:T ratio at 1. This will reduce the tritium throughput to the CD by a factor of 10. The tritium inventory in the CD unit will be reduced correspondingly. The number of CD columns will also be reduced, from 4 to 1.

The tritium inventory and plasma-driven permeation in the first wall and in the impurity control system are always of concern in a fusion reactor design. This problem may even be more severe here because of the selection of the structural material, silicon carbide (SiC). In SiC, the tritium diffusivity is very low and the solubility is very high, especially at low temperatures [1]. The tritium inventory in the SiC due to the pressure-driven permeation does not appear to be a serious problem because of the slow kinetics on the surface and the low diffusivity. However, the kinetic energy of the plasma particles will drive the tritium into the SiC where it will remain. To minimize the tritium inventory in the SiC, the material temperature is kept above 750 °C. To reduce the tritium permeation to the divertor coolant, the divertor target is made with a tungsten coating on the SiC-composite coolant tubes.

Tritium inventory is also a concern in the breeding blanket. Because of the high blanket operating temperature and the good tritium release characteristics of Li_2ZrO_3 , the

reference breeding material, the tritium inventory in the breeding material is calculated to be only 0.3 g. However, there is the potential of high tritium inventory in the beryllium. Because of the intimate mixing of the breeder and multiplier in the sphere-pac configurations and the proximity of the SiC to the breeder, the tritium will be deposited by recoil in the multiplier and the surrounding structure. This recoil-deposited tritium will diffuse back to the purge very slowly because of the low tritium diffusivity in those materials and the formation of the oxides on the surface. Since the tritium release mechanism from Be is not clear, only an estimate of tritium inventory is made at this time.

6.2. DIVERTOR

6.2.1. Divertor Plasma Parameters

The high power density and plasma temperatures in a fusion power reactor must be reduced considerably in the divertor region to avoid sputtering where the plasma contacts a material target surface. Evidence shows that this can be accomplished with a divertor operating in the high recycling mode (*i.e.*, exhibiting a high density and low temperature at the divertor target). The high recycling mode at reactor-relevant parameters has not been demonstrated experimentally. In this study, analytical and computational models, calibrated to present-day experimental data and extrapolated to predict reactor performance, are used.

The ARIES-I design operates with high-recycling poloidal divertors in a double-null configuration. The BRAAMS [2] and EPIC [3] 2-D computational models, calibrated in the present-day experimental regime, are used to predict the performance of the impurity-control and particle-exhaust system. Both codes predict that for a mid-plane separatrix electron density of $\sim 10^{20} \text{ m}^{-3}$ (corresponding to a recycling coefficient of 0.98), the plasma temperature on the divertor target can be lowered to ~ 20 to 25 eV and the peak heat flux on the divertor target is $\sim 4.5 \text{ MW/m}^2$ (for a target inclination of 10° at the separatrix). The low net particle efflux required, as described above for global particle balance, is consistent with a relatively modest pumping rate. An estimate shows that an effective pumping speed of $100 \text{ m}^3/\text{s}$ is required for neutrals with an average energy of 1 eV in the pumping ducts.

First-wall erosion rates were determined by assuming that the carbon is sputtered from the SiC and the average sputtering yield is taken to be 0.04. No credit is taken for redeposition since the sputtered material is swept towards the divertor. The sputtering

contributions from D, T, and ^4He are comparable and the total first-wall erosion is computed to be 0.3 mm/y. The resulting carbon impurity concentration in the plasma is found to be almost half of the DT plasma density at the first wall ($\sim 3.0 \times 10^{18} \text{ m}^{-3}$), while the overall carbon concentration is less than 1% of the core plasma density.

The divertor target plate is designed with a thin tungsten (W) coating on the SiC-composite coolant tubes. Erosion of the W coating was determined for a range of likely flux conditions predicted by the computational codes. The nominal steady-state plasma solution resulted in an erosion rate of 2 mm/y. This is due entirely to the alpha particles, and a conservative value of 0.15 is used for the ratio of net to gross erosion. A comparable contribution to erosion could result from transient power fluxes to the target that are consistent with experimental observations of present-day divertor operation. The resulting W concentration in the plasma was calculated and found to be negligible. Implantation and permeation of tritium in the W target is found to be small and to result in a negligible tritium inventory in the target.

The computational study indicates that the desired low-temperature divertor plasma can be achieved with some caveats in the computational modeling and assumption. Erosion of the SiC first wall and the W divertor target are at acceptable levels for these solutions. It is assumed that these solutions are valid steady-state solutions, although the uncertainty in achieving these solutions remains. There are two key R&D directions: (1) improve the computational model and reduce the uncertainty in the solution, and (2) pursue active techniques for controlling the edge-plasma region to ensure that the high-recycle, low-temperature solution is attained.

The divertor parameters that are used in the tritium calculations are summarized in Table 6.2-I.

6.2.2. Plasma-Driven Permeation

The plasma-driven permeation of tritium was a concern during the ARIES-I study. Because of the high solubility and low diffusivity of tritium in SiC [1], the tritium inventory was calculated (by using the DIFFUSE code [4]) to be few kilograms at the end of life of the first wall. With a decrease in temperature, the tritium solubility increases while the diffusivity decreases. Therefore, the tritium inventory problem will be even more severe with a lower first-wall temperature.

The DIFFUSE code treats the structure with a stationary boundary. The first wall of a fusion reactor, however, recedes because of the erosion and sputtering. If the erosion

Table 6.2-I.
ARIES-I Divertor Parameters

DT flux ($\text{m}^{-2}\text{s}^{-1}$)	3×10^{19}
Divertor area (m^2)	124
DT energy (eV)	23.8
W thickness (mm)	2.0
SiC thickness (mm)	0.5
W temperature ($^{\circ}\text{C}$)	985
W/SiC interface temperature ($^{\circ}\text{C}$)	800
SiC/coolant interface temperature ($^{\circ}\text{C}$)	670

velocity is similar to or faster than the diffusion velocity, then the DIFFUSE code does not apply. For such a case, the implanted tritium will be released back to the plasma because of wall erosion before it has time to diffuse toward the coolant (*i.e.*, because of the sputtering and not the back-diffusion process). Thus, the tritium inventory will be much less than the case with diffusion as the only mechanism.

The erosion rate of the first wall is calculated in Sec. 5.2 using the following assumptions. The DT flux to the first wall is calculated to be $3 \times 10^{19} \text{ m}^{-2}\text{s}^{-1}$, with an average ion energy of 25 eV. The ion energy will increase to 80 eV when it strikes the first wall. The He particle flux is $4.3 \times 10^{18} \text{ m}^{-2}\text{s}^{-1}$, with an energy of 140 eV when it strikes the wall. The first-wall erosion yield is computed, including the effects of the energy distribution of the particles, by the IONSPUT code. The first-wall erosion is assumed to be limited by the erosion of carbon from the SiC matrix, with an effective specific gravity of C inside SiC of 1.5. No redeposition is assumed because the sputtered material, once ionized, will be swept along the field lines to the divertor target. The results of the erosion rate calculations are shown in Table 6.2-II.

The particle implantation depth is calculated using the ion transport code, TRIPOS [5]. The results indicate that the mean particle-penetration depth is 1.1 nm. The lifetime of this penetration layer is, therefore, only 100 seconds. The first-wall sur-

Table 6.2-II.
First-Wall Erosion Rates (mm/y)

Due to T	0.15
Due to D	0.10
Due to ⁴ He	0.07
Total	0.32

face area of ARIES-I is about 700 m². The tritium flux impinged on the first wall is $1.5 \times 10^{19} \text{ m}^{-2}\text{s}^{-1}$. Therefore, the maximum total tritium inventory in the first wall is only 6 g.

This is the first time that the effect of sputtering is included in the plasma-driven permeation calculations. Since the back transport of the tritium is enhanced by the sputtering process, the tritium inventory in the first wall is much reduced, from a few kg to 6 g. The same methodology can be used to calculate the tritium inventory in the divertor.

Tritium permeation through the divertor plate structure to the divertor coolant is usually of serious concern. For the ARIES-I design, SiC structural material with a W coating is used for the divertor. The primary purpose of the W coating is to reduce the sputtering. Of additional benefit is the reduction in the forward tritium permeation, as well as reducing the tritium inventory. Because the diffusivity of tritium in W is much larger than in SiC, the tritium preferentially diffuses backward to the plasma chamber rather than forward across the SiC tube to the coolant. Based on the DIFFUSE code calculations, the forward tritium diffusion to the coolant is essentially zero. The tritium inventory in the structure, mainly inside the W coating, is about 10 g.

6.3. BLANKET

6.3.1. Blanket Parameters

The reference ARIES-I blanket design uses SiC composite as the structural material, 10-MPa helium as the coolant, Li_2ZrO_3 as the solid tritium breeder, and beryllium-metal sphere-pac pellets as the neutron multiplier. The blanket is segmented toroidally into 32 inboard and 32 outboard poloidal modules. Each poloidal module comprises 17 nested, U-shaped, SiC-composite shells, as discussed in Sec. 8. The sphere-pac solid-breeder and Be neutron-multiplier mixture is located between the shells. Cylindrical helium-coolant channels with 1-mm-thick walls are embedded in each of the 17 SiC-composite shells. The helium coolant enters and exits the blanket at, respectively, 350 and 650 °C. The coolant enters the blanket from the inlet plena which are located in the shield behind the blanket and reflector. It then flows radially inward through the shells. It cools the shells while flowing in the toroidal direction before it turns and flows radially back into the coolant outlet plena. This routing configuration was selected to provide adequate cooling of the blanket materials and to minimize the blanket pressure drop. For the extraction of bred tritium, we use a separate helium-purge stream at 0.4 MPa flowing throughout the breeder sphere-pac material. To form the purge-flow stream, 1-mm-thick porous SiC-fiber sheets are brazed on each side of the tube-bank-geometry coolant-channel shell. The triangular-shaped channel that is formed between adjacent cylindrical coolant tubes and the porous sheet becomes the purge flow channel. This allows the shortest flow path (*i.e.*, half of the individual breeder-zone thickness) for the bred tritium to migrate before it reaches a purge flow channel. The Li breeder and Be multiplier sphere-pac is composed of 1- and 0.1-mm pellets. If these sizes lead to excessive purge-flow pressure drop, a larger size combination of 1.5- and 0.2-mm-diameter pellets will be acceptable. Radially, the blanket comprises a 1.05-cm-thick first wall, 20-cm-thick tritium breeding zone, 10-cm-thick beryllium reflector, 38-cm-thick SiC reflector, and 30-cm-thick SiC reflector and plena. Details of different blanket-zone dimensions and material compositions are give in Secs. 8.4 and 8.5. The summary of the outboard blanket parameters given in Table 6.3-I should also be representative of the inboard blanket modules.

6.3.2. Blanket Tritium Inventory

A tritium transport model for calculating the blanket tritium inventory in the ceramic breeding material has been developed at ANL within the solid-breeder base program [6]. This model accounts for the effects of diffusion, desorption, adsorption, and solubility.

Table 6.3-I.
ARIES-I Blanket Design Parameters

Coolant, helium at 10 MPa	
Inlet temperature (°C)	350
Outlet temperature (°C)	650
Radial zone thickness (cm)	
First wall	1.05
Breeder/multiplier	20.0
Beryllium reflector	10.0
SiC reflector	38.0
SiC reflector and plena	30.0
First wall: SiC composite	
Coolant channel inside diameter (cm)	0.8
Coolant wall thickness (mm)	1.0
Thermal conductivity (W/K-m)	10.
Temperature at top and bottom of the blanket module (°C)	
Minimum/maximum	553/878
Temperature at mid-plane of the blanket module (°C)	
Minimum/maximum	636/977
Breeder zone: sphere-pac 80:20 vol % mixture of Be and Li, respectively. Two pellet sizes, 1 mm and 0.1 mm in diameter. ^(a)	
Characteristic thickness (cm)	1.1 – 1.2
Effective thermal conductivity (W/K-m)	4.6
Temperature at top and bottom of the blanket module (°C)	
Minimum/maximum	442/878
Temperature at mid-plane of the blanket module (°C)	
Minimum/maximum	471/873
Be reflector zone: sphere-pac ^(a)	
Characteristic thickness (cm)	4.1
Effective thermal conductivity (W/K-m)	7.3
Temperature at top and bottom of blanket module (°C)	
Minimum/maximum	419/786
Temperature at mid-plane of the blanket module (°C)	
Minimum/maximum	446/843
SiC reflector zone: sphere-pac ^(a)	
Minimum characteristic thickness (cm)	8.6
Effective thermal conductivity (W/K-m)	7.2
Temperature at top and bottom of the blanket module (°C)	
Minimum/maximum	429/903
Temperature at mid-plane of the blanket module (°C)	
Minimum/maximum	461/1020

^(a)SiC structure with cooling channel inside diameter of 0.5 cm.

The input parameters to the model include the dimensions of the breeding zone, neutron wall loading, tritium production rate, volume fraction of the breeder and multipliers, temperature distribution, and purge gas conditions. These input parameters are given in previous sections and are consistent with the blanket configuration and thermal-hydraulic design.

The desorption rate constant of tritium from Li_2ZrO_3 has not been measured. The ITER activity, however, recommended using the Li_2O desorption-rate constant [7]. The diffusivity of tritium in Li_2ZrO_3 has been reported [8]. All other rate constants have been taken from the ITER activity [7]. The grain size of the material is not a material property, but rather it is a requirement for material fabrication. A grain size of 10 microns is assumed. With the input parameters established by the blanket design and the material properties obtained by the experiments, the tritium inventory is calculated based on a unit first-wall area (cm^2). The tritium inventory is calculated at each of the three separate blanket zones and the total tritium inventory is the sum over the three zones.

The tritium inventory in the breeding material is calculated to be 8×10^{-9} mole/ cm^2 of first-wall area, as shown on Fig. 6.3-1. This corresponds to a total blanket-tritium inventory in the breeding material of only 0.3 g. This small inventory is caused by the good tritium-release characteristics, as well as the temperature range (442–878 °C) of the selected breeding material. Another factor is the relatively small mass fraction of the breeding material in the blanket. This small tritium inventory number is consistent with the U.S. ITER calculation [7], which gave a breeding material tritium inventory of 0.11 g (more recent ITER calculations show a higher tritium inventory). The steady-state tritium inventory is reached quickly (~ 3000 s), as shown in Fig. 6.3-1, and is consistent with the low blanket-tritium inventory.

The calculation summarized above is for tritium inventory in the breeding material only. The ARIES-I reactor has a very large Be mass fraction in the blanket, and the breeder and multiplier are intimately mixed in a sphere-pac configuration. For such a system, the following activities occur:

1. The Be is oxidized by the moisture in the purge gas and also in the breeding material. In this way, Be is very similar to Al which is also easily oxidized. BeO will form a tight and stable layer that will prevent further oxidation. BeO will also prevent any tritium in the Be from being released to the purge gas if the temperature is below 610 °C. Recent experimental results show that rapid tritium release may occur if the temperature exceeds 610 °C.

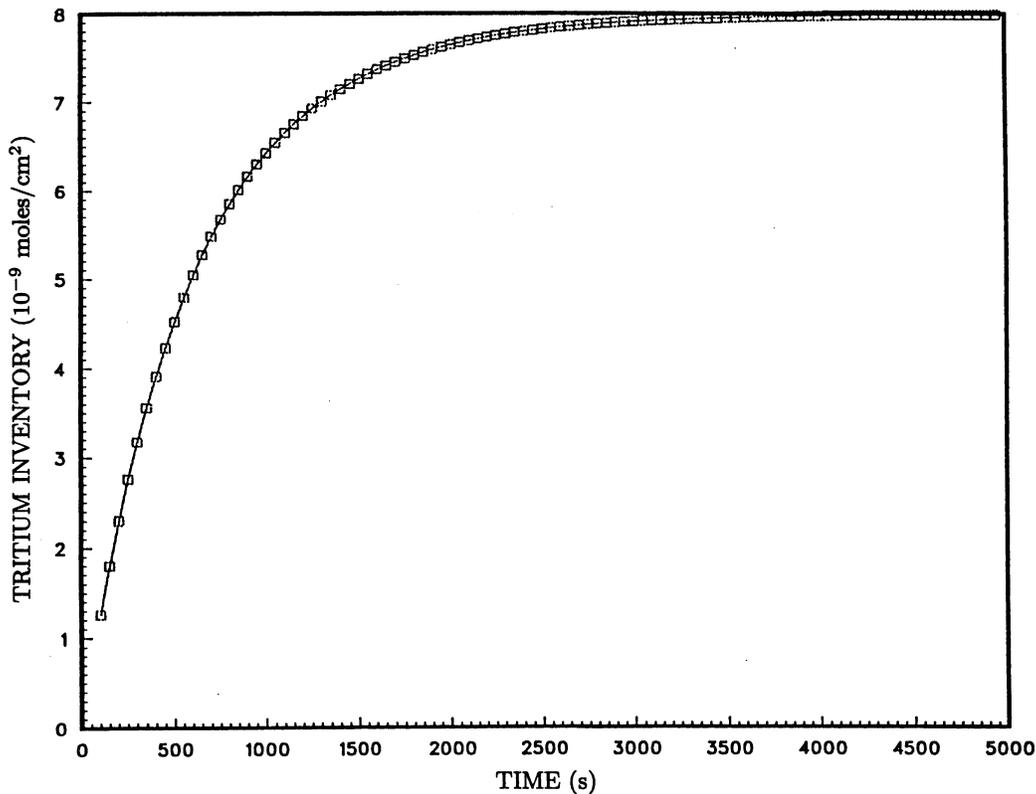


Figure 6.3-1. Tritium inventory in ARIES-I breeding material.

2. Approximately 1% of the tritium generated in the blanket is produced in the Be and amounts to ~ 3 g per full power day (FPD). Therefore, the total tritium production due to the Be(n,T) reaction is ~ 1.1 kg per full power year (FPY).
3. The tritium produced in the breeding material has a recoil energy of 2.5 MeV. Therefore, the tritium generated on the outside edge of the breeding material will be released by this recoil energy. Since 80% of the blanket volume is occupied by the Be, we may assume that the breeding material is surrounded by the Be. The tritium will thus be deposited inside the Be spheres. The penetration depth of the 2.5-MeV tritium atom in Li_2ZrO_3 is estimated to be 10 microns. The Li_2ZrO_3 sphere diameter is 1 mm. Therefore, 6% of the breeder volume is within the 10-micron range, or 6% of the tritium bred in this volume can be released by the recoil energy. Assuming 3% has sufficient energy left for deposition, and half of that will be released outward toward the Be, 4 g/FPD of tritium will be deposited inside the Be. This calculation is very rough and should be considered as only an indication of the magnitude of the problem. Therefore, the total tritium deposited into the Be is estimated to be 1.4 kg/FPY.

The tritium release mechanism from the Be is highly uncertain. There has been only one measurement made of hydrogen diffusivity in Be, and that result is questionable [9]. The difficulty is to separate the effect of the bulk diffusion inside the Be from the surface effect caused by the oxide. Transport and retention in Be has been highlighted as an ITER R&D issue. Experimental work (Hollenberg and Baldwin, PNL; and Longhurst, INEL) is underway.

Figure 6.3-2 summarizes data for tritium release from Be from three different measurements [9–11]. The new data by Baldwin show very little tritium release (0.01%–4%) from the irradiated Be in the temperature range of 300–510 °C. At 610 °C, however, there is a sudden burst of tritium released. For ARIES-I calculations, we are assuming that there is no tritium released below 610 °C and a complete release of tritium above 610 °C. Since the existing experimental results show contradicting release characteristics, this assumption needs to be verified by the future experiments. It should be noted that this assumption is different from that predicted by the ITER modeling, which is also included on Fig. 6.3-2.

Since we are using the results from Baldwin, it is important to summarize the experimental conditions. The samples of the experiment come from hot-pressed Be material (~100% dense, ~2% BeO content) that was irradiated to $\sim 5 \times 10^{22}$ n/cm² fast fluence (~2500 appm tritium and ~30,000 appm He) at low temperature (≥ 75 °C) in ATR. Therefore, both the tritium content and He content are much higher than that of ARIES-I. The high He content may be a very important factor in the tritium release. At room temperature, this high He content exists in the form of individual atoms and small bubbles. However, at elevated temperature (*e.g.*, 610 °C), the He will migrate to grain boundaries, coalesce to form large bubbles, and form interconnected pathways to enhance tritium release. Therefore, this tritium release depends on the high helium content in the Be and this observed tritium release with the high helium content of the ATR samples may not occur for Be with less He content (as is the case for Be in the ARIES-I blanket).

With the assumption that the tritium will be released from the Be with $T > 610$ °C, it is possible to estimate the tritium inventory inside the Be. The volume fraction of Be with temperatures below 610 °C is calculated to be 25%. The tritium produced in the Be is 1.15 kg/FPY, while the tritium deposited inside the Be is 1.4 kg/FPY. Therefore, the total amount of tritium retained inside the Be is 0.64 kg/FPY. If the blanket is heated above 610 °C once a year while the plasma is down (or in partial power), the maximum tritium inventory in the Be is then 0.64 kg.

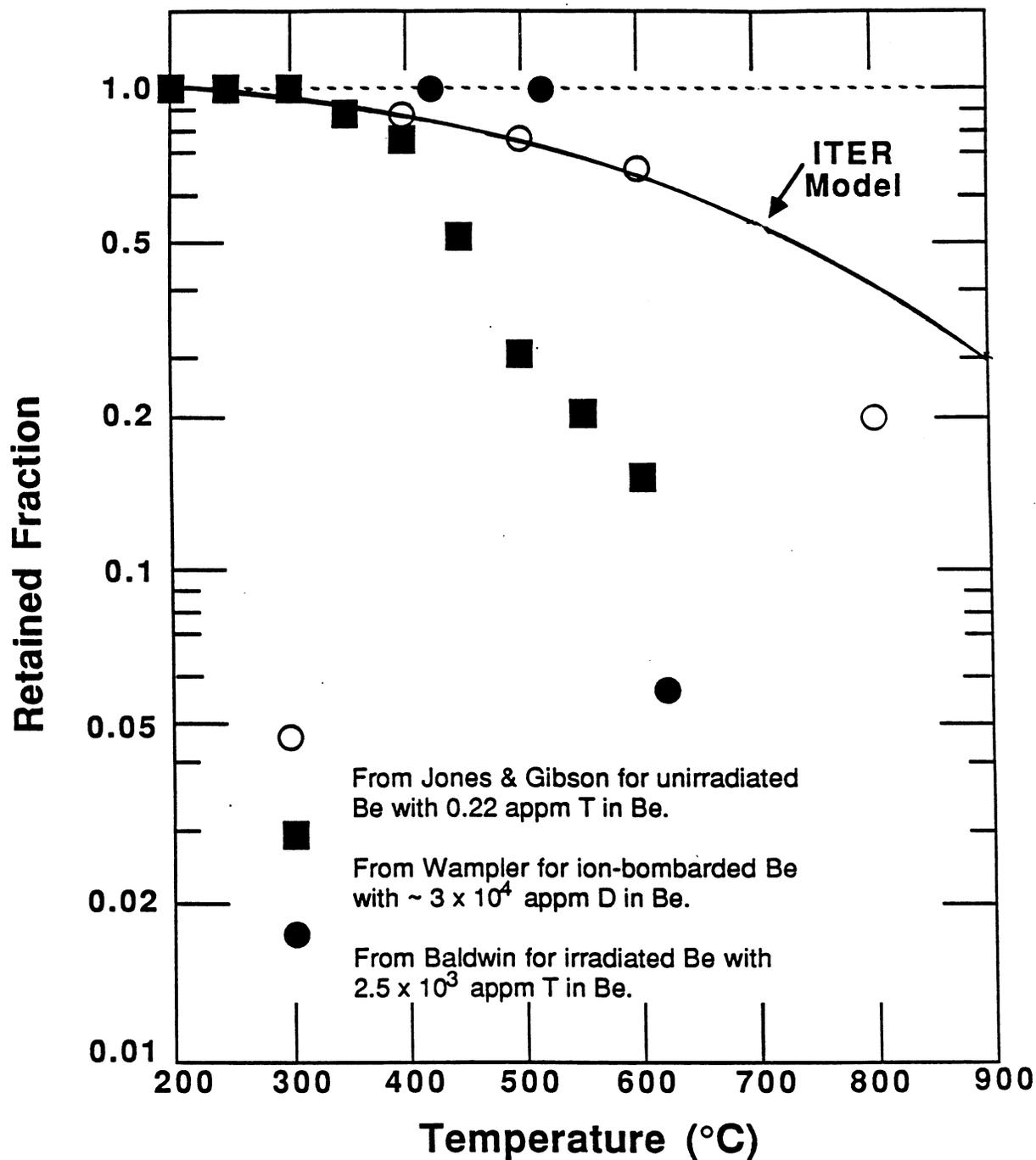


Figure 6.3-2. Fraction of "trapped" hydrogen isotope in Be as a function of annealing temperature for 25- to 100-h anneals of unirradiated Be soaked with tritium [9] and 10-min anneals of deuterium-ion-implanted Be.

6.4. PLASMA EXHAUST SYSTEM

The largest tritium unit in a DT fusion reactor is the plasma-exhaust processing system. A conventional plasma-exhaust system uses a cryogenic pump to pump the D, T, H, ^4He , and impurities from the divertor channel. First, the hydrogen isotopes are separated from the exhaust. Next, the isotopes of hydrogen are separated from each other by the cryogenic distillation (CD) process. The processing system involves a number of units that may have a large tritium inventory (*e.g.*, CD pumps and columns). In addition, the throughput of the tritium in the exhaust is usually large, often many times larger than the burn rate in the plasma. The ARIES-I design aims at reducing the tritium inventory in this system by: (1) increasing the plasma burn fraction to reduce the tritium throughput in the exhaust and (2) eliminating or modifying the units that have large tritium inventories.

The plasma burn fraction for ARIES-I is calculated to be 19.3% (Sec. 3), which is a factor of 7 higher than that of ITER. Even at this high burn fraction, however, the tritium throughput in the plasma exhaust is still over four times larger than the next largest tritium unit (*i.e.*, the blanket recovery system). Therefore, it is important to reduce the tritium inventory and the complexity of the plasma exhaust system. There are three components in this system that usually have large tritium inventories:

1. The cryogenic pumps, which operate in a pulse mode between the regenerating time (tritium inventory is proportional to the pulse time);
2. A gettering bed for tritium separation, which is often used to separate hydrogen isotopes from the waste (also operates in a pulse mode); and
3. The CD unit, which is used to separate H, D, and T (can be large because of the large throughput and the high separation factor sometimes required).

The complexity of the plasma exhaust system is primarily due to the multiple CD units and the equalizers required for the separation of H, D, and T.

A key concept developed in the ARIES-I study is that of using the CD process to separate only the H from D and T, while keeping the D:T ratio at 1. The fueling to the plasma is by DT pellet injection and because the D:T ratio in the exhaust is also 1, there is no need for separating D and T. The H is generated by the DD reaction in the plasma. Since the DD reaction cross section is a factor of 100 smaller than that of the DT reaction,

the H production rate is also a factor of 100 smaller than that of the ^4He production rate. In order to minimize the throughput to the CD unit, the H concentration in the plasma is allowed to accumulate to $\sim 1\%$. Even at this level of H concentration, the plasma impurity concentration is still dominated by He, which is 10.7%. For such a system, only 10% of the plasma exhaust is required to pass through the CD unit to remove H. Since no additional separation of D&T is required, one CD column is sufficient, compared with three or four CD columns for a conventional fuel-processing system.

Because of the steady-state operation, a Pd diffuser is used to separate the hydrogen isotopes from the ^4He and other wastes (hydrogen will diffuse through the Pd diffuser, while the other gases will bypass it). The performance of a Pd diffuser in fuel cleanup has been demonstrated [11]. Waste processing is conventional (*i.e.*, using an oxidizer to convert all of the T in the waste to the HTO form, using a molecular sieve to separate HTO from other gases, and using a reduction unit to reduce the HTO to HT which will then recombine with the hydrogen stream from the Pd diffuser).

The cryogenic vacuum pump is another component that usually has a large tritium inventory. This is caused by the pump's pulsed mode of operation. To reduce the T inventory in the pumps, the ARIES-I design uses turbo-molecular pumps. Although a metallic turbo-molecular pump has to be shielded from the magnetic field, a ceramic pump, which doesn't require shielding, is being developed in Japan. At this time, the size of the pump is limited [12].

The flow diagram of the ARIES-I plasma-exhaust processing system is shown in Fig. 6.4-1 and the major parameters are listed on Table 6.4-I. The tritium inventory in the CD unit is shown and compared to that for ITER in Table 6.4-II. By careful selection of the equipment and/or processes to be used in the plasma-exhaust fuel-processing system, the tritium inventory in the system is minimized.

6.5. SUMMARY

The key issues for a tritium system are the tritium inventory and the tritium release. For the ARIES-I design, the structural material is SiC, through which the tritium permeation rate is very low. Therefore, tritium release, if any, will be by leakage and not by permeation. Tritium leakage is caused mainly by cracks that develop in the coolant tubes. Because the coolant pressure (10 MPa) is much larger than the purge-gas pressure (0.4 MPa), any leakage will be toward the purge. Therefore, it can be assumed that the

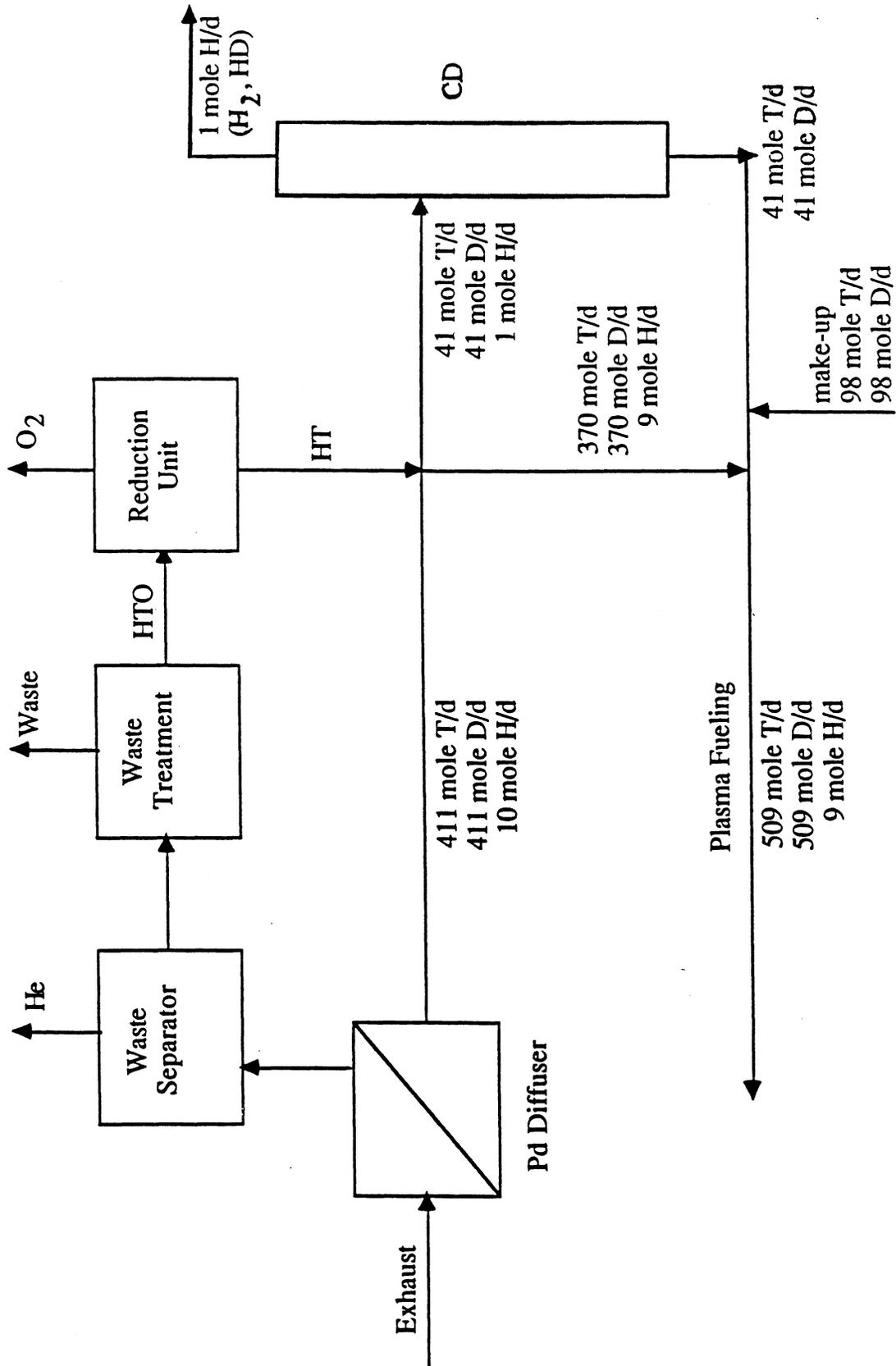


Figure 6.4-1. ARIES-I fuel-processing flow diagram.

Table 6.4-I.
ARIES-I Fuel-Cycle Parameters

Fusion power (MW)	1,925
Tritium burn rate (g/d)	295
Tritium burn fraction	0.193
Tritium feed rate (g/d)	1,528
Tritium exhaust rate (g/d)	1,233
D exhaust rate (g/d)	822
He exhaust rate (g/d)	393
H generation rate (g/d)	1
H/(D + T)	1%
He/(D + T)	12%

Table 6.4-II.
Cryogenic-Distillation Capacity Comparison

	ITER	ARIES-I (Conventional)	ARIES-I (Only separation of H)
Throughput (mole T/d)	2,500	411	41
Number of columns	4	4	1
Tritium inventory (g)	800	150	50

tritium leakage to the primary coolant and to the steam generator will be extremely small.

The tritium inventories of the ARIES-I major components have been calculated and summarized in Table 6.5-I. Minimizing the inventory is a key effort and the important assumptions used for this activity are also included on this table. The total tritium inventory in the reactor is 700 g, 90% of which is in the beryllium.

The most uncertain tritium inventory is in the beryllium. Tritium production in Be of 1.15 kg/FPY is probably within 50% accuracy. The recoil-deposited tritium inventory of 1.4 kg/FPY is possibly accurate to within a factor of 2 or 3. The most uncertain are the tritium release characteristics. The data used here are based on some recent experimental results, with a helium content in Be of 30,000 appm. If this data proves not to be relevant to ARIES-I conditions, the tritium inventory in the Be can be as high as 5 to 10 kg/FPY. Experimental work in this area is in progress and more definite answers should be available in about one year.

Table 6.5-I.
Estimates of Tritium Inventory in the ARIES-I Design

Components	T Inventory	Comments
Fuel cycle		
Vacuum pumping	Small	Using ceramic turbo-molecular pump
Helium separation	Small	Using Pd diffuser
Cryogenic distillation	50 g	Using CD unit to remove H only
First wall	6 g	Sputtering removes C and T
Divertor wall	10 g	Using W coating on SiC
Blanket		
Breeder material	1 g	Selection of breeding material and operating temperature
Beryllium	640 g	1. Complete tritium release >610 °C 2. Blanket heat to 610 °C once a year.

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