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ARIES-I TRITIUM SYSTEM*

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ABSTRACT

A key safety concern in a D-T fusion reactor is the tritium inventory. There are three components in a fusion reactor with potentially large inventories, i.e., the blanket, the fuel processing system and the plasma facing components. The ARIES team selected the material combinations, decided the operating conditions and refined the processing systems, with the aiming of minimizing the tritium inventories and leakage. The total tritium inventory for the ARIES-I reactor is only 700 g. This paper discussed the calculations and assumptions we made for the low tritium inventory. We also addressed the uncertainties about the tritium inventory.

I. INTRODUCTION

The tritium systems in a fusion reactor has to reduce the tritium inventory in the system and simplify the system design. The tritium system is usually small. Therefore, the cost of the tritium system will be of secondary importance. The design of the ARIES-I tritium system is focused on the reduction of the tritium inventory and simplification of the design. However, due to the material selection and system configuration, some new problems have been uncovered during the study. The problems will be documented. Further work, both by theory and by experiments, will be required to address those problems.

The largest tritium stream in a D-T fusion reactor is the plasma exhaust. The throughput of the ARIES-I plasma exhaust is reasonably small due to 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 separation of hydrogen isotopes from the waste; and we use a turbo-molecular pump for pumping the exhaust. Another unit which has large tritium inventory is the cryogenic distillation (CD) unit. 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 is estimated to be 50 g. The number of the CD columns will also be reduced, from 4 to 1.

The tritium inventory and plasma driving permeation in the first wall and the impurity control system are always concerns in a fusion reactor design. 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 2 mm tungsten coating on the SiC composite coolant tubes.

The third region with tritium inventory concerns is the breeding blanket. Li_2ZrO_3 is selected as the breeding material partly due to its good tritium release characteristics. The tritium inventory in the breeding material is calculated to be only 1 g. However, there is the potential of high tritium inventory in the Be. Due to the intimate mixing of the breeder/multiplier in the sphere-pac configurations, the tritium will be deposited in the multplier due to recoil. The tritium release mechanism from Be is uncertain Therefore. Only an estimate of tritium inventory is made at this time.

II. FIRST WALL AND DIVERTOR

A. First Wall and Divertor Description

The high power density and plasma temperatures in the fusion power reactor regime must be reduced considerably in the divertor region to avoid sputtering where the plasma contacts a material target surface. 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. Theory predicts that for a mid-plane separatrix electron density of $\sim 10^{20} m^{-3}$ (corresponding to a recycling coefficient of 0.98), the plasma temperature on the divertor target can be lowered to ~20 to

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25 eV and the peak heat flux on the divertor target is ~4.5 MW/m^2 (for a target inclination of 10 at separatrix).[1]

First wall erosion rates were determined assuming the carbon is sputtered from the SiC. 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 He are comparable and the total first wall erosion is computed to be 0.3 mm/y. The sputtering of the first wall may have a major impact on the first wall tritium inventory, as will be discussed later in this paper.

The divertor target plate is designed with a 2 mm tungsten coating on the silicon carbide composite coolant tubes. 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.

The divertor parameters which will be used in the tritium calculations are summarized in Table 1.

Table 1.
Divertor Parameters

DT flux, 1/m^2 - s	3 x 10^{19}
Divertor area, m^2	124
DT energy, eV	23.8
W thickness, mm	2
SiC thickness, mm	0.5
W temperature, °C	985
W-SiC interface temperature, °C	800
SiC-coolant interface temp., °C	670

B. Plasma-Driven Permeation

The plasma driven permeation tritium was a concern during the ARIES study. Due to the high tritium solubility and low tritium diffusivity in SiC,[2] the tritium inventory was calculated to be few kilograms at the end of life of the first wall by using the DIFFUSE code.[3] The tritium solubility increases, while the diffusivity decreases, with a decrease in temperature. 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. In a fusion first wall, the first wall recesses due to the erosion and sputtering. If the erosion 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 due to wall erosion before it has time to diffuse toward the coolant. Therefore, the tritium will be released back to the plasma chamber due to sputtering process, not due to 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. The DT flux to the first wall is calculated to be 3 X 10^{19}/m^2-sec, with an energy of 25 eV. This energy will increase to 80 eV when it strikes the first wall. The He particle flux is 4.2 X 10^{18}/m^2-sec, with an energy of 140 eV when it strikes the wall. 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. With these assumptions, the first wall erosion rate is calculated to be 0.32 mm/y.

The particle implantation depth is calculated to be 1.1 nm. The lifetime of the penetration layer is, therefore, only 100 seconds. The first wall surface area of ARIES-I is about 700 m^2. The tritium flux impinged on the first wall is 1.5 X 10^{19}/m^2-sec. Therefore, the maximum total tritium inventory in the first wall is only 6 grams.

The tritium permeation through the divertor plate structure to the divertor coolant is a usually serious concern. For ARIES-I design, a SiC with W coating is used for the divertor. The primary purpose of the W coating is to reduce the sputtering. An added benefit is to reduce the forward tritium permeation, as well as the tritium inventory. The diffusivity of tritium in W is much larger than that in SiC. Therefore, the backward diffusion of the tritium to the plasma chamber is much easier than the forward diffusion of the tritium 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.

III. BLANKET SYSTEM

A. Blanket Description

The selected 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 Be as the neutron multiplier. The helium coolant comes into the blanket at 350°C and leaves the blanket at 650°C. For the extraction of bred tritium, we used a separate helium purge stream flowing

throughout the breeder sphere-pac material. The helium purge stream pressure is 0.4 MPa. The Li breeder and Be multiplier sphere-pac composes of 1 mm and 0.1 mm pellets. The detailed blanket design description can be found in Ref.4.

B. Blanket Tritium Inventory for ARIES Design

A tritium transport model to calculate the blanket tritium inventory in the ceramic breeding material has been developed at ANL within the solid breeder base program.[5] This model takes the account of the the effect of diffusion, desorption, adsorption, and solubility. The input parameters to the model includes the dimension of the breeding zone, the neutron wall loading, the tritium production rate, the volume fraction of the breeder and multipliers, the temperature distribution, as well as the purge gas conditions. The parameters used here are consistent with the blanket configurations and thermal hydraulic design.[4]

The desorption rate constant of tritium from Li_2ZrO_3 has not been measured. The ITER activities recommended to use the desorption rate constant for Li_2O.[6] The diffusivity of tritium in Li_2ZrO_3 has been reported.[7] All other rate constants have been summarized by the ITER design studies.[6] A grain size of 10μ is assumed. With the input parameters established by the blanket design, and the material properties obtained by the experiments, the tritium inventory is calculated to be 1 g.

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

1. The Be is being oxidized by the moisture in the purge gas and also by the breeding material. In this way, Be is very similar to Al. It is easily oxidized. Be_2O will form a tight and stable layer which will prevent further oxidation. Be_2O will also prevent any tritium in the Be being released to the purge gas if the temperature is below 610°C. Recent experimental results show that rapid tritium release may occur if temperature exceeds 610°C.

2. Approximately 1% of the tritium generated in the blanket is produced in the Be, or about 3 g/FPD. Therefore, the total tritium production due to the Be(n,T) reaction is about 1.1 kg/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 10μ. The Li_2ZrO_3 sphere diameter is 1 mm. Therefore, 6% of the breeder volume is within the 10μ 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 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

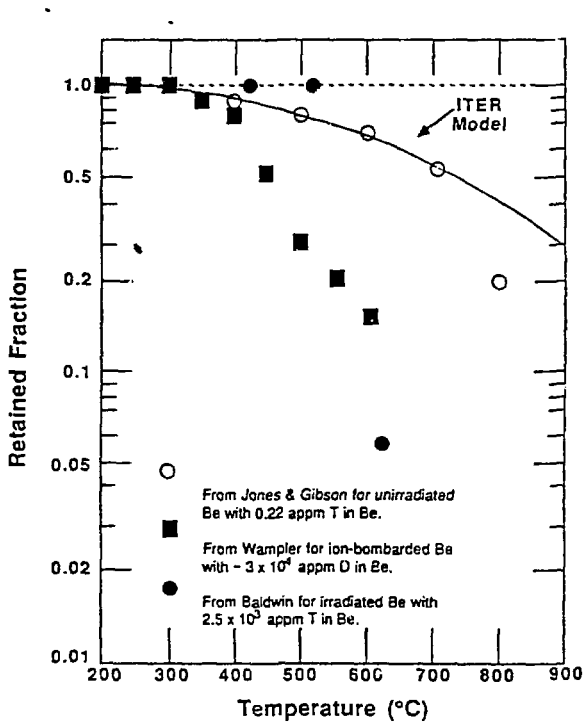

Fig. 1 Fraction of "trapped" hydrogen isotope in Be as a function of annealing temperature for 25-100 hour anneals of unirradiated Be soaked with tritium and 10 minute anneals of deuterium-ion-implanted Be.

measurement of hydrogen diffusivity in Be, but the result is questionable.[8] 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) are underway.

Figure 1 summarized tritium-release-from-Be data from three different measurements.[8-10] The new data by D.L. Baldwin shows very little tritium release (0.01% - 4%) from the irradiated Be in the temperature range or 300-510°C. At 610°C, there is a sudden burst of tritium release. For ARIES-I calculation, we are assuming that there is no tritium release 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.

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) which was irradiated to ~5 X 10^{22} n/cm^2 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 to form interconnected pathways to enhance tritium release. Therefore, this tritium release depends on the high helium content in the Be. Thus, this observed tritium release with the high helium content of the ATR samples may not occur for Be with less He content.

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 remained inside the Be is 0.64 kg/FPY. If the blanket is heated above 610°C one a year while the plasma is down (or in partial power), the maximum tritium inventory in the Be is then 0.64 kg.

IV. PLASMA EXHAUST SYSTEM

The largest tritium unit in a D-T fusion reactor is the plasma exhaust processing system. A conventional plasma exhaust system uses a cryogenic pump to pump the D, T, H, He and impurities from the divertor channel. Upon separation of the hydrogen isotopes from the waste, the hydrogen isotopes are separated by a number of cryogenic distillation columns (CD). The process involves a number of units which has a larger tritium inventory, such as the cryogenic pumps and the CDs. The ARIES-I design aimed to reduce the tritium inventory in this system by two ways:

1. Increase the plasma burn fraction to reduce the tritium throughput in the exhaust.

2. Elimination or modifying the units which have a large tritium inventory.

By assuming a high recycle divertor, the plasma burn fraction is calculated to be 19.3%. There are three components in this system which usually have large tritium inventories:

1. The Cryogenic Pumps. The conventional cryogenic pumps operate in a pulse mode between the regenerating time. The tritium inventory is proportional to the pulse time.

2. Gettering for Tritium Separation. A gettering bed is often used to separate hydrogen isotopes from the waste. The gettering bed operates also in a pulse mode.

3. The CD Unit. The CD unit is used to separate H, D, and T. The unit can be large due to the large throughput and the high separation factor sometimes required.

The complexity of the system is caused mainly by the multiple CD units and the equalizers required for the separation of H, D, and T.

A key concept developed is to use the CD to separate only the H from D, T, while keeping the D/T ratio at 1. The fueling to the plasma is by pellet injection with a DT pellet, while the D/T ratio in the exhaust is also 1, there is no need of D/T separation. The H is generated by the D-D reaction in the plasma. Since the D-D reaction cross-section is a factor of 100 smaller than that of D-T reaction, the H production is small. In order to minimize the throughput to the CD, the H concentration in the plasma is allowed to accumulate to about 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 CD to remove H. Since no additional D/T

separation is required, only one CD column is required, comparing to three or four CD columns for a conventional fuel processing system.

To separate the hydrogen isotopes from the He and other waste, a Pd diffuser is used. A key reason to use the Pd diffuser is the steady state operation. The performance of a PD diffuser on fuel cleanup has been demonstrated.[11] Hydrogen will diffuse through the Pd-diffuser, while the other gases will be by-passed.

Another component which usually has large tritium inventory is the cryogenic vacuum pump. The large tritium inventory is caused by the pulsed operation mode of the pump. For ARIES-I, we suggest the using of a turbo-melecular pump. A ceramic turbo-melecular pump is being developed in Japan to remove this shielding requirement.[12] At this time, the size of the pump is limited.[13] A concept of continuously-regenerating cryopumps has also been proposed.[13]

The flow diagram of the ARIES-I plasma exhaust processing system is shown in Fig. 2. The major parameter is listed on Table 2.

Table 2.
ARIES-I Fuel Cycle Parameters

Fusion power, MW	1925
Tritium burn rate, g/d	295
Tritium burn fraction	0.193
Tritium feed rate, g/d	1528
Tritium exhaust rate, g/d	1233
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
Tritium inventory in CD,g	50

V. 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 made of SiC. The tritium permeation rate through the SiC structure is very low. Therefore, tritium

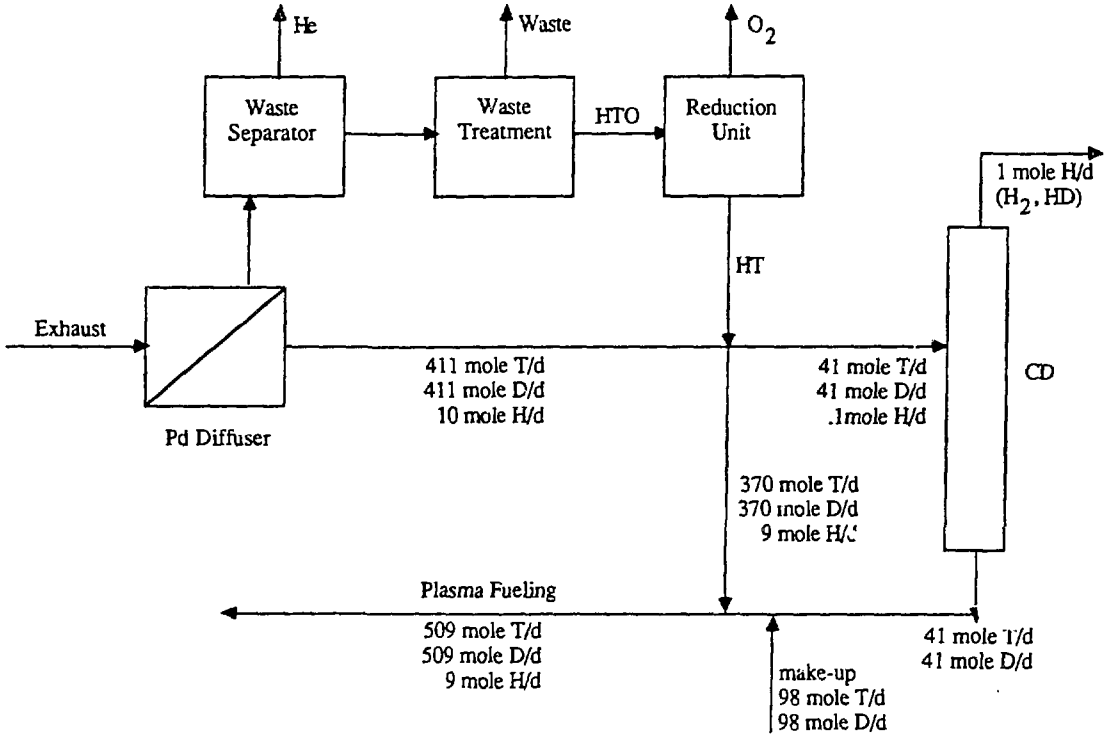

Figure 2. ARIES-I fuel processing flow diagram.

release if any, will be by leakage, not by permeation. The tritium leakage is dominated by the cracks developed in the coolant tubes. However, 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 tritium leakage to the primary coolant, and to the steam generator, will be extremely small.

The tritium inventory of the major components have been calculated, as summarized on Table 3. The key efforts are to minimize the inventory. The important assumption used for this activity is also included on this table. The total tritium inventory in the reactor is 700 g, with 90% of that in the Be.

Table 3. Tritium Summary

Components	Tritium Inventory	Comments
Fuel Cycle		
Vacuum pumping	small	using ceramic TMP
He separation	small	using Pd diffuser
Cryogenic distl.	50 g	using CD to remove H only
First Wall	6 g	Sputtering removes C and T
Divertor Wall	10 g	Using W coating SiC
Blanket		
Breeder matl.	1 g	Selection of breeding material and temperature
Be	640 g	1. Complete tritium release >510°C 2. Blanket heat at 610°C once a yr

The most uncertain tritium inventory is also in the Be. The tritium production in the Be, 1.15 kg/FPY, is probably within 50% accuracy. The recoiled tritium deposition, 1.4 ky/FPY, is possibly accurate to within a factor of 2 or 3. The most uncertain is the tritium release characteristics. The data used here is based on some recent experimental results, with a He content in Be of 30,000 appm. This result is not consistent with the other two experimental results. However, at this time, we can only make a best estimate. The tritium inventory in the Be can be as high as 5 to 10 kilograms/FPY. Experimental works are in progress. We will have a more definite answer in about one year.

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