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WITH A SELF-PUMPED LIMITER*

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THE TRITIUM SYSTEM FOR A TOKAMAK WITH A SELF-PUMPED LIMITER*

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ABSTRACT

The self-pumping concept was proposed as a means of simplifying the impurity control system in a fusion reactor. The idea is to remove helium in-situ by trapping in freshly deposited metal surface layers of a limiter or divertor. Trapping material is added to the plasma scrape-off or edge region where it is transported to the wall. Some of the key issues for this concept are the tritium inventory in the trapping material and the permeation of protium and recycling of tritium. These quantities are shown to be acceptable for the reference design. The tritium issues for a helium-cooled solid breeder reactor design with vanadium alloy as a structural material are also examined. Models are presented for tritium permeation and inventory calculation for structure materials with the effect of a thin layer of coating material.

1. INTRODUCTION

The self-pumping concept was proposed¹ as a means of simplifying impurity control in a fusion reactor. The basic concept is to remove helium in-situ by trapping in freshly deposited metal surface layers of a limiter or divertor. The design of the self-pumping limiter has to satisfy the requirement of trapping helium, recycle deuterium and tritium (DT) and permeate protium simultaneously. At the same time it has to satisfy all the thermal hydraulic requirements associated with the first wall. A key requirement for the deposited material is to trap helium much better than hydrogen. It has been demonstrated experimentally that nickel preferentially traps helium² and that several other materials (iron, vanadium, niobium, molybdenum, and tantalum) are believed to be capable of preferential trapping. The selective trapping in certain metals is because of the negligible solu-

bility of helium in the lattice. The injected helium will diffuse through the lattice until it reaches a nearby trapping site where it can come out of solid solution. Hydrogen, on the other hand, remains in the solid solution until it diffuses to the surface and escapes. Other plasma contaminants, notably oxygen, can be removed by the self-pumping system by chemically combining with the deposited metal. Protium formed by a D-D reaction needs to be removed by permeation for efficient plasma operation.

Benefits of the self-pumping system are the elimination of vacuum ducts, pumps, and penetration shielding (except for a very small startup system), and the reduction of tritium recycling and refueling. In addition, a self-pumped system may perform better and last longer than alternative systems such as a pumped limiter. The first case here is a self-cooled lithium/vanadium blanket with first wall as a limiter. This concept combines the functions of first wall and limiter into a single first wall structure. The wall is shaped in accordance with the outermost plasma flux surface. Trapping material is added to the plasma scrape-off or edge region where it is transported to the wall. The entire wall area is used for helium trapping.

The important issues for the self-pumping concept are the tritium inventory in the trapping material and the permeation of tritium and protium into the coolant. The retained tritium inventory should be as low as possible. The tritium permeation rate should be less than the blanket breeding rate. The protium permeation should be high enough to remove protium produced by the D-D reaction at very low protium concentration (H/DT ratio ~ 0.1). The tritium permeation and inventory as well as protium permeation are calculated for the entire lifetime of the reactor.

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A second system considered is a helium-cooled solid breeder reactor design concept. The Li_2O breeder with the coolant using V-alloy as a structural material appears to be the most attractive design.³ The tritium system for this concept is analyzed and both the permeation and inventory for V-alloys with and without this layer of tungsten as coating are calculated.

II. TRITIUM PERMEATION AND INVENTORY AND PROTIUM PERMEATION FOR SELF-PUMPING

The retained tritium inventory is an important factor for this concept and should be as low as possible. The tritium permeation rate is less critical but also should be minimized. To first order, the permeation rates of tritium and protium are governed by a similar processes. The permeation should be low enough for acceptably low tritium throughout, but must be high enough to remove protium. Protium is formed by the D-D reaction at a rate of about 1/400 of the α -production rate (at the plasma temperatures considered in TPSS).⁴ This is equivalent to 0.1 gms of protium per day, for a 2.5 MW/m² wall loading.

In steady state, the protium concentration in the plasma can be related to the permeation as follows:

$$\frac{N_p}{N_{DT}} I_{DT} \epsilon = \dot{M}_p \quad (1)$$

where I_{DT} is the current of DT ions to the boundary and \dot{M}_p is the protium production rate. N_{DT} and N_p are the number density per unit volume for both deuterium-tritium and protium respectively. The self-pumping efficiency, ϵ , is defined as the removal rate via permeation to the coolant divided by the particle impingement rate. This expression assumes equal transport properties of hydrogen isotopes in the plasma.

The DT current is approximately scales linearly with neutron wall loading. For a wall loading of 2.5 MW/m², and for typical plasma edge conditions given in Ref. 4, $I_{DT} = 2.5 \times 10^{24} \text{ s}^{-1}$ and $\dot{M}_p \approx 7.9 \times 10^{17} \text{ s}^{-1}$. This gives for the protium concentration:

$$\frac{N_p}{N_{DT}} = \frac{3.2 \times 10^{-7}}{\epsilon} \quad (2)$$

An acceptable protium concentration is about 1%, therefore ϵ has to be $\geq 3 \times 10^{-2}$. In order to have an acceptable level of tritium leakage to the first wall, the maximum allowable leakage rate should not exceed the blanket breeding rate. The tritium breeding rate is calculated to be $3 \times 10^{20} \text{ T/sec}$. This corresponds to self-pumping efficiency of $\epsilon < 2.5 \times 10^{-4}$. Therefore the first wall has to be designed with

$$2.5 \times 10^{-4} > \epsilon > 3 \times 10^{-5}.$$

To compute the permeation and inventory of hydrogen isotopes in candidate self-pumped materials, we have developed a steady state computer code. The permeation model assumes a hydrogen isotope atom implantation flux J_i at a depth δ in the wall less than the wall thickness d . This implantation depth depends primarily on the energy of D-T particles which is assumed to be in the 100's of eV range.

It is assumed that gas molecules leave either wall surface (front or back) by recombination limited desorption according to

$$J = 2 K_r C^2 \quad (3)$$

where C is the dissolved hydrogen isotope concentration near the surface, and K_r is the recombination coefficient given by:⁵

$$K_r = \frac{4 \alpha}{K_{SO}^2 \sqrt{\pi K M T}} \exp\left(-\frac{2 E_s - E_x}{K T}\right) \quad (4)$$

where

- α = sticking coefficient (= 1 for clean surfaces)
- K_{SO} = pre-exponential Sievert's solubility constant
- K = Boltzmann constant
- M = mass of the molecule formed by recombination
- T = absolute temperature
- E_s = energy of solution for hydrogen in metal
- E_x = $E_s + E_d$ (E_d = diffusion energy)

and

$$E_x = \begin{cases} E_s + E_d & \text{if } E_s + E_d > 0 \\ 0 & \text{otherwise.} \end{cases}$$

The protium concentration in the material is much less than the tritium or deuterium concentration. The effect of the protium isotope is taken into consideration by assuming that the protium flux leaving either surface is given by:

$$J = K_r C_H C_{DT} \quad (5)$$

where

- C_H is protium concentration.
- C_{DT} is tritium-deuterium concentration.

The code was run for the first wall/limiter system using vanadium as structural material. This system is designed for 10 years of operation at 2.5 MW/m² neutron wall loading. The temperature gradient in the wall is mainly due to the surface heat flux (q) and is given by

$$K \frac{\partial T}{\partial x} = q \quad (6)$$

where K is the thermal conductivity of the wall material. Initially, the vanadium wall thickness is about 2 mm with the back surface temperature fixed at 640°C. The average surface heat flux is about 0.92 MW/m².⁴ This yields a front surface temperature of 697°C.

There are two key requirements for the self-pumping material pertaining to lifetime. These are the fraction of helium capable of being trapped inside the material and the maximum surface temperature at which the material will still trap helium. Based on limited experimental data^{6,7} the average trapping atomic fraction is estimated to be 30% and the maximum surface temperature is estimated to be 0.7 of the melting temperature. Since He trapping rate of 10¹⁴/cm²-sec is required, fresh vanadium is being deposited at the rate of 3.4 × 10¹⁴/cm²-sec, which is equivalent to 1.5 mm/yr. The thermal conductivity of the deposited vanadium is assumed to be 70% of that for full density vanadium. The rate of the temperature increase as a result of the added material is given from Eq. 6 by

$$\frac{dT}{dt} = \frac{Q}{K} \left(\frac{dx}{dt} \right) \quad (7)$$

This yields a temperature increase of about 62 °K/yr. After 10 years of operation the surface temperature becomes about 0.7 of the melting temperature.

Figure 1 shows the tritium permeation and inventory for the vanadium wall as a function of number of years of operation and wall thickness. As the wall thickness increases, the front wall temperature increases which increases the hydrogen isotope flux leaving the front surface and reentering the plasma. This has the effect of reducing the permeation rate. Thus, the permeation rate is largest in the early years of operation and continuously decreases thereafter. At the start of operation the permeation rate is equal to about five times the tritium burn up rate. This is an order of magnitude smaller than what would be removed by a regular pumped limiter or divertor. If necessary, the initial permeation rate could be made smaller by simply starting with an additional wall thickness. For example, an initial added thickness of 1.5 mm would reduce the permeation rate approximately in half. Although this might impact the system lifetime, there is already a safety margin in determining the lifetime. Another means of reducing the tritium permeation, if necessary, is to use an oxide barrier over a portion of the wall. This is possible since the protium concentration in the plasma is lower than required, being less than 0.5% over most of the lifetime as shown in Table 1. For example, using a permeation barrier over half of the wall structure would reduce the tritium permeation in half while still maintaining a low protium concentration.

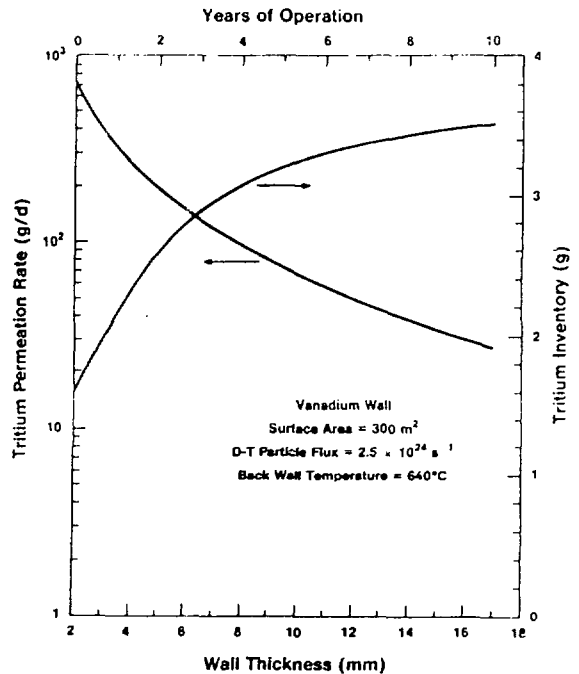


Fig. 1. Tritium permeation and inventory in a self-pumped limiter.

The tritium inventory is acceptable, being extremely small i.e., several grams for the full 10 years as shown in Fig. 1. Although the thickness of the wall increases by almost an order of magnitude over the lifetime, the inventory only increases by about a factor of 2. This is mainly because as the surface temperature increases the amount of tritium leaving the front surface facing the plasma increases and consequently this tends to reduce the permeation to the coolant.

III. HELIUM-COOLED SOLID BREEDER SYSTEM

Of the solid breeder designs considered in the Blanket Comparison and Selection Study (BCSS),³ the Li₂O breeder with the He coolant received the highest overall ranking for both tokamak and tandem mirror systems. In TPSS vanadium alloys was proposed as a structural material so that the coolant outlet temperatures can be increased to optimize the gross power conversion efficiency of the system. This will result in reducing the cost of electricity.

The high permeability of vanadium alloys to hydrogen isotopes has positive and negative consequences for the design of solid breeder

Table 1
Tritium and Protium Permeation and Inventory
Calculation for Self-Pumped First Wall- Limiter

Vanadium First Wall as Limiter
Total Wall Area = 300 m²
Back Wall Temperature = 640°C
D-T Particle Flux = 2.5 x 10²⁴ s⁻¹

Operation Time (years)	Wall Thickness (mm)	First Wall Temp. (*K)	Permeation Coefficient, c	T-Permeation (g/d)	Plasma Protium Concentration	Tritium Inventory (g)
0	2	970	.0013	697	2.5 x 10 ⁻⁴	1.6
1	3.5	1032	6.0 x 10 ⁻⁴	324	5.3 x 10 ⁻⁴	2.1
2	5	1094	3.6 x 10 ⁻⁴	197	8.8 x 10 ⁻⁴	2.6
3	6.5	1156	2.5 x 10 ⁻⁴	133	.0013	2.8
4	8	1210	1.8 x 10 ⁻⁴	96.5	.0018	3.1
5	9.5	1280	1.4 x 10 ⁻⁴	73	.0024	3.2
6	11	1342	1.1 x 10 ⁻⁴	57.3	.0030	3.3
7	12.5	1404	8.5 x 10 ⁻⁵	46.2	.0037	3.38
8	14	1466	7.0 x 10 ⁻⁵	38.1	.0045	3.44
9	15.5	1528	5.9 x 10 ⁻⁵	31.9	.0054	3.47
10	17	1590	5.0 x 10 ⁻⁵	27.2	.0063	3.5

blankets. By using a vanadium alloy for breeder cladding, tritium can be allowed to permeate the cladding from the breeder to the primary helium coolant which then acts as the medium for both tritium recovery and heat transport. To reduce the tritium inventory in the solid breeder, D₂ is maintained in the coolant at the level of 10-100 Pa partial pressure. The high hydrogen isotope pressure and the large permeability of vanadium alloys may result in excessive leakage to the plasma. Two design innovations were used to solve the problem associated with leakage to the plasma. First, the coolant side of the first wall is coated with a thin layer (~1 μm) of tungsten which acts as a diffusion barrier for hydrogen isotopes. Calculations are then performed to ensure that this design change does not induce excessive hydrogen embrittlement. Second, the helium coolant is doped with deuterium rather than protium for the purposes of tritium recovery from the solid breeder.⁴ Deuterium is chosen over protium because it is a fuel for the D-T reaction. Thus, proper design of the first wall can reduce the D and T leakage to the plasma to be less than the rate of consumption of D and T in the plasma.

A. Hydrogen Isotope Permeation and Inventory

Because of the deuterium added to the helium coolant, its permeation through the

first wall could be a serious feasibility issue. Another factor of concern is the hydrogen isotope concentration in the vanadium first wall. If the concentration exceeds a certain limit, embrittlement of vanadium may occur.

To model this problem, the first wall is subjected to two simultaneous boundary conditions. The boundary near the plasma side is exposed to an intense D-T particle flux while the back surface facing the helium coolant is subjected to the deuterium partial pressure in the helium coolant. The model assumes a deuterium or tritium atom implantation flux J_i to a depth δ in the front wall facing the plasma which is less than the wall thickness d.

It is assumed that gas molecules leave either surface by recombination-limited desorption according to Eq. (3). On the coolant side, it is assumed that nonequilibrium flow exists at the wall surface, i.e., recombination-limited desorption and dissociation-limited adsorption. The molecular adsorption rate J_d is related to the deuterium gas pressure P above the surface by a dissociation coefficient K_d according to

$$J_d = K_d P \quad (8)$$

where

$$K_D = 2K_S^2 K_r \quad (9)$$

and K_S is the Sievert's constant.

The governing equations for the surface currents can then be written as

$$J_1 = 2 K_{r1} C_1^2 \quad (10)$$

$$J_2 = \rho_{d2} p - 2 K_{r2} C_2^2 \quad (11)$$

also

$$J_1 = J_2 + J_i \quad (12)$$

where the subscripts 1 and 2 refer to the front (plasma) and the back (coolant) surfaces respectively, and J_i is again the implantation flux. The surface current J_2 then represents the net deuterium flux going into the wall (and then to the plasma) from the helium coolant. The above equations were solved numerically using efficient iteration techniques to yield very accurate solutions.

B. Analysis of Vanadium Alloy with Coated Tungsten

The primary purpose of coating the coolant side of the first wall is to limit the leakage of hydrogen isotopes to levels acceptable for plasma operation. A secondary advantage of the coating may come from inhibiting the rate of oxidation of V-alloys due to O_2 and H_2O impurities in the helium coolant.

The vanadium (V) wall coated with a very thin layer of tungsten (W) can mathematically be treated as if the wall were consisted of pure vanadium with an "artificial" sticking coefficient, α_{eff} , in the surface recombination factor K_{rv} . This artificial sticking coefficient can be deduced as follows.

The hydrogen isotope concentration in the material (C) is given by Sievert's solubility law:

$$C = K_S \sqrt{P} \quad (13)$$

and

$$K_S = K_{s0} e^{-E_S/KT} \quad (14)$$

where

$$K_S = \text{Sievert's solubility constant}$$

$$P = \text{pressure of hydrogen isotope}$$

At the interface between the coating and the bulk material, the pressure should be the same, then

$$\frac{C_w}{K_{s_w}} = \frac{C_v}{K_{s_v}} \quad (15)$$

where the subscripts w and v are for tungsten coating and vanadium first wall respectively.

The current of hydrogen isotope leaving the coated surface is

$$J = 2 K_{r_w} C_w^2 \quad (16)$$

substituting from (15)

$$J = 2 K_{r_w} \left(\frac{K_{s_w}}{K_{s_v}}\right)^2 C_v^2 \quad (17)$$

rearranging

$$J = 2 \left(\frac{K_{r_w}}{K_{r_v}} \cdot K_{r_v}\right) \left(\frac{K_{s_w}}{K_{s_v}}\right)^2 C_v^2 \quad (18)$$

or

$$J = \alpha_{eff} \cdot 2 K_{r_v} C_v^2 \quad (19)$$

where

$$\alpha_{eff} = \left(\frac{K_{r_w}}{K_{r_b}}\right) \left(\frac{K_{s_w}}{K_{s_v}}\right)^2 \quad (20)$$

Substituting values for K_s and K_r with proper units and rearranging, yield

$$\alpha_{eff} = \alpha \left(\frac{\rho_v}{\rho_w}\right)^2 \frac{1}{e} (E_x^w - E_x^v)/KT \quad (21)$$

where ρ is the density of the material and E_x is defined in Eq. 4.

Then the current of hydrogen isotope leaving the tungsten coated vanadium surface can be given as

$$J = 2 K_{r_v}^{eff} C_v^2 \quad (22)$$

where

$$K_{r_v}^{eff} = \frac{4 \alpha_{eff}}{K_{s_0}^2 \sqrt{\pi KMT}} e^{(2E_s^v - E_x^v)/KT} \quad (23)$$

So the hydrogen isotope flux leaving the coated surface can be given as that leaving the original surface with no coating but with effective recombination coefficient given by the above equation.

IV. RESULTS AND DISCUSSIONS

The calculations presented in this section are for a vanadium wall material assuming 100% first wall as a limiter. The deuterium permeation rate from the helium coolant to the plasma side is shown in Fig. 2 for different deuterium partial pressures in the coolant.

The calculation is shown for both a pure vanadium wall and for vanadium coated with a very thin layer of tungsten on the coolant side. This thin layer acts as a permeation barrier for the deuterium in the coolant. As can be seen from Fig. 2, the permeation of deuterium (with partial pressure in the required range of $10 + 100$ Pa in helium coolant) through an uncoated vanadium wall is in the 10's of kg/d which is clearly unacceptable. However, for a tungsten-coated vanadium, the permeation rate to the plasma side is about 2 + 3 orders of magnitude lower which may be acceptable from the design point of view. The tritium inventory in the wall is on the order of a few grams in the entire wall.

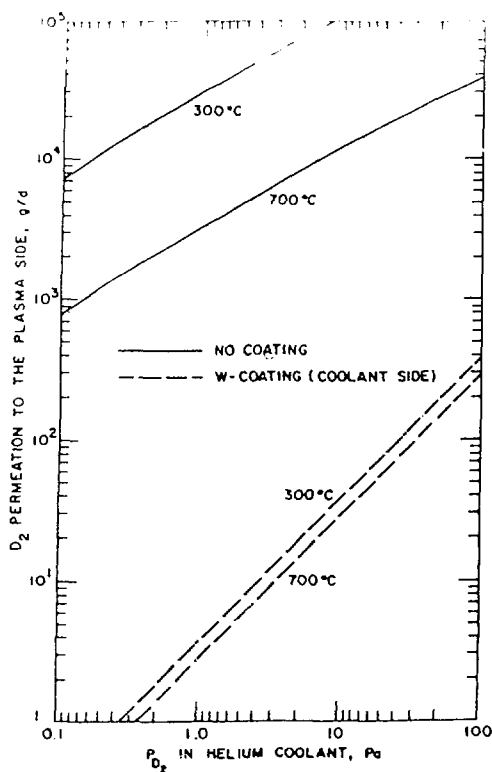


Fig. 2. Permeation of deuterium from the He coolant to the plasma for bare V and coated wall.

Another important factor is the hydrogen isotope concentration in vanadium. If the concentration exceeds a certain limit, it may cause embrittlement of vanadium and subsequent failure of the structure. The embrittlement of V and V-alloys may become a problem at hydrogen concentrations of ~ 100 wppm.³ As the primary hydrogen isotope in the vanadium will

be deuterium for this reactor design, a design limit of 200 wppm is imposed. Figure 3 shows the maximum hydrogen isotope (D) concentration in the vanadium wall for different deuterium partial pressures in the coolant. For the case of vanadium with no coating, and at the lower temperature portions of the wall, the concentration of the hydrogen isotope exceeds 200 wppm which may cause embrittlement in the required range of partial pressures. However for the higher temperature portions of the wall, the hydrogen isotope concentration is lower than 200 wppm. For tungsten-coated vanadium the concentration is well below the limit that may cause embrittlement for both temperature limits: For bare V, deuterium is the primary hydrogen isotope in solution, while for coated V, about half of the concentration is T and the other half is D. Table 2 summarizes the permeation and the concentration calculations along with the reference values used in the calculations. As a general conclusion it seems that a perfect thin layer of tungsten coating on the vanadium wall near the coolant side will substantially reduce both the deuterium permeation through the wall and the hydrogen isotope concentration in the vanadium wall to an acceptable limit.

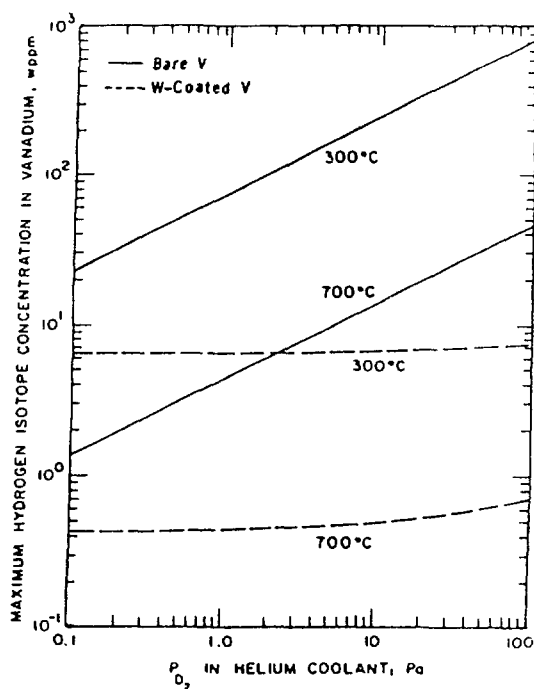


Fig. 3 Maximum hydrogen isotope concentration in the first wall as a function of D_2 partial pressure in the He coolant.

Table 2
Hydrogen Isotope Concentration in Vanadium

DT particle flux = 2.5×10^{24} #/s (100% first wall as a limiter)
Wall thickness = 2.5 mm
Wall area = 300 m²

		(P _{D₂}) Pressure (Pa)	0.1	1.0	10.	100.
T = 700°C	without coating	concentration (wppm) ^d	1.4	4.3	13.6	44
		permeation (g/d) ^b	7.9×10^2	3.4×10^3	1.2×10^4	3.7×10^4
T = 700°C	with W coating	concentration (wppm) ^c	.44	.45	.48	.70
		permeation (g/d) ^b	.25	2.76	27.8	279
T = 300°C	without coating	concentration (wppm) ^d	23	73	231	732
		permeation (g/d) ^b	7.38×10^3	2.96×10^4	10^5	3.22×10^5
T = 300°C	with W coating	concentration (wppm) ^c	6.48	6.49	6.56	7.3
		permeation (g/d) ^b	.34	3.6	36.2	363

^a Concentration of O₂ due to P_{D₂}

^b Flux of O₂ to plasma due to P_{D₂} in coolant (back diffusion caused by T-implantation not included)

^c Concentration of D and T

The above analysis was performed for a pure V first wall and a "perfect" W coating. The permeation rate of hydrogen isotopes through V-15Cr-5Ti might be less than it is for pure V. However this effect is overshadowed by the fact that the W coating may not act as a perfect layer. If plasma-arc-spraying is used as a fabrication method for coating the first wall, then the W coating will be brittle at the lower temperature.⁴ A better fabrication technique should be developed or a more-compatible coating would have to be explored. Aluminum is an interesting possibility as it has a low permeability and its thermal expansion coefficient is matched reasonably well with that of V. However, its low melting point, while an advantage in fabrication, would cause problems during transients. Another good possibility is V-Al alloys in which Al segregates to the surfaces and form a few monolayers on top of the vanadium. These few monolayers can act as a good permeation barrier for hydrogen isotope and can provide a better performance from the impurity control view points.⁸

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