

12. TITAN-I TRITIUM SYSTEMS

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12. TITAN-I TRITIUM SYSTEMS

12.1. INTRODUCTION

The major units in the tritium system of a fusion reactor are: (1) the the blanket tritium-recovery system, (2) plasma-exhaust processing system, (3) the fuel-processing and safety systems. The complete tritium system has to be designed under the constraints of tritium inventory, system cost, and tritium leakage rate. Significant relaxation of any one of the constraints will have a major impact on the overall design of the complete system.

In the TITAN design, plasma fueling is performed by injection of pellets. Because the separation of the D and T of the plasma exhaust is not required, the cryogenic distillation system needs only sufficient capacity to separate protium generated by the DD reaction. Since only about 1% of the plasma exhaust will be processed, the cost of the plasma-exhaust processing is much reduced and a redundant unit is affordable. A double plasma-exhaust processing system can significantly improve the reliability of the system, and the reactor tritium-fuel storage can be reduced.

The molten-salt-recovery process [1] is selected for tritium recovery from the lithium blanket. This process has been demonstrated on the laboratory scale [1] to recover tritium from lithium down to 1 wppm. Therefore, the tritium inventory in the blanket will be moderate. A unique advantage of using lithium as the primary coolant and the breeder is associated with the high tritium solubility in the lithium. For a tritium concentration of 1 wppm, the tritium partial pressure is only 10^{-7} Pa. With such low tritium partial pressure, tritium containment is usually not a severe problem. This reduces the required capacity of the room air-detritionation system and the secondary containment systems.

Sodium has been selected as the intermediate coolant for most fusion reactor designs [2]. Using the molten-salt extraction method to remove tritium from the primary coolant, the tritium partial pressure will be about $\sim 10^{-9}$ torr which is much smaller than the obtainable pressure by cold trap in sodium at 115°C [3]. Therefore, the maximum partial pressure of the tritium in the secondary coolant (sodium or lithium) and tritium permeation to the steam side would be set by the tritium pressure in the primary coolant. One advantage of sodium is the low solubility of tritium in sodium compared to that of lithium resulting in a smaller tritium inventory in the secondary loop by about

two orders of magnitude. For the TITAN-I design, it was decided that this advantage is not significant and to also use liquid lithium as the secondary coolant since this choice avoids the requirement of two separate liquid-metal systems (sodium and lithium) and the related technologies. Since tritium solubility is much higher in lithium than sodium, the TITAN-I design has a moderate amount of tritium inventory in the secondary loop.

In Section 12.2, the blanket tritium systems will be discussed, including estimates of tritium diffusion and inventory in the fusion power core (FPC) and the description of the molten-salt recovery system chosen for the TITAN-I design. Fuel-processing and tritium-cleanup and safety-related systems are discussed in Sections 12.3. Results and conclusions are summarized in Section 12.4.

12.2. BLANKET TRITIUM SYSTEM

12.2.1. Plasma-Driven Permeation

The TITAN-I plasma operates in a highly radiative regime with a high-recycling, open divertor configuration to reduce the heat flux on the divertor plates to acceptable levels. This configuration also decreases the plasma temperature at the divertor plate and at the first wall to minimize erosion. As a result, most particles impinging on the first wall have very low temperatures (5 eV or less). Vanadium alloys, such as those used in the first-wall and blanket structure (V-3Ti-1Si), are known to be susceptible to tritium permeation [4]. The potential, therefore, exists for a large plasma-driven permeation (PDP) into the first-wall coolant. At low particle energies, a phenomenon of major concern called “super-permeation” [5] has been observed. This phenomenon can be best explained by Figure 12.2-1. With a high particle energy, PDP reaches a peak about 250 s after initial implantation, but then decreases to a much lower, steady-state value. As the plasma-side surface is cleaned by the high energy particles, surface diatomic recombination and release of tritons becomes much easier, enhancing diffusion of the implanted tritons toward the plasma side. However, low energy particles will not clean the surface, hindering plasma-side re-emission and giving rise to super-permeation. In Figure 12.2-1, the SS-304 structural material has PDP by 20 eV ions 750 times higher than that by 320 eV ions. Energetic plasma ions of energies 40 eV or more are known to clean the surface and prevent super-permeation [6]. Although a small fraction of the neutral particles in TITAN-I edge-plasma have energies of this magnitude, their ability to clean the surface adequately is unclear. The energy spectrum of these neutrals, estimated by the neutral transport calculations (Section 5.5), is given in Table 12.2-I.

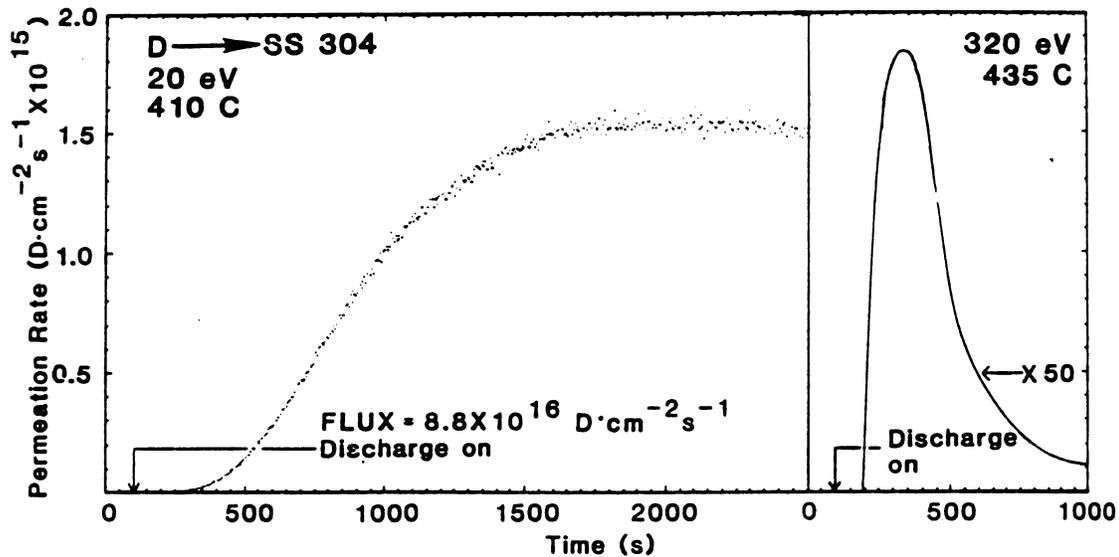


Figure 12.2-1. Typical data showing a transient permeation rate during high-energy ion implantation of stainless steel, as opposed to monotonic approach to steady state for low-energy implantation [6].

In the TITAN-I design, 95% of the tritons (both charged and neutral) have energies below 5 eV at the first wall. Two major uncertainties arise in evaluating the resulting PDP: (1) the surface cleanliness and the rate of plasma-side re-emission after implantation, as mentioned above and (2) the fraction of ions reflected from the first wall at energies below 10 eV. The fraction of hydrogen reflection off vanadium and vanadium alloys at these low energies is uncertain and implantation at this low energy range is poorly modeled. One analysis [7] shows the fraction of hydrogen, reflected off nickel, remains at 90% down to particle energies of about 3 eV. Below this energy, surface sticking and implantation of the ions increases drastically.

For evaluation of typical values of PDP, 90% reflection of the first-wall tritium flux (10% implantation) and a clean first wall (recombination sticking coefficient, $\alpha = 1$) are assumed. Incident tritium flux on the first wall is estimated to be $1.5 \times 10^{21} \text{ m}^{-2}\text{s}^{-1}$. Depth of implantation is evaluated using the TRIPOS code [8]. The PDP is evaluated using the DIFFUSE code [9] with boundary conditions of diatomic recombination and $\alpha = 1$ on both sides [2]. The first wall of TITAN-I is made of V-3Ti-1Si alloy and consists of a poloidal bank of 0.8-cm-diameter tubes with initial (beginning-of-life) wall thickness of 1.25 mm and a 0.25-cm erosion allowance for one year of operation. The

Table 12.2-I.
ENERGY SPECTRUM OF NEUTRAL DT FLUX
AT THE TITAN-I FIRST WALL

Energy (eV)	Percent of Flux
< 5	85
5 - 25	10
25 - 100	4
100 - 500	1

DIFFUSE code indicates a PDP of about 100 g/d of tritium into the first-wall coolant, increasing daily tritium extraction requirements by a manageable 25% over that required for tritium-breeding extraction.

If the plasma-side surface is dirty ($\alpha \sim 10^{-3}$ to 10^{-5}), PDP of many kg/d of tritium is possible. For example, a first-wall surface with a sticking coefficient decreased by two orders of magnitude ($\alpha = 0.01$) would increase the PDP to about 1 kg/d, requiring a larger and more costly extraction system. If in actuality such large PDP occurs and poses a design problem, a permeation barrier such as several microns of tungsten deposited by chemical-vapor deposition on the inside or outside of the first-wall tubes would reduce this permeation to acceptable levels.

In the TITAN-I design, a tungsten-rhenium alloy (W-26Re) is chosen for the divertor plates. Tungsten is very resistant to permeation and, therefore, PDP through the divertor plate is not a concern. The DIFFUSE code indicates permeation through the divertor of about 0.1 g/d of tritium, which is insignificant relative to the first-wall permeation.

The problem of PDP which is described here is not unique to compact RFP reactors. Any fusion reactor design with a combination of a low edge-plasma temperature and a vanadium-alloy first wall must consider the severity of this problem. Experiments are needed to determine the extent of plasma-driven permeation and the sputtering rate of the first-wall structure at low edge-plasma temperatures.

12.2.2. Molten-Salt Recovery Process

The molten-salt extraction process has shown considerable promise for removing hydrogen isotopes down to the ppm level for liquid lithium [1]. In this method, liquid lithium and a molten salt are in contact, and LiH is preferentially distributed to the salt phase. The salt is then electrolyzed to yield H₂ at an emf of about 0.9 V; this voltage is significantly below that of the decomposition potential of the salt (> 2 V). The hydrogen is recovered from the molten salt by sweeping the porous stainless-steel hydrogen electrode with a circulating stream of inert gas. The tritium is subsequently recovered from the inert gas with a getter.

The LiH is preferentially dissolved in the salt. The volumetric distribution coefficient,

$$D_v = \frac{(\text{wppm T in salt}) \times (\text{density of salt})}{(\text{wppm T in Li}) \times (\text{density of Li})}, \quad (12.2-1)$$

is 3.1 for protium and 3.9 for deuterium in liquid lithium. This favorable distribution coefficient is the key reason that molten-salt recovery is effective for tritium recovery from lithium.

The flow rate of the lithium which is processed per unit time, X , is given by

$$X = \left(\frac{T_b}{I_{ss}} \cdot \frac{\varepsilon D_v \eta + 1}{\varepsilon D_v \eta} \right) M_{Li}, \quad (12.2-2)$$

where T_b is the tritium breeding rate, ε is the efficiency of tritium recovery from the salt, η is the efficiency factor which accounts for nonequilibrium tritium distribution during contacting, I_{ss} is the steady-state tritium inventory, and M_{Li} is the lithium inventory.

It has generally been assumed that η is equal to unity, yielding $D_v \eta \simeq 4$. An important trade-off [10] occurs between the recovery efficiency, which determines the number of contacting units, and the average current density, which dictates the volume of the electrolysis tank. With all other parameters fixed, the efficiency of the salt processing, ε , determines the number of contacting units required to maintain a given tritium steady-state inventory. On the other hand, the average current density decreases as the recovery efficiency increases, since the average current density is proportional to the tritium concentration. Therefore, the total electrode area and the size of the electrolysis tank must be large to extract the tritium at the lower current density and still recover tritium at a rate equal to the breeding rate. The salt processing step can be operated at high recovery efficiency and low current density or the recovery efficiency can be reduced to increase the average current density. It is believed that maximizing the recovery efficiency will be

the preferred mode of operation, since the contacting units will be more complex devices than the salt processing tanks and the contacting operation will require substantially more electrical power.

As can be seen in Table 12.2-II, the processing requirements are small for 1 wppm of tritium in lithium. These requirements increase linearly as the tritium concentration decreases so that the processing requirements at 0.1 wppm are a factor of 10 higher than those given in Table 12.2-II. Furthermore, the molten-salt extraction process lends itself to a modular design in which relatively small units are connected in parallel with the liquid-lithium blanket. This design allows additional flexibility for accommodating changes in the allowable tritium inventory of the blanket. If future safety regulations, economic constraints or design difficulties require a reduction in the tritium inventory, additional processing units could be attached easily to meet the new requirements.

The lithium that is separated from the molten-salt contactor will be saturated with the molten salt. The corrosion characteristics of the lithium saturated with the salt may be different from that of the pure lithium. In the molten-salt extraction experiment at Argonne National Laboratory, a stirrer was exposed to the Li/salt mixture for 2000 h at 500°C [10]. The stirrer was in motion for approximately 1000 h and for 6 h, the temperature was 600°C. Upon inspection of the stirrer, it was discovered that: (1) the corrosion rate is nearer that expected for lithium corrosion than that for molten salt corrosion; (2) nickel is preferentially depleted from grain boundaries, as would be expected for lithium penetration; and (3) chromium appears to accumulate at the grain boundaries, again suggesting lithium penetration rather than salt penetration, since molten salts would more than likely cause chromium depletion. Therefore, it is concluded that salt has a small impact on the corrosion characteristics of lithium. However, potential effects of MHD and radiation need further investigation.

12.2.3. Tritium Diffusion

Calculations of tritium permeation from the lithium coolant through the vanadium-alloy structure were made using DIFFUSE [9], with boundary conditions of diatomic recombination on any outer surfaces (recombination coefficient $\alpha = 0.1$) and Sievert's law on the coolant side of the pipes. Sievert's law corresponds to solubility of gas phase species in an adjacent solid [9]:

$$c(x, t) = S[T(x, t), P(t)] X(t), \quad (12.2-3)$$

with c the concentration of the species in the solid at position x and time t , S the solubility of the species which is a function of temperature and pressure, and X the

Table 12.2-II.
ANALYSIS OF MOLTEN-SALT EXTRACTION SCHEME
FOR A LIQUID-LITHIUM BLANKET SYSTEM

	Breeding rate	420 g/d ^(a)															
	Recovery rate	520 g/d ^(a)															
	Lithium exit temperature	556 °C ^(b)															
	Extraction system temperature	556 °C ^(b)															
Estimated blanket inventories																	
	Lithium	2.12×10^8 g															
	Tritium	212 g (1 wppm)															
	Tritium recovery efficiency, ϵ	90%															
	Capacity per extractor unit	23 m ³ /h															
	Electrical power per unit	3.7 kW															
<table style="width: 100%; border-collapse: collapse;"> <thead> <tr> <th style="width: 20%; text-align: center;">Tritium Concentration (wppm)</th> <th style="width: 20%; text-align: center;">Effective Distribution Coefficient ($D_e\eta$)</th> <th style="width: 20%; text-align: center;">Lithium Processed per Hour (kg/h)</th> <th style="width: 20%; text-align: center;">Number of Units</th> <th style="width: 20%; text-align: center;">Required Electrical Power (kW)</th> </tr> </thead> <tbody> <tr> <td style="text-align: center;">1</td> <td style="text-align: center;">4</td> <td style="text-align: center;">22,000</td> <td style="text-align: center;">7</td> <td style="text-align: center;">26</td> </tr> <tr> <td style="text-align: center;">1</td> <td style="text-align: center;">1</td> <td style="text-align: center;">88,000</td> <td style="text-align: center;">28</td> <td style="text-align: center;">104^(c)</td> </tr> </tbody> </table>			Tritium Concentration (wppm)	Effective Distribution Coefficient ($D_e\eta$)	Lithium Processed per Hour (kg/h)	Number of Units	Required Electrical Power (kW)	1	4	22,000	7	26	1	1	88,000	28	104 ^(c)
Tritium Concentration (wppm)	Effective Distribution Coefficient ($D_e\eta$)	Lithium Processed per Hour (kg/h)	Number of Units	Required Electrical Power (kW)													
1	4	22,000	7	26													
1	1	88,000	28	104 ^(c)													

(a) Based on a tritium-breeding ratio of 1.2 and 100 g/d of PDP.

(b) Parameters of the Scoping Phase design,
the blanket and first-wall coolant were mixed in the outlet.

(c) Reference case.

mole fraction of the species in the gas phase. To evaluate the ambient tritium pressure corresponding to some concentration of tritium in liquid lithium, the following expression was employed [1]:

$$C = 17.9 P_{T_2}^{1/2} \exp(12,284/RT), \quad (12.2-4)$$

where C is the concentration of tritium in the lithium (wppm), P_{T_2} is the tritium partial pressure (torr), R is the ideal gas constant (1.987 cal/mole K), and T is the temperature (K). Given the temperature of the lithium within different regions of the blanket and piping and using a typical tritium concentration of 1 wppm in the lithium, the tritium partial pressures were calculated and input into DIFFUSE.

Tritium permeation rates through the intermediate heat exchangers into the steam-generator system is a major concern with respect to the release of the radionuclides into the environment under normal operating conditions. This permeation was estimated using an expression from Reference [2] for permeation through HT-9 tubing and an assumed oxide barrier factor of 100 on the water side of the pipes. The steam-generator piping consists of 1-mm-thick HT-9 tubing with surface area $2.7 \times 10^4 \text{ m}^2$. The permeation rate into the water for ultimate release to the atmosphere is estimated to be 6.8 Ci/d, which is less than the target release rate of 10 Ci/d to the environment. The tritium concentration within the steam system is negligible.

Out-of-blanket piping structure in the primary and secondary loops is a stabilized ferritic steel with a 15-cm inner diameter and a 3-mm wall thickness. The DIFFUSE calculations using the boundary conditions mentioned above indicate permeation through the bare pipes into containment would total about 2 Ci/d for both loops. However, insulation on the pipes acts as a barrier and reduces permeation to a negligible level. The inventory within this out-of-blanket piping structure is estimated to be less than a milligram of tritium.

The tritium inventories within the first-wall structure are evaluated by DIFFUSE, using the first-wall boundary conditions described in Section 12.2.1. The typical case of a clean first wall gives an inventory of 0.72 g, while a dirty first wall ($\alpha = 0.01$) increases the inventory to 4.53 g. The structural inventories within the balance of the fusion reactor core are evaluated using the boundary conditions mentioned above. These inventories are given in Table 12.2-III.

Table 12.2-III.
TRITIUM INVENTORIES IN THE TITAN-I REACTOR

Unit	Tritium Inventory (g)
Storage	1,100
Primary-coolant loop	212
Secondary-coolant loop	300
Molten-salt extraction	10
Fuel processing	20
First wall:	
typical case	0.72
excessive PDP	4.53
Integrated blanket coil	2.20
Hot shield, zone 1	0.14
Hot shield, zone 2	0.25
Divertor shield	0.08
Divertor	< 0.01
Out-of-blanket piping	≪ 0.01
 TOTAL TITAN-I inventory	 1,650

12.3. FUEL-PROCESSING AND SAFETY SYSTEMS

The fuel-processing loop will process the reactor exhaust gas to remove and decompose all impurities so that all of the deuterium and tritium are recovered; will perform isotopic separation of the hydrogen isotopes to provide any or all of the species D_2 , DT , and T_2 as required by the various fueling systems; and will provide safe, long-term storage of the tritium inventory during shutdown periods. The fuel-processing system will also process tritium recovered from the blanket-extraction system and from the blanket module experiments and prepare this tritium supply for use in the fueling system. These interactions are shown in the tritium flowchart, given in Figure 12.3-1.

The tritium-cleanup and safety system is necessary to provide personnel, public, and environmental protection. This system includes secondary-containment-cleanup, tritium-contaminated-waste-cleanup, and room-air-detritionation subsystems. The latter subsystem will be used to recover tritium which may be released into the confinement building and/or the reactor hall and to process the air in the hot cell and the decontamination area. Other safety and environmental protection systems are included as part of the tritium-cleanup system (*e.g.*, tritium-monitoring equipment and modification to the ventilation system to accommodate maintenance operations in the tritium system).

Tritium-related subsystems for accomplishing the fuel processing and tritium cleanup and safety include the following:

- Tritium receiving and storage
- Fuel cleanup
- Isotope separation
- Fuel preparation
- Transfer and circulation pumps
- DT fueling
- Glovebox-atmosphere detritionation
- Room-air detritionation
- Tritium-waste treatment
- Tritium monitoring

- Breathing air
- Liquid-waste disposal
- House vacuum
- Flexible-ventilation duct
- Tritium data acquisition and control

The tritium subsystems must maintain the concentrations of the airborne tritium below $20 \mu\text{Ci}/\text{m}^3$ in spaces occupied by unprotected operating personnel and below $4000 \mu\text{Ci}/\text{m}^3$ for protected (in bubble suit) operating personnel. This assumes a protection factor of 200 for the bubble suit.

12.3.1. Fuel-Processing System

The fuel-processing system will process all gases exhausted from the plasma chamber, tritium recovered from the blanket and coolant streams, and the fueling-system gas-recovery system. The impurities which are expected to be in the exhaust gas and which must be removed are shown in Table 12.3-I. The fuel-processing system must not only remove these impurities, but also decompose them to recover the contained deuterium and tritium.

Key parameters of the fuel-processing system are summarized in Table 12.3-II and the capital costs for fuel-processing and associated safety systems are given in Table 12.3-III. The subsystems of the fuel-processing system are discussed below.

12.3.1.1. Fuel-cleanup system

A schematic flow diagram of the proposed fuel-cleanup (FCU) system for the TITAN reactor is shown in Figure 12.3-2. The exhaust gas from the plasma chamber first enters a palladium-alloy membrane diffuser. The hydrogen isotopes rapidly permeate through the alloy while impurities are unable to pass through. This gives an extremely pure hydrogen isotope stream to feed the isotope separation system (stream 3). A bleed stream containing the impurities is continuously removed from the diffuser (stream 4).

If these impurities contain hydrogen isotopes (off-normal condition), this stream passes over a heated catalytic-oxidation bed where the reactive impurities are oxidized

Table 12.3-I.
ESTIMATED CONTAMINANT-GAS CONCENTRATION
IN THE VACUUM-PUMP EXHAUST OF TITAN REACTOR

Gaseous Component	Atomic %	Probable Chemical Species
A. Normal Operation		
D	~ 45	DT
T	~ 45	DT
He	5-10	He
H	1-2	HD, HT
Xe	≪ 1	Xe
B. Off-Normal Operation (Air Contamination)		
D	~ 45	DT
T	~ 45	DT
He	5-10	He
H	1-2	HD, HT
Xe	≪ 1	Xe
N	2	N(D,T) ₃
O	< 1	DTO
Ar	≪ 1	Ar

Table 12.3-II.
PARAMETERS OF THE FUEL-PROCESSING SYSTEM

Tritium flow rate to fuel processing	5,920	g/d
Deuterium flow rate to fuel processing	3,950	g/d
Tritium inventory in processing	20	g
Volume of fuel-processing equipment	1,500	m ³
Power required for processing (maximum)	830	kW

Table 12.3-III.
CAPITAL COSTS OF THE FUEL-PROCESSING SYSTEM

Subsystem	Cost, k\$ (1986)
Fuel cleanup, isotope separation	3,225
Glovebox detritiation	1,525
Data acquisition and control	2,315
Emergency power	1,180
Transfer pumps, storage	690
Tritium monitors, gas analysis	940
Secondary containment, waste disposal, inventory control	475
TOTAL	10,350

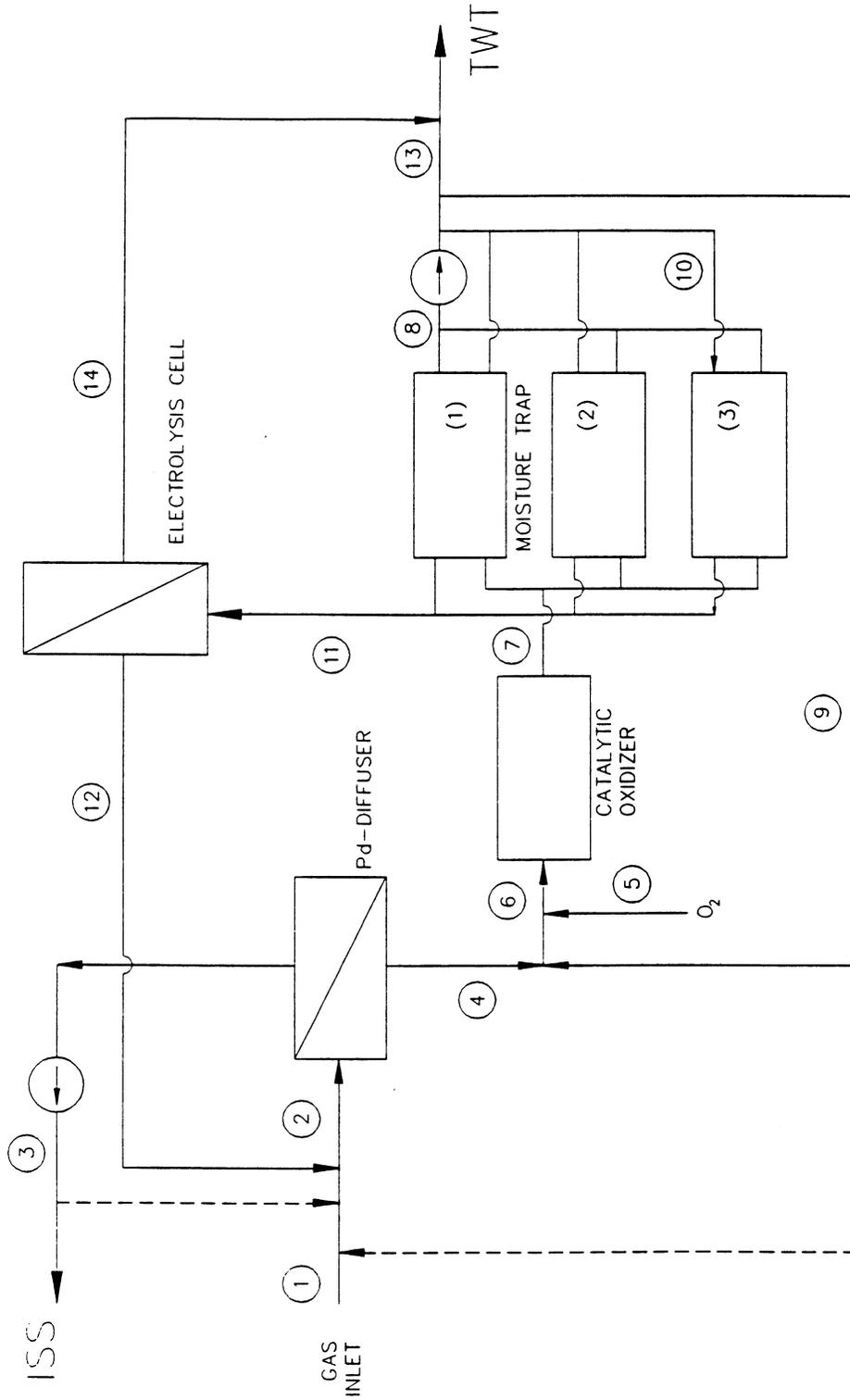


Figure 12.3-2. Flow diagram of the fuel-cleanup system.

and all hydrogen isotopes end up as water. This water is collected in the molecular-sieve moisture traps while the nontritiated impurities (He, CO₂, N₂, Ar) pass through the moisture trap to the tritium-waste-treatment system. Periodically the moisture traps are heated and the water released is passed to a ceramic electrolysis cell (stream 11) where the water is electrolyzed. The released oxygen permeates through the ceramic (stabilized zirconia) and is fed to the tritium-waste-treatment system (stream 14). The hydrogen isotopes released in the electrolysis cell (stream 12) are fed back to the inlet of the palladium membrane and ultimately to the isotope-separation system (stream 3). This FCU is based on components developed by the Japan Atomic Energy Research Institute [11] and tested with tritium at the Tritium System Test Assembly (TSTA) [12].

Tritium recovered from the blanket is fed into the main inlet stream to the FCU subsystem (stream 1). Some additional preprocessing of this tritium may be required, depending the nature of the breeding material in the blanket modules. For instance, if the tritium released from the breeder is in the form of HTO, it may be profitable to decompose the HTO before it enters the main fuel-cleanup unit.

The FCU is designed to recover > 99.9% of all hydrogen isotopes in the plasma exhaust gas. The impurity level in the FCU output stream is of the order of 1 part per million.

12.3.1.2. Isotope-separation system

A stream of the purified hydrogen isotopes from the fuel-cleanup system is sent to the isotope-separation system (ISS). The ISS is based on fractional distillation of the hydrogen species at cryogenic temperatures, a well developed technology [12]. The distillation columns are packed with commercial, stainless-steel packing material and are operated at ~ 20 K.

The ISS is designed basically in accordance with the TSTA [12] design concept and equipment. Each column is a few centimeters in diameter and 3 to 4 m in height. The refrigeration unit required to provide helium gas coolant at 20 K is contained within the isotope-separation package.

12.3.1.3. Fuel-preparation system

The fuel preparation system consists of a series of transfer pumps, calibrated volumes, and small holding tanks. Feed gas for this system comes from the tritium storage beds, deuterium storage tanks, and the isotope-separation unit.

12.3.1.4. Transfer and circulation pumps

A number of circulation and transfer pumps are required in the fuel processing system. These pumps are used to provide the driving force and pressure differentials to move the gas stream through the fuel-cleanup, isotope-separation, and fuel-preparation systems. These pumps must be all-metal, dry, tritium-compatible pumps. The pumps selected are manufactured by the Metal Bellows Corp., Sharon, Massachusetts [13] and have been in tritium service at various laboratories for several years. The Metal Bellows pump is a completely tritium-compatible alternative for the low-vacuum region. The vacuum chamber of this pump is carefully sealed from the atmosphere and from all moving mechanical parts. The Metal Bellows pump is limited by low throughput when a high compression ratio is required, hence, using more pumps with lower compression ratios in series is preferred.

The second transfer pump is a tritium-compatible, orbiting scroll pump manufactured by Normetex (France) [14]. Standard sizes for these pumps range from 15 to 1300 m³/h. Operation of these pumps relies on the relative motion of two spiral vanes: one is stationary, while the other moves in a circular orbit. The chambers are sealed from both the atmosphere and from all lubricated parts by a bellows seal. The two spiral vanes are noncontacting so no lubricant is required. As a result, the pump does not contaminate the pumped gases and there is no tritium-contaminated pump oil to contend with. The Normetex scroll pumps can be operated from a few Pa to atmospheric pressure. Although the scroll pumps are more efficient in the low vacuum region than in the intermediate region, they are most useful in the intermediate region because of the lack of other tritium-compatible pumps for this pressure regime.

12.3.1.5. Fuel-storage system

The fuel in the fuel-storage system consists of unburned fuel and bred tritium recovered by the ISS, as well as new fuel received from off-site sources. The isotopic components of the fuel (D₂, DT, and T₂) are fed to the fueling systems during operation of the reactor. The storage of the DT and T₂ fuel components is on a metal matrix. The metal storage beds consist of double-jacketed vessels containing a metallic getter in the form of sponge, powder, or turning. The gettering material must have good thermal stability and hydrogen-release properties. There are several potentially suitable gettering materials such as uranium, zirconium-cobalt alloy, or LaNi_{5-x}Al_x alloys. These metals all have a low dissociation pressure at room temperature. The temperature required to

reach a dissociation pressure of one atmosphere is in all cases 400°C or less. The metal storage beds cannot help in the removal of decay helium from the fuel. Because helium is not adsorbed on a metal getter bed, tritium can be transferred from one bed to a second bed. Any helium in the one atmosphere of the gas will remain gaseous and, therefore, can be pumped away by a small vacuum pump. About 5 kg of $\text{LaNi}_{5-x}\text{Al}_x$, 37 kg of ZrCo , or 115 kg of uranium metal are needed to store 1 kg of deuterium or 1.5 kg of tritium.

12.3.1.6. Redundancy of fuel-processing subsystems

To ensure a highly reliable fuel supply at a minimum cost, certain key fuel-processing subsystems should be duplicated. These key systems are the fuel-cleanup system (FCU), the isotope-separation system (ISS), the transfer pumps, the fuel-storage system (FSS), the associated gloveboxes, tritium monitors, and controls. The cost for these redundant systems is \$4.5 million. This cost is very small compared to the cost of storing even one day's throughput of standby tritium, which would amount to over \$60 million.

12.3.2. Tritium-Cleanup and Safety System

The tritium-cleanup system includes the safety subsystems for radiation and tritium monitoring, protection of the public and employees against exposure to radiation, and preparation of waste tritium for shipment off-site for reprocessing or safe disposal. Systems for both normal operations and emergency situations are discussed below.

12.3.2.1. Glovebox- and room-atmosphere-detrition systems

Tritium may be accidentally released into gloveboxes or the room atmosphere. In either case, the tritium will be purged from the atmosphere and solidified on a matrix as an oxide for eventual recovery or disposal. Two detrition subsystems are planned: one with a capacity of 50 cfm (cubic feet per minute) for continuously processing glovebox atmospheres, and another larger system for processing room atmospheres.

The smaller glovebox-atmosphere-detrition system (GBADS) will be a batch system consisting of redundant blowers, catalysts with preheaters and heat exchangers, and desiccant beds. This redundancy may be achieved by a single subsystem with parallel components or by two separate and independent units, each with a capacity of 25 cfm. The cost difference between the two configurations is not expected to be significant. In

Table 12.3-IV.
ROOM-AIR-DETRITIATION SYSTEM (RADS)
DESIGN ASSUMPTIONS

Ideal cleanup time ^(a)	3 d
Tritium spill size	50 g
Tritium concentration after cleanup ^(b)	20 $\mu\text{Ci}/\text{m}^3$

(a) No absorption on walls and surfaces. Surface absorption will extend cleanup time.

(b) Perfect mixing in room air. Imperfect mixing may allow shorter cleanup time by selective use of portable ventilation ducting.

either case, the maximum capacity of 50 cfm can be achieved with both blowers in operation. Common to the systems will be two large receiving tanks, normally kept below atmospheric pressure. Four desiccant beds are to be used, two for each of the units if the separate unit configuration is adopted. A separate regeneration unit will be available for drying the desiccant beds as needed. A possible alternative to the above-described method for removing tritium from inert glovebox atmospheres is the adsorption through chemical bonding of elemental tritium by metal "getters." This technology is currently under development and is not available at the present time.

The large room-air-detritiation system (RADS) will consist of three separate and independent 5000-cfm units that can operate individually or in parallel, depending on the size and the number of the rooms to be cleaned up (Figure 12.3-3). The rooms to be ducted to the RADS include the test cell, hot cell, warm cell, tritium facility, and GBADS and RADS rooms. The RADS unit capacity was determined by the requirement (Table 12.3-IV) that it be capable of reducing the tritium concentration in the TITAN test cell ($170,000 \text{ m}^3$) following a release of 50 g to the derived-air concentration (for HTO in air) of $20 \mu\text{Ci}/\text{m}^3$ in 72 h, which is satisfactory for worker exposure with no protection.

The relationship of system cost to the detritiation time is shown in Figure 12.3-4. Each unit (Figure 12.3-5) comprises a blower, a catalyst with heater and heat exchanger, and two redundant desiccant beds. A cooler/refrigerator dryer may be incorporated to

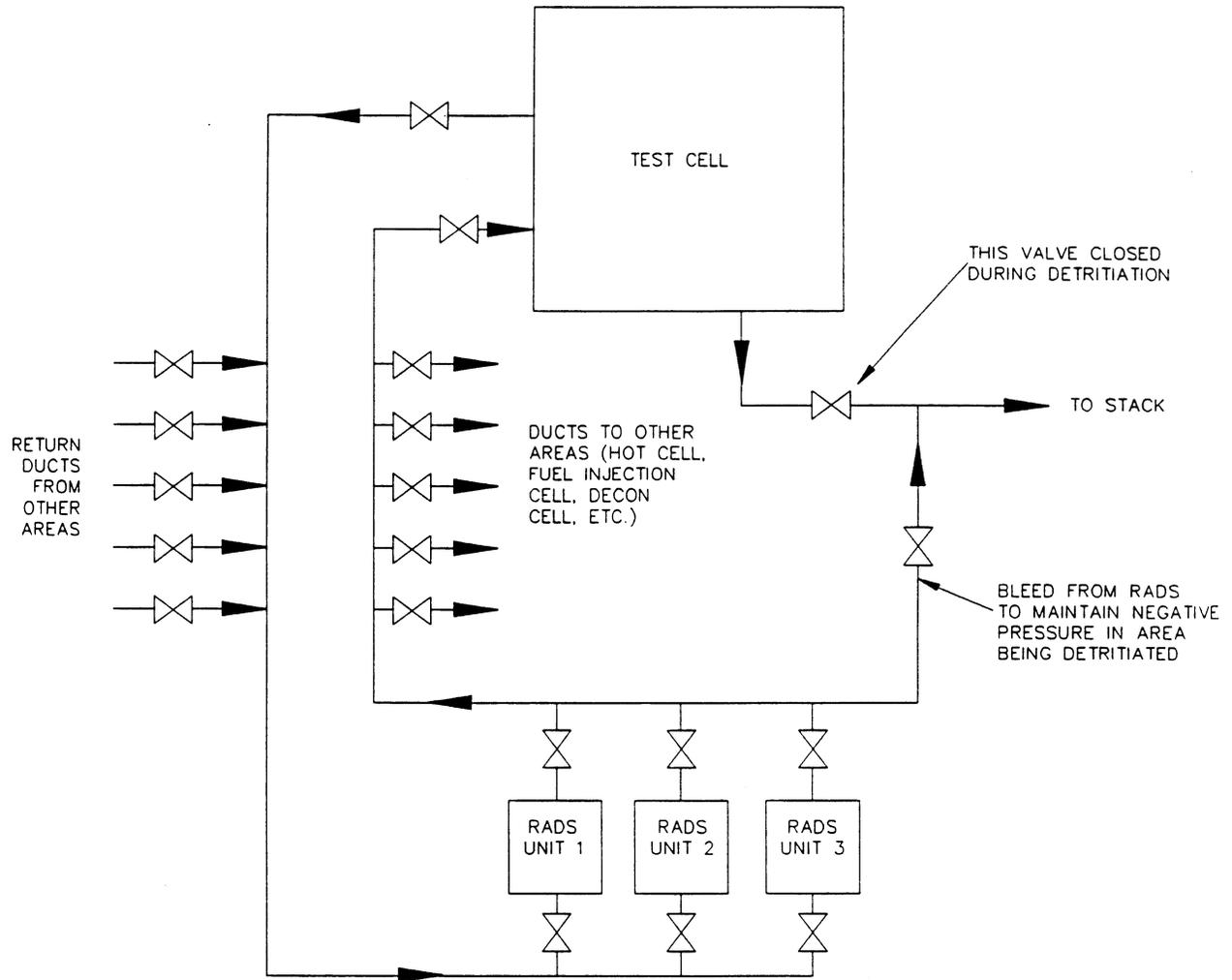


Figure 12.3-3. Room-air-detritiation system (RADS).

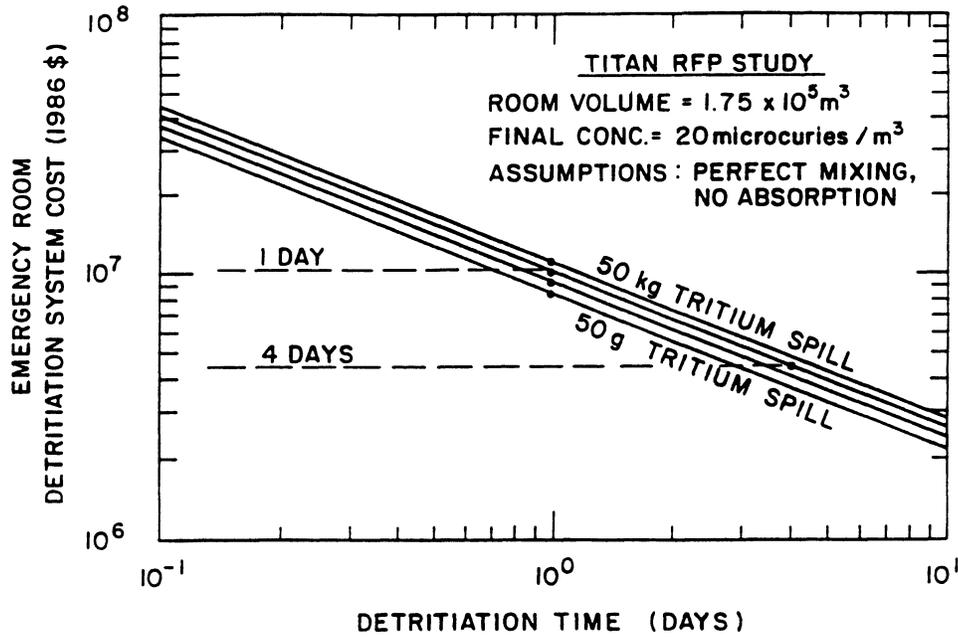


Figure 12.3-4. Cost of the detritiation system as a function of room detritiation time.

reduce the size of the desiccant beds. Alternatively, the components of the units may be linked together so as to provide component redundancy as opposed to unit redundancy, at little increase in cost. Regeneration of the desiccant beds will be done by unloading and replacing with dry desiccant. Once removed, regeneration of the moist desiccant may be accomplished by the GBADS regeneration unit, or alternately, it may be treated as waste. Table 12.3-V summarizes the parameters of the RADS equipment.

Table 12.3-V.

**PARAMETERS OF
 THE ROOM-AIR-DETRITIATION SYSTEM (RADS)**

Equipment capital cost	\$4,350,000	(1986\$)
Equipment volume	3,000	m^3
RADS power requirement (maximum)	1,200	kW

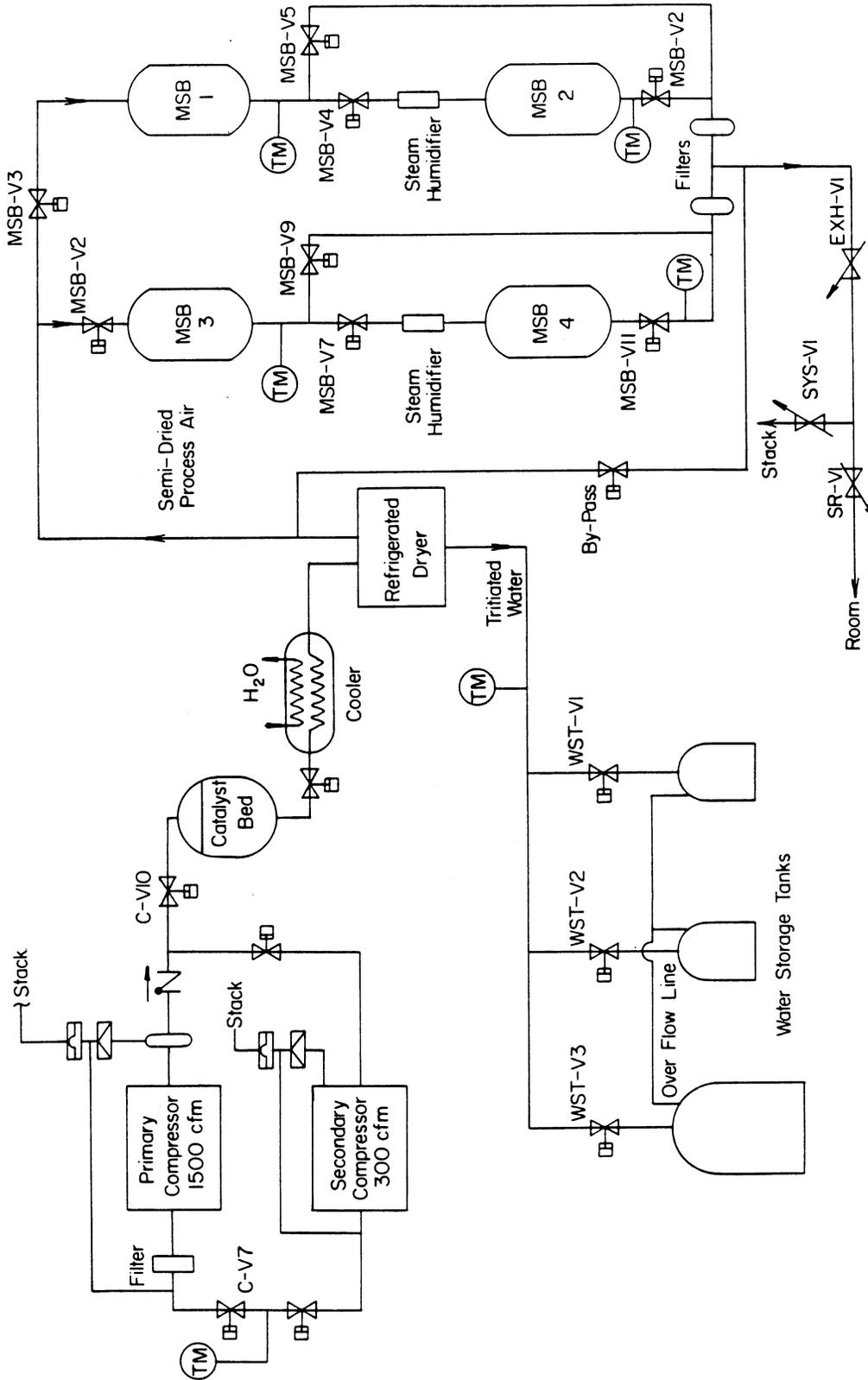


Figure 12.3-5. Layout of a room-air-detritiation system.

12.3.2.2. Tritium-waste-treatment system

The primary purpose of this system is to receive and process the effluent gas from gloveboxes. In addition, other gaseous effluents generated throughout the facility (such as vacuum-pump exhausts) that could possibly contain tritium will be processed by this system. Tritium removal is based on the catalytic oxidation of tritium-containing compounds and subsequent removal of the tritiated water by adsorption on molecular sieve beds. The decontamination factor (ratio of the tritium input to tritium exhausted to environment) will be better than 10^5 . The system includes holding tanks, blowers, catalyst bed, and molecular sieve beds. The tritiated water is recovered from the molecular-sieve beds by regeneration with a heater-purge gas. The water is collected and packaged for shipment to an external disposal or reprocessing facility.

12.3.2.3. Radiation monitoring system

The radiation-monitoring system involves monitoring for safety as well as for the purpose of diagnostics and process control. The instruments to be used include gamma and neutron monitors as well as tritium monitors. Many are environmental (*i.e.*, room, stack, or duct) monitors whereas others are for monitoring glovebox atmospheres or processes for tritium concentrations.

Approximately 90 to 100 fixed, active tritium monitors are projected for the TITAN facility. Tritium instruments are of the flow-through ionization-chamber type and are capable of measuring tritium concentrations from a few $\mu\text{Ci}/\text{m}^3$ to about one Ci/m^3 , the actual range depending on the application. Some selected monitors will be capable of discriminating between activated-air products, tritium gas, and tritium oxide. This type monitor is somewhat more expensive, but the safety implication of knowing the form (gaseous or oxide) is quite significant, because the oxide is about 25,000 times more hazardous.

Monitors will be needed in all gloveboxes, handling rooms for activated and/or contaminated equipment (*e.g.*, hot cell, decontamination room) and for the rooms containing the tritium-cleanup subsystems (glovebox-air detritiation, tritium-waste treatment, and emergency room-air detritiation). All of these proposed instruments are fixed or transportable and do not include portable instruments and passive monitoring devices such as thermoluminescent dosimeters or neutron- or gamma-sensitive film.

12.3.2.4. Breathing-air system

The test cell and tritium-handling areas will be equipped with a supplied breathing-air system. Personnel will use breathing air for emergency access to areas where tritium has been accidentally released and for maintenance operations where there is the possibility that significant tritium may be released into the room.

The major components of the system are supplied air suits and connecting air hoses, breathing-air stations located throughout the facility, a pressure regulating and distribution system, and a source of breathing air. As a source for breathing air, either a compressor and a holding tank or a tube trailer filled with breathing air (approximate capacity 45,000 scf) will be used. One or more portable breathing air stations (on carts) will also be available, either as an alternative or a supplement to the fixed system.

12.3.2.5. Liquid-radioactive-waste system

This system includes all the piping, drains, pumps, and storage vessels necessary to collect liquid radioactive wastes, and to prepare for shipment off site for disposal. These wastes may be generated as a result of normal operations (washing, decontamination, *etc.*) or accident conditions which may, for example, produce water used for extinguishing a fire. The waste may be tritiated or may contain other radioactive materials. The system will normally handle only water. Other radioactive liquids will be packaged separately.

12.3.2.6. House-vacuum system

This system will provide a service vacuum to evacuate tritium-contaminated or potentially contaminated lines located throughout the tritium areas. Components will include an oil-free vacuum pump ($\sim 200 \text{ m}^3/\text{h}$ in size), and piping and outlets distributed in areas in which tritium will be handled. The pump will discharge to the tritium-waste-treatment system.

12.3.2.7. Flexible-ventilation-duct system

The flexible-ventilation-duct system will be used to provide local ventilation to areas where tritium has been or may be released into the room air because of an accident or a planned maintenance operation. Examples of uses are maintenance on the plasma

chamber through an open port or removal of a glovebox window. The exhaust of the flexible ventilation duct can be either routed to the stack, or if the tritium concentration in the air is higher than allowable release levels, to one of the tritium cleanup systems.

12.3.2.8. Local instrumentation and control system

The tritium system will have its own control system that will interact with the main control system in a supervisory manner. Several options are available for controlling the tritium subsystems. Either a central tritium-control system can monitor and control the separate subsystems, or distributed processors can control the individual subsystems with a central tritium-control computer having supervisory monitoring and control functions. The latter method is preferable. Manual (computer independent) control of each of the tritium subsystems may be included. The cost and space requirements for these options are similar.

12.4. SUMMARY AND CONCLUSIONS

The parameters of TITAN-I which are important to the design of the tritium systems are summarized in Table 12.4-I. A flowchart of the TITAN-I tritium systems is shown in Figure 12.4-1. The tritium throughput and inventory of each component has been calculated, as can be seen in Figure 12.4-1 and Table 12.2-III. The cost of the TITAN-I tritium system is summarized in Table 12.4-II. The cost estimates are based on the Tritium System Test Assembly scaling [15] and Blanket Comparison and Selection Study scaling [2]. The tritium inventory (1650 g), the leakage rate (7 Ci/d), and the tritium-system cost (\$35M) are all very reasonable.

A potential critical issue facing TITAN-I is the plasma-driven permeation of low energy tritons through the permeable vanadium-alloy first wall. The ability of the small fraction of high-energy plasma ions to adequately clean the first-wall surface is uncertain. Any fusion reactor design with low edge-plasma temperature may face similar problems.

An important feature of the TITAN-I design is the cryogenic distillation of only 1% of the plasma exhaust. Because of the associated cost reductions, redundant units are used in the plasma-exhaust-processing system to reduce significantly the tritium fuel storage (and cost) requirements.

Table 12.4-I.
TITAN-I TRITIUM PARAMETERS

Fusion power	2,301	MW
Thermal power	2,935	MW
Net electric output	970	MWe
Plasma tritium burnup fraction	5	%
Tritium-breeding ratio	1.2	
Tritium production rate	420	g/d
First-wall plasma-driven permeation	100 - 1000	g/d
Tritium extraction rate	520 - 1500	g/d
Tritium concentration in lithium	1	wppm

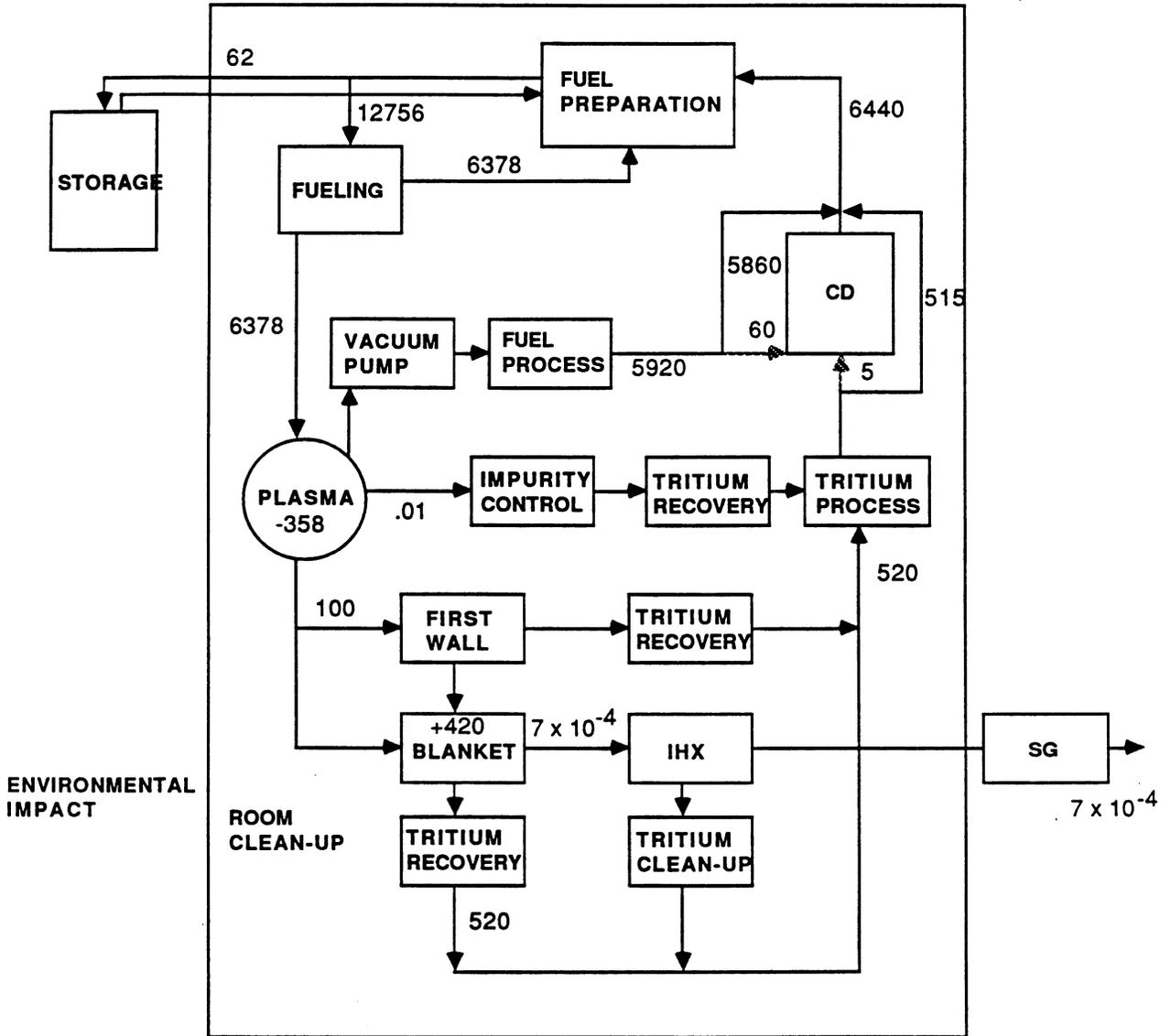


Figure 12.4-1. Flowchart of the TITAN-I tritium systems.

Table 12.4-II.
TITAN-I TRITIUM SYSTEM CAPITAL COSTS

Subsystem	Cost (k\$, 1986)
Fuel processing:	
Fuel cleanup, isotope separation	3,225
Glovebox detritiation	1,525
Data acquisition and control	2,315
Emergency power	1,180
Transfer pumps and storage	690
Tritium monitors and gas analysis	940
Secondary containment, waste disposal, and inventory control	475
Backup unit for fuel processing	4,500
TOTAL	14,850
Room-air detritiation	4,350
Molten-salt extraction (typical case)	5,000
Molten-salt extraction (excessive PDP)	15,000
 TOTAL cost for TITAN-I (typical case)	 24,200
TOTAL cost for TITAN-I (excessive PDP)	34,200

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