

18. TITAN-II TRITIUM SYSTEMS

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18. TITAN-II TRITIUM SYSTEMS

18.1. INTRODUCTION

In this section, specific tritium-system design issues that relate to the use of the aqueous-salt blanket coolant in TITAN-II are considered. A critical issue is the need to recover and control tritium that is bred directly in the water coolant of the primary heat-transport system. In particular, tritium recovery from water is required on a scale larger than existing water-detrification systems. However, there is considerable industrial experience with recovery of hydrogen and its isotopes from water, and some relevant process equipment is used on a larger scale in non-tritium applications.

The TITAN-II design considers higher tritium levels (50 Ci/kg) in the primary-coolant water relative to previous design studies (*e.g.*, 1 Ci/kg in the "Blanket Comparison and Selection Study," BCSS [1]) in order to minimize the cost of water-processing equipment required for tritium recovery. Design features which make this possible are: (1) lower pressure in the primary system resulting from an elevation of the fluid boiling point caused by the addition of the Li salt, (2) possible use of double-walled steam generators, (3) presence of the water pool which captures a large part of the tritiated-water leakage, (4) routine use of welded joints, and (5) removal of tritiated water to safe storage during major maintenance operations. Component leakage rates and air-drier technology are based on CANDU systems performance. The overall tritium-loss rate for the TITAN-II design is estimated at 50 Ci/d.

The reference design for the blanket tritium-recovery system is based on a tritium level of 50 Ci/kg in the primary water coolant. This system recovers 430 g/d of tritium, primarily through a five-stage vapor-phase catalytic-exchange (VPCE) system, which transfers the tritium from the water to hydrogen gas, and then by cryogenic distillation for isotope separation. The TITAN-II fusion power core is submerged in the pool of water to achieve a high level of safety. The water pool contains tritium from primary-coolant system leakage, which is maintained at 0.37 Ci/kg by water distillation, with the enriched tritiated water from the distillation columns mixed with the primary-coolant water for final tritium recovery. The water-feed rate to the VPCE system is about 4000 kg/h at 50 Ci/kg. The estimated installed cost of the TITAN-II tritium-recovery system is 130 M\$ (1986), not including building, air-cleanup, and indirect costs. Although the water-feed rate is about 10 times larger than the Darlington Tritium-Removal Facility,

the cost is only 3 to 4 times larger because of the economy of scale, fewer VPCE stages, and the lower reflux ratio needed in the cryogenic columns by the light-water feed.

The other TITAN-II tritium-related systems and flow rates are also assessed. The fuel-processing systems are similar to those of TITAN-I, which are described in Section 12. Unique features include a redundant impurity-removal loop (rather than relying on large tritium storage capacity) and a small feed to the isotope-separation system because of the use of mixed DT fueling. Plasma-driven permeation is less important in TITAN-II than in TITAN-I because the first wall is at a lower temperature and is made of ferritic steel rather than vanadium (Section 18.2). Back diffusion of protium is significant but acceptable. The air-detritionation system has a larger drier (but not recombiner) capacity to recover most of the tritiated water leaking from primary-system components.

The analysis of the TITAN-II tritium systems is given in the subsequent sections. The tritium flow rates are estimated in Section 18.3. The blanket tritium-recovery system, the plasma-exhaust-purification system, and the room-air-cleanup system are discussed, respectively, in Sections 18.4 through 18.6.

18.2. PERMEATION

18.2.1. First Wall

The first wall of the TITAN-II design is made of a ferritic steel alloy (9-C) in contrast with vanadium-alloy first wall of TITAN-I. Ferritic alloys are more resistant to hydrogen permeation than vanadium, so plasma-driven permeation should be less of a concern for TITAN-II.

As described in Section 12, plasma-driven permeation is modeled by using the code TRIPOS [2] to estimate the depth of triton implantation into the first wall, and the DIFFUSE code [3] to model the ensuing diffusion through the first wall. Values for the activation energies and coefficients for diffusion and solubility of hydrogen in the 9-C alloy (or HT-9) have not been reported. Consequently, the stainless-steel properties are used since both stainless steel and 9-C have 12% chromium and, therefore, the stainless-steel properties should approximate those of 9-C. The plasma flux and energies incident on the first wall are similar to those of TITAN-I.

The first wall of TITAN-II consists of half-cylindrical channels separated from the blanket lobes, with the cylindrical face towards the plasma. The direct area facing

the plasma is 101.6 m^2 , although the cylindrical shape increases the effective area for permeation. The first-wall channels are 30-mm wide and 1.5-mm thick at the beginning of life. The channel-wall temperature ranges from 638 to 345 °C on the plasma side, 370 to 335 °C on the coolant side, and varies by 32 °C from coolant inlet to outlet. For DIFFUSE calculations, a slab geometry with an effective area is used, with average temperatures of 530 and 240 °C, respectively, for the plasma and coolant side.

Diatomic-recombination boundary conditions are used for both sides of the first wall. The plasma-side calculations consider both a sputter-cleaned and dirty surface (recombination sticking coefficient of 1 or 0.01, respectively). Ferritic surfaces in contact with water coolant are known to form oxide layers which act as permeation barriers. A barrier factor of 100 was used for HT-9 in water in the BCSS [1] and is adopted here, although the effect of the lithium-nitrate solution on the oxide layer is uncertain.

The result is a permeation of tritium from the plasma into the coolant of 0.34 to 4 g/d, depending on the cleanliness of the plasma surface. Comparable amounts of deuterium also permeate. The recombination coefficient is not well known and lower values than assumed here would give higher permeation rates.

In an aqueous-salt-cooled system, the hydrogen over-pressure in the coolant may be significant, perhaps as much as several atmospheres. Such pressures raise the possibility of hydrogen permeation through the first wall from the coolant into the plasma. To estimate the size of hydrogen back-diffusion, DIFFUSE calculations are made assuming a Sievert's-law boundary condition on the coolant side of the first wall, and diatomic recombination on the plasma side (sticking coefficient of unity). The back permeation is found to be between 12 and 39 g/d, respectively, for over-pressures of 1 to 10 atm. If an oxide layer at the steel-water interface is assumed with a barrier factor of 100, then back permeation is reduced to tenths of a gram per day. Similar calculations assuming a dirty plasma-side surface (sticking coefficient of 0.01) result in comparable values.

18.2.2. Divertor

The TITAN-II divertor design is similar to that of TITAN-I. The divertor armor is made of a tungsten-rhenium alloy (W-26Re) and is 1-mm thick with 7.4-m^2 surface area. The armor is brazed onto W-26Re tubes with 10-mm outer diameter and 1-mm wall thickness which are cooled with an aqueous- LiNO_3 solution. Data for hydrogen diffusion are not available for the W-26Re alloy, so corresponding properties of pure tungsten are used for calculations.

The particle flux and energies incident on the divertor plates are similar to those for TITAN-I (Section 12). The TITAN-II divertor-plate temperature ranges from 675 to 870°C on the plate surface, peaks at 360°C on the coolant side of the tubes, and varies by 40°C from coolant inlet to outlet. For DIFFUSE analysis, average temperatures of 730 and 340°C are used, respectively, for the plasma side and coolant side of the divertor plate. For these conditions, a plasma-driven permeation of 0.035 to 0.07 g/d is calculated, depending on the cleanliness of the plasma-facing surface.

18.3. TRITIUM CONTROL

In this section, the tritium-release rates from the TITAN-II reactor are estimated as a function of the tritium levels in the water coolant. Previous fusion-reactor studies have used a tritium level of 1 to 10 Ci/kg in the primary coolant [1,4], but such a low level would require a very large water-detrification system for TITAN-II. In fact, there is considerable utility operating experience with tritium levels above 1 Ci/kg in water coolant. For example, CANDU reactors are designed to accommodate up to 70 Ci/kg in the moderator and 2.5 Ci/kg in the primary circuit at equilibrium. In present CANDU reactors at Pickering and Bruce, the heavy-water moderator (70°C and 0.1 MPa) contains 20 to 30 Ci/kg tritium while the heavy-water primary coolant (300°C and 9 MPa) contains 1.5 to 2 Ci/kg [5]. With these tritium concentrations (and inventories of about 4 kg of tritium in the water), the release rates from CANDU power stations have averaged 30 to 50 Ci/d-reactor over the past several years [5]. For fusion, a tritium level of 10 to 100 Ci/d is generally considered acceptable [1]. Based on the CANDU experience, tritium levels in the range of 10 to 100 Ci/kg were considered and a level of 50 Ci/kg was selected for TITAN-II.

There are two generic pathways for tritium losses to the environment: water leakage and permeation across heat exchangers and steam generators, and air losses from the evaporation of water leaked from components. For TITAN-II with steel piping and aqueous solution at $\leq 350^\circ\text{C}$, tritium permeation across the coolant piping is unimportant.

18.3.1. Tritium Losses to Power Cycle

Tritium losses across heat exchangers include both permeation and leakage. Heavy-water losses across the steam generators at Pickering (550 MWe) and Bruce (880 MWe) were, respectively, 4 and 0.5 kg/d, because of permeation and small leaks [5]. Large leaks

are not included in these values, as they are rare and are quickly repaired. Reference [1] (BCSS) reports a water-loss rate of 45 kg/d for light-water-reactor steam generators, 10 kg/d for CANDU reactors, and adopts 24 kg/d-GWe as a reference value. More recent data also support a low CANDU steam-generator leakage. For example, 0.02% of CANDU tubes were plugged or repaired (as of 1986) compared to the industry average of 2% [6]. This small percentage was achieved through care in design, fabrication, and operation, and is necessary because of the cost of lost D₂O (*e.g.*, a 1.6-mm hole in 24,000 m² of steam-generator area will be unacceptable). For TITAN-II, similar care would be necessary with the proposed tritium levels. If a water-loss rate of 1 to 10 kg/d by leakage and permeation is assumed, then the tritium-loss rate through the steam generator is 1 to 10 (Ci/d)/(Ci/kg).

Alternately, the permeation rate through the steam generator can be estimated directly. The permeating flux is proportional to the material permeability, area, inverse of the wall thickness, and square-root of up-stream minus down-stream pressures. A value of 2×10^{-10} N-cm²/s-kPa^{0.5} is reported in the literature for the permeability of the oxidized Inconel [5], with some CANDU data indicating somewhat lower values. Based on a heat-transfer area of 21,500 m² (Bruce uses 18,920 m² for 880 MWe), a tube thickness of 1.12 mm (Bruce uses 1.12 mm with 9-MPa primary and 4.3-MPa steam-side pressures), a conservative primary H₂ pressure of 10 MPa, and a negligible steam-side hydrogen pressure, the permeation rate for TITAN-II is about 3 (Ci/d)/(Ci/kg).

This permeation rate assumes that the T:H ratio in the gas phase is the same as that in the water. There is some evidence that about 5%-10% of the tritium is initially formed as HT gas [7]. This will eventually equilibrate with the HTO phase (about 16 appm T in H at 50 Ci/kg) such that the T:H ratio in the hydrogen gas phase would be three times less than that in the water phase. However, the rate of equilibration is normally slow at room temperature. In the presence of radiolysis, 300 °C water in contact with metals, an H₂ over-pressure, and recombiners, the level of HT partial pressure is not clear. If the HT partial pressure were much higher than the equilibrium levels, it could be used to aid tritium extraction.

These tritium-loss rates could be reduced by three system-design options:

1. Use of an intermediate heat exchange (IHX) loop,
2. Use of double-walled steam-generator tubes,
3. Use of equal or higher pressure on the steam side.

An IHX provides a good barrier between the steam side and the primary circuit. The intermediate loop could be maintained at a much lower tritium level so that subsequent losses into the steam generator contain a low tritium content. The lower tritium levels would be maintained by an additional flow to the primary-coolant tritium-recovery system. A particular advantage of the IHX is that it prevents the highly-tritiated primary water from reaching the steam circuit in the event of a major tube rupture. The disadvantages are: a lower thermal efficiency because of the loss of peak steam temperature, the capital cost of an extra heat-transport loop, extra load on tritium-recovery and room-air-detritiation systems because of the extra coolant loop, and increased leakage and maintenance due to the extra equipment.

Double-walled steam generators have been considered for use in sodium-cooled fission reactors and in some fusion studies (BCSS [1], MARS [8]). TITAN-II could use this concept to provide additional leak resistance (failure of one boundary need not propagate into the second boundary) and permeation resistance (interior-surface coatings inhibit transport). Estimates for MARS indicate a reduction in the tritium permeation of up to a factor of 10^5 is possible [8]. One experiment [9] reported no improvement over single-wall permeation rates, but it seems likely that a significant effect could be obtained with other interior-surface preparations. The double-walled steam-generator approach provides similar advantages as an IHX, but with less cost and complexity and reduced degradation of peak steam temperature.

The third approach, equal or lower primary-system pressure is possible for TITAN-II because the salt content of the primary circuit elevates the boiling point. Consequently, the primary-system pressure can be reduced while maintaining a high temperature, resulting in the elimination of the driving force for leakage. In the TITAN-II design, the primary-coolant system is at 7 MPa, about equal to that of the secondary side. Lower pressure in the primary system will also reduce the water-leakage rate from other components of the primary system and keep the leakage rates within the range of CANDU experience (primary system at 9 MPa).

In summary, the tritium-loss rate across a good but conventional steam generator is estimated to be about 1 to 10 (Ci/d)/(Ci/kg). With improvements such as equal primary and secondary pressures, increased thickness of tubing, double-walled tubes, or other concepts, it seems plausible that the leakage rates could be reduced an additional factor of ten. It is assumed here that a tritium-loss rate of 0.2 (Ci/d)/(Ci/kg) by permeation and leakage can be achieved.

18.3.2. Component Tritium Losses

Water leakage from components leads to losses of tritiated-water vapor generally up the stack. This loss is minimized by using leak-tight designs and components, segregating major systems, and the use of air-drier technology. Again, because of the economics of heavy water, CANDU reactors have developed leak-tight systems and efficient air driers. TITAN-II has the further advantages of no mechanical pressure-tube end fittings (eliminating a major water-loss pathway), and of the primary system being immersed in a water pool (capturing a large part of the tritiated-water leakage).

Characteristics of tritium leakage and recovery from several CANDU reactors are summarized in Table 18.3-I. Total tritium-escape rates (leakages) and loss rates are reasonably well-known because the air-drier water is monitored and any lost heavy water must be made up by external supplies. Uncertainty exists in leak rates from specific components or systems, and the data are not necessarily reported in a directly accessible form. Overall, however, the data indicate that operating power plants with tritium levels in the range 1 to 30 Ci/kg have demonstrated overall water leakage rates of 40 to 400 kg/d, loss rates of 5 to 16 kg/d, and air-recovery efficiencies of 75% to 96%.

Several possibilities exist for reducing water leakage from present CANDU levels. The improvements are being incorporated into the newer fission reactors, and could certainly be used in a fusion reactor. These include [5]:

- Minimizing piping joints and components and using welds where possible;
- Using live-loading and bellows seals on valves, taper-lock or equivalent fittings on flanges, and similar leak-tight components;
- High maintenance priority for the fixing of leaks;
- Better room sealing.

For TITAN-II, tritium leakage during maintenance is minimized by removing the tritiated water from the affected loop to a dump tank and replacing it with unpressurized, low-tritium water. In general, the routine use of remote welding and maintenance equipment as expected in fusion machines will encourage welded connections and on-line repair of leaky components.

In order to estimate the tritium losses, the TITAN-II water-blanket systems are divided into four major areas: the reactor vault, coolant-chemistry room, blanket tritium-recovery room, and main reactor hall, as is shown in Figure 18.3-1. The main steam

Table 18.3-I.
TRITIUM-RELEASE RATES FROM CANDU REACTORS

Unit(s)	Year	Escape Rate (kg/d-reactor)	Recovery (%)	Loss Rate (kg/d-reactor)
Pickering A (1-4)	1972-86	170 ^(a)	94 ^(a)	10 ^(a)
Pickering A (3,4)	1985	360	96	16
Pickering B (5-7)	1985	290	95	15
Pickering B (5-8)	1983-86	82 ^(a)	94 ^(a)	4.8 ^(a)
Bruce A (1-4)	1985	175	93	12
Bruce B (5,6)	1985	—	—	7
Cordoba	1984	36	85	5
Le Preau	1984	45	75	11
Le Preau	1985	98	89	11
Wolsung	1985	125	91	11

(a) Estimated values.

generator, the primary pumps, and the pool heat exchanger are located within the pool. Other components such as the primary-circuit ion-exchange resins and filters, which are gamma active, could also be placed in the pool.

A CANDU average of 200 kg/d leakage may be inferred from Table 18.3-I. The distribution of water leakage between CANDU systems is not known precisely, but it appears to be dominated by the hot primary-coolant system. Of this, a substantial amount is believed to be caused by the pressure-tube end fittings which are mechanical joints that are routinely accessed while on line. Taking 75% of the nominal CANDU leak rate to be from the primary system, and taking credit for the lack of end fittings in TITAN-II but recognizing that the primary loop is not easily accessible and the salt may affect leakage,

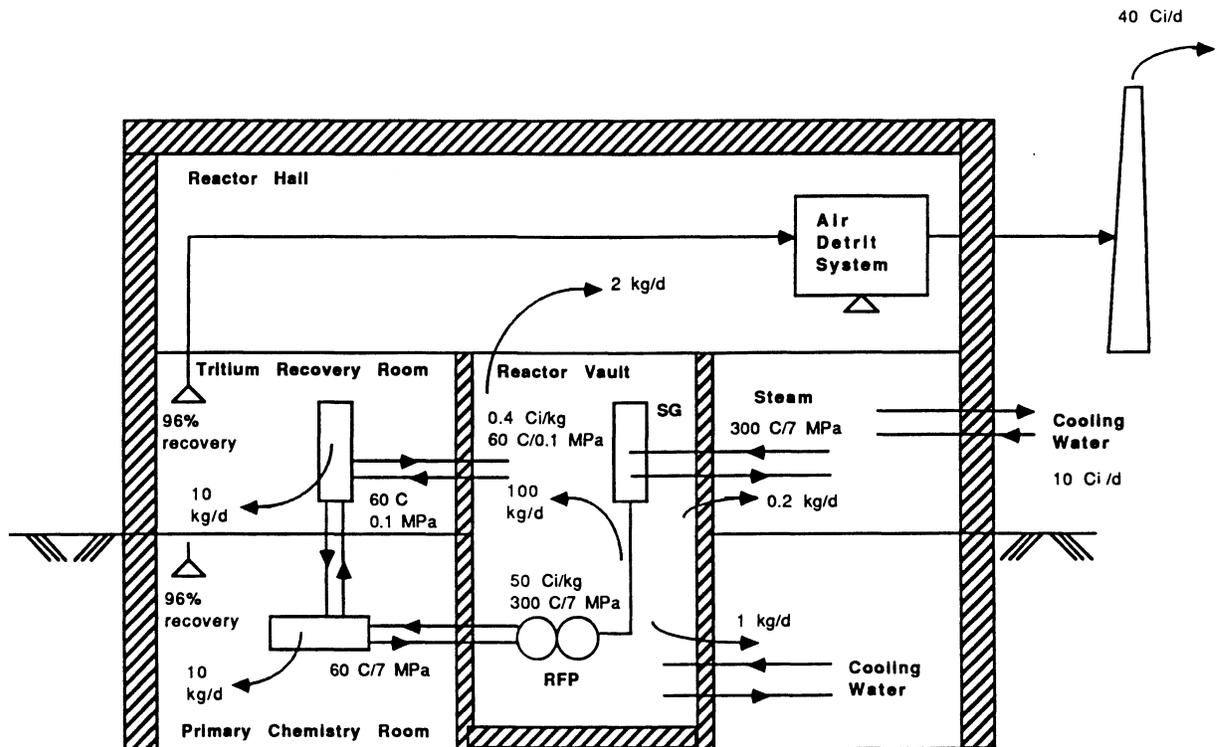


Figure 18.3-1. Water and tritium pathways within the TITAN-II reactor building, based on a 50 Ci/kg primary-system concentration. The net loss to the environment, allowing for the air-drier recovery efficiencies shown, is 50 Ci/d.

the estimated primary-system leakage rate in TITAN-II is 100 kg/d. Note that as this water leaks into the pool, the associated tritium is not released to the environment.

The remaining 25% of the nominal component water-leak rate occurs in other water-processing systems: the coolant-chemistry system and blanket tritium-recovery system in TITAN-II. It is assumed that welded joints and other procedures noted earlier are used, which reduce the leak rate by half to about 20 kg/d. A small loss also occurs from the pool, which is also sealed from the reactor hall. Figure 18.3-1 illustrates the general tritium levels and leakage pathways, based on a primary-coolant concentration of 50 Ci/kg.

It is further assumed that the equipment within the chemistry and tritium-recovery rooms is confined so that its environment can be efficiently dried at a 96% water-removal

rate. This rate represents the upper end of proven CANDU performance and implies use of room seals, segregation of leakage-prone equipment, and good drier capacity.

18.3.3. Tritium-Loss Rates

Based on the above permeation and leakage rates, the tritium-loss rates can be determined as a function of the tritium concentration in the primary coolant. In general, these releases linearly increase with tritium concentration. Clearly, the lower the tritium levels in the water, the lower the potential tritium-release-rate levels are. However, this must be balanced against the increasing cost of tritium extraction at low coolant concentration. These costs are evaluated in Section 18.4 with a tritium level of 50 Ci/kg selected as the TITAN-II design point.

From a maintenance viewpoint, the water temperature and tritium levels are of less concern than the air-tritium levels. In CANDU reactors, the moderator presently contains 20 to 30 Ci/kg without having a major impact on maintenance, and the design value is 70 Ci/kg. Under these conditions, workers would use plastic suits, system flushing, and automatic equipment to minimize the dose. In an accident where a worker was exposed to, for example, 10% direct skin wetting by 100 Ci/kg water, the committed dose is 15 rem. External dose from gamma-active corrosion products will also contribute and are much larger than the tritium dose (in present CANDU circuits). Overall, it is not expected that a tritium level of 50 Ci/kg will pose an unusual or unacceptable maintenance risk.

The TITAN-II reference design based on this trade-off has a tritium concentration of 50 Ci/kg in the primary-coolant system and 0.4 Ci/kg in the pool. The corresponding tritium-release rates to the environment are about 50 Ci/d total from the water-related systems. Based on the estimates for TITAN-I (Section 12), the additional contribution to the routine tritium releases from the remaining tritium systems, such as fuel cleanup and storage, are small.

18.4. BLANKET TRITIUM-RECOVERY SYSTEM

18.4.1. Tritium-Recovery Options

Some options for hydrogen-isotope separation are summarized in Table 18.4-I, with the hydrogen in the elemental or oxide (water) form. The effectiveness of the process is

Table 18.4-I.
THEORETICAL SEPARATION FACTORS
FOR SOME ISOTOPE-ENRICHMENT PROCESSES [10]

Process	HD/H ₂	HT/H ₂	DT/D ₂	H ₂ O/D ₂ O	HTO/H ₂ O	DTO/D ₂ O	²³⁵ U/ ²³⁸ U
Distillation	1.5 ^(a)	1.8	1.2 ^(a)	1.05 ^(a)	1.06	1.01	1.00002
H ₂ S exchange	NA	NA	NA	1.07 ^(a)	–	–	NA
Gas diffusion	1.22	1.41	1.12	1.027	1.054	1.025	1.0043 ^(a)
Gas centrifuge	1.051	1.10	1.051	1.038	1.078	1.038	1.16 ^(a)
Laser	–	–	–	–	10,000 ^(b)	–	–

(a) Processes in industrial use

(b) Tritium must be transferred into CF₃H molecule first.

roughly proportional to the separation factor, although there are other equally important parameters.

Distillation is widely used for hydrogen isotopes: cryogenic distillation at 20 K with elemental gas feed, and water distillation at 320 K with water feed (depending on the isotopes and concentrations). The bi-thermal hydrogen-sulfide (and the related ammonia) exchange process is used in industry for heavy-water extraction from ordinary water. This process would have good separation factors for tritiated water also, but would be difficult to use since it involves toxic and flammable gases. The newer technologies of gas centrifuge and laser enrichment offer particular advantages for uranium separation. Gas centrifuge offers reasonable theoretical separation for water, but not sufficiently better to justify replacing passive columns with high-speed rotating machinery. Laser separation offers very high separation factors in the laser stage, but requires substantial chemical processing as well as high-power lasers and optics. Laser separation has not proven better than the H₂S-exchange process for H₂O/HDO separation from natural water (at 150 ppm HDO). However, laser enrichment is still under development and may be effective for TITAN-II.

In the TITAN-II study, water and hydrogen distillation are considered the fundamental isotope-separation processes. These processes represent conventional technologies, have reasonable separation factors, and are mechanically fairly simple. Water distillation is used for pre-enrichment of the tritium, and cryogenic distillation is used for the final separation. A practical system involves other processes needed to transfer the tritium from the water into hydrogen gas. The options are vapor-phase catalytic exchange (VPCE), liquid-phase catalytic exchange (LPCE), direct electrolysis, or a combination of catalytic exchange and electrolysis. Some energy for electrolysis could be recovered by recombining the detritiated hydrogen with oxygen in a fuel cell. The water would be separated from the salt by flashing.

For the TITAN-II reactor, the tritium-production rate is 430 g/d. For an average tritium level in the primary heat-transport circuit of 50 Ci/kg, the water-feed rate to the tritium-recovery system would be about 4000 kg/h (200 kmol/h). For comparison, in other water-cooled blanket concepts in which the water tritium levels were deliberately kept lower than the levels considered here, maintaining 1 Ci/kg in water would require a 20-kmol/h water-detritiation system for 1-g/d tritium permeation into the water from the first wall or breeder [1,11].

Table 18.4-II summarizes some of the existing large hydrogen-separation facilities. Relative to TITAN-II with 50 Ci/kg coolant-tritium levels, VPCE, electrolysis, water distillation, and cryogenic distillation have been (or soon will be) demonstrated on a 1/10 scale with tritium; and electrolysis and H₂S exchange have been used in industry on a larger scale without tritium.

The effect of coolant-tritium levels on tritium-system cost is determined by considering a range of process designs at primary-coolant levels between 10 to 100 Ci/kg. Representative water-feed rates to the tritium-recovery system are given in Table 18.4-III as a function of the primary-system tritium content.

The water feed to the salt separator (flasher) is about 2.5 times the tritium-system feed assuming that 40% of the water is evaporated while going from the high-temperature, high-pressure feed conditions to the 0.1-MPa and 100 °C salt-free vapor. If the water-feed salt concentration is at 40% of its solubility limit, then the product is at 70% and precipitation should not occur. The pool is maintained at less than 0.5 Ci/kg in order to keep its tritium inventory acceptable. The tritiated water from the air-detritiation and other waste streams is estimated at about 6 kg/h (and several Ci/kg) based on the assumed leak and recovery rates. For comparison, these flow rates are much smaller than the coolant flow in the primary circuit (about 10⁸ kg/h) and its flow through the chemistry-control system (10⁵ kg/h).

Table 18.4-II.
PARAMETERS OF
VARIOUS LARGE WATER ISOTOPE-SEPARATION SYSTEMS

Site	Feed Fluid	Product Fluid	Process ^(a)	Feed Rate (kmol/h)	Start-up Date
Darlington (Canada)	D ₂ O	T ₂	VPCE/CD	18	1988
Grenoble (France)	D ₂ O	T ₂	VPCE/CD	1	1972
Chalk River (Canada)	D ₂ O	T ₂	LPCE/CD	1	1988
Mound Lab (US)	H ₂ O	T ₂	CECE/CD	0.06	1986
Chalk River (Canada)	D ₂ O	T ₂	DE	0.13	1985
Chalk River (Canada)	D ₂ O/HDO	D ₂ O	Elec. cascade	3-30	1950
Aswan Dam (Egypt)	H ₂ O	H ₂	DE	1700	1984
Norsk Hydro (Norway)	H ₂ O	H ₂	DE	180	1950
Bruce (Canada)	H ₂ O	D ₂ O	H ₂ S exchange	110,000	1975
Pickering (Canada)	D ₂ O/HDO	D ₂ O	WD	1-6	1972
Bruce (Canada)	H ₂ O/HDO	D ₂ O	WD	20	1975
Nangal (India)	H ₂ /HD	D ₂	CD	223	1964
Los Alamos (US)	DT	T ₂	CD	0.02	1985
Tokyo (Japan)	H ₂	H ₂ O	FC	150	1984
TITAN at 50 Ci/kg	H ₂ O/HTO	T ₂	–	220	20xx

(a) VPCE: Vapor-phase catalytic exchange;

LPCE: Liquid-phase catalytic exchange;

CD: Cryogenic distillation with H₂ or He refrigerant;

CECE: Combination of catalytic exchange and electrolysis;

DE: Direct electrolysis;

WD: Water distillation;

FC: Fuel cell.

Table 18.4-III.
REPRESENTATIVE WATER-DETRITIATION FEED RATES^(a)

Coolant T Level (Ci/kg)	From Blanket Circuit		From Pool ^(b) (kg/h)	From Air Cleanup (kg/h)
	To Flasher (kg/h)	To Recovery (kg/h)		
100	5050	2020	930	6
70	7200	2880	650	6
50	10100	4040	460	6
30	16800	6730	280	6
10	50450	20180	90	6

(a) Based on 460 g/d of T recovered from blanket, with 90% detritiation per pass.

(b) Based on 0.5 Ci/kg in pool or 100 kg/d leakage of primary coolant from blanket.

For the TITAN-II water-detritiation system, several process options were considered. Light-water coolant is used in the final design, although heavy water was considered [12]. A heavy-water tritium-recovery system has more efficient phase transfer but less efficient cryogenic distillation, so it uses less steam but more electricity, and has more hydrogen gas but contains less tritium. The overall tritium system costs are comparable, but the cost of the heavy water itself makes light water the coolant of preference.

A simple model was developed containing correlations for water distillation, VPCE, electrolysis, and cryogenic distillation [13]. The model was developed for and calibrated against equipment at the Darlington or ITER scale (100 to 800 kg/h water-feed rate), so the extrapolation to TITAN-II conditions is approximate. The results of these scoping calculations are given in Table 18.4-IV at various blanket-coolant tritium levels. The costs are direct equipment costs and must be doubled to include installation.

The trade-off between tritium-recovery-system cost and tritium release as a function of coolant-tritium level is shown in Figure 18.4-1. Tritium-release rates increase linearly with blanket-coolant tritium levels, but the system costs decrease roughly as the inverse

Table 18.4-IV.

**SCOPING EVALUATION OF SEVERAL TRITIUM-RECOVERY-SYSTEM DESIGNS
(460 g/d RECOVERY AND 90% DETRITIATION PER PASS)**

Coolant T Level (Ci/kg)	Process ^(a)			Maximum T Level (Ci/kg)	Process T Inventory (kg)	Process H Inventory (kmol)	Electrical Power (MWe)	Thermal Power (MWth)	Direct Cost (M\$)
H₂O COOLANT:									
10	5WD	5VPCE	CD(H ₂)	50	1.8	560	31	240	230
10	10WD	5VPCE	CD(He)	100	2.8	280	16	230	230
20		5VPCE	CD(H ₂)	20	1.4	2100	28	51	125
20		5VPCE	CD(He)	20	1.4	1400	73	51	150
20		DE	CD(He)	240	12.0	280	78	0	190
20	10WD	5VPCE	CD(He)	200	2.6	140	8	120	125
50		5VPCE	CD(H ₂)	50	0.9	860	12	21	73
50		5VPCE	CD(He)	50	0.9	580	32	21	89
50	5WD	5VPCE	CD(He)	250	1.3	110	7	48	63
50	10WD	5VPCE	CD(He)	500	2.6	60	4	50	62
50	5WD	5VPCE	CD(H ₂)	250	1.4	170	3	50	60
50		DE	CD(H ₂)	600	12.0	170	3	0	86
70		5VPCE	CD(H ₂)	70	0.8	600	8.5	15	60
70	5WD	5VPCE	CD(H ₂)	350	1.3	121	1.8	35	46
70	5WD	5VPCE	CD(He)	350	1.3	81	5	35	51
100		5VPCE	CD(H ₂)	100	0.7	460	6.5	11	51
100	5WD	5VPCE	CD(H ₂)	500	1.3	91	1.4	26	38
D₂O COOLANT:									
20		5VPCE	CD(H ₂)	20	0.9	3800	51	38	118
50		5VPCE	CD(H ₂)	50	0.6	1500	22	16	70
70		5VPCE	CD(H ₂)	70	0.5	1100	16	11	57
100		5VPCE	CD(H ₂)	100	0.4	820	12	8	49
34 ^(b)		8VPCE	CD(H ₂)	34	0.04	140	2	1.4	20

- (a) nWD: Water distillation with enrichment by factor of n;
 mVPCE: m-stage vapor-phase catalytic exchange;
 DE: Direct electrolysis;
 CD: Cryogenic distillation with H₂ or He refrigerant.

(b) Darlington Tritium Recovery Facility at 26 g/d of tritium and 97% detritiation per pass.

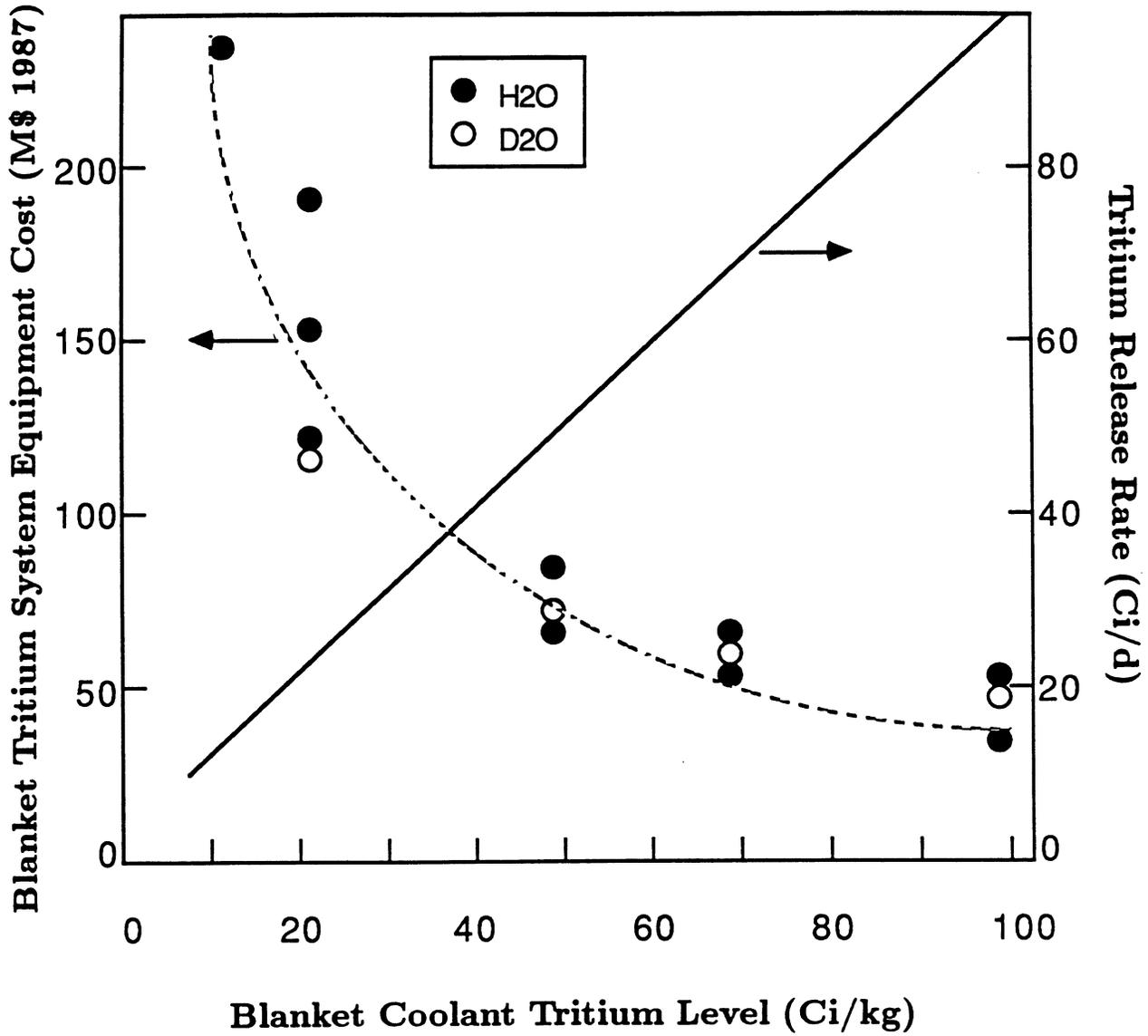


Figure 18.4-1. Comparison of direct cost of the tritium-recovery system and tritium-release rate as a function of blanket-coolant tritium levels. Design points are shown for the various tritium-system designs analyzed.

Table 18.4-V.

TITAN-II BLANKET TRITIUM-SYSTEM DESIGN REQUIREMENTS**Blanket:**

Volume	274 m ³
Coolant	H ₂ O at 50 Ci/kg
Lithium content	6.4 at.% Li as aqueous-LiNO ₃ solution (98.5 g LiNO ₃ per 100 g H ₂ O)
Conditions	330 °C, 7 MPa
Tritium-extraction rate	≥ 430 g/d of T
Tritium-extraction efficiency	≥ 90% per pass

Pool:

Volume	22,640 m ³
Tritium inventory	≤ 1 kg T
In-leakage from blanket coolant	100 kg/d at 50 Ci/kg

Tritiated Waste Water:

Production rate	6 kg/h at 10 Ci/kg
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of the tritium level. Beyond about 50 Ci/kg, the additional cost savings become relatively small while the tritium-release rates (and coolant-tritium inventory) continue to increase. At 50 Ci/kg, the estimated tritium-release rate is about 50 Ci/d, which is within the acceptable design range for fusion reactors (nominally 10 to 100 Ci/d), and is consistent with present CANDU performance.

The TITAN-II reference design must recover 430 g/d of tritium from 50-Ci/kg water. The system conditions are summarized in Table 18.4-V. Based on the scoping evaluations (Table 18.4-IV), the main options for tritium recovery are compared below.

VPCE versus LPCE versus electrolysis. The catalytic-exchange processes need more hydrogen gas because of the less efficient HTO/HT transfer, but they have much

smaller tritium inventories for the same peak process HTO concentration and are generally less expensive. The total power consumption is similar. A catalytic process is preferred for the TITAN-II scale. The LPCE has some advantages because of mechanical simplicity and lower temperature operation. However, VPCE is more efficient and thus less expensive. Therefore, the vapor-phase catalytic-exchange process is chosen for TITAN-II.

Hydrogen versus helium refrigerant. For TITAN-II, as with Darlington, the additional hydrogen gas for the refrigerant cycle should be less important (given the large amount already present as process gas) than the cost and power savings (a factor of three to five).

Water distillation. Distillation provides simple HTO enrichment, operating at 50 to 60°C and 13 to 17 kPa using low-pressure steam. Equipment costs could potentially be reduced by 10M\$ by using water distillation. However, the cost of the large building needed to house the towers (8 towers, each about 2.6 m in diameter and 30 to 40-m high, for 50-Ci/kg feed) and the high tritium levels in water should be considered, which appear to offset any cost savings. Distillation would be useful, however, for enriching the pool and waste water because of the lower feed rate and tritium content.

18.4.2. Blanket Tritium-System Design

Based on the scoping analysis, a reference design was selected based on:

- Water distillation for the detritiation and pre-enrichment of the low-tritium feed from the pool and air-cleanup systems,
- Five-stage VPCE for transferring tritium into the hydrogen gas phase from the primary coolant and enriched water-distillation output,
- Cryogenic distillation to separate the protium from the tritium.

A block diagram of the tritium-recovery system is shown in Figure 18.4-2, giving the overall mass balance and the flow rates of the main forward and recycle streams within the tritium system. This design is larger than necessary and could remove 465 g/d of tritium (rather than 430 g/d) at 50 Ci/kg.

Tritium levels in the pool are maintained at less than 0.5 Ci/kg by processing 465 kg/h through a single water-distillation tower. The tower returns the water with 90% detritiation, and forwards 5.4 kg/h at 50 Ci/kg for mixing with the blanket-coolant water

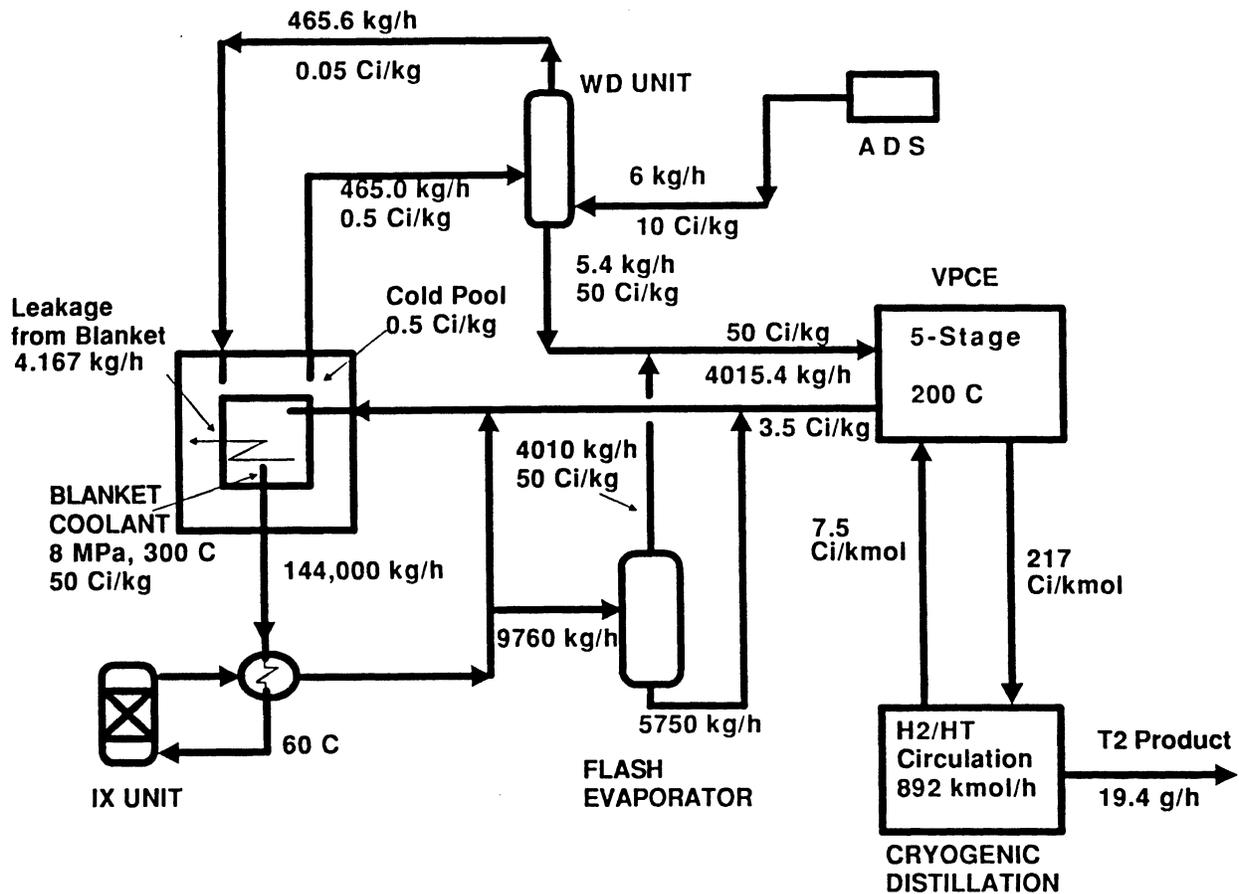


Figure 18.4-2. Blanket tritium-extraction system for TITAN-II showing water and hydrogen flow rates and tritium level in the flows, based on extracting 465 g/d of tritium at 50 Ci/kg.

for processing. This tower is about 2.6 m in diameter and 30- to 40-m high, about the size and duty of the heavy-water-upgrader distillation column at Darlington. The water-distillation system includes the tower with packing, reboiler, overhead condenser, vacuum system to provide a condenser pressure of 13 kPa, pumps, surge tanks, piping, control, and instrumentation. Accounting for tritium decay in the pool, the peak tritium levels would actually be 0.37 Ci/kg and would only be reached after several years of operation.

A purification unit is required to keep gamma levels low in the blanket coolant and to provide a clean feed stream to the tritium-recovery system. The purification unit uses high-pressure, low-temperature technology widely used in CANDU reactors: a heat exchanger, filters, and ion exchange (IX) columns to remove both cationic and anionic impurities. A bleed stream of 144,000 kg/h is withdrawn from the blanket-coolant loop and cooled to 60°C to avoid thermal degradation of the IX resin. This flow rate re-

moves about 50% of the suspended impurities in the primary circuit each hour, typical of CANDU design. The filter vessels will have integral lead shielding and replaceable cartridge baskets. The IX columns contain a lithiated mixed-bed resin that will not remove lithium or nitrate ions. The IX columns will be located in pits provided with concrete shielding, and the resin will be slurried in and out of the columns.

The purified salt solution leaving the IX columns is reheated to 280°C in the heat exchanger, and a fraction is diverted to a flash evaporator. The flashing of water from 8 MPa and 280°C yields about 40% salt-free vapor at 0.1 MPa and 100°C which is sent to the VPCE unit for tritium recovery. Liquid from the flash evaporator, concentrated in lithium salt but below the solubility limit, combines with the detritiated water from the VPCE and the main purification stream from the heat exchanger.

A five-stage VPCE unit provides a water-detritiation factor of 93% and reduces the hydrogen flow to the cryogenic distillation, which is then four times the molar feed-water rate. Each VPCE stage consists of an evaporator, superheater, catalyst bed, and condenser. The catalyst operates at 200°C and slightly above atmospheric pressure. There are no valves in this equipment; the Grenoble VPCE system has operated reliably for many years.

The cryogenic-distillation unit receives mostly H₂/HT and separates the hydrogen isotopes into 99+% T₂ product and an almost pure H₂ stream. The cryogenic-distillation unit is a cascade of packed distillation columns, reboilers, condensers, heat exchangers, and catalytic equilibrators inside an insulated cold box. The feed gas from the VPCE is dried and purified to remove contaminants (*e.g.*, water vapor, nitrogen, and oxygen) and then compressed and cooled before entering the columns. Hydrogen refrigeration is used because it cools efficiently through the latent heat of vaporization. This leads to much reduced refrigeration power and smaller cryogenic cold boxes because of the smaller heat-exchange surface area.

The largest cost component in the cryogenic-distillation unit is the first column which separates the bulk H₂ from HT. With hydrogen refrigerant, the operating conditions of this first column are 0.2 MPa and 23 K. A 40-stage stripping section is adequate to provide a tritium-poor hydrogen-return stream at 10 Ci/kmol. The first column condenser heat load, which essentially determines the refrigeration requirement, is about 360 kW for 890-kmol/h feed. The tritium inventory is based on a scale-up of Darlington, accounting for differences in flow rate and reflux ratios, with a 25% reduction for an optimum design of the high-tritium cascade.

Table 18.4-VI.

**TITAN-II BLANKET TRITIUM-SYSTEM
CAPITAL COST AND TRITIUM INVENTORY
(BASED ON EXTRACTING 465 g/d of T at 50 Ci/kg)**

Process Unit	Equipment Cost (M\$ 1987)	Tritium Inventory (g)
Purification	1.5	4.5
Flash evaporator	1.0	25.4
Water distillation	6.0	1.7
VPCE ^(a)	41.0	12.5
Cryogenic distillation ^(b)	20.0	550.0
TOTAL	69.5	594.1

(a) Cost includes driers, feed and return tanks, gas compressors, five-stage VPCE (catalytic beds, evaporators, superheaters, and condensers), low-tritium column-expansion tank, vacuum pumps, piping, controls, analyzers, and instrumentation.

(b) Cost includes hydrogen-refrigeration unit.

The characteristics of the TITAN-II blanket tritium system are given in Tables 18.4-VI and 18.4-VII. The process-equipment capital cost includes engineering but not site preparation, structures and foundations, site services, installation, commissioning, freight, taxes, or indirect costs such as corporate overheads.

The overall cost of TITAN-II tritium systems is estimated as 70 M\$ (1987) for direct cost of the process equipment. Although the feed throughput is about a factor of 10 larger than Darlington and the process is similar, the costs are only about 3 to 4 times larger. This reduction reflects general economies of scale (the dominant equipment scales as the 0.7 power), fewer VPCE stages for TITAN-II because of the lower detritiation needs (5 rather than 8), and the lower reflux ratio (a factor of 3 to 5) in the first and largest cryogenic-distillation column by separating H₂/HT rather than D₂/DT.

Table 18.4-VII.

**TITAN-II BLANKET TRITIUM-RECOVERY SYSTEM
(BASED ON EXTRACTING 465 g/d of T at 50 Ci/kg)**

Maximum tritium concentration	50 Ci/kg in water
Tritium-extraction rate	465 g/d of T
Tritium inventory as water	44 g T
Tritium inventory as gas	550 g T
Blanket detritiation factor	93% per pass
Hydrogen-refrigeration power	5.7 MWe
Low-pressure steam to water distribution	5.7 MWth at 300 kPa
Low-pressure steam to VPCE	1.2 MWth at 600 kPa
High-pressure steam to VPCE	8.5 MWth at 2.5 MPa
Hydrogen-gas inventory	1500 kg
Building volume	36,000 m ³

The installed cost for the TITAN-II tritium system is estimated as 140 M\$ (1987), based on doubling the direct cost of the equipment. The exact multiplier is not clear; installation costs for similar types of equipment have been quoted as 10% to 400% of the direct costs. A factor of two is roughly consistent with Darlington, if the building and air-cleanup systems are treated separately.

There are several factors which will reduce costs. First, the actual system is about 8% oversized relative to the final TITAN-II requirements. Second, the TITAN-II base year for costing is 1986. Consequently, the reference TITAN-II blanket tritium-system cost is estimated to be 130 M\$ (1986).

Other factors are not explicitly included since their individual contributions are less than the overall accuracy of the cost estimate. First, TITAN-II is a commercial reactor so there will be economies of production. Engineering costs (10% of Darlington, roughly 25% of the process-equipment cost) can be shared over several reactors. Second, there will be

a learning curve which can reduce costs by 15% to 30% over 10 installed systems. Third, the VPCE equipment (boiler, superheater, catalyst, and condenser) can be combined into factory-assembled columns for each stage. This would complicate maintenance, but reduce water leakage and provide up to 30% savings per stage. Finally, the tritium-recovery system will reduce the natural deuterium present in the primary circuit, which will decrease the work load on the cryogenic columns and also provide some reactor deuterium needs (about 200 kg of deuterium is available, mostly in the pool). Another blanket tritium-system concept could allow higher tritium levels in the pool, and circulate the detritiated blanket-coolant water directly through the pool. This would eliminate the water-distillation column presently used on the pool, but would require more stages on the VPCE. This concept was not optimized, but with pool tritium levels of 1 to 1.2 Ci/kg (rather than the present 0.4-Ci/kg concept) there is little overall effect on the direct capital cost.

A substantial improvement in costs requires a new approach. Laser separation is under investigation, but it is not yet clear that this is attractive for water detritiation. Radiolysis may be useful if a large amount of the tritium stays in the HT form, and if the system can control the consequences of the associated O₂ production. One may imagine advances in technologies such as lasers, permeation membranes, or magnetic systems that take advantage of the large differences in mass, charge-to-mass ratios, nuclear moments, and molecular vibrational levels between the different water molecules. These differences have not been fully exploited in present hydrogen-separation systems.

18.5. PLASMA-EXHAUST-PURIFICATION SYSTEM

The plasma-exhaust-purification system is virtually identical to that of TITAN-I described in Section 12. Back diffusion of protium from the coolant into the plasma may occur if there is a substantial hydrogen over-pressure from the radiolysis. For an H₂ pressure of 1 to 10 bar, the permeation rate is 0.1 to 10 g/d of hydrogen. Assuming 1 g/d of H permeation, a fuel-feed rate of 10 kg/d of DT, and a flow to the fuel-isotope system of 1% of the main fuel flow, the resulting protium concentration in the plasma is about 2.5 at.%, which is acceptable. The present fuel-processing system is designed to handle up to about 5% protium. If necessary, the fuel-isotope system could be increased in size at small overall increase in cost, or the fuel-isotope system could be merged with the blanket-isotope system.

18.6. ROOM-AIR-CLEANUP SYSTEM

Air driers will be widely used in TITAN-II to recover HTO lost from component leakage. Air from rooms containing leakage-prone equipment will be segregated using CANDU-like confinement technology and ventilated through air driers. These driers are not needed for HT conversion duty so they do not have heaters, recombiners, or coolers. They are reliable and proven in CANDU service.

The overall building volume containing exposed primary-coolant circuit and blanket tritium-system equipment is roughly 40,000 m³. In the TITAN-II reference design, 40 Ci/h of HTO escapes from this equipment and is recovered or lost. For a room decontamination factor of 100, this implies air tritium levels of about 1 MPCa averaged over the whole volume. In practice, the tritium levels will fluctuate and be higher during spills and maintenance. On the other hand, the most leakage-prone equipment will be segregated into smaller volumes which can be dried more effectively. A detailed layout has not been attempted, but if driers are used with several air changes per hour on average, then the cost for the installed drier system is estimated at 13 M\$ for 200,000 m³/h. In addition, some capacity is needed for other rooms (*e.g.*, the fuel-purification room). In TITAN-II, tritium is generally more in the water form and handled using the normal driers, so the additional emergency air-cleanup system has a capacity of 15,000 m³/h at a cost of 2 M\$.

18.7. SUMMARY

The TITAN-II tritium flow paths are shown in Figure 18.7-1. The tritium inventories and tritium-systems costs are given in Tables 18.7-I and 18.7-II, respectively.

The final design of the TITAN-II tritium system is based on 50 Ci/kg in the blanket-water coolant, and 0.37 Ci/kg in the pool. The TITAN-II tritium-release rates are dominated by those associated with leakage of HTO from the blanket-coolant systems and are estimated to be 50 Ci/d. The maximum tritium concentration in the blanket tritium-recovery system is also 50 Ci/kg since no enrichment is used in the present design. This level of tritium is only modestly beyond present utility experience, presents no unusual risks (*i.e.*, accidental splashing is not fatal), and may be handled using standard precautions such as protective suits.

The overall cost of the TITAN-II tritium system is 170 M\$ (1986) installed. The costs of the TITAN-II tritium system (Table 18.7-II) are dominated by the blanket tritium-

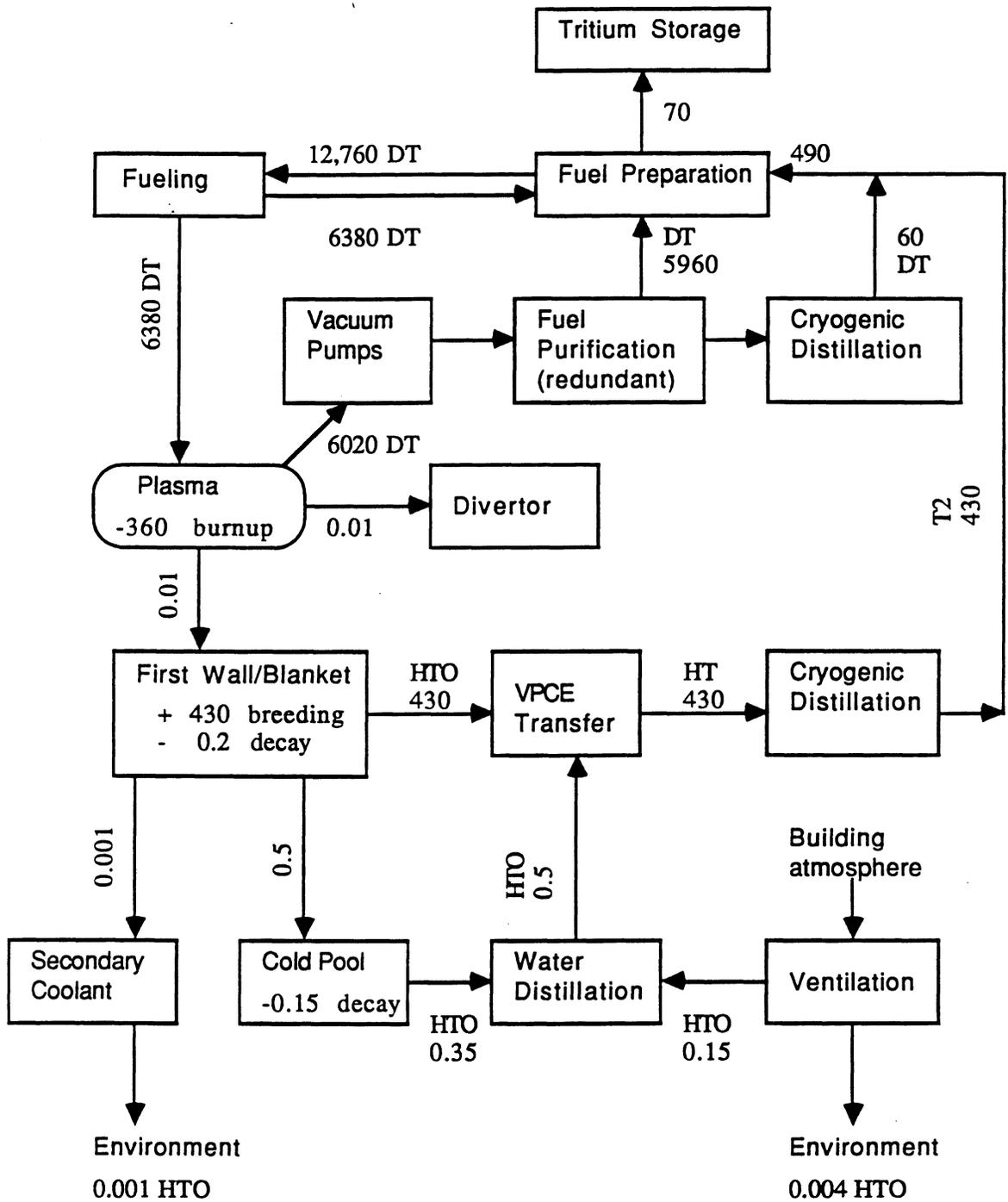


Figure 18.7-1. Overall tritium flow rates (g/d) and form in the TITAN-II reactor.

recovery system. Since tritium recovery in TITAN-II involves isotope separation of tritium from low concentrations in water, it is expected to be more expensive than for other fusion-blanket concepts. The present design approach is based on proven chemical exchange and distillation concepts. Costs for other tritium systems are similar to those for TITAN-I (note the larger air-drier capacity). Some costs are estimated from Reference [14].

The total tritium inventory in TITAN-II is 4 kg, roughly comparable to the inventory in some CANDU reactors at present. The largest inventory is in the primary circuit, which requires a larger blanket processing system.

Table 18.7-I.
TITAN-II TRITIUM INVENTORIES

System	T Inventory (g)	Form
Primary heat-transport coolant	1420 ^(a)	HTO
Beryllium	10	T in metal
Piping and structure	< 1	T in metal
Plasma chamber and vacuum	5	DT
Fuel processing	20	DT
Blanket tritium recovery	44	HTO
	550	HT
Shield	< 10	HTO
Tritium storage	1000	Metal tritide
Pool	940 ^(b)	HTO
TOTAL	4000	

(a) Based on 274 m³ at 50 Ci/kg.

(b) Based on 22,640 m³ at 0.4 Ci/kg.

A major reduction in the costs and tritium levels requires a new water-detrification approach. At present, laser separation is under investigation, but probably improvements in the lasers and optical materials are required for this approach to be attractive. Radiolysis might be helpful if a high yield of HT is obtained (not clear from present experiments), and if the associated O₂ production is acceptable.

Relative to the TITAN-I tritium system (Section 12), the TITAN-II tritium system is more expensive, the total tritium inventory is larger, the overall tritium system is physically larger, and the chronic tritium releases are larger. However, the TITAN-II tritium inventory is much less at risk for major release because of the lack of reactive chemicals, the low temperatures and pressures of most of the tritiated water, and the pool surrounding the fusion power core. Plasma-driven permeation is not a concern for TITAN-II, although the amount of protium back-diffusing to the plasma is significant.

Table 18.7-II.

TITAN-II TRITIUM-SYSTEM INSTALLED COSTS

System	Installed Cost (M\$ 1986)
Fuel processing	14
Blanket tritium recovery	130
Air detrification	15
Fueling	4
Waste-water cleanup	2
Tritium storage	1
Tritiated-waste disposal	3
Primary-coolant-storage tanks	1
TOTAL	170

REFERENCES

- [1] D. L. Smith, G. D. Morgan, M. A. Abdou, *et al.*, "Blanket Comparison and Selection Study, Final Report," Argonne National Laboratory report ANL/FPP-84-1 (1984).
- [2] P. Chou and N. M. Ghoniem, "Precipitate Dissolution by High-Energy Collision Cascades," *J. Nucl. Mater.* **117** (1983) 55.
- [3] M. I. Baskes, "DIFFUSE 83," Sandia National Laboratory report SAND-83-8231 (1983).
- [4] S. Sood, O. K. Kveton, *et al.*, "Tritium Systems Concepts for the Next European Torus (NET)," Canadian Fusion Fuels Technology Project report CFFTP-G-86020 (1986).
- [5] K. Y. Wong, T. A. Khan, F. Guglielmi, *et al.*, "Canadian Tritium Experience," Ontario Hydro report (1984).
- [6] G. A. Pon, "Significant Engineering Developments," in *Proc. Eng. Centennial Conf.*, Montreal, Canada (May 1987).
- [7] T. Kamara, R. M. White, and F. S. Rowland, "The Reactions of Recoil Tritium Atoms in Aqueous Solution," *J. Inorg. Nucl. Chem.* **21** (1961) 210.
- [8] B. G. Logan, C. D. Henning, *et al.*, "MARS, Mirror Advanced Reactor Study, Final Report," Lawrence Livermore National Laboratory report UCRL-53480 (1984).
- [9] S. Sinharoy, A. R. Keeton, and W. J. Lange, "Permeation of Deuterium through Double Walled Croloy Tubes," Westinghouse report 86-7B6-HYPER-R1 (1986).
- [10] M. Benedict, T. Pigford, and H. Levi, *Nuclear Chemical Engineering*, 2nd edition, McGraw-Hill, Toronto (1981).
- [11] The NET Team, "NET Status Report," NET-51 (1985).
- [12] A. Dombra, Chalk River Nuclear Laboratories, private communication (August 1986).
- [13] A. Busigin and K. Kalyanam, "Water Detritiation System Module for ETR System Code," Ontario Hydro report (1987).
- [14] M. Abdou, C. Baker, J. Brooks, *et al.*, "A Demonstration Tokamak Power Plant Study (DEMO)," Argonne National Laboratory report ANL/FPP/82-1 (1982).