Chapter XI

VACUUM AND TRITIUM SYSTEMS

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1. VACUUM SYSTEM

The reactor vacuum system as a whole includes: the torus, the neutral beam injectors, the SC magnet cryostats, the fuel pellet injector, and the special vacuum system for other components. Only the vacuum systems for the torus and neutral beams are discussed here.

1.1. Torus vacuum system

The features of the torus vacuum system are as follows:

(a) Pumping of a high helium gas load
(b) Presence of tritium in the pumped gases
(c) Operation near sources of high electromagnetic and particle radiation
(d) Long-term quasi-continuous operation under remote maintenance.

These operating conditions put a great strain on the type and pumping speed of the high-vacuum and fore-vacuum pumps, as well as on the reliability of pumps, valves and gauges.

The basic vacuum physical parameters of the plasma, vacuum chamber and pumping system are summarized in Table XI-1.

The vacuum system of the torus must be able to pump the plasma chamber before operation with plasma, in order to obtain the initial base pressure, and also to pump during dwell time and plasma burn.

The toroidal vacuum chamber should be pumped down to a pressure of 10^{-7} torr. The gas load is mainly determined by gas release from the vacuum chamber. For the vacuum boundary of INTOR, the surfaces of the first wall, breeding blanket, test modules, and parts of the radiation shield are placed within

TABLE XI-1. PARAMETERS RELEVANT TO THE VACUUM SYSTEM

Plasma volume (V_{pl})	240 m^3
Average D-T ion density (n_i)	1.4×10^{20} m^{-3}
Burn time: Stage I	100 s
Stages II, III	200 s
Dwell time	20 s
Initial base pressure	10^{-7} torr
Pre-shot base pressure	3×10^{-5} torr
Gas-load composition	47% D; 47% T; 5% He; 0.5% C; 0.5% O
Chamber pumping speed (He) (S_{eff})	2×10^5 l/s
Number of divertor exhaust ducts	12
Vacuum chamber volume (V_{ch})	450 m^3
Surface area inside vacuum chamber (F_{ch})	10^4 m^2
Maximum temperature of the first wall (T_{ch})	623 K
High-vacuum pump type	cryosorption

the chamber. The total area of the internal surfaces is ~10^4 m^2. Baking and discharge cleaning are used to outgas these surfaces. Those construction elements which are rather difficult to bake after assembly may be outgassed before the reactor is assembled. The assembled chamber is baked to 200–300°C. The maximum baking temperature is limited by the adjoining lead blanket zone. A baking technique can be selected only when the design has been developed in detail. One of the possible techniques is heating the chamber by a hot gas, for instance argon. In this case, water vapour is removed from the chamber. Oxygen and carbon are removed from the surfaces facing the plasma by hydrogen discharge while the chamber walls are hot. After baking and discharge cleaning the specific thermal outgassing rate will be reduced to 10^{-7} torr·l/s·m^2. In this case the effective pumping speed required to provide the initial base pressure should be 5×10^4 l/s.

During the dwell time the vacuum chamber should be pumped down to 3×10^{-5} torr within 20 s. The main gas load source is the neutralized and thermalized plasma at the first wall, i.e. a non-reacted D-T mixture and helium. An additional gas load during dwell time is created by hydrogen isotopes released from the first wall. It is estimated that the particle release rate from the wall surface at 573 K is 5×10^{15} $cm^{-2} \cdot s^{-1}$ at the beginning of the pumping interval;

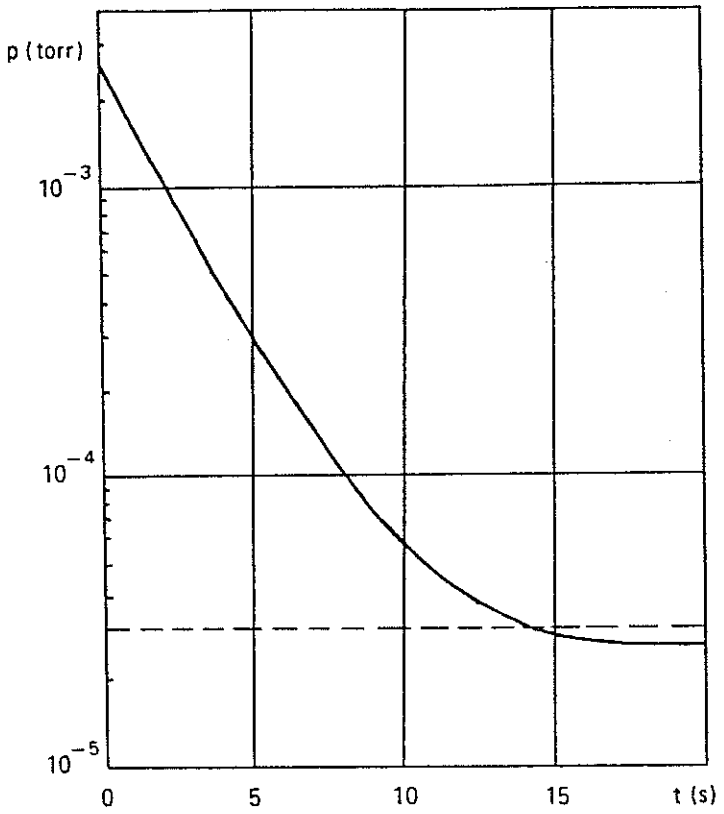


FIG. XI-1. Change of pressure in the vacuum chamber at pumping during dwell time.

then it decreases by two orders of magnitude within several seconds and is about 10^{13} cm$^{-2} \cdot$s^{-1} at the end of the pumping interval [2]. This saturation value is nearly reached after ten cycles of reactor operation. When recombination and release of hydrogen isotopes in molecular form are taken into account, the average particle release rate during the pumping interval can be estimated as $q = 2 \times 10^{13}$ cm$^{-2} \cdot$s^{-1}, with the minimum chamber pressure determined by the hydrogen gas load from the first wall being 2.5×10^{-5} torr. The chamber pressure change during dwell time is described by the expression:

$$P = \left(n_i \frac{V_{pl}}{V_{ch}} - \frac{qF_{ch}}{S_{eff}} \right) kT_{ch} \, e^{-S_{eff}/V_{ch}} + q \frac{kT_{ch} F_{ch}}{S_{eff}}$$

where k is the Boltzmann constant and the other symbols are the same as those used in Table XI-1.

For the vacuum system parameters given in Table XI-1, the required pressure of 3×10^{-5} torr is reached 14 s after the beginning of pumping (Fig. XI-1). To pump the chamber down to 3×10^{-5} torr during 20 s, an effective pumping speed of 1.5×10^5 l/s is adequate.

TABLE XI-2. BASIC CHARACTERISTICS OF HIGH-VACUUM CRYOPANEL PUMPS

Pump type	Pumped gas	Working temperature (K)	Specific pumping speed ($l/s \cdot cm^2$)	Maximum sorption capacity ($torr \cdot l/cm^2$)	Maximum heat load (W/cm^2)	Energy consumption (kW per l/s)
Condensation	D-T	~ 4	7 – 10	30	0.1	10^{-3}
Cryosorption (microporous adsorbents)	D-T	15 – 30	7 – 10	30	1.0	10^{-4}
Cryosorption (condensed gases)	D-T	10 – 30	4 – 7	30	1.0	10^{-4}
Sorption (non-evaporable metallic getters)	D-T	500 – 700	5 – 10	5 – 10	–	10^{-4}
Cryosorption (microporous adsorbents)	He	~ 4	1.5 – 3.5	1.0	0.1	10^{-3}
Cryosorption (condensed gases)	He	~ 4	2.0	1.0	0.1	10^{-3}
Cryosorption (microporous adsorbents)	He	20	0.1 – 0.15	$(3-5) \times 10^{-3}$	1.0	10^{-4}

In the stationary plasma burn mode it is necessary to keep the helium concentration in the plasma below 5%; to ensure this condition, a few per cent of the helium atoms escaping from the plasma should be pumped [1]. The effective helium pumping speed should not be less than 2×10^5 l/s. Thus, the pumping mode during burn time should be most effective and is decisive when choosing the pumping speed.

The pumping system of the toroidal chamber consists of 12 pumping units symmetrically placed relative to the reactor axis. Each pumping unit is connected with the divertor chamber by an exhaust duct of 1.0–1.2 m diameter and 9 m length. To provide the effective speed of the helium chamber pumping of 2×10^5 l/s, each of the 12 pumps should pump helium at a speed of up to 5×10^4 l/s [5].

The most attractive pumps to be used for the reactor system are listed below (see also Table XI-2).

Pumps for D-T mixture

(a) Condensation pumps (~ 4 K): Their advantages are simple design and maintenance, short regeneration time and an unlimited number of regeneration cycles; their disadvantage is incompatibility with high radiation and heat fluxes.

(b) Cryosorption pumps based on microporous adsorbents or condensed gases at 10–30 K: Their advantage is a higher working temperature; their disadvantages are incompatibility with high radiation and heat fluxes, and a relatively long regeneration time.

(c) Sorption pumps based on non-evaporable metallic getters: Their advantage is that they can be arranged close to high radiation zones; their disadvantages are a relatively long regeneration time and a limited lifetime.

Pumps for helium

Cryosorption pumps (~4 K) based on microporous adsorbents or condensed gases: The disadvantages of these pumps are the complicated technology of adsorbent deposition onto a substrate, the low mechanical strength limiting the number of regeneration cycles, and the low heat conductance limiting the sorption layer thickness and therefore sorption capacity. Helium can be pumped by means of adsorbent-based cryosorption pumps at 20 K, but in this case there is a substantial reduction both in specific pumping speed and in maximum sorption pump capacity. The development of a highly efficient pump on a condensed gas base is complicated because of the lack of sufficient experimental data on helium pumping and on helium mixtures with hydrogen nuclides.

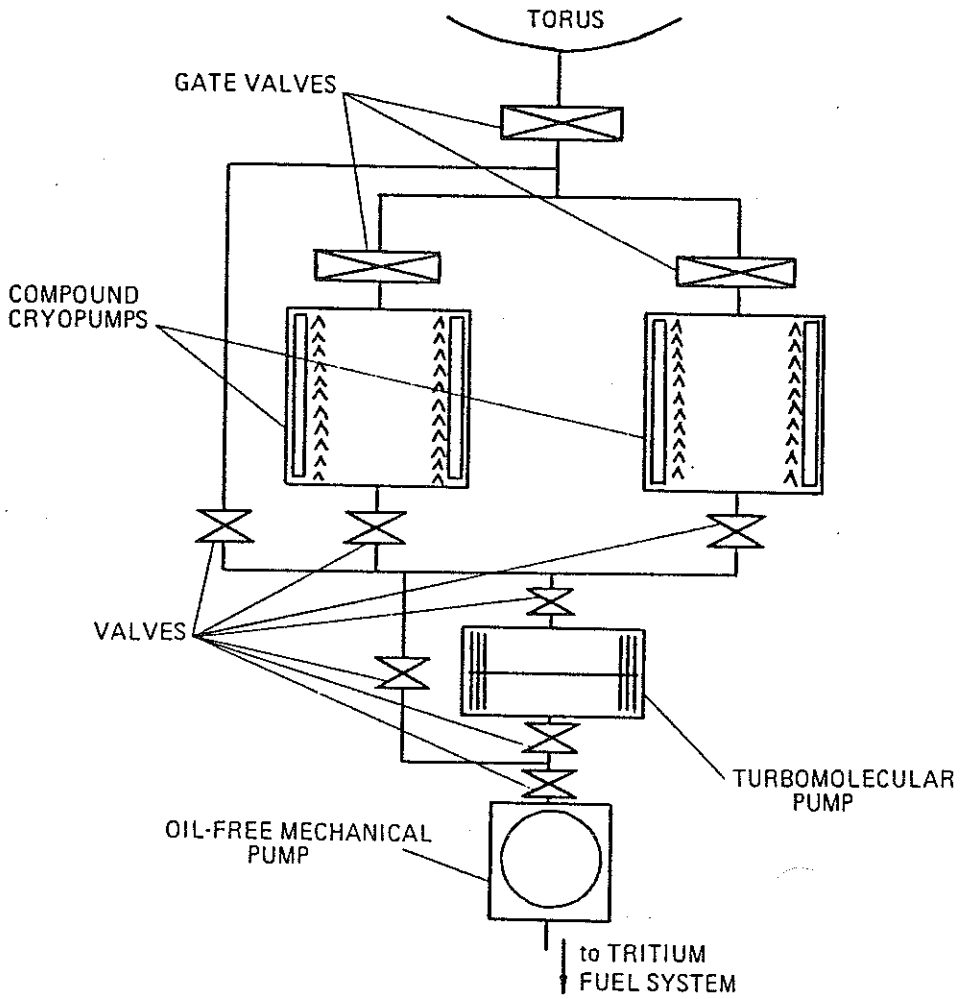


FIG.XI-2. Pumping unit of toroidal vacuum chamber.

With conventional cryosorption pumps of the refrigerator type the helium pumping speed reaches only 10–15% of the pumping speed for hydrogen; with modern cryopumps of the same type, however, the efficiency is up to 60% [2]. It is unlikely that the helium pumping speed can be increased by decreasing the cryosorption panel temperature or by using other cryosorbent materials.

Cryosorption pumps using microporous adsorbents, operated at ~4 K, can pump either hydrogen mixtures or helium, but not both simultaneously. However, it has been shown [6] that at the 4 K temperature required to cryosorb helium gas, these pumps actually pump the hydrogen isotopes by cryocondensation rather than by sorption. This solid hydrogen forms an effective hydrogen ice barrier between the gaseous helium and the cryosorption surface. Thus, these pumps are not acceptable for the torus vacuum system. Acceptable pumps are compound cryocondensation/cryosorption pumps [7]. These consist of a cryocondensation

panel working at 4 K, followed by a cryosorption panel also operating at 4 K. The hydrogen isotopes are efficiently pumped on the first-stage cryocondensation surface and the helium is pumped on the cryosorption panel. These compound cryopumps will meet all of the requirements of the torus vacuum system. The successful operation of compound cryopumps has recently been demonstrated [7]. Pumps of this type will be used to pump helium and D-T mixtures in INTOR.

Two compound cryopumps are arranged in parallel in each of the 12 pumping units. There are two pumps in each unit, so that toroidal chamber pumping by 12 pumps and simultaneous regeneration of the other 12 pumps is provided [3]. The regeneration time should be minimized in order to reduce tritium inventory in the reactor. The estimated time of continuous cryopanel operation is 2 hours, the tritium inventory in cryopumps being ~120 g.

In addition to compound cryopumps, turbomolecular and "dry" mechanical pumps (Fig. XI-2) are included in the system. Their purpose is to pump air from the chamber and to provide the base pressure and pumping when the cryosorption pumps are regenerated. The development of oil-free turbomolecular pumps with a pumping speed of about (2-3) $\times 10^4$ l/s is necessary for fusion devices of the INTOR type.

1.2. Vacuum system of neutral beam injectors

Four deuterium injectors operating simultaneously are used for heating the plasma. The power of the beams injected into the plasma is 75 MW, the particle energy is 175 keV; the injection time is 6 s for each working pulse.

The injector vacuum system includes the vacuum chamber in which a gas neutralizer and basic means for high-vacuum pumping are located and to which positive ion sources are connected. In addition, there are other high-vacuum and fore-vacuum pumps, valves and gauges.

The injector vacuum system should provide:

(a) An initial base pressure of 10^{-7} torr
(b) Vacuum conditions in the ion sources, neutralizers and injection lines that provide effective ion neutralization, minimum beam losses due to re-ionization not exceeding a few per cent, and minimum diffusion flow of cold deuterium from the injector through the drift tube into the discharge chamber
(c) Minimum tritium flow from the discharge chamber into the injector
(d) The capability of replacing the ion sources without breaking the injector vacuum
(e) The capability of pressure measurement and vacuum interlocking in the operating pressure range.

The ions can be effectively neutralized at 10^{-3} torr at the neutralizer input and at 10^{-4} torr at the neutralizer output. To minimize re-ionization losses and cold deuterium flow into the discharge chamber, the atom pressure in the tube

should be $\sim 5 \times 10^{-6}$ torr. Condensation pumps cooled by liquid helium are used as a basic means for high vacuum pumping. Helium pumping is not required in the injectors and therefore there is no need for more complex cryosorption pumps. The total cryopanel area depends on the maintenance of a tolerable pressure in the injector chamber and on the gas load value, which is determined mainly by the deuterium from the ion sources. The gas load due to the ion sources can be estimated by the ratio of 1 torr·l/s per 5 A of ion current [4]. For each injector the deuterium flow is 70—80 torr·l/s. The specific deuterium pumping speed on liquid-helium-cooled cryopanels is 8 l/s·cm^2. To obtain a mean pressure of 5×10^{-6} torr in the injector chamber, a cryopanel area of 200 m^2 is required, which is almost equal to the total area of all the surfaces in the chamber. To reduce the cryopump speed, a differential pumping system is used in the injectors, the liquid-helium-cooled panels being placed in diaphragm-separated chambers. Most of the gas is pumped in the neutralizer region at a relatively high pressure and a small amount of gas is condensed on the cryopanels close to the drift tubes. The differential pumping system permits a reduction of the cryopanel area by 45—50 m^2. The cryopumping speed in each injector is (3-4) $\times 10^6$ l/s. The number of differential pumping stages can vary from two to four and their quantity should be evaluated in future design.

Turbomolecular and oil-free mechanical pumps are used to obtain the initial base pressure of 10^{-7} torr and during cryopanel regeneration. To obtain the initial base pressure, the inner injector surfaces are outgassed by baking at 200°C. The cryopanel regeneration frequency depends on the quantity of deuterium consistent with the safety conditions in the event of a loss of vacuum in the injector, or on the tolerable quantity of tritium flowing from the discharge chamber into the injector. When the reactor availability is 50%, the tritium accumulation is ~ 1 g/day. To exclude explosion hazards during a loss of vacuum, the cryopanels should be regenerated twice a month.

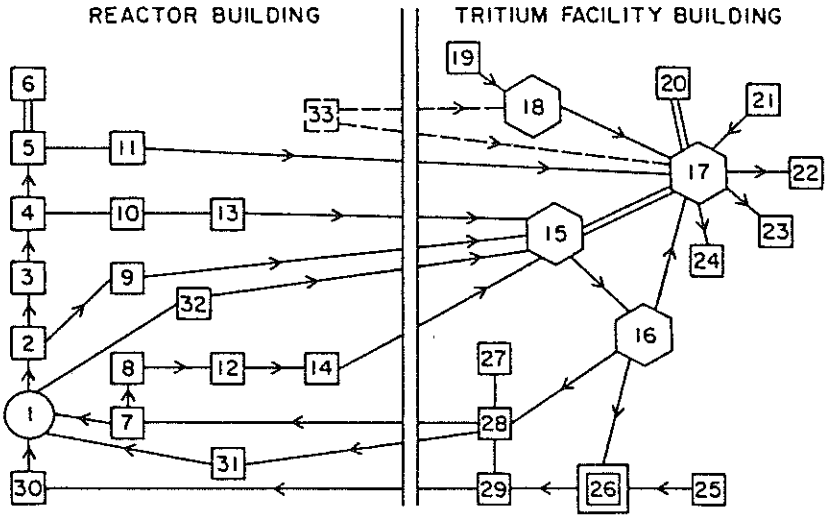
Vacuum gates arranged at the drift tubes provide for the maintenance of reactor chamber vacuum during injector replacement and cryopanel regeneration. Also, the fast shutter valves in the injector lines are opened only during injection, in order to limit tritium flow into the injector.

All ion sources are connected to the injector through gate valves so that the sources can be replaced without warming the cryopanels.

2. TRITIUM SYSTEM

2.1. Introduction

The INTOR fusion reactor device incorporates a complete deuterium-tritium-lithium fuel cycle. An important concern at the present level of the study is to estimate the magnitude of the tritium inventory with an accuracy as high as is compatible with a conceptual design.



1 PLASMA CHAMBER
2 DIVERTOR
3 DEBRIS SEPARATOR
4, 8 DT CRYOCONDENSATION PUMPS
5 HELIUM PUMP
6 SURGE TANK
7 NEUTRAL BEAM INJECTOR
9 TRITIUM RECOVERY, DIVERTOR
10, 11, 12 REGENERATION PUMPS
13, 14 METAL BELLOWS PUMPS
15 FUEL CLEAN-UP UNIT (FCU)
16 ISOTOPIC SEPARATION UNIT (ISU)
17 TRITIATED WASTE TREATMENT (TWT)
18 ATMOSPHERIC TRITIUM RECOVERY SYSTEM (ATR)
19 TERTIARY ENCLOSURE
20 WATER – TRITIUM RECOVERY UNIT (TWRU)
21 SECONDARY ENCLOSURE
22 DETRITIATED GASES: N_2, O_2, CO_2
23 HELIUM (TRITIUM – FREE)
24 TRITIUM WASTES
25 T_2 – SHIPMENT/RECEIVING
26 T_2 AND DT STORAGE
27 D_2 SUPPLY
28 D_2 STORAGE
29 FUEL BLENDER
30 PELLET FUELLER
31 D_2 – GAS FUELLER
32 TRITIUM RECOVERY BLANKET
33 EVACUATED BUILDING VACUUM SYSTEM AND/OR ATR

FIG.XI-3. Fuel cycle for INTOR.

2.2. General design considerations

The primary goal in the design of the tritium containment and handling systems for INTOR is to produce a completely integrated tritium fuel cycle, independent of any off-site tritium sources when the plant is operational.

The major objectives for the design are as follows:

(a) The tritium impact on the environment should be minimized
(b) Worker exposure should be reduced to levels as low as practicable
(c) Tritium waste generation should be minimized
(d) The tritium systems should be operated in areas free of gamma or neutron irradiation, where possible
(e) The tritium systems should be designed to have maximum reliability and availability, including redundancy of components.

The complete tritium fuel cycle is shown in Fig. XI-3. The tritium handling systems perform the following functions:

(a) Reprocessing of tritium for fuelling
(b) Processing of tritium produced in the blanket
(c) Processing of tritiated wastes
(d) Processing of tritium contained in the coolant
(e) Recovery of atmospheric tritium
(f) Control of the amount and location of tritium in the plant.

Tritium reprocessing includes units for chemical purification and isotopic enrichment of the fuel. The tritium generated in the blanket, mainly as T_2O (THO), is electrolysed and then passed to the main fuel cycle to be processed. The tritium-contaminated deuterium is pumped out of the neutral beam lines, chemically purified and then processed in the isotopic separation unit. The location of tritium in the plant is regulated by a storage and distribution system. Several treatment systems maintain low tritium levels in the secondary enclosures (controlled-atmosphere glove boxes and double-walled pipes) and in the coolant systems. The atmospheric tritium recovery systems are designed to scrub the atmosphere of a tritium handling area in the event of a tritium release. The tritiated wastes which accrue as by-products from various handling systems are detritiated and/or consolidated by several different units.

2.3. Tritium parameters for the INTOR design

Starting from different scenarios foreseen for INTOR and using the main parameters given in Table XI-1, the tritium parameters given in Table XI-3 are derived.

TABLE XI-3. TRITIUM PARAMETERS FOR THE INTOR DESIGN

	Stage I	Stage II	Stage III
Plasma parameters			
Thermal power (MW)	620	620	620
Plant availability	0.15	0.25	0.50
Average ion density (ions/m^3)	1.4×10^{20}	1.4×10^{20}	1.4×10^{20}
Plasma volume (m^3)	241	241	241
Fractional tritium burn-up	0.05	0.05	0.05
Burn time (s)	100	200	200
Duty cycle (%)	69	82	82
Tritium mass flow rates (100% availability)			
Tritium burned (g/d)	65	77	77
Tritium fuelled (g/d)	1305	1547	1547
Tritium exhausted (g/d)	1240	1470	1470
Tritium bred (g/d)	42	50	50
Breeding ratio	0.65	0.65	0.65
Deuterium exhausted (g/d)	826	977	977
Deuterium burned (g/d)	43	51	51
Neutral beams (100% availability)			
Deuterium pumped (g/d)	276	158	158
Deuterium injected (g/d)	78	45	45
Tritium pumped (g/d)	1	1	1
Annual balance			
Tritium burned (kg/a)	3.6	6.8	13.6
Deuterium burned (kg/a)	2.35	4.65	9.31
External tritium supply (kg/a)	1.3	2.23	4.46

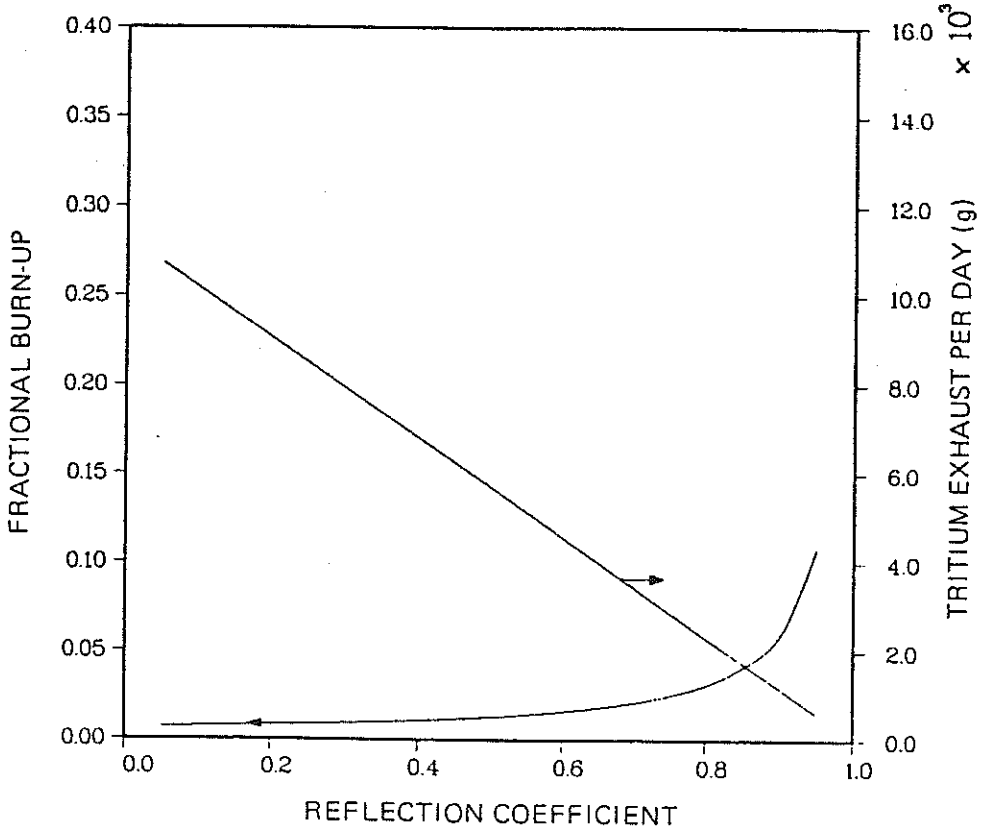


FIG.XI-4. Dependence of fractional burn-up and tritium exhaust per day on the reflection coefficient.

The INTOR tritium inventory must be a minimum. The impact of the INTOR plasma parameters (burn-up, particle confinement time, reflection coefficient) and engineering parameters (cryopump regeneration times) on this inventory was studied in detail [1] and is illustrated in Figs XI-4 to XI-10. One of the most important concerns is to decrease as much as practicable the regeneration time for cryopanels. With the present technology, a regeneration time of 2 h will ensure reasonable reliability and will decrease the pump inventory to ~120 g.

Tritium storage sufficient for 30 days of full operation is foreseen for INTOR. This requirement corresponds to a tritium inventory of 1950 g during Phase I and 2300 g during Phases II and III.

In the plasma reprocessing system, the minimum tritium inventory corresponds to a continuous process. Unfortunately, the initial operation is the regeneration of cryopanels, which is discontinuous. Also it is necessary to use batch components in parallel and optimized operation to minimize tritium inventory in the chemical purification step. The total tritium inventory in the plasma reprocessing system

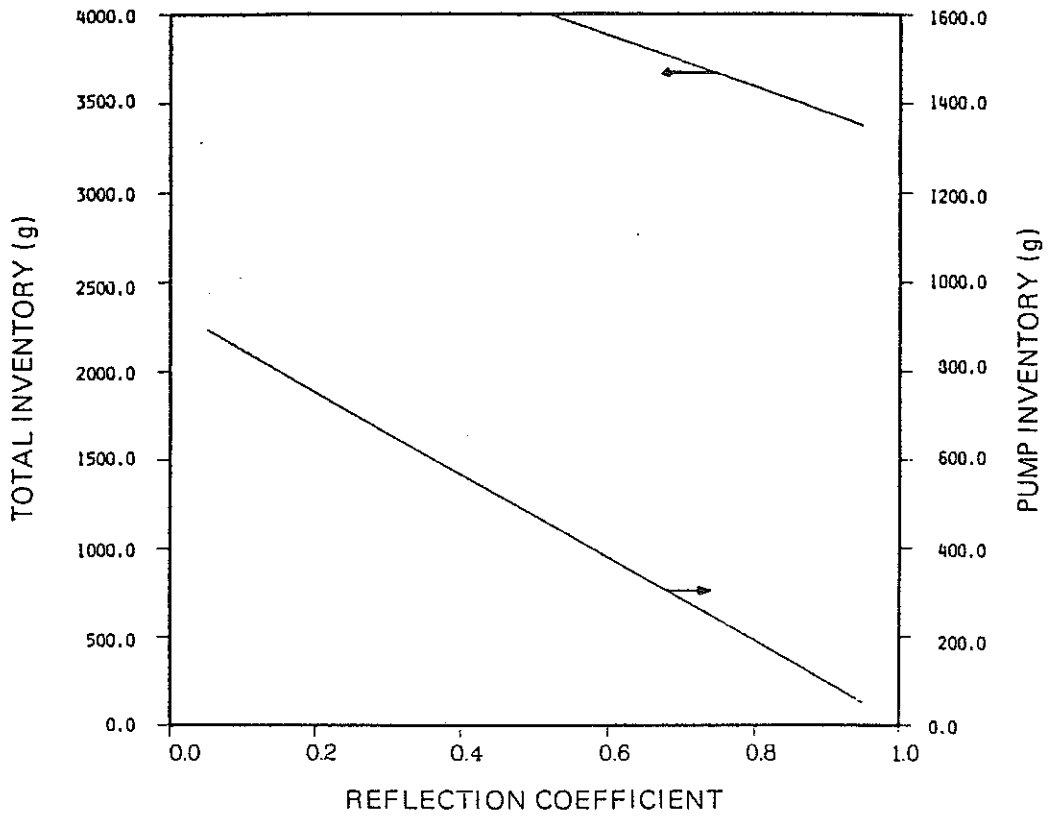


FIG.XI-5. Dependence of total tritium inventory and pump inventory on the reflection coefficient.

could be decreased to ~200 g, including the tritium inventory in the cryogenic distillation step.

The tritium inventory in pellet injectors is ~100 g for each injector, which gives 200 g of tritium inventory in the pellet fabrication system.

The tritium inventory in the blanket itself is not well known; it depends on a number of parameters as well as on radiation effects on tritium release from solid breeders. A reasonable estimate of the tritium inventory in the blanket is ~500 g to 1 kg. Estimates of the tritium inventory in various components of INTOR are given in Table XI-4. The designation "vulnerable" or "non-vulnerable" refers to the degree of control which can be enforced on the associated system. The tritium residing in the INTOR blanket is relatively immobile since it is in the oxide form and retained by the solid breeding material. The tritium in storage, ~2.3 kg, in the form of small units, each containing 12 g of tritium, is located in a barricaded vault with an inert cover gas for fire protection, thus making it relatively "non-vulnerable" to an accident.

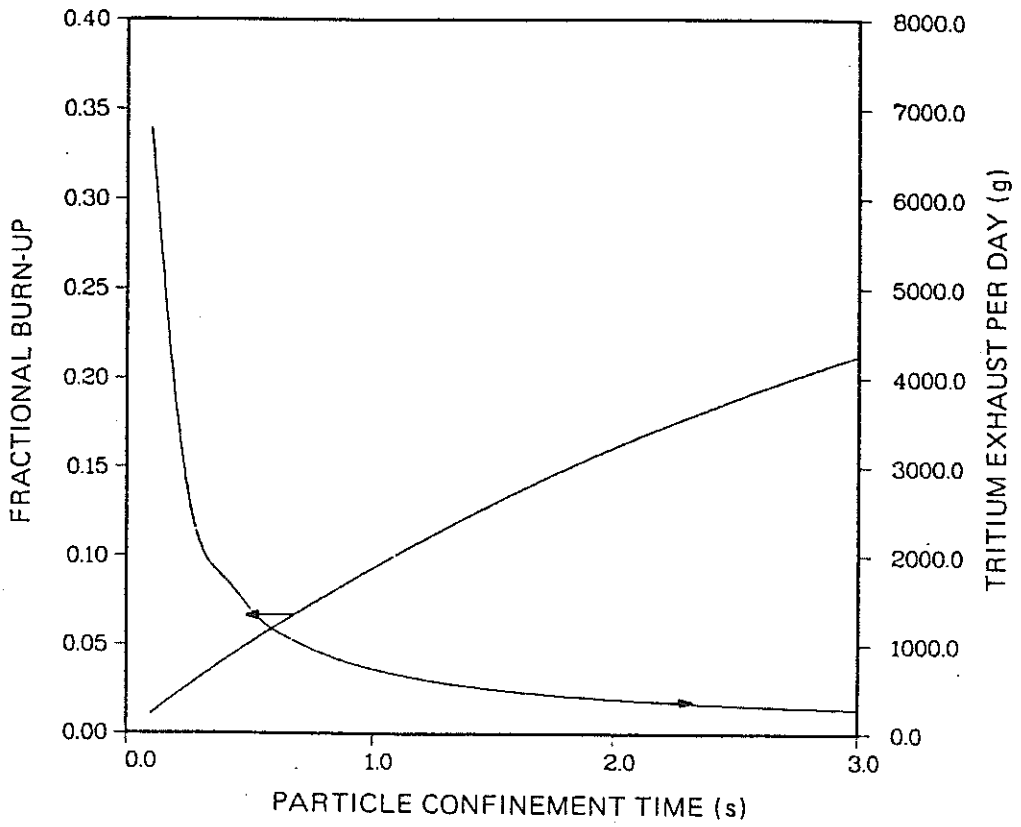


FIG.XI-6. Dependence of fractional burn-up and tritium exhaust per day on particle confinement time.

The fuel processing system contains tritium in the unoxidized form. Each tritium-processing unit is doubly contained and under monitor control to ensure maximum accident protection. Vacuum or an inert cover gas is present to reduce the fire hazard. The isotopic separation unit, which contains the highest tritium inventory, has high reliability. Nevertheless, this inventory is conservatively considered as "vulnerable" in safety analysis.

The tritium in the fuel handling system (vacuum pumps, gas puffing units, pellet fabrication and breeder tritium-recovery units in the reactor building) is considered to be "vulnerable" since the reactor area is subjected to severe thermal, magnetic and radiation loads which may interfere with the function of the units or may accelerate their aging. Chapter XVI gives details on this area of concern.

2.4. Fuel reprocessing

The main fuel pathway begins at the plasma chamber, proceeds to the cryopumps, and continues to the fuel clean-up unit. (The effluent from the neutral beams follows the same pathway.)

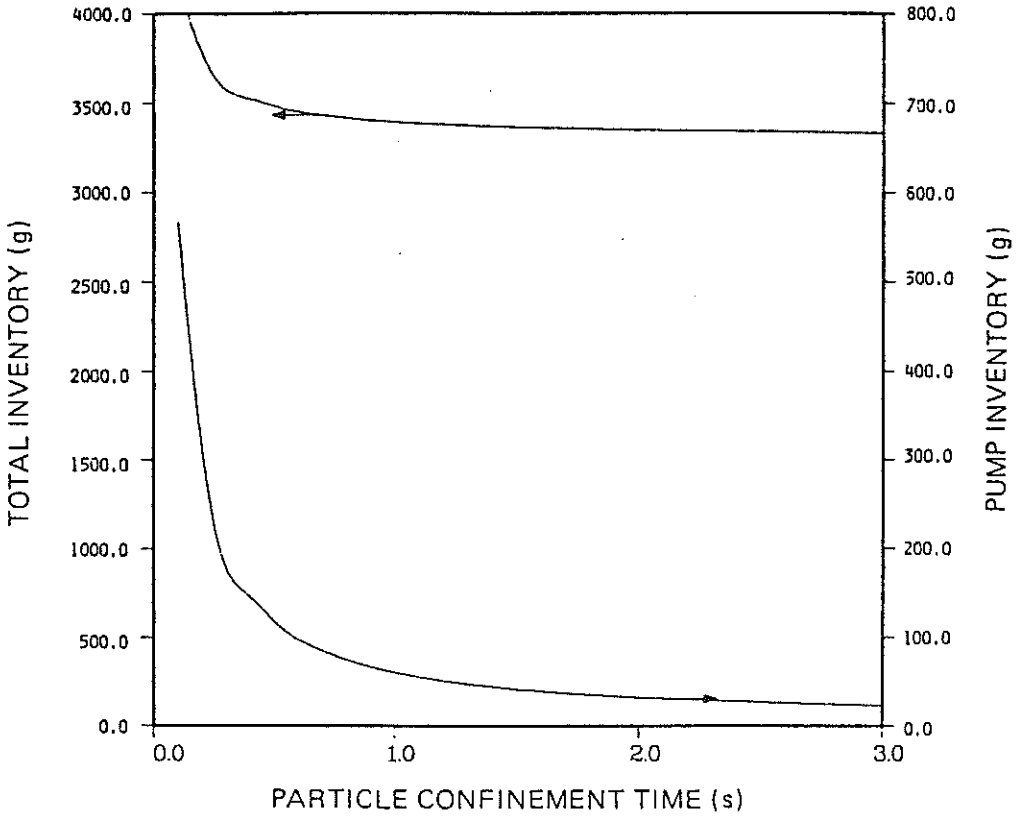


FIG.XI-7. Dependence of total tritium inventory and pump inventory on particle confinement time.

The following procedures for fuel reprocessing have been proposed by the INTOR participants: conversion of free oxygen into water at 450 K followed by cold-trapping, and helium removal from the deuterium-tritium mixture by means of a cryogenic film condenser or a cryogenic stripping column [4]; water cold-trapping followed by diffusion of the deuterium-tritium mixture through a palladium-silver membrane at 570 K [2, 3]; and trapping of chemically active impurities (O_2, N_2, CH_4) and hydrogen isotopes on a layer of hydride-forming metals, including separation of inert gases such as helium, followed by separation of impurities from the deuterium-tritium mixture onto a second metallic bed [5].

In all these concepts, in one step or another of the fuel clean-up system, it is necessary to use such a high temperature that tritium diffusion through the vessel wall is enhanced. Cold trapping of all gaseous impurities is necessary. Two units are needed in parallel: when the cold trap is exhausted, the flow is switched to the regenerated cold trap. The first trap is heated to room temperature and the impurities pumped to a high-temperature (800 K) catalytic oxidizer where their tritium content is recovered as DTO and can be re-used after reprocessing. The tritium-free compounds receive final detritiation in the tritium waste treatment

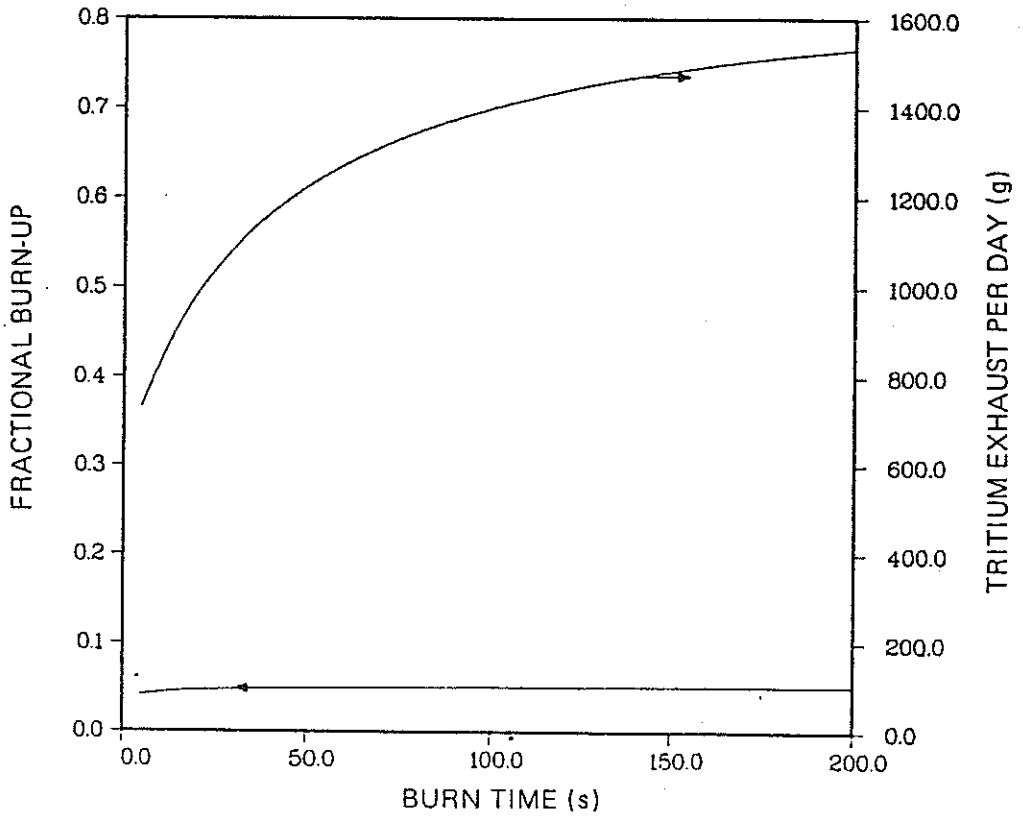


FIG.XI-8. Dependence of fractional burn-up and tritium exhaust per day on burn time.

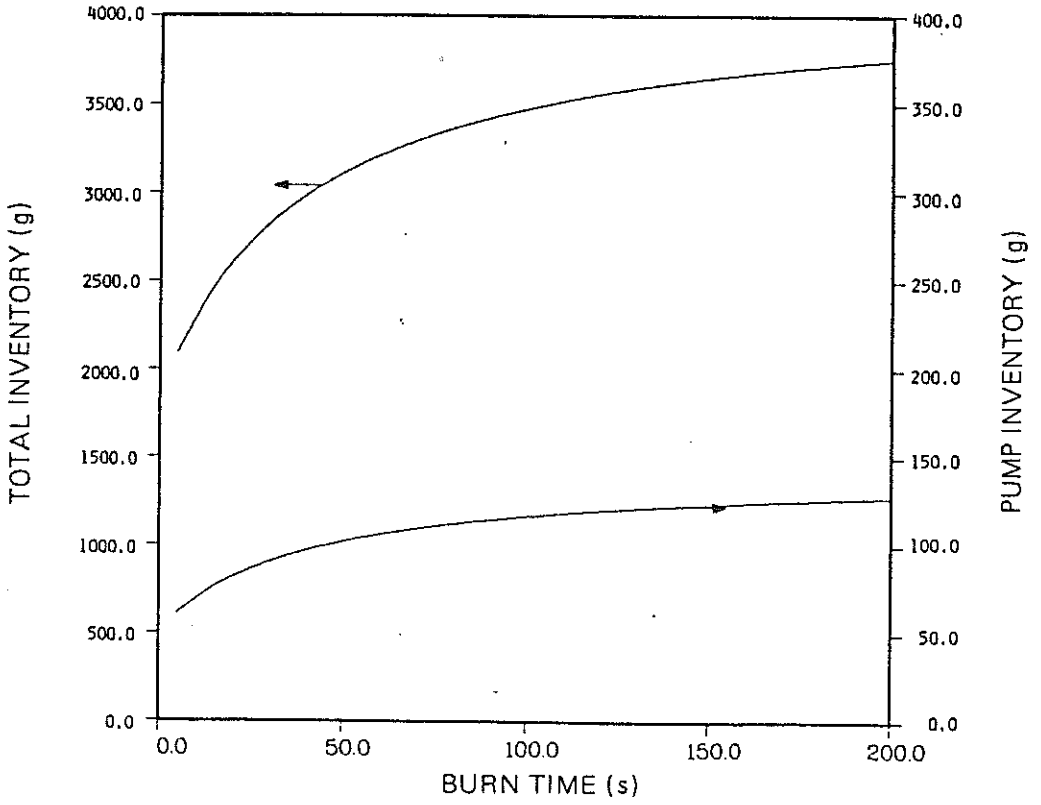


FIG.XI-9. Dependence of total tritium inventory and pump inventory on burn time.

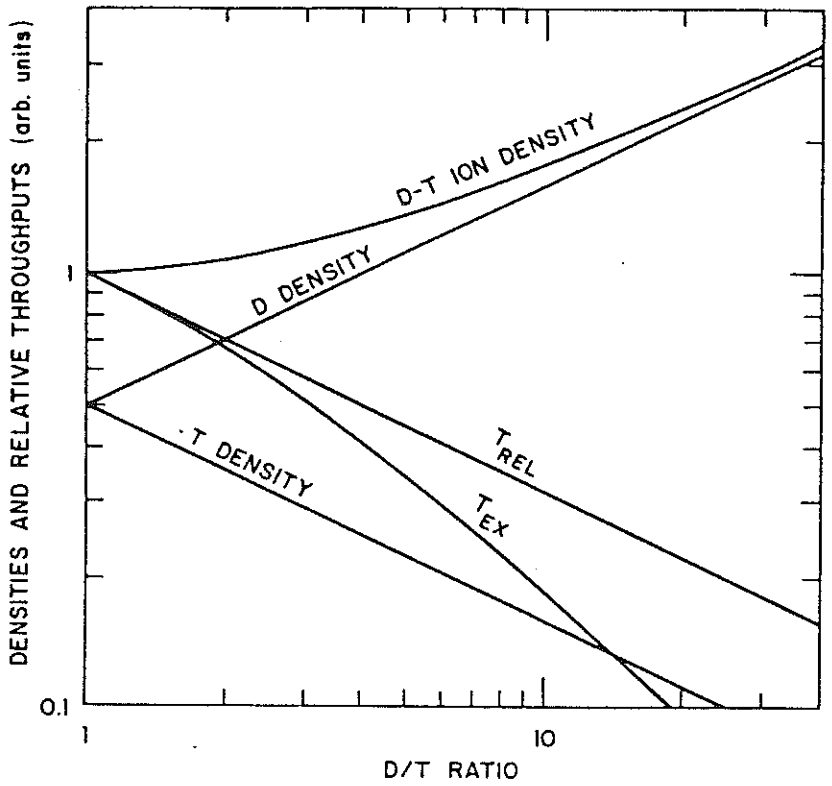


FIG.XI-10. Dependence of the tritium throughput rate for a fixed fusion power on the ratio of the deuterium/tritium ion density.

system (catalytic recombiner and drying beds) before being exhausted to the atmosphere.

The remaining elements of the fuel clean-up system are two cold (77 K) molecular sieve beds which collect small amounts of impurities from the neutral beam line. During their regeneration the accumulated impurities are transferred to the inlet of the fuel clean-up unit. A diagram of the fuel clean-up unit is shown in Fig. XI-11.

The common basis of the isotopic separation system is cryogenic distillation. To fulfil INTOR requirements ($\geqslant$90% D_2, $\geqslant$90% T_2, $\leqslant$ 1% H_2 in either stream), the system consists of four distillation columns and two equilibrators. The three product streams consist of a 99% pure T_2 stream ($<$ 1% D_2), a 99.96% pure D_2 stream ($< 3 \times 10^{-4}$ at.% tritium), a 48.4 at.% deuterium, 51.6 at.% tritium stream, and a waste protium-deuterium stream (99% HD) containing a residual amount of tritium ($< 10^{-4}$%). If a tritium extraction system for the coolant is required, this waste could be burned on a catalytic bed, separated, electrolysed, and directed to the detritiation unit for reprocessing.

Having passed the isotopic separation unit, the pure deuterium and tritium output is either stored until needed or directed to the fuel injectors where the needed fuel mixtures are prepared for use in the reactor fuelling devices.

TABLE XI-4. TRITIUM INVENTORY FOR THE INTOR DESIGN

Vulnerable tritium inventory (g)		
(a) Reactor building		
Vacuum pumps		120
Fuellers		20
Pellet fabrication		200
Blanket tritium recovery		50
Additional tritium inventory[a]		TBD[b]
(b) Tritium building		
Fuel processing		200
	Subtotal	590
Non-vulnerable tritium inventory (g)		
(a) Reactor building		
Breeding blanket		500–1000
(b) Tritium building		
Storage		2300
(c) Additional tritium inventory		TBD[b]
	Subtotal	2800–3300
Total tritium inventory (g)		3400–3900
Initial tritium inventory (g)		3000

[a] Expected to be relatively small compared with the total inventory.
[b] To be determined.

2.5. Bred tritium processing

Tritium is removed from the blanket mainly as T_2O by means of a helium purge stream. Taking into account the possibility of a small fraction of tritium gas being present in the purge gas, the first step of the tritium-breeding processing system consists of a conversion of this small amount of gas into water by passing it through a catalytic reactor in which, after injection of free oxygen, hydrogen is combined with oxygen over a "DEOXO" type catalyst at 450 K.

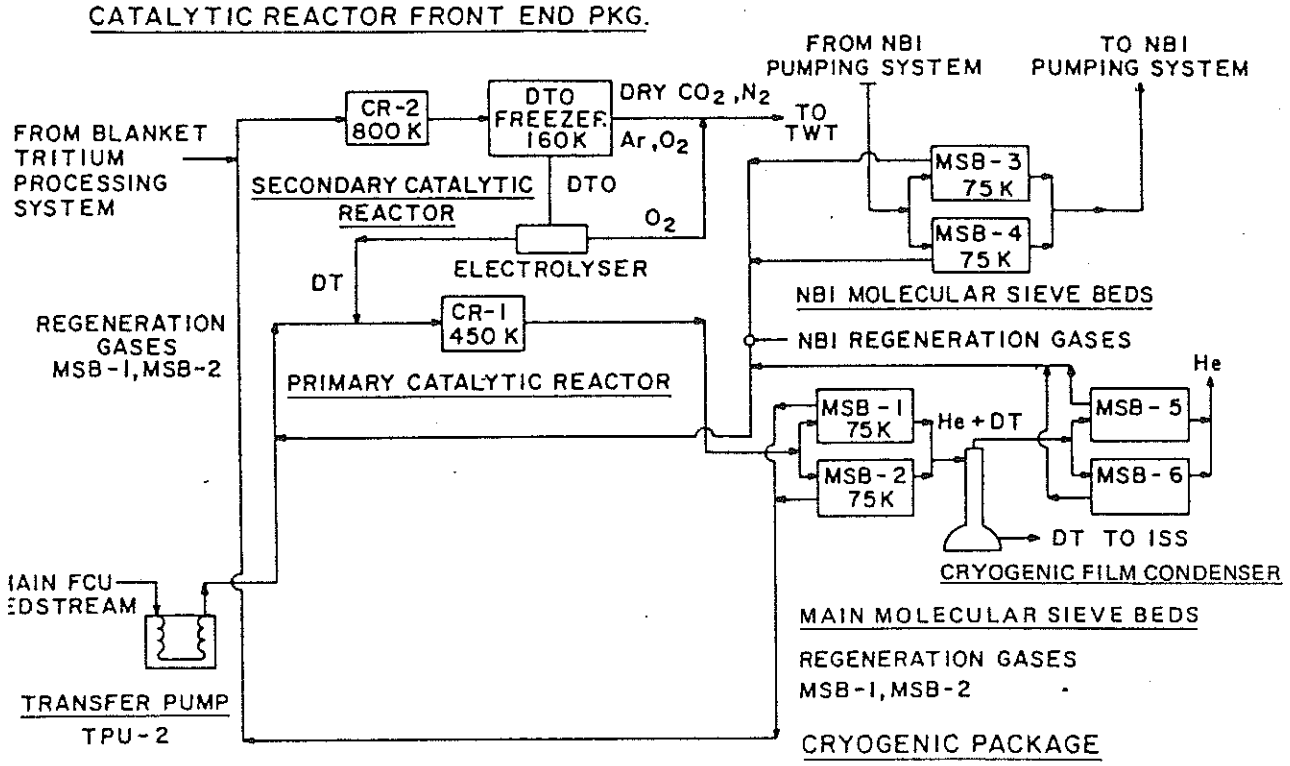


FIG.XI-11. Schematic diagram of the fuel clean-up unit process.

Thereafter, helium has to be cooled to a temperature of 80 K over molecular sieve beds to remove all its water content. Following the separation of the sweep gas from the water vapour, helium is heated again to room temperature before being sent back into the breeding blanket.

When required, the cold trap is heated to a temperature that makes it possible to recover its water content in liquid form. This water is transferred to an electrolytic cell where its decomposition into tritium and oxygen takes place. The recovered tritium is directed to the fuel clean-up system in the main stream and some of the recovered oxygen is sent back to the catalytic bed.

2.6. Permeation of tritium through the first wall and divertor plates

The permeation of hydrogen isotopes through structural materials has been studied extensively in the past. By classical diffusion theory, the permeation rate per unit area, at a given temperature, is proportional to the square root of the

TABLE XI-5. TRITIUM PERMEATION THROUGH THE FIRST WALL

Time of exposure at 573 K		10 h	100 h	1000 h	10000 h	33300 h
N[a]	(Ci/m^2)	5.9	19	53	59	59
J[b]	(Ci/m^2)	3.6×10^{-10}	5.9×10^{-6}	7.5	220	780

[a] N is the quantity of tritium contained in a 1-cm-thick austenitic stainless steel first wall.
[b] J is the quantity of tritium released through the outside surface.

pressure and inversely proportional to the square root of the atomic mass. Thus the permeation rate ϕ of T_2 through a membrane of area A and thickness t is given by:

$$\phi = \frac{1}{\sqrt{3}} \, k \left(\frac{A}{\tau}\right) (P_1^{1/2} - P_2^{1/2})$$

with P_1 being the upstream pressure and P_2 the downstream pressure.

Starting from this model for an upstream pressure of 1.3×10^{-3} Pa [2, 8], the tritium atom permeation through the first wall could be illustrated by the quantities given in Table XI-5. Extrapolating the values given in this table to the INTOR burning time, the amount of tritium permeation from the first wall into the coolant is of the order of 30 g at 573 K. This value is reduced by a factor of 10–100 for the average wall temperature in INTOR.

A thin oxide film on the coolant side of the wall will tend to serve as a tritium barrier that will decrease the permeation rate by at least two orders of magnitude. This would correspond to 0.01 Ci/d or a total of ~1 g of tritium in the coolant during the reactor lifetime.

Several factors could affect this value. A lower temperature gradient in the wall (~380 K on the coolant side) would probably decrease the permeation rate significantly. Radiation effects could either increase the permeation rate by enhancing the diffusion or decrease the rate because of trapping at defects. The permeation rate may also be higher because the tritium atoms coming out of the wall can be directly captured by water ions, thus by-passing the potentially rate-controlling step of recombination into gaseous molecules at low pressure. Possibly a more important factor is the effect produced by implantation of the tritium into the surface of the wall. A credible mechanism by which the tritium

permeation rate through the first wall could be increased over the calculated value includes the following steps:

(a) Implantation of high concentrations of tritium atoms in the inner surface of the first wall
(b) Relatively slow recombination of the tritium atoms at the surface
(c) Increased permeation of tritium through the wall because of the higher pressure gradient established.

At present, insufficient data exist to assess clearly the importance of this mechanism. Limited experimental data indicate that substantial increases in permeation rates occur in stainless steel at about 773 K [9]. However, recent data [10] have shown only modest effects (enhancement by a factor of ~ 2) from 10 keV implantations for wall temperatures below 500 K. Further investigations are required to assess more reliably the importance of this mechanism.

2.7. Coolant

To ensure that the tritium levels in the primary coolant are maintained at a low enough level, a tritiated water recovery unit must be included in the tritium system design. Theoretically, all processes used to produce heavy water are suitable to extract tritium from the water coolant. But, taking into account the complexity of many of these processes, the choice is restricted to two or three processes: water distillation, isotopic exchange between water and hydrogen followed by cryogenic distillation, and electrolysis plus cryogenic distillation.

A small fraction of the coolant (depending on the tritium permeation flow rate) has to be processed. Before the tritium can be extracted, the activated materials must be removed.

2.8. Atmospheric tritium recovery

To minimize the impact of tritium on the environment, the tritium containment systems are designed so that all tritium releases, whether due to normal leakage or to accidents, remain within the confines of the plant. Several tools have to be utilized to accomplish this goal. The outer walls of potential tritium-containing areas are lined with a tritium barrier. The materials and equipment used in a tritium area are selected to minimize surface adsorption. Tritium areas are operated at reduced atmospheric pressure to minimize tritium leakage. The atmospheric tritium level (HTO) in all areas is maintained at $<5\ \mu Ci/m^3$. Tritium systems utilize triple containment where reasonable, and double containment is used throughout. The tritium transport lines between different buildings are kept at a minimum in number and length and must withstand the design basis earthquake.

TABLE XI-6. TRITIUM RELEASES IN THE INTOR DESIGN

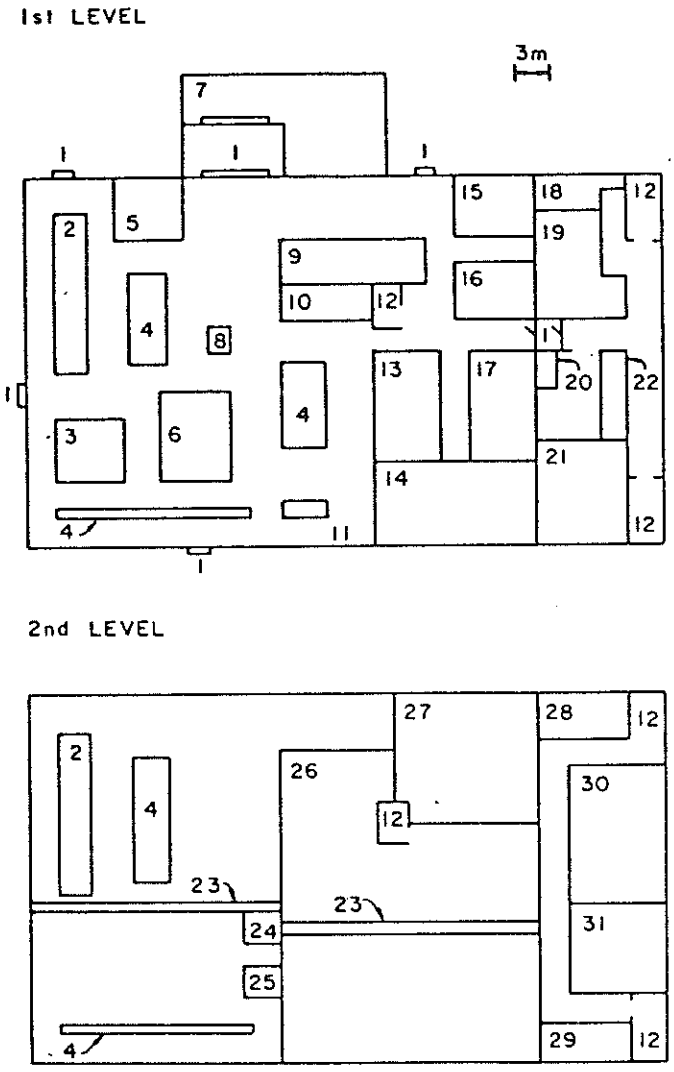
Coolant	
Steady-state concentration	0.003 – 0.03 Ci/l
Flow rate	50 – 1000 l/d
Tritium release	3 – 10 Ci/d
Buildings	
Volume	3×10^5 m^3
Tritium level	5 μCi/m^3
Tritium release	$\leqslant$ 1 Ci/d
Ventilation	
Volume	3×10^5 m^3
Tritium level	5 μCi/m^3
Number of releases	5 – 10
Tritium release	8 – 16 Ci/d
Solid waste	
Tritium waste	$\leqslant$ 1 Ci/d
Total tritium release	11 – 20 Ci/d

A timely decontamination of potential tritium spills within the containment is provided by:

(a) Dedicating an appropriately sized atmospheric tritium recovery unit to each tritium-containing area
(b) Interfacing the heating, ventilation and air conditioning systems with the air tritium recovery system to prevent a tritium release to the environment.

Multiple, discrete flow paths between units are used to partition the tritium inventory and thus reduce the magnitude of a tritium release.

The building atmosphere will be detritiated after each tritium release in order to minimize tritium release to the environment and tritium absorption on exposed surfaces. The basic operation consists of processing the atmosphere by catalytic oxidation of tritium gas to T_2O (HTO), followed by its absorption on molecular sieves.



1. AIRLOCK
2. ISOTOPE SEPARATION*
3. FUEL CLEAN-UP*
4. TRANSFER PUMPS*
5. SOLID WASTE DISPOSAL
6. D_2/H_2 STORAGE
7. DOCK
8. BREATHING AIR
9. TRITIATED WATER RECOVERY*
10. TRITIATED WASTE TREATMENT
11. FUEL BLENDER*
12. STAIRS
13. ANALYTICAL LABORATORY
14. TRITIUM VAULT
15. MAINTENANCE SHOP
16. USED BUBBLE SUIT
17. ELECTRONICS LABORATORY
18. LOBBY
19. RADIATION SAFETY
20. EMERGENCY SHOWER
21. LOCKER/SHOWER
22. NEW BUBBLE SUIT
23. CRANE
24. HELIUM STORAGE
25. NITROGEN STORAGE
26. ATMOSPHERIC TRITIUM RECOVERY*
27. HEATING/VENTILATION (-) PRESSURE
28. HEATING/VENTILATION (+) PRESSURE
29. LAVATORY
30. CONTROL ROOM
31. OFFICES

*Multiple units.

FIG.XI-12. Equipment of the tritium facility.

The quantity of tritium waste generated is minimized by: selecting and developing processes which inherently produce a minimum of solid waste; processing tritiated water; minimizing the use of organics; minimizing the contamination of organics where they are functionally necessary; using all-metal seals in valves, pumps, etc.; and reprocessing tritiated waste. Any waste generated within the plant is packaged in the smallest volume possible.

2.9. Tritium release design goal

The tritium release design goal for INTOR is $\leqslant$ 1 g/a in all forms (gas, liquid and solid waste), averaged over operating and maintenance phases and including in-plant releases. The sources of all releases are listed in Table XI-6.

2.10. Tritium building

The tritium building is completely separate from the reactor building, but is located nearby to minimize the length of the tritium feed-lines. The building is designed for contact maintenance, having an air environment and being free of gamma and neutron radiation. All tritium-handling units are under remote, independent computer control. The tritium storage area is sited in the tritium building. The tritium processing units are of modular design to facilitate replacement. Multiple units and full redundancy are provided to ensure maximum reliability and availability. The primary containment systems are made of metal (no elastomers). An inert dry atmosphere glove box is the secondary containment for most components. Double-wall piping is used between units.

To minimize worker contact with tritium in the tritium building, which is designed for contact maintenance, the following procedures are applied:

(a) Atmospheric tritium levels are maintained at <5 μCi/m^3
(b) An atmospheric tritium recovery unit with a flow rate of 0.5 vol·%/min is provided
(c) Normal hydrogen fire prevention practices are exercised.

An illustration of the tritium building equipment is shown in Fig. XI-12.

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INTERNATIONAL TOKAMAK REACTOR
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REPORT OF THE
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