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IAEA INTOR WORKSHOP REPORT, GROUP 14
—— VACUUM ——

October 1979

Fusion Research and Development Center

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IAEA INTOR Workshop Report, Group 14
—— Vacuum ——

Fusion Research and Development Center
Tokai Research Establishment, JAERI

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Vacuum systems data base assessment and evaluation of the principal vacuum parameters for INTOR-J concepts were made. This report covers the torus vacuum vessel and its pumping system including vacuum wall location, pumping means, available maximum pumping speed, and vacuum wall cleaning. Emphasis in the conceptual studies was on helium pumping by cryopumps. Regeneration cycle and tritium inventory of the cryopumps were evaluated from the published data. Radiation effects on vacuum systems and remote leak detection were also studied.

Key words: INTOR, Tokamak Reactors, Vacuum Systems, Vacuum Technology, Vacuum Vessel, Cryopumps, Helium Pumping, Wall Cleaning, Radiation Effects, Remote Leak Detection, Data Base Assessment, Conceptual Studies

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IAEA INTOR ワークショップ検討報告書・グループ14

――― 真 空 技 術 ―――

日本原子力研究所東海研究所核融合研究開発推進センター

（1979年9月27日受理）

国際トカマク炉（INTOR）の概念設計に必要な真空技術に関するデータベースについて調査検討を行うとともに，INTOR－Jの主要な真空パラメータの評価を行った。本報告では，トーラス真空容器とその排気系について記述されており，真空壁の位置，真空ポンプの選定，採りうる最大排気速度，真空壁の清浄化などについて検討されている。特に，クライオポンプによるヘリウム排気の問題点や再生の頻度等について詳しく評価している。真空系に及ぼす放射線の影響や遠隔操作による洩れ検知法についても述べている。

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CONTENTS

目　　次

1. Introduction and Summary

Vacuum systems data base assessment and evaluation of the principal vacuum parameters for INTOR-J concepts were performed by the authors after First Session of IAEA INTOR Workshop.

Since the INTOR concept tries to sustain 100 - 200 second burning, impurity control, especially pumping of produced helium is one of major issues of the vacuum systems. There have been proposed two major Japanese concepts about helium exhaust from plasma. One of them is a concept of simple poloidal divertors[5] and the other is a concept called "Wall Lapping Plasma".[6] For both concepts, the essential requirement is to pump out helium of 1.6×10^{20} atoms/sec and accompanying other impurity gases during the burn.

Other important problems arise from the use of D-T fuel and the irradiation due to photon and neutron production. It is necessary to recover the pumped gases as quickly as possible in order to achieve low inventories of tritium. It also is required that no significant quantity of tritium escape to the environment. The vacuum system should be adaptable to the circumstances of limited space, electromagnetic field, and photon and neutron radiation.

These new issues necessarily restrict the pumping means or devices of INTOR. The pumps which should be considered include cryopumps, turbomolecular pumps without lubricating oils and mercury diffusion pumps with cold traps. Although there is considerable experience with oil-lubricated turbomolecular pumps in torus vacuum systems of tokamaks and with cryocondensation pumps in neutral beam injectors, larger dry turbomolecular pumps and new configurations for cryopumps may be needed for INTOR. Other pumps such as mechanical booster pumps without oils and bulk getter pumps may be used as an auxiliary means. Novel pumping means such as a palladium-alloy membrane pump for hydrogen isotopes and ion implantation for helium have

been proposed, but these are untested ideas.

The available maximum pumping speed to the vacuum vessel depends on the vessel structural design; specifically on the proximity of pumping ports to magnetic field coils or primary shields. In the INTOR-J design, the vacuum vessel is divided into six individual segments each of which can be removed and replaced by remote handling techniques. If we assume six pumping ducts with 1.5 m high, 1 m wide and 5 m long, and that the pumping speed of the cryopumps is the same as the duct conductance, the total effective pumping speed at the pumping duct inlet or the divertor vessel will be about 500 m^3/sec for helium, with six pumps of 170 m^3/sec each. It is sufficient to say that the helium production rate of 1.6×10^{20} atoms/sec can be removed by the cryopump system only if the helium pressure (at 300 K) is higher than 1.2×10^{-3} Pa at the duct inlet. If higher pressures can be attained, smaller cryopump systems or dynamic pump systems such as using turbomolecular pumps and mercury diffusion pumps will fulfill the requirement.

Cryosorption pumps for helium pumping are now under investigation and have not been put to practical use. When molecular sieves are used as the adsorbent, it has been recognized that the pump may not be able to accommodate helium-hydrogen isotopes mixtures, because condensed deuterium and tritium will block the adsorbent surface and prevent helium pumping. This probably means that the cryosorption panels (4.2 K) for helium will be surrounded by two chevrons, one at 77 K and the other at 4.2 K. In this case the expected maximum speeds per unit projected area are about 30 $m^3/sec \cdot m^2$ for helium and about 78 $m^3/sec \cdot m^2$ for deuterium (at 300 K), and the ratio of helium speed to deuterium speed is smaller than 0.4. Thus, the required area for the helium pumping speed of 170 m^3/sec for a single pumping unit can be estimated to be about 6 m^2.

Regeneration time of the cryopumps must be short enough not to affect

the duty cycle or the availability of the INTOR experiments. The regeneration cycle will be determined by considering several items which include safety problems, tritium inventory and heat loading to cryopanels due to tritium decay. In a preliminary design, there are 12 cryopumping units, six of which are in operation and the remainder are in regeneration. The regeneration cycle is 12 hours (6 hours in operation and 6 hours in regeneration). The inventory of tritium in the cryopumping system will be lower than 0.15 kg. Incidentally, cryopanels will be placed in well-shielded locations without sacrifice of conductance to reach the panels in order to reduce the heat deposition due to neutrons and gamma-rays.

The adsorbed and occluded species in the surface layer of the vacuum walls are the source of low-Z impurities in the tokamak plasma. Therefore, cleaning of the walls is an important factor in controlling the impurity release. There exist four methods of cleaning which have been employed to remove adsorbed and occluded impurities in the surface layer; baking, chemical treatment with gases, discharge cleaning, and in-situ coating. It may be possible to use the radiation-induced outgassing by electrons and gamma-rays.

The vacuum walls of INTOR will have to be baked in vacuum at a temperature of around 250°C. However, baking at higher temperatures does not seen very practical because of the bulky and complicated structure. Other ways of surface cleaning such as pre-baking to a temperature in excess of 450°C will be needed before the final assembling. No organic elastomers should be used in the vacuum boundaries because of their sensitivity to bake-out temperature, radiation and tritium.

The dwell time of 30 seconds is long enough to pump out the volume gases in the vacuum vessel. However, the transient behavior of the residual gases are not simply related to the pumping time constants, and provides evidence for the role of surface adsorption and diffusion on the evacuation

time for the vacuum vessel.

The leak detection system in fusion reactors will be designed on the assumption that all vacuum vessel joints including major structural weld joints of the vessel must be capable of being leak checked. Leak location would also be done by detecting a local pressure rise within the vacuum vessel due to gas influx from a leak. Other R&D items for remote leak detection will be high resolution mass analyzers, remotely controlled movable mechanisms for leak sensors in three dimensional vacuum space, and radiation effects on the leak sensors and their related systems.

The vacuum development program for INTOR will include design, fabrication, testing, and evaluation of vacuum components and systems as necessary in order to meet the objectives. Development and testing of advanced methods of helium pumping, surface cleaning, remote leak detection, etc. must be performed to support the specific vacuum system designs.

2. Vacuum Wall Location

The choice of vacuum boundary location as well as the choice of repair and maintenance scheme will greatly influence the concept of the fusion reactor design. Therefore, detailed design study must be conducted before a choice of vacuum boundary location.

In Figs. 1(a) - 1(c), configurations after Japanese designs of INTOR are shown for both concepts with divertors and without divertor. The vacuum wall is located at the back of the blanket and inside the shield. Location of the vacuum wall behind the blanket means that the blanket and its related structures are included in the vacuum vessel. The immediate implications are an increase in vessel volume which influences the evacuation time and an additional increase in outgassing surface area which affects the ultimate pressure in the vacuum vessel. In this arrangement, the neutron and gamma-ray radiation level on the vacuum wall would be 0.02 MW-year/m^2 and the temperature of the wall would be around 100°C during operation.

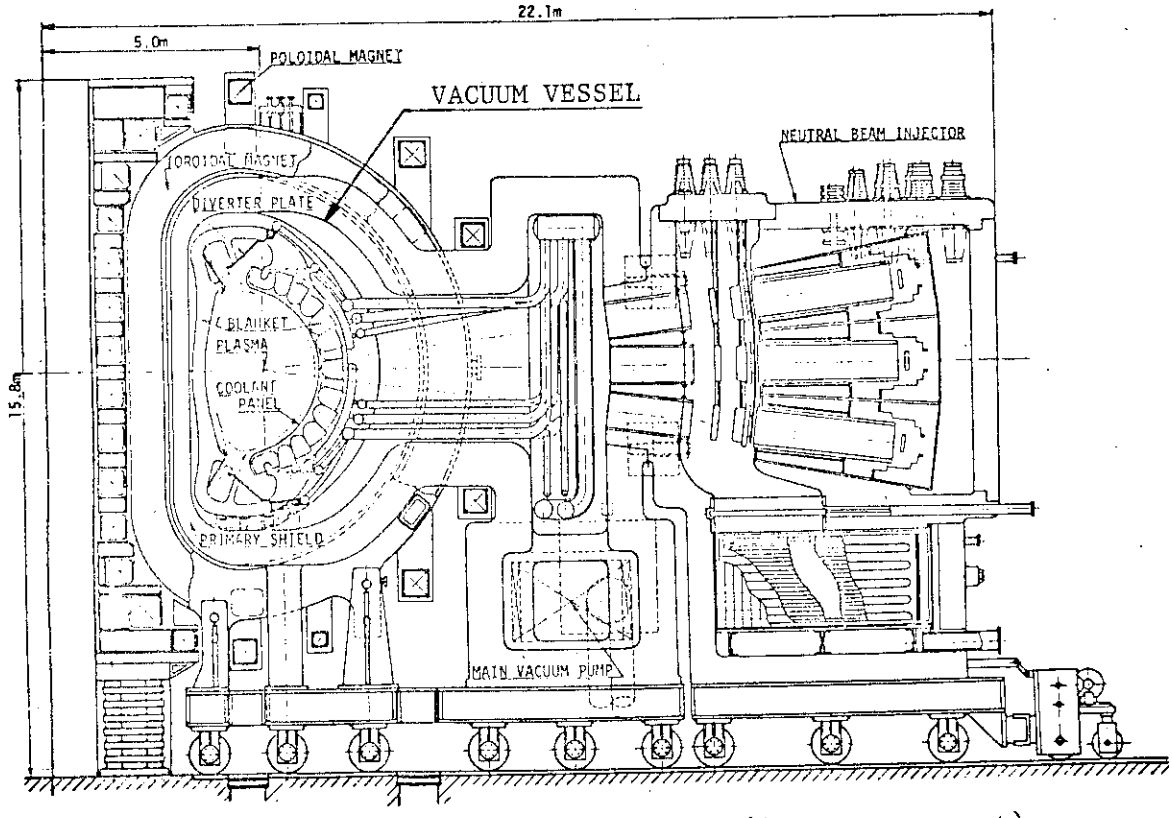

Fig. 1(a) Vertical view of INTOR-J (divertor concept).

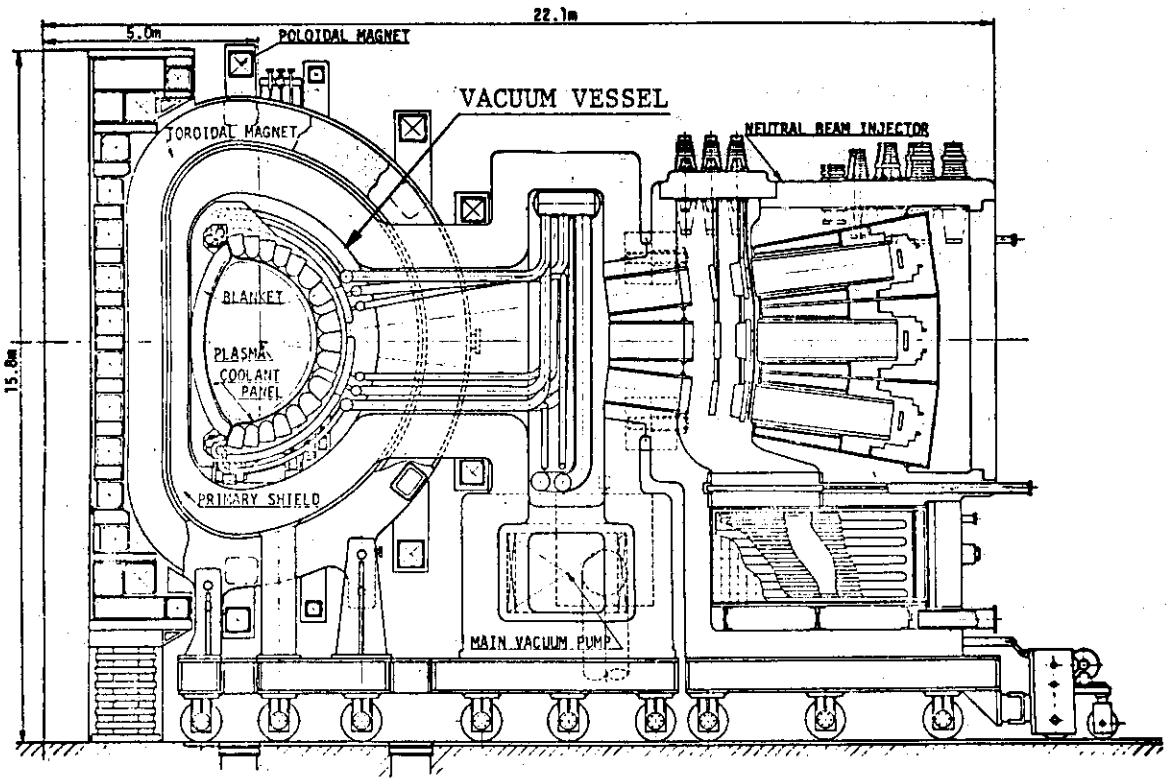

Fig. 1(b) Vertical view of INTOR-J ("Wall Lapping Plasma" concept).

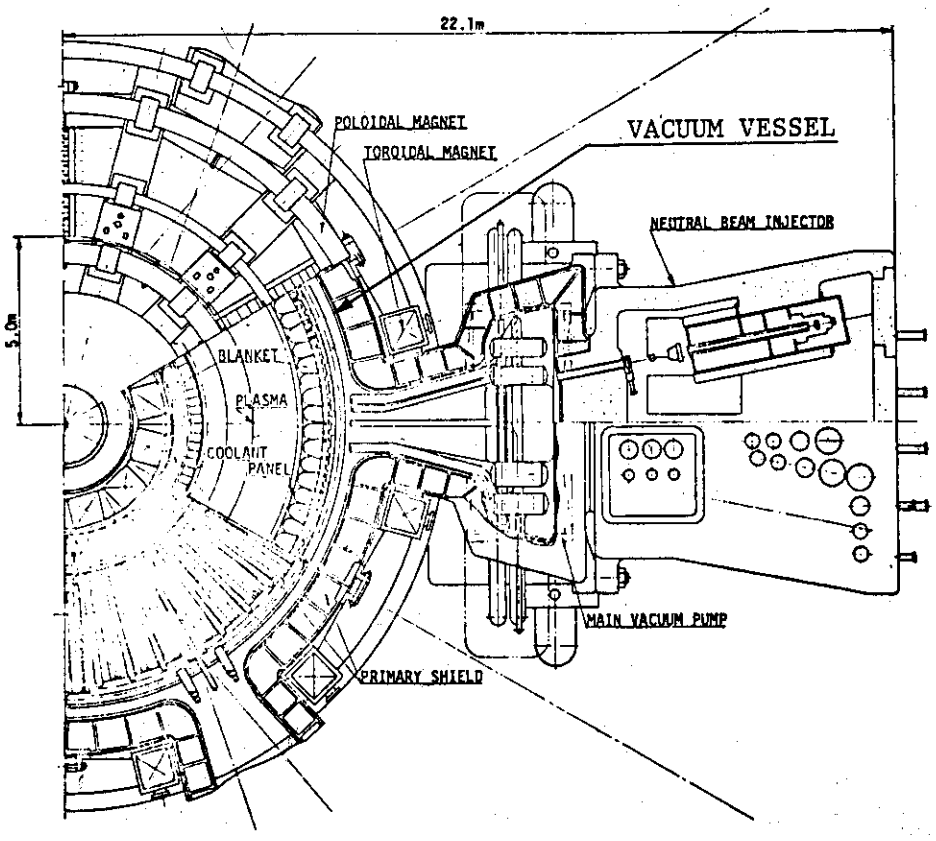

Fig. 1(c) Plane view of INTOR-J ("Wall Lapping Plasma" concept).

3. Data Base for Large (>100 m^3) Vacuum Systems in Japan

In Japan, the space simulation program has experienced very large vacuum systems of fast pumping down and low ultimate pressures. The Space Environmental Simulation Testing Facility was built at Tsukuba Space Center of NASDA in 1975.[1] The 1,350 m^3 vacuum vessel is pumped with a 200 m^3/sec and a 3,000 m^3/sec cryopumps (20 K), two 50 m^3/sec Ti-sublimation pumps and three 0.8 m^3/sec sputter-ion pumps. The ultimate pressure of about 10^{-7} Pa can be obtained within 15 hours from atmospheric pressure.

As to the large tokamak development, tests of the major components of JT-60 and integrated preconstruction design were finished. These tasks were performed by Japanese five major nuclear industry groups under the contracts with JAERI. For the tests, some components in full scale (e.g. a full scale sector of the vacuum vessel) were manufactured. As a first step of JT-60 construction, a contract for manufacturing the tokamak machine including the vacuum system was made in April, 1978 with Hitachi Ltd. and ULVAC is subcontracted with Hitachi Ltd.

The main vacuum parameters of JT-60 are summarized in Table 1. The vacuum vessel is an all-metal and completely welded composite torus. Inconel 625 has been chosen as the material for the vessel because of its high mechanical strength at high temperatures, high electrical resistivity, good weldability, etc. The vessel is bakeable to 500°C. The limiter material is chosen as molybdenum in JT-60, but it may be altered at a later stage of the experiments, depending on the progress made with impurity control.

A base pressure of $p \leq 1.33 \times 10^{-6}$ Pa in the vessel is aimed at. The total leak rate should not exceed $Q_L = 6.65 \times 10^{-8}$ Pa·m^3/sec. An outgassing rate of $q \leq 1.33 \times 10^{-8}$ Pa·m^3/m^2·sec may be attained for both Inconel 625 and molybdenum after 500°C bake out. The total gas release

Table 1 Main vacuum parameters of JT-60

Major/minor radii of torus	3m/1m x 1.5m
Inner volume	~100 m^3
Materials vacuum vessel liner, limiter	 Inconel 625 molybdenum
Surface area vacuum vessel liners and limiters	 ~500 m^2 ~750 m^2
Baking temperature	up to 500°C
Main pump	turbomolecular pumps with liquid nitrogen traps
Pumping speed at the vacuum vessel	~14 m^3/sec for nitrogen
Permissible outgassing rate	$\leq 1.33 \times 10^{-8}$ Pa·m^3/sec·m^2
Permissible total leak rate	$\leq 6.65 \times 10^{-8}$ Pa·m^3/sec
Ultimate pressure	$\leq 1.33 \times 10^{-6}$ Pa

from 1.25×10^3 m^2 wall (liner and vessel) surface is estimated to be $Q_o \leq 1.25 \times 10^3 \times 1.33 \times 10^{-8} = 1.66 \times 10^{-5}$ Pa·m^3/sec. The consideration of these requirements led to the design of the torus pumping system consisting of turbomolecular pumps with liquid nitrogen cold traps and their backing mechanical pumps. The effective pumping speed is about 14 m^3/sec. Metal gate valves and large vacuum seals are currently being tested. Much experimental work of these components remains to be done to establish their reliability.

The development of cryopumps for neutral beam injectors of JT-60 is now under way.[2] Each pumping unit has a hydrogen pumping speed of about 1.3×10^3 m^3/sec. The required cryopanel area is about 14.5 m^2. Since the typical operating pressure at the cryopump is below 6.7×10^{-3} Pa,

Table 2 Heat loading to the cryopanel in a JT-60 NBI unit

	Pulsed load	Continuous Load
Radiation from 77 K plate	-	9.9 W
Radiation from room temperature materials	9.5 W	19.0
Support loss	-	2.0
Condensation of hydrogen 4 $Pa \cdot m^3/sec$	3.6	0.12
Backscattered hydrogen	0.1	0.003
Cyclotron radiation	25	0.42
Eddy current	15	0.25
Total	53.2 W	31.7 W

the cryopanel temperature should be lower than 3.6 K. Two types of cryopanels are being evaluated. One of them is composed of a set of aluminum pipes with fin plates, and the other is a stainless steel kilting vessel. The former is cooled by two phase flow helium and the latter by liquid helium pooled in the vessel. The expected heat loads to a single cryopanel for the former case are tabulated in Table 2. The total refrigeration capacity for the 14 pumping units is estimated to be about 1,000 W at 3.5 K. Parallel distribution of liquid helium to 14 units is an important problem for the neutral beam injection system. A simulation test for the system will be made using a prototype injection unit.

4. Torus Vacuum System for INTOR

4.1 Selection of pumping means

The torus vacuum system is one of key components to achieve reactor relevant long burning. Since the INTOR concept tries to sustain 100 - 200 second burning, pumping of produced helium is one of major issues of INTOR. Other important problems come from the use of D-T fuel and the irradiation due to photon and neutron production. It is necessary to recover the pumped gases as quickly as possible in order to achieve low inventories of tritium. It also is required that no significant quantity of tritium escape to the environment. The vacuum system should be adaptable to the circumstances of limited space, electromagnetic field, and photon and neutron radiation.

These new issues necessarily restrict the pumping means or devices of INTOR. The use of oil diffusion pumps and mechanical pumps with oil lubricated bearings and elastomer seals should be eliminated because tritium tends to exchange with hydrogen in organic materials causing contamination and deterioration of the materials. As the tritium decays, it releases beta rays which cause damage by breaking chemical bonds leading to cross-linking and polymerization of the materials.

The remaining pumping means or devices which should be considered for INTOR include turbomolecular pumps without lubricating oils, mercury diffusion pumps with cold traps (dynamic pumps), cryocondensation pumps, cryosorption pumps, bulk getter pumps such as Zr-Al getters, and perhaps a palladium-alloy membrane pump (surface pumps). There is considerable experience with oil-lubricated turbomolecular pumps in torus vacuum systems of tokamaks and cryocondensation pumps are to be used in neutral beam injectors. These pumping mechanisms are reasonably well understood, but larger dry turbomolecular pumps and new configurations for cryopumps may be needed for INTOR. Cryocondensation pumps, getter pumps and palladium-

alloy membrane pumps are not effective for pumping helium and should be considered as hydrogen (or D-T) pumps. The palladium-alloy membrane pump may have a large pumping speed for hydrogen,[3] but it has not been tested yet.

The efficiency of these pumps for pumping deuterium may be evaluated with the pumping speed per unit surface area or occupied volume (surface or volume efficiency) and with that per unit power input (power efficiency). Table 3 shows a comparison of deuterium pumping speed per unit projected

Table 3 Comparison of deuterium pumping speed per unit surface area and that per unit power input for different pumping means

Pumping means	Deuterium pumping speed		Notes
	per unit surface area ($m^3/sec \cdot m^2$)	per unit power input (m^3/kJ)	
Turbomolecular pump	30- 60	> 10	
Mercury diffusion pump	20- 30	~ 0.3	with a baffle and a trap
Cryocondensation pump	70- 80	~ 10	with a chevron baffle
Bulk getter pump	~ 70	~ 30	working temp. = 200°C
Pd-alloy membrane pump	50-100	3 -10	at 10^{-3} Pa.

area of the intake port (in case of non-surface type pumps) or the active surface (in case of surface type pumps) and that per unit power input for the above mentioned pumps. In general, dynamic pumps have lower unit pumping speeds than surface pumps and are larger in physical size.

Here, the comparison is made on rather large pumps in order to reduce the size effect. Technical data were taken from the catalogues of some vacuum companies for the conventional pumps and from literatures for the cryopump[2),4)] and the palladium-alloy membrane pump.[3] It is clear where each pump ranges,

although the design and thus the operating condition will vary in a certain range.

4.2 Available maximum pumping speed

There are two major Japanese concepts about helium pumping from the plasma. One of them is a concept of simple poloidal divertors[5] and the other is a concept called "Wall Lapping Plasma",[6] at which plasma is lapping the wall with rotating resonant helical islands by external helical coils of small currents and helium is exhausted through simple valves. For both concepts, the essential requirement is to pump out helium of 1.6×10^{20} atoms/sec and accompanying other impurity gases and D-T during the burn.

The maximum size of the ports on the side of the vacuum vessel is dependent on the vessel structural design; specifically on the proximity of a port to toroidal field coils, poloidal field coils or primary shields. In an INTOR design without divertors, for example, the vacuum vessel is divided into six individual segments each of which can be removed and replaced by remote handling techniques. The above considerations result in six ports that are 1.5 m high with 1.0 m wide. The ports are commonly used with neutral beam injectors.

With a tubing length ℓ = 5.0 m, we can get a conductance per pumping port of

$$\Gamma = \frac{97.4\sqrt{T}}{\sqrt{M}} \alpha \frac{a^2 b^2}{(a+b)\ell} = \frac{19.8\sqrt{T}}{\sqrt{M}} \ (m^3/sec), \qquad (1)$$

where α = 1.13 is a correction factor and M and T are the mass number and temperature of the gas, respectively. On the other hand, the volume of the gas impinging into the same pumping port per second is given by

$$\Gamma' = \frac{36.5\sqrt{T}}{\sqrt{M}} A = \frac{54.8\sqrt{T}}{\sqrt{M}} \ (m^3/sec). \qquad (2)$$

Thus, the Clausing factor of the tube, which is the probability of a gas molecule impinging into the port passing through the tube, can be expressed as

$$C = \frac{\Gamma}{\Gamma'} \simeq 0.36, \tag{3}$$

where we assume that the angular distribution of the desorbed molecules from the wall obeys the cosine law.

From Eq.(1), if the pumping speed of the pump could be the same as the port conductance, the effective pumping speed at the port inlet will be about 86 m^3/sec for helium at 300 K. The total effective pumping speed applicable under these conditions is about 514 m^3/sec, with 6 pumps of 171 m^3/sec each.

The pumping speed of 170 m^3/sec for a single pumping unit is fairly high as compared with those for torus pumping systems of present-day large tokamaks such as JT-60 and TFTR. On the other hand, a cryopump unit for the neutral beam injector of JT-60 is now under design with a pumping speed of about 1.3×10^3 m^3/sec for hydrogen. Although the helium pumping speed per unit projected area in cryosorption pumps is estimated to be lower than 40 % of the deuterium speed in cryocondensation pumps, it may be possible to construct a cryosorption pumping unit with a helium pumping speed of 170 m^3/sec.

From these considerations, it is sufficient to say that the proposed helium production rate of 1.6×10^{20} atoms/sec can be removed by the above-mentioned cryopump system only if the helium pressure (at 300 K) is higher than 1.2×10^{-3} Pa at the port inlet. If higher pressures can be attained, smaller cryopump systems or dynamic pump systems such as using turbomolecular pumps and mercury diffusion pumps will fulfill the requirement.

4.3 Cleaning of the vacuum walls

The adsorbed and occluded species in the surface layer of the vacuum walls are the source of the low-Z impurities in the tokamak plasma. Therefore, cleaning of the walls is an important factor in controlling the impurity release. Chemical cleaning treatments are carried out to reduce gross contamination such as oil, grease and small particulates. After this process the surface is still covered with water vapor, oxygen and carbon. There exist four methods of cleaning which have been employed to remove adsorbed and occluded impurities in the surface layer; baking, chemical treatment with gases, discharge cleaning, and in-situ coating.

4.3.1 Baking

Baking the walls can reduce the surface concentration of adsorbed and occluded molecules by thermal desorption. Outgassing rate of various wall materials have been measured at both room temperature and 500°C in JAERI in cooperation with ULVAC[7] (see Table 4). The outgassing rate of unbaked stainless steel, Inconel and molybdenum is normally in the range 10^{-4} - 10^{-7} $Pa \cdot m^3/sec \cdot m^2$, depending on the pre-treatments and the time for which they have been maintained under vacuum. The main desorbed species is water vapor, with lesser quantities of carbon monoxide and hydrogen. After baking for 100 - 200 hours at 500°C, however, the outgassing rate decreases to less than 10^{-8} $Pa \cdot m^3/sec \cdot m^2$ at room temperature.

On the other hand, it has been reported[8] that the total quantity of gas desorbed from an as-received stainless steel sample during a vacuum bake-out at 450°C is 4.7 $Pa \cdot m^3/m^2$. This is equivalent to 114 monolayers of adsorbed gas. The surface was then exposed to air for 24 hours, and a subsequent bake-out desorbed a further 11 monolayers of gas. The large quantity of gas released during the initial bake-out is principally gas

Table 4 Outgassing rates from SS-304L, YUS-170, Inconel 625, Hastelloy-X, molybdenum, pyrolytic graphite and SiC-coated graphite [Pa·l/sec·cm^2]

		Before bake-out				During bake-out				After bake-out			
		Time from starting pumping (hour)				Time after the temperature reaches 500°C (hour) *1				Time after the heater is turned off (hour)			
		1	5	10	20	1	5	10	50	1	5	10	20
Stainless steel SS-304L	Total	2.9E-6*4	5.3E-7	2.7E-7	1.3E-7	3.5E-5	7.6E-6	4.0E-6	2.3E-7	9.0E-9	2.5E-9	1.3E-9	6.7E-10
	m/e= 2(H_2)	1.9E-7	5.6E-8	2.9E-8	1.7E-8	2.5E-5	6.5E-6	2.7E-6	1.9E-7	8.5E-9	2.0E-9	9.3E-10	-
	18(H_2O)	1.5E-6	2.4E-7	1.3E-7	7.8E-8	3.9E-7	-	-	-	1.7E-10	-	-	-
	28(CO)	1.5E-7	3.6E-8	1.5E-8	1.1E-8	8.2E-6	7.3E-7	2.3E-7	3.9E-8	1.5E-10	3.4E-11	-	-
Stainless steel YUS-170	Total	5.2E-7	3.1E-8	1.3E-8	4.1E-9	3.7E-7	1.1E-7	5.7E-8	1.7E-8	3.4E-9*2	(4.0E-10)	(2.0E-10)	(8.8E-11)
	m/e= 2(H_2)	8.4E-8	3.2E-9	1.6E-9	3.8E-10	5.7E-8	2.5E-8	2.3E-8	4.3E-9	3.1E-9*2	-	-	-
	18(H_2O)	1.7E-8	1.3E-8	2.3E-9	2.0E-9	1.0E-8	1.3E-9	-	-	-	-	-	-
	28(CO)	2.7E-8	2.9E-9	1.3E-9	5.6E-10	2.3E-7	4.9E-8	2.7E-8	4.9E-9	1.3E-11*2	-	-	-
Inconel-625	Total	9.3E-6	1.3E-6	6.4E-7	2.6E-7	-	-	5.2E-6	2.4E-7	3.3E-10	1.4E-10	1.1E-10	7.3E-11
	m/e= 2(H_2)	9.0E-7	1.7E-7	1.2E-7	9.0E-8	-	-	4.1E-6	5.7E-8	1.7E-10	6.7E-11	6.9E-11	5.3E-11
	18(H_2O)	2.1E-6	4.5E-7	2.2E-7	1.6E-7	-	-	-	-	-	-	-	-
	28(CO)	7.7E-7	2.1E-7	1.3E-7	1.0E-7	-	-	8.3E-7	1.7E-7	6.0E-12	4.1E-12	3.4E-12	2.9E-12
Hastelloy-X	Total	1.7E-6	1.3E-7	6.7E-8	3.3E-8	-	1.6E-6	9.6E-7	2.0E-7	(3.7E-10)	(3.6E-11)	-	-
	m/e= 2(H_2)	2.8E-7	1.9E-8	1.0E-8	5.3E-9	-	2.0E-7	6.9E-8	2.4E-8	(2.3E-10)	(1.1E-11)	-	-
	16(CH_4)	5.1E-8	5.1E-9	2.8E-9	1.3E-9	-	-	-	-	(1.7E-11)	-	-	-
	18(H_2O)	7.3E-7	5.3E-8	2.7E-8	1.1E-8	-	3.6E-8	-	-	-	-	-	-
	28(CO)	8.0E-8	6.7E-9	3.7E-9	2.1E-9	-	1.2E-6	6.0E-7	1.3E-7	(1.1E-11)	(6.7E-12)	-	-
Molybdenum	Total	1.1E-5	1.6E-7	3.2E-8	4.7E-9	2.5E-6	6.3E-7	4.0E-7	1.6E-7	1.7E-10	9.3E-11	6.9E-11	5.1E-11
	m/e= 2(H_2)	1.1E-6	1.3E-8	4.0E-9	5.3E-10	6.2E-7	1.6E-7	8.0E-8	2.0E-8	9.3E-11	4.4E-11	2.7E-11	1.5E-11
	18(H_2O)	4.0E-6	7.6E-8	1.3E-8	1.9E-9	5.2E-7	1.3E-7	8.0E-8	1.7E-8	1.7E-12	9.3E-13	8.0E-13	8.0E-13
	28(CO)	8.0E-7	1.2E-8	3.1E-9	3.3E-10	7.7E-7	1.7E-7	1.3E-7	7.3E-8	3.2E-11*3	2.5E-11*3	2.7E-11*3	2.7E-11*3
	44(CO_2)	3.6E-7	2.9E-9	6.7E-10	-	-	-	-	-	-	1.6E-12	1.1E-12	7.7E-13
Pyrolytic graphite	Total	1.1E-6	1.3E-7	5.3E-8	2.0E-8	1.1E-5	4.7E-6	2.6E-6	1.0E-6	(1.2E-9)	(1.6E-10)	(1.3E-10)	(1.3E-10)
	m/e= 2(H_2)	2.4E-7	1.7E-8	7.3E-9	3.7E-9	1.5E-6	1.8E-7	1.0E-7	4.0E-8	(8.6E-10)	(1.3E-10)	(1.0E-10)	(8.0E-11)
	16(CH_4)	-	-	-	-	-	-	-	-	(6.0E-11)	(4.7E-12)	(2.7E-12)	(2.0E-12)
	18(H_2O)	4.8E-7	6.4E-8	2.5E-8	9.3E-9	6.9E-7	6.7E-8	-	-	-	-	-	-
	28(CO)	1.0E-7	1.1E-8	4.7E-9	1.9E-9	7.3E-6	3.3E-6	2.0E-6	7.3E-7	(3.3E-11)	(8.0E-12)	(5.6E-12)	(2.7E-12)
SiC coated onto graphite	Total	1.5E-5*1	3.6E-6	1.3E-6	4.8E-7	2.4E-5	5.2E-6	3.5E-6	-	-	-	-	-
	m/e= 2(H_2)	4.8E-6*1	8.0E-7	4.0E-7	1.5E-7	1.5E-5	4.8E-6	1.9E-6	-	-	-	-	-
	16(CH_4)	4.5E-7*1	6.7E-8	-	-	-	-	-	-	-	-	-	-
	18(H_2O)	2.7E-6*1	4.0E-7	1.6E-7	3.7E-8	-	-	-	-	-	-	-	-
	28(CO)	2.3E-6*1	8.0E-7	2.7E-7	1.3E-7	2.8E-6	5.3E-7	1.5E-7	-	-	-	-	-

*1: The outgassing rates at 2 hours from starting pumping.
*2: The outgassing rates at 0.35 hours after the heater is turned off.
*3: Nitrogen is a dominant gas in this stage.
*4: The expression 2.9E-6 shows 2.9×10^{-6}.

occluded in the surface layer during manufacturing, welding, etc. and this is permanently removed by a single bake to a temperature in excess of 450°C. The much smaller quantity of gas adsorbed on the clean surface when exposed to air, principally of water vapor and oil vapors, can be removed by a subsequent 250°C bake-out.[9)]

It has been reported that baking to higher temperatures can be effective in causing carbon and oxygen to dissolve in the bulk material.[10)] Figure 2 shows the change in surface concentration of a stainless steel sample which is heated up in steps of 100 degrees from room temperature to 900°C at intervals of one hour. As the temperature is increased, phosphorus and sulfur appear on the surface, while carbon and oxygen disappear. In Inconel, similar changes occur, but at slightly higher

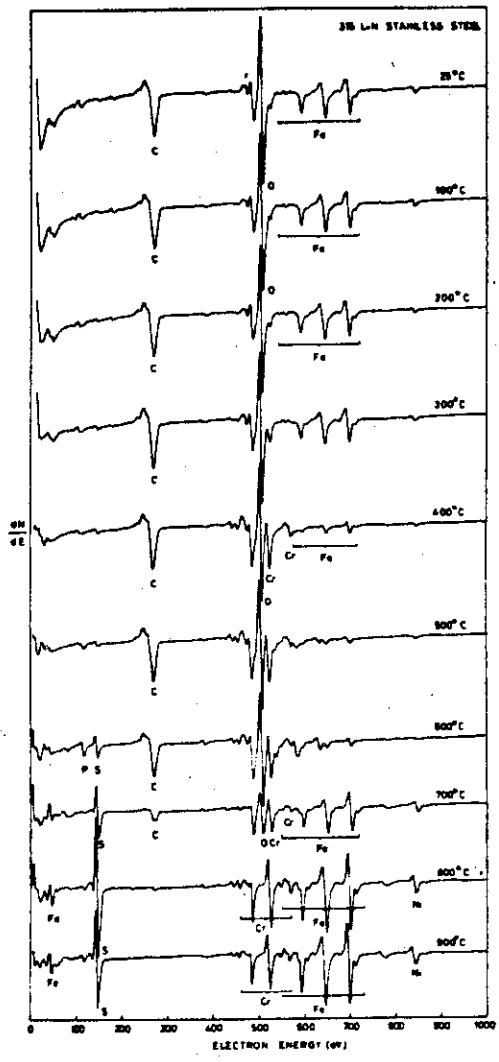

Fig. 2 Auger spectra from 316L+N stainless steel as a function of temperature.

temperatures.

In a proposed design concept for INTOR the blanket and its related structures are included in the vacuum vessel. The temperatures of the protection wall (low-Z coated refractory metal), cooling pipes, blanket walls, and vacuum vessel (stainless steel) are ∿1,400°C, ∿300°C, ∿200°C and ∿100°C, respectively. Assuming the total inner surface area A $\simeq 1 \times 10^4$ m^2, the pumping speed S $\simeq$ 200 m^3/sec (for the impurity gases except helium), and the ultimate pressure Pu $\leq 10^{-5}$ Pa (lower than 10^{-4} of the filling pressure), we can find the required outgassing rate being less than 2×10^{-7} Pa·m^3/sec·m^2.

In conclusion, the vacuum walls of INTOR will have to be baked in vacuum for several hours at a temperature of around 250°C. However, baking at higher temperatures does not seen very practical because of the bulky and complicated structure. Other ways of surface cleaning such as pre-baking to a temperature in excess of 450°C will be needed before the final assembling. No organic elastomers should be used in and through the vacuum vessel because of their sensitivity to bake-out temperature (as well as radiation and tritium).

4.3.2 Chemical treatment with oxygen and hydrogen

It has been shown that carbon and oxygen on molybdenum surface can be reduced to 1/4 of the initial concentration by successive exposing to oxygen (∿10 Pa) and hydrogen (∿100 Pa) at temperatures above 500°C.[11] Figure 3 shows a typical change in surface composition of a molybdenum

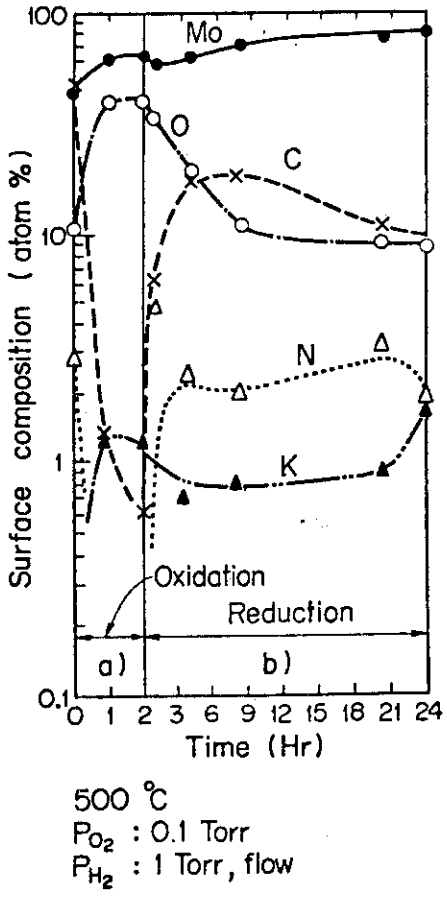

Fig. 3 Typical change in surface composition of molybdenum during high temperature treatment with oxygen and hydrogen.

sample during the treatment. Prior to the treatment the sample was exposed to air for 4 days after a glow discharge cleaning.

As to graphite surface it has also been demonstrated that the surface can be made passive to chemical reaction with hydrogen by an atomic hydrogen shower to a fluence in excess of 10^{21} atoms/sec·m^2 at around 500°C.[12] A measurement of methane production rate from pyrolytic graphite exposing to atomic hydrogen is shown in Fig. 4 as a function of sample

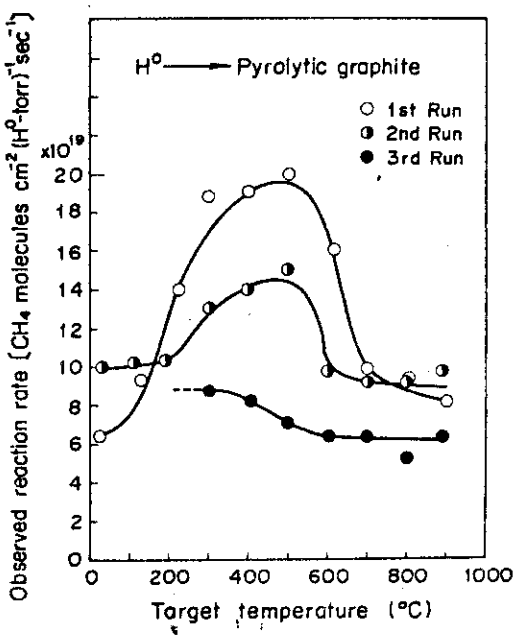

Fig. 4 Methane production rate from pyrolytic graphite as a function of temperature.

temperature. An incident flux of 2 x 10^{18} hydrogen atoms/sec·m^2 was used for the measurement. The sample was heated up in steps of 100 degrees from 100 to 900°C. In the first and second runs a peak in the methane production rate was observed between 500 and 600°C. In the third run, for which the sample had been exposed to atomic hydrogen to a fluence of about 10^{21} atoms/sec·m^2 at above 500°C and subsequently cooled down in high vacuum, the rate decreased and the peak disappeared in the measured temperature range. The inactive graphite surface has been found to be restored again to the former state if it is exposed to an oxygen containing gas such as air.

4.3.3 Discharge cleaning

The adsorbed species which are not removed by baking or chemical reaction can be reduced by hydrogen discharge cleaning. Two modes of discharge cleaning have been studied to date. One of them is high-power discharge cleaning (HDC) with a power density to the wall of $\sim 10^4$ W/m^2 and an electron temperature of $\sim$100 eV. The other is low-power (Taylor-type) discharge cleaning (LDC) with a power density of $\sim$500 W/m^2 and an electron temperature of $\sim$5 eV. The predominant residual gases produced during HDC are CH_4 and CO, whereas LDC produced primarily H_2O and CH_4.[13] The optimum plasma temperature in LDC may be in a region such as

[energy of dissociation of H_2] < T_e < [energy of dissociation of H_2O],

i.e., $4.5 \text{ eV} < T_e < 9.6 \text{ eV}$.

Recently, a LDC mode with an electron cyclotron resonance (ECR) plasma has been proposed by IPCR[14] and been tested in JFT-2. By this technique, a continuous plasma can be produced and it may increase the conditioning speed.

4.3.4 In-situ coating

A technique of impurity control by in-situ coating with titanium has now been applied successfully to most tokamaks. The long-term effects of coating the walls are not known, but there is some concern that the coated film may peel and cause a deterioration in the system.

4.4 Evacuation time for the vacuum vessel

The vessel evacuation time will depend on the design of the INTOR vacuum system. Unless the adsorbed and occuluded gases in the surface layer could be concerned, the pressure (p) vs. time (t) relation in the vacuum vessel is given by the equation

$$p = p_i \exp(-\frac{S}{V} t) + p_u, \tag{4}$$

where p_i is the initial pressure at $t = 0$, p_u the ultimate pressure in the system, V the volume of the vacuum vessel, and S is the effective pumping speed of the torus pumping system.

Figure 5 shows the predicted pumpdown of a vacuum vessel following

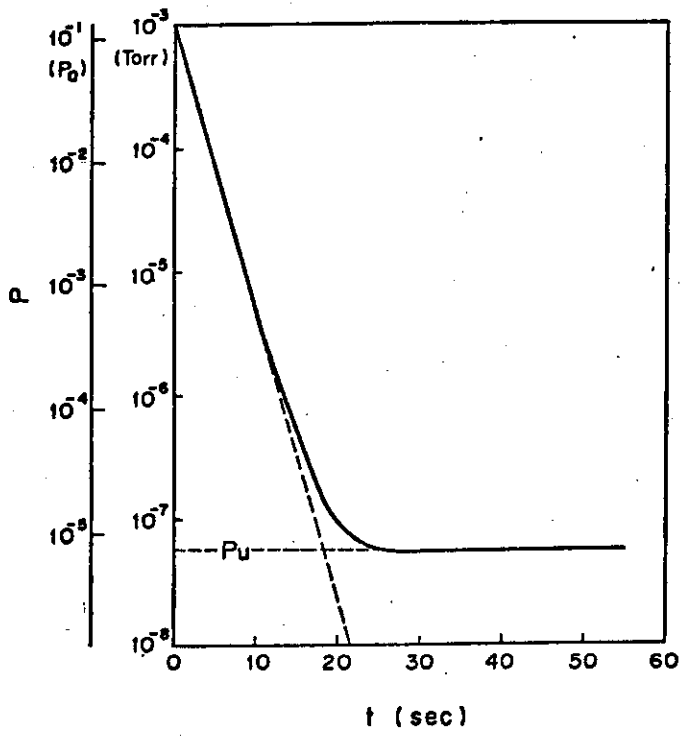

Fig. 5 A predicted pumpdown of volume gas in the vacuum vessel following a D-T plasma pulse.

a D-T plasma pulse. Here, we use the following parameters which have been discussed in previous sections.

- o average ion density (n) $\sim 1.2 \times 10^{20}$ m^{-3}
- o plasma volume (V_p) ~ 200 m^3
- o V $\sim 1{,}000$ m^3
- o p_u $\sim 1 \times 10^{-5}$ Pa
- o S ~ 500 m^3/sec for D-T and He
- o V/S = τ ~ 2 sec for D-T and He

Thus, the dwell time of 30 seconds is long enough to pump out the volume gases in the vacuum vessel.

In the actual machines, however, the transient behavior of the residual gases are not simply related to the pumping time constants, and provides evidence for the role of surface adsorption and diffusion on the evacuation time for the vacuum vessel.

5. Data Base for Helium Pumping

5.1 Physical considerations of helium exhaust from plasma

Two major concepts have been proposed for INTOR to exhaust helium from the plasma considering compromise with related engineering issues.

One of them is the concept of poloidal divertors with their coils located outside the toroidal field coils[5] (see Fig.6). In this simple

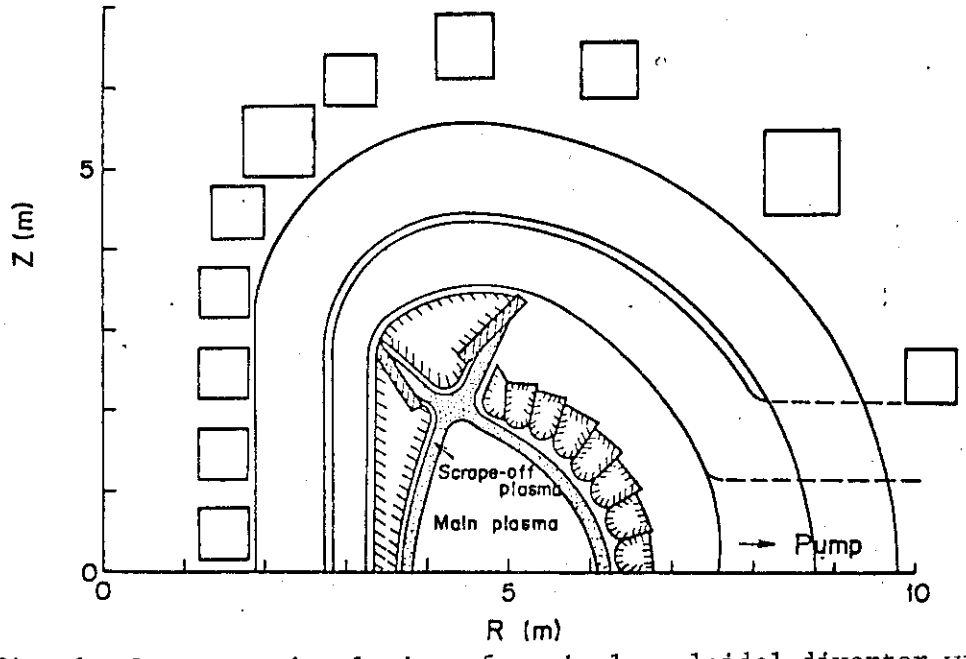

Fig. 6 Cross-sectional view of a simple poloidal divertor with its coils locating outside the toroidal field coils.

divertor, almost all of the reflected helium neutrals from the neutralizer plate are ionized and flow back to the plate by sheath potential, or freely into the pumping region. The thermal helium is easily ionized and hardly flow into the main plasma. Therefore, helium can be enriched in the pumping region and exhausted by pumps whose total effective pumping speed is lower than 500 m^3/sec. The required pumping speed can be mitigated if some slits are employed at the divertor throat to reduce the throat conductance.

The other is a concept called "Wall Lapping Plasma",[6] at which plasma is lapping the wall with rotating resonant helical islands by external helical coils of small currents and helium is exhausted through simple

valves. This has possibility to become a charming alternative because of its simple mechanical structure which can keep the larger available region for the blanket.

5.2 Selection of pumping means for helium

Helium is considered to be the most difficult gas to pump of all gases in fusion reactors because of its physicochemical nature. The difficulty in helium pumping by cryopumps arises from the small heat of condensation and adsorption of helium. Only cryosorption panels (using porous materials such as molecular sieves, charcoals and frozen gases) at 4.2 K are capable of pumping helium at a reasonable rate. The difficulty in pumping by getter pumps arises from the extremely small binding energy of helium with getter materials such as Zr-Al as compared with that of most reactive gases. In fact, getter pumps of any type do not pump helium. In turbomolecular pumps and mercury diffusion pumps, the helium pumping characteristics are associated with back diffusion of helium through high speed turbine blades and through jet streams because of the small mass of molecules. However, this is not a fatal problem for fusion applications.

There is probably no way to pump helium at a higher rate than the deuterium speed except a novel method using ion implantation. Helium pumping speeds of cryosorption pumps will be significantly lower than deuterium speeds particularly because the 4.2 K sorption panels must be surrounded by hydrogen pumping panels. Expected helium speeds per unit projected area of the intake port or the active surface for the above-mentioned pumps are summarized in Table 5.

5.2.1 Cryosorption pumps

Large cryocondensation pumps are currently used in space simulators and neutral beam injectors. However, cryosorption pumps for helium are

Table 5 Comparison of helium pumping speed per unit surface area for three candidate pumping means

Pumping means	Helium pumping speed per unit surface area ($m^3/sec \cdot m^2$)	Notes
Turbomolecular pump	30- 60	
Mercury diffusion pump	20- 30	with a baffle and a trap
Cryosorption pump	30- 40	with two chevrons

now under investigation and have not been put to practical use.

At the beginning of the D-T burn, the torus vacuum pumps will remove a mixture of deuterium and tritium, and during the burn and the dwell time, the pumps must remove a mixture containing as much as 10 % helium. Operating pressures of the mixture at the pumps are expected to be as high as 10^{-2} Pa. When molecular sieves are used as the adsorbent, it has been recognized that the pump may not be able to accommodate helium mixed with hydrogen isotopes, because condensed deuterium and tritium will block the adsorbent surface and prevent helium pumping.[15] This probably means that the cryosorption panels for helium will be surrounded by two chevrons, one at 77 K and the other at 4.2 K (in case of condensation type) or 20 K (in case of sorption type). In this case the expected maximum speeds per unit projected area are about 30 $m^3/sec \cdot m^2$ for helium and about 78 $m^3/sec \cdot m^2$ for deuterium (at 300 K) assuming the unity sticking (or condensation) coefficient and the chevron baffle transmission coefficient being 0.25. The ratio of helium speed to deuterium speed will be smaller than 0.4. Thus, the required surface area for the helium pumping speed of 170 m^3/sec for the single pumping unit (see Section 4.2) can be estimated to be larger than 5.7 m^2. From experiments using type 5A molecular sieve, it is also recognized that the pumping behavior for

pure helium is very complex.[16] Pumping speeds are a function of pressure (feed rate), total amount of the adsorbed helium, and the temperature history of the cryosorption panel prior to the run. This probably means that there is a maximum feed rate to the pumping surface that will permit constant pumping speeds. At higher feed rates, diffusion into the molecular sieve does not keep pace with the feed rates. It is suggested that these pumping instabilities may be avoided by using other adsorbents with larger pore sizes which allow higher diffusion rates.[17] In conclusion, the following recommendations for improving the performance of cryosorption pumps using porous solid materials are presented:

(1) The pump should be designed so that cryosorption panels are surrounded by hydrogen pumping panels and only helium is adsorbed on the sorption panels.

(2) Adsorbents with larger pore sizes should be investigated to determine whether higher pumping speed may be achieved in the helium pressure range of 10^{-3} Pa.

Frozen gas layers formed under suitable conditions are also used as adsorbents for helium. They have porous structures and good thermal contact with the cold substrate and are easily removed by warming up the surface. Sorption characteristics of the condensed layers are closely related to their structures and are influenced by the following factors;[18] (a) the temperature of cryosurface at which the condensed layer is formed, (b) the rate of formation of the layer, (c) the amount of gas condensed, and (d) the temperature history of the layer after condensation. The pumping characteristics of frozen CO_2 layers for helium have been investigated under various conditions.[19] The CO_2 layers condensed at temperatures below 8 K show a maximum sorption capacity but have an unstable structure which tends to change to a more stable structure of lower sorption capacity. During the structure change, abrupt temperature rises

of the condensed layer, thermal spikes, have been observed. These results seem to be important for practical applications of cryosorption pumps with frozen gas layers. The relative merits of porous solid materials and frozen gas layers occur in different aspects of pumping characteristics, and direct comparisons are difficult.

The principal advantages of cryopumps are the possibility of being constructed at relatively low cost and the flexibility of design for the cold surface to almost any size and shape desired. On the other hand, the shortcomings of the pumps are their sensitivity to heat loads and the necessity for regeneration which doubles the required pumping capacity. These problems will be discussed in other sections.

5.2.2 Turbomolecular pumps

Turbomolecular pumps are available from a number of manufacturers. The largest units known to be commercially available are of approximately 5 m^3/sec with a 0.47 m-diameter intake port,[20] but these are with conventional oil lubricated bearings. The pumps with gas and magnetic bearings are also available but in smaller sizes. Larger dry turbomolecular pumps which can be placed in any direction may be highly attractive for fusion applications. If such innovating pumps could be developed, the required helium pumping speed of 170 m^3/sec will be obtained in the occupied area of approximately 10 m^2.

The technical limitations of turbomolecular pumps using in magnetic fields and at elevated temperatures come from the high rotating speeds and the small clearances between rotor and stator blades. Eddy currents and thus heat will be generated in the rotor. Thermal expansion of the blades will limit the maximum bake-out temperature.

5.2.3 Diffusion pumps

Mercury diffusion pumps were used at Princeton and Kurchatov Institure, but have not been used for fusion applications in Japan. The major problems arise from the use of the high vapor pressure liquid and the lack of availability of large units. Diffusion pumps have by nature a relatively high pumping speed per unit port area. But, the effective speed of the pump system will be lowered to about 1/5 of the nominal speed by cold traps above the pump.

6. Some Considerations on Cryopumping

6.1 Regeneration of the pumps

Regeneration time of the cryopumps must be short enough not to affect the duty cycle or the availability of the INTOR experiments. The regeneration cycle must be determined by considering the following items; (1) the allowed thickness of condensed D-T films at which the condensation coefficient of D-T mixtures decreases to, for example, 90 % of the initial value, (2) the allowed adsorbent thickness and helium concentration in the adsorbent at which the sticking coefficient of helium decreases to, for example, 80 % of the initial value, (3) the safety problems including the possibility of hazardous explosion on the occasion of air leakage, and (4) the allowed amount of tritium on cryopanels which is of importance from the points of view of (a) tritium inventory and (b) heat loading to cryopanels due to tritium decay.

In case of cryocondensation pumping, it has been reported that only a 10 - 15 % drop in pumping speed occurs when the amount of condensed deuterium varies from 0 to 5 x 10^4 Pa·m^3/m^2.[21] High pumping speeds are maintained with deuterium feed rates up to 2 Pa·m^3/sec·m^2. This implies high thermal conductivity of the frozen gas.

In case of cryosorption pumping, the maximum helium concentration in porous adsorbents at 4.2 K may be in the order of 10 atomic per cents of the bed material, and the amount of adsorbed helium on a molecular sieve 5A cryopanel is reported to be more than 3 x 10^4 Pa·m^3/m^2.[22] On the other hand, it has been shown that the measured pumping speed for helium is most influenced by pressure (feed rate), total amount of the adsorbed helium and the temperature history of the panel.[16] Although we cannot determine in a practical sense the allowed helium concentration in an adsorbent at the present time, we may expect the allowed amount of adsorbed helium of more than 1 x 10^4 Pa·m^3/m^2 on some improved materials.

If an air leak were to occur in a cryopump, the resultant oxygen-hydrogen mixture would pose an explosive hazard. From this point of view, the longest operation time should be determined by the time required to accumulate sufficient hydrogen isotopes such that if the leak were to occur in a closed volume, the partial pressure of the evaporated hydrogen would exceed 1.7×10^3 Pa.

On the other hand, it is most desirable to minimize the tritium inventory on the cryopanels. The heat load to the cryopanel due to tritium decay is estimated to be 2.9 W/10^3 Pa·m^3. This would force the shortest possible time for the regeneration cycle.

The allowed amounts of pumped gases on cryopanels derived from the different limitations are summarized in Table 6. It is apparent that the allowed amount of hydrogen isotopes should be less than 2.8×10^3 Pa·m^3/m^2.

Table 6 Allowed amounts of gases on cryopanels derived from different limitations

Item No.	Allowed amounts of gases on cryopanels (Pa·m^3/m^2)		Notes
	helium	hydrogen isotopes*	
(1)		$< 5 \times 10^4$	from Reference (21)
(2)	$< 1 \times 10^4$		expected value
(3)		$< 2.8 \times 10^3$	panel area = 6 m^2 and closed volume = 10 m^3 for a single pumping unit
(4)-(a)		as little as possible	
(4)-(b)		$< 3.5 \times 10^3$	heat load < 5 W/m^2

(* D:T = 1:1)

6.2 Preliminary design study on cryopump systems

A preliminary design of the torus vacuum system for INTOR has been made under following conditions:

o burn time	100 sec
o dwell time	30 sec
o duty	0.77
o He production rate (during burn)	1.6×10^{20} sec^{-1} = 0.6 $Pa \cdot m^3/sec$ at 300 K
o pumping rate of hydrogen isotopes (during burn)	1.6×10^{21} sec^{-1} = 3.0 $Pa \cdot m^3/sec$ at 300 K
o average ion density	1.2×10^{20} m^{-3} → 45 $Pa \cdot m^3$ per one dwell time
o plasma volume	200 m^3 → 45 $Pa \cdot m^3$ per one dwell time

The torus pumping system is composed of 12 cryopumping units, 4 roughing units, and 4 regeneration and fuel recovery units. Six cryopumping units are alternately operated at an interval of 6 hours. The cryosorption panels for helium pumping are surrounded by two chevrons, one at 77 K and the other at 4.2 K. Hydrogen isotopes are accumulated mainly on the 4.2 K chevron. Table 7 summarizes specifications of the torus pumping system.

Table 7 Specifications of a torus cryopumping system

Duct conductance (total)	
He & $D_2(T_2)$	$\sim$1,000 m^3/sec
Pumping speed per unit projected area	
He	30 $m^3/sec \cdot m^2$
$D_2(T_2)$	78 $m^3/sec \cdot m^2$
Projected area of cryopanels (total in operation mode)	$\sim$36 m^2
Number of pumping units	
in operation mode	6
in regeneration mode	6
Pumping speed at 77 K chevrons (total)	
He	$\sim$1,080 m^3/sec
$D_2(T_2)$	$\sim$2,800 m^3/sec
Effective pumping speed (total)	
He	$\sim$500 m^3/sec
$D_2(T_2)$	$\sim$700 m^3/sec
Regeneration cycle	12 hr
Operating time	6 hr
Regeneration time	< 6 hr
Pumped amount after 6-hour operation (total)	
He	$\sim 1.0 \times 10^4$ Pa·m^3 ($\sim 2.8 \times 10^2$ Pa·m^3/m^2)
hydrogen isotopes	$\sim 5.8 \times 10^4$ Pa·m^3 ($\sim 1.6 \times 10^3$ Pa·m^3/m^2)
Tritium inventory in cryopumping system	$\sim 5.8 \times 10^4$ Pa·m^3 ($\sim$0.15 kg)

7. Radiation Effects on Vacuum Systems

7.1 Cryopanels

7.1.1 Nuclear heating

The major radiation effect on the liquid helium cooled cryopanels would be an additional heat deposition in the cryopanels due to neutrons and gamma-rays (nuclear heating). If the cryopanel is assumed to be operating at 4.2 K and the heat capacity of copper at that temperature is 0.1J/kg·K, the nuclear heating should be suppressed to less than 1×10^3 W/m^3 to keep the small temperature rise of 1 K/sec. This corresponds to a heat deposition of 10 W/m^2 for a 10^{-2} m thick cryopanel. This gives a significant load to the refrigerator since the total heat load due to support conduction, infrared ray radiation, eddy current and the enthalpy change of the pumped gas is several watts per square meter.

In the preliminary design of JXFR,[23] the attenuation of neutron and gamma-ray fluxes has been calculated as a function of the distance from the plasma axis (see Fig.7). The amounts of the nuclear heat deposition in the

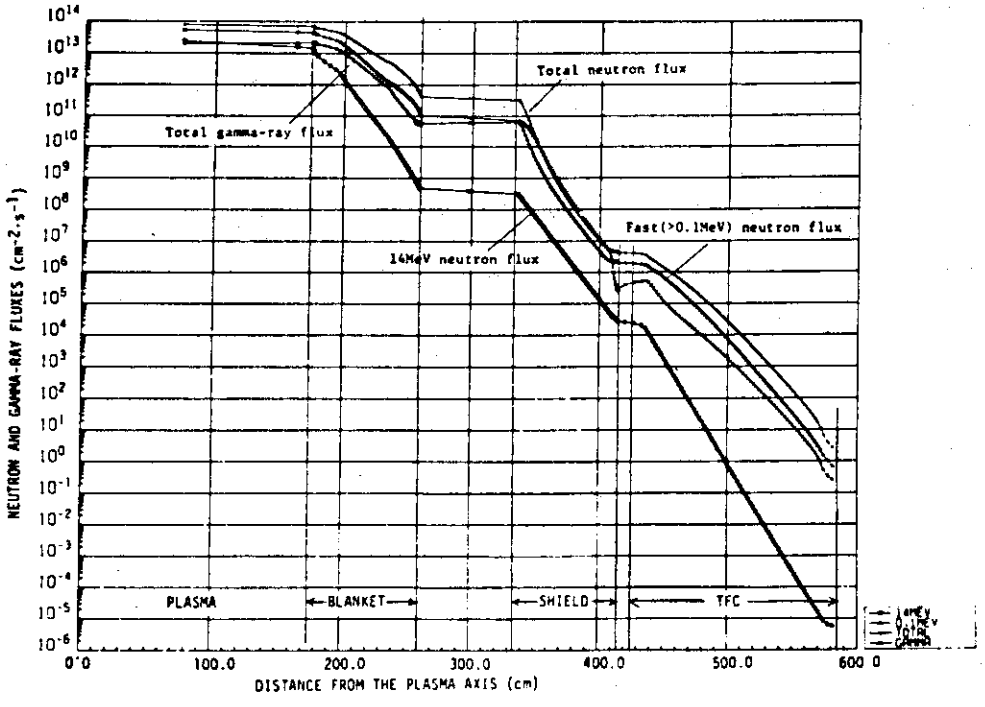

Fig. 7 Attenuation of neutron and gamma-ray fluxes in the outer blanket, shield and toroidal field coil of JXFR.

cryopanels have also been evaluated using the Monte Carlo method.[24] In this case the heat load due to nuclear heating (mostly due to gamma-rays) in the cryopanels for the torus pumping system is about 1×10^4 W/m^3. Therefore, a little bit more shielding or better selection of panel material will be needed to suppress the temperature rise. To reduce the heat deposition, cryopanels must be placed in well-shielded locations without sacrifice of conductance to reach the panel. The nuclear heat deposition in the NBI cryopumps is calculated to amount to less than 1/10 of that in the torus cryopumps.[24]

7.1.2 Cyclotron radiation

As plasma temperature is in the range of 1 keV, it is necessary to take cyclotron radiation into account. The heat deposition due to the radiation increases in proportion to $\beta^2 B^{5/2}$. The wave length of the cyclotron radiation will be in the submillimeter to millimeter range. Because the surface reflectivity of metal walls to such electro-magnetic waves is expected to be very high, a large heat deposition may occur on the cryopanels unless adequate wave absorbers are provided inside the pumping ports.

7.1.3 Desorption by radiation

It may be possible for radiation from a D-T plasma to interact with the condensed or adsorbed gases on cryopanels, leading to desorption (radiation-induced desorption).[25] Further investigation is required on the mechanism of radiation-induced desorption.

7.1.4 Radiation effect on liquid nitrogen

In the case where liquid nitrogen is used as a coolant, some of the oxygen which condensed in the liquid nitrogen is converted to ozone due to ionization by radiation.[26] This ozone may explode into a hazardous condition,

as known from the field of fission reactor and accelerator experiments. Therefore, the purity of liquid nitrogen is required to be as high as possible.

7.2 Seals and valves

Mechanical seals and valves in high radiation environment will be made of metals. The problems associated with swelling, reduced ductility, and lubrication of the mechanism may be of concern, but there is considerable fission reactor experience in this field. Elastomer seals should be eliminated because tritium tends to exchange with hydrogen in organic materials causing contamination and deterioration of the materials.

7.3 Other materials

In order to secure good accessibility for maintenance, it may be desirable to fabricate vacuum components of low activation metals. Recently, aluminum alloy uhv components including metal seals have been developed in National Laboratory for High Energy Physics.[27] Another concern would be for electrical insulating ceramics. Large sized metal-ceramic seals will be needed in the vacuum system.

8. Remote Leak Detection

In general, the leak detection system in fusion reactors will be designed on the assumption that all vacuum vessel joints including the major structural weld joints of the vessel must be capable of being leak checked. Leak location, however, may be limited to those joints which can be replaced or repaired without segment removal. These include all vacuum seals on the diagnostic, pumping and neutral beam ports and the segment parting joints.

Locating a leak to a specific seal will be accomplished by providing for each joint a gas tube which will permit the introduction of a tracer gas into the joint. This approach is typical of all seals, whether bolted or welded. All other welded joints on the vacuum vessel will be surrounded by a secondary structure which provides gas tight enclosures for the introduction of the tracer gas.

When leaks are to be checked, a tracer gas is sequentially fed into every leak detection volume and a specific mass analyzer or a leak detector is utilized to determine the presence of the tracer gas. It is an important problem whether it would be possible to use helium as the tracer gas for fusion reactors in which helium will be produced by the fusion reaction and will be embedded in the vacuum walls. Even if helium is used as the tracer gas, a high resolution gas analyzer which is capable of separating helium from deuterium must be utilized since deuterium will be fed to the vacuum vessel as a fuel gas.

Leak location would also be done by detecting a local pressure rise within the vacuum vessel due to gas influx from a leak. In this case, a small pressure sensor must be moved inside the vacuum vessel by a skilful manipulator. The movable mechanism must be designed and constructed such that it can perform in the high vacuum and high radiation environment without contaminating the interior of the vacuum vessel.

In JAERI, a development program of a leak sensor with a bundle of capillaries at the entrance of an ionization gauge is now under way. The device is intended to detect only derectional molecular beams from leak points. This is based on the analysis about a derectional detector of molecular density.[28] Other R&D items for remote leak detection are (1) high resolution mass analyzers, (2) remotely controlled movable mechanisms for leak sensors in three dimensional vacuum space, and (3) radiation effects on the leak sensors and their related systems.

9. R & D Items

The vacuum development program for INTOR will include design, fabrication, testing, and evaluation of vacuum components and systems as necessary in order to meet the objectives. R & D items for INTOR vacuum technology would be as follows:

(1) Advanced methods of helium pumping

- (1-1) improvement of cryosorption panels
- (1-2) testing of radiation effects on cryopanels
- (1-3) design and fabrication of large cryosorption pump systems and confirmation of their performance
- (1-4) development of large dry turbomolecular pumps
- (1-5) improvement of mercury diffusion pump systems
- (1-6) evaluation and testing of novel pumping method by ion implantation

(2) Materials and components

- (2-1) development and testing of components of low activation metals
- (2-2) development and testing of large metal-ceramic seals and gate valves
- (2-3) evaluation and testing of low-Z materials and low-Z surface coatings

(3) Surface conditioning technique

- (3-1) optimization of baking and discharge cleaning conditions
- (3-2) testing of radiation-induced outgassing
- (3-3) development of in-situ surface coating technique

(4) Remote leak detection

- (4-1) development of high resolution mass analyzer
- (4-2) development of remotely controlled movable mechanisms
- (3-3) testing of radiation effects on leak sensors

(5) Diagnosis and control of large vacuum systems

- (5-1) measurement of dynamic charactor of vacuum systems
- (5-2) system control with computers

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