R&D Activities of Tritium and Breeder Technology at KHI

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Kawasaki Heavy Industries, Ltd. (KHI) has developed tritium breeding blanket and tritium handling systems. In current design studies of tritium breeding blanket, small spherical solid breeding material has been desired. A mass-production process of small spherical breeder was developed by using a rotating-granulation/sintering method. The fracture toughness of fabricated solid breeder of 1mm ϕ was measured.

Several tritium implantation apparatus has been developed and constructed to investigate the implanted tritium behavior on first wall and divertor plate. The characteristics of these apparatus are also examined.

1. Introduction

Kawasaki Heavy Industries, Ltd. (KHI) has performed development of fusion nuclear technologies in field of tritium breeding blanket and tritium handling systems, and to apply them to fusion projects such as ITER/JAERI and national laboratories.

In tritium breeding blanket, solid breeder has been proposed as a tritium breeding material. One of the concerns for solid breeder is the thermal stress cracking under the fusion blanket condition. Current design studies of breeding blanket[1], therefore, offer a small spherical solid breeder to reduce the induced thermal stress in the breeder. In order to supply several kinds of spherical breeder for in-pile and out-of-pile tritium release studies, a mass-production process for making about 1mm diameter of lithium ceramic was developed, using a rotating-granulation/sintering method.

In tritium handling systems, tritium interaction with first wall and divertor plates subjected to low energetic tritium bombardment is identified as a key issue. There are deep uncertainties in process that govern tritium trap and release. KHI constructed the tritium permeation test apparatus for JAERI to investigate the permeation behavior in candidate material[2]. Recently, a new implantation apparatus with high flux and low energy triton beam was constructed for Toyama University to examine the retention/recycle characteristics. This apparatus was also designed to handle pure tritium installed in glovebox and to analyze the sample surface with two kinds of optical analysis method.

In this paper, some characteristics of spherical breeder using rotating-granulation method will be discussed. Furthermore, the outlines of new tritium plantation apparatus will be introduced including the preliminary experimental data.

2. Development of Tritium Breeding Material

Solid breeder such as Li_2O, $LiAlO_2$, or Li_4SiO_4 has been proposed as a tritium breeding material in a fusion reactor blanket. One of the great concerns for solid breeder, is the thermal stress cracking under the fusion blanket condition. In order to reduce the induced thermal stress in breeder, small spherical form is needed.

2.1 Fabrication of small spherical solid breeder

Development of small spherical breeding material has been carried out by adopting rotating-granulation/sintering method[3]. Typical fabrication procedure for 1mm ϕ solid breeder is described in Fig.1. This procedure includes nucleus production process to prepare nucleus, granulation process to grow the nucleus to pebbles of desired diameter, and sintering process to increase density and mechanical strength of pebbles. Small spherical pebbles of 1mm ϕ Li_2O fabricated by this method are shown in Photo 1. Photo 2 shows the SEM micrograph of broken section of Li_2O pebble, and this photo shows grain size to be 40 μm in average.

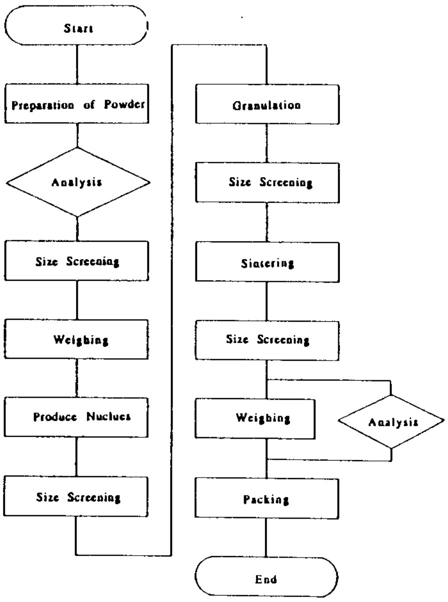

Fig. 1 Pebble fabrication procedure

2.2 Mechanical Strength

The fragmentation of breeding material results in the blockage of sweep gas flow path and production of local hot spot. Especially, local hot spot leads the breeder temperature to unacceptable level. In order to evaluate the integrity of pebbles

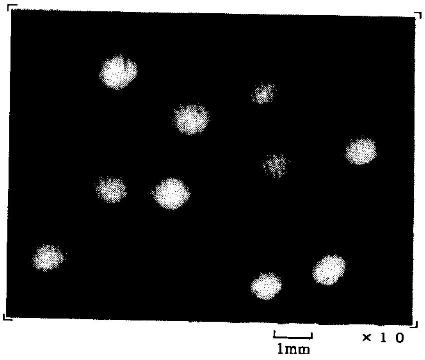

Photo 1 Micrograph of 1mm broken section for Li_2O pebbles

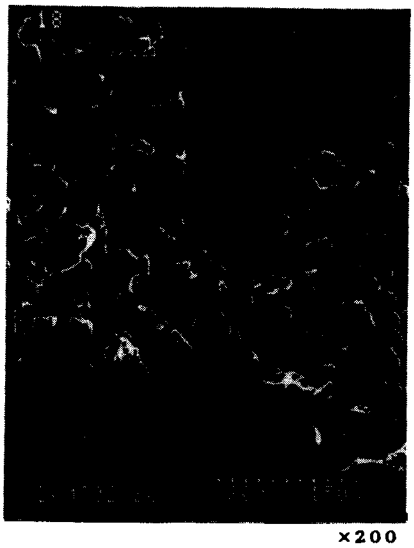

Photo 2 SEM micrograph of broken section for Li_2O pebble

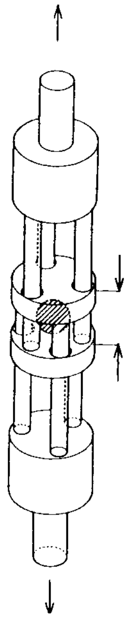

Fig. 2 Test apparatus for compression toughness test

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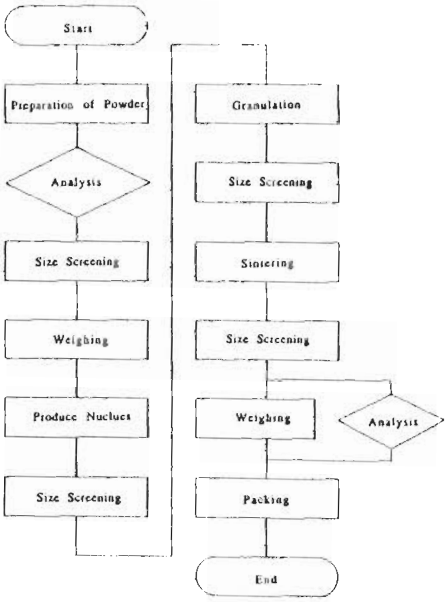

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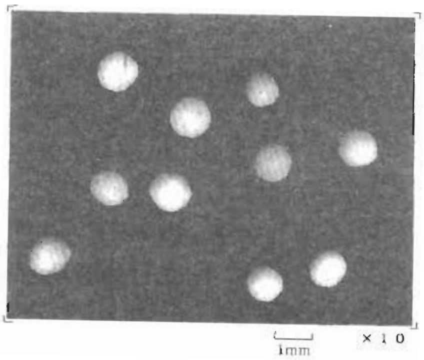
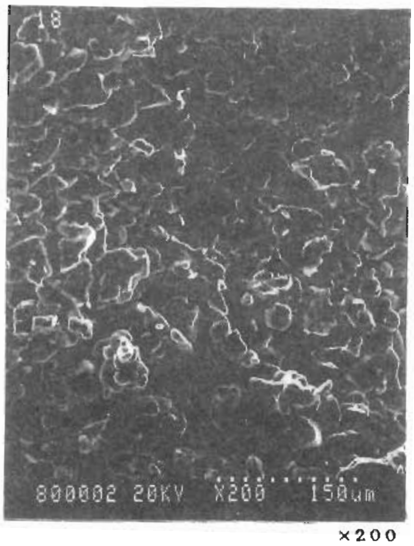


Photo 1 Micrograph of 1mm broken section for Li_2O pebbles



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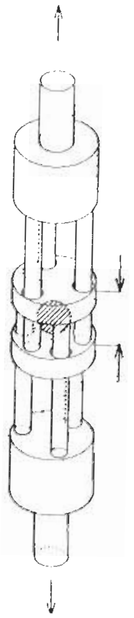


Fig. 2 Test apparatus for compression toughness test

under blanket condition, mechanical strength of the pebbles were measured by compression toughness test. In this test, test apparatus shown in Fig.2 is used to transfer tensile force to compression one. Compressive stress was measured for the fabricated pebbles of 1mm diameter and is show in Fig.3. The fracture toughness decreases with increasing temperature. The Li_2O pebble reveals the highest compressive stress in our fabricated solid breeders.

For the representative breeder such as Li_2O, the calculated relation between maximum induced stress and pebble diameter is shown in Fig.4 as a function of volumetric heating rate. This figure indicates that thermal stress of 0.9kg/cm^2 is generated inside the Li_2O pebble under the volumetric heating rate of 10W/cm^3 (ITER condition). Comparing this estimated stress and observed compressive stress of 80-100kg/cm^2 at 800℃, fabricated Li_2O pebble has sufficient mechanical strength under the breeding blanket condition.

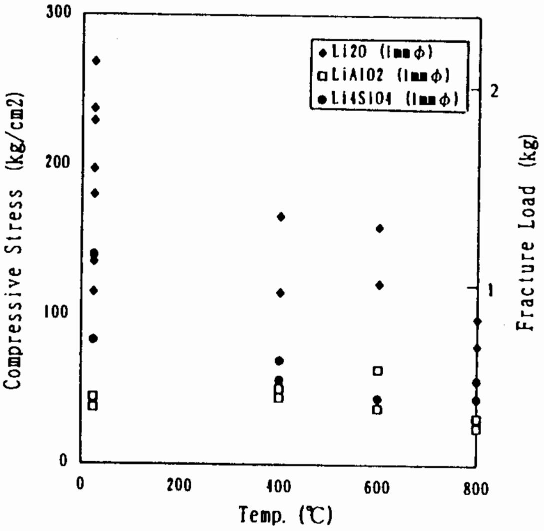


Fig. 3 Compressive strength of sintered pebbles

3. Development Tritium Implantation Apparatus

3.1 JAERI Tritium Permeation Apparatus

For D-T fusion reactor, a key safety issue is the release of tritium implanted into first wall and divertor plates. To understand the permeation behavior of the implanted tritium, a tritium permeation apparatus[2] had been designed and constructed for the Tritium Process Laboratory in JAERI. This apparatus has been designed for tritium implantation driven permeation experiment as shown in Fig.5. The ion source is a modification of the quartz capillary duoPIGatron proton source. A dense narrow plasma is generated by the duoplasamtron and PIG discharge mechanism in a quartz tube to which a strong magnetic field is applied along the tube axis. Very high density of ion current is obtainable with a high proton percentage as high as 90%. Ion beam energy is varied from about 20eV to 1keV continuously. Ion flux larger than 0.2mA/cm^2 for 200eV D^+ and 2mA/cm^2 for 1keV D^+ is expected at the target position.

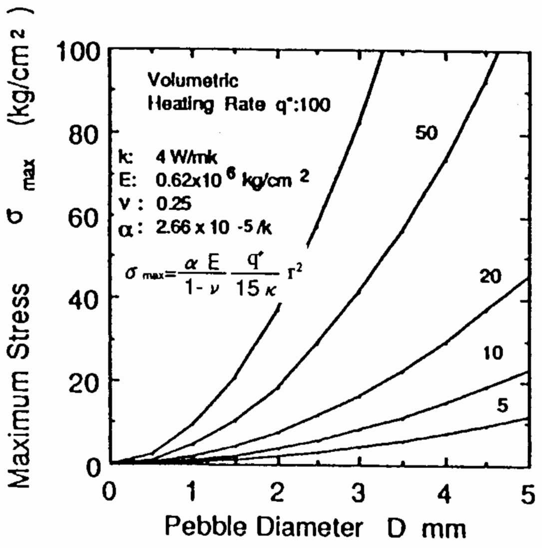


Fig. 4 Calculated thermal stress in Li_2O pebbles

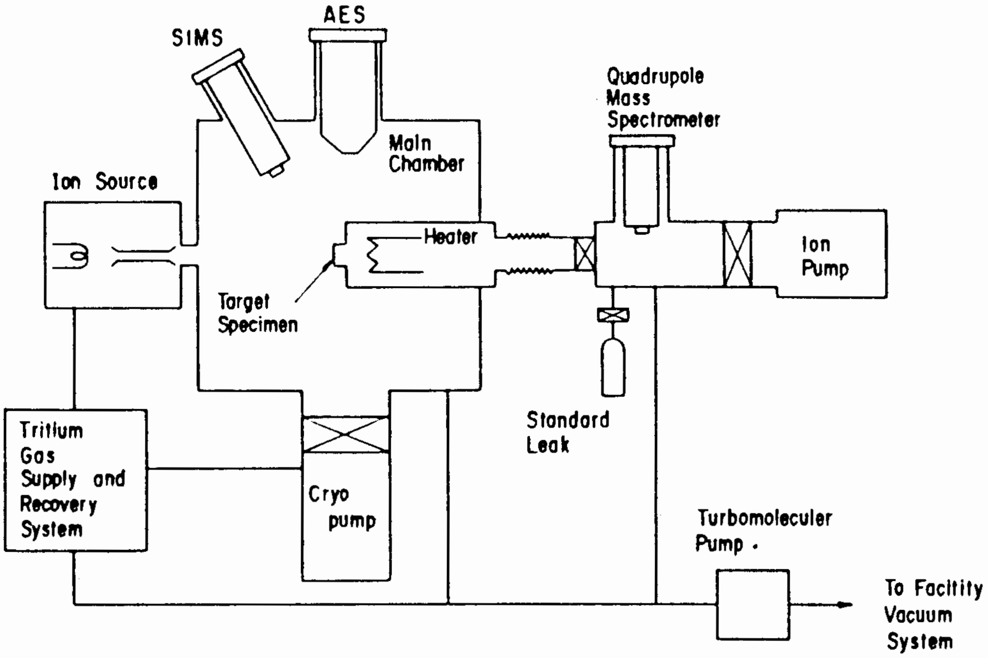


Fig. 5 Schematic illustration of JAERI tritium permeation apparatus

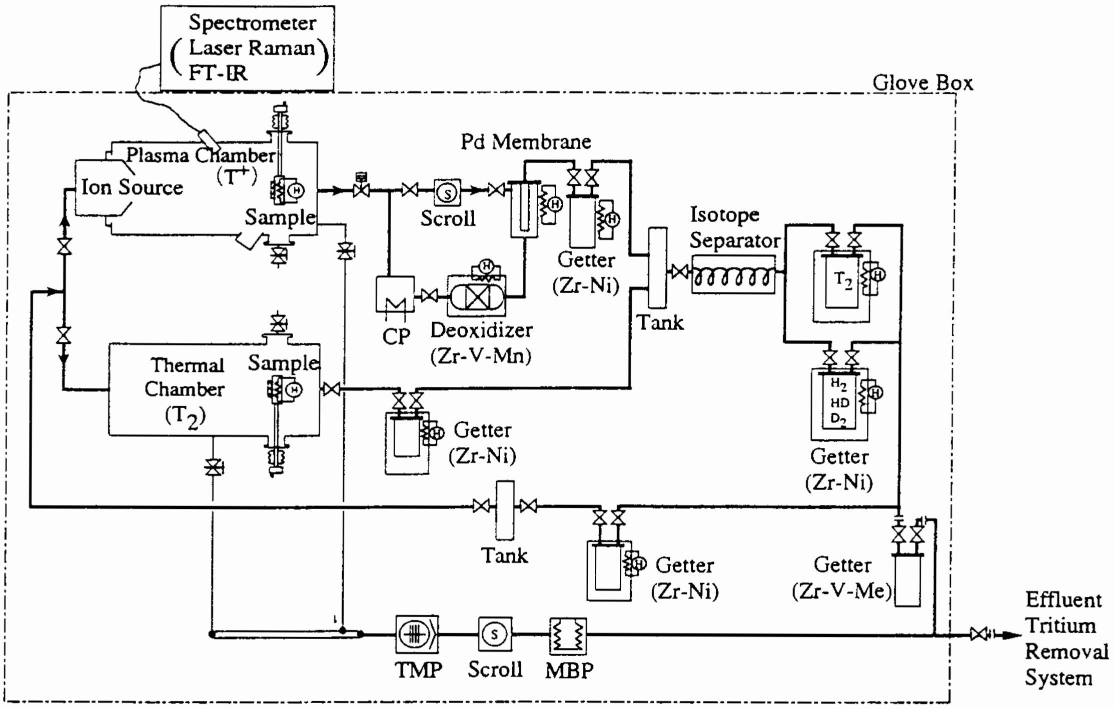


Figure 6 Schematic illustration of Toyama tritium implantation

3.2 TOYAMA Tritium Implantation Apparatus

Based on the above implantation apparatus, recently, a new implantation apparatus has been constructed for Hydrogen Isotope Research Center (HRC) in Toyama University to investigate the retention/recycle characteristics. Major characteristics of this apparatus are introduced below.

a) System Description

Figure 6 shows a process diagram of this apparatus. Tritium inventory limit at HRC is 100Ci. This apparatus, therefore, has been designed to circulate the small quantity of tritium through the system in an expeditious manner, and to investigate the plasma implantation and thermal gas absorption behavior. The apparatus consists of plasma chamber, thermal chamber, recovery unit, purification unit, isotope separation unit, storage-supply unit, and evacuation unit.

Plasma chamber is a 304SS cylindrical vacuum vessel which is 300mm long and 200mm in diameter. The ion source in plasma chamber is the ECR plasma source with a 250W microwave power generator and a 1kW magnet power supply. Very high density of ion current is obtainable at target sample. As shown in Fig.7, the Laser-Raman and FT-IR (Fourier transform infrared absorption) spectroscopy are installed to investigate the chemical state of tritium remained in the target material.

The thermal chamber whose volume is limited to approximately $50cm^3$ has been designed to examine the absorption/adsorption behavior of pure tritium gas on the sample material. The chemical species in chamber are analyzed by quadrupole mass spectrometer. The chamber pressure is also measured by using a capacitance manometer.

Recovery unit is designed to recover the residual tritium in both chambers. Tritium in plasma and thermal chambers is recovered effectively by using a cryopump (750l/second) and Zi-Ni getter, respectively.

Storage-supply unit consists Zr-Ni metal getters for tritium supply and storage, tank for measurement, and mass flow controller. Tritium is released from the metal getter by heating up to 450℃, and stored in the tank. After that, tritium gas in the tank is fed to the ion source in plasma chamber or to the thermal chamber at an adequate flow rate through a mass flow controller.

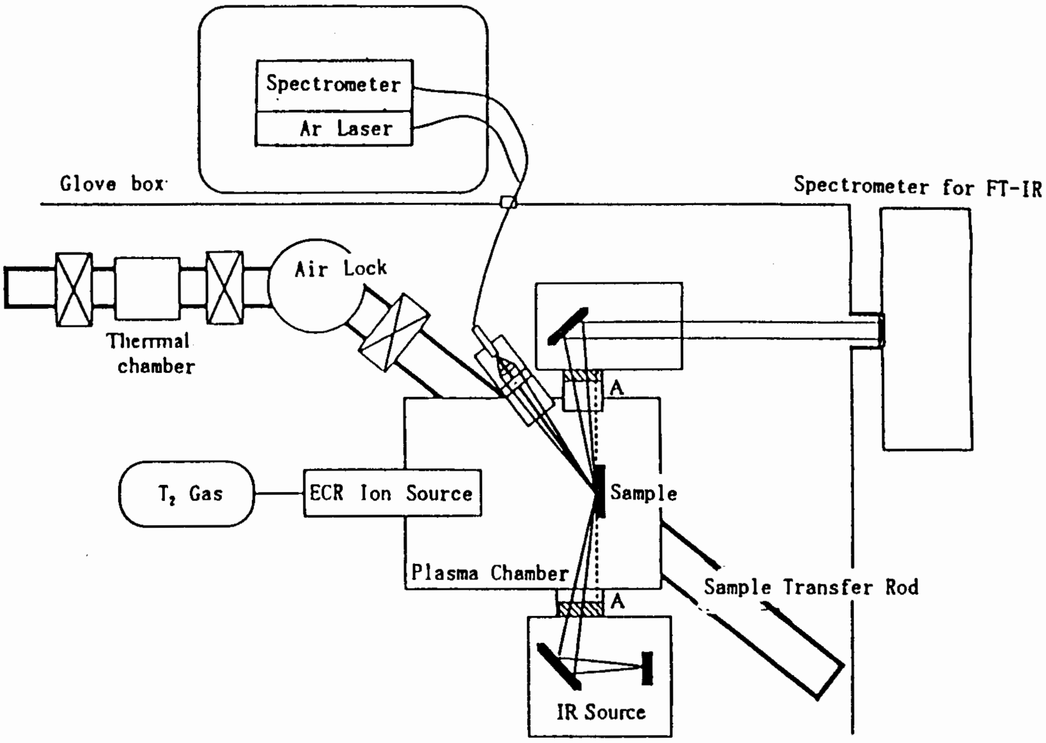


Fig.7 Scheme of the Laser-Raman and FT-IR spectrometer arrangement

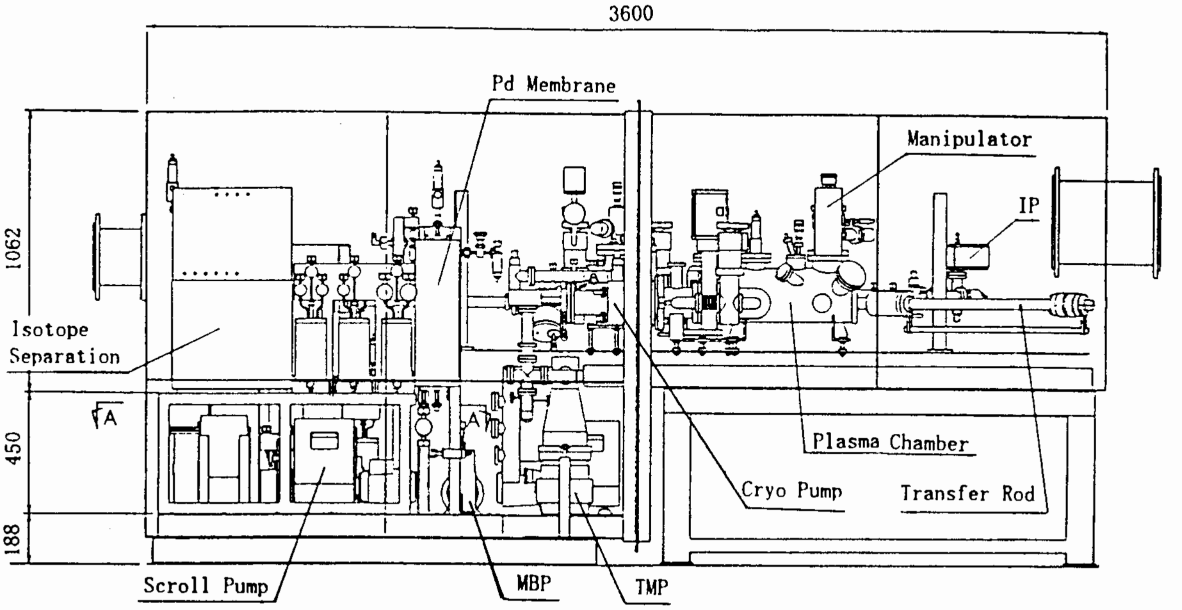


Fig. 8 Layout of tritium implantation apparatus

Purification unit is removes the impurities such as C, O, and N in hydrogen isotope stream. Deoxidizer packed Zr-V-Mn is used to crack the tritium oxide and hydrocabon, and palladium diffuser is also used to extract hydrogen isotope only. Approximately 1mol/h of hydrogen isotopes are purified by this diffuser.

For the isotope separation unit, gas chromatography method is adopted because of small amounts of hydrogen stream in one batch operation. In order to enhance the separation performance, separation column is designed to be kept at liquid nitrogen temperature.

Evacuation unit which consists a 500l/second turbo molecular pump backed by a 200l/min scroll pump and 5Nl/min metal bellows pump, is used for the first stage evacuation of all the systems.

All most of experimental hardware are placed in a glovebox (3.6m × 1.0m × 1.7m) as shown in Fig.8. Laser unit, surface spectrometer, and control rack for system are placed outside the glovebox.

b) Operational Test

Prior to tritium implantation experiments, some examination tests using D_2/H_2 gas were carried out.

Characteristic of ion source

The plasma is generated by heating the gas by the microwave. The ions in the plasma can be accelerated by the bias voltage between the ion source and the target. The ion beam current increased generally with increasing a bias voltage. As a function of microwave power, Fig.9 shows the ion beam current (600eV) at the target at a distance of 150mm from the ion beam exit hole. At the microwave power above 130W, the ion beam current did not change with increasing the microwave power. However, below 130W, it decreased remarkably with decreasing microwave power.

Therefore, it was found that ion current density into target did not depend on the microwave power, but was strongly dependent on the bias voltage.

Characteristic of Isotope separation unit

A portion of the D_2-H_2 mixture gas was taken into a sampler (1.0Ncc). The sample gas was carried by Ne flow

(5.5cc/min.) into a gas chromatography assembled as shown in Fig.10. The separation column packed with alumina coated with $MnCl_2$ is cooled to liquid nitrogen temperature. A typical chromatogram is illustrated in Fig.11. It shows a very sharp peak for H_2 followed by HD and D_2 peaks.

It was confirmed that gas chromatography column at liquid nitrogen temperature is applicable for the isotope separation of a small amount of hydrogen isotopes.

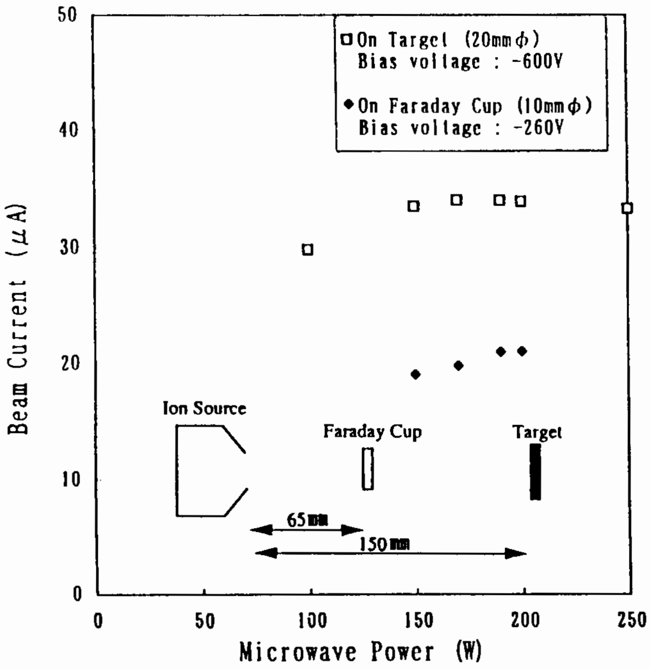


Fig. 9 Ion beam current

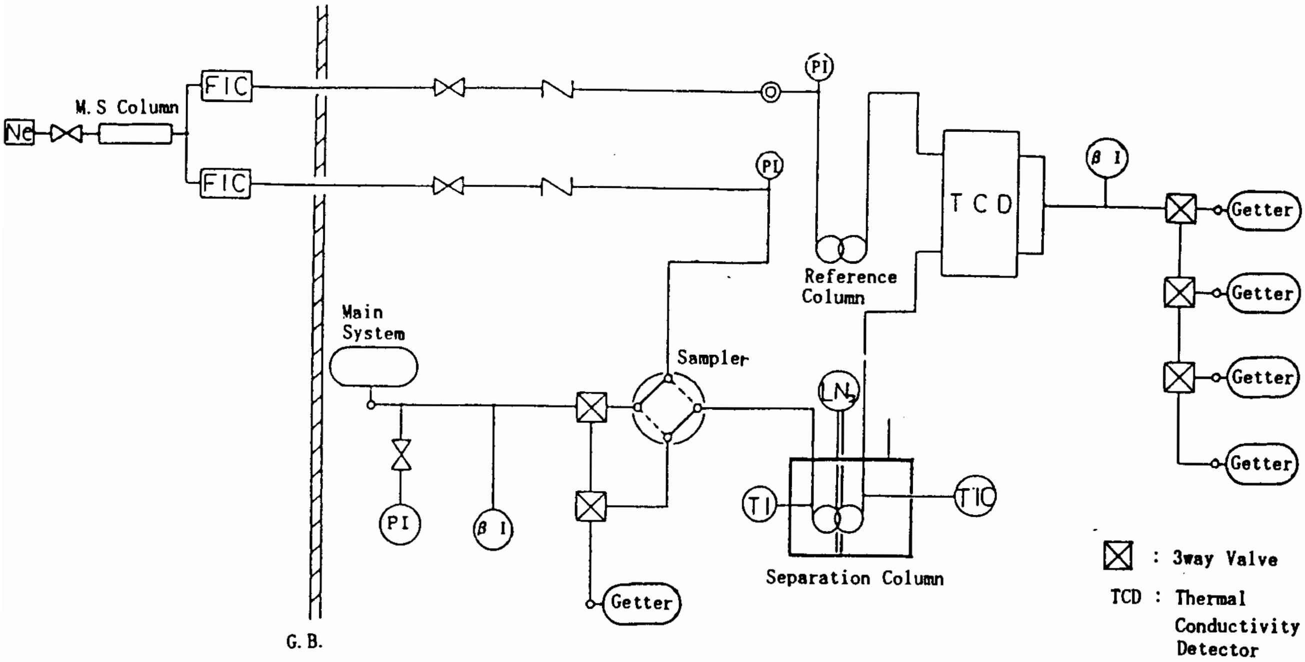
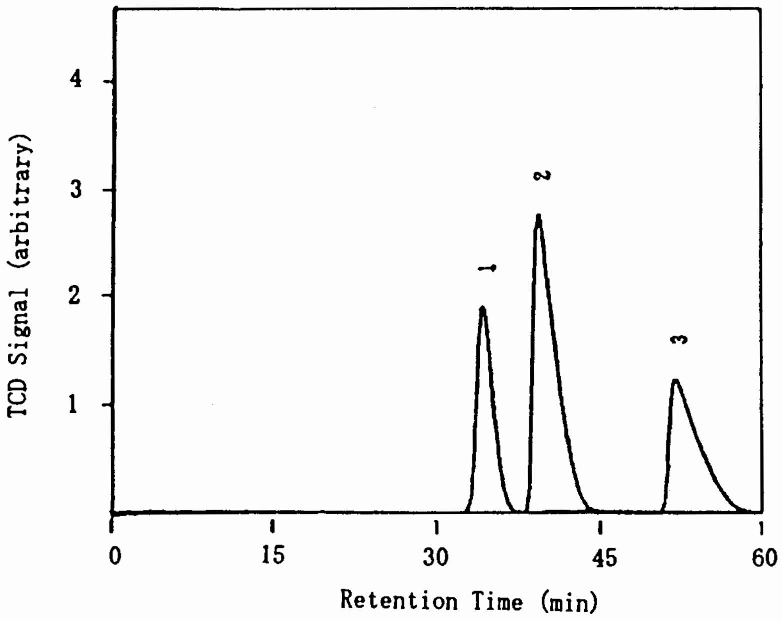


Fig. 10 Schematic flow-diagram of isotope separation



Carrier Gas : 5.5Ncc/min(Ne)
Column : 5mm^{ID}×2.8m^{L}
Hydroisopack (Al_2O_3+$MgCl_2$)
Liq. N_2 Temp.

Sample : H_2/HD/D_2=1:2:1
1Ncc(charge)

Fig.11 A typical chromatogram (peak1 : H_2 peak2 : HD peak3 : D_2)

4. Conclusions

Fabrication technologies of small spherical solid breeder have been developed to supply a large quantity of breeder for feasible blanket. Tritium implantation apparatus has been also developed and constructed to apply for national fusion project. Conclusions of these developments are summarized as follows.

1) The rotating-granulation/sintering method proves applicable to fabricate the spherical solid breeder with high productivity and high sphericity.
2) The mechanical strength of fabricated soiled breeder seems to be sufficient for use in the tritium breeding blanket.
3) Tritium implantation apparatus with high flux and low energy ion beam was developed and constructed successfully to examine the retention/recycle behavior on first wall and divertor.
4) Preliminary operational test prior to the pure tritium implantation were performed. A current density of 34μA/target (6.8×10^{13} D^+/s・cm^2) at an ion energy of 600eV was achieved.

References

[1] IAEA, "ITER Blanket, Shield and Material data Base", ITER Documentation Series, No.29, IAEA, Vienna (1991)
[2] K. Okuno, S. O'hira, H. Yoshida, Y. Naruse, T. Suzuki, S. Hirata, and M. Misumi, Fusion technology, Vol14, No.2, P713 (1988)
[3] T. Suzuki, O. Murata, and S. Hirata, Ceramic Transaction, Vol.27: Fabrication and Properties of Lithium Ceramics III, P37, The American ceramic Society, Inc. (1992)