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IMPLICATIONS OF TRITIUM IN NEUTRAL BEAM INJECTORS*

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ABSTRACT

Neutral injectors for heating plasmas of D-T burning fusion reactors are subject to tritium contamination. This paper discusses relevant questions and problem areas pertinent to tritium environment, including calculations of tritium contaminations in different neutral injectors, gas handling and pumping systems, and implications on beam line components.

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Introduction

Plasmas of D-T fusion reactors are expected to be heated to ignition temperatures by deuterium neutral injection or by tritium neutral injection although less likely. Since D/T gas is the fusion fuel, the tritium breeding and handling system is one of major components of a fusion reactor. Tritium contamination in neutral beam injection (NBI) systems results from particle backstreaming from a reactor vessel and, to a much lesser degree, from D(d,P)T reactions from D-beam-bombarded surfaces. In the case of tritium injection, of course, the entire NBI system is filled with a large inventory of tritium gas. Therefore, gas pumping and recycling is of prime concern in NBI system design as well as in reactor fuel cycle system (FCS) design.

The Tokamak Fusion Test Reactor (TFTR) under construction at Princeton employs a central system to handle gases from both the reactor vessel and the NBI systems.[1] Design studies on the Experimental Test Facility (ETF)[2] and other prototype reactors address these problems extensively. In comparison with the reactor design problems associated with tritium handling, the tritium implications on NBI systems are not extensive. A NBI system compatible with a prototype reactor environment, however, is still a large extrapolation from the present NBI technology.

In this paper, we examine tritium-related problem areas in NBI systems, including test-stand beam lines, tritium injector systems, and deuterium injector systems. Gas pumping, controlling, and recycling scenarios are discussed along with beam line component technology that would be affected by tritium contamination. For numerical calculations, design parameters of a prototype reactor relevant to our discussions are established, based on a tokamak geometry.

Reference Design

In order to make our discussion more quantitative, we have adopted a set of reference beam system parameters and some relevant reactor parameters. These are based on our understanding, at the time of writing (April, 1980), of the ETF baseline designs. The parameters were augmented with other considerations when found necessary for quantification. The NBI system parameters and some relevant reactor parameters are as follows:

NBI System

Injected D° Power (P_{inj}) --------65 MW total

Energy of D° (E_b) ----------150 keV

Pulse length (t_p) ----------6 s

Power Efficiency

η_c(a) ----------0.4

η_c^*(b) ----------0.24

Gas efficiency (τ_{ig})(c) ----------0.5

Atomic ion species ----------0.9

No. of beam lines ----------4

Reactor Vessel

Major Radius ----------5.4 m

Minor Radius ----------1.3 m

Burn time (t_b) ----------500 s

Duty cycle ----------89%

Particle loss Rate ----------(a few) x 10^{22} s^{-1}

Injection port & duct ----------0.8 x 1.2 m; 3 m long

(a) *Overall system efficiency including direct ion energy recovery*

(b) *System efficiency without direct recovery. Obtained as the product of the equilibrium fraction (0.32) and a geometric transmission efficiency (∿0.75).*

(c) *Ratio of extracted ion mass-flux to the input gas mass-flux.*

It is noteworthy that the two power conversion efficiencies listed above are to be used for different purposes: η for the electrical power requirement and η^* for calculations of required gas input. While direct recovery increases the electrical power efficiency (e.g., from 0.24 to 0.4) by recovering the energy of unneutralized ion particles, it does not reduce the total extracted ion current from a source. The input gas throughput is determined by the total extracted ion current, and thus can be estimated as $P_{inj}/(E_A \eta^* \eta_g)$, i.e., using η^* instead of η.

Tritium Contaminations

We will examine sources of tritium contamination for three different beam lines: a test-stand beam line is likely needed within a reactor site for testing and servicing replacement ion sources. The tritium beam line is included for discussion because a need may arise for tritium NBI for a beam-driven mode of operation or a possible fueling means.[3] Deuterium NBI is the most promising heat method based on the present assessment.

(1) Test-Stand Beamline. Tritium contamination will result only from D(d,P)T reactions between energetic deuterium beam particles and implanted particles in beam targets. Assuming that the embedded D concentration (C_D) is independent of the depth, the reaction rate can be expressed as

$$Y = \frac{I_b C_D}{e} \int_0^R \sigma[E(x)]dx$$

where $\sigma[E(x)]$ is the reaction cross section at an energy $E(x)$ of incoming particles at a depth x, I_b is the total D-beam current, e is the electronic charge, and R is the stopping range of deuterium with energy E_b. The

tritium yield calculated in Fig. 1 is about the same as the neutron yield (or He yield) from the other branch reaction, $D(d,n)^3He$. Calculations of these yields were reported[4] for various NBI systems. Neutron shielding is certainly required. In a beam line with the given reference design parameters (i.e., 150 keV, 205 A, and t_p = 6 sec), a total tritium yield of ∿6.6 μCi is estimated from Fig. 1. A total extracted ion current of ∿450 A per beam line (based on η*) requires an input gas flow of 90 torr-ℓ/s using η_g = 0.5 and 1 A = 0.1 torr-ℓ/s. Thus, the relative tritium impurity is several parts per billion, which may be low enough to be vented into the atmosphere. If the inventory gas were to be repeatedly circulated for reuse, however, the tritium content will build up. It is also noted that the D-T neutron yield (∿14 MeV) will exceed the D-D neutron yield (∿2.5 MeV) when the tritium content is greater than 0.5%.

(2) Tritium NBI. If we use η_g = 0.5 and an extracted tritium ion current of 450 A, the required tritium input flow per 6-sec pulse is 540 torr-ℓ, or 0.19 g, or 1.82 kCi. The needed tritium inventory could reach a few hundred grams (e.g., for 1000 pulses). A unique problem of tritium NBI will be the beta-decay heat of accumulated T_2-gas on a cryogenic pump surface. Typical heat loading of cryogenically pumped gas consists of the heat of sublimation and the enthalpy (5/2 kT). For tritium gas, the heat of sublimation of ∿237 J/g and the enthalpy of ∿265 J/g (T ∿ 77K) yield ∿95 Joule per shot (∿0.19 g). On the other hand, the beta-decay of tritium particles (0.5 ~~watts per gram~~[5]) yields a total steady-state heat loading of ∿0.1 J/sec per shot, or a total heat load of ∿50 Joules between shots (560 sec). The rate of beta-decay heat load

increases as more gas is condensed with successive shots, which may become a dominant source of heat load. For other hydrogen isotopes, the gas regeneration cycle (or the number of injection pulses between gas regeneration off the cryosurface) is determined by a permissible partial pressure (~13 torr). Insofar as the liquid helium boiloff is concerned (e.g., 14 ℓ/hr with 10 watts), however, the beta-decay heat is likely the determining factor of the regeneration cycle in tritium NBI systems.

(3) Deuterium NBI. The primary source of tritium contamination is the backstreaming gas from a reactor vessel during each injection pulse. The bakcstreaming rate depends on the gas pressure near the reactor wall, the particle loss rate from the reactor plasma, and the cross sectional area of the drift duct that connects the beam line to the reactor vessel. Although estimates exist of global particle loss rates for future machines, ~~last~~ most fuel particles would be concentrated on the divertor region if employed, thus making it difficult to estimate the net backstreaming rate through an opening to the injector system. To get an estimate of the backstreaming rate of tritium, we approximate the rate to be represented by the random molecular flux, ~~1/4 nv~~ $nv/4$. Assuming that the gas pressure near the reactor wall is $\sim 10^{-6}$ torr and the gas (50% D_2 and 50% T_2) is in thermal equilibrium at a wall temperature of ~400°C, then the backstream flux of tritium molecules (T_2) is estimated as $\sim 3 \times 10^{15}$ $cm^{-2} sec^{-1}$.

For a duct size of 80 cm x 120 cm, the backstream rate is $\sim 2.9 \times 10^{19}$ per second. The total amount of tritium depends on how fast a valve is open and closed (we will discuss later fast shutters in this respect).

Assuming that the additional "open time" is ~4 sec in addition to the beam pulse length of 6 sec, the total amount of backstreamed tritium would be 2.9×10^{-4} g per shot per beam line. It is of interest to see the relative tritium concentration to the deuterium gas loading. For a beam line with a total ion current capability of I, the deuterium gas loading is given as

$$I(1/\eta_g - \eta^*).$$

Using the reference design parameters (I = 450 A; the effective gas pulse length of 7 sec), the total deuterium molecules remaining in the beam line is 0.15 g. Therefore, the tritium contamination corresponds to ~0.12% by the number of molecules.

Gas Pumping and Handling

Gas Handling Scenarios

In the previous section, tritium sources for three beam line systems were examined. The presence of tritium affects gas handling or cycling scnearic. In a test-stand beam line within a reactor site, the used gas can be simply vented to environment, or recirculated until the tritium content builds up. In the tritium NBI system, an independent gas recycling system may be employed, although the deuterium contamination (from back-streaming from the reactor) will build up, making it necessary after a number of reuses to transfer the NBI gas inventory to the main reactor fuel recycle system.

For the deuterium NBI system which is the most likely choice for reactor, three gas handling scenarios are considered as shown in Fig. 2: (A) a total integration of the NBI gas cycle into the reactor fuel recycle system (FRS), in which the tritium-contaminated deuterium gas from NBI systems is treated in the same way as the unburnt fuel gas; (B) a partially independent gas handling scenario where the gas is recycled until the

tritium impurity reaches an intolerable level at which the entire inventory is tranfered to the reactor FRS and the NBI system receives a fresh D_2 inventory; (C) a one-way cycle[6] in which the regenerated D_2-gas is "continuously" transfered to the reactor FRS to be used as fuel while a fresh source of D_2 is "continuously" supplied to the NBI system.

Scenarios (A) and (C) are similar except the way of supplying the gas to the NBI system. Scenario (A) is rather impractical because isotope separation of near-pure D_2-gas needed for the NBI system is very complex and costly. Scenario (C) is most simple and economical; however, it would require that the amount of the deuterium gas regenerated from the NBI should not exceed too much the amount of burnt D_2 so as not to saturate the reactor FRS capacity by D_2 accumulation. Scenario (B) is only useful for the case where the NBI inventory gas can be recycled a number of times, i.e., the tritium contamination per cycle is small (<<1%).

Gas Pumping

Regeneration of gas from getter or cryocondensation surfaces poses a serious problem in the NBI as well as in the reactor vessel since it affects the reactor duty cycle with its frequency required and the finite time taken up. For a beam line of the reference design parameters a minimum pumping speed of $>10^7$ ℓ/sec is needed.

Cryocondensation pumps have been successfully used in the present-day NBI systems. The rate of liquid helium consumption could reach an uncomfortable level (in terms of cost) under radiation heat load from a reactor or together with the additional beta-decay heat in the case of tritium injectors. The calculated tritium accumulation levels in deuterium injectors, however, are low enough not ~~be~~ to affect the regeneration cycle of a cryopump.

In comparison with cryogenic pumping, Zr-Al getter pumps have some advantages. The getter surface chemically holds the gases at room temperatures and thus a large release of gas under calamities is less likely. Since they operate at ∿400°C and regenerate at ∿700°C the effect of radiation heat load (neutron, gamma, or beta) is not critical. For a low duty cycle operation necessary in our reference design, the possibility may exist for continuous regeneration bettween injector pulses. At operating temperatures of 500-600°C, the getter pump can be designed to either absorb or desorb depending on the gas pressure, which would make a continuous regeneration possible.[7] We strongly recommend that an investigation of this mode of operation be carried out and that beam system designers consider the getter pump as an alternative. While the getter pumps do not pump helium, cryopumps that can pump both He and D_2 have yet to be demonstrated.

Beam line Shutter

In case where the presence of tritium in the NBI system is a significant design constraint, a fast-acting shutter instead of or in addition to the vacuum valve can be employed to substantially decrease the accumulation for a relatively short heat-to-ignition beam pulse. It may be advantageous to allow the beam line shutter to be "leaky", so that it can be opened and closed fast and/or that it could be combined with a neutron shield.

Implications on Beamline Components

For the consideration of radiation damages only, hydrocarbon materials are not desired as the vacuum seal. They are in general noted for a source of hydrogen impurity and a sink for tritium through isotope exchanges. There appears no particular vacuum wall material requirement that is associated with the presence of tritium. Gas absorption and desorption properties of candidate metals may be important for other reasons, such as gas evolution problem from drift duct walls.[8] In the case of tritium injectors, however, tritium permeation through beam target materials into cooling water may pose a problem worth examining owing to the radiation levels around a reactor.

Owing to the radiation levels around a reactor remote maintenance is generally required. Because most tritium accumulation occurs in the pumps and the pumps can be regenerated prior to maintenance, the presence of tritium has no direct effect on remote maintenance. If a getter pump is used, however, one can imagine a failure in the regeneration system such that the getters remain contaminated with tritium. Replacing the cathodes of the ion source, which may be the most vulnerable part in a beam line, involves a break-in-the-vacuum and thus is subject to tritium safety.

Regarding to ion extraction mechanism from the source, a few implications of the tritium injector are noted. Ion extraction is generally governed by the Child-Langmuir law,[9] and thus scales as the extraction voltage to the 3/2-power and inversely with the square root of the ion mass. Thus, if an identical source is used to produce either T^+ or D^+ at the same energy, the T^+ current is reduced by the square root of mass ratio, but the ion-to neutral conversion efficiency is higher. The ratio of T° power to D°

power is then given roughly as $f_o(E_b/3)/\sqrt{1.5}\ f_o(E_b/2)$, yielding ~1.18 for E_b = 150 keV, where f_o is the near-equilibrium fraction of neutrals. Because the neutral penetration into the reactor plasma is approximately linear with the energy per nucleon of the neutrals, however, tritium ions must be accelerated to an energy 1.5 times higher than deuterium ions for the similar degree of penetration.

Finally, we note that in the TFTR project, the NBI systems and the main tokamak vessel are located in "not seismically protected" area and thus the maximum inventory of tritium recommended for this area is merely 24,000 Ci (or 2.5 gram). One way to increase the allowable inventory level is to design the entire fusion device to be seismically protected. This would likely be difficult and expensive, especially with the design of beam ducts and vacuum connectors, and adversely affect the already difficult remote maintenance problem.

Conclusions

We have examined tritium-related implications in NBI systems. The amount of tritium accumulation in a beam line affects to some degree the gas recycling scenario. The choice of vacuum pump depends not only on its pumping capability but also by the ease of regeneration, and the possible use of Zr-Al getters for a continuous regeneration is recommended to be studied. The beta-decay heat of tritium may become a serious constraint on cryopumps in the tritium NBI systems. The rough estimates of backstreamed tritium in a deuterium NBI system of the reference design showed that the tritium contamination does not pose a serious problem on beam line components, nor on pumps, and thus the regenerated gas can be reused a number of times without purification.

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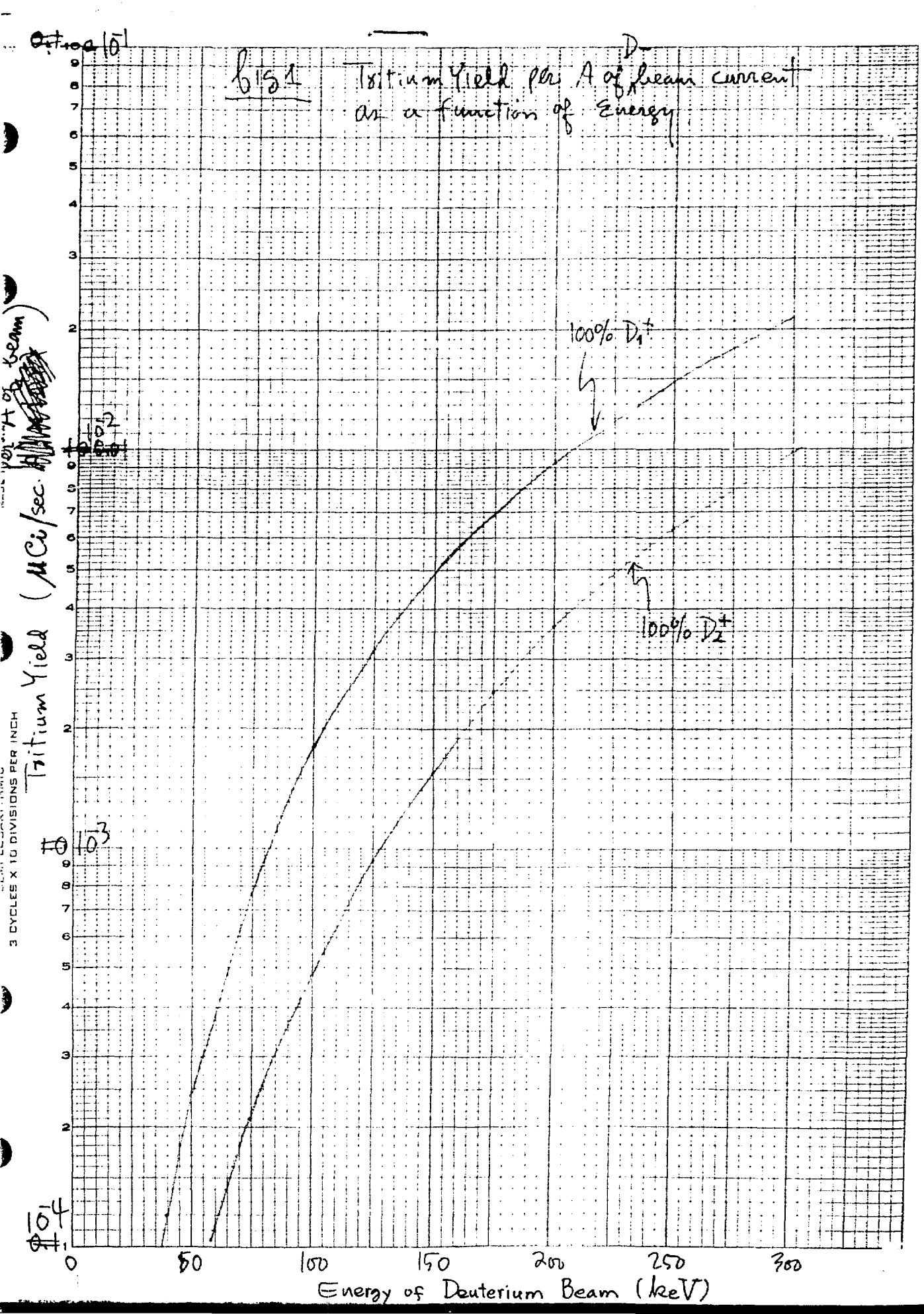
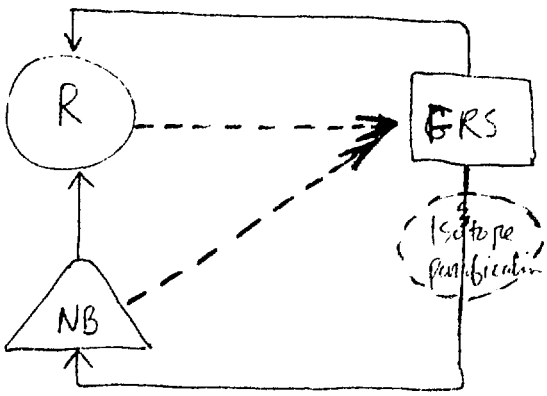
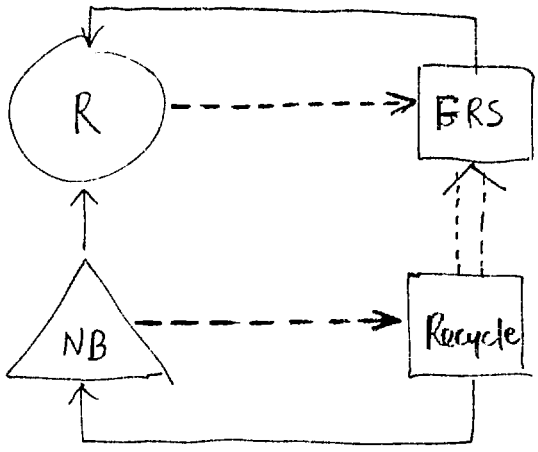


Fig. 1 Tritium Yield per A of beam current as a function of Energy
100% D_1^+
100% D_2^+
Tritium Yield (μCi/sec. per A of beam)
Energy of Deuterium Beam (keV)
0
50
100
150
200
250
300
3 CYCLES X 10 DIVISIONS PER INCH



(A)



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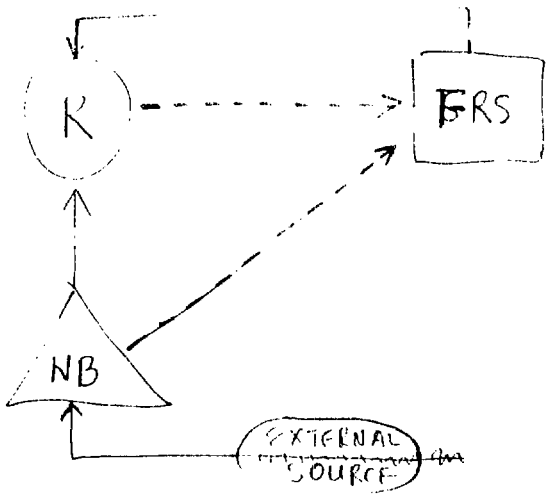

(C)

Fig. 2. Three Gas Handling Scenarios. Dotted lines indicate exhaust gas flows. R ··· Reactor, FRS ··· fuel recycle system, NB ··· Neutral Injectors