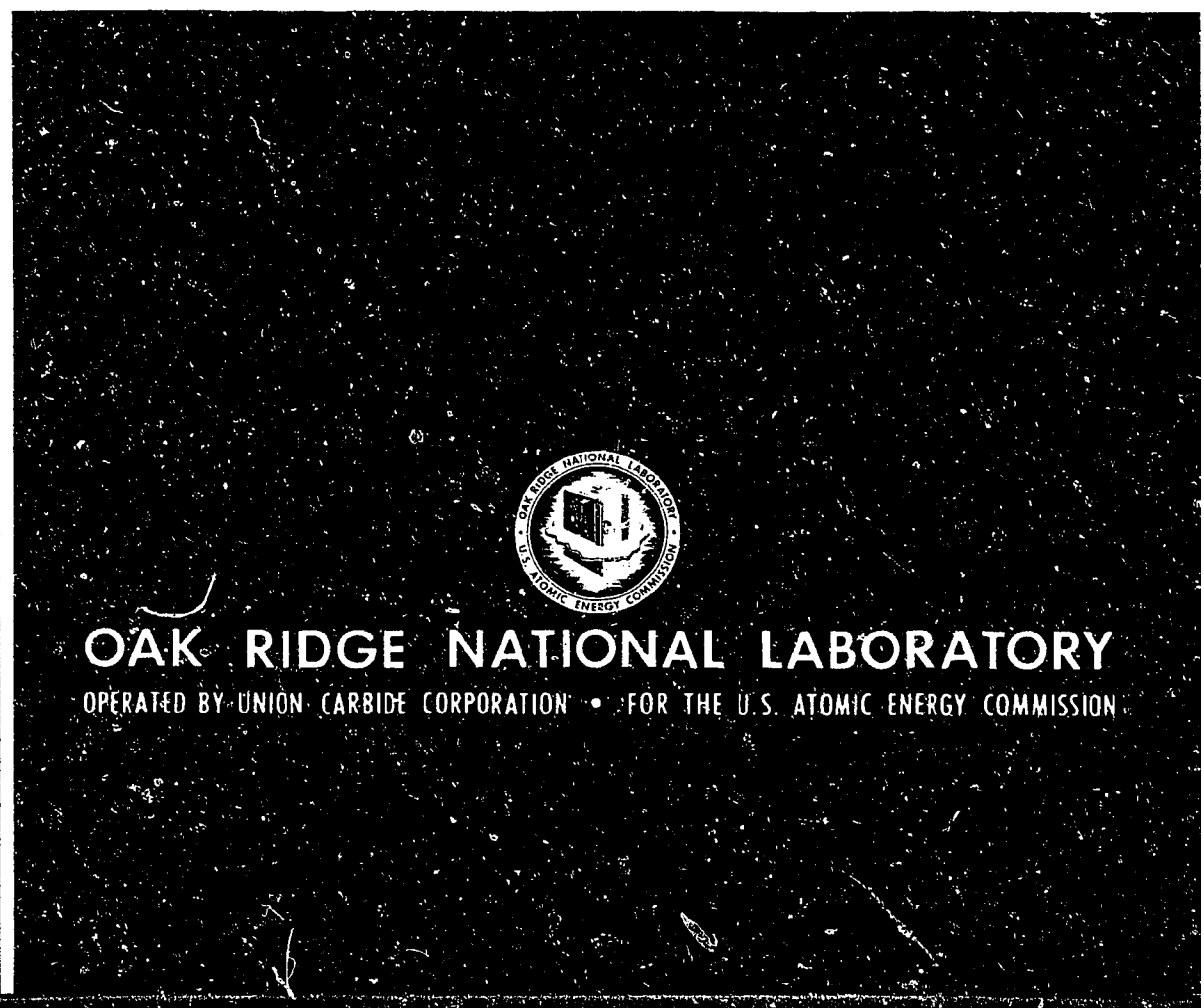
ORNL-TM-5080

Scoping Studies of Tritium Handling in a Tokamak Experimental Power Reactor

J. S. Watson
R. Cherdack
S. D. Clinton
P. W. Fisher



BLANK PAGE

Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy $5.50; Microfiche $2.25

ORNL-TM-5080

Contract No. W-7405-eng-26

CHEMICAL TECHNOLOGY DIVISION

SCOPING STUDIES OF TRITIUM HANDLING IN A TOKAMAK EXPERIMENTAL POWER REACTOR

J. S. Watson
R. Cherdack*
S. D. Clinton
P. W. Fisher

JANUARY 1976

---

*Burns and Roe, Inc. Work done under Subcontract No. 3928.

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

CONTENTS

SCOPING STUDIES OF TRITIUM HANDLING IN A TOKAMAK EXPERIMENTAL POWER REACTOR

J. S. Watson S. D. Clinton
R. Cherdack P. W. Fisher

ABSTRACT

Tritium handling techniques in an experimental fusion power reactor (EPR) are evaluated to determine the requirements of the system and to compare different equipment and techniques for meeting those requirements. Tritium process equipment is needed (1) to evacuate and maintain a vacuum in the plasma vessel and the neutral beam injectors, (2) to purify and recycle tritium and deuterium for the plasma fuel cycle, (3) to recover tritium from experimental breeding modules, and (4) to provide tritium containment and atmospheric cleanup. A development program is outlined to develop and demonstrate the required techniques and equipment and to permit confident design of an EPR for operation by the mid-1980s.

1. INTRODUCTION

The experimental power reactor (EPR) operating at 150 MW(t) from fusion reactions would consume 200,000 Ci, or 20 g, of tritium per day. If the fraction burned is 1%, this would correspond to a tritium process rate of 20 MCi, or 2 kg, per day. This will be, by far, the largest tritium handling system built for a fusion experiment by the mid-1980s. The tritium handling system will include equipment to evacuate the reactor vessel, purify and recycle fuel to the reactor, separate deuterium and tritium for injection, extract tritium from the breeding blanket, and control tritium releases by containment and atmosphere cleanup. In size and complexity, the EPR system will represent significant extrapolations from the Tokamak Fusion Test Reactor (TFTR) tritium system, including larger quantities of tritium, more complete and more reliable containment, and larger equipment, as well as additional process steps. The EPR may provide the first realistic demonstration of all major components of a fusion power reactor tritium handling system. It will be the first operating

fusion device to provide on-site purification and recycle of unburned fuel. (TFTR plans currently call for collection of unburned fuel on solid getters. The fuel is, perhaps, recovered and purified off-site.) The EPR will include the first on-site isotope separation equipment and, probably, the first on-line experiments for recovering tritium bred in a CTR blanket.

The EPR will be a major experiment in many areas of plasma physics and fusion reactor technology. Tritium handling operations and experiments will constitute only a portion of the overall program; therefore, the tritium handling equipment (like most other parts of the system) must have a relatively high degree of reliability. The first goal of the tritium handling system will be to ensure that the probability and consequence of releases to the environment will be acceptably low. The second goal is to ensure, with relatively high probability, that tritium releases or equipment malfunction will not hinder other portions of the experiment. The third goal is to obtain data and experience in designing tritium handling equipment for later fusion devices, and for demonstration and commercial reactors.

During FY 1975, tritium handling experiments were limited to scoping studies (i.e., definition of problems and listing process options which could be used); no attempt was made to develop a specific design. Comparisons were made of the merits of various process options and the conditions under which they appear practical; however, in most cases, no firm choice was made. During FY 1976, many options in all portions of the EPR will be eliminated; specific (preliminary) designs will then become useful.

Perhaps the most useful results of the scoping studies of EPR tritium handling system are the establishment of estimates of the magnitude of the tritium handling problems and a better understanding of the developmental work needed to evaluate some proposed techniques and to provide the process systems required for an EPR. Although ORNL tritium experimental programs have been conceived and planned specifically to meet the anticipated needs of the CTR development, the scoping studies have shown

a need for additional or expanded efforts in two areas: recovery of tritium from lithium metal blankets, and tritium handling in solid feed systems. Methods for recovering tritium from lithium blankets are currently being studied at ORNL (and elsewhere), but a significant expansion of this effort is needed. Specific process techniques needing evaluation are discussed. Far less emphasis has been devoted to the feed system, and all of the process and containment aspects of solid pellet feed should be studied. Additional work is needed to properly couple the injector gas recovery stream to the main plasma exhaust system.

Further advances in the tritium handling system design will eventually require additional information from other portions of the EPR studies. In the next few years, the EPR program must attempt to define the fueling system, divertor system (if needed), and blanket cooling and breeding systems. In addition, an effort is needed to estimate purity requirements for all EPR feed streams and operating cycles for all systems.

2. RADIOLOGICAL HAZARDS

Tritium, an isotope of hydrogen with two neutrons in the nucleus, decays by low-energy beta emission to ^{3}He with a half-life of 12.33 years. The peak energy of the beta particle is 0.0186 MeV, and the average beta energy per disintegration is 0.006 MeV. Because the radioactive emission is a low-energy beta particle, there are only small external dose hazards and no shielding requirements associated with tritium. However, there are significant internal dose hazards which are closely related to the length of time the tritium remains in the body.

The biological half-life, the length of time required for an initial body burden of a radionuclide to be reduced by one-half (assuming no further ingestion), is the most commonly used measure of the duration of a radionuclide's residence in the body. The biological half-life takes into account all removal mechanisms, including radioactive decay, perspiration, respiration, and excretion. Thus, for many radionuclides, the various organs have different biological half-lives, and certain organs tend to concentrate some elements (e.g., the thyroid concentrates iodine).

The rate of tritium uptake by the body is very sensitive to the chemical form of the tritium. For example, if uptake in the form of HT, T_2, or DT is very slow and the resulting dose can be small, a dose calculation cannot be based simply on the assumption that the T-containing gas molecules come to equilibrium with the hydrogen in the body. Only a small amount of tritium will exchange with the hydrogen in the water present in the lungs.

Tritiated water behaves biologically in a manner similar to that of normal water. The biological half-life of water in the human body varies from 4 to 18 days. Twelve days was chosen for calculations in this study. This figure can be used for water absorbed through the digestive tract, the lungs, or the skin. It is generally assumed that all water entering the lungs and digestive tract is absorbed into the body. In addition, the amount of moisture absorbed through the skin may range from nearly zero to approximately the amount absorbed in the lungs. For this study, the moisture absorbed through the skin was assumed to be equal to that absorbed through the lungs.

The exchange of tritium in tritiated water with hydrogen in organic compounds in the body is not a large factor in dose calculations since the exchange rate very nearly corrects for the longer biological half-life of tritium in organic compounds. Ingestion of tritiated organic matter through the digestive tract results in about the same dose per curie as ingestion of tritiated water for similar reasons. It has been suggested that about 10% of organic compounds entering the lungs are absorbed as compared with 100% of the water. There is essentially no absorption of organic compounds through the skin.

Because tritium in the form of tritiated water represents the more serious hazard, because large, hot metal surfaces abound in the EPR, and because many tritium removal mechanisms which may be used in safety systems rely on oxidizing tritium, safety-related dose calculations for this study of an EPR assumed that all tritium is in the form of tritiated water.

In calculating doses resulting from tritium, the important quantity to be developed is the total energy absorbed per gram of affected body

tissue. In accordance with the discussions above, the affected tissue is considered to be the entire body, and the tritium is assumed to be uniformly distributed. Because the emission is a low-energy beta, it is appropriate to assume that essentially all the emitted energy is absorbed in the body.

The unit of dose used in most safety guidelines is the roentgen equivalent man (rem). In the case of beta particles, 1 rem is equal to 100 ergs of energy absorbed per gram. Thus, the dose received from ingestion of a number of curies of tritium, Q, in the form of tritiated water may be calculated from the following equation:

$$D = Q(Ci)\left[3.7 \times 10^{10}\left(\frac{dis}{sec \cdot Ci}\right)\right]\left[0.006\left(\frac{MeV}{dis}\right)\right]\left[1.6 \times 10^{-6}\left(\frac{erg}{MeV}\right)\right]\left[10^{-2}\left(\frac{rem}{erg \cdot g}\right)\right]$$

$$\times \left[\frac{12 \times 86{,}400}{0.693}\ (sec)\right]\left[\frac{1}{W}\ (g^{-1})\right] ,$$

where W is the body weight in grams.

The International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection have formulated recommendations concerning maximum radiation doses to both the population at large and workers in the nuclear industry. These recommendations[1,2] can be summarized as follows:

(1) No member of the general public should receive a whole-body dose of more than 500 millirems per year.

(2) The average whole body dose to the general public should not exceed 170 millirems per year.

(3) Workers in the nuclear industry shall not receive a whole-body dose of more than 5 rem per year.

(4) Exposures to radiation should be kept to a minimum.

There are other recommendations concerning specific organs, but these are not relevant to tritium. The federal government has established

and continually revised regulations in accordance with these recommendations. The bulk of these are contained in Title 10 of the Code of Federal Regulations (10 CFR). The portions of 10 CFR that are most relevant to power plants are Parts 20, 50, and 100. While an EPR, as a government-operated facility, would not be legally bound to conform with these standards, it seems reasonable to expect the predecessor of fusion power plants to be designed to operate within limits specified for fission power plants.

Part 20 covers maximum allowable doses to both members of the population at large and people employed within nuclear facilities. It sets standards for direct irradiation and maximum permissible concentrations (MPCs) of radionuclides in air and water. The MPCs for tritium in restricted and unrestricted areas from 10 CFR 20 are shown in Table 1.

Table 1. Maximum permissible concentrations for tritium in restricted and unrestricted areas

Tritium concentration (μCi/cc)	Restricted area	Unrestricted area
In water	1×10^{-1}	5×10^{-3}
In air		
Ingestion	5×10^{-6}	2×10^{-7}
No ingestion	2×10^{-3}	

The MPCs that are most relevant to EPR studies are those for air in restricted areas and for air and water in unrestricted areas. The MPCs for air in restricted areas fall into two categories. The first assumes that the tritium in the air is in the form of tritiated water and is ingested into the body, where it remains in accordance with the mechanisms discussed in the previous section. Exposure to the MPC on this basis for 1 year would result in a dose of approximately 5 rem. Allowable concentrations are permitted to increase if it can be assured that only a fraction of an employee's time will be spent in the area. For the second case,

the MPC is based on no ingestion of tritium (i.e., protective clothing and breathing equipment are used). While protective clothing allows personnel to work in relatively high concentrations of tritium, the effect of such clothing on productivity implies that the personnel will spend more time in the restricted area and, perhaps, be subject to greater doses from gamma or neutron sources in the area.

The MPC for air in unrestricted areas is also based on the assumption that the tritium is in the form of tritiated water and takes into account appropriate ingestion mechanisms and the biological half-life. If an individual continuously remains in air containing the MPC of tritium, he will receive a dose of approximately 170 millirems per year from the ingested tritium. The MPC for water is based on the ingestion of 1.1 liters of tritiated water per day continuously, and also results in a dose of about 170 millirems per year.

Along with the detailed MPC requirements, 10 CFR 20 contains the general statement that all releases of radionuclides shall be as low as practicable.

Part 50 of 10 CFR contains various design criteria for light-water fission power plants. A proposed Appendix I to 10 CFR 50 covers those portions of the plant that affect the normal releases of radionuclides to the environment. The basic criteria contained in this Appendix are as follows:

(1) The concentration of radioactive iodine isotopes with half-lives greater than 8 days shall not exceed 10^{-5} times the MPC at the site boundary.

(2) The total dose from noble gases to an individual at the site boundary shall be no greater than 10 millirems/year.

(3) The whole body dose from gaseous effluents to any member of the general public shall not exceed 5 millirems/year.

(4) The total quantity of radionuclides other than tritium in liquid effluent shall not exceed 5 Ci/year.

(5) The total radionuclide concentration in the liquid effluent shall not result in an annual dose of more than 5 mrem to any individual.

These design criteria were developed to help quantify the "minimum practicable" release requirement of 10 CFR 20 and to provide guidance to the designers of light-water fission power plants. Two aspects of Appendix I should be kept in mind. First, the design criteria are not expected to be applied in an absolutely rigid manner. Information obtained from recently published documents[3] indicates that, at least with respect to the radioiodine limits, designs which represent a bona-fide attempt to comply with the minimum release criteria will be acceptable even if calculations indicate that the numerical limits may be exceeded. The second aspect is that Appendix I has not yet been formally adopted and, even after adoption, will be subject to review and possible revision. For example, it has been recommended that significantly higher skin doses from gaseous effluents be allowed.

Even though 10 CFR 50 generally applies to commercial installations and Appendix I applies only to commercial light water reactors, the release criteria may be used as guidelines to acceptable standards for an EPR. The third and fifth criteria appear to be the most relevant. [Even though tritium is not mentioned as a gaseous effluent in the proposed Appendix I, it will probably be included in the whole body dose limit (5 millirem) for an EPR.] Systems discussed later in this report have been studied to assess their ability to comply with these criteria.

Siting criteria are provided by 10 CFR 100. The basic site-related considerations involve geology, hydrology, meterology, and demography. The principal relevant numerical limits related to radioactive release concerning the site are as follows: The site shall be sufficiently large that (1) no person outside the site boundary shall receive a whole body dose of more than 25 rem in the 2-hr period following the release of radionuclides associated with a maximum credible accident; (2) a zone around the plant exists whose perimeter is large enough such that a person at

the outer boundary shall receive a whole body dose of no more than 25 rem from the radioactive cloud resulting from a maximum credible accident, but whose population is small enough to permit action to be taken to evacuate or protect this population; and (3) the distance from the plant to the nearest population center of 25,000 or more is at least one and one-third times that to the outer edge of the low-population zone.

These criteria are to be considered as reference points; the fact that a site meets them does not ensure acceptability of the site. The whole body dose limit of 25 rem is taken from an NCRP recommendation that a once-in-a-lifetime accidental whole body dose of 25 rem may be disregarded in determining the radiation exposure of a worker in the nuclear industry.

While there are no guidelines as to what would comprise a maximum credible accident for an EPR, existing general criteria on quality assurance and redundancy of containment and cleanup systems to prevent and mitigate releases could be applied to an EPR.

3. RADIOLOGICAL SAFETY CRITERIA

The preceding section outlined the hazards associated with tritium and discussed the federal regulations that may apply to an EPR. Performance criteria were established concerning tritium concentrations and releases which lie within the numerical limits of these regulations. Design criteria were then formulated to determine the various features that the tritium handling systems and related auxiliaries must have to ensure compliance with the performance criteria and to minimize the probability of accidental tritium releases. These two classes of criteria are described below.

3.1 Performance Criteria

The reactor building will, of course, be a restricted area; thus the applicable regulatory limit for tritium is 5×10^{-6} μCi per cubic centimeter of air. The limit on tritium may be increased by requiring personnel to use protective clothing and breathing equipment or by ensuring

that they will be exposed to the atmosphere of the reactor building for only small fractions of their work time. However, it was decided not to adopt a criterion based on the use of protective clothing and equipment since these items tend to decrease productivity as well as prolong the amount of time that personnel would spend in the reactor building to perform a given task, and hence increase the dose received from external neutron, gamma, and beta sources. In fact, the criterion chosen was that the reactor building atmosphere should contain no more than 5×10^{-7} μCi of tritium per cubic centimeter during maintenance periods so that the tritium would not contribute significantly to the dose received by maintenance personnel. Thus most of the maximum allowable dose would arise from exposure to external neutron, gamma, and higher-energy beta radiation.

In selecting a numerical performance criterion for the release of tritium to the environment during a maximum credible accident, the considerations that should determine the maximum allowable release were not clearly defined. As discussed in Sect. 2, a site might be considered acceptable if the whole body dose at the boundary for the 2-hr period following a release from a maximum credible accident is less than 25 rem. However, 25 rem seems fairly high in view of the small size of an EPR and the alleged safety advantages of fusion over fission. In any case, it is not clear that the entire 25-rem allowance could be devoted to tritium since other radionuclide releases may be associated with the as-yet undefined maximum credible accident.

At the other extreme, an AEC Safety Guide[4] states that, if the failure of a given component of a nuclear power plant results in a dose at the site boundary of less than 170 millirems, then it may be designed and manufactured in accordance with applicable commercial codes (as opposed to the more stringent nuclear codes). Therefore, if the reactor building containment and cleanup systems are such that the 170-millirem dose limit can be met, the major tritium bearing systems can be designed and constructed in a more economical manner.

While the two relevant criteria discussed above are two orders of magnitude apart, they are expressed in a similar manner (i.e., in terms

of whole body dose at the site boundary). Thus, it was determined that the maximum tritium release limit should be expressed in terms of site boundary dose. For reasons discussed above, it would be desirable to choose a dose limit far below 25 rem — in fact, as close to 170 millirems as possible. However, because so many detailed design data are lacking in the scoping study phase and many arbitrary assumptions had to be made in dose calculations, including the amount of tritium released to the reactor building, the site boundary distance, and the meteorology, we decided to focus on the more nominal goal of a whole body dose of 1.0 rem. In more detailed EPR design studies, it will be possible to examine more precisely what changes in containment and cleanup systems, if any, will be required for a sixfold decrease in the site boundary dose.

A limit of 5 millirems per year for the whole body dose at the site boundary was selected from normal releases in accordance with the spirit of the guidelines of Appendix I of 10 CFR 50.

Design criteria and preliminary design information which result from these numerical performance criteria are discussed later in this section and also in Sect. 4.

3.2 Design Criteria

The following criteria were formulated to provide guidance for the design of the tritium handling systems:

(1) All systems containing significant quantities of tritium in either gaseous or potentially gaseous form shall be enclosed. The atmosphere of these enclosures shall be maintained at a slight negative pressure and processed through a cleanup system designed to remove tritium.

This criterion was established to limit normal leaks to the reactor building as well as to reduce the probability of large leaks due to accidents. It is consistent with present practice for all new systems being constructed to handle large quantities of tritium.

(2) Tritium-bearing systems shall not release their inventories as a result of design-basis earthquakes, tornadoes, or the missiles generated by them. Protection must be provided for support systems, such as cryogenic refrigerators or electric power supplies, which could affect tritium containment.

This criterion was established to reduce the likelihood of releases resulting from natural disasters.

(3) If possible, all tritium handling systems shall be located in the reactor building. In any case, such systems shall be enclosed within containment structures that are reasonably leak-tight (comparable to BWR secondary enclosures), maintained at a slight negative pressure, are missile protected, and are of seismic I design.[5] All containment structures shall have elevated release points.

This criterion, which is aimed particularly at mitigating the effects of the maximum credible accidents, also leads to the control of releases due to normal leaks and smaller accidents.

(4) All containment buildings shall be equipped with atmosphere cleanup systems which remove tritium at a rate and an efficiency necessary to meet the site boundary dose criterion (1 rem) for a maximum credible accident and the air concentration criterion for normal maintenance conditions (5×10^{-7} μCi/cc) established above.

(5) Where possible, tritium handling systems shall operate at temperatures and pressures below those at which significant diffusion of tritium through metal walls occurs.

(6) Monitors shall be provided throughout containment structures, particularly at possible tritium leakage points. All release points from the containment structures shall be monitored.

Monitors will be needed to protect personnel. Local monitors at potential leak points will provide fast response, aid in the protection of personnel, and help locate failures. Monitors at release points will aid in evaluating the performance of the containment and cleanup system and in determining the environmental impact of any release.

(7) Those systems containing significant quantities of highly flammable materials shall have inert gas as a blanket in their enclosures.

4. TRITIUM CONTAINMENT AND ATMOSPHERE CLEANUP SYSTEMS

The tritium containment and atmosphere cleanup systems minimize the leakage of tritium and its consequences. These systems consist of (1) enclosures around tritium-bearing components and piping, (2) the cleanup devices that maintain a slight negative pressure and remove tritium from the atmosphere in these enclosures, (3) the reactor containment building and the concomitant atmosphere cleanup and HVAC (heating, ventilating, and air conditioning) equipment, and (4) other tritium containment structures, if any, plus their atmosphere-cleanup and HVAC components.

The enclosures and their cleanup systems are designed to prevent leakage to the containment buildings. The atmosphere cleanup systems associated with the containment building are intended to decrease tritium concentrations in the buildings to facilitate maintenance and to reduce the release of tritium to the environment. The function of the containment building(s) is to prevent the release of tritium to the environment. It also serves to control the releases of various activation products as well as to provide biological shielding.

4.1 Summary of Accomplishments in FY 1975

During this scoping study, several criteria were formulated for performance and design of tritium containment and atmosphere cleanup systems. These criteria, which were presented in the preceding section, include the

following performance guidelines: a limit of 5×10^{-7} μCi/cc for tritium in the containment building atmosphere during maintenance, a maximum annual dose of 5 millirems from normal tritium release at the site boundary, and a whole body dose of 1 rem at the site boundary for 2 hr following the maximum credible tritium release. These criteria result in the following design requirements: double enclosure of tritium systems; provisions to ensure resistance to seismic and tornado damage; containment structures; atmosphere cleanup systems; low-temperature, low-pressure operation where possible; adequate monitoring; and inert gas blankets for flammable systems.

Analyses were then performed to determine the feasibility of meeting the performance criteria and the necessity for some of the criteria. Preliminary conceptual design of the systems was begun.

4.2 Range of Objectives

The minimum objectives for the tritium containment and atmosphere cleanup are the limiting of off-site releases to the lowest practicable level and the reduction of tritium content in the reactor containment to a level which will permit periodic maintenance. The purpose would be to provide for safety of the general public.

The systems could be designed to accomplish this minimal goal; if so, they would have little detailed relevance to future power reactors. If the EPR were located on a large, remote site, low tritium removal rates or high leakage rates might be tolerable while still keeping off-site doses within acceptable limits.

The maximum objectives for these systems would be to maintain tritium levels in the reactor building in accordance with the performance criterion of 5×10^{-7} μCi/cc (discussed earlier) so that the tritium dose to maintenance personnel would be negligible, and to maintain the off-site dose limits of 5 millirems for normal and 1 rem for maximum credible tritium releases, even in the case of a fairly small site. It would also be beneficial if they could be used as prototypes for similar systems in future power reactors.

Systems achieving these maximum goals would ensure the safety of the general public, allow greater flexibility in maintenance and operations, and provide useful design, construction, and operating experience for commercial reactors.

4.3 Possible Solutions

The design performance requirements for these systems have been outlined in preceding paragraphs and in Sect. 3. To meet these requirements, it is necessary to achieve the following:

(1) minimize leakage from tritium-bearing systems to the reactor building atmosphere,

(2) remove tritium from the reactor containment building atmosphere,

(3) control leakage from the reactor building to the environment.

The minimization of tritium leakage to the building requires very tight systems, with barrier temperatures below those where tritium diffusion is significant, and/or double-enclosure techniques. Removal of tritium from the building atmosphere requires a system which can oxidize tritium and subsequently adsorb or otherwise collect the tritiated moisture. Control of leakage to the environment requires a tight containment building with minor, uncontrolled leakage or a building maintained at a slight negative pressure with a continuous, controlled elevated exhaust to compensate for inleakage.

4.4 Selection of Techniques

In this phase of the scoping study, an attempt was made to evaluate the best way of meeting the performance criteria that have already been outlined in Sect. 3. Some of these evaluations resulted in confirmation of design criteria contained in Sect. 3, as well as in the conclusions discussed below.

An examination of the size, operating conditions, diversity, and complexity of the various systems that will pump, process, or otherwise

handle large quantities of tritium indicates that some leakage will be inevitable. Therefore, the use of double enclosures at slight negative pressures, with cleanup systems to remove tritium from their atmospheres, is considered to be the only way to ensure minimum leakage to the reactor building. Brief analyses, which will be discussed in a later subsection, indicate the extremely small amount of leakage that is tolerable to ensure compliance with even less stringent criteria than 5×10^{-7} μCi/cc. Also, the use of double-enclosure and cleanup systems is accepted practice today, and certain portions of the tritium handling systems (e.g., uranium beds or gas storage cylinders) may represent fire or explosion hazards unless contained within inert-atmosphere enclosures. Many systems for removing tritium from enclosure or reactor building atmospheres are commercially available, but no attempt has been made to choose the most promising of these in the scoping study.

A containment building with a slight negative pressure and a controlled, elevated exhaust was judged to be more effective in reducing doses from a maximum credible tritium accident and probably more economical than a PWR-type containment building. However, future studies of the effect of the normal release of air activation products and the costs of PWR- and BWR-type containments might alter this conclusion.

4.4.1 Further design considerations

The exact nature of double enclosures of the tritium-containing system will depend on detailed system designs, access requirements, and physical arrangements. As discussed above, it may be necessary to maintain inert atmospheres in some enclosures, depending on the quantities of hydrogen isotopes or other materials contained by the enclosed system. All enclosure atmospheres will be maintained at a slight negative pressure and will be processed continuously through cleanup systems to remove tritium. Process designs for these cleanup systems will be similar to those for the reactor building.

The atmosphere cleanup system for the reactor building will contain heaters and catalytic or CuO beds to oxidize tritium to tritiated

water. The tritiated water will then be collected on dryer beds probably consisting of refrigerated molecular sieves which are regenerable. These beds are usually operated at about 100 psi. The exact system flow rate, although not yet developed, will probably be on the order of one-fourth to one-half building volume per hour for rapid tritium removal. A tritium decontamination factor (DF) of 100 is considered both achievable and desirable. Addition of hydrogen or steam may be necessary for isotopic dilution in order to achieve this DF for low concentrations of tritium. Analyses of the performance requirements of this system are discussed in the following subsection.

The volume of the reactor building will be on the order of 4,000,000 ft^3. Based on data from buildings of a similar nature, it should be possible to maintain a pressure of -0.25 in. H_2O with an inleakage rate of less than one-half building volume (2,000,000 ft^3) per day. An amount similar to the inleakage would be exhausted from an elevated release point about 100 m high. This exhaust stream would be taken from the outlet side of the reactor building's atmosphere cleanup system.

Tritiated liquids resulting from the regeneration of dryer beds of cleanup systems will be pumped to a small facility for solidification, drumming, and off-site shipment.

4.4.2 Analyses

The analyses performed involved determination of the allowable tritium leak rate to the reactor building and evaluation of the performance of the reactor building, its atmosphere cleanup system, and stack

The allowable leak rate to the building, L, with a given removal system flow rate, W, to maintain a maximum concentration of 5×10^{-7} Ci/m^3 is given by:

$$L\left(\frac{Ci}{sec}\right) = W\left(\frac{m^3}{sec}\right) \times 5 \times 10^{-7}\left(\frac{Ci}{m^3}\right). \qquad (1)$$

If W is approximately one-quarter building volume per hour (17,000 cfm or 7.9 m^3/sec), $L = 4 \times 10^{-6}$ Ci/sec or 0.3455 Ci/day. Thus, even

with a substantial increase in flow or relaxation of the concentration criteria, the need for double containment of systems handling several millions of curies of tritium per day would probably remain.

The following assumptions were used to evaluate the effectiveness of reactor building containment and atmosphere-cleanup schemes in reducing the 2-hr site boundary doses:

(1) Minimum dispersion conditions exist for 2 hr following a maximum credible accident.

(2) A total of 5×10^{7} Ci of tritium is released to the reactor building following the maximum credible accident.

(3) The breathing rate of an individual at the site boundary is 3.47×10^{-4} m^3/sec.

(4) All tritium is in the form of tritiated water.

The equation used to calculate the dose, D, from the maximum credible accident was:

$$D\ (\text{rem}) = (Q)\left(\frac{X}{Q}\right)(BR)(2)(T)\left(\frac{\tau_{\frac{1}{2}B}}{0.693}\right)(E)(3.7 \times 10^{10})(1.60 \times 10^{-8})\left(\frac{1}{W}\right), \quad (2)$$

where

Q = release rate from the reactor building, Ci/sec;

$\frac{X}{Q}$ = dispersion factor from AEC Safety Guide 25,[4] sec/m^3;

BR = breathing rate in m^3/sec; AEC Safety Guide 25 specifies a rate of 3.47×10^{-4} m^3/sec;

T = length of exposure, assumed to be 7200 sec;

$\tau_{\frac{1}{2}B}$ = biological half-life, assumed to be 1.04×10^{6} sec;

E = average energy absorbed per disintegration, assumed to be 0.006 MeV;

W = weight of the affected organ chosen conservatively as 4×10^{4} g (whole body).

A factor of 2 is included to account for adsorption of moisture through the skin; the factor 3.7×10^{10} is the number of disintegrations per sec per Ci; the factor 1.60×10^{-8} converts from MeV to rem g.

The two system-dependent factors in the equation are Q and X/Q (X/Q depends on the elevation of the release point). The two containment and cleanup system combinations evaluated were: a very tight containment building which leaked at the rate of 0.01 building volume per day, and had a cleanup system with a flow rate of one building volume per hour and a tritium DF of 100 (or higher); and a containment building which operated under a slight negative pressure, had an inleakage rate of one-half building volume per day, and exhausted a similar amount from the outlet side of a cleanup system with a tritium DF of 100 through a 100-m-high stack.

The release rate from the first case varies significantly with time, and the integrated release over a 2-hr period is given by:

$$\int_0^{7200\ \text{sec}} Q\ dt = \int_0^{7200} \frac{0.01}{86{,}400} (5 \times 10^7)\left(\frac{-0.99t}{3600}\right) dt \qquad (3)$$

$$\approx 5.8\ (1-e^{-2})(3600) = 17{,}900\ \text{Ci}\ .$$

The X/Q for a ground-level release during minimum dispersion conditions at a site boundary distance of 200 m is approximately 3×10^{-3} sec/m^3, including the effect of turbulence from the building wake. Thus, the dose, D, at the site boundary calculated from Eq. (2) would be:

$$D = (17{,}900)(3 \times 10^{-3})(3.47 \times 10^{-4})(2)\left(\frac{1.04}{0.693}\right)(10^6)(0.006)(3.7 \times 10^{10})$$

$$\times\ (1.60 \times 10^{-8})\left(\frac{1}{4 \times 10^4}\right) = 4.9\ \text{rem.}$$

The second case with the controlled exhaust from the elevated release point was evaluated using the conservative approximation that the quantity

of tritium in the building (and hence the release rate) remains constant for 2 hr. Thus, the release rate Q is given by:

$$Q = \left(\frac{0.5}{86,400}\right)(5 \times 10^{7})[10^{-2}\ (DF)] = 2.89\left(\frac{Ci}{sec}\right).$$

The X/Q value for a 100-m-high release point during fumigation conditions is 5×10^{-4} sec/m^3 at a site boundary distance of 200 m. Therefore, the site boundary dose for the second case is:

$$D = (0.289)(5 \times 10^{-4})(3.47 \times 10^{-4})(2)\left(\frac{1.04 \times 10^{6}}{0.693}\right)(7200)(0.006)$$

$$\times\ (3.7 \times 10^{10})(1.72 \times 10^{-8})\left(\frac{1}{40,000}\right) = 0.96 \text{ rem.}$$

Note that taking into account the effect of a moderate recirculating cleanup system with a flow rate of one-fourth building volume per hour would reduce the 0.96 rem to approximately 0.62 rem.

An additional analysis was performed to determine the dose resulting from tritium released during normal operation of the EPR. With a 100-m release point elevation, a release rate of 2,000,000 ft^3 (one-half building volume) per day (0.66 m^3/sec), a concentration of 5×10^{-7} μCi/cc in the building, and a X/Q value of 7×10^{-5} sec/m^3 at 200 m for nonfumigation conditions were taken from AEC Safety Guide 25.[4] The annual dose, D, is given by:

$$D = (0.66)(5 \times 10^{-7})(7 \times 10^{-5})(2.4 \times 10^{-4})(2)(3.15 \times 10^{7})\left(\frac{1.04 \times 10^{6}}{0.693}\right)$$

$$\times\ (3.7 \times 10^{10})(1.60 \times 10^{-8})\left(\frac{1}{4 \times 10^{4}}\right) = 4.6 \times 10^{-5} \text{ rem/year.}$$

Note that this calculation assumed that the exhaust concentration was equal to the building concentration when, in fact, it could be a factor of 100 lower (if the exhaust tritium removal system was operating under normal conditions). In any case, the calculation shows that, even

with considerable increases in building or exhaust tritium concentration, the annual average dose rate at a 200-m site boundary due to normal tritium releases will still be below 5 millirems.

4.5 Program Requirements

Definitive designs for the double enclosures cannot begin until the designs and layouts for major tritium-bearing systems for the EPR are available.

Additional information on the behavior of tritium in the environment and the modes and consequences of tritium releases from fusion facilities would be useful in establishing final criteria and designs for tritium containment and atmosphere cleanup systems. Work in these areas is being performed at ORNL and Pacific Northwest Laboratory.

Work to be done in the future on containment and cleanup systems for the EPR includes design of secondary enclosures, further detailed examination of tritium-related accidents, selection of optimum cleanup systems, design of tritiated liquid waste disposal system, and tritium containment in blanket modules.

5. VACUUM SYSTEMS

5.1 Main Vacuum Systems

The main vacuum system consists of the pumps, piping, and valving that serve to lower the pressure appropriately prior to initiation of a discharge and maintain the proper conditions for operation throughout the discharge. This system must remove impurities that are dislodged or outgassed from the walls, fusion ash (helium), and deuterium and tritium which have left the plasma. The system may pump from a divertor exhaust chamber similar to the MATT-1050 design or directly on the main vacuum chamber, perhaps in conjunction with a gas blanket. Because the main vacuum system pumps large quantities of tritium (probably hundreds of grams per day), it is considered a tritium-handling system.

5.1.1 Summary accomplishments in FY 1975

Preliminary information was used to determine the operational requirements for the main vacuum system. Data on divertors, design and operation, and plasma confinement were used to determine the performance requirements for the vacuum system. Some safety criteria were established (Sects. 3 and 4). Various pumping techniques were examined in light of the system requirements. The feasibility of using cryosorption or mercury diffusion pumps was established, along with the preliminary sizing of vacuum ducts.

5.1.2 Range of objectives

The minimum goals for the main vacuum system are to maintain those pressures which are required in the plasma vessel and divertor channels, if any, and to limit tritium leakage to the lowest practicable amounts. The purpose of this system would be to allow successful operation of the EPR while helping to minimize doses to plant personnel and the general populace.

The maximum goals for the main vacuum system include: extreme relevance to reactor design; flexibility for testing various pumping

speeds and gas blanket concepts; minimum accident potential or consequences; reliable remote maintenance or negligible activation; very low tritium losses; rapid regeneration if cryosorption pumps are used; belch-proof design if mercury diffusion pumps are used; favorable economics; long life; high reliability; commercial availability (or nearly so); capability for scale-up to larger tokamaks; and good interface with the next stage in the fuel recycle system. A system achieving these goals would furnish proper conditions for plasma operation, including gas blanket experiments, yield experimental data and operating experience for reactor-like systems, provide maximum safety, and ensure minimum tritium costs.

At this time, it seems that the main vacuum system for the EPR could achieve most of the important maximum goals.

5.1.3 Possible solutions

Requirements. In order to achieve the goals outlined above, the system will have to meet the following requirements:

(1) Must be available in appropriately large speeds and capacities, either now or in the future, without major developments or breakthroughs.
(2) Must provide complete tritium recovery in a reasonable manner (tritium may cost as much as $7000/g, and several grams will be pumped per day).
(3) Must pump helium.
(4) Must achieve low base pressures.
(5) Must perform in radiation field.
(6) Must have relevance to future (i.e., beyond EPR) reactors.
(7) Must demonstrate reliability and maintainability.
(8) Must show reasonable economics.
(9) Must be compatible with the general requirements and physical configuration of the EPR.

Selection process. The main vacuum system will consist of piping, valves, and pumps. Since the pumps are the key components in determining the nature of a vacuum system, the initial step in the selection process in this study was to evaluate the capability of eight types of vacuum pumps to meet the requirements listed above. The results of this analysis, which are discussed below, are summarized in Table 2.

Table 2. Summary of results of evaluation of eight vacuum pumps

Type of pump	Requirement met[a]								
	1	2	3	4	5	6	7	8	9
Cryosorption	✓	✓	✓	✓	✓	✓	✓	✓	✓
Mercury diffusion	✓	✓	✓	✓	✓	✓	✓	✓	✓
Oil diffusion	✓	?	✓	✓	?	X	✓	✓	✓
Titanium gettering	✓	X	X	✓	✓	X	?	✓	X
Turbomolecular	X	?	X	✓	?	X	?	✓	?
Zirconium gettering	?	?	X	✓	✓	?	✓	✓	✓
Cryosorption panel in vessel	X	?	✓	✓	✓	X	?	✓	X
Ion	X	X	?	✓	✓	X	X	X	✓

[a] ✓ - acceptable; X - rejected; ? - applicability is in doubt.

Cryosorption pumps were found to meet all nine requirements to a reasonable degree. These pumps, which consist of an array of molecular sieves cooled by a liquid helium reservoir surrounded by liquid-nitrogen-cooled chevrons, operate by adsorbing volatile gases such as hydrogen and helium on the helium-cooled array and less volatile gases on the outer chevrons. Special-order modules with speeds on the order of 25,000 liters/sec have been manufactured. Because the pumps function by adsorption, they must be regenerated periodically by removing the liquid helium and heating. A regeneration period of several hours at 300°C is required

to drive off water vapor, but rapid regeneration at cryogenic temperatures may be possible if only helium and hydrogen isotopes have to be removed.

Mercury diffusion pumps were also found to be capable of meeting the nine requirements. These pumps use directed streams of mercury vapor to sweep the gas being pumped from the inlet side toward the exhaust. The mercury is then condensed and recycled to the boilers and is subsequently transferred in the form of vapor to the spray jets. Liquid-nitrogen-cooled baffles are required to reduce the backstreaming of mercury into the vacuum chamber. These baffled traps reduce the pumping speed by almost 50%, but the resulting speed is still acceptable, as will be discussed later. Only minor developmental work is required to produce pumps of the scale needed for an EPR. These pumps are now widely used in CTR applications. Diffusion pumps, however, also require forepumps, and the two-part pump system should be considered as a unit.

Oil diffusion pumps are essentially the same as mercury pumps, except that oil vapor is used as the pumping medium. Because of the oil's greater volatility, concern over backstreaming and contamination of the vacuum chamber is intensified. The effects of gamma, beta, and neutron radiation on the oil and its contamination by tritium were other concerns which downgraded oil pumps with respect to the mercury pumps. Forepumps are also required for oil diffusion pumps.

Titanium gettering was found to be unacceptable, principally because of difficulties with tritium recovery and helium pumping. Titanium gettering involves evaporation of the metal, which condenses as a very thin film on a surface. This film then adsorbs active gases as oxides, nitrides, hydrides, etc. Once the film is exhausted, a subsequent layer must be deposited. Because of the chemical nature of the process, helium cannot be pumped. The stability of the hydride and the many layers that would be required in a short time would make tritium recovery extremely difficult. Because of these failings and concerns over evaporation rates, required surface areas, and possible peeling problems, titanium gettering has been ruled out for an EPR and judged to have no relevance to future reactors.

Turbomolecular pumps are basically similar to rotary compressors operating at 20,000 to 30,000 rpm. The largest pumps commercially available have a speed of less than 2000 liters/sec for air. Turbomolecular pumps have difficulty in achieving low base pressures for hydrogen. It was judged highly unlikely that turbomolecular pumps could be designed to achieve the required speeds and base pressures for hydrogen isotopes with reasonable outlet pressures in an economically feasible manner.

Zirconium gettering pumps use an 85% zirconium--15% aluminum metal mixture, arranged for maximum surface area, to pump gases by forming zirconium oxides, hydrides, etc. These pumps differ from titanium gettering pumps in that the getter material is in a permanently fixed configuration and is regenerable. Questions exist concerning the economic feasibility of scaling the presently available pumps to the sizes needed for an EPR. In addition, they require regeneration at 750°C and the number of cycles before deterioration has not been established. As with all getter techniques, the zirconium pumps cannot pump helium.

A cryosorption or cryogenic panel within the vessel, or the divertor chamber itself, was ruled out because of the anticipated effects of bombardment by charged and energetic neutral particles. Also, regeneration would involve complex extraction equipment in a highly activated area, or it could use a multistep process to raise the pressure of the regeneration gas to a suitable level for the subsequent step in the fuel cycle. The latter would interfere with continuous operation of an EPR.

Ion vacuum pumps use currents of accelerated ions to propel the gas being pumped onto a titanium gettering surface. This technique yields some effective helium pumping, but at tremendous economic costs as well as increased complexity. Complete tritium recovery from these pumps would probably not be possible. It is difficult to visualize how an ion pump could be scaled up to the size required for an EPR. Difficulties are involved with the performance in a magnetic field; in addition, the lack of a reasonable tritium recovery technique remains.

System design. The preceding evaluations indicate that cryosorption and mercury diffusion pumps are the two types that are reasonably capable of meeting the requirements for an EPR. Each type would be used in a system consisting of several pumps attached to the ends of many large ducts connected either to the main vessel or to the divertor exhaust chambers. The pumps would be located outside the shield, and the ducts penetrating the shield would be bent to allow shielding against neutron and gamma streaming.

Based on the estimates of plasma confinement time and the vacuum requirements stated in the Princeton Reference Design,[6] a pumping speed of 3.1×10^5 liters/sec for DT is required. This could be obtained from 14 ducts, each 1 m in diameter by 5 m long with a 1-m-diam cryosorption pump attached. A larger number of smaller ducts and pumps could be used if physical arrangements do not allow the use of 1-m-diam ducts. The length of the pump would be about 3 m if proportions were the same as for smaller existing pumps. However, simple extrapolation from existing designs indicates that a 1-m-diam by 3-m-long pump would have a capacity on the order of 300 liters (STP). This represents only 8 hr of running time between regenerations, and changes in internal arrangement and an increase in overall size will be required to raise this to a 24-hr operating cycle, which will be more reasonable.

The major advantages of cryosorption pumps are their capability for meeting the nine requirements listed in Sect. 5.3, their simplicity of design and operation, their flexibility of orientation, and their cleanliness.

Disadvantages include the need for frequent regeneration, the requirements for liquid helium and nitrogen, and the large quantities of tritium stored in what may be considered to be the vulnerable form of a condensed gas.

Mercury diffusion pumps with appropriate liquid-nitrogen-cooled baffles perform nearly as well as cryosorption pumps with similar flange

diameters. Since they are throughput rather than adsorption-type devices, there are no concerns over regeneration cycles or the buildup of very large tritium inventories. However, they do require forepumps for efficient operation. The forepumps can be similar to the mercury ejectors proposed for the Princeton Reference Design,[6] which will increase the outlet pressure to about 100 torr. This pressure should be suitable for further processing steps.

The major advantages of mercury diffusion pumps are their capability for satisfying the nine requirements stated in Sect. 5.3, their lack of large tritium inventory, and a fairly long history of successful operation. Disadvantages include the need for forepumps, the requirement for cold baffles, the fixed orientation, and the possibility of backstreaming of mercury into the plasma vessel and the subsequent contamination this causes.

5.1.4 Analyses

Analyses of the main vacuum system were made to determine the required pumping speed and the size of the system necessary to achieve it. The pumping speed was derived from the plasma particle confinement time, the plasma density and volume, and the vacuum requirement. The plasma data indicated that approximately 10^{22} ions per second would leave the plasma. Initial scoping calculations indicated that it would be impossible to maintain a pressure of 10^{-7} to 10^{-8} torr outside the plasma without a divertor. Consequently, design calculations proceeded on the assumption that either a divertor or a gas blanket concept would be used. Although the gas blanket concept for tokamaks has not yet been developed, preliminary discussions indicated that the vacuum requirements would probably be no more rigorous than for a divertor. MATT 1050 gives the requirement for the divertor[6] as 4.5×10^{-4} torr. This led to a pumping speed of 3.1×10^{5} liters/sec, based on diatomic gas (i.e., 5×10^{21} molecules/sec). The system was then sized, based on diatomic gas of mass 5 at 300°K. It was determined that a 1-m-diam by 5-m-long duct had a conductance of approximately 59,000 liters/sec; when coupled with a 40,000-liter/sec pump, 14 ducts were required.

The analysis is sensitive to the vacuum requirements and ion confinement time. Drops in confinement time or allowable pressures much greater than a factor of 2 will make design of the system considerably more difficult.

5.1.5 Program requirements

Because the vacuum pumps may have to function in gamma and neutron radiation fields, the changes in the properties and performance of all materials due to such exposure used in candidate pumps should be determined. Possible changes in pump design or materials of construction for lower activation and greater endurance may be suggested. The first step in the program would consist of detailed calculations of the radiation flux at the pumps, based on a reasonable physical layout.

Data required from plasma physics concern the vacuum requirements for divertors and/or gas blankets and the particle confinement time. Such data should be forthcoming from future EPR studies using input on scaling laws, plasma feeding, and divertor performance from plasma experiments.

Information is needed on the long-term reliability, maintainability, and performance of large cryosorption pumps and mercury diffusion pumps. Rapid regeneration capabilities and storage capacities of cryosorption pumps should be determined.

Studies on rapid regeneration of cryosorption pumps are being done at ORNL. The commercial development of large, high-speed cryosorption and mercury diffusion pumps should be encouraged. Reliability and remote maintenance possibilities should be studied.

Work to be done in future EPR studies directly concerning the main vacuum system includes development of roughing systems and preparation of optimum physical arrangements. Work with those involved with blanket and shield design and neutronics calculations, and with manufacturers, will further clarify research and development needs.

5.2 Injector Vacuum System

The injector vacuum system consists of the piping, valves, and pumps that are necessary to maintain the appropriate vacuum in the neutral beam injector drift tubes, ion sources, and gas cells. Because a substantial amount of tritium, perhaps hundreds of thousands of curies per day, will enter the system from the main plasma vessel via the drift tubes and because the deuterium feed to the injectors may become contaminated with tritium, this system is considered a tritium-handling system. The quantity of tritium involved necessitates a reasonably high recovery of tritium from the injector vacuum systems.

5.2.1 Summary of accomplishments in FY 1975

In the scoping phase of the EPR studies, the requirements (i.e., speed and pressure) for the injector vacuum system were assessed using preliminary data as a basis. Scoping calculations were performed to determine the necessity for tritium recovery. Performance and safety criteria were developed, and preliminary conceptual designs were prepared.

5.2.2 Range of objectives

The minimum goals for the injector vacuum system are to provide the proper conditions for operation of injectors and to handle tritium safely. This would allow the neutral beams to heat and fuel the EPR plasma and to induce beam-plasma fusions while providing for the safety of plant personnel.

The maximum goals for the system are the following: maintenance of proper pressure in injectors, minimum practicable release of tritium, maximum relevance to reactor, simple scale-up to larger tokamak systems, complete tritium recovery, favorable economics, high reliability, easy maintainability, and minimum impact on beam operations. A system that could achieve these goals would improve the economics of EPR operation by reducing tritium costs and providing useful design data and operating experience for systems to be used on subsequent power reactors; it would

also allow proper beam (and hence EPR) operation as well as minimize tritium doses to plant personnel and the general public.

At present, it appears that most of the maximum goals for the injector vacuum system for an EPR can be met.

5.2.3 Possible solutions

Requirements. Based on preliminary data on injectors, a pumping speed and an ultimate pressure requirement were developed. Thus, to satisfy the above criteria, it was determined that the injector vacuum system should have the following characteristics:

(1) ultimate pressure below 10^{-4} torr,
(2) capability for pumping small amounts of helium,
(3) high speed in a system of reasonable size,
(4) continuous operation or short, infrequent regeneration periods,
(5) high degree of tritium recovery,
(6) compatibility with the physical nature of injectors,
(7) a design based on commercial or potentially commercial technology,
(8) favorable economics,
(9) high reliability,
(10) easy maintainability,
(11) a reasonable outlet pressure, but no interference with injector operation, if regeneration is required,
(12) low contamination residue, and
(13) low backstreaming.

Selection process. The injector vacuum system consists of valves, pumps, and piping. The major selection process involved determining the most desirable type of vacuum pumps. Eight types of pumps were evaluated based on their capability for satisfying the 13 requirements listed above. Table 3 summarizes the results of the evaluations, which were similar to those for the main vacuum systems.

System design. Table 3 indicates that both mercury diffusion pumps and cryosorption pumps will probably satisfy all the requirements.

Table 3. Summary of evaluation of eight types of pumps

Type of pump	Requirement met[a]												
	1	2	3	4	5	6	7	8	9	10	11	12	13
Cryosorption	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓
Oil diffusion	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	X	✓
Mercury diffusion	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓
Titanium gettering	✓	X	✓	?	X	✓	✓	✓	?	?	X	X	✓
Turbomolecular	✓	?	X	✓	✓	✓	✓	X	?	✓	✓	?	?
Zirconium gettering	✓	X	?	✓	?	✓	✓	?	?	✓	✓	?	✓
Cryosorption panel in injector	✓	✓	✓	✓	✓	X	✓	✓	✓	?	X	✓	✓
Ion	✓	?	?	✓	X	✓	✓	X	?	?	?	X	✓

[a] ✓ - acceptable; X - rejected; ? - applicability is in doubt.

The vacuum pumps would be attached to the chambers enclosing the gas cells and drift tube on each injector. Valves between the pumps and chamber would provide isolation during regeneration of the cryopumps, roughing down of the injector, or pump maintenance.

Based on the preliminary data provided, a pumping speed of 70,000 to 100,000 liters/sec is required for a 20-A, 150-keV injector. This can be attained using four or five cryosorption pumps so that one can be regenerated without having a significant effect on the pumping speed. Each pump must be able to store approximately 200 std liters for 24-hr cycling. Two or three large mercury diffusion pumps of a size and a design similar to those of currently available commercial diffusion pumps would also provide the required pumping speed.

The advantages of cryosorption pumps are their capability for meeting the requirements listed above, their cleanliness, and their flexibility of orientation. Disadvantages include the need for regeneration, requirement of liquid helium, and storage of significant quantities of tritium as a condensed gas.

Mercury diffusion pumps also meet the specified requirements, operate continuously, and do not accumulate tritium; however, they require forepumps, maintain a fixed orientation, and have the potential for contaminating vacuum chambers with mercury vapors.

Analyses. Analyses of the injector vacuum system were made to determine the required pumping speed for an injector of reasonable size and the amount of tritium that would be conducted up the drift tube of such an injector from the plasma vessel.

The computation of the required overall pumping speed requirement was based on a 3-MW, 150-keV injector with an ionization efficiency of 0.50 and a neutralization efficiency of 0.33. All ions were assumed to be D_1^+. Based on preliminary criteria, the gas cell pressure was in the millitorr range and the drift tube was kept at or below 10^{-4} torr. It was determined that a pumping speed of approximately 80,000 liters/sec was required. Approximately 900 standard liters would have to be pumped during a 24-hr period. This leads to a capacity requirement of about 200 std liters for each of five cryosorption pumps on a 24-hr cycle.

A scoping calculation of the amount of tritium that infiltrates from the plasma into the injector was based on twenty-four injectors, 40-cm-diam, 3-m drift tubes, and a DT pressure of 10^{-4} torr outside the plasma. This calculation showed that the injector vacuum system would handle on the order of 7.5×10^5 Ci of tritium per day.

The conclusions of these analyses are sensitive to changes in the data concerning the injector efficiencies, the gas cell and drift tube pressure requirements, the injector geometry, and the pressure outside the plasma.

A decrease in the ionization or neutralization efficiencies by much more than a factor of 2 would make engineering of the injector vacuum system difficult. Large increases in gas cell pressure or decreases in drift pressure would also be difficult to accommodate.

An increased drift tube length or a lower pressure outside the plasma would decrease the amount of tritium handled, but probably not sufficiently to eliminate the need for recovery. Shorter drift tubes or a

higher pressure outside the plasma would increase the amount of tritium handled; however, major increases (i.e., more than a factor of 5) in the mass flow rate up the drift tubes would be necessary before significant changes in the injector vacuum system designs would be warranted.

5.2.4 Program requirements

The actual beam requirements, both voltage and total power, can be determined only by utilizing plasma physics data. Data are also needed on the long-term reliability and performance of large cryosorption and mercury diffusion pumps. Rapid regeneration techniques and large-storage-capacity designs for cryosorption pumps should be formulated. Work on rapid regeneration is being done at ORNL. The commercial development of high-capacity cryosorption pumps and low-backstreaming mercury diffusion pumps should be encouraged.

Information needed from the beam program involves the efficiency, pressure requirements, and geometry of the injectors. The beam technology program is expected to generate such information.

Future work on the injector vacuum system will include an evaluation of roughing systems and development of vacuum systems for a more complete range of possible injector designs.

6. TRITIUM PROCESSING SYSTEMS

Tritium processing systems include equipment for removing DT mixtures from the main vacuum systems, purifying the DT mixture for recycle to feed systems, separating isotopic (H) impurities from DT mixtures, recovering tritium from blanket modules, preparing tritium-containing wastes for disposal, transferring feed, providing storage for tritium, and vent or atmosphere cleanup equipment directly associated with the process systems. Although the major vacuum pumps are considered with the main vacuum system, they are connected directly to the process equipment and, therefore, must also be considered in process studies.

6.1 Range of Objectives

The performance objectives of an EPR have not been defined completely yet; thus we are forced to look at systems with a variety of capabilities. The final selection of EPR performance criteria will involve considerations of the reliability of desired information, technological extrapolation (scale-up), and costs. These considerations must be made for individual components of the EPR as well as for the overall EPR program. The range of objectives for the EPR tritium handling system can be divided into three levels: (1) minimum acceptable, (2) intermediate, and (3) maximum reasonable. Obviously, several levels of performance are possible, and these three levels only represent relative ranges of effort. A minimum objective for the tritium system is more likely (but not necessarily) to be adopted if minimum performance is chosen for several other components of the EPR. This would correspond to a "relatively simple" EPR with minimal performance.

Minimum acceptable performance. The tritium systems with a performance level rated as "minimum acceptable" would ensure safe containment of tritium and provide only basic services for plasma experiments. No additional effort would be made to advance tritium technology for CTRs. Containment of tritium is the first objective of any tritium processing

system, and much of the atmosphere cleanup equipment, secondary containment methods, and emergency stack gas process equipment will be similar for any performance level chosen. Since this will be a significant portion of tritium process costs, differences in the total cost for the various performance levels may not be as large as initially expected; the savings would occur only in the main process equipment. The process steps involving current or near-term technology would be chosen; no special effort would be made to use "more difficult" techniques simply because they could be used in future reactors. No attempt would be made to provide a tritium breeding blanket or the necessary tritium recovery equipment.

In summary, the minimum acceptable system would only provide support for other portions of the EPR development program; it would make no extra effort to advance tritium technology. Thus additional and separate work would be required to generate tritium technology for future reactors. This approach could speed the implementation of an EPR; however, it would reduce the benefits obtainable from the EPR. The path would require additional tritium handling development and demonstration systems prior to future reactors.

Intermediate system. Perhaps a more reasonable approach would involve the selection of process steps that could be applied to future reactors. However, this approach still would not provide all of the tritium technology for future reactors because an intermediate EPR is not likely to include all components of later versions. Little or no breeding blanket experiments would be attempted, and the feed and removal systems might be significantly simplified. Although all process steps should be potentially useful for future reactors, an attempt would be made to ensure conservative selections and avoid more advanced concepts even if they might eventually prove more desirable for such reactors. Thus all EPR process techniques may not be selected for use in future reactors.

Provisions for tritium containment will not be significantly different for the intermediate and the minimum acceptable systems.

Maximum reasonable system. A tritium handling system with a performance level rated as "maximum reasonable" would include an evaluation of techniques for handling all process steps required for future reactors, including a blanket recovery system. The EPR would provide data for extrapolation of techniques to future reactor conditions. The techniques tested would be those expected to be most appropriate for the later versions; they could, of course, prove less desirable than expected. The EPR would then be the principal tritium experiment in preparation for future reactors.

As now envisioned, the EPR tritium processing system should be designed to:

(1) limit tritium release rates to acceptably low levels and minimize the probability and consequence of accidental releases,
(2) contain acceptable inventories of tritium,
(3) involve reasonable (hopefully, small) costs,
(4) be compatible with all other portions of the EPR, and
(5) use current technology or technology which can be expected to be developed early enough for use in the EPR.

The first goals of the tritium handling system will be to provide acceptable containment and to minimize the hazard of personnel exposure and/or machine downtime due to tritium releases. Excessive release of tritium is not only a hazard to personnel, but can also contaminate expensive equipment and jeopardize the mission of the EPR. The desired degree of containment is discussed in Sects. 3 and 4. Secondary containment of all process equipment is anticipated, and emergency tritium removal equipment will be installed in the reactor containment building.

Tritium inventories must be limited because (1) they represent a significant cost to the system, (2) high inventories probably imply a higher maximum environmental risk in the event that a major release occurs, and (3) tritium supplies available to the EPR will be limited. A 150-MW(t) EPR would burn 20 g of tritium per day (200,000 Ci/day). If the fraction burned is 1%, this suggests a tritium process rate of 2 kg, or 20 MCi, per day. If a 2-day inventory is maintained throughout the

process and storage systems, this would correspond to an inventory of 4 kg, or 40 MCi. At this point, one cannot make an accurate estimate of the tritium inventory required for an EPR. Different process techniques will require different inventories; also, most of the process steps have not been chosen. The cost of tritium to the EPR is uncertain. Present ERDA policy would make tritium available to ERDA programs such as the EPR at no significant cost, provided the tritium could be supplied by existing facilities without hindering other programs. The quantities of tritium required for the EPR are expected to be available under these terms. If it became necessary to charge the EPR program for the tritium supply, the cost would be significant. If additional production capacities were required for the EPR, the costs would become considerable.

Although the potential hazard is often considered to increase as tritium inventories increase, one should be careful about drawing such simple conclusions. Inventories can vary, depending on the process techniques used, but the difference in the probability of a release may be more important than differences in the inventories.

Fusion power in general, and the EPR in particular, will require considerable advances in several areas of technology. It is necessary to choose existing or near-term techniques, when they are adequate, to increase the probability of overall EPR success to an acceptable level. This situation applies to all portions of the EPR system. The economics or costs of tritium handling systems do not impose sharp clearly defined limits, but no process system that significantly increases the overall EPR cost should be selected.

The principal tritium process steps required for the EPR (or any fusion reactor) are shown in Fig. 1. Vacuum pumps play an obvious and important part in the removal of tritium, deuterium, and impurities from both the plasma chamber (or divertor) and the injectors. Although vacuum pumps must be considered as part of the tritium processing system, they are more conveniently treated as part of the vacuum systems. The size and pumping characteristics of the vacuum pumps are discussed as part of the vacuum system in Sect. 5. Recovery of D-T from the pumps is discussed as part of the tritium processing system. Cryosorption pumps permit

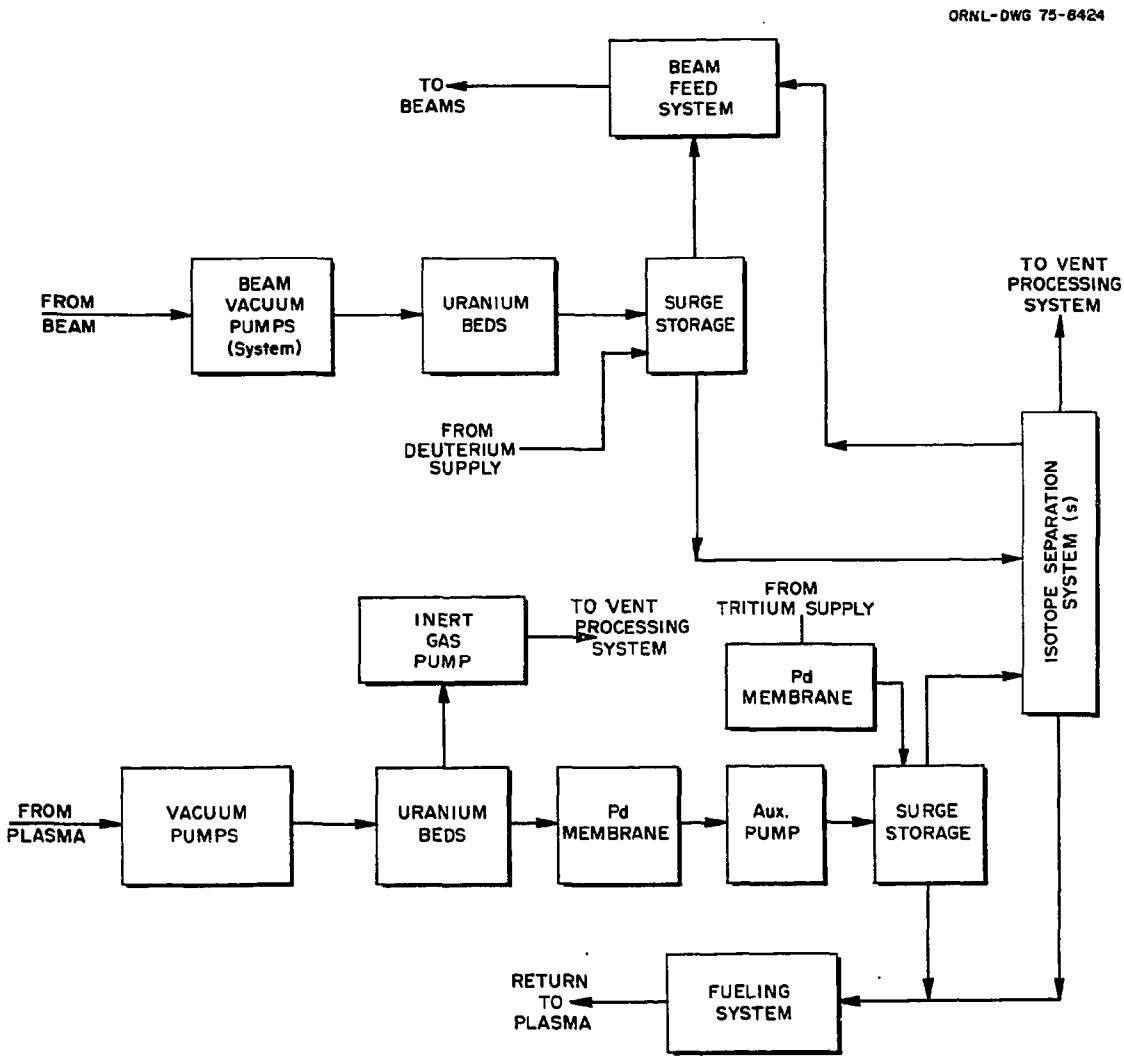


Fig. 1. Schematic view of the reference EPR tritium system.

complete regeneration of tritium and have no shaft seals to permit leakage to or from the system. They also introduce no impurities, such as lubricants or working fluids, to the system. Thus cryosorption pumps must be considered for the EPR. Mercury diffusion pumps appear to be the most likely competitor to cryosorption pumps; however, mechanical pumps and getter pumps were also investigated.

Two disadvantages of cryosorption pumps are their cyclic operation and their limited capacity for holding D-T. The cyclic nature of these pumps increases the potential tritium inventory; the pumps would have to be regenerated frequently, probably daily. The degree of disadvantage this represents depends on the other process steps. If some of these steps are cyclic with the same period, the steps may mesh very well. The most likely purification technique, sorption on uranium beds, is a cyclic process, and thus coordinates satisfactorily with cryosorption pumping. The principal disadvantage of mercury diffusion pumps is the possibility of introducing mercury into the plasma system or downstream process equipment. Purification systems would probably be damaged by mercury accumulations. Cold traps would drastically reduce the chance of steady mercury contamination of other equipment, but one must still consider the probability of a slow accumulation of mercury and the possibility of severe contamination in the event of equipment malfunction.

Some form of protection may be needed to protect cryosorption pumps from sputtered materials. This protection may be provided by the vacuum duct walls. If necessary, cold baffles could be inserted in the duct for further protection. If a divertor is used, the divertor channels themselves may serve this purpose. Small quantities of material reaching a cryosorption or mercury diffusion pump would probably have little effect on the pump. The sputtered material would coat the outer chevrons which are cooled to liquid nitrogen temperatures. Very little material would be expected to reach the molecular sieve or mercury. Small quantities of material sticking to the outer chevrons would be expected to have little effect. Larger quantities could damage the chevrons or even

significantly affect the liquid nitrogen heat load. The need for additional protection for the pump system can be estimated more accurately when more reliable data on EPR particle fluxes at the wall and sputtering yields become available.

Once tritium is recovered from the vacuum pump, it must be purified to remove chemical impurities such as helium, oxygen, and nitrogen. Helium, which is formed from the D-T reaction in the plasma, may be of the order of 1% of the removed gases. Other impurities may result from leakage or outgassing of the equipment. The purification technique most likely to be employed for the EPR uses uranium beds. These beds will sorb essentially all likely impurities except the noble gases.[7,8] When heated, they release the hydrogen isotopes but retain the impurities. Uranium beds obviously have limited lifetimes determined by the impurity content of the process streams. The life of uranium in the EPR, however, is expected to be long (years). Noble gases swept into the bed can be pumped off; and if necessary, additional purification can be achieved by routing the uranium bed outlet stream through a palladium membrane. Uranium beds, which are frequently used for tritium storage and purification, are considered current technology. Although other purification methods also look promising, the existing experience with uranium beds gives them an edge. An alternative method will have to promise significant advantages in order to be preferred for the EPR. The cyclic nature of uranium beds couples well with cryosorption pumping.

Other hydride-forming metals besides uranium will need to be considered. Since different metals have different affinities for hydrogen (isotopes) as well as oxygen and nitrogen, a better combination of properties may be found. However, the existence of a more suitable material is not now apparent. Uranium beds are also proposed for storing tritium. Although it is possible that other hydride-forming metals or alloys may be selected for the storage function, uranium will probably be used for purification.

Palladium membranes have frequently been used for purifying hydrogen. (The application of palladium for final purification of tritium from uranium beds has already been mentioned.) The membranes could be selected

as the sole purification method, but the principal disadvantage of this approach is that tritium compressors would be required. Mechanical compressors are likely to require significant maintenance and perhaps allow small leaks to or from the system (to or from secondary containment atmosphere). Although palladium membrances alone can readily produce a high-purity hydrogen isotope stream, the production of a reject stream containing little or no hydrogen isotopes without larger surfaces and/or high total pressures is difficult.

It is possible to combine the purification operation with the isotope separation steps. The advantages of this operation obviously depend upon which isotope separation technique is chosen. Cryogenic distillation, one of the most promising isotope separation techniques, could separate helium as well as other impurities likely to be present. The merits of this approach, however, also depend on the throughput capacity required for the isotope separation system. All material recovered from the plasma will have to be purified before it is returned to the plasma. However, all material will not have to undergo isotope separation. The material used to fill the torus before each run and the material feed in the form of gas or solid pellets (probably) will not require isotope separation. Only D (and possibly T) supplied through injectors need undergo isotope separation, and a separate purification step is likely to be chosen.

Isotope separation will probably be required at the EPR site to prepare "tritium-free" deuterium for the injectors and to remove the trace of hydrogen produced by D-D reactions and outgassing from the vacuum vessels and equipment. Although these operations could be carried out in a single column or cascade, separate systems are likely to be selected because the required throughputs will be different. It is even possible that different separation techniques will be chosen. The first system (DT separation) will be the larger. A specific design for the EPR cannot be prepared at this time because the purity requirements for the injectors and the plasma feed have not been estimated. Hydrogen isotope separations have been performed in equipment of reasonably large size, and the data and experience gained from these separations can be

expected to be used in the EPR. However, impurity requirements should be specified as soon as possible. Then specific designs can be made to determine whether, as expected, the isotope requirements of EPR can be met at modest cost with current technology. Although we expect no unusually large expenditures for isotope separation equipment, one could specify plasma and feed concentrations which would be difficult to meet.

Satisfactory storage of tritium either as a gas or as a solid tritide (usually uranium tritide) has been demonstrated in other ERDA programs. This is current technology, and either form should be considered "safe." Because solid storage is often considered less hazardous, it will probably be used for most large-scale and long-term storage. Small-scale and especially short-term (surge) storage will be as low-pressure gas.

The blanket recovery system is one of the least well understood areas in the EPR tritium handling system. Detailed evaluations of all of the blanket recovery systems considered for the EPR or later fusion power reactors will not be presented here since a separate report is being prepared on the subject.[9] Considerable work will be required to select and develop a practical and reliable recovery system. Lithium is currently considered the most likely blanket material; therefore most of the major efforts should involve the study of recovery of tritium from lithium. A small side stream of lithium will have to be removed for processing. We are currently considering three processes for recovering tritium from lithium: extraction,[10] probably with a molten salt; permeation; and sorption on hydride-forming metals. The present experimental effort is deficient in all three areas. A significant study of extraction at ORNL is proposed. The work will focus on recovery of tritium from the salt (after extraction) because this appears to be the most critical step of the process. Alternative techniques have also been proposed. A smaller effort on recovery by permeation is needed, while current studies on sorption processes should be continued. Expanded studies of sorption processes will be recommended when (or if) sorbers with more practical sorption rates are found.

The feed system has not been extensively evaluated in the EPR tritium study because the plasma physics aspects of the feed system have not been

clearly defined. If subsequent developments indicate that solid pellet feed should be used in the EPR, relatively more complex tritium handling equipment would be involved. This is another area where expanded tritium studies will be recommended.

6.2 Information Required for Evaluating Tritium Processing Systems for the EPR

Plasma physics area. Many aspects of the tritium process system will have to be reevaluated as requirements and performance of the plasma system become better understood. Purity requirements must be known for all feed streams: fill, supplementary feed (solid?), and injector feed. How much H_2, He, and higher-Z impurities can be tolerated in these streams? How closely must the DT ratio be maintained? What is the maximum permissible T concentration in the injector feed? To a lesser extent, the impurity concentrations in all exit streams will be important. We will be particularly interested to find out whether massive quantities of high-Z sputter products affect pump performance.

Several options in the EPR design could affect tritium handling equipment. The probability of adopting these techniques and their performance characteristics should be evaluated. The selection of feed systems is obviously important to the tritium handling program. Solid-feed equipment will involve different techniques and containment systems. Although no divertor is currently proposed for the EPR, some potential divertor designs would affect tritium removal techniques. Diagnostic devices can also affect tritium containment. Any opening to the plasma chamber must be considered in tritium containment studies. A graphite liner facing the plasma could lead to tritium retention and result in different sputter products and outgassing rates (and composition).

Fundamental data required. The fundamental chemical data required are, for the most part, associated with blanket recovery processes. Sputter products and tritium retention on the graphite liner (if adopted) could become more important as studies proceed.

Tritium technology data required. Considerable data are needed on the recovery of tritium from vacuum pumping systems. Cryosorption pumps

or mercury diffusion pumps are currently believed to be the most suitable choice for the EPR, but alternative systems should also be considered. Cryosorption pumping is currently under study at ORNL. Particular interest is centered on regeneration at low temperatures and pump sorption capacity for hydrogen isotopes and helium at various pressures and pumping rates. If mercury diffusion continues to be a promising alternative system, study of these pumps should be initiated.

The recovery of tritium from lithium blankets is the area of most immediate interest. This is perhaps the most difficult developmental effort required for tritium handling in the EPR. However, as mentioned earlier, a minimal EPR device would not include provisions for blanket experiments. It would also be possible to use a less-efficient recovery system for the EPR. This would allow breeding and material studies in some blanket modules, but it would be less effective in advancing tritium handling technology. Recovery processes for 500°C lithium blankets are limited by inventory considerations; that is, the maximum allowable tritium concentration is restricted by the allowable tritium inventory in the blanket. If only a small portion of the EPR blanket region is devoted to breeding experiments, much higher concentrations could be permitted before the inventory becomes excessive. Since heat from this module does not necessarily need to be discharged directly to a stream, tritium release rates may also be controlled even when the tritium concentrations are relatively high. Thus it may be possible to operate the recovery system at concentrations significantly higher than those required for future reactors. However, one would prefer to simulate future reactors as effectively as possible and operate the EPR blanket at comparable concentrations.

Isotope separation. Existing engineering data should be more fully evaluated before new experimental work in this area is initiated. As soon as information on allowable DT rates, permissible H concentrations, and injector feed rates are estimated, one or more conceptual designs of an isotope separation system should be prepared. This will clarify the magnitude of the isotope separation problem. Can the proposed separations be made at minor or acceptable cost? Should the purity requirements

be reexamined? Are the estimating data and experience on isotope separation adequate for EPR needs?

Existing related programs. Most of the experimental development required for tritium handling in the EPR is also needed for demonstration reactors, other CTR devices, and future power reactors. These efforts need not be funded solely or even principally as EPR programs. The general-purpose goals can be achieved by continuing and expanding existing tritium handling studies. Specific design and eventual demonstrations, however, should be considered a part of the EPR program. Development of tritium technology for the EPR, demonstration reactors, and other CTR devices has been the goal of the DCTR-funded Tritium Handling Studies in the ORNL Chemical Technology Division. This program is directed at solving some of the problems just discussed. It is, however, only a modest-size effort which will need to be expanded during the coming years to meet all the probable needs of the EPR. This program is supported by fundamental studies, funded by the ERDA Division of Physical Research, that are being carried out in the ORNL Chemical Technology and Chemistry Divisions. All of these programs benefit from association and consultation with related studies in the ORNL Thermonuclear and Metals and Ceramics Divisions.

6.3 Recommendations

All of the tritium studies in progress at ORNL have been outlined and directed in accordance with our best judgments of the needs of the EPR and future reactors; thus far, no significant redirection of these programs has been indicated. Significant expansion, however, will be required during the next few years to meet the needs of the EPR. This expansion will need to occur largely in the area of DCTR development effort. Fundamental efforts funded by the ERDA Division of Physical Research need to be continued; in addition, some expansion (beyond inflation) would be desirable.

Specific expanded efforts for FY 1976 and 1977 include:

(1) An increased effort to experimentally evaluate the alternative methods for recovering tritium from lithium.

(Studies of tritium recovery from other proposed blanket materials should be continued, but no major expansion is recommended unless the most probable blanket choice for EPR is changed.)

(2) Studies of tritium recovery from vacuum pumps should be expanded. Additional pumping systems (especially mercury diffusion) should be considered.

(3) ORNL personnel should become involved in the design of the tritium handling system for the TFTR. Although the tritium systems for the TFTR and the EPR will probably be quite different, useful interchange of information and experience can significantly aid both programs. Such involvement could include direct participation in most of the design efforts or only consultation and review. More benefit, however, will accrue from active participation.

Specific efforts during FY 1976 relative to the preliminary conceptual design of the EPR should include:

(1) Preparation of an integrated layout of the tritium system. This layout will have to be modified several times in the future as information becomes available from elsewhere in the EPR studies. However, such a layout is necessary to ensure that several minor items are not ignored in more-detailed studies made subsequently.

(2) Investigation of the needs of the isotope separation system. The size and cost of isotope separation will be estimated using the best current estimates of feed rates and allowable isotope concentration ranges in all feed streams.

(3) Consideration of alternative methods for recovering tritium from lithium blankets.

These efforts, along with existing experimental programs, will attempt to narrow the choice of alternative recovery techniques.

(4) Initiation of studies of tritium handling in solid DT feed systems for the EPR.

7. REFERENCES

1. International Commission on Radiological Protection Publication 9 (September 1965).

2. International Commission on Radiological Protection Publication 39 (January 1971).

3. Code of Federal Regulations, Title 10, Part 50, Appendix I (Mar. 4, 1974).

4. AEC Safety Guide 25, Quality Group Classifications and Standards (March 1972).

5. NRC Regulatory Guide 129 (1975).

6. R. G. Mills (ed.), A Fission Power Plant, MATT-1050 (1975).

7. R. S. Carlson, Mound Laboratory, Miamisburg, Ohio, personal communication, 1974.

8. G. L. Powell and S. B. Condon, Hydrogen in Uranium Alloys, Y-DA-5317 (1973).

9. J. S. Watson, S. D. Clinton, P. W. Fisher, and J. B. Talbot, "Evaluation of Several Liquid and Solid Tritium Breeding Blankets," Proceedings of the International Conference on Radiation Effects and Tritium Technology for Fusion Reactors, Held at Gatlinburg, Tennessee, October 1-3, 1975.

10. V. A. Maroni, R. D. Wolson, and G. E. Staahl, "Some Preliminary Considerations of a Molten Salt Extraction Process to Remove Tritium from Liquid Lithium Fusion Reactor Blankets," Proceedings of the First Topical Meeting on the Technology of Controlled Nuclear Fusion, Held at San Diego, California, April 16-18, 1974, CONF-740402.