Tritium Source Terms for Fusion Power Plants

by
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September 1976

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PREFACE

Fusion reactor technology has developed far enough to expect laboratory demonstration of practical levels of fusion employing the D-T reaction to occur in the early 1980s. Following that demonstration, and depending upon the national priorities for energy from D-T fusion, construction and operation of experimental reactors and demonstration power reactors could occur before the end of this century. Operation of the first commercial power plants could then follow, starting about 2010.

Development and adoption of a new power system eventually will require a description of the environmental effects in an environmental statement providing a comparison to the effects of competitive systems. In anticipation of that statement, an environmental analysis (BNWL-2010) has been prepared for the ERDA Division of Magnetic Fusion Energy. That analysis estimates the environmental effects of constructing and operating D-T fusion reactors as an economically competitive source of electricity in the 21st century.

The analysis has four primary purposes:

1. To describe the general nature of the environmental effects,
2. To determine current ability to estimate the effects,
3. To determine methods for reducing the effects, and
4. To determine research necessary for increasing capability to define and reduce the effects.

Timely identification of needed research and methods for reducing effects will permit the performance of that research and the revision of conceptual fusion power plant designs before preparation of the program environmental statement. This would improve the quality of the environmental statements and could reduce the estimated adverse environmental effects due to fusion power plants.

The environmental analysis (BNWL-2010) concludes that the following assumed characteristics are the best set for the first operating fusion power plants:

- The D-T fusion reaction
- Large quantities of activation products
- Kilogram quantities of tritium in the plant systems
- Massive reactor structures
- Large lithium inventories
- Large inventories of liquid metals and salts
- Standard electricity generation
- Standard radioactive waste systems
- Large magnetic fields
- A self-contained fuel cycle
- Rural siting

Using these characteristics a reference reactor was analyzed to determine the environmental effects by using available concepts of plant subsystems designs that control interactions with the environment or by assumption that best current technology would be used in subsystems design.

Because this analysis does not take into account advances in both fusion and waste control technology during the next thirty years, the estimated effects probably are significantly higher than the actual effects will be for the first fusion power plants. The estimated environmental effects should be interpreted only as being the probable upper limit for the actual effects.

Preparation of the fusion power plant environmental analysis required development and use of specially developed data and analysis methods not used in the preparation of current environmental statements for fossil and fission power plants. These data and analysis requirements are documented in a series of reference topical reports to make this information publicly available and to assure understanding of the basis for the conclusions made in the environmental analysis.

These reference topical reports summarize the state-of-the-art as applicable to preparation of environmental statements for fusion power plants. They present the data and analytical techniques used in the environmental analysis to estimate the interactions with the environment and the resultant environmental effects. This information then was analyzed for adequacy and the need was determined for additional research to assure satisfactory ability to prepare environmental statements for the fusion development program and experimental facilities in the early 1980s. Estimated environmental effects are presented in these reference documents only as necessary to illustrate use of the data and analytical techniques.

This report is one of those reference documents for the environmental analysis. The other documents in this series contain more details of the power plant concepts and the probable environmental effects of fusion power plants with the assumed characteristics listed above. These documents are available through the National Technical Information Service:

An Environmental Analysis of Fusion Power to Determine Related R&D Needs, BNWL-2010
Review of Fusion Research Program: Historical Summary and Program Projections, BNWL-2011
Fuel Procurement for First Generation Fusion Power Plants, BNWL-2012
Current Fusion Power Plant Design Concepts, BNWL-2013
Reference Commercial Fusion Power Plants, BNWL-2014
Siting Commercial Fusion Power Plants, BNWL-2015
Materials Availability for Fusion Power Plant Construction, BNWL-2016
Projected Thermodynamic Efficiencies of Fusion Power Plants, BNWL-2017
Tritium Source Terms for Fusion Power Plants, BNWL-2018
Management of Nontritium Radioactive Wastes from Fusion Power Plants, BNWL-2019
Methodology for Estimating Radiation Doses Due to Tritium and Radiocarbon Releases, BNWL-2020
Magnetic Field Considerations in Fusion Power Plant Environs, BNWL-2021
Biological Effects of Tritium Releases from Fusion Power Plants, BNWL-2022
Biological Effects of Activation Products and Other Chemicals Released from Fusion Power Plants, BNWL-2023
Safety Review of Conceptual Fusion Power Plants, BNWL-2024
An Investigation of the Transportation Requirements of Fusion Power Plants, BNWL-2025
Considerations of the Social Impact of Fusion Power, BNWL-2026
Environmental Impacts of Nonfusion Power Systems, BNWL-2027
Environmental Cost/Benefit Analysis for Fusion Power Plants, BNWL-2028
Biomagnetic Effects: A Consideration in Fusion Reactor Development, BNWL-1973
An Analysis of Tritium Releases to the Atmosphere by a CTR, BNWL-1938

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TRITIUM SOURCE TERMS FOR FUSION POWER PLANTS

1.0 INTRODUCTION

The radioactive hydrogen isotope, tritium, is a probable fuel for early-generation fusion reactors because it participates with deuterium in a fusion reaction which appears to be more readily ignited when compared to other candidate fusion reactions. Eventually it may be replaced by nonradioactive reactants. However, the early fusion power systems will employ tritium. Therefore, we must plan for effective tritium utilization on a technology basis which has acceptable environmental impacts.

In the high-temperature regimes where fusion reactors are expected to operate, tritium tends to be pervasive. With tritium inventories estimated at up to ~25 kg (25×10^7 Ci) for a 5000 Mwt plant,⁽¹⁾ it is inevitable that some tritium releases to the environment will occur. The need to minimize these releases will influence fusion reactor design and operation. Fortunately, substantial technology already exists to produce and control tritium. However, advances in tritium control will be required which are specific to the needs of fusion technology. Early detailed assessment of tritium control requirements is essential to identify priorities for development of tritium control technology for fusion. This development must consider tritium impacts to the public and to the plant personnel.

This report presents the results of a study which generated preliminary estimates of the tritium source terms for several fusion power reactor reference designs. These reference or conceptual designs were analyzed to calculate or estimate the amounts and locations of tritium releases within the plants, and the pathways of tritium escape to the environment. Only steady-state plant operation was addressed in this study. Tritium releases associated with accidents are referenced in a separate publication.⁽²⁾

Where possible, the tritium source terms and releases were calculated from descriptions of plant equipment and processes. Frequently, however,

the reference designs did not contain sufficient detail to permit rigorous calculations. In those cases, the tritium source terms and releases were estimated from details presented in designs for similar plants, or from the experience base of the fission power industry and tritium production facilities.

The tritium releases and source terms identified in this study are based on first-generation fusion reactor conceptual designs which provide a useful, but preliminary perception of an operating fusion reactor. However, a major objective of this work was to identify areas of tritium control and handling technology which will require further development as fusion reactor technology evolves. The analysis was structured to identify tritium control inadequacies in the reference designs and in the available data base, and to identify available technology which can be assimilated into fusion reactor designs.

2.0 SUMMARY AND CONCLUSIONS

Because tritium is radioactive and potentially pervasive, fusion reactor technology based on deuterium-tritium fuels must develop tritium control procedures which will assure protection of the public and of plant personnel.

The purpose of this study is to assess tritium control aspects of fusion reactors, to derive a preliminary estimate of the probable tritium releases and source terms^(a) under steady-state reactor operation, and to identify areas where advancement of fusion reactor tritium control technology will be required.

Tritium release aspects of six fusion reactor conceptual designs were evaluated, including four Tokamaks, one mirror and one theta pinch reactor. The principal effort was directed to Tokamaks because they represent the current main-line Division of Magnetic Fusion Energy (DMFE) effort. Two Tokamaks which had the most detailed conceptual designs at the time of the analysis (PPPL and UWMAK-I) were selected for detailed analysis to define tritium source terms.

Methodology for estimating tritium releases first required detailed identification of the reactor components where the releases can occur. The potential pathways to the environment then were identified and the tritium source terms were quantified. The procedures were first applied to the plant as designed. However, the designs sometimes did not have sufficient detail for a given component to permit calculations. Also, the designs frequently did not consider technology advances which can be expected to reduce releases and source terms. The analysis therefore compares releases and source terms as a function of the tritium control options which are demonstrated on commercial or laboratory scales, or are promising for further development.

The principal primary tritium release mechanisms from fusion reactor components are (not necessarily in order of importance):

- accidents;
- major and minor maintenance;

(a) The terminology used in this report identifies tritium which escapes from primary containment (pipes, vessels, etc.) to reactor containment as tritium releases. Tritium emissions which cross the reactor site boundary to the environment are identified as tritium source terms.

- leakage from fluid system connections, valves and pumps;
- permeation through pipes and vessel walls.

Tritium releases as a result of accidents were not addressed in this study because they are included in a separate document.⁽²⁾ Tritium releases from major maintenance were not studied in detail because of a lack of information on the actual maintenance requirements for the power plants evaluated. However, incorporation of multiple containment boundaries in fusion plant designs will greatly minimize tritium source terms from these release mechanisms, because of provisions for processing and recovery of the released tritium from containment atmospheres. A rough analysis suggests that source terms from major maintenance of systems such as PPPL or UWMAK-I would probably not exceed a range of 100 to 1000 Ci/yr if secondary and tertiary containment are designed into the plant.

Tritium releases during normal operations which occur as a result of minor maintenance, leakage from fluid systems, and permeation through primary containments were the major focus of this study. Table 2-1 summarizes the tritium source terms for normal operations based on PPPL and UWMAK-I conceptual designs. The fusion source terms in Table 2-1 assume a conservative approach to tritium permeation pressure dependence and take credit for oxide barriers on steam generator piping. They assume 90 percent recovery of tritium released to containment. Tritium source terms from the PPPL design were low due to specification of double jacketing on tritium blanket piping and vessels. This may be difficult to engineer for a high-temperature nuclear system, suggesting that the UWMAK-I source terms may be more credible, based on foreseeable technology.

Other tritium source term data are compared in Section 5.3 for scenarios which evaluate barrier effects, tertiary containment, and first power tritium pressure dependence.

Two major uncertainties in the physical and chemical behavior of tritium exist which significantly impact calculation of tritium permeation in fusion design. The first uncertainty is the lack of data on physical

TABLE 2-1. Tritium Source Terms During Normal Operations^(a,b)

	(Ci/GWe·y)
UWMAK-1 ^(c)	
Gaseous	800
Liquid	10
PPPL ^(d)	
Gaseous	180
Liquid	None

- (a) Source terms do not include tritium releases from storage. Estimates from ERDA facilities suggest routine storage source terms in the range of 40 to 600 Ci/GWe.y.
- (b) Source terms do not include tritium releases resulting from major maintenance, roughly estimated to be in the range of 100 to 1000 Ci/yr.
- (c) Tritiated water from steam released partially through a wet cooling tower and partially to natural streams.
- (d) Tritiated water from steam system released entirely through stack. Double-jacketed piping and equipment assumed to intercept tritium permeating from lithium-beryllium fluoride blanket.

and chemical behavior of tritium in low pressure regimes down to 10^{-10} to 10^{-12} mm Hg, where extrapolations indicate that some fusion systems may operate. It appears that tritium permeation may be significantly reduced at these low pressures relative to permeation rates extrapolated from higher pressures. More information is needed before the effect of this phenomenon can be accurately determined. Secondly, the effect of oxide film barriers on tritium permeation, particularly through steam generators, is currently under study. Laboratory results underway at ORNL suggest that oxide films on steam generators will reduce tritium permeation substantially, by a factor of 100 or more over permeation rates through clean metal. Effects of this information on fusion reactor

tritium releases are included in this analysis and significantly affect the conclusions. However, additional work will be required to further define oxide barrier characteristics as a function of time, material, and operating conditions.

Several conclusions were reached about tritium release mechanisms. Tritium leakage must be minimized, which will require advances in valve and closure technology as vital for fusion reactor tritium control. As mentioned earlier, tritium permeation to the steam system is a problem not easily resolved because of the cost of tritium separation after the tritium has exchanged with protium to form HTO. However, the importance of tritium permeation is not yet clear. Tritium removal from contaminated cover gas atmospheres is a major aspect of tritium control. This will require improvement and scale-up of existing tritium removal equipment, and modular design of free space in the reactor hall and component rooms. An important aspect of cover gas tritium control is to minimize the conversion of elemental tritium to the oxide form, due to the lower Regulation Control Guide (RCG) value for the elemental form.

Analysis of the various fusion designs and the tritium characteristics of the designs pointed out several significant conclusions:

- In the UWMAK-1 design more tritium accumulates as waste than is routinely released to the environment; therefore, careful attention must be paid to the generation and disposal of tritiated waste from fusion reactors.
- The matter of tritium fuel storage also has a significant impact on the overall tritium distribution in a fusion reactor.
- Raising coolant temperatures to increase thermal efficiency is a desirable design parameter. However, this has a negative effect on tritium releases. Thus, some difficult design challenges can be anticipated for the fusion plant designer wherein tradeoffs between tritium control and other plant operating performance are involved.

As pointed out above, the detailed analysis of tritium release was performed for the Tokamak designs where the most detailed information

was available. General comparisons were made between mirror and Tokamak fusion reactors. It was concluded that mirror fusion reactors would probably exhibit tritium source terms to the environment approximately equal to those of the Tokamak designs. The mirror reactor may generate more tritiated waste inside containment than Tokamak reactors. The theta pinch fusion reactor appears to differ significantly from Tokamaks in the very large containment volumes which must be purged or evacuated for effective tritium control. This would mean that tritium control for the theta pinch design may be more costly than for Tokamaks but principally only because of plant size. The theta pinch and the mirror fusion reactor both share the problem of tritium control problems associated with large numbers of vacuum pumps. However, if cryosorption pumping develops on a large scale, problems associated with tritium-contaminated oil and/or mercury may largely disappear.

It is impossible to project the probable radiation emission rules into the 21st century, which is the earliest that fusion is expected to be commercialized. Therefore, it is not now possible to establish the maximum tritium releases which are likely to be allowable from fusion reactors. The low relative whole body dose which tritium now contributes suggests that reasonable tritium releases from fusion will not significantly increase radiation doses to the world population. On the other hand, as-low-as-reasonably-attainable guidelines suggest that fusion must minimize further tritium additions to the biosphere. Consequently, a substantial investment in tritium control technology for fusion reactors is essential. That technology must develop on several fronts:

- improved definition of fusion reactor designs;
- assimilation of existing tritium control technology;
- advances in basic tritium data specific to fusion reactor regimes;
- advances in engineering technology specific to fusion reactor tritium control requirements;
- development of plant models to optimize tradeoffs between tritium control and plant operation.

Recommendations regarding each aspect of the control technology are summarized in Section 6.0.

3.0 TRITIUM ENVIRONMENTAL PERSPECTIVES

Tritium has a half-life of 12.26 years, and decays to less than 1 percent of its original activity in about 85 years. It emits a weak beta ($E_{\max} = 18.6$ keV) with no associated penetrating radiation (gamma or x-ray). Therefore, no extra shielding is required beyond normal piping or vessel walls. It is absorbed by biological systems, but the biological half-lives are relatively short: less than 1 day for T_2 and about 8 days for T_2O .^(3,4)

3.1 GLOBAL TRITIUM INVENTORY

As we consider the probable tritium environmental impacts from fusion, it is of interest to relate them to the current and projected tritium inventories in the environment. The worldwide tritium inventory in air and surface waters arises from the following sources:⁽⁵⁾

- nuclear weapons testing in the atmosphere - the largest single source, peaking at $\sim 2 \times 10^9$ Ci in 1962; with no further additions, is expected to decay to the level of natural production early in the 21st century;
- natural production by cosmic ray interactions with oxygen and nitrogen; the steady-state inventory is estimated to be $\sim 10^8$ Ci;
- nuclear fission power, expected to exceed the natural tritium production rate by ~ 1985 to 1990.

Releases from tritium production facilities in several countries do not appear to have been tabulated, but probably are on the order of $\sim 2 \times 10^7$ Ci (cumulative).

3.2 RADIATION DOSE COMPARISONS

Table 3-1 compares radiation doses to the human population from natural, fallout and fission sources. The comparisons emphasize the minor tritium contribution to the whole-body dose in the general population. The estimated average whole-body dose from tritium in the year 2000 is 0.03 mrem,⁽⁵⁾ which is 0.02 percent of the natural radiation dose. On

TABLE 3-1. Whole-Body Radiation Doses^(a)

Basis: Average Person - U.S. 1974

Source	Total Dose mrem	Tritium Dose ^(b) mrem
Natural ^(c)	130	0.004
Fallout	4	0.06
Nuclear Power	2×10^{-4}	---

(a) Values adapted from References 5 and 6.

(b) Total est., all tritium sources in 2000 AD = 0.03 mrem; lower than 1974, due to decay of fallout T₂.

(c) Terrestrial, cosmic, etc.

the other hand, in some fission reactors, tritium contributes a substantial fraction of the total radiation dose to plant personnel. For example, in Canadian heavy water reactors (HWRs), tritium concentrations in the heavy water inventory may reach 10 Ci/kg.⁽⁷⁾ The tritium contributions to total man-rem exposures (tritium plus gamma) for three Canadian HWRs are summarized in Table 3-2, arising from tritium-contaminated water leaking into work locations. Tritium concentrations in fusion reactor power circuits are not likely to approach 10 Ci/kg; however, fusion reactors will have more locations where plant personnel may be subjected to tritium than fission reactors do.

Tritium releases to the public have been the principal concern to date for those assessing fusion reactor impacts. The data in Table 3-2 suggest that tritium doses to plant forces will also be an important consideration.

It is not clear what radiation dose limits will apply to nuclear facilities when fusion reactors begin operation. Current rules specify a limit of 0.5 rem/y, total body at the site boundary. However, a proposed EPA standard aims to lower the limit to 0.025 rem/y from all nuclear fuel cycle facilities.⁽⁸⁾

TABLE 3-2. Tritium Doses as Percent of Total Occupational Dose at Canadian HWRs(7)

<u>Plant</u>	<u>Years</u>	<u>Tritium (Internal)</u> <u>% of Total Annual Radiation Dose</u>	
		<u>Average</u>	<u>Range</u>
NPD	1962-74	31	12-57
Douglas Point	1967-74	20	7-30
Pickering	1971-74	26	18-30

Note: Total dose limits are 5 rem/man.y and cannot exceed 3 rem/man.y for the station average. The Pickering Station average exposure (1974) was 0.79 man-rem/MWe, installed (4 x 500 MWe = 2000 MWe), compared to ~1 man-rem/MWe, installed for U.S. PWRs.

4.0 ANALYSIS OF TRITIUM CONTAINMENT IN FUSION REACTORS

A 5000 MWe fusion reactor will burn ~0.7 kg of tritium per day, have a tritium inventory of up to ~25 kg, and will circulate tritium at rates 20 to 30 times the consumption rate.⁽⁹⁾ Figure 4-1 is a simplified flow diagram for tritium in a plant with a mobile breeding blanket. The figure illustrates the following major aspects of tritium flow:

- fuel preparation and injection
- plasma burn and exhaust
- processing of unburned fuel and helium ash
- tritium extraction from the breeding blanket
- tritium storage.

The tritium will diffuse from the heat transfer circuit into the power system, which in most designs is stream-driven. Numerous piping runs and vessels will contain tritium in several temperature regimes: at elevated temperatures, near room temperature, and possibly at cryogenic temperatures for storage, fuel preparation or isotopic separation.

The type and size of fusion power plants examined in this analysis will probably not be built until after 2010. Obviously, the conceptual designs for those plants contain many areas of uncertainty, including concepts and equipment either not yet developed or not even conceived. However, these designs do perform their major intended function--to identify the scale and complexity of technologies which must be developed or invented before practical engineering designs are possible.

The principal primary tritium release mechanisms from fusion reactor components are (not necessarily in order of importance):

- accidents
- leakage from fluid system connections, valves, pumps, etc.
- major and minor maintenance
- permeation through pipes and vessel walls.

The impacts of the release mechanisms will not apply equally to plant and offsite populations, and in some cases may be in conflict. For example, allowing the tritium concentration to build to high values in a steam

FIGURE 4-1. Tritium Flow in a Fusion System

power system would preclude doses to the public, but would subject the plant work force to increasing tritium exposure from local leaks of tritium-laden steam or water. Therefore, the interests of both populations must be considered.

For purposes of this study, the following terminology was adopted to describe tritium flow pathways:

- Tritium source or source term - refers to the transport of tritium from the fusion reactor or associated structures to the environment. Tritium source terms are used in dose or health effects calculations.
- Tritium release - refers to the transport of tritium from one area of "containment" to another. Tritium is "released" by leakage from valves, by permeation of pipes, by spillage of contaminated water, by deliberate actions (power system blowdown), etc. A tritium "release" directly to the environment is also a tritium source term.
- Tritium wastes - tritium contaminated materials that are captured and contained to prevent release to the environment or containment spaces.

The methodology followed in this study to determine the tritium sources and releases from large fusion power plants had three major components: calculation, substitution, and estimation. Where sufficient details were present in the plant design, a calculational approach was taken. This usually involved calculating a tritium permeation rate through a given component, or calculating a fluid leak rate from some item of hardware containing a tritium-contaminated fluid. For example, steam generator design was generally sufficiently well-defined to permit direct calculation of tritium permeation into the steam system.

Where a process or piece of equipment was identified but not described in sufficient detail to permit calculations, substitution was used. Since several fusion reactor designs of nearly the same power rating were reviewed, it was occasionally possible to substitute calculational results from a component or process in one plant design for a similar, but less well-defined one in another plant design. For example, this was particularly true of the diverters.

When design details were sketchy or lacking, source term estimates were based on experience in the nuclear fission industry. These estimates were based on comparisons with similar processes or equipment in light water power reactors and/or ERDA tritium production facilities. For example, estimates of leakages from complex gas piping systems were based on Mound and Savannah River experiences. Estimates of contaminated vacuum pump fluids were based on experience at Mound Laboratory.

Tritium releases as a result of major maintenance or accidents were not included in the scope of this study. Radioactive materials problems associated with major maintenance are addressed in Reference 10; tritium releases from accidents in fusion power systems are addressed in Reference 2.

4.1 TRITIUM CONTAINMENT BOUNDARIES

The general method used to determine the estimated tritium source terms or releases for each reference design involved the consideration of the plant and its components as being within a series of tritium containment boundaries. These boundaries are represented graphically in Figure 4-2.

FIGURE 4-2. Tritium Containment Boundaries

First, the tritium releases from individual components were determined. Next, this information was used to determine the tritium releases and/or source terms for the reactor containment buildings and/or major subdivisions of those buildings. Finally, the calculations for the building containments were used to determine the tritium source terms for accumulation of tritiated wastes, and the releases of tritium to the environment.

This general approach identifies those areas where significant tritium losses originate within the plant, where it is usually easier to apply tritium control than just prior to release at the stack or waste water stream where the tritium has been diluted with large amounts of air or water. This method also highlights the differences among the various reference designs more distinctly.

Tertiary containment was considered as a potential tritium control method and was applied to two of the reference designs evaluated. This design philosophy proved to be a powerful element for tritium control in fusion power systems, for both accidental and chronic tritium releases. Releases of large tritium volumes have occurred at tritium production facilities, where multiple containment boundaries were not available.⁽²⁾ Multiple containment appears to be an essential requirement in fusion reactors in both operational and storage areas.

4.2 TRITIUM PERMEATION

Most permeation regimes in current tritium control technology involve small piping and relatively low temperatures. Materials with low tritium permeation rates are selected (e.g., copper alloys). All-welded construction is common, to obviate losses at flanges. In some cases double jacketing is used, usually on small bore piping. In fusion reactors, the combination of large piping and high temperatures will require advances in methods and materials to control permeation. Specification of double jacketing is tempting, but may be expensive and difficult to engineer on large-diameter pipe runs, particularly if periodic pipe and equipment inspection is required. However, other methods of multiple containment may prove to be effective in localizing tritium which permeates the primary containment boundary. Use of tritium barriers and/or getters is another potentially fruitful approach, but much basic and applied development work is needed to define optimum methods, particularly using extended exposures under prototypic conditions.

Of more basic concern is the lack of data on physical and chemical behavior of tritium in the low-pressure regimes down to 10^{-10} to 10^{-12} mm Hg, where extrapolations suggest that some fusion systems may operate. Preliminary data^(11,12) suggest that first power tritium pressure dependence may apply at pressures below 1 mm Hg, compared to half-power dependence ($p^{1/2}$) at higher pressures where most hydrogen isotope permeation data have been generated. At low tritium pressures (10^{-2} to 10^{-12} mm Hg), calculated permeation rates differ by orders of magnitude depending on the pressure law which applies. For example, the tritium permeation rate into the power system via the heat exchangers and steam generators was calculated to be 10.1 curies per day in UWMAK-I, assuming one-half power pressure dependence. If the first power pressure dependence held below 10 mm Hg, the calculated permeation rate would be 7×10^{-5} curies per day. Confirmation of low tritium permeation rates would have dramatic effects on the calculated tritium releases from fusion reactors.

Results of tritium permeation measurements with niobium are also encouraging.⁽¹³⁾ In systems containing reactive tritium species, corrosion may also be a significant permeation mechanism.⁽¹⁴⁾ Much basic kinetic and thermodynamic data are needed in the pressure ranges where tritium is expected to occur in fusion reactors.

In this analysis, tritium permeation of structural and containment materials was identified as a major contributor to tritium source terms and releases. However, the magnitudes depend on the assumed permeation mechanism. To determine the magnitude of such permeation in the designs under consideration, Webb's calculational method and data⁽¹⁵⁾ were used. In this approach, the tritium permeation rates through a metal are given by:

$$V = \frac{CA}{X\sqrt{3}} \left(P_1^{1/2} - P_2^{1/2} \right) \exp \left(- \frac{Q}{RT} \right) \quad (1)$$

where: V = tritium permeation rate
 C = permeation constant of gas-metal system
 X = thickness of metal
 A = area of metal barrier
 P₁ = upstream tritium partial pressure
 P₂ = downstream tritium partial pressure
 Q = activation energy of diffusion
 R = gas constant
 T = absolute temperature.

To use equation (1), a detailed description of a component, including the dimensions, temperatures and chemical compositions of the component environment, is necessary. This, in turn, requires a rather complete understanding of the processes taking place in and/or around the component. Detailed information was not always available. However, because tritium permeation of containment materials is an important tritium release mechanism, sufficient details were usually given to permit evaluation of the major tritium permeation releases in most reference designs, for example, permeation through steam generators to the power system.

The effect of oxide film barriers on tritium permeation, particularly through steam generators, is also under study.⁽¹⁶⁾ Laboratory results from tests underway at ORNL suggest that oxide films normally present in steam generators will reduce actual tritium permeation by at least a factor of 100 over permeation rates through clean metal. The effects of this new information on fusion reactor tritium releases are included in this analysis.

As stated above, tritium permeation rates based on half-power dependence were used in this study. This approach yields upper limits in the calculations of tritium source terms and releases. Such conservatism is warranted in an environmental assessment of a future energy technology. However, for comparison purposes, the results of the tests at ORNL⁽¹⁶⁾ were also factored into the analysis. In those calculations, the tritium permeation rates into the steam generators were reduced by a factor of 100; all permeation of other system boundaries was regarded as unchanged from the base case, which assumed permeation based on half-power dependence everywhere in the plants. The net result of this change was to reduce substantially the source terms for tritium-contaminated liquids. This is not surprising because most tritium-contaminated liquid sources from fusion reactors originate with releases from power system components, principally blowdown. The results of these analyses are presented in Section 5.0.

4.3 TRITIUM RELEASE MECHANISMS

In addition to permeation, several other tritium release mechanisms were included in this study. However, most are less amenable to quantitative evaluation than the tritium permeation releases. In the following sections, these other release mechanisms and the methods used to evaluate them are described.

4.3.1 Releases During Normal Operation

The major thrust of this study was to evaluate tritium releases and source terms during normal fusion power plant operation. Tritium release mechanisms evaluated for routine plant operations were:

- leakage from fluid system connections and penetrations in the blanket and shield regions of the plants;
- leakage from valves, blowers, pumps, etc.;
- losses of tritiated water from the power systems via spills, leaks, etc.

The release mechanisms proved very difficult to quantify by direct calculation, due primarily to the paucity of details in the reference designs. Because details such as number, size, and type of valves, pipe fittings, gaskets, etc. are usually not found in reference designs, an estimation technique was used to determine the magnitude of the releases. The technique involved comparison of equipment and processes in the reference designs with similar items in light water power reactors, fuel reprocessing plants, and ERDA tritium production and handling facilities. Known tritium source terms and release rates from current nuclear facilities were then used to estimate the tritium releases from corresponding fusion equipment. While this is far from a precise method, it reflects the current ability of technology to deal with tritium. In that sense, the method also yields conservative results, because future technology undoubtedly will improve on current tritium control technology.

4.3.1.1 Leakage

The obvious approach to minimize leakage is to eliminate valves, flanges and other entries to tritium-bearing components. However, all-welded construction is not fully compatible with engineering requirements. Valve leak rates may rise to $\sim 10^{-5}$ cm³/sec, which would result in the release of ~ 1 Ci T₂/day. Advances in valve and closure technology should be considered a requirement for fusion reactor tritium control technology.

4.3.1.2 Tritiated Water Release from Power Systems

Tritium arrives in the steam system by permeation and can possibly be controlled to a major extent by selection of low-permeation materials, including composite metal construction⁽¹⁷⁾ of steam generator piping. Oxide films which form on the steam side also may be effective barriers, but their long-term stability needs to be evaluated. Calculated tritium entry to the steam varies from a fraction of a curie per day to ~30 curies per day, depending on the credit taken for barriers.⁽¹⁷⁾ After permeation occurs into the stream, release of the tritiated water to the environment is partially controllable, but does depend on blowdown requirements.

There is general agreement that tritium contamination of the steam system is the most difficult control problem to contend with, because once the tritium exchanges with protium to form HTO, tritium separation is technically feasible, but so far economically unattractive.⁽¹⁸⁾

4.3.1.3 Tritium Releases from Contaminated Atmospheres

Tritium releases by other mechanisms will generally be gaseous and, in principle, easier to control. Previous authors^(9,19,20) have identified the need for multiple containment boundaries to intercept and recover tritium from chronic or accidental releases. The technology for tritium recovery from air or cover gases has been developed for recovery to ~1 part per billion (volume) for small-scale applications;⁽²¹⁾ however, the current 168 hour occupational controlled area MPC for the oxide form would require 1 part per trillion. Improvement and scaleup of existing tritium removal equipment and modular design of free space in the reactor hall and component rooms will be needed. It also will be important to minimize free space. For example, the "apparent" free space in 5000 MW_{Th} conceptual designs is ~250,000 m³, based on building dimensions. By comparison, the largest existing flow rate in tritium cleanup equipment is ~30 m³/min. The implications of this comparison to tritium atmosphere control are addressed elsewhere.⁽²²⁾ Air locks appear to offer the principal path for tritium escape from spaces with well-designed containment. However, definition of optimum containment materials and design will be needed for building layout.

The RCG^(a) for tritium in the elemental and oxidized forms for 40 hr occupational controlled areas are 2×10^{-3} $\mu\text{Ci/cc}$ and 5×10^{-6} $\mu\text{Ci/cc}$ respectively. This comparison emphasizes the incentive to maintain the tritium in the elemental form inside the plant. In some plant operations the form will be dictated by the process. In other operations, the form may be controllable. For example, low-oxygen and low-humidity atmospheres will minimize reversion to the oxide form. This aspect of occupational dose control needs further evaluation.

4.3.2 Releases During Maintenance

Maintenance of fusion reactors includes a range of operations, from replacing a first-wall module to changing tritium-contaminated pump oil. Opening any tritium-contaminated system will result in tritium releases to the cover gas. Preflushing of the component before disassembly may minimize but will not likely eliminate the contamination potential. A procedure called "green-housing" has been developed which utilizes a plastic tent around openings in contaminated equipment to localize tritium and channel it to cleanup facilities.⁽²³⁾

Methods have been developed to minimize tritium doses to maintenance personnel. Use of protective clothing is often effective where tritium is the only radiation source. However, the clothing generally reduces worker efficiency. This can work counter to dose control in maintenance areas where substantial penetrating radiation contributes to the total dose to maintenance personnel.⁽²⁴⁾ Thus, protective clothing may not always offer the solution to dose control to maintenance personnel working on a tritium-contaminated fusion reactor component which is in a gamma zone.

Tritium releases during maintenance of fusion power reactors are functions of not only the reactor design, but also of the plant management operating philosophy. Put simply, a certain amount of tritium control can be designed into a plant to limit releases during maintenance. However, careful and clean work habits and procedures are also an integral part of tritium control during maintenance operations. Neither item is specifically addressed in the fusion reactor design concepts reviewed for this study. Therefore, this class of tritium releases was evaluated using

(a) Radiation Concentration Guide

the same comparison technique described in the previous section. Again, it was felt that current practices represented a conservative estimate of future events. Maintenance releases addressed in this study included:

- general maintenance on fluid piping systems;
- maintenance on power system components;
- contamination of work areas and personnel during maintenance operations.

Specifically not addressed in this study are tritium releases associated with replacement of a first-wall module or modules, because replacement procedures and the tritium evolution characteristics of the module are not defined. Rough estimates of tritium evolution from a first-wall module during maintenance suggest that the original source term would not exceed 100-1000 Ci/yr.

4.3.3 Tritium-Contaminated Wastes

Tritium releases during waste handling are a function of methodology and of management insistence on adherence to work procedures. Much effective methodology has been developed for routine management and disposal of tritiated wastes.⁽¹⁸⁾ However, tritium control will be a significant consideration in packaging and disposal of wastes from tritium-contaminated fusion systems.

Disposal of large volumes of tritium-contaminated water and other fluids has similar significance for fission and fusion. Fission programs are underway to deal with the problem, including investigation of tank storage, geologic storage, fixation on solids, and tritium separation.⁽¹⁸⁾

Tritium-contaminated wastes considered in this study include:

- clothing
- vacuum pump oil
- vacuum pump mercury
- sorbates for tritium cleanup systems
- ion exchange resins from the power system
- discarded equipment.

Not included in this study were the wastes generated during replacement of first wall components. These components are a special case, and their replacement was not considered a routine or normal operation.

The quantity of tritiated wastes from the reference fusion plants was estimated by both calculation and comparison with existing nuclear facilities. Details for each reference design are presented in Section 5.0.

4.4 GENERAL ASSUMPTIONS

A number of assumptions were made for this analysis, some of which significantly influence the final results. In most cases, these assumptions were conservative, i.e., they lead to predictions of higher rather than lower tritium releases.

For this analysis, a "base case" was first determined. Then a range of options for tritium control was evaluated around this base case. The general assumptions employed in analysis of the base case were:

1. The reference designs were evaluated as presented. No effort was made to "improve" a design even when changes which would reduce tritium releases were evident.
2. No credit was taken for permeation barriers such as oxide films or special composite materials except as specified in the reference design.
3. Tritium permeation rates in metals were assumed to be proportional to the square root of the tritium partial pressure.
4. All tritium released to the containment building atmosphere(s) was assumed to be oxidized to water and processed in a tritium recovery system. The oxidation and subsequent recovery of the tritium were assumed to be accomplished with an overall efficiency of at least 90 percent. This assumption is the exception to the first item listed above. Most reference designs indicated some type of containment atmosphere cleanup process but did not provide specifics.

In laboratory tritium removal systems, tritium concentrations of less than one part per billion in the effluent gas have been reported,⁽²¹⁾ and other systems are under development.⁽²⁵⁾ In principle, it appears possible to achieve substantial scaleup of these systems. Whether they can operate in a favorable economic regime on the scale required for large fusion reactor containment volumes is not clear. For this environmental assessment however, a more conservative approach was taken. Moisture and chemical pollutants are routinely removed from large volumes of air with efficiencies exceeding 90 percent. We assume that it would be possible to do at least as well with tritium.

These assumptions were modified in the analysis of the "optional cases." The most important optional case evaluated was the one which examined the effects of an oxide film in the steam generators. An oxide film is normally present on the steam side of a steam generator and, depending upon the heat carrying fluid, sometimes on both sides of a steam generator. Tests underway at ORNL⁽¹⁶⁾ suggest that an oxide film will reduce actual tritium permeation by at least a factor of 100 over the permeation rate through clean metal. To simulate this effect for our analysis, the tritium permeation rate through steam generators was reduced by a factor of 100 over the base case. All else remained the same.

Another optional case evaluated modified the assumption of tritium permeation rate dependence on tritium partial pressure. For this case, the permeation rate of tritium through metals was assumed to be directly proportional to the tritium partial pressure. This calculation provides a lower limit to tritium permeation rates.

Also evaluated was the use of tertiary containment. For this case, the base case assumption of 90 percent recovery of tritium from containment atmospheres was changed to 99 percent. This reflected the use of the 90 percent efficient recovery systems in two successive containment atmospheres, one of which is released into the other.

5.0 TRITIUM SOURCE TERMS FOR FUSION REACTORS

Six fusion reactor reference designs were evaluated in this study.^(17,26-32) Four were Tokamaks; the others were a Theta-Pinch and a Mirror reactor. The specific designs reviewed were:

- UWMAK-I
- Princeton Fusion Reactor
- UWMAK-II
- BNL Fusion Reactor
- LLL Mirror Fusion Reactor
- ANL/LASL Reference Theta-Pinch Reactor

Four Tokamak designs were evaluated because the Tokamak-type reactor is currently favored to be the most likely first-generation fusion power reactor. Of the four Tokamak designs, the UWMAK-I and Princeton Fusion Reactor designs were the most complete at the time of this study. Therefore, they were analyzed in detail. The other Tokamak designs were evaluated by comparing similar and dissimilar features of each to the two more detailed UWMAK-I and PPPL designs. This method of evaluation was also used for the two non-Tokamak designs. Comparisons of the most important features of the Tokamak and non-Tokamak designs were considered to be of greater value than simple analysis of the non-Tokamak designs. This was primarily due to the great similarity of many systems and components in both reactor types.

For the UWMAK-I and Princeton reference reactor designs a detailed tabulation of tritium releases and source terms was prepared. Each "release" was identified as to form (gas, liquid, solid, adsorbed on solid) and disposition (released to containment atmosphere, to the environment, or recovered and stored on-site). As noted previously, 90 percent of the tritium released to containment was assumed to be adsorbed on molecular sieves, with the remaining 10 percent released to the environment.

The reactor systems studied included:

Blanket and Shield	Diverter
Fuel Preparation and Handling	Tritium Extraction
Piping	Tritium Storage
Heat Exchangers	Waste Handling
Power	Containment

5.1 UWMAK-I

UWMAK-I is a Tokamak-type fusion power reactor with a thermal power rating of about 5000 MW.^(27,28) Its heat transport system consists of a liquid lithium primary system, a sodium secondary system, and a steam-turbine power system with wet cooling towers. Tritium is bred in a blanket region, with the lithium serving as both primary coolant and fertile material. From a tritium control viewpoint, lithium is an excellent coolant because of the low tritium partial pressure due to the relatively strong Li-T interaction. A summary of the tritium releases and source terms for UWMAK-I is presented in Table 5-1. A more detailed tabulation of these results is presented in Appendix A.

There are two published versions of UWMAK-I. The first version⁽²⁷⁾ had the following temperatures in the lithium primary system: 483°C(max); 283°C(min). The second version⁽²⁸⁾ addressed effects of mildly higher coolant temperatures on increased thermal efficiency; the maximum lithium temperature was 489°C and the minimum was 359°C. The second version also had a helium-gap barrier in the steam generator to reduce tritium permeation. Tritium source terms and releases presented in Table 5-1 relate to the original UWMAK-I design. However, we analyzed impacts of the higher system temperature in the second version on tritium releases. This is discussed later in this section.

It is instructive to note in Table 5-1 that more tritium accumulates as wastes than is routinely released to the environment. Obviously, careful attention must be paid to the generation and disposal of tritiated wastes from fusion reactors. The current reference design for UWMAK-I provides much detail on the permeation of tritium through the power system

into the environment, but much less detail on tritiated waste handling. Advances in future reference designs should include more detailed consideration of tritium control technology.

The matter of tritium fuel storage also has a significant impact on the overall tritium distribution in a fusion reactor. Table 5-1 does not include a figure for tritium leakage from a fuel storage system. The form of the stored tritium in UWMAK-I was not specified. Tritium releases are approximately 9000 curies per year from an existing gas phase storage system at the Savannah River Plant. Improved tritium storage technology is under development and would be applied to fusion reactors. However, it is not clear how effectively releases can be reduced in systems having large inventories, numerous valves, and frequent need to revert to the gas phase for tritium transport to fueling facilities.

The effect of raising coolant temperatures to increase thermal efficiency on tritium control is demonstrated in a comparison of the two versions of UWMAK-I. For example, in the second version both the hot and cold leg temperatures of the lithium system were raised. So was the temperature of the yttrium hot trap, which is in thermal equilibrium with the cold leg. The temperature increase had a dual effect on tritium losses: a) it raised the partial pressure of tritium over the lithium coolant, and b) it increased tritium permeation through the lithium system piping. These effects caused a two- to fourfold (depending on location) increase in the tritium permeation rate through the primary system containment. The penalty of increased tritium release must be weighed against the small increase in the plant thermal efficiency. This illustrates the complex interties of tritium control with nearly all of the thermal aspects of fusion power plants. It produces some difficult design challenges and trade-offs for the fusion plant designer, and suggests the need for detailed modeling to analyze the trade-offs of tritium control and other plant operating procedures.

The effects of an oxide film permeation barrier in the steam generator were also examined. Using recent information from ORNL,⁽¹⁶⁾ the tritium permeation rate through the power system was assumed to be a

TABLE 5-1. Tritium Source Terms: UMMAK-I

<u>Tritium Source</u>	<u>Release to Containment</u>	<u>Tritiated Wastes</u> (all in curies/year)	<u>Release to Environment(a)</u>
1. Blanket and Shield	400	4060	40
2. Shield Cooling System	400	360	40
3. Piping Systems (Li + Na)	2800	2500	300
4. Heat Exchangers	20	18	2
5. Power Systems (Steam)	730	660	3770
6. Tritium Extraction System	6400	5760	640
7. Diverter System	570	880	60
8. Fuel Handling Systems	1100?	1000?	100?
9. Waste Handling Systems	--	1840	--
Totals	12,400	17,100	5,000

(a) Environmental source terms

Note 1: Assumes operation 100% of the year

Note 2: Question mark denotes substantial uncertainty

Note 3: No releases from storage or major maintenance included.

factor of 100 less than that calculated for Table 5-1. Results of this analysis are presented in Table 5-2. From an environmental viewpoint, the large reduction in the tritium-contaminated liquid source term is encouraging. The significance of the reduced tritium source terms would be even greater if viewed as a reduction in offsite radiation dose potential.

The work currently underway at ORNL strongly suggests that tritium permeation through a steam generator will be much less than that estimated by assuming a half-power dependence on pressure for the tritium permeation rate and extrapolating existing data to the very low tritium pressure regimes of a fusion reactor. Further work will be necessary to define the relationships between the tritium permeation rate and the properties of the oxide layer for various steam generator materials, tritium pressures, and service conditions. To date, the results of this work look very promising.

5.2 PRINCETON FUSION REACTOR

The fusion power reactor designed by the Princeton Plasma Physics Laboratory is a Tokamak with a thermal power rating of about 5300 MW.⁽²⁶⁾ The maximum primary system temperature is ~660°C. Tritium is bred in a molten salt blanket (flibe, a mixture of LiF and BeF₂) and heat is removed from the blanket by pressurized helium flowing through a complex array of coolant piping. The helium coolant transfers the reactor heat to steam generators which produce steam for a conventional steam-turbine electrical generating plant. This design involves a high tritium partial pressure (up to 10^{-2} mmHg) over the flibe. The higher tritium partial pressure, however, makes tritium control more difficult. Tritium releases and source terms for this reactor are summarized in Table 5-3. A more detailed tabulation of the results of the analysis for this reactor is presented in Appendix B.

TABLE 5-2. Tritium Source Terms: UHMAK-I

Tritium Source	Release to	Tritiated Wastes (all in curies/year)	Release to
	Containment		Environment (a)
1. Blanket and Shield	400	4060	40
2. Shield Cooling System	400	360	40
3. Piping Systems (Li + Na)	2800	2500	300
4. Heat Exchangers	20	18	2
5. Power Systems (Steam)	<4	<3	<38
6. Tritium Extraction System	6400	5760	640
7. Diverter System	570	880	60
8. Fuel Handling Systems	1100?	1000?	100?
9. Waste Handling Systems	--	1840	--
Totals	11,700	16,400	1,200

(a) Environmental source terms

Note 1: Assumes operation 100% of the year

Note 2: Question mark denotes substantial uncertainty

Note 3: No releases from storage or major maintenance included.

TABLE 5-3. Tritium Source Terms: Princeton Reactor

<u>Tritium Source</u>	<u>Release to Containment</u>	<u>Tritiated Wastes</u> (all in curies/year)	<u>Release to Environment(a)</u>
1. Blanket and Shield	400	360	40
2. Helium Coolant System	1100	990	110
3. Piping Systems (He + Flibe)	400	360	40
4. Heat Exchangers	--	Included in 3	--
5. Power Systems (Steam)	730	660	2270
6. Tritium Extraction System	?	?	?
7. Diverter System	360	3230	40
8. Fuel Handling Systems	1100	1000	100
9. Waste Handling Systems	--	1840	--
Totals	4,100	8,400	2,600

(a) Environmental source terms

Note 1: Assumes operation 100% of the year

Note 2: Question mark denotes substantial uncertainty

Note 3: No releases from storage or major maintenance included.

Tritium releases are minimized in this design by using double-walled pipe and vessels in all systems containing tritium and operating at elevated temperatures. The only significant tritium source term identified is a 2200-curie per year loss from the steam system. Tritium reaches the steam system by diffusion from the molten salt to the helium coolant and then by diffusion through the steam generators.

In general, the Princeton design has treated the problems and hazards associated with tritium in a reasonably thorough manner. However, there are some areas where the existing design may prove impractical and thus different methods of tritium control may have to be applied.

In this reactor, control of tritium releases to plant atmospheres and ultimately to the environment depends primarily on a system of double-walled pipe and components in the molten salt and helium coolant systems. The double-walled system is necessary to prevent the release of about 20,000 curies per day of tritium which otherwise would diffuse into the plant atmospheres. Whether or not such a double-walled system is a practical design solution to the control of tritium in this reference design is a critical question. The answer affects nearly every aspect of tritium control and release in the plant, with the possible exception of the steam system. With the double-walled systems the tritium releases are very small; without them whole areas of the plant must be isolated and large volumes of containment atmosphere must be processed to keep tritium losses at acceptable levels.

Another area where tritium source terms and releases are difficult to assess for this plant is maintenance. Tritium control and releases during maintenance depend upon whether or not the double-walled piping systems prove to be practical. If they are, then the maintenance problems associated with tritium are primarily those associated with disposing of tritium-saturated components and piping. If double-walled systems prove difficult to use in practice, then the maintenance problems involve access to contaminated atmospheres as well as tritium-saturated equipment.

In the Princeton reactor, tritium fuel would be stored as a metal hydride. Tritium leakage from such a system may be considerably less than losses from a gas storage system. However, no firm estimates were possible because no design details were given for the fuel storage system. Thus Table 5-3 includes no estimate for tritium leakage from fuel storage system components.

As in the UWMAK-I analysis, the effects of an oxide film permeation barrier in the steam generators were evaluated for the Princeton design. The results of this evaluation are summarized in Table 5-4. Because the Princeton design called for venting power system blowdown (9500 gal/day) up a 100-meter stack as water vapor, reducing tritium permeation had no effect on the source term for tritium contaminated liquids. Thus the net effect of the tritium permeation reduction is almost entirely confined to the gaseous source term. This illustrates quite well one of the design options for tritium control available to the fusion reactor designer. In UWMAK-I, power system blowdown was directed to the cooling towers, where about two-thirds of it was vaporized and the rest released as liquid from cooling tower blowdown. Thus UWMAK-I had a significant tritium-contaminated liquid source term. Princeton avoided this by stacking the blowdown. The reactor designer thus has several options, all with different environmental consequences, for dealing with tritium-contaminated power system effluents.

5.3 COMPARISON: UWMAK-I AND PRINCETON FUSION REACTOR

The gaseous and liquid tritium source terms for UWMAK-I and the Princeton Fusion Reactor are presented in Table 5-5. For purposes of this comparison, the source terms for the fusion plants have been normalized to curies per gigawatt electric per year.

The first column reflects tritium escape to the environment in liquids from the steam power system and from cover gas containment. Recovery efficiency from containment was assumed to be 90 percent, based on moisture recovery efficiency in large heating and ventilating systems. Scaleup of existing tritium recovery technology almost certainly will make this a conservative assumption. In the PPPL design, blowdown from the power system

TABLE 5-4. Tritium Source Terms: Princeton Reactor

Tritium Source	Release to Containment	Tritiated Wastes (all in curies/year)	Release to Environment (a)
1. Blanket and Shield	400	360	40
2. Helium Coolant System	1100	990	110
3. Piping Systems (He + Flibe)	400	360	40
4. Heat Exchangers	--	Included in 3	--
5. Power Systems (Steam)	<4	<3	<23
6. Tritium Extraction System	?	?	?
7. Diverter System	360	3230	40
8. Fuel Handling Systems	1100	1000	100
9. Waste Handling Systems	--	1840	--
Totals	3,400	7,800	350

(a) Environmental source terms

Note 1: Assumes operation 100% of the year

Note 2: Question mark denotes substantial uncertainty

Note 3: No releases from storage or major maintenance included.

TABLE 5-5. Tritium Releases During Normal Operations^(a)

	Base Case	Base Case with Oxide Film Barrier	Permeation Proportional To Pressure
	----- (Ci/GWe.y) -----		
UWMAK-I (1.5 GWe)			
Gaseous	2500 (1750)	800 (100)	260
Liquid	800	~10	None
PPPL (2 GWe)			
Gaseous	1300 (1100)	180 (30)	170
Liquid	None	None	None

(a) No releases from storage or major maintenance included in above figures; 40-600 Ci/GWe.y are estimated from experience in tritium storage facilities at ERDA tritium production laboratories.⁽¹⁹⁾ A rough estimate places releases due to major maintenance in the range of 100-1000 Ci/yr.

was released through a stack, precluding liquid releases to streams. The estimated gaseous tritium release rate from the PPPL system is lower than the UWMAK-I rate because credit was taken for extensive double-wall pipe and component system construction, which intercepted ~20,000 Ci/d tritium permeation releases, promoted by the relatively high tritium pressure (10^{-3} mm Hg) in the flibe blanket. The numbers shown in the parentheses next to the gaseous tritium release rates in Table 5-5 were derived by applying the assumption of triple containment of all reactor components with containment atmosphere cleanup efficiencies of 90 percent in each containment atmosphere (secondary and tertiary). This design option, partially incorporated in the existing PPPL design, offers further reductions in gaseous tritium release rates.

The second column reflects lower tritium releases if credit is taken for oxide film barriers on the steam side of heat exchanger piping, based on the recent results from ORNL.⁽¹⁶⁾ Tritium leakage to the power system then becomes a small fraction of a curie per day for both systems. The third column reflects further reductions if the permeation depends on the first power of the tritium pressure.

The releases do not include contributions from accidents, major maintenance or storage, because there were insufficient bases to permit detailed analysis. A range of storage releases is given at the bottom of the table, based on experience at tritium laboratories.

Clearly, both reactor designs lead to low tritium source terms. The most interesting result of the comparison presented in Table 5-5 is the display of the many options and tradeoffs for tritium control available to the plant designer. The point and form of release, design of containments, presence of permeation barriers (either planned or unplanned), etc., all impact the tritium source terms for fusion power plants. Ultimately, all of these elements and many more must be factored into a reactor design which not only must be safe for both the public and plant personnel, but which must produce economically competitive power as well.

5.4 OTHER TOKAMAK FUSION REACTORS

UWMAK-II and the Brookhaven fusion reactor designs were also evaluated during this study. Their lack of completeness at the time of the analysis relative to the UWMAK-I and Princeton designs made analysis and direct comparisons with the Princeton and UWMAK-I designs difficult. Thus the approach taken was to compare features of the UWMAK-II and BNL designs with the UWMAK-I and Princeton designs and evaluate the similarities and differences in the designs.

5.4.1 UWMAK-II

UWMAK-II is a Tokamak-type fusion power reactor with a thermal power rating of 5000 MW.⁽²⁹⁾ A helium coolant transfers heat from a blanket region to a conventional steam-turbine power system with wet cooling towers. The maximum primary system temperature is ~650°C. Tritium is bred in a solid lithium blanket (LiAlO_2) where it is collected in a manifold system. Some tritium diffuses into the helium coolant, where it is recovered.

In general, UWMAK-II contains many systems which are similar to either UWMAK-I or the Princeton reactor. The problems associated with tritium removal from the helium coolant and tritium releases from this coolant system were not as completely addressed in the UWMAK-II design as they were in the Princeton design. The UWMAK-II design specified none of the elaborate double-walled construction found in the Princeton design and relied heavily on barrier materials in the steam generator to prevent large tritium releases to the power system.

Because of the above uncertainties, it was difficult to determine whether or not the UWMAK-II would have larger tritium source terms than either UWMAK-I or the Princeton Fusion Reactor. The conclusion reached was that with proper clarification of and modifications to the UWMAK-II design, tritium source terms for that reactor could be held to values similar to those estimated for the UWMAK-I and Princeton reactors.

5.4.2 Brookhaven Fusion Reactor

The BNL fusion reactor design is a Tokamak with a thermal power rating of about 5000 MW.⁽³⁰⁾ It is cooled with a combination of water and helium heat transfer systems, which transfer energy to standard steam-turbine power systems. The maximum primary system temperature is ~760°C. Tritium is bred in solid lithium-aluminum or lithium-aluminate materials.

The major portion of the BNL fusion reactor design study that was evaluated for this report dealt with the design of a low activation blanket for a Tokamak similar to UWMAK-I. There were few descriptions of equipment sizes or process flow rates. Essentially, only the blanket and shield region and the power system were described. One unique feature of this design concerns removal of tritium from the power system. It was stated that the power system would be designed for zero blowdown and that the 0.5 curies per day of tritium which diffuses into the power system would be removed. No further discussion or description of how this was to be accomplished was given.

Given the many similarities between the BNL design and the UWMAK-I design, one must tentatively conclude that the tritium source terms for the two plants would be similar. The low activation blanket of course has major implications for the magnitude of radiation exposure to plant personnel during a change of the first wall. It would also undoubtedly impact the tritium releases during such a major maintenance operation, but the degree of that impact and its significance relative to other tritium releases is undetermined.

5.5 LLL MIRROR FUSION REACTOR

Lawrence Livermore Laboratory (LLL) has prepared a design study for a 2000 MWe mirror fusion reactor.^(31,32) This reactor would produce 590 MWt of fusion energy by burning a D-T plasma magnetically contained by a Yin Yang magnet system. Electrical power would be produced by both direct conversion of charged particle energy and by a standard steam-cycle thermal energy conversion system. Liquid lithium, helium, and water were the alternative heat transfer fluids chosen for this study.

A detailed analysis of tritium source terms for this reactor was not attempted. Rather, the general concept of a mirror fusion reactor was examined to identify those design features which would result in either substantially larger or smaller tritium source terms than those encountered in the Tokamak designs. The similarities and differences between the LLL Mirror Fusion Reactor and the Tokamaks analyzed in the earlier sections of this report are discussed below.

5.5.1 Similarities to Tokamaks

In many respects the LLL Mirror Fusion Reactor resembles both UWMAK-I and the Princeton Fusion Reactor in the thermal energy conversion systems. In the Livermore machine, tritium is bred in and extracted from a liquid lithium primary coolant, similar in concept to UWMAK-I. Heat is transferred from the lithium to a helium secondary system and then to a steam-cycle

power system, similar in concept to the Princeton Reactor. The LLL designers estimate that about 330 curies per year of tritium would reach the reactor power system via permeation of heat transfer surfaces. On a per MWe basis, this is slightly more than the rate postulated for the Princeton Reactor and slightly less than that estimated for UWMAK-I. The differences, however, are small enough to conclude that tritium releases via the thermal power system of a mirror fusion reactor would probably be similar to those of similarly rated Tokamaks.

5.5.2 Differences from Tokamaks

The requirement for continuous injection of high energy D-T particle beams into the mirror reactor plasma will present some problems (and offer some advantages) over the fueling method used in the Tokamaks. First, the solid pellet injection method used in UWMAK-I and the Princeton Reactor is likely to be more efficient than the neutral beam injection process of a mirror machine. Thus the mirror reactor employs an array of ion recovery devices, direct converters, and vacuum systems to recover energy and material not successfully injected into the plasma. (These recovery devices are, in some respects, analogous to the Tokamak diverter.) However, since the neutral beam injection process is less efficient, a higher tritium recycle rate will be required which increases the size and volume of process equipment needed. These factors combine to provide a greater opportunity for tritium releases from leaking valves, flanges, and other gas system fittings. Tritium contamination of vacuum system working fluids is also likely to be greater for mirror fueling systems than for the Tokamak systems.

From a tritium control standpoint, the advantages (if any) of the neutral beam injection fueling system of a mirror fusion reactor are related to its steady-state operation. It is usually easier to optimize a process which operates at one condition than to optimize a process with several operating conditions. The transient operating mode of a Tokamak

reactor requires intermittent operation of the fueling devices. Thus tritium control must be designed for repeated cycles of dead time, start-up, burn, and shut-down. Whether this would present tritium control problems substantially more difficult than those encountered with neutral beam injectors is not possible to determine at this time.

Another significant difference between the mirror fusion reactors and the Tokamak fusion reactors is the use of direct converters in the mirror machines. Direct conversion of charged particle beams into electricity is a significant feature of the LLL Mirror Reactor. The most significant feature of the direct conversion system, from a tritium control viewpoint, is the enormous, fan-shaped vacuum vessel around the expander-collector. In the 200-MWe LLL design, 400 forty-eight-inch diffusion pumps are required to service the direct conversion system. Tritium contamination of vacuum pump working fluids is thus likely to present a significant although potentially manageable problem in preventing tritium releases from the plant. One could therefore expect a mirror fusion reactor to generate more tritiated waste material (in the form of pump fluids) than would Tokamak fusion reactors. By way of comparison, the 1500-MWe UWMAK-I requires 96 thirty-inch diffusion pumps to operate its diverter system.

5.5.3 Summary

Mirror fusion reactors of the type described in the LLL study would probably exhibit tritium source terms to the environment approximately equal to those of similarly rated Tokamak fusion reactors. This is primarily due to the similarities of the two reactor types at their interfaces with the environment, e.g., containments, thermal power systems, waste disposal requirements, etc. The major differences between the two reactor types are related to the high energy recycle and the large vacuum system requirements of the mirror reactor. The mirror reactor is likely to generate more tritiated waste inside containment than Tokamak reactors. While this will require extra care in the design of tritium control features for mirror reactors, it is not expected to present any major problems beyond those that would be encountered in any large fusion power reactor.

5.6 ANL/LASL THETA-PINCH REACTOR

Los Alamos Scientific Laboratory and Argonne National Laboratory (LASL/ANL) have jointly prepared a reference design for a theta-pinch fusion power plant.^(17, 33) Called the Reference Theta-Pinch Reactor (RTPR), the design includes two alternative thermal energy conversion systems. One system uses a lithium-sodium-steam heat transport system and operates at conservatively rated temperatures (550°C max.), with a net plant efficiency of 40 percent. The other system uses a lithium-potassium-steam heat transport system with a potassium vapor topping cycle. This advanced system operates at higher temperatures (1130°C max.), with a net plant efficiency of 56 percent. Both energy conversion systems were evaluated for a duty cycle time of 10 seconds. The output thermal power is 3600 MWt for both systems. The net electrical power is 1200 MWe for the conventional system and 1700 MWe for the advanced system.

The conventional sodium-steam power system was also evaluated for a duty cycle time of three seconds. This represents a more severe design case and thus was chosen as the basis for an environmental analysis.⁽¹⁷⁾ For the three-second duty cycle, the power outputs are 12,000 MWt and 4100 MWe.

In many respects, the LASL/ANL RTPR design is really several partial designs around a single reactor concept. As such, it is difficult to perform a single analysis of tritium source terms and call it representative of a theta-pinch fusion power plant. Indeed, the environmental analysis of the RTPR done by ANL and LASL⁽¹⁷⁾ contains several sub-cases, each with tritium control problems of different magnitudes. For this analysis, it is therefore more instructive to examine the general theta-pinch reactor concept to determine how it compares with the Tokamak and mirror reactors vis-à-vis tritium source terms. Such comparisons are presented below.

5.6.1 Similarities to Other Fusion Reactor Concepts

The RTPR combines many features of both UWMAK-I and the Princeton Fusion Reactor. Its main heat transport system (for the conventional thermal energy conversion option) is quite similar to that in UWMAK-I. From a tritium standpoint, the primary differences between the two systems are a higher tritium partial pressure in RTPR (10^{-10} torr versus 10^{-11} torr in UWMAK-I) and the use of composite barrier materials in the RTPR steam generators. These differences result directly from the use of a molten salt contactor for tritium extraction from the lithium in RTPR, in contrast to the UWMAK-I system which uses yttrium beds. The higher tritium partial pressure in the RTPR lithium results in greater permeation releases, hence the barrier materials. Based on a tenfold higher tritium pressure, an RTPR, similar in power level to UWMAK-I, likely would have permeation releases about three times greater than those calculated for UWMAK-I.^(a) For the advanced thermal energy conversion system in RTPR, higher lithium and potassium temperatures are employed to enhance plant thermal efficiency. The penalty paid for this includes greater tritium permeation releases than for the lower temperature conventional system.

The remedy for the higher tritium permeation releases in RTPR is similar to that proposed for the Princeton Reactor: double containment with helium-purged tritium control zones. In the RTPR, all regions of the plant containing equipment with high tritium concentrations (or substantial permeation losses) are either under vacuum or are purged continuously with helium. The use of helium-purged jackets around critical components is also considered as an alternative helium-conserving option.

(a) This assumes that tritium permeation rate depends on the square root of the tritium partial pressure. If, as some investigators propose, the permeation rate is directly proportional to partial pressure then tritium releases via permeation from RTPR, although still very low, would be ten times greater than those calculated for UWMAK-I.

The RTPR design of isolated, helium-purged cells in high tritium contamination areas is an excellent tritium control technique which should find application in future fusion reactor reference designs of all types. However, the use of helium-purged, double-jacketed equipment should probably be kept to a minimum, for reasons discussed in Section 5.2. Because of the extensive system of purged zones in RTPR, tritium releases via the permeation mechanism to the main building atmospheres are assumed to be negligible. The RTPR most closely resembles the Princeton Reactor with regard to tritium releases due to permeation of structural materials.

The RTPR shares at least one tritium control liability--large size vacuum chambers--with the mirror fusion reactor. The main vacuum chamber is several meters in minor diameter and over 100 meters in major diameter. A secondary vacuum chamber with an even larger minor cross-sectional area than the plasma chamber and a similar 100⁺ meter major diameter is also included in RTPR. As with the mirror machine, large vacuum chambers contaminated with tritium pose problems for the large array of vacuum pumps required to sustain the system. Contamination of vacuum pump fluids will be a significant tritium control problem in the RTPR.

5.6.2 Differences from Other Fusion Reactor Concepts

Several design features unique to the theta-pinch reactor concept will also present tritium control problems. The requirement for pulsed operation at a frequency of 10 seconds per cycle or less places severe demands upon plasma chamber exhaust and fueling systems. The RTPR fuel supply and removal system requires 352 first-stage vacuum pumps and 176 second-stage pumps. Such an elaborate gas system will certainly be a potential source of leaks. Thus, one tritium source for the helium purge system serving the reactor-core buffer region and access tunnel will probably be leaks from the hardware of the fuel supply and removal system. The complex fuel supply and removal system will almost certainly be a major source of tritium-contaminated pump fluids and hardware.

A second feature of the RTPR likely to contribute to tritium control problems is its enormous containment building. RTPR has the largest containment structure (over 260 meters in diameter) of any power plant reviewed for this study. By comparison, the containment structures of UWMAK-I and the Princeton Reactor are 120 meters and 85 meters in diameter, respectively. Removal of tritium from the atmosphere of this building would be an extremely difficult task. Therefore, while RTPR certainly shares this problem with the other designs, the size of RTPR would make control of tritium released to the building atmosphere more difficult, if only because of the scale factor.

5.6.3 Summary

The theta-pinch type fusion reactor, as exemplified in the RTPR, possesses most of the same tritium control problems as the other types of fusion reactors. However, a significant difference between the RTPR and the other reactors is associated with the very large containment volumes in the RTPR which must be purged or evacuated for effective tritium control.

Tritium releases to the environment from an RTPR appear to be amenable to the same order of control as for the Tokamaks. As in the cases of the other fusion plants, tritium releases to the environment are largely under the control of the designer. For RTPR, control of tritium releases may prove more costly than for the Tokamaks, but principally because of plant size.

The RTPR shares with the mirror fusion reactor the tritium control problems associated with large numbers of vacuum pumps. Control and disposal of significant quantities of tritium-contaminated vacuum pump fluids in the RTPR will be a major tritium handling activity. As with the mirror fusion reactor, this is one control problem that can be managed to prevent tritium releases from the plant. The major impact of this activity comes from the potential for contamination of plant equipment and personnel, as well as increased waste disposal burdens.

6.0 RECOMMENDATIONS TO DEFINE AND ADVANCE TRITIUM CONTROL TECHNOLOGY FOR FUSION REACTORS

Advances in tritium control technology for fusion reactors must develop on several fronts:

- improved definition of fusion reactor designs;
- assimilation of existing tritium control technology;
- advances in basic tritium data specific to fusion reactor regimes;
- advances in engineering technology specific to fusion reactor tritium control requirements;
- development of plant models to optimize tradeoffs between tritium control and plant operation.

The activities outlined above interact and in several respects need to develop simultaneously. As the next generation of fusion reactor conceptual designs develop, they need increased focus on tritium control technology, including available aspects from existing technology, such as tritium monitoring, maintenance procedures, and waste handling. The designs now depend on tritium data extrapolated from relatively high pressure regimes. A broad experimental base needs to develop which is specific to the fusion reactor operating regimes identified in the designs. The reactor designs also will define tritium control requirements on an engineering level. The basic studies will provide a data base which is important to the design and operation of equipment and processes. Inclusion of tritium control in plant modeling is in early stages. It will become increasingly effective as plant designs improve, and will become increasingly important as the time for plant design and operation approaches.

The following sections outline specific recommendations under each topic listed above.

6.1 ADVANCES IN FUSION REACTOR DESIGNS

The first generation fusion reactor conceptual designs provide useful insights regarding plant size, layout, and operation. In several aspects they do not provide engineering detail needed for a finely-tuned

treatment of tritium releases and source terms, which leads to the following recommendations:

- The design efforts should begin with the premise that the plant will be designed and operated to minimize tritium releases to the environment and occupational exposures to plant personnel.
- Improvements in design detail are required to assess the adequacy of tritium control in fusion power plants. Details such as pipe and component dimensions, temperatures, numbers and types of valves, flanges and pipe connections, fluid stream compositions, temperatures and flow rates, and internal and external containment building design are needed to quantify tritium releases.
- The designs need to address an approach to modular compartments in tritium control areas, minimizing cover gas volumes to the extent possible. Use of multiple containment boundaries to intercept tritium in all routine or accidental releases appears to be essential.
- Efforts should be made to permit easier comparison of point designs. Too often one reference design differs sufficiently from other designs that meaningful comparisons are difficult.

6.2 ASSIMILATION OF EXISTING TRITIUM CONTROL TECHNOLOGY

Tritium has been handled in large quantities at numerous tritium production facilities for over twenty years. The state-of-the-art in tritium control has been summarized in detail.⁽²³⁾ Fusion technology has available substantial technology in the following aspects of tritium control:

- monitoring and alarm systems
- tritium removal from small cover gas volumes
- maintenance procedures
- protective clothing

- solid and liquid waste disposal
- storage procedures.

For example, implementation of recent advances in tritium control technology has resulted in major reductions in tritium emissions from the Mound Laboratory.⁽³⁴⁾ Fusion reactors can draw from much of that technology, but will also have specific tritium control problems not addressed by existing technology.

6.3 ADVANCES IN BASIC TRITIUM DATA SPECIFIC TO FUSION REACTOR REGIMES

Tritium pressures in most fusion reactor systems are estimated to be orders of magnitude below the pressures where hydrogen isotope/materials interactions have been defined. Most available data were generated at tritium pressures above one mm Hg(torr). Extrapolation of existing data to fusion reactor regimes suggests that tritium pressures will be in the range of 10^{-2} to 10^{-14} torr in blanket systems.^(26,35) The kinetics and thermodynamics of tritium/material interactions under these conditions are poorly defined. Solubilities, diffusivities, and permeation rates are important examples where data are largely lacking under fusion blanket conditions. We have pointed out earlier that permeation calculations may be in error by several orders of magnitude if the permeation mechanism is not known. A few low-pressure permeation rate studies have been initiated.^(12,13,16) However, permeation is one of the most important tritium release mechanisms and its quantification is an important aspect of tritium control definition, suggesting that development of a substantial data base is justified. This should include definition not only of the common gas-phase permeation mechanism described by Webb,⁽¹⁵⁾ but also other potential permeation mechanisms such as corrosion.⁽¹⁴⁾

Barrier technology is another tritium control area where improved basic understanding is needed. Effects of surface oxides and other films on tritium adsorption and permeation need extensive definition for the full range of fusion reactor candidate materials.

Basic studies are needed to advance the state-of-the-art in tritium removal from cover gases, since existing methods process to lower tritium levels which still are well above maximum permissible concentrations for both T_2 and T_2O . An important aspect of cover gas tritium control is to identify measures to maintain the tritium in the elemental form, due to

the lower RCG value. Additional work appears justified to determine whether the economics of tritium removal from liquid wastes can be improved. In summary, basic studies are needed in the following areas relevant to tritium control technology, giving highest priority to the first three:

- the kinetics and thermodynamics of tritium/material interactions, with early emphasis on permeation;
- barrier technology;
- advances in tritium removal from cover gases;
- advances in tritium removal from liquid wastes.

6.4 ADVANCES IN ENGINEERING TECHNOLOGY FOR TRITIUM CONTROL

The principal aspects of tritium control engineering technology appear to be:

- plant design and operation to minimize tritium releases to containment;
- provisions to intercept accidental and routine tritium releases at all plant locations;
- provisions to remove tritium from cover gases rapidly, to acceptable concentrations;
- provision for economical and safe disposal of tritiated wastes.

The above procedures must assure protection of the public and the plant personnel from excessive tritium doses.

The principal engineering developments which appear essential for near-term priority action are:

- Engineering-scale demonstration of promising barriers identified in the basic studies; oxide film barriers on steam generator materials appear capable of reducing permeation rates by two orders of magnitude;⁽¹⁶⁾ however, they need further qualification in extended tests under prototypic conditions. Composite metal barriers have shown some promise,^(17,33) but need further technical and economic evaluation for large-scale application.

- Evaluation of the practicality of double jacketing for large, high-temperature nuclear piping runs; if this approach to intercept tritium does not prove feasible, jacketing with a getter material may offer an effective method of tritium interception, again requiring engineering evaluation.
- Scaling of existing methods for detritiation of cover gases and evaluation of their applicability to tritium control in large-volume fusion reactor halls; this must include evaluation of modular design which will be compatible with reactor operation and major maintenance requirements.
- Evaluation of tritium storage options for all stages of fusion reactor operation, emphasizing tritium control aspects.

The following represent additional tritium control activities for longer-term implementation:

- Evaluation of valving and closure methods for high-temperature, large-diameter piping and vessels handling tritium-bearing fluids.
- Identification of acceptable work procedures for plant maintenance personnel in tritium and tritium/gamma environments, drawing from experience in tritium laboratories and fission reactors, but assuring that the procedures apply in all aspects of fusion reactor operation and maintenance.
- Identification of acceptable tritium handling procedures during all aspects of fusion reactor maintenance, focussing particularly on major maintenance of tritiated, radioactive first-wall modules.
- Evaluation of tritiated waste handling, packaging, and long-term storage, drawing from existing technology.
- Evaluation of the alternative scenarios for control and handling of large-volume tritiated water from the power system, including controlled releases, low-blowdown operation, and long-term storage.
- While this report addresses tritium containment under normal reactor operation, it is clear that fusion reactor accident

scenarios must be evaluated, and engineering capability must evolve to deal with them. As indicated earlier, the most important single aspect appears to be the design of multiple containment boundaries to intercept and process tritium released from primary containment before it reaches the environment, and without exceeding dose limits to plant personnel.

6.5 DEVELOPMENT OF PLANT MODELS TO OPTIMIZE TRITIUM CONTROL AND PLANT OPERATION TRADEOFFS

Changes in plant operation which are otherwise attractive may work counter to effective tritium control. Increases in operating temperature are desirable to improve thermal efficiencies, but at the same time may promote higher tritium system pressure and permeation rates. Tritium removal from some blankets is improved as the tritium concentration increases, but the increasing concentration may also increase permeation. Modeling of plant operation must include tritium control as a major consideration. Darvas has suggested preliminary approaches to tritium modeling in fusion plants.⁽⁹⁾ More recent publications also provide useful insights to tritium modeling in fusion plants.^(22,36,37)

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APPENDIXES

APPENDIX A

UWMAK-I

Detailed tabulations of the tritium releases and source terms for UWMAK-I are presented here in Tables A-1 and A-2. Table A-1 contains the data used to prepare summary Table 5-1 and includes an estimate for tritium releases from a gas phase storage system. Table A-2 was used to prepare summary Table 5-2 and also includes estimates of storage releases.

The differences between Tables A-1 and A-2 are found in permeation releases through the steam generators. In Table A-2 credit was taken for a factor of 100 reduction in the tritium permeation rate through the steam generators due to the presence of an oxide film. Details are discussed in Section 5.1.

TABLE A-1. UWMAK-I Tritium Releases and Source Terms

Tritium Releases	Release To Containment	Stored On-Site	Release To Environment	Total Tritium
	Curies/Year			
1. Blanket and Shield				
a. Diffusion from blanket into vacuum gap	---	102(AL)	---	102
b. Diffusion from shield to vacuum gaps in shield	---	1200(AL)	---	1200
c. Diffusion from shield into helium cooling system	---	2400(AL)	---	2400
d. Leakage from B & S fluid connections and penetrations	<37?(G,L)	<33?(AL)	<4?(G)	<37?
e. Maintenance on B & S equipment	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
2. Shield Cooling System				
a. Leakage from valves, pumps, etc.	<37?(G)	<33?(AL)	<4?(G)	<37?
b. Cleanup of Shield Cooling System helium	---	See 1c	---	See 1c
c. Maintenance on Shield Cooling System	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
3. Piping Systems (Li & Na)				
a. Diffusion through lithium system piping	2300(G)	2070(AL)	230(G)	2300
b. Diffusion through sodium system piping	55(G)	49(AL)	6(G)	55
c. Diffusion through walls of sodium storage tanks	59(G)	53(AL)	6(G)	59
d. Leaks from valves, pumps, etc.	<37?(G,S)	<33?(AL,S)	<4?(G)	<37?
e. Maintenance on Lithium and Sodium System piping	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?

G - Gas
L - Liquid
AL - Adsorbed Liquid
S - Solid
? - Estimate or Unknown

TABLE A-1 (continued)

Tritium Releases	Release To Containment	Stored On-Site	Release to Environment	Total Tritium
4. Heat Exchangers				
a. Diffusion through IHX shell	17(G)	15(AL)	2(G)	17
b. Diffusion through steam generator shell	2(G)	1.8(AL)	0.2(G)	2
5. Power Systems				
a. Diffusion through steam generator into power system, released via blowdown	---	---	3700(L)	3700
b. Leaks from Power System	<365?(G,L)	<328?(AL)	<37?(G)	<365?
c. Maintenance on Power System components	<365?(G,L,S)	<328?(AL,S)	<37?(G)	<365?
6. Tritium Extraction Systems				
a. Permeation losses	6000(G)	5400(AL)	600(G)	6000
b. Leaks from valves, pumps, etc.	<37?(G)	<33?(AL)	<4?(G)	<37?
c. Maintenance on ³ H Extraction Systems	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
7. Diverter Systems				
a. Diffusion through lithium system piping	12(G)	11(AL)	1(G)	12
b. Diffusion through sodium system piping	small(G)	small	negligible	small
c. Diffusion through IHX and Steam Generator to Power System	---	---	<0.1(L)	<0.1
d. Tritium Extraction System permeation losses	58(G)	52(AL)	6(G)	58
e. Permeation losses from vacuum system	102(G)	92(AL)	10(G)	102
f. Maintenance on Diverter System Equipment	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
g. Leakage from valves, pumps, etc.	<37?(G)	<33?(AL)	<4?(G)	<37?
h. Contaminated mercury from vacuum pumps	---	365?(L)	---	365?

TABLE A-1 (continued)

Tritium Releases	Release To Containment	Stored On-Site	Release To Environment	Total Tritium
8. Fuel Handling Systems				
a. Losses during fuel preparation and injection	<365?(G)	<328?(AL)	<37?(G)	<365?
b. Losses during storage (assume storage as gas)	9,100?(G)	8,200?(AL)	900?(G)	9,100?
c. Leakage from isotopic separation processes	<365?(G)	<328?(AL)	<37?(G)	<365?
d. Maintenance on fuel handling systems	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
9. Waste Handling Systems				
a. Contaminated Clothing	---	37(S)	---	37
b. Contaminated pump oils	---	1,800?(L)	---	1,800?
c. Disposal of contaminated yttrium from tritium extraction beds	---	?	---	?
d. Disposal of contaminated charcoal from diverter	---	?	---	?
e. Disposal of contaminated ion exchange resins from Power Systems	---	?	---	?
Total for All Systems: (Curies/Year)	21,540	25,292.8	5,851.3	31,144.1

TABLE A-2. UWMAK-I Tritium Releases and Source Terms

Tritium Releases	Release to Containment	Stored On-Site	Release to Environment	Total Tritium
	Curies/Year			
1. Blanket and Shield				
a. Diffusion from blanket into vacuum gap	---	102(AL)	---	102
b. Diffusion from shield to vacuum gaps in shield	---	1200(AL)	---	1200
c. Diffusion from shield into helium cooling system	---	2400(AL)	---	2400
d. Leakage from B & S fluid connections and penetrations	<37?(G,L)	<33?(AL)	<4?(G)	<37?
e. Maintenance on B & S equipment	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
2. Shield Cooling System				
a. Leakage from valves, pumps, etc.	<37?(G)	<33?(AL)	<4?(G)	<37?
b. Cleanup of Shield Cooling System helium	---	See 1c	---	See 1c
c. Maintenance on Shield Cooling System	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
3. Piping Systems (Li & Na)				
a. Diffusion through lithium system piping	2300(G)	2070(AL)	230(G)	2300
b. Diffusion through sodium system piping	55(G)	49(AL)	6(G)	55
c. Diffusion through walls of sodium storage tanks	59(G)	53(AL)	6(G)	59
d. Leaks from valves, pumps, etc.	<37?(G,S)	<33?(AL,S)	<4?(G)	<37?
e. Maintenance on Lithium and Sodium System piping	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?

G - Gas
 L - Liquid
 AL - Adsorbed Liquid
 S - Solid
 ? - Estimate or Unknown

TABLE A-2 (continued)

Tritium Releases	Release To Containment	Stored On-Site	Release to Environment	Total Tritium
4. Heat Exchangers				
a. Diffusion through IHX shell	17(G)	15(AL)	2(G)	17
b. Diffusion through steam generator shell	2(G)	1.8(AL)	0.2(G)	2
5. Power Systems				
a. Diffusion through steam generator into power system, released via blowdown	---	---	37(L)	37
b. Leaks from Power System	-----Negligible-----			
c. Maintenance on Power System components	<4?(G,L,S)	<3?(AL,S)	<1?(G)	<4?
6. Tritium Extraction Systems				
a. Permeation losses	6000(G)	5400(AL)	600(G)	6000
b. Leaks from valves, pumps, etc.	<37?(G)	<33?(AL)	<4?(G)	<37?
c. Maintenance on ³ H Extraction Systems	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
7. Diverter Systems				
a. Diffusion through lithium system piping	12(G)	11(AL)	1(G)	12
b. Diffusion through sodium system piping	small(G)	small	negligible	small
c. Diffusion through IHX and Steam Generator to Power System	-----Negligible-----			
d. Tritium Extraction System permeation losses	58(G)	52(AL)	6(G)	58
e. Permeation losses from vacuum system	102(G)	92(AL)	10(G)	102
f. Maintenance on Diverter System Equipment	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
g. Leakage from valves, pumps, etc.	<37?(G)	<33?(AL)	<4?(G)	<37?
h. Contaminated mercury from vacuum pumps	---	365?(L)	---	365?

TABLE A-2 (continued)

Tritium Releases	Release To Containment	Stored On-Site	Release To Environment	Total Tritium
8. Fuel Handling Systems				
a. Losses during fuel preparation and injection	<365?(G)	<328?(AL)	<37?(G)	<365?
b. Losses during storage (assume storage as gas)	9,100?(G)	8,200?(AL)	900?(G)	9,100?
c. Leakage from isotopic separation processes	<365?(G)	<328?(AL)	<37?(G)	<365?
d. Maintenance on fuel handling systems	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
9. Waste Handling Systems				
a. Contaminated Clothing	---	37(S)	---	37
b. Contaminated pump oils	---	1,800?(L)	---	1,800?
c. Disposal of contaminated yttrium from tritium extraction beds	---	?	---	?
d. Disposal of contaminated charcoal from diverter	---	?	---	?
e. Disposal of contaminated ion exchange resins from Power Systems	---	?	---	?
Total for All Systems: (Curies/Year)	20,814	24,639.8	2,115.2	26,755

APPENDIX B

PRINCETON FUSION REACTOR

Detailed tabulations of the tritium releases and source terms for the Princeton Fusion Reactor are presented here in Tables B-1 and B-2. Table B-1 contains the data used to prepare summary Table 5-3 and includes an estimate for tritium releases from a gas phase storage system. Table B-2 was used to prepare summary Table 5-4 and also includes estimates of storage releases.

The differences between Tables B-1 and B-2 are found in permeation releases through the steam generators. In Table B-2 credit was taken for a factor of 100 reduction in the tritium permeation rate through the steam generators due to the presence of an oxide film. Details are discussed in Section 5.2.

TABLE B-1. Princeton Fusion Reactor Tritium Source Terms

Tritium Releases	Release To Containment	Stored On-Site (All in Curies/Year)	Release To Environment	Total Tritium
1. Blanket and Shield				
a. Diffusion from B&S into helium coolant	---	$7.3 \times 10^{+6}$ (AL)	?	$7.3 \times 10^{+6}$
b. Leakage from B&S fluid connections and penetrations	<37?(G)	<33?(AL)	<4?(G)	<37?
c. Maintenance on B&S equipment	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
2. Helium Coolant System				
a. Leakage from Helium Coolant System (purge helium)	730(G)	657(AL)	73(G)	730
b. Maintenance on Helium Coolant System	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
c. Recovery of ^3H from purge, diverter, and blanket helium streams for use as fuel	?	See 1, 3a and 7a	?	See 1a, 3a, and 7a
3. Piping Systems (He & Flibe)				
a. Diffusion through primary helium containments into helium purge stream	?	$7.3 \times 10^{+6}$ (AL)	?	$7.3 \times 10^{+6}$
b. Leakage from helium systems	-----	Included in 2a	-----	
c. Maintenance on helium piping systems	-----	Included in 2b	-----	
d. Diffusion through primary flibe containments into helium purge stream	-----	Included in 3a	-----	
e. Leakage from flibe systems pumps, valves, etc.	<37?(G,S)	<33?(AL,S)	<4?(G)	<37?
f. Maintenance on flibe systems	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?

G - Gas
L - Liquid
AL - Adsorbed Liquid
S - Solid
? - Estimate or Unknown

TABLE B-1 (continued)

Tritium Releases	Release To Containment	Stored On-Site	Release To Environment	Total Tritium
4. Heat Exchangers				
a. Diffusion through steam generator shell	-----	Included in 3a	-----	
5. Power Systems				
a. Diffusion through steam generators into power system, release via steam blowdown	---	---	2,200(G)	2,200
b. Leaks from Power System	<365?(G,L)	<328?(AL)	<37?(G)	<365?
c. Maintenance on Power System components	<365?(G,L,S)	<328?(AL,S)	<37?(G)	<365?
6. Tritium Extraction System (Flibe)				
a. Permeation of flibe-containing system boundaries by diffusion	-----	Included in 3a	-----	
b. Leaks and Maintenance	-----	Included in 3e and f	-----	
c. Process losses from tritium purification train	?	?	?	?
d. Permeation of flibe-containing system boundaries by TF corrosion	?	?	?	?
7. Diverter Systems				
a. Diffusion from diverter into helium coolant loop	---	43.8×10^6 (AL)	?	43.8×10^6
b. Permeation losses from Diverter System	?	?	?	?
c. Maintenance on Diverter System	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
d. Contaminated vacuum pump oil from diverter	---	2,900?(L)	---	2,900?

TABLE B-1 (continued)

Tritium Releases	Release To Containment	Stored On-Site	Release To Environment	Total Tritium
8. Fuel Handling				
a. Losses during fuel preparation and injection	<365?(G)	<328?(AL)	<37?(G)	<365?
b. Losses during storage (cryogenic liquid)	?	?	?	?
c. Leakage isotopic separations processes	<365?(G)	<328?(AL)	<37?(G)	<365?
d. Secondary Fractionator overhead stream	---	9(AL)	?	9
e. Drag Stream Fractionator waste	---	?	?	?
f. Maintenance on Fuel Handling System	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
9. Waste Handling				
a. Contaminated clothing	---	37(S)	---	37
b. Contaminated pump oils and mercury	---	1,800?(L)	---	1,800?
Totals for All Systems Less Tritium Recovered For Fuel: (Curies/Year)	4,089	8,421	2,614	11,035

TABLE B-2. Princeton Fusion Reactor Tritium Source Terms

Tritium Releases	Release To Containment	Stored On-Site (All in Curies/Year)	Release To Environment	Total Tritium
1. Blanket and Shield				
a. Diffusion from B&S into helium coolant	---	$7.3 \times 10^{+6}$ (AL)	?	$7.3 \times 10^{+6}$
b. Leakage from B&S fluid connections and penetrations	<37?(G)	<33?(AL)	<4?(G)	<37?
c. Maintenance on B&S equipment	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
2. Helium Coolant System				
a. Leakage from Helium Coolant System (purge helium)	730(G)	657(AL)	73(G)	730
b. Maintenance on Helium Coolant System	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
c. Recovery of ^3H from purge, diverter, and blanket helium streams for use as fuel	?	See 1, 3a and 7a	?	See 1a, 3a, and 7a
3. Piping Systems (He & Flibe)				
a. Diffusion through primary helium containments into helium purge stream	?	$7.3 \times 10^{+6}$ (AL)	?	$7.3 \times 10^{+6}$
b. Leakage from helium systems	-----	Included in 2a	-----	
c. Maintenance on helium piping systems	-----	Included in 2b	-----	
d. Diffusion through primary flibe containments into helium purge stream	-----	Included in 3a	-----	
e. Leakage from flibe systems pumps, valves, etc.	<37?(G,S)	<33?(AL,S)	<4?(G)	<37?
f. Maintenance on flibe systems	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?

G - Gas
 L - Liquid
 AL - Adsorbed Liquid
 S - Solid
 ? - Estimate or Unknown

TABLE B-2 (continued)

Tritium Releases	Release To Containment	Stored On-Site	Release To Environment	Total Tritium
4. Heat Exchangers				
a. Diffusion through steam generator shell	-----	Included in 3a	-----	
5. Power Systems				
a. Diffusion through steam generators into power system, release via steam blowdown	---	---	22(G)	22
b. Leaks from Power System	-----Negligible-----			
c. Maintenance on Power System components	<4?(G,L,S)	<3?(AL,S)	<1?(G)	<4?
6. Tritium Extraction System (Flibe)				
a. Permeation of flibe-containing system boundaries by diffusion	-----	Included in 3a	-----	
b. Leaks and Maintenance	-----	Included in 3e and f	-----	
c. Process losses from tritium purification train	?	?	?	?
d. Permeation of flibe-containing system boundaries by TF corrosion	?	?	?	?
7. Diverter Systems				
a. Diffusion from diverter into helium coolant loop	---	$43.8 \times 10^{+6}$ (AL)	?	$43.8 \times 10^{+6}$
b. Permeation losses from Diverter System	?	?	?	?
c. Maintenance on Diverter System	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
d. Contaminated vacuum pump oil from diverter	---	2,900?(L)	---	2,900?

TABLE B-2 (continued)

Tritium Releases	Release To Containment	Stored On-Site	Release To Environment	Total Tritium
8. Fuel Handling				
a. Losses during fuel preparation and injection	<365?(G)	<328?(AL)	<37?(G)	<365?
b. Losses during storage (cryogenic liquid)	?	?	?	?
c. Leakage isotopic separations processes	<365?(G)	<328?(AL)	<37?(G)	<365?
d. Secondary Fractionator overhead stream	---	9(AL)	?	9
e. Drag Stream Fractionator waste	---	?	?	?
f. Maintenance on Fuel Handling System	<365?(G,S)	<328?(AL,S)	<37?(G)	<365?
9. Waste Handling				
a. Contaminated clothing	---	37(S)	---	37
b. Contaminated pump oils and mercury	---	1,800?(L)	---	1,800?
Totals for All Systems Less Tritium Recovered For Fuel: (Curies/Year)	3,363	7,768	363	3,131

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L. R. Bunnell
L. L. Burger
S. H. Bush
N. E. Carter
T. D. Chikalla
R. G. Clark
T. L. Criswell
S. D. Dahlgren
M. T. Dana
D. E. Deonigi
R. L. Dillon
D. A. Dingee
P. J. Dionne
B. H. Duane
J. W. Finnigan
J. C. Fox
J. J. Fuquay
J. E. Garnier
R. D. Gastil
B. F. Gore
J. N. Hartley
A. J. Haverfield
U. P. Jenquin
A. B. Johnson, Jr.
R. H. Jones
10 T. J. Kabele
W. S. Kelly
H. E. Kissinger
D. A. Kottwitz
N. Laegried
B. R. Leonard, Jr.
D. L. Lessor
H. B. Liemohn
R. C. Liikala
M. A. McKinnon
R. F. Maness
R. P. Marshall

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E. S. Murphy
R. D. Nelson
D. F. Newman
R. E. Nightingale
D. E. Olesen
L. T. Pedersen
R. T. Perry
D. R. Pratt
L. A. Rancitelli
J. F. Remark
D. S. Renne
R. E. Rhoads
W. D. Richmond
W. F. Sandusky
L. C. Schmid
N. M. Sherer
E. P. Simonen
R. I. Smith
J. K. Soldat
C. W. Stewart
K. B. Stewart
R. W. Stewart
J. A. Strand
D. L. Styris
A. M. Sutey
V. L. Teofilo
G. L. Tingey
M. T. Thomas
R. C. Thompson
L. H. Toburen
T. J. Trapp
R. Wang
R. E. Westerman
L. D. Williams
10 J. R. Young
M. G. Zimmerman
1 Technical Publications (BH)
5 Technical Information