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and Plasma Physics (UC-20)
Issued: March 1975

**An Engineering Design Study of
A Reference Theta-Pinch Reactor (RTPR):
Environmental Impact Study**

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scientific laboratory
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ARGONNE, ILLINOIS

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ARGONNE NATIONAL LABORATORY
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An Engineering Design Study of A Reference Theta-Pinch Reactor (RTPR): Environmental Impact Study

A Joint Report

Prepared for the U.S. Atomic Energy Commission

by

**Argonne National Laboratory and
Los Alamos Scientific Laboratory**

by

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FOREWORD AND ACKNOWLEDGMENTS

The purpose of this report, Vol. II of LA-5336/ANL-8019, is to assess the environmental impact of the Reference Theta-Pinch Reactor (RTPR) power plant design that is described in Vol. I. Controlled thermonuclear research (CTR) is now in an optimistic growth phase in which serious consideration is being given to conceptual designs of fusion power plants as one aspect of a program to demonstrate the scientific feasibility of controlled fusion by about 1980-82. The conceptual reactor design detailed in Vol. I, therefore, makes possible an environmental assessment. Furthermore, the worldwide concern about the environmental impact of all energy-generating systems requires that the major CTR concepts — including the theta pinch — be subjected to serious studies of their possible environmental impact, if CTR is to continue to be a viable endeavor of the technologically advanced nations.

This statement of environmental impact pertains specifically to a 12 GWt (where Wt denotes watts thermal) fusion power plant detailed in Vol. I, and, therefore, generality or finality of conclusions based on this particular power plant design should not be

drawn; the results and conclusions contained herein should not be viewed as scalable quantities nor be used to infer environmental effects of other fusion power plant designs. In this context, Vol. II should provide feedback for future designs. Furthermore, the presentation here represents a first step in placing environmental studies for fusion reactors on a more quantitative basis. The results of this effort still depend on estimates and extrapolations, although areas where information and experience are lacking are clearly acknowledged in the hope that future, more detailed assessments will eradicate these uncertainties.

This study has been made jointly by the Los Alamos Scientific Laboratory (LASL) and the Argonne National Laboratory (ANL). The authors wish to acknowledge the significant contributions to this report by the following people:

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ABSTRACT

The recently completed engineering design study for the Reference Theta-Pinch Reactor (RTPR) has allowed an assessment of the potential environmental impact of an RTPR power plant to be made. During normal operation of the plant, tritium is expected to be released at a rate of 6 Ci/day, an amount that would lead to low maximum doses (0.06 to 0.8 mrem/yr, depending on site and cooling options). These doses, and the anticipated integrated population doses, are considerably less than doses now considered acceptable.

Problems related to (i) the required commitment of some natural resources (e.g., beryllium and niobium), (ii) the disposition of activated structural materials and other radioactive waste (76,700 kg/yr or 12 GCi/yr at shutdown) for a five-year niobium component lifetime, and (iii) land despoilment are substantial but do not appear to compromise the viability of RTPR's as a useful power source. Consideration of the occurrence of severe accidents (e.g., liquid-metal fires) and other unusual incidents indicate that their effects will be contained within the plant, but that they may be costly. Radiation doses resulting from accidental release of the operating tritium inventory during a liquid-metal fire would probably be below the level now permitted for fission plants.

1. INTRODUCTION

Since the beginning of controlled thermonuclear fusion research in the early 1950s, proponents have been fascinated with the overwhelming benefits of fusion energy: the worldwide availability of a potent, cheap, and abundant fuel to fulfill mankind's energy requirements. Furthermore, early enthusiasm for fusion power gave rise to forecasts of "pollution-free" energy sources, and to some extent this attitude persists.

However, the cheapness, abundance, and environmental impact of fusion power is critically dependent upon the fuel cycle used. The use of tritium-deuterium (D-T) fuel requires that the tritium be bred from lithium; the use of tritium in large quantities poses a potential environmental effect. In addition to being radioactive (tritium is a beta emitter with a 12.3-yr half-life), the D-T reaction produces more high-energy neutrons than, for instance, the fusion of D-D. In addition to the effects of tritium, the activation of structural materials and impurities by energetic neutrons represents a major contribution to the environmental impact of a D-T fusion reactor. Because a fusion power plant generates ~5 times more neutrons than a fission power plant for the same amount of thermal power, structural activation (Ci/Wt) in fusion reactors is expected to be equal to or greater than in fission reactors. The biological and afterheat problems associated with radioactive products, however, do not appear as severe. In spite

of the environmental advantages offered by the D-D fuel, however, the D-T cycle is considered here because of its relative ease of ignition and high power density.

Environmental effects in which the fusion power plant differs little from present-day generating stations (waste heat, chemical waste, land allotment, etc.) are included for the sake of completeness but are only briefly treated. The environmental impact of a theta-pinch fusion power plant has been evaluated herein the basis of an engineering design described in Vol. I.¹ This design is based on extrapolations of plasma stability criteria, techniques presently used in Scyllac experimentation, material properties, high-voltage technology, and tritium chemistry. A discussion of the effect of these extrapolations and other technological problems upon the design of the power plant will form the basis for Vol. III of this report. Volume III of this report assesses technological problems which have been quantified by this conceptual design, and the research and development required for solving these problems are summarized. Generally, extrapolations of technology can be accommodated by slight design changes. This flexibility in design also exists to mitigate or alter undesirable effects upon the environment. Thus, the environmental impact described herein is strongly dependent upon the reactor design and should not be considered unalterable.

2. DESCRIPTION OF TWO TYPICAL REACTOR SITES

A realistic assessment of the effect of a power plant on the environment should be based on an environmental model which can be described in some detail. To facilitate such an evaluation two specific locations have been chosen: the sites for the Bailly and the Oconee Nuclear Power Stations. The Bailly site is presently under consideration for the installation of a fission nuclear power plant, and the Oconee plant is operational. Hence, environmentally significant features for both sites have been thoroughly documented and investigated. These sites were chosen for this study to represent the contrast of possible typical locations having a high population density (Bailly) and a relatively low population density (Oconee).

The 350-acre Bailly site² (Site 1) is located on the shore of Lake Michigan about 30 miles southeast of downtown Chicago. The Illinois-Indiana boundary is about 21 miles due west; Gary, Indiana is about 12 miles to the WSW; Michigan City, Indiana, is about 13 miles to the ENE; and the Indiana-Michigan boundary is about 18 miles ENE. This site is situated at a lake-shore interface between a 25-mile industrial complex (steel mills and other heavy industry) on the west and a 12-mile recreational region (state and national park areas and private residences) on the east. Population (1970) within a 10-mile radius is 83,600, and within a 50-mile radius the population is 7.3

million. Figure 2-1 gives geographic, population, and meteorological maps of Site 1.

The Oconee site³ (Site 2) is located in Oconee County, South Carolina, within 25 miles of the North Carolina and Georgia state boundaries. Oconee is at the southern reach of Appalachia where the Piedmont Hills join the Blue Ridge mountains to form the Piedmont Crescent of South Carolina. The largest city within 100 miles (97 miles NW) is Knoxville, Tennessee, with a population (1970) of about 175,000. The nearest urban center is Anderson, South Carolina (21 miles away) with a 1970 population of about 28,000. In 1970, about 36,000 people lived within a 10-mile radius; about 694,000 people live in the 0- to 50-mile radius. Figure 2-2 gives the geographical, population and meteorological maps for Site 2.

Figure 2-3 gives the average temperature and humidity for Sites 1 and 2. The population, meteorological, and hydrological data available for Sites 1 and 2 will be used in evaluating design and accident conditions with respect to environmental effects; the degree of public hazards created by the operation of a RTPR at each specific site will be evaluated both in terms of individual and annual integrated population (AIP) doses.

Fig 2-1A.
Geographical map of the Bailly site

Fig 2-1C
Wind Rose for the Bailly site (Ref 4).

Fig. 2-1B.
Population map of the Bailly site.

Fig. 2-2A.
Geographical map of the Oconee site.

Fig. 2-2C.
Wind Rose for the Oconee site (Ref. 5).

Fig. 2-3.
Summary of daily averaged temperature and
humidity data for Site 1 and Site 2 (Refs. 4,5).

3. PLANT DESCRIPTION

3.1 OVERALL DESIGN

This chapter summarizes environmentally relevant features of the RTPR power plant design. Details of this design are described in Vol. I of this report.¹

The RTPR power plant is composed of 10 major systems, as indicated diagrammatically on Fig. 3.1-1. Details of the plant layout are shown in Figs. 3.1-2, 3.1-3, and 3.1-4. The theta-pinch reactor is in the form of a toroid with a major radius of 56 m and a minor radius of 50 cm. The torus is formed from 176 reactor modules* about 2 m long, as shown in Fig. 3.1-5. Each reactor module consists of 100 wedge-shaped blanket segments encircled by the implosion heating and adiabatic compression coils. Structural support around the adiabatic compression coil is

*In order of increasing radius the major components of a reactor module are: plasma chamber, first wall (insulator, metal composite), blanket, implosion heating coil, adiabatic compression coil, titanium support hoop, and vacuum envelope. The word "reactor" is used only to describe the assemblage of reactor modules, i.e., all components surrounded by the vacuum wall. The words "power plant" describe the entire system shown in Fig. 3.1-1.

Fig. 3.1-1.

Schematic of RTPR power plant subsystems.

provided by a titanium hoop external to the compression coils as shown in Fig. 3.1-5. The modular approach was selected for ease of maintenance and assembly and leads naturally to enclosure in a vacuum for protection of the refractory metals from oxidation, prevention of tritium leakage, and provision of vacuum and fueling requirements. The biological shield is, therefore, the vacuum and "cold-wall" boundary; the reactor modules contained within it are functionally independent of each other and are connected only to individual fluid lines and electrical leads. In addition to alleviating thermal expansion problems, this arrangement permits the vacuum to be maintained when replacing a defective reactor module.

Provisions for access to the reactor are made from above the 2-m thick concrete biological shield. A continuous chamber above the reactor allows the use of a remotely operated machine capable of removing and installing reactor modules. This module replacement machine also serves as a "coffin" to transfer radioactive reactor modules to the hot-cell area and to supply replacement units (Fig. 3.1-2). A rotary vacuum-to-atmosphere transfer cell is provided for this purpose. The access area is separated from the reactor cell by isolation plates to make the critical vacuum pumping volume as small as possible and to avoid dispersion of reactor fuel. Specific values for design parameters are summarized in Appendix A.

3.2 TRITIUM HANDLING SYSTEMS AS THEY RELATE TO POTENTIAL LEAKAGE PATHS TO THE ENVIRONMENT

Figure 3.2-1 delineates those components of the RTPR power plant that potentially could release tritium to the environment. Regions of the plant with high tritium concentrations are either under vacuum or are purged continuously by an inert gas (helium). The reactor is not impervious to tritium, and significant tritium leakage is expected to occur in this region. The reactor is situated within a vacuum chamber defined by the cool, steel lining of the biological shielding, this lining represents the

Fig. 3.1-2.
Cutaway view of RTPR power plant.

Fig. 3.1-3.
Floor plan of RTPR power plant.

Fig. 3.1-5.
Cutaway view of RTPR module.

Fig. 3.2-1.

Diagram of RTPR power plant showing purge system and zones of major tritium content.

primary tritium barrier surrounding the reactor core. Tritium injected and generated in the reactor must (1) enter the vacuum system, (2) pass through a helium-purged room surrounding the biological shield, or (3) be transported by the lithium-sodium-steam coolant system to the turbogenerator area. A major portion of the bred tritium is separated from the lithium coolant by the tritium processing system (Fig. 3.1-1).

Four isolated volumes within the RTPR power plant can be identified which contain significant tritium inventories and therefore are subject to helium purge.*

*The design under consideration purges the entire room(s) with helium. The use of helium purge jackets around critical components is an alternative, helium-conserving option.

These regions are indicated in Fig. 3.2-1 and quantified in Table 3.2-1. Although the helium-filled buffer region around the biological shielding is represented in Fig. 3.2-1 as an isolated region, reference to Fig. 3.1-3 will show that portions of this region are defined by other helium-filled areas. The helium purge system forces gas continually through the lithium process area, the reactor-core buffer region, the access tunnel, and finally back into the chamber which houses the fuel and ash processing, fuel injection, and fuel storage systems. Tritium contained in the helium-purge stream is removed as tritiated water (THO) on an ion-exchange column, and the purified helium is again recirculated.

Tritium permeating through the lithium-sodium heat exchanger, subsequently exchanging with protium, and entering into the steam system

TABLE 3.2-1
DESCRIPTION AND VOLUME OF MAJOR TRITIUM
ZONES IN THE RTPR

Region	Region Description	Volume ^a (m ³)
1	Buffer region surrounding the RTPR core and biological shield.	18,280
2	Access tunnel below core and biological shield which houses vacuum ducts and pumps.	31,280
3	Process room which houses fuel and ash processing system, fuel storage, fuel injection system, and helium purge-gas system.	51,660
4	Process room which houses lithium processing and lithium-sodium intermediate heat exchanger.	176,250
Total		277,470

^aSee footnotes p. 22.

represents a potential leakage path of tritium into the environment. This leakage may occur either by leakage directly from the turbine or from the steam generator through the condenser cooling water during blowdown operations. These leakage paths are indicated on Fig. 3.2-1.

Because tritium release is a potentially serious problem, the present RTPR power plant design includes a number of features to minimize the operational release of tritium. The maximum permissible tritium concentrations recommended by the International Commission on Radiological Protection (Table 4.1-1) were chosen to establish on a preliminary basis the following design requirements.

(1) The molten-salt extraction system (described in Vol. I, Sec. 4.9.2) must operate with a tritium chemical activity in the lithium at a level corresponding to a uniform partial pressure throughout the blanket of less than 10^{-10} Torr. The molten salt extraction process proposed in Vol. I potentially is capable of achieving the required tritium (chemical) activity, but more thermodynamic data are needed before definite conclusions can be made.

(2) The gaseous purge system must recover tritium that permeates through the outermost containment of the tritium handling systems. By adjusting the oxygen level in the helium, the purge flow can be operated with a partial pressure of molecular tritium (HT, DT, or T₂) below about 10^{-15} Torr (Appendix B). The outer containment of the purge system is

coated or laminated with materials relatively impermeable to molecular tritium if cold-wall techniques to inhibit tritium permeation prove unacceptable. The flowing helium stream is processed continuously to recover the tritium (THO).

(3) The ceramic liner that provides electrical insulation between the plasma and the first metallic wall should limit permeation of fuel through the first wall. Hence, the flux of deuterium and tritium from the plasma to the blanket probably can be ignored, despite the large concentration gradient. The ceramic liner, however, must be continuous, must not spall or crack during operation, and must have an extremely low hydrogen permeability. Tritium leakage between the plasma and blanket does not represent a leakage path to the atmosphere, but establishes a trade-off with the tritium extraction system; large quantities of hydrogen isotopes diffusing into the blanket region may compromise the ability to maintain the tritium partial pressure in the blanket region at or below 10^{-10} Torr.

(4) Primary and secondary heat exchangers must be provided with hydrogen diffusion barriers, i.e., they must be constructed from composites of two or more metals. The surfaces of these composites must be compatible with the contacting fluids, and at least one of the metals of the composite must be relatively impermeable to hydrogen. Detailed calculation of the sodium-steam heat exchanger permeation rates are given in Appendix C.

The four helium purge zones and the Li-Na-H₂O coolant circuit represent all potential sources for tritium leakage. The coolant circuit is the only interfacial path not amenable to disengagement from the environment by encapsulation in a migration barrier formed either by a relatively impermeable coating, a purge jacket, or both. Since continuous leakage of water from the steam system to the environment will occur, it is presumed that all the tritium entering this system will eventually escape.*

To facilitate an effective control and inventory of tritium throughout the plant, a "mapping" model has been developed which accounts for tritium inventories in and release rates from all systems of the RTPR. Although this mapping model (Appendix C) does not describe all forms of the tritium in the RTPR power plant (oxides, hydrides, adsorbed, absorbed, gaseous, liquid), the model is adequate to assess major leakages of tritium by the lithium-sodium-steam circuit.

3.3 ENERGY CONVERSION AND REJECTION OF THERMAL ENERGY

As is the case for any power plant operating on a thermal cycle, the RTPR power plant must reject an appreciable portion of generated thermal energy to the environment. The significant amount of direct-conversion energy** generated by the RTPR makes possible a major reduction of the recirculating power, keeping the circulating power fraction to a tolerably low value.

A conventional Rankine steam cycle is used as the thermal converter. The high-temperature, potassium topping cycle discussed as an alternative power conversion system in Vol. I is not considered in this environmental study. Lithium is the primary coolant, absorbing the fusion neutron energy and cooling the

reactor first wall. The lithium is pumped to a heat exchanger where it transfers heat to a secondary coolant (sodium) loop. The sodium in turn flows to a steam generator that generates steam to drive four turbogenerators. An energy balance for the RTPR power plant is given in Fig. 3.3-1. The plant gross thermal efficiency is 40%, and a gross power of 4800 MWe is produced; a net power of 4030 MWe is available. With a circulating power fraction of 0.16 the overall plant efficiency is 33% and a total of 8873 MWt appears as waste heat. The RTPR power plant directly converts an equivalent 1150 MWe, which amounts to 24% of the gross electrical power.

The steam-generating system was chosen to be nearly identical to that originally proposed for the demonstration LMFBR. Although this choice results in some sacrifice in thermal conversion efficiency (more waste heat must be accepted) the design detail available for the sodium-steam cycle lends considerable credence to design data on the power plant portion of the RTPR. A piping schematic is shown in Fig. 3.3-2. Specific design parameters will be introduced in the following sections as required and detailed design numbers are given in Appendix A. Volume I also considers an advanced, high-temperature scheme (potassium topping cycle) to achieve increased thermal conversion efficiencies, thereby further mitigating environmental effects caused by thermal discharge. Considerable technical development is required before this high-temperature system becomes a reality.

As previously mentioned, tritium that reaches the steam system is assumed lost to the environment. This loss could probably be reduced to a negligible level if the steam working fluid were replaced by an inert gas. The use of a helium-turbine power-conversion system and helium intermediate coolant loop is dis-

*Although not considered as an alternative in this study, the possibility exists of totally containing the tritium in a leakproof steam system. The tritium concentration in the steam system would be determined by the equilibrium established between the leakage into the steam system from the sodium system and the radioactive decay of tritium. If a means could be found to retain all tritium in the steam system, the total tritium in this system would build-up to ~7 g (12.6 mCi) (1).

**Of the 1850 MWt contributed by the alpha particles to the total 12 GWt power generated by the RTPR, approximately 62% appears as direct-conversion power and is fed directly to the magnetic energy storage system.

Fig. 3.3-1
Summary energy balance for the RTPR power plant

Fig. 3.3-2.
Piping diagram for RTPR power plant.

cussed in Vol. III, and it is believed that the tritium losses from this plant can be greatly reduced with only a small penalty in reduced thermal efficiency.

Tritium clean-up of the steam system presents an additional means of curtailing tritium leakage to the environment. A tritium purification device is now being marketed by Sulzer Bros., Inc.⁶ in Switzerland which reduces the THO concentration in H₂O by a

distillative separation process. This separation process, however, does not seem practical for the purposes of this study because of the low tritium level desired in the steam plant. Technological advances in this distillative process, however, might make this technique of tritium clean-up economically attractive in the future.

4. ENVIRONMENTAL EFFECTS OF NORMAL OPERATION

4.1 TRITIUM

4.1.1 Radiology of Tritium in the Environment⁷⁻¹⁴

Some pertinent data related to the radiology of tritium are given in Table 4.1-1. Tritium decays to helium-3 by beta emission with a half-life of 12.3 yr. The beta particle has a maximum energy of 18.6 keV and an average energy of 5.7 keV. Chemical equilibria in the presence of water are given by the following relationships.

Since the mass-action equilibrium coefficients (Table B-3) for these reactions favor, respectively, the formation of HT and HTO at ~300 K, most of the gas-

eous tritium released from a thermonuclear power plant is expected to be found in the environment as tritiated water.^{8, 10, 13, 4}

The biological half-life of tritium ingested into the body as tritiated water is about 12 days. Little evidence exists to suggest that tritium entering the body in this form is assimilated into organic substances *in vivo*. Evidence does indicate, however, that other forms of plant and animal life do assimilate tritium from tritiated water in biologically active organic substances.¹² Should tritiated proteins enter the human body in vegetable, dairy, or meat products, the in-body residence time for tritium in human tissues might be increased considerably. As a result, the decay of tritium so incorporated in biologically active substances could conceivably result in increased somatic and genetic (e.g., tritium in thymidine) damage to body tissue. Clearly, additional experimental work is needed to elucidate the effects of low levels of tritium on biological systems.

TABLE 4.1-1

RADIOLOGIC PROTECTION DATA FOR ELEMENTAL TRITIUM AND TRITIATED WATER^a

	Form of Tritium	
	HTO or T ₂ O	HT or T ₂
Biological Half-life	12 days	---
Critical Organ	Body Tissue	Skin
Dose Limit	0.5 rem/yr	3 rem/yr
Maximum Permissible Concentration in Air	2×10^{-7} Ci/cm ³	4×10^{-5} Ci/cm ³
Maximum Permissible Concentration in Drinking Water	3×10^{-3} μ Ci/cm ³	---

^aData taken from Ref. 15. The indicated dose limits apply to routine exposure of members of the public as recommended by the International Commission on Radiological Protection.

Releases of tritium from thermonuclear power plants to the environment must be sufficiently low so that tritium concentrations in man or his food chains do not increase to critical levels. The maximum permissible water and air concentrations listed in Table 4.1-1 reflect this concern. Since tritium appears in the environment predominantly as tritiated water (THO), the related radiation dose to man is strongly dependent on the local hydrologic cycle. An atmospheric source of tritium may contaminate a local landscape either by direct deposition of moisture or by exchange of tritiated water molecules of the atmosphere with ordinary water of soils, plant, animals, and surface waters. Once contaminated, soils, plants, and animals can carry tritium to man through the food chain. An example of such a pathway is the soil-plant-cow-milk-man food chain. The total dose to man from such a pathway depends on the turnover rate of tritium in each of the links of the pathway. In general, the overall turnover rate in a given eco-pathway depends on the rate of the slowest transport mechanism. Tritiated soil water has turnover times ranging from 24 days in the tropics to 1.7 yr in the desert. In temperate zones, turnover time for soil water in the rooting zone of plants may vary from 30 to 120 days, depending on location, soil type, and season of the year. Intrinsic turnover rates of plants, animals and man are much faster.

Tritium is considered to be a low-hazard isotope because of its low beta energy and its relatively short residence time in man. When the longer residence-times associated with certain food chains are considered, the conclusion that tritium is a low-hazard isotope is still valid. Anspaugh *et al.*¹⁶ have calculated an anticipated dose to man from all avenues of exposure including foods, inhalation, and skin absorption. These calculations assumed a steady concentration of 10^3 pCi per cubic meter of air and concluded that men living in such an environment and deriving their food from it would acquire an annual dose of 3.7 mrem/yr. The data of Anspaugh *et al.*¹⁶ indicate that when ecological pathways to man are considered, approximately 40% of the total dose to man from tritium comes from inhalation and skin absorption, the remaining 60% being attributable to ingestion of tritiated meat, milk, and vegetables. Tritium release from a fusion reactor will result in local contamination; the release point represents a point source and tritium fall-out by precipitation is rapid.^{17,18} In principle, therefore, the food chain pathway should be less significant in delivering tritium to man, in that the derivation of appreciable amounts of food from the area close to the point of release seems unlikely. Far from the release point, the natural dilution of tritium in both ground waters and air should

significantly mitigate the effects of biological concentration.

4.1.2 Tritium Distribution and Leakage

The model used to describe the tritium distribution and leakage profiles in the RTPR power plant is described in this section. These profiles are determined by the detailed technical and design features of the RTPR power plant and ultimately have an important bearing on the operational characteristics of the RTPR power plant. The motivation for investigating the location and quantities of tritium in D-T fueled fusion reactors as they are currently conceived can be simply stated as follows: First, the plant design must limit the tritium release rate to the lowest practicable level. Secondly, some guidance on how low the release rate must be can be developed from the concentrations listed in Table 4.1-1, using the permissible tritium concentration in air and an atmospheric diffusion model.

In addition to environmental considerations, other potentially significant consequences of the tritium in fusion reactors can be identified. These include hold-up of tritium in the structure (particularly niobium used in the blanket, coolant pipes, primary heat exchanger, and lithium processing equipment), accelerated corrosion and loss of mechanical integrity resulting from hydriding, and assorted maintenance problems. These influences couple with radiological considerations to create an incentive to minimize the tritium inventory.

The principal objective with regard to the control and handling of tritium will be to minimize the total tritium inventory and loss rate in each portion of the power plant. As outlined in Sec. 3.2, minimization of both the tritium inventory and the loss rate will be accomplished by (1) effectively processing the lithium blanket to achieve the lowest possible concentration of tritium, (2) employing a contiguous system of barriers (both metal and ceramic) that are impervious to the hydrogen isotopes in any form, and (3) continually operating a flowing, gaseous purge throughout the plant. As will be shown in subsequent analyses, the principal tritium loss path in the RTPR power plant is through the heat transfer systems and into the steam system; for this reason, emphasis is placed on this particular tritium loss path.

The properties that determine the solubility and permeation rate of hydrogen through structural metals of interest to fusion reactor technology are treated in Appendix B. The calculational model used to "map" the tritium in important RTPR power plant systems is described in Appendix C. The results of

these numerical analyses are presented in this section, and the subsequent individual and population doses are estimated for Site 1 and Site 2.

4.1.2.1 Model Used to Map Tritium Distribution in RTPR Power Plant. Figure 3.2-1 illustrates schematically a coarse zoning on the tritium handling system within the RTPR. Four critical regions can be identified: (1) buffer region surrounding biological shielding, (2) vacuum pump tunnel, (3) fuel processing room, and (4) lithium processing and heat exchanger room. Within any of the four zones are located various tritium-containing units or subsystems. For the purposes of the calculational model, any given subsystem is classified into one of two categories: *isobaric* or *interfacial*. An isobaric system contains hydrogen (isotopes) at a constant chemical activity, and, therefore, the partial pressure of hydrogen bearing species is constant. The lithium and niobium are considered isobaric systems. Interfacial systems are those which support a hydrogen (chemical) activity gradient, and this gradient drives a flow of hydrogen through the interface. The sodium-steam heat exchanger and the containment walls of the four zones described above represent examples of the interfacial category.

The evaluation of the tritium inventory and leak rate for every subsystem within the RTPR is an ultimate goal, but the power plant design is not sufficiently advanced to allow this kind of detailed analysis. As noted in Sec. 3.2, however, a major contribution to the environmental impact by tritium is expected to occur by the lithium-sodium-steam coolant circuit.

4.1.2.2 Tritium Leakage for the Li-Na-H₂O Path. The RTPR uses eight heat-exchanger steam-generator circuits. Components of these circuits which handle lithium will be constructed entirely from niobium. The transport of tritium through the lithium-sodium heat exchanger, through the steam generator, and into the steam system represents a potentially serious tritium leakage path. Design parameters for the lithium-sodium heat exchanger and the sodium steam-generator are summarized in Table 4.1-2. The sodium-steam interface includes a copper diffusion barrier to limit the rate at which tritium enters the steam system (and eventually the environment).

The solubility and permeability data given in Appendix B (Tables B-1 and B-2) are used in conjunction with the model described in Appendix C to estimate the rate of tritium transport into the steam

TABLE 4.1-2
PARAMETERS USED FOR TRITIUM MAP IN THE RTPR HEAT EXCHANGER

Exchanger	Composite Layer	Total Area ^a (m ²)	Thicknesses (mm)	System Temperature ^b (K)	
				Highest	Lowest
Li-Na	Nb ^c	3.7 x 10 ⁴	0.9	814	683
	304-SS ^d	3.7 x 10 ⁴	1.0	794	622
Na-steam	Incoloy-800 ^d	4.0 x 10 ⁴	1.0	794	622
	Copper ^e	4.0 x 10 ⁴	2.5	794	622
	Incoloy-800 ^f	4.0 x 10 ⁴	1.0	757	533

^aThe combined area of eight independent units.

^bThe temperature profile was determined by dividing each heat exchanger wall into eight isothermal zones. The fractional areas from the lowest through the highest temperature zones were as follows: 0.05, 0.10, 0.15, 0.20, 0.20, 0.15, 0.10, 0.05.

^cLithium side.

^dSodium side.

^eIntermediate layer.

^fSteam side.

Fig. 4.1-1.
Tritium map used for RTPR coolant circuit.

system. Figure 4.1-1 illustrates the calculational geometry and notation. Detailed results of these computations are given in Appendix D.

The partial pressure of tritium in the lithium was fixed at 10^{-10} Torr, in anticipation of the low limits of the molten-salt extraction process (Vol. I). No provisions have been made for tritium removal from the sodium. As a result, the partial pressure of tritium in the sodium should reach but not exceed 10^{-10} Torr.

This situation requires a low leakage into the steam cycle and continual monitoring of the steam system. The results of a series of calculations with and without a copper diffusion barrier in the sodium-steam heat exchanger are summarized in Table 4.1-3. The copper layer provides an additional barrier to tritium migration if the continuity of the laminated structure is maintained. Figure 4.1-2 gives the tritium leakage rate vs thickness of copper. It should be

TABLE 4.1-3
SUMMARY OF CALCULATIONS
ON THE SODIUM-STEAM HEAT EXCHANGER FOR THE RTPR

Na-Steam Heat Exchanger Wall	Daily Tritium Leakage To Steam System	
	grams	Curies
With 2.5-mm-thick copper layer (99% effective)	$< 6 \times 10^{-4}$	< 6
Without copper layer	$> 3 \times 10^{-3}$	> 30

Fig. 4.1-2.

Tritium leakage rate vs thickness of copper diffusion barrier.

noted that portions (flanges, weldments) of the primary heat exchanger will not be amenable to the maintenance of a continuous copper barrier. Since the total area where the copper barrier will not be effective is estimated to be $\sim 0.01\%$, or considerably less than 1% of total heat-exchanger area, this lack of continuity is neglected in the calculations.

The effectiveness of the copper diffusion barrier in containing tritium is surprisingly good. Although the curve presented in Fig. 4.1-2 is calculational, ongoing work at ANL generally substantiates this behavior. Figure 4.1-3 illustrates the good bond integrity of 316-SS/Cu/316-SS composites. Although the application of pressure bonding techniques to the construction of "triplex" steam generators represents an extension of technology, no insurmountable problems are foreseen. Figure 4.1-3 also shows a pressure-bonded composite of Nb/304-SS of the kind which will be required by the primary lithium-sodium heat exchanger.

4.1.2.3 Miscellaneous Wall Leakages to the Purge System. The leakage of tritium through coolant piping walls and other tritium-lithium containment structures and handling equipment into the helium purge flow system is estimated for a 304SS structure of total area $4 \times 10^5 \text{ m}^2$ and 10 mm

Fig. 4.1-3.

Photomicrographs of pressure bonded 316-SS Cu 316-SS and Nb 304-SS composites.

thickness.* The area and thickness are commensurate with present and projected design details for most of the important systems. The temperature used (800 K) is somewhat higher than the anticipated average temperature of that portion of the containment structure enshrouded by the helium purge flow system; hence, the results of these leakage calculations should be conservative.

This model uses the parameters for tritium permeation through 304-SS given in Table B-2 and an average tritium driving pressure of 10^{-6} Torr (a conservatively high value). The daily tritium leakage to the purge flow system is estimated to be ~ 2 Ci/day (2×10^{-4} g/day). Although this leakage rate is small compared to the daily tritium injection rate (28.8 kg/day) or breeding rate (> 1.58 kg/day), the tritium collected in the purge gas must still be recovered, processed, and returned to the fueling system to avoid long-term economic penalties and additional waste disposal problems.

4.1.3 Calculation of Tritium Inventory and Release Rates

Table 4.1-4 gives a breakdown of the tritium inventory in the RTPR. A (maximum) total volume equal to $277,500 \text{ m}^3$ is under helium purge and in one form or another the RTPR contains a total of ~ 3800 g of tritium (Table 5.1-1); 53 g in the blanket, 404 g in lithium which is external to the blanket, 1426 g in storage,** 2 g dissolved in the niobium structure, and

* All niobium piping external to the vacuum region is externally clad with a thin layer of stainless steel to provide oxidation resistance. This design is not finalized, but insulation may be placed between the niobium and steel, or a low-pressure purge gas could be interspersed. It should also be noted that all rooms with high tritium concentration or components handling high concentrations of tritium will be surrounded by a buffer room purged with helium. Furthermore, both rooms will be lined with a tritium barrier, e.g., aluminum. In addition to the purged buffer rooms, a helium purge stream will either flow through the room in which the processing equipment is situated or flow through an external jacket which surrounds the apparatus. For calculational purposes, flow through the room is selected as being calculationaly more conservative, in that a greater tritium leakage results.

**Contained in double-wall I.C.C. Shipping containers or possibly as a metal hydride.

the remainder associated with the fueling, vacuum, helium purge, and secondary coolant systems.

The tritium leakage to the steam system is assumed to be released to the environment as THO. Except for small amounts of water vapor released with gaseous discharge from the ejector and other leaks (turbine gland-seals, etc.) tritium release will be in the form of liquid water added to the condenser circulating water.

To determine the tritium leakage to the atmosphere from each of the four tritium zones identified in Table 3.2-1, a leakage time constant, λ_i (volume fraction/day), and a leakage rate of subsystems into a given volume, S_i (g/day), must be specified. A first-order balance equation for each tritium zone could be written based on the helium purge configuration depicted in Fig. 4.1-4.

The design of the RTPR power plant is not sufficiently advanced to estimate the leakage rate constants, λ_i , or the subsystem leakage rates, S_i . A very rough estimate of S_i (lithium processing-heat exchange zone) indicates a leakage rate of 2 Ci/day into the helium-purged chamber to be reasonable. On the basis of existing containment technology for fission reactors,¹⁹ values for λ_i on the order of 10^{-3} /day are considered good. Assuming an average tritium concentration, $\langle x \rangle$ in kg/m^3 for all four tritium zones, the total tritium leakage to the atmosphere equals $2.78 \times 10^9 \langle x \rangle$ (Ci/day).

The maximum permissible concentration (MPC) of tritium in air based on the dose to people living in that air amounts to $40 \mu\text{Ci/m}^3$ (Table 4.1-1), which corresponds to 4.31×10^{-8} Torr at 1 atm. Since the volumes under helium purge will be subject to entry for periodic maintenance, the tritium levels within this volume cannot be too high. Selecting $\langle x \rangle$ equal to the MPC gives a leakage rate to the environment of 0.11 Ci/day.

Since tritium leakage through the coolant circuit (~ 6 Ci/day) far exceeds the atmospheric leakage anticipated from the helium purge system, the lack of design details on process equipment not related directly to the coolant circuit does not have a significant influence on calculated results for steady-state releases. This situation, however, may not be valid during off-normal operation, particularly during periods of maintenance and repair; some of the most significant releases of radioactivity from fission reactors have occurred during periods of shut-down for routine repair and maintenance. The situation in which a significant, instantaneous release of the tritium inventory may occur is treated in Sec. 5.

TABLE 4.1-4
SUMMARY OF TRITIUM INVENTORY IN THE RTFR

Tritium Zone Subsystem Component	Volume m ³	Mass ^b kg	Tritium Inventory ^c (grams)
Reactor	918	...	54.7 (subtotal)
*Plasma Chamber	878	...	0.99
*Blanket	842	1,217,300	53.4 (subtotal)
-Niobium	44	383,680	0.38
-Beryllium	28	51,800	
-Graphite	301	681,650	
-Lithium-7	119	51,170	27
-Lithium-6	114	49,000	26
*Vacuum space	1,412,782	...	0.32 (subtotal)
-Adjacent core	2,782	...	0.31 ^d
-Atop core	1,410,000	...	0.02 ^e
Lithium Processes			
*Primary coolant (Niobium)	21.8	2,202,300	2.20 (subtotal)
-Headers	41.0	356,700	0.36
-Manifold	8.9	77,600	0.08
-Miscellaneous piping	40.5	352,000	0.35
-IHX	93.3	812,000	0.81
-Lithium processing ^g	47.0	540,000	0.54
-Lithium pump	5.1	58,000	0.06
*Primary coolant circuit (Lithium)		863,550	404.6 (subtotal)
-Manifold	73.5	30,632	16.2
-Headers	446.4	183,924	98.5
-Miscellaneous piping	29.5	12,273	6.5
-IHX	730.7	304,363	161.3
-Lithium processing ^g	490.	200,900	106.5
-Lithium pump	72.	29,450	15.6
Buffer zone around Biological Shield	18,280	...	0.00 ^f
Vacuum pump tunnel	31,280	...	0.00 ^f
Vacuum ducts	680	0.010	5.04 ⁱ
Vacuum pumps	148	0.002	1.18 ^j
Fuel processing zone	51,660	...	3339 (subtotal)
*Extraction system	...	...	270
*Fuel ash process	...	...	143 ^g
*Fuel storage	...	...	1426 ^h
*Fuel gas blanket injection	...	...	1500 ⁱ
*Helium purge	51,660	...	0.00 ^j
Secondary coolant			
*Sodium	...	3,026,000	0.3 ^k
*Water	...	5,000,000	7.0 ^l
Total	...	...	3814.1

^aVolume of lithium processing unit sized for 10% bypass flow.

^bTotal lithium mass ~ 863,550 kg.

Total niobium mass ~ 2,585,300 kg.

^cTritium concentration in lithium equals 0.53 ppm ($P_H = 10^{-10}$ Torr at 800 K)

Tritium concentration in niobium equals 10^{-5} ppm (800 K)

^dComposition assumed to be same as plasma (50% D, 50% T) but pressure is a factor of 10 less

^eTritium partial pressure assumed to be 10^{-10} Torr (300 K)

^fTritium partial pressure assumed less than 10^{-15} Torr as a result of trace oxygen impurities in helium

^gTritium in fuel-ash processing equal to 10% tritium in storage

^hTritium storage equal to 1 day fuel supply (0.017 g/s at 5% burn-up)

ⁱEqual to the total tritium required for 300 pulses (15 in)

^jThe nominal pressure in the pump throat and inlet ducts is 0.1 Torr. A 50% D-T mixture is assumed to exist at 300 K.

^kAssumed concentration equal to 0.2 ppb.

^lThis value represents a worst case, where the in-leakage to the steam system equals the natural decay of tritium (i.e., a completely closed system). For the actual case considered here, this value will be near zero.

Release to the Atmosphere. The atmospheric diffusion model of Pasquill²⁰⁻²² is used to calculate the maximum tritium concentration in the atmosphere at ground level; from these concentrations the biological dose is estimated. The magnitude and location of the maximum tritium concentration varies with wind direction, wind speed, and other atmospheric conditions. In calculating the concentrations and doses presented herein, the annual average concentration at many locations around the point of (continual) release is computed, and the greatest of these is used to calculate the maximum annual-averaged dose. This procedure is currently employed in calculating maximum doses as a part of the environmental impact of fission power plants.²³

The ground-level concentration downwind of a release depends on the release rate, the release height, the distance downwind, the wind velocity, the degree of turbulence of the air, and the constancy of wind conditions. The analysis of Pasquill²⁰ has become adopted as standard for calculations of concentration, as illustrated in Fig. 4.1-5²². Depending on wind conditions, the maximum concentration and the location of the maximum with respect to the release point show considerable variation. The ordinate displays the product of the ratio of the local concentration (χ) to the release rate (Q) and the average wind speed ($\bar{u}$). The values have been averaged with respect to variations in speed and turbulence within each of six stability categories given in Table 4.1-5. Separate figures are available for different release heights.

The instantaneous dose rate at a given distance from the release point is computed from airborne concentration given by relationships such as those depicted in Fig. 4.1-5 once the air stability category, windspeed (m/s), release rate (Ci/s), and the

Fig. 4.1-5.

Generalized dose vs distance curves used in this study for release from a 30-m stack (wind conditions summarized on Table 4.1-2).

relationship between concentration (Ci/m³) and dose rate are given. Such an instantaneous dose rate would not provide an estimate of doses from normal operation of the plant because of changes in wind direction velocity and stability category with time. Instead, annual average values of χ/Q are calculated for each of the 16 cardinal directions and each of a series of distance ranges (11 were used for the Bailly plant) on the basis of meteorological statistics available for the site. For Site I (Bailly) these values are readily available for a release height of

TABLE 4.1-5
DESCRIPTION OF WIND CONDITIONS USED IN FIG. 4.1-5
PASQUILL'S STABILITY CATEGORIES

<u>Stability Category</u>	<u>Typical wind velocity</u>	<u>Description</u>
	m/s	
A	1	very sunny, summer weather
B	2	sunny and warm
C	5	partial cloud during day
D	5	overcast day or night
E	3	partial cloud during night
F	2	clear night

150 m. The maximum value for χ/Q (4.75×10^{-8} s/m³) occurred at 1-2 miles to the north-northeast of the release point. This value was therefore used in calculating the maximum annual average dose from airborne release at Site 1 for case (2) with a 150-m-high cooling tower.

Detailed average χ/Q values are not available for 30-m or 150-m releases at Site 2 (Oconee). For these cases maximum annual average χ/Q values were prorated using the figures in Ref. 23. Based on the data summarized in Figs. 2.-1 and 2.-2, Site 1 is most often characterized by wind condition C or D (Table 4.1-5) and Site 2 is most often characterized by wind condition B. For Site 1, the factor by which the maximum χ/Q for 150-m release must be multiplied to obtain the maximum χ/Q for a 30-m release is 34.0 s/m³; the ratio of the distance of maximum concentration for the 30-m release to that for the 150-m release is 0.133. A factor was also used to reduce the maximum 30-m release (at a distance of ~440 m) to that for the nearest populated location, assumed to be at the edge of a 600-m exclusion zone. This factor is 0.66. For Site 2, the maximum release factor, the distance ratio, and the 600-m correction factor were determined in the same way, and are based upon the Site 1, 150-m value of maximum χ/Q . The maximum annual average χ/Q values obtained by applying these factors to the known maximum annual average value for 150-m-high releases at Site 1 are shown in Table 4.1-6.

Two methods of calculating doses from the values of the maximum annual average concentrations calculable from the χ/Q values in Table 4.1-6 were employed. For a lower limit, the ICRP method was used; the 500 mrem/yr dose was prorated from the 0.2 μ Ci/m³ maximum permissible concentration to the actual appropriate concentration. For an upper limit the method of Morley and Kennedy¹⁷ was

employed. These authors used the atmospheric diffusion model of Pasquill²⁰ as interpreted by Bryant²¹ to estimate the local consequences of atmosphere tritium releases. Similar atmospheric diffusion and dose data are available in Ref. 22. The assumption was made that all tritium released to the atmosphere quickly equilibrates with atmospheric moisture and that at any point downstream of a continuous point-source of tritium the body fluids are in equilibrium with the (tritiated) atmospheric moisture.¹⁷ The result can be expressed as: $D(\text{mrem/yr}) = 1.82 \times 10^5 \chi/\rho$, where χ is in μ C/m³ and ρ is in g/m³. Data in Fig. 2-3 show that typical humidities are 3.5 g/m³ for Site 1 and 11.0 g/m³ for Site 2.

The variation of dose from airborne tritium is shown in Fig. 4.1-6 for one of the options under consideration. About 72% of the 6 Ci/day of tritium is released at a height of 150 m for case (2) at Site 1. No credit is taken for reduction in dose from the dilution associated with the vertical air current leaving the cooling tower, and hence the doses shown are conservatively high. The ICRP method of dose was used. Note that the dose to people close to the cooling tower calculated by this method would be quite small, and beyond about 2 km the dose decreases rapidly. The directions shown in Fig. 4.1-6 are those for which average χ/Q values (and hence doses) are maximum (NNE) and minimum (W).

TABLE 4.1-6

MAXIMUM ANNUAL AVERAGE
VALUES FOR χ/Q

Release Height (m)	χ/Q (s/m ³)	Distance (km)
Site 1	150 4.75×10^{-8}	2.4
	30 1.06×10^{-6}	0.60
Site 2	150 2.14×10^{-7}	0.89
	30 9.95×10^{-7}	0.60

Fig. 4.1-6

Dose vs distance for conditions most commonly prevailing at Site 1 and Site 2.

4.1.4.3 Maximum Doses. Using the methods described in the preceding sections, maximum individual doses were calculated for the three cases chosen. These are given in Table 4.1-7. For all cases and for both methods of calculating dose, the annual doses are far below the ICRP limit of 500 mrem and the average background of about 125 mrem. These doses are also less than the 5 mrem allowed in 10 CFR 50, (Appendix I)²⁴ as the "lowest practicable" standard for maximum total dose from fission plants. The dose for once-through cooling is quite low, even with the unrealistic assumption that persons would drink only undiluted plant effluent. Highest doses are calculated for the wet cooling tower case (case (2)). Here the greatest contribution is in the exaggerated drinking water dose.

It is concluded that risk of adverse effects on individuals due to tritium released during normal operation of the RTPR is so small as to be of no practical concern for 6 Ci/day releases.

4.1.4.4 Population Doses. The annual integrated population (AIP) dose is measured in man-rem and represents a product of the annual radiation dose and the number of individuals under risk. The man-rem unit is a more effective measure of the radiological impact at a given site than the individual dose; this unit is applied to Sites 1 and 2. The atmospheric and aquatic release rates used for the individual dose determinations are used also to determine the AIP dose.

Site 1 was selected as representative of sites near a large metropolitan area (with a 50-mile radius population of 7.3×10^6) and using a lake as the receiving water. About 5.8×10^6 individuals within a 90-km (50-mile) radius of Site 1 obtain their drinking water from

various lake-front intakes. Using the steady-state diffusion model of A. Okubo²⁵, the annual AIP dose for the various intakes of water was computed to be 46 man-rem for the case where the 6 Ci/day enters the hydrologic system directly (case (1)). Using an atmospheric diffusion model, the annual average site meteorology, and the ICRP dose model, the AIP doses resulting from exposure to airborne tritium were computed for cases (2) and (3). For the wet cooling tower case, the AIP hydrologic dose was 12.7 man-rem/yr and the airborne dose 0.61 man-rem/yr for a total of 13.3 man-rem/yr. For case (3) dry cooling towers, the airborne dose was 0.84 man-rem/yr. For comparison the natural background of 125 mrem/yr results in an AIP dose of about 990,000 man-rem/yr to the residents within a 50-mile radius of Site 1. Prorating from the Bailly design, a BWR plant with the same net capacity as the RTPR (4030 MWe) would lead to a total AIP dose of 248 man-rem/yr. These results are summarized in Table 4.1-8.

Site 2 is intended to represent a rural area (50-mile population of 730,000) utilizing a river as the receiving water. Within 90-km (50-mile) downstream of the site, about 25,000 individuals obtain their drinking water from the river. The AIP dose to these individuals from a 6-Ci/day release in case (1) is computed to be 1.7 man-rem/yr, based on an expected average river dilution of 10. For wet cooling towers, case (2), the hydrologic dose is 0.4 man-rem/yr and the airborne dose is 0.50 man-rem/yr; the total AIP dose is 0.9 man-rem/yr. For dry cooling towers, case (3) the airborne AIP is 0.69 man-rem/yr. For comparison, the natural background of 125 mrem/yr results in AIP dose of about 91000 man-rem to residents within the 90-km (50-mile) radius of Site 2. Scaling from the Oconee Plant, a 4030-MWe plant

TABLE 4.1-7

**MAXIMUM ANNUAL AVERAGE INDIVIDUAL DOSES FROM THE RELEASE
OF 6 Ci/day OF TRITIUM (mrem/yr)**

	Site 1 (Bailly)		Site 2(Oconee)	
	ICRP ^a	M&K ^b	ICRP ^a	M&K ^b
Case (1) (Once-Through Cooling)	0.064(water)		0.064 (water)	
Case (2) (Wet Cooling Tower)	0.80(water)		0.80(water)	
	0.0059 (air)	0.12	0.027 (air)	0.18
	0.81 (total)	0.92	0.83 (total)	0.98
Case (3) (Dry Cooling Tower)	0.18 (air)	0.77	0.17 (air)	0.14

^aMethod of International Commission on Radiological Protection⁽¹⁵⁾

^bMethod of Morley and Kennedy⁽¹⁷⁾.

TABLE 4.1-8

**SUMMARY OF ANNUAL INTEGRATED POPULATION (AIP)
DOSE (MAN REM/YR) AT 6 Ci/day RELEASE RATE**

Cooling Case	Site 1 Bailly (Lake, 5.8 x 10⁶ People/ 50-Mile Radius)	Site 2 Oconee (River, 7.3 x 10⁵ People/ 50-Mile Radius)
Once-through cooling	46 (Water)	1.7 (Water)
	12.7 (Water)	0.4 (Water)
Wet cooling tower	0.6 (Air)	0.5 (Air)
	13.3 (Total)	0.9 (Total)
Dry cooling tower	0.8 (Air)	0.7 (Air)
Bkg/50-mile radius (125 mrem/yr)	9.9 x 10 ⁵	9.1 x 10 ⁴
Fission plant (Prorated)	248	15.4

would lead to a AIP dose of 15.4 man-rem/yr from a fission plant.

The combined annual integrated population doses from exposure to the gaseous and liquid effluents of the plant at either site are very small compared to the AIP dose received each year from the natural radiation background. Hence, no discernible radiological impact on individuals and the population is expected from normal operation of the RTPR. Also, RTPR tritium population doses would be substantially less than total doses from fission plants with the same net generating capacity. Section 5.1 presents a cursory analysis of the dose out expected for abnormal (accident) conditions.

4.2 STRUCTURAL AND IMPURITY ACTIVATION AND RADIOACTIVE WASTE

Some reactor components will not last the life of the RTPR plant. The reactor first wall represents the most crucial and failure-prone component of the RTPR; the intense neutron flux will couple with the cyclic stresses induced by thermal and flow transients to shorten a module lifetime to a few years. Failure will probably be manifested by either loss of first-wall dielectric strength or by lithium seepage into the plasma chamber. Neutron-induced swelling, particularly within the graphite regions, may also be a lifetime determinant. The RTPR design facilitates easy and rapid replacement of defective modules, although the high radiation fields will necessitate

remote handling. In addition to neutron activated structure material, each module will contain tritium dissolved both in the structure and the lithium coolant as well as impurities which have been activated by neutron irradiation. These impurities may remain within the blanket structure, be transported to concentration points within the lithium-cooling circuit, or remain dissolved in the lithium coolant. Major contributions to the radwaste problem can, therefore, be identified: (1) radioactive structural wastes (primarily ^{93m}Nb and ⁹⁴Nb), (2) impurity activation in the lithium, and (3) activation of graphite and beryllium blanket materials.

4.2.1 Radioactive Structural Waste

In the present design, Nb-1%Zr alloy was selected as a major structural material of the blanket. After long-term exposure (<5 yr) to radiation, the blanket will need to be replaced for structural reasons. This "spent" structure will be very radioactive and must be handled as a radioactive material. Based on an estimated minimum average lifetime of 5 yr, the amount of material that must be stored or disposed is estimated to be no greater than 19 kg/MWe-yr. Depending upon initial purity, it may be possible to recycle blanket materials after some years of "cooling." In view of the rapidly growing use of automation in the metals industry, remote handling and recycling of this material may eventually prove practical and economical. Remote refabrication and

repair would eliminate the need for extremely long-term storage of radioactive wastes and would effectively remove some of the long-lived activity by further transmutation during reuse in the fusion blanket.

A detailed analysis of the effects of first-wall loading I_w , operating time T , and structural alloy has been presented elsewhere,²⁶ and Appendix E summarizes the calculational model and analytical results for radioactivity and nuclear afterheat. Figure 4.2-1 illustrates the decay of radioactivity induced in the Nb-1%Zr blanket structure after five years of operation at a wall loading of 6.7 MW/m² (duty factor of 100%), or an average flux of 2.71×10^{19} n m⁻²s at the first wall. Figure 4.2-2 illustrates the

Fig. 4.2-1.

Radioactivity of niobium structure as a function of time after shutdown.

Fig. 4.2-2.

(Ci/Wt) and BHP vs time for Nb-1%Zr and V-20%Ti at $I_w = 6.7$ and 2.0 MW/m².

time dependence of the radioactivity (Ci/Wt) and the biological hazard potential (km³/Wt) for both Nb-1%Zr and V-20%Ti blanket structures which have operated for five years at values of I_w equal to 6.7 and 2.0 MW/m², respectively. The dependence of radioactivity and nuclear afterheat (P/P_o) on wall loading is summarized on Fig. 4.2-3.

Increased wall loading for a given operating period generally increases the short-term activity and BHP for Nb-1%Zr, although increased wall loadings reduce the long-term activity and hence storage/waste problems. In essence, the large contribution to the total radioactivity of Nb-1%Zr fusion reactors by second-order neutron absorption renders higher radioactivities (Ci/Wt or Ci/Wt yr) for higher wall loadings at times < 100 yr after shutdown because of the burnout of ^{93,94}Nb. Therefore, a long-term advantage exists for higher wall loadings as far as possible storage requirements are concerned. A summary of the interaction between I_w , T , and t_s for Nb-1%Zr is given in the tables and figures of Appendix E. The effects on radioactivity of substituting V-20%Ti for Nb-1%Zr in the blanket are also summarized in Appendix E.

Because of the close proximity of the copper magnetic coils to the RTPR blanket, considerable neutron activation of the copper is expected. The major portion of this activity is short-lived (≤ 12.74 h) and therefore presents more of an afterheat problem than a radiological (storage) problem; the copper activity is $\sim 20\%$ of the Nb-1%Zr activity (Ci/Wt) and $\sim 5\%$ of the Nb-1%Zr biological hazard potential at shutdown ($I_w = 2.0$ MW/m², $T = 5$ yr). The isotope ⁶³Ni ($t_{1/2} = 100$ yr) represents a major long-term contributor to the radioactivity from a radwaste point

Fig. 4.2-3.

Dependence of (Ci/Wt) and P/P_o at shutdown on I_w .

of view, although this situation may change if impurities found in commercially pure copper (or copper alloy) are incorporated into the analysis. The 5 272-yr ^{60}Co may be the most important contributor to maintenance and recycling problems for the coils. Employing aluminum coils should reduce the induced activity by orders of magnitude.

The cross-section data used in these analyses are inadequate for radiological and afterheat assessments with greater than ~25% accuracy (~ factor of 2 for $^{95,95\text{m}}\text{Nb}$); for this reason the results and conclusions presented herein should be viewed as preliminary. Much more work and refinement must be done before these kinds of calculations for fusion reactors will be on an equal footing with similar estimates for fission reactors. This singular fact should be kept in mind when making radiological fission-fusion comparisons. Estimates of $^{95,95\text{m}}\text{Nb}$ activity most likely err on the high side, overpredicting BHP and P/P_o, whereas ^{94}Nb predictions probably err on the low side.

In addition to neutron activation, the tritium dissolved in the structural alloy contributes to the radioactivity. The amounts of dissolved tritium in the structural niobium metal (the Sr's in Fig. 4.1-2) are small considering the volumes of these materials. For example, if it becomes necessary to remove and refabricate the entire wall in one of the eight lithium-sodium heat exchangers, and the tritium dissolved in this wall cannot be removed prior to disassembly, the wall will contain only about 635 mg (1 mg in the stainless steel and 634 mg in the niobium) of tritium. This amount does not present a severe handling or storage problem. The dissolved tritium, however, may have to be removed before disposal or reworking of the assembly.

4.2.2 Impurity Activation in the Lithium Coolant

The specific radioactivity of lithium results from one or more of the following mechanisms.

(1) The isotopes initially present as impurities in lithium are activated by the neutron flux during passage through the reactor.

(2) A fraction of the surface isotopes of the structural material in the reactor section and in contact with the lithium are sputtered or recoiled into the coolant from a capture or scattering reaction.

(3) Radioactive and nonradioactive isotopes will be introduced into the coolant as a result of corrosion processes.

The processing system proposed to remove tritium from the lithium coolant will also be an effective

purification system for the lithium. It is expected that, in addition to the hydrogen isotopes, it will remove the isotopes of carbon, nitrogen, and oxygen on a continuous basis. Furthermore radioisotopes of boron, sulfur, silicon, or phosphorus, which are initially present in the lithium or blanket materials exposed to the lithium, will also be eventually removed by the purification system. No more than 1 ppm of each of these isotopes would be expected in any case. Larger amounts of calcium, potassium, and sodium are expected in the initial lithium charge, but these impurities should be removed during the start-up operation of the plant and before a significant radiation problem is incurred. Table 4.2-1 summarizes typical impurities found in high-purity lithium.

Constituents and impurities of the blanket materials in contact with the lithium, e.g., iron, nickel, chromium, niobium, tantalum, hafnium, and tungsten, will also be removed during the processing step and will eventually require disposal. Experience over many years of operation with a similar liquid-metal system on the EBR II²⁷ indicates that the accumulation of these materials is not significant (amounting to about a few to tens of liters per year). Typical impurities found in commercial Nb-1%Zr are summarized in Table 4.2-2. Determination of the total activation introduced into the lithium coolant by the neutron activation of these impurities is not within the scope of this study. An estimate of the activity derived from these impurities can be obtained from the niobium corrosion rate of 10.0 $\mu\text{m/yr}$; this corrosion

TABLE 4.2-1
TYPICAL IMPURITIES FOUND IN
IN HIGH-PURITY LITHIUM

Element	Content (ppm)
Si	5-50
O	100
N	30-500
Ca	100
Na	100
K	100
Fe	100
Ni	100
Cr	100
Ta	100
F	3000
Cl	3000
Nb	---

TABLE 4.2-2
TYPICAL IMPURITIES PRESENT IN
COMMERCIAL Nb-1% Zr ALLOY

<u>Element</u>	<u>Content (ppm)</u>
W	300
Mo	100
Ta	1000
Hf	100
Zr	10,000
Nb	985,000
O	300
N	200
H	20
C	100
Fe	50
Ni	20

rate includes the contribution from neutron sputtering and mechanical erosion. For instance, if R (m/s) represents the sum of the niobium corrosion, erosion, and sputtering rates at the lithium coolant, then the ratio of the rate of dissolved niobium activity (C_i/W_i yr) to the structural activation (C_i/W_i) is given by $(A/V)R$; A is the total niobium area in contact with the lithium coolant within the blanket ($19,540\text{m}^2$) and V is the total volume of niobium in the blanket (44m^3). The quantity $(A/V)R[C_i/W_i]$ represents a worst case in that all dissolved niobium is assumed to be removed from the lithium coolant upon each pass through the reactor, and depletion of the blanket niobium by corrosion is not taken into account. Hence for R equal to $10.0\text{ }\mu\text{m/yr}$, the quantity $(A/V)R$ equals 0.0044 (I/yr) or each year the lithium clean-up system must remove, store, and dispose of an activity equal to 0.44% of the total niobium activity in the blanket. After five years of operation (C_i/W_i) = 5.13 (see Fig. 4.2-1 or Table E-2), or the rate of activity handled by the lithium clean-up system at this time is 6.1×10^6 Ci/yr.

4.3 THERMAL EFFECTS

During normal operation approximately 2.0 MW is dissipated to the cooling water or cooling tower air for every 1 MW of electricity generated (Fig. 3.3-1). This high value of reject heat is a result of the choice of a conservative, low-temperature power plant (thermal efficiency $\sim 40\%$). More efficient plant cycles utilize higher coolant temperatures. The

higher temperatures will cause approximately the same technological difficulties and capital cost increases for fusion power plants as for plants using other energy sources. The thermal reject problem is inherent to any large-scale generation of electrical energy. Fusion power has the potential for appreciable direct conversion, although Miley²⁸ indicates a limit of $60\text{-}65\%$ direct conversion even for the most efficient (direct-conversion) fusion reactions. The development of systems and methods for the utilization of low-grade heat represents a promising solution to the thermal effects problem and should be seriously considered in the long-range planning of a fusion economy.

4.4 CHEMICAL, SANITARY, AND OTHER WASTES

Generation of wastes of this nature and the release to the environment will be the same as for other types of thermal power plants. No detail on these discharges will be provided here.

4.5 TRANSPORTATION OF FUEL AND RADWASTE

After start-up, no radioactive fuel will be transported to the plant; deuterium and lithium will be required at rates of 970 and 3360 g/day , respectively. Product tritium will be shipped from the plant at a rate determined by plant capability or the extent to which the DT cycle is used in a growing fusion economy. If no market exists for tritium, the breeding ratio is easily reduced to unity. Currently tritium is routinely shipped in DOT-approved, double-wall shipping containers in 10-g lots. Solid waste will be shipped intermittently from the plant as discussed in Sec. 4.2. The greatest mass of material is likely to be the niobium blanket structural material. Distributed evenly over a 5-yr life, the shipments of radioactive niobium would average about $76,700\text{ kg/yr}$.^{*} Shipments of debris collected in the cold traps and from the molten-salt, lithium purifying system will also be required.

Management of long-lived radioactivity associated with the copper compression coils represents another

^{*}This magnitude of niobium wastage will probably be intolerable when prorated to a sizable fusion economy using this metal. Remote recovery, refinement, and refabrication of activated niobium will most likely be required.

consideration. If all the copper is replaced every 10 yr, shipment of the order of 560,000 kg/yr are to be expected. The copper, however, will probably be reclaimed rather than treated as waste. Shipments of radioactive graphite and beryllium could average 136,000 and 10,000 kg/yr, respectively, for a 5-yr blanket life-time. However, the copper, graphite, and beryllium activity will be low level.

4.6 IRREVERSIBLE AND IRRETRIEVABLE COMMITMENT OF RESOURCES

The resources committed for the construction of a theta-pinch power plant generally are the same as for any large industrial facility. The materials required by a fusion power industry supplying 10^7 MWe* have been estimated.²⁹ Table 4.6-1 summarizes the resource requirements anticipated for the RTPR.

Although the approximate nature of the estimates presented in Table 4.6-1 precludes a detailed analysis of the resource impact of the RTPR, some general comments on the supply of some of the principal materials can be made:**

Deuterium - Available in the oceans in essentially unlimited supply. Extraction is anticipated to cause negligible environmental effect.

Lithium - Available in plentiful supply. Land-based resources are sufficient for long-term operation of a fusion economy. Much more lithium is available in oceans at undetermined cost and environmental impact for extraction.

Copper - Large quantities are used in the RTPR compared to other fusion reactor concepts. The demand for copper resulting from a completely theta-pinch fusion reactor economy would have a strong impact on the copper industry. The U.S. resource of copper at the present price is estimated to be about twice the amount needed for a RTPR power economy. The worldwide resources are about four times the U.S. resources. For

the amount of copper available at increased cost, the demand for fusion reactors represents about a quarter of the U.S. supply and a thirtieth of worldwide resources. Additionally a considerable effort is underway to develop processes for extracting copper from low-grade ores. Nuclear chemical mining, for instance, could create a significant increase in availability with no cost increase.³⁰

Niobium - According to the present RTPR power plant design, approximately 1/20 of the present U.S. niobium resource is required for each 4 GWe plant (Table 4.6-1). Replacement of niobium blanket structure because of radiation damage also represents a potentially large consumption of niobium. These levels of niobium usage appear to make essential the recycle of blanket niobium. Heat exchangers and primary coolant ducts could probably be made from other materials (stainless steels, vanadium, molybdenum), thereby considerably reducing the niobium requirements. Even so, a large portion of the total niobium (30%) is required by the superconductor in the magnetic energy and transfer system. The demand on niobium resources will be significant, particularly if niobium alloys come into general usage for energy storage and transfer lines.³²

Beryllium - The magnitude of beryllium reserves is poorly known, but Table 4.6-1 indicates that only about 500 4-GWe plants can be built with the known U.S. reserves, although better data on the ultimate beryllium reserves are needed.

Helium - The magnetic energy transfer and storage system requires a vast inventory of helium refrigerant; however, available resources appear adequate. The helium requirement for the purge system is insignificant compared to the METS system requirement. Large quantities of helium are available in the atmosphere, although the cost to separate this helium, though unknown, will certainly be higher. The helium production in the D-T reaction is sufficient to balance all losses in the RTPR power plant.

*The world's "steady-state" electrical generating capacity is estimated to be 10^7 MWe. Presently, the doubling time for electrical consumption is 7 years, although the hope exists that a plateau will be reached, perhaps by the year 2100-2200.

**These comments are applicable to a universal, D-T fusion economy of $\sim 10^7$ MWe.⁽²⁹⁾

TABLE 4.6-1
RESOURCE REQUIREMENTS ANTICIPATED FOR THE RTPR

Material	12 GWt (4 GW _e) RTPR			10 ⁷ MWe from 2440 4 GW _e RTPR's		Known Resources At Present Prices		Estimated Resources At Increased Prices	
	Const. (10 ⁶ kg)	Oper. (10 ⁶ kg/yr)	Requirement for 30 yr. Plant Life (10 ⁶ kg)	Const. (10 ⁶ kg)	Oper. (10 ⁶ kg/yr)	U.S. (10 ⁶ kg)	World (10 ⁶ kg)	U.S. (10 ⁶ kg)	World (10 ⁶ kg)
Niobium(Total)	3 608	0767	5 91	8,800	189	70 ^a	6,000 ^a	140 ^a	...
Blanket	383	0767 ^b 0 011 ^d 6 76 x 10 ^{-6c}	0 33 ^d		26 ^{c,d}	5 4 ^e	100 ^e	7,800 ^e	17,200 ^e
Headers	356								
Manifolds	077								
Misc Piping	352								
H Exch	812								
Li Processing	540								
Li Pumps	058								
METS	1 029								
Copper(Total)	9 304	1 7 x 10 ^{-3c}	9 355	22,700	4 15 ^c	77,000 ^a	280,000 ^a	180,000 ^a	1,100,000 ^a
Compression	5 582	1 7 x 10 ^{-3c}				2 x 10 ^{4d}	10 ^{5e}	10 ^{5e}	3 x 10 ^{5e}
Coil		1 3							
Leads	3 722		3 722						
Lithium									
Natural Total	1 58 ^b	016 ^b	2 06 ^b	3,840 ^b	39 ^b	5,000 ^a	6-8,000 ^a	9,000 ^a	100,000 ^a
Coolant Loop (natural)	864					800 ^a	1,000 ^a	2,700 ^a	...
Blanket ⁶ Li	051	00238 ^c							
Blanket ⁶ Li	049	00113 ^c							
Blanket Natural Li	716 ^b		0 48 ^{e,g}						
Beryllium	052	0104 ^b 6 x 10 ^{-4c}	3648 ^b 018 ^c	126	25 ^{b,h} 1 46 ^c	26 ^a 18 ^a	380 ^a ...	72 ^a 250 ^a	1,000 ^a 280 ^a
Helium	1 253	0 0015 ⁱ 0 002 ^j	1 298	3,056	3 658	1,200 ^h	1,300 ^{h,i}	5,000 ^{h,i}	29,000 ^{h,k}
Carbon	68	136 0 000,0 ^o	1 76 ^b 0 0074 ^c	1,653	332 ^b 0 697 ^c	500 ^a	100,000 ^a	...	...
Tin	0 3	...	0 3	730	...	6 ^a 0 ^a	4,000 ^a 6,000 ^a	42 ^a 100 ^a	7,000 ^a 11,000 ^a
Deuterium	0 003 ⁿ	0 0004	0 0162	...	...	...	...	...	...

^aFrom Reference 29

^bAssumes 6 years useful lifetime

^cTransmutation

^dCrossion-sputtering products collected in cold trap

^eFrom Reference 31

^fCan be reprocessed

^g2.5 x 10⁴ g in vessel of 1.5 ft dia per 10³ gal/hr (Ref. 30)

^hOne kg natural Li gives 0 074 kg ⁶Li

ⁱLittle information available

^jIn the ground

^kIn the atmosphere

^lFrom impurity

^mGas from transmutation: ³He, ⁷Li, ⁶Li, T, and D. Only about 50% may be captured

ⁿSupplied to RTPR in METS

^oExcess stored in 10³ ft³ supply at 5% burn up

4.7 LAND DESPOILMENT

The land despoilment issue has two aspects. The first is the direct land use by the power plant itself imposed by the requirements associated with buildings, transmission equipment, cooling ponds or towers, and exclusion zones. In this respect, the RTPR power plant is the same as other nuclear power plants. The second aspect of land despoilment concerns the procurement and eventual disposal (where

applicable) of construction materials and fuel. This facet of the environmental impact of the RTPR is concerned primarily with mining operations and the disposal of radioactive structural material. An estimate of material quantities needed implies that the mining operations would be very extensive for a mature fusion power economy.

Evaluation of the environmental impact is a matter that may be decided only by an eventual cost-benefit analysis that will balance the conflicting needs of the

populace for power required to maintain a high standard of living and a desire to maintain the unspoiled countryside. Such a cost-benefit analysis is not within the scope of this work. At a time when various fusion reactor concepts are developed to the point of yielding more detailed operational characteristics, the environmental impact may weigh heavily in the selection of the most attractive design. At present it is not obvious that significantly different materials requirements for the construction of different designs exist; the lithium and superconductor requirements will be comparable, for instance. Furthermore, it appears that the use of niobium and copper are comparable within the uncertainty of current design descriptions. The use of superconducting vs normal conducting magnets implies differing demands for niobium and copper. In fact, the construction of superconducting magnets employs large quantities

of copper for stabilization, and the theta-pinch concept employs superconducting materials for the energy storage device. The lifetime of the components may introduce different long-range requirements; RTPR design anticipates frequent changing of blanket assemblies. However, the maintenance requirements of other designs have not been shown to be better in this regard, and the issue may be avoided if recycling of the materials proves feasible.

In terms of land usage, disposal of the neutron-activated structural material would not be extreme. The problem, in fact, could be reduced to negligible proportions if refabrication and reuse of these materials proves feasible when their structural properties become degraded. Considering the low volatility of fusion-originated radwaste, the storage requirements of a fusion economy should be considerably less than an equivalent fission economy.

5. ACCIDENT ANALYSES

Environmental impact statements generally do not treat large-scale accidental releases of radioactivity; accident analyses are usually treated separately in a safety analysis report. However, because rigorous accident analyses for fusion reactors are not available, and even a brief treatment may help to bring the magnitude of the problem into perspective relative to the seriousness of accidents in other systems, the accident situation is briefly reviewed.

Six accident categories are considered from the viewpoint of their effect on the plant and the resulting possible damage to the environment. No large-scale uncontrolled nuclear reaction is possible, because of the limited amount of fuel present at any given time in the plasma (1.00 g). Ignition and burn of a DT plasma will require high performance from all of the systems involved; the overwhelmingly probable result of a malfunction will be nonignition. Moreover, relatively little radioactive fuel is available in the plasma chamber for dispersal in the event of an accidental release. The credible accidents appear to be those that might occur in the liquid-metal system. A liquid-metal fire and the release of all tritium contained by the coolant systems is postulated as the maximum credible accident, and the resulting maximum possible release of radioactivity is found not intolerable.

5.1 MAXIMUM CREDIBLE ACCIDENT

The tritium inventory represents the potential for public hazard for the RTPR. The estimates of tritium inventory summarized in Table 4.1-6 clearly indicate that the total tritium inventory is widely dispersed throughout the RTPR. Table 5.1-1 gives a concise summary of the tritium inventory.

A severe liquid-metal fire and the attendant instantaneous release of tritium contained in the coolant systems appears to represent a maximum credible accident (MCA).^{*} The assumption is made

^{*}The development of fusion reactor technology is not advanced to the point where a unique MCA can be identified. However, this brief review of the potential for a publicly hazardous accident for the RTPR indicates strongly that liquid-metal fires rank high as a driving force for a MCA.

TABLE 5.1-1
SUMMARY OF TRITIUM INVENTORY^a

Location	Inventory (g)
Reactor Blanket	53
Lithium Circuit (Not including blanket)	404
Fuel-Ash Processing, Tritium Extraction, and Fuel Injection	1913
Sodium ^b	0.3
Steam	7
Storage	1426
Other	10
Total	3814

^aSee Table 4.1-4, p. 22 for detailed breakdown.

^bBased on 0.1 ppb concentration and 3.026×10^6 kg Na.

that the fuel storage area containing 1.4×10^7 Ci of tritium can be sufficiently isolated and protected so that no substantial part of the tritium inventory can be discharged to the environment by any conceivable accident. Therefore, the MCA treated here involves the sudden release to the atmosphere of 4.6×10^6 Ci of tritium contained primarily in the coolant circuits (Table 5.1-1).

To estimate the effect on the environment of such a release, it is assumed that the hot alkali metals will react (burn) with air and available moisture at a constant rate for a period of a few days. Thus, the reaction products (a mixture of nitrides, oxides, hydrides, hydroxides, and gases) contain 8.5×10^5 kg of lithium and 3.0×10^6 kg of sodium.^{*} Although some of the discharged tritium would probably remain in the solid alkali metal reaction products dispersed by the fire, calculation of the effects of the accident is made on the basis that the entire tritium inventory (excluding storage) would be dispersed as gaseous HT or HTO during the fire.

^{*} Eight separate coolant systems generate steam for four turbogenerator units. The degree of realism in assuming the all eight coolant systems simultaneously experience a serious fire is yet to be tested analytically.

The dispersion model is based upon a model presented in the AEC Safety Guide 3.³³ Using this model, the maximum ground level concentrations can be estimated as a function of distance from the point of release in accordance with the values of χ/Q shown in Fig. 5.1-1. For this figure a number of Pasquill diffusion categories (see Table 4.1-5) and variable wind direction within a 22.5° sector were employed. In view of the strong upward convective action of the heat from the burning alkali metals, an effective release height of 50 m was assumed. At the 600-m distance, $\chi/Q = 1.32 \times 10^{-5}$ s/m³, and the 3-day concentration is 3.7×10^{-4} Ci/m³. The direct beta "shine" (in rads) from an "infinite cloud" was calculated³³ as $0.23 E \chi t$, where E is the average beta energy of the tritium emissions (0.0057 MeV) and t is the release time(s). For the "infinite cloud," this energy is all absorbed so the calculated dose is 0.13 rem. It was also assumed that the ICRP skin and whole body exposure will result, prorated for three days from the 0.5 rem/yr expected at a concentration of 2×10^{-7} Ci/m³. The calculated skin and whole body dose is 7.6 rem, and is below the 25-rem whole-body dose limit in 10 CRF 100³⁴ for low-probability accidents. Figure 5.1-2 gives the dose vs distance relationship for release from a 50-m stack.

Fig. 5.1-1.
Diffusion factor vs distance for MCA conditions.

Fig. 5.1-2.
External beta-ray dose vs distance for MCA conditions for 50-m release height (Site 1).

These calculations are considered unrealistically pessimistic or conservative for at least three reasons. First, an unlikely combination of events must occur nearly simultaneously to initiate the fire. Secondly, more unlikely circumstances must occur to sustain the fire for several days. Finally, the doses could be obtained only by persons who remained at a 600-m distance from the accident scene for a period of three days while a raging fire consumes nearly a gigagram of lithium.

It should be noted that the lithium, which contains 457 g of tritium as well as some dissolved radioactive corrosion products, is buffered from the steam plant by a relatively benign sodium system. The sodium contains less than 1 g of tritium. The lithium is physically isolated from the steam system by two to four concrete walls (depending upon the particular location) and from external air by three to five walls. Therefore, breaching of some combination of these barriers must be postulated to initiate a lithium fire. Such a sequence of events is not impossible; a severe earthquake, a very large meteor, or simultaneous failure of several safety systems could be postulated to initiate the lithium fire.

Sustaining the lithium fire requires that more safety devices fail. For an example, should any of the extreme occurrences listed above happen, all lithium normally would quickly empty into dump tanks, which are secured in the most remote basement compartment; although some lithium may burn, the ma-

por portion of the lithium inventory would be safely stored in the dump tanks, away from any conflagration. Furthermore, it must be postulated that fire extinguishment systems fail or are ineffective in reducing the fire severity and extent, and that no other corrective action is taken to mitigate the extent of the fire.

Finally, a person exposed to the fallout from the fire would probably find it quite uncomfortable to breathe alkali metal hydroxide fumes for even a few seconds, to say nothing of several days. In fact, the lung damage from breathing these fumes would be much more severe than the radiation from tritium or corrosion products, if continued exposure was maintained. A much more likely course of action would result in very rapid evacuation of regions downwind of a fire such as postulated above.

The available design detail for the RTPR power plant and the changing nature of this design does not warrant a detailed, quantitative analysis of a "maximum credible accident" at this time for the RTPR. Granted that a profound accident analysis will be required some time prior to the construction of this power plant, the foregoing qualitative analysis indicates that a severe liquid-lithium fire is not impossible, but is unlikely. Furthermore, the consequences of the postulated accident to the general populace, as far as tritium is concerned, is not expected to be catastrophic.

5.2 MAGNET MALFUNCTION

An increase in magnet field beyond the design value would, in principle, tend to burst the magnet, and increase the reaction rate. Since the energy for the pulsed field is delivered from a limited store, a significant increase in magnetic field above design values is unlikely, if not impossible. In any case the excessive forces on the magnet structure would be the most serious problem, resulting in damage or destruction of the magnetic coils. This situation could possibly lead to a violation of the blanket lithium containment, but would probably be restricted to a few core modules. Tritium released from the plasma chamber and portions of failed blanket would still be contained in the vacuum system.

5.3 CRYOGENIC ENERGY STORAGE FAILURE

If by loss of refrigerant or some other mechanism the METS systems were to lose superconductivity, a large quantity of stored electrical energy would be

released. However, through judicious magnet and circuit design the rate of energy release can be limited and a serious explosion prevented. The METS system is also modularized and reactor modules are removed by remote operation; a catastrophic failure of one or more of the 200-MJ units would not result in release of toxic materials.

The arrangement of each METS unit must, however, be planned to assure that explosive failure of a given unit does not lead to a failure propagation to adjacent METS units. Failure propagation can be prevented either by providing blast shielding for each METS unit or by assuring that appropriate dump resistors are automatically switched across a unit which has lost refrigeration or has otherwise lost superconductivity.

5.4 FUEL INJECTION ACCIDENT

If a malfunction causes the rate of fuel injection to increase, harmful effects seem improbable; in all likelihood an increased rate of fuel injection will prohibit ignition. For fuel injection rates below design values, the thermonuclear reaction rate would be smaller (if, indeed, ignition occurs). Moreover, significantly lower injection will lead to failure of the ignition sequence.

5.5 LIQUID-METAL ACCIDENTS

A sodium-to-water steam heat exchanger poses a special hazard. In the worst case a rupture of the heat exchanger would allow a massive injection of water and steam into the liquid metal. A rapid reaction would result producing hydrogen gas and solid corrosion products (sodium oxide, hydroxide, and hydride). The resulting pressure surges and potential hydrogen explosion represents a major hazard. Loss of system integrity could lead to large-scale fires and release of the tritium contained in the sodium. The plant has been designed with eight sodium-water heat exchangers to prevent simultaneous loss of all the sodium with its tritium (~0.3 g tritium). It is unlikely, however, that all of the hot alkali metal(s) would react quickly because of the protective effects of the reaction products. The products of reaction, however, could block the coolant flow channels in or near the heat exchanger and lead to a decrease or a loss of coolant capability.

Small leaks in the heat exchanger can be expected to grow considerably larger, either by the rapid formation of new small holes in contiguous surfaces from jet impingement of reacting species or by the

slower formation of larger holes by corrosion downstream of a seepage-type leak. In the former case, the new hole(s) would be small; in the latter case, a warning period of the order of hours would probably precede a massive reaction or a large-scale breakdown. A significant amount of pertinent information on the behavior of alkali metal systems under accident conditions is available from the fission reactor program and will be invaluable in assuring a safe and reliable liquid-metal system for a fusion reactor.

5.6 LOSS-OF-COOLING ACCIDENT

The loss-of-cooling accident (LOCA) in the theta-pinch fusion reactor does not appear capable of affecting the environment. Within the reactor the LOCA may be serious and, in the worst case, could result in melting portions of the blanket. It does not seem possible for local melting to lead to effects external to the power plant. To examine the results of an accidental loss-of-cooling, several scenarios can be postulated. The most severe case appears to be a sudden loss of cooling to a module combined with a simultaneous failure of all detection instrumentation such that the reactor continues to operate. Two alternative accidents are possible for this situation: a loss of coolant *flow* and the loss of coolant *per se*.

In the former case (loss of coolant flow) the coolant failure is external to the module, and coolant does not escape to either the vacuum trough or the plasma chamber. In this case the module can be assumed filled with lithium. Furthermore, since neither plasma contamination nor shorting of the electrical systems occurs, normal operation of the plasma portions of the reactor is assumed to continue. If flow meters, heat-exchanger pressure, or temperature sensors do not signal for shut-down of the reactor, the first wall will fail after about 20 pulses subsequent to the coolant stoppage. It is expected that the first-wall insulator will subsequently fail mechanically because of the large temperature excursion (600 K), and the resultant electrical arcing would melt the first wall, thereby allowing lithium into the plasma chamber. This situation would lead to a serious economic penalty and would require replacement of the overheated modules. The vacuum trough would require cleaning of spilled lithium and insulator debris. Replacement of the shorted high-voltage leads might also be required.

The second class of LOCA's to be considered is that in which coolant leaks either into the vacuum duct or the plasma chamber with a failure of detection circuits to shut-down the reactor. In this case the

lithium in the vacuum duct or plasma chamber would increase the impurity level and the plasma probably would not ignite or burn. Conceivably, the reactor could continue to pulse for several cycles and, during these pulses, the first wall of the failed module would be receiving radiation from the lithium-laden plasma. This energy would be approximately equal to the plasma internal energy. However, little lithium will remain in the failed module and heat transfer must occur by radiation rather than by conduction through the lithium. A single pulse will raise the first-wall temperature to approximately 1900 K, and mechanical stress or melting or both will probably cause failure, resulting in shorting of the subsequent pulse. Hence, to avoid excessive repair of the electrical circuits, it is necessary to prevent more than one pulse after the lithium has leaked away from the first wall and into the vacuum chamber.

Gross MHD plasma instabilities could result in a large portion of the plasma energy being deposited in a single module; melting and large lithium spills would certainly occur. These spills as well as the radioactive structure and tritium are completely contained within the plant. If for some unlikely reason this spill escapes the vacuum trough, it must still penetrate additional walls (most being lined with tritium permeation barriers) to escape the plant. Chemical reactions with materials of construction (e.g., graphite, niobium, alumina) or shield material appear to be slow and should not cause problems before significant cooling occurs. In any case, if the detection circuits fail to shut the reactor down, severe module and local plant damage can be expected, but no discernible environmental effects outside the plant should occur.

The most probable set of circumstances involving a LOCA will allow the reactor to be shut down as soon as the leak has occurred and before the leak has grown to significant proportions. In this case the blanket structure continues to generate heat from exoergic nuclear decay. As shown in Fig. 5.6-1 and as discussed in Appendix E, the afterheat is about 1.6% of the operating power (P_o) immediately after shutdown and for some days thereafter.

To a first approximation afterheat is conservatively assumed to be deposited in the niobium. The RTPR generates 12,000 MWt in a 718-m^3 blanket corresponding to an operating power density of 16.7 MW/m^3 (of blanket). The niobium volume fraction is 6.1% (44 m^3), leading to a volume average of 4.4 MW/m^3 (of niobium) and a maximum power density of 10.0 MW/m^3 at the first wall. Two cases must be considered: (1) the lithium drains away from the first wall and out of the module (loss of coolant) and (2)

Fig. 5.6-1.

Nuclear afterheat vs time for Nb-1%Zr and V-20%Fe blankets.

failure is external to the module (loss-of-coolant flow).

In the latter case the afterheat raises the temperature of the module at a linear rate of about 0.13 K/s. If no corrective action were taken for one hour, the blanket temperature would rise to less than 1200 K, since the blanket first wall can be kept nearly isothermal by conducting heat from the axial center, through the lithium coolant, and to the ends of the module.

If the failure occurs within the module itself and the lithium drains from the module, the temperature rise is about 0.17 K/s and, again, nearly an hour is allowed before untoward temperature excursions occur. Radiation from a single module is not as effective in this case, however, since the center of the first wall

can no longer conduct heat freely toward the end. If thermal conduction within the module is neglected, central first-wall temperatures will reach 1200 K at steady state.

Although the afterheat power density is small, it will nevertheless be necessary to cool the module while it is stored. If helium at one atmosphere pressure is used to cool the expended module, helium velocities of about 90 m/s (300 fps) in the coolant channel nearest the first wall will be required. This flow rate represents no more than a $\sim 10^4$ Pa pressure loss, and only about 0.5 kg/s of helium is required per module.

5.7 VERY UNUSUAL OCCURRENCES

Although a detailed accident analysis has not been made, the above calculations are indicative of the worst consequences expected to be incurred in the event of a serious cataclysm (earthquakes, tornadoes, airplane crashes, sabotage, dedicated vandalism, or acts of war). A major portion of the plant is located underground, and rather special means must be employed to make even the rupture by external forces of one system a credible occurrence. The plant contains nothing of interest to thieves, international bandits, or terrorists except the stored tritium inventory. This inventory can be made very secure by storage as a metal hydride with a very limited extraction system. Except for the initial shipment and possible surplus tritium exported to other plants, no tritium shipments are envisioned.

6. SUMMARY

This report indicates that the environmental impact of theta-pinch fusion reactors would be minimal. In terms of radioactivity and other hazards, material needs, and land despoilation, the anticipated advantages associated with the fuel supply and harmless reaction products appear to be without seriously offsetting disadvantages. More profound engineering design studies, however, are needed to substantiate these initial indications.

In summarizing the environmental impact of the specific RTPR power plant design given in Vol. I, effects can be identified which are either common to other large electrical power sources or which are unique to the RTPR power source (and by and large unique to fusion power in general). Although this exercise has focused primarily on environmental effects which are unique to fusion (tritium inventory, structural activation, recirculating power requirements, unique resource requirements), the former category is included in this summary to give some perspective to the environmental potential of fusion.

6.1 ENVIRONMENTAL ASPECTS OF FUSION POWER WHICH ARE COMMON WITH OTHER LARGE SOURCES OF ELECTRICAL ENERGY

Several environmental aspects appear identical with other large energy sources (fossil, fissile). The waste heat from a fusion power plant, like all other power plants, is dependent on the maximum thermal conversion temperature. If advanced (high-temperature) energy conversion cycles are used, the waste heat is reduced regardless of the specific heat source used. The waste heat problem could, in principle, be worse for fusion reactors than for fissile or fossil fuel power plants because of the large amount of circulating power required by the former. This problem to a large extent is alleviated in the RTPR power plant by the direct conversion of a portion (9.0%) of the total fusion energy.

Land despoilment by fusion reactors appears to differ little from other heat sources. Presently, the likelihood of siting the RTPR power plant (or any fusion power plant, for that matter) in an urban setting

and close to the load point appears to differ little from a fission or fossil power plant. Although differing significantly in kind, the mining requirements for fuel and structural materials taken together do not appear to differ significantly between fusion, fission, or fossil power plants.

6.2 ENVIRONMENTAL EFFECTS UNIQUE TO THE RTPR

The major environmental effects of a fusion reactor that employs D-T and magnetic confinement can generally be associated with (1) the need to breed tritium (tritium inventories, lithium burnout), (2) the need to store inductively vast quantities of energy and the associated resource requirements on niobium, helium, copper, and steel, (3) the need to operate radioactive components as well as to transport and to store radioactive waste, and (4) the need to replace on a regular basis substantial quantities of radiation-damaged metals for a given amount of energy generated. The RTPR power plant actively circulates 28.8 kg/day of tritium, stores 1.43 kg of tritium, and consumes 1.38 kg/day of tritium. Under optimistic conditions (10^{-10} Torr partial pressure of tritium in the lithium coolant, copper diffusion barrier in the heat exchanger) the leak rate of tritium to the environment (condenser water for the case of the present RTPR power plant design) amounts to 6 Ci/day. Since 81% of the breeding occurs in ^6Li , a total of 14,600 kg/yr of natural lithium is consumed for a breeding ratio of 1.11. Approximately 1.0×10^6 kg of niobium and 1.3×10^6 kg of helium is required for inductive energy storage, and approximately 76,700 kg/yr of radioactive niobium from the blanket (assumed 5-yr lifetime) must be handled (transported, stored, and reprocessed). An additional 1800 kg/yr of low-level radioactive niobium must be processed as a result of lithium corrosion.

The magnitudes of tritium losses, resource depletion, and radwaste problems pertain specifically to the 12 GWt (4.0 GWe) RTPR power plant design described in Vol. I. Most of these effects are highly design-dependent. For instance, depending on the

emphasis placed upon minimizing the tritium release rate and inventory, different coolant and breeding schemes can be adopted which, in principle, could reduce the 6-Ci/day tritium release rate to a level determined by the leak integrity of the helium purge system (0.01 to 0.1 Ci/day). The level of radwaste handled by the RTPR power plant could be reduced by the replacement of niobium in the blanket by metals which activate to a lesser extent. These examples serve to illustrate the variability of important environmental effects afforded by engineering design changes.

Once structural and blanket materials have been selected, however, the annual requirements for these materials show little sensitivity to design change. For a given total energy output, the total number of neutrons produced is roughly a constant. These neutrons will, therefore, lead to a given number of displaced atoms in the surrounding structural material. This radiation-damaged material must be replaced at a rate (19.1 kg/MWe-yr) which is roughly independent of the blanket design details. Hence, the need for blanket materials is nearly proportional to the integrated thermal output of the fusion power plant, and a strong economic incentive exists to attain that integrated thermal output in as short a period of time as possible (i.e., higher power ratings). The economic implications of the opposing requirements of duty cycle and materials limitations are discussed in Vol. III.

6.3 ACCIDENT POTENTIAL FOR THE RTPR POWER PLANT

As indicated in Sec. 5.1, a detailed safety analysis of the RTPR power plant would be premature and meaningless at this time. The discussion in Sec. 5 attempts to establish bounds on the magnitude of the safety question. In the unlikely event that the total circulating tritium inventory is suddenly released to the atmosphere, the external radiation dose to the surrounding populace (Fig. 5.1-1) would amount to 7.6 rem at 600 m and 1.8 rem at 3 km from the release point. A liquid-metal fire appears to represent a potential driving force for this kind of release, although the likelihood of an instantaneous and total release of all actively circulating tritium appears highly improbable. The loss-of-cooling accident appears more probable. However, if the LOCA is detected such that the reactor can be shut down, the damage to the reactor should not be serious. None of the possible consequences of the LOCA should propagate outside the reactor containment.

Although other accident situations have been discussed in Sec. 5., more must be known about the details of plasma physics and the engineering design before conclusive remarks can be made. However, based on the tentative evidence accumulated thus far, the RTPR power plant appears to have an extremely low potential for a publicly hazardous accident.

7. CONCLUSIONS

The potential environmental problem for the RTPR power plant design presented in Vol. I can readily be identified as either being common with other large-scale energy sources (fission or fossil) or being unique to fusion or the specifics of the RTPR design. Some of the environmental problems which fall into the latter category can be eliminated or greatly alleviated by design innovation.

A cursory evaluation of the accident potential for the RTPR indicates an extremely safe plant; the potential for a publicly hazardous accident is extremely low, and borders on the incredible. Even with preposterous postulations, the accidental

release of the entire circulating tritium inventory is not particularly serious. The most serious environmental effect is caused by the choice of niobium as a structural material. This choice seems particularly poor because of limited resources of the material and excessive care required to handle this metal; special techniques are needed to manufacture, fabricate, and weld niobium. Finally, large amounts of relatively long-lived radioactive waste material are produced. However, the radioactive waste generated is not toxic nor volatile and should be relatively easily handled and stored.

APPENDIX A

A.1 DESIGN DATA FOR THE REFERENCE THETA-PINCH REACTOR (RTPR)

RTPR Systems

1. Implosion (shock) heating system
2. Adiabatic compression system
3. Magnetic energy transfer system
4. First wall system
5. Blanket system
6. Biological shielding system
7. Fuel supply/ash removal system
8. Fuel processing system
9. Coolant processing system
10. Energy conversion system
11. Alternate potassium topping energy conversion system

1. IMPLOSION (SHOCK) HEATING SYSTEM

1. Plasma parameters at start of burn
2. Implosion heating coil (IHC) parameters
3. Design requirements

1. Plasma Parameters at Start of Burn

1. Compression field, kG	110
2. Compression ratio at start of burn	0.226
3. Radius of first wall, cm	50
4. Electron and ion temperature at start of burn, keV	6.1
5. Plasma radius at start of burn, cm	11.3
6. Temperature of ions after implosion heating (before equilibration), keV	2.28
7. Electron and ion temperature after implosion heating (after equilibration), keV	1.14
8. Implosion heating field, kG	13.8
9. Implosion velocity, cm/ μ s	20.1
10. Plasma back EMF, kV	650
11. Implosion heating electric field, kV/cm	2.0
12. Ambient filling pressure (300 K), mTorr	17
13. Ion density at start of burn, ions/cm ³	2.46E+16
14. Ratio of plasma to magnetic pressure	1

2. Implosion Heating Coil (IHC) Parameters

1. Mean radius of IHC, cm	90
2. Length of IHC, cm	100
3. Thickness of IHC winding, cm	1
4. Filling fraction for IHC	0.7
5. Implosion heating field, kG	13.8
6. Voltage of the IHC capacitor, keV	60
7. Total azimuthal voltage around IHC, keV	650

3. Design Requirements

1. Implosion heating field, kG	13.8
2. Time for field, ms	10
3. Risetime, microsec	0.1
4. Energy required by the implosion heating stage, MJ/m	1.66

2. ADIABATIC COMPRESSION SYSTEM

1. Design requirements
2. Mechanical stresses in the compression coil
3. Compression coil cooling parameters

1. Design Requirements

1. Coil winding thickness, mm	1.5
2. Number of turns in compression coil	187
3. Width of compression coil, m	0.5
4. Neutron heating in coil, MJ/m	0.57
5. Gamma-ray heating in coil, MJ/m	4.88
6. Total energy generated in coil, MJ/m	11.6
7. Increase in temperature of copper coil, K/pulse	2.0
8. Energy necessary to remove from coil (for a 10 sec duty cycle), MW/m	1.5
9. Magnetic field requirements:	

Field Strength (kG)	Risetime* (ms)	Duration* (ms)
13.8-110	20	
110		80
110-32.2	20	
32.2-0	1000	
0		~ 8000

* Nominal values used (see footnote on page 7).

2. Mechanical Stresses in the Compression Coil

1. Adiabatic compression field, kG	110
2. Outward pressure exerted by field on compression coil, MPa	43
kpsi	6.3
3. Average hoop stress, MPa	125
kpsi	18.5

3. Compression Coil Cooling Parameters

1. Neutron heating, MJ/m	0.57
2. Gamma-ray heating, MJ/m	4.88
3. Transport current joule heating, MJ/m	5.96
4. Eddy-current joule heating, MJ/m	0.18
5. Length of compression coil, m	0.5
6. Inner diameter of compression coil, m	1.9
7. Outer diameter of compression coil, m	2.7
8. Conductor (turn) thickness, mm	1.5
9. Number of turns	187
10. Current per turn, kA	24
11. Insulation thickness, mm	0.12
12. Insulation thickness (glass), mm	0.95
13. Mean coil temperature, C	25
14. Coolant inlet temperature, C	20
15. Coolant outlet temperature, C	23
16. Mean heat transfer coefficient, BTU/hr ft ² °F	500
17. Average temperature rise in coil per pulse, C	2.4
18. Weight of copper, kg/module	37,000
19. Weight of coolant per module, kg	600
20. Coolant pressure drop through coil, psi	<1
21. Total coolant pressure drop, psi	1

3. MAGNETIC ENERGY TRANSFER SYSTEM

1. Design parameters for a superconducting energy transfer module

1. Design Parameters for a Superconducting Energy Transfer Module

1. Coupling coefficient, k_T^2	0.98
2. Coupling coefficient, k_m^2	0.9
3. Transfer efficiency, %	62.3
4. Field strength of coil 1, kG	60
5. Total magnetic energy stored, MJ	286
6. Energy transferred to load, MJ	178
7. Outer coil radius, cm	182
8. Inner coil radius, cm	145
9. Intermediate coil radius, cm	176

10. Ampere turns, A	2.60E + 07
11. Ampere turns, A	2.07E + 07
12. Ampere turns, A	1.41E + 07
13. Current density in superconductor, A/cm ²	1.0E + 05
14. Coil thickness, mm	7.16
15. Outer coil thickness, mm	7.16
16. Intermediate coil thickness, mm	4.02
17. Inner superconductor volume, cm ³	2.34E + 05
18. Outer superconductor volume, cm ³	1.49E + 05
19. Intermediate superconductor volume, cm ³	1.23E + 05
20. Moment of inertia of inner superconductor winding, kg/m ²	2431
21. Moment of inertia of outer superconductor winding, kg/m ²	978
22. Moment of inertia of intermediate superconductor winding, kg/m ²	1185
23. Refrigerator power, MWe	600

4. FIRST WALL SYSTEM

1. First wall thermal and radiation flux loading
2. Parameters used in first wall heat transfer calculations

1. First Wall Thermal and Radiation Flux Loading

14-MeV neutron loading, MW/m ²	2.0
Burn time, sec	0.080
Cycle time, sec	10.0
Radius of the first wall, cm	50.0
1. Uncollided (14.08 MeV) neutron current averaged over the reactor cycle, n/cm ² sec	8.91E + 13
2. Uncollided (14.08 MeV) neutron current averaged over the burn, n/cm ² sec	1.10E + 16
3. Total neutron flux averaged over the reactor cycle, n/cm ² sec	9.03E + 14
4. Fluence of 14.08 MeV neutrons, n/cm ² year	2.84E + 21
5. Current of 14.08 MeV neutrons/pulse, n/cm ² pulse	8.91E + 14
6. Neutron power density in alumina, J/cm ³ pulse	156.3
7. Gamma ray power density in alumina J/cm ³ pulse	63.5
8. Neutron power density in niobium, J/cm ³ pulse	23.6
9. Gamma ray power density in niobium, J/cm ³ pulse	183.4
10. Bremsstrahlung energy flux at the first wall, J/cm ² pulse	71.62
11. Bremsstrahlung power density in alumina, J/cm ² pulse	1432.4
12. Bremsstrahlung power density in niobium, J/cm ³ pulse	286.5
13. Total power density in alumina, J/cm ³ pulse	1652.2
14. Total power density in niobium, J/cm ³ pulse	493.5

2. Parameters used in First Wall Heat Transfer Calculations

1. Inside radius of niobium wall, cm	50.00
2. Thickness of alumina insulator, cm	0.030
3. Thickness of niobium wall, cm	0.100
4. Thickness of lithium, cm	10.00
5. Neutron energy deposited in blanket, MJ/m	93.1
6. Bremsstrahlung energy during burn, MJ/m	2.25
7. Internal energy of plasma at the time of cooling, MJ/m	4.13
8. Heat capacity of alumina, J/gK	$0.4184T^{0.18}$
9. Heat capacity of niobium, J/gK	0.335
10. Heat capacity of lithium, J/gK	4.184
11. Thermal conductivity of alumina, W/cmK	$390.0T^{1.20}$
12. Thermal conductivity of niobium, W/cmK	0.700
13. Thermal conductivity of lithium, W/cmK	0.500
14. Density of alumina, g/cm ³	3.80
15. Density of niobium, g/cm ³	8.55
16. Density of lithium, g/cm ³	0.43

5. BLANKET SYSTEM

1. Position and thickness of components in blanket and shield
2. Blanket parameters: area, volume, density, weight
3. Surface areas per 2-m segment

1. Position and Thickness of Components in Blanket and Shield

Layer	Inner Radius (cm)	Outer Radius (cm)	Thickness (cm)
1. First wall	50.0	50.1	0.1
2. Lithium	50.1	53.1	3.0
3. Beryllium	53.2*	55.7	2.5
4. Lithium	55.8*	60.3	4.5
5. Graphite	60.4**	64.9	4.5
6. Lithium	65.0**	67.7	2.7
7. Graphite	67.8**	82.6	14.9
8. Enriched ⁶ Li	82.7**	88.9	6.2
9. Alumina + Vacuum	89.0*	89.5	0.5
10. Implosion coil	89.5	90.5	1.0
11. Alumina + vacuum	90.5	93.5	3.0
12. S.S. ring	93.5	93.5	0.01
13. Compression coil	93.5	133.5	40.0
14. Titanium	133.5	143.5	10.0
15. Vacuum	143.5	150.0	6.5
16. Biological shield	150.0	Variable	> 200.0

*Indicates allowance for thickness of Nb wall.

**Indicates allowance for thickness of Nb-alumina-Nb wall.

2. Blanket Parameters: Area, Volume, Density, Weight per Module

Layer	Cross Sec- tional Area (cm ²)	Volume (cm ³)	Density (g/cm ³)	Weight (kg)
1. Lithium (1)	894	178,800	0.43	77
2. Beryllium	790	158,000	1.85	292
3. Lithium (2)	1521	304,200	0.43	131
4. Graphite	1601	320,200	2.26	723
5. Lithium (3)	977	195,000	0.43	84
6. Graphite	1957	1,391,400	2.26	3150
7. Lithium 6	3234	646,800	0.43	278
8. Insulator	3306	661,200	3.80	2520
9. Niobium		250,000	8.58	2180

3. Surface Areas per Module

1. Niobium segment walls, m ²	320
2. Niobium dividers, m ²	135
3. Alumina wall coating, m ²	320
4. Alumina divider coatings, m ²	64
Length of niobium welds, m	4580

6. BIOLOGICAL SHIELDING SYSTEM

1. Reactor module assembly

1. Reactor Module Assembly

1. Thickness of concrete shield, m	1.8
2. Neutron flux at inner surface of shield, n/s cm ²	4.5E + 11
3. Neutron flux at outer surface of shield, n/s cm ²	2.5E - 02
4. Photon flux at inner surface of shield, gammas/s cm ²	2.5E + 10
5. Photon flux at outer surface of shield, gammas/s cm ²	2.3
6. Biological dose rate at outer surface of shield, mrem/h	0.02
7. Maximum permissible dose rate for free access areas, mrem/h	0.08
8. Maximum radiation heating rate in concrete shield, W/g	5.7E - 04

7. FUEL SUPPLY/ASH REMOVAL SYSTEM

1. Design parameters
2. Vacuum pumping system

1. Design Parameters (per Module)

1. Pressure of D-T mixture at 810 K, mTorr	46
2. Temperature of gas, °K	810
3. Filling pressure at 300 K, mTorr	17
4. Filling pressure at 810 K, mTorr	46
5. Volume of 50-cm-radius chamber which must be filled, liters,	1500
6. Required flow rate, Torr-liters/s	250
7. Pumping speed in chamber, liters/s	14,000
8. Residual helium, atom percent	0.6

2. Vacuum Pumping System

1. Number of first-stage vacuum pumps (19 kW)	352
2. Number of second-stage vacuum pumps (15 kW)	352
3. Number of other vacuum pumps (16 kW)	20
4. Total power requirement for vacuum pumps (MWe)	12.3

8. FUEL PROCESSING SYSTEM

1. Design parameters

1. Design Parameters

1. Fuel ash flow, g/hr of gas	10,000
2. Molar composition of gas, %	
A) Deuterium	49
B) Tritium	49
C) Helium	1
D) Protium	<1
3. Mean temperature of gas, K	800
4. Mixture equilibration point, K	50
5. Number of plates in each distillation column	60
6. Space between columns, in.	2.5
7. Diameter of column 1, in.	6
8. Diameter of column 2, in.	3
9. Height of columns 1 and 2, ft	15
10. Added diameter due to insulator, in.	2.4
11. Plate efficiency, %	30
12. Relative volatility of H-D compared to H-T	1.4

9. COOLANT PROCESSING SYSTEM

1. Design requirements

1. Design Requirements

1. Partial pressure of tritium in lithium, Torr	<1.0E - 10
2. Temperature of lithium, C	550
3. Number of salt contactors (10 s cycle)	50
4. Contactor diameter cm	50
5. Contactor height cm	43
6. Contactor power MW (10 s cycle)	1.0

10. ENERGY CONVERSION SYSTEM

1. Surge tanks
2. Primary lithium pumps
3. Secondary sodium pumps
4. Intermediate heat exchanger
5. Intermediate heat exchanger heat transfer data
6. Steam generator
7. Steam cycle parameters
8. Power plant auxiliary power
9. Power summary

1. Surge Tanks

1. Number of surge tanks	352
2. Flow velocity, ft/s	0.5
3. Capacity ft ³	11

2. Primary Lithium Pumps, $\tau_c = 3$ s, $\eta_{TH} = 40\%$

1. Number of pumps	8
2. Pump stages	Single
3. Pump type	Centrifugal
4. Flow rate, lb/s	6250
gpm	103,500
5. Head, ft Li	160
psi	30
6. NPSH (operating) ft Li	80.5
(Limit)	50
7. Speed, rpm	450
8. Port size, intake diameter, in.	48
Outlet diameter, in.	48
9. Power required, hp	2485
MWe	1.85

3. Secondary Sodium Pumps, $\tau_c = 3$ s, $\eta_{TH} = 40\%$

1. Number of pumps	8
2. Pump stages	Single
3. Pump type	Centrifugal
4. Flow rate, lb/s	14,800
5. gpm	122,000
5. Head, ft NA	115
psi	43.3
6. NPSH, ft NA	92
psi	34
7. Speed, rpm	811
8. Port size, intake diameter, in.	48
Outlet diameter, in.	34
9. Power required, hp	3820
MWe	2.85

4. Intermediate Heat Exchanger, $\tau_c = 3$ s, $\eta_{TH} = 40\%$

Parameter	Shell	Tubes
1. Fluid	Lithium	Sodium
2. Inlet temperature, F	1004	675
3. Outlet temperature, F	775	971
4. Inlet pressure, psia	29	100
5. Outlet pressure, psia	17	88
6. Flow rate, lb/s	6250	14,800
7. Flow velocity, ft/s	5	11.3
8. Length, ft	42.5	32
9. Diameter, in.	168	0.75
10. Thickness, in.	3.5	0.042
11. Sectional areas, sq ft	36	24.6
12. Surface area, sq ft	1800	56,520
13. Number of tubes		9000
14. Holdup, cu ft	3100	1780
15. Intake diameter, in.	36 (2)	48 (1)
16. Outlet diameter, in.	36 (2)	34 (2)
17. Weight, lb	300,000	106,000

5. Intermediate Heat Exchanger Heat Transfer Data, $\tau_c = 3$ s, $\eta_{TH} = 40\%$

1. Number of heat exchangers	8
2. Power rating, MWT (each)	1500
	91.42
4. Heat transfer area, sq ft	56,520
5. Overall heat transfer coefficient, BTU/h·sq ft·F	1400
6. Log mean temperature difference, F	63
7. Material (alloy)	Niobium
8. Total dry weight, lb	500,000

6. Steam Generator, $\tau_c = 3$ s, $\eta_{TH} = 40\%$

Parameter	Shell	Tubes
1. Number of steam generators	8	
2. Fluid	Sodium	Steam
3. Inlet temperature, F	965	500
4. Outlet temperature, F	650	903
5. Inlet pressure, psia	32	2660
6. Outlet pressure, psia	17	2435
7. Flow rate, lb/s	14,800	1560
8. Power rating, MWt	1500	1500
9. Flow speed, ft/s	1.42	
10. Length, ft	74.6	142.6
11. Diameter, in.	180	0.5
12. Thickness, in.	2.5	0.065
13. Material	304SS	INCO800
14. Surface area, sq ft	3532	67,275
15. Number of tubes		3600
16. Holdup, cu ft	8597	552
17. Intake diameter, in.	30 (2)	8 (6)
18. Outlet diameter, in.	30 (2)	12 (6)
19. Weight, lb	400,000	200,000

7. Steam Cycle Parameters, $\tau_c = 3$ s, $\eta_{TH} = 40\%$

Four Turbine-Generator Type: Tandem-Compound Quadruple Flow

1. Heat input, MWt	3000
2. Steam flow rate, lb/s	3134
3. Temperature at throttle, F	900
4. Pressure at throttle, psig	2400
5. Gross power output, MWe	1275
6. Gross thermal efficiency, %	42.5
7. Number of condenser units	2
8. Number of feedwater heaters	14
9. Condenser steam pressure, in. Hg	1.5
10. Condenser heat transfer surface, sq ft/unit	247,000
11. Feedwater pump power, MW	37
12. Condenser cooling water pump power, MW	3.3
13. Condenser cooling water inlet temperature F	57

8. Power Plant Auxiliary Power, $\tau_c = 3$ s, $\eta_{TH} = 40\%$

Subsystem	Rated Power (MW)	Total (MW)
1. Helium liquefiers for S.-C. energy transfer modules	600	600
2. Fuel/ash system:		
1st stage vacuum pumps, 352 at 20 kW	7.2	
2nd stage vacuum pumps, 352 at 16 kW	5.8	
Tunnel pumps, 20 at 16 kW	0.3	
Compressors, fuel/ash, compressor, interstage, compressor, to storage, compressor, inert gas cooling, 10 at 40 kW	0.4	13.7
3. Coolant processing system:		
Molten salt pumps, 200 at 1 kW	0.2	
Contactor drive, 200 at 20 kW	10	0.7
4. Implosion heating and compression coils:		
Cooling pumps	16	
Ohmic loss in busbars (6000 ft per module, 2-in. sq busbars)	120	136
5. Power plant:		
Lithium pumps, 8 at 1850 kW	14.8	
Sodium pumps, 8 at 2850 kW	22.8	
Boiler feed pumps, 4 at 37 MW	149	
Condenser coolant pumps, 8 at 1.6 MW	13.2	
Miscellaneous pumps, compressors	8.0	207.8
9. Power Summary		
Total auxiliary power, MW	968.2	
Gross electric power, MW		5100
Net electric power, MW		4132

11. POTASSIUM TOPPING ENERGY CONVERSION SYSTEM, $\tau_c = 10$ s, $\eta_{TH} = 56.4\%$

1. System parameters
2. Potassium boilers
3. Major lithium piping runs
4. Major potassium piping runs

1. System Parameters $\tau_c = 10$ s, $\eta_{TH} = 56.4\%$

1. Thermal power, MWt	3600
2. Lithium temperature, in/out, K	1273/1400
3. Lithium pressure, in/out, psi	30/25
4. Lithium flow rate, kg/h	25E + 06
5. Lithium pump, hp	0.4E + 06
6. Potassium temperature from primary heat exchanger, K	1255
7. Potassium temperature from turbine, K	866
8. Potassium temperature from feed heater, K	1190
9. Potassium pressure from primary heat exchanger, atmospheres	5.5
10. Potassium pressure from turbine, atmospheres	0.15
11. Enthalpy rise in boiler, BTU/lb	800
12. Potassium flow rate, kg/s	1950

2. Potassium Boilers

1. Height, ft	60
2. Diameter, ft	6
3. Tube diameter, in.	1
4. Mean temperature difference, F	5
5. Heat transfer coefficient, BTU/sq ft·h·F	2660
6. Number of tubes per boiler	910
7. Tube wall thickness, in.	0.042
8. Weight of tubes, lb	27,000
9. Weight of shell, lb	50,000

3. Major Lithium Piping Runs

Parameter	Location			
	Module to Manifold	Outlet Manifold	Manifold to K Boiler	K Boiler to Contactor
1. Temperature, F	2050	2050	2050	1832
2. Pressure, psig	17	14	14	2
3. Density, lb/cu ft	25	25	25	27
4. Flow rate, lb/s	85.2	1560	3750	3750
5. Velocity, ft/s	17	9	2	1
6. Length, ft	50	290	130	75
7. Diameter, in.	6	36	48	48
8. Thickness, in.	0.085	0.25	0.5	0.25
9. Sectional area, sq ft	0.2	7.06	12.5	12.5
10. Surface area, sq ft	78.5	2730	1650	960
11. Weight/pipe, tons	0.15	0.15	17.5	6
12. Pressure loss, psi	1	0.1	0.3	0.2
13. Number of pipes	176	4	2	2
Total weight, tons	26	60	35	12

Parameter	Location			
	Contactor to Pump	Pump to Manifold	Inlet Manifold	Manifold to Modules
1. Temperature, F	1832	1832	1832	1832
2. Pressure, psig	0	30	30	30
3. Density	27	27	27	27
4. Flow rate, lb/s	3750	3750	1560	85.2
5. Velocity, ft/s	1	1	8	16
6. Length, ft	25	120	2	50
7. Diameter, in.	48	48	36	6
8. Thickness, in.	0.25	0.25	0.072	
9. Sectional area, sq ft	12.5	12.5	7.06	0.2
10. Surface area, sq ft	320	1520	1730	78.5
11. Weight, tons	1.8	8.4	0.15	0.15
12. Pressure loss, psi	0.1	0.2	0.1	1
13. Number of pipes	2	2	4	176
Total weight, tons	3	16	60	26

Total weight of lithium piping = 238 tons

4. Major Potassium Piping Runs

Parameter	Location			
	Boiler to Turbine	Turbine to Condenser	Condenser to Pump	Pump to Boiler
1. Temperature, F	1800	1100	1100	1700
2. Pressure, psia	80	2.4	2.4	80
3. Density, lb/cu ft	0.15	0.0057	43	43
4. Flow rate, lb/s	1072	1072	1072	1072
5. Velocity, ft/s	300	1200	15	15
6. Length, ft	50	--	20	60
7. Diameter, ft	5.5	14	1.5	1.5
8. Thickness, in.	1.0	0.42	0.035	0.25
9. Sectional area, sq ft	23.75	154	1.75	1.75
10. Surface area, sq ft	863	--	95	246
11. Weight, tons	20			1.5
12. Pressure loss, psi	0.5	--	0.5	1.0
13. Number of pipes	4	4	4	4
Total weight, tons	80	--	--	6

Total weight of potassium piping, tons:

Niobium	80
S.S.	6

APPENDIX B

B.1 DETERMINATION OF PHYSICAL PROPERTIES WHICH AFFECT TRITIUM DISTRIBUTION AND LEAKAGE

The important properties that will determine hydrogen (isotope) distribution and leakage characteristics in the RTPR are solubilities in structural components (niobium, Al_2O_3 , stainless steel) and the stability of the various chemical forms in which the hydrogen is expected to exist.

To make a reasonable estimate of the amount of tritium dissolved in the structure and fluids, an expression for the solubility of tritium in the structural materials must be obtained. For the case of dilute solutions of hydrogen (isotopes) in metals, the relationship between solubility and pressure is adequately given by Sieverts' law³⁵

$$C = K_s \sqrt{P} \quad (\text{B-1})$$

where C is the concentration, P is the partial pressure of gas in equilibrium with the metal, and K_s is exponentially dependent on temperature.³⁶ Hence, the general expression relating the concentration of hydrogen dissolved in the metal to the equilibrium hydrogen pressure and temperature is given by

$$C = K_s^0 \sqrt{P} e^{-Q_s/kT} \quad (\text{B-2})$$

where K_s^0 is a material-dependent constant, Q_s is the activation energy for solution of hydrogen in the metal, k is the Boltzmann constant, and T is the absolute temperature. Values of K_s and Q_s for materials of interest to fusion reactors are summarized in Table B-1.

The migration rates of hydrogen isotopes through structural interfaces have been reviewed by Webb⁴¹ and Stickney⁴² and can be determined from permeation data available for the materials of construction. The relationship between permeation rate, surface areas, thicknesses, and pressure is given by Eq. (B-3)

$$\Phi = \left(\frac{A}{X} \right) \cdot (\sqrt{P_H} - \sqrt{P_L}) \cdot B \cdot e^{-Q_p/kT} \quad (\text{B-3})$$

where Φ is the flux of hydrogen through the interface, A and X are, respectively, the area and thickness of the interface, P_H and P_L are the hydrogen pressures on each side of the interface ($P_H > P_L$), and B and Q_p are parameters related to the material under consideration. Values of B and Q_p for the permeation of protium through materials used in the reference design model are summarized in Table B-2. In most cases only limited data are available for deuterium and tritium permeation through these materials; the deuterium and tritium permeation rates were estimated by using the values of B and Q_p for protium and multiplying the result obtained from Eq. (B-3) by the square root of the mass of the hydrogen isotope of interest.*

Most of the above properties (particularly the permeation rates for the hydrogen isotopes) are strongly affected by the nature of the hydrogen-bearing species at various interfaces. The species H_2 , D_2 , and T_2 have been identified as being important in a lithium blanket system. The extent to which other species (HD , HT , and DT) are formed is fixed by the concentrations of H_2 , D_2 , and T_2 . In the case of the RTPR, a number of other hydrogen-bearing species will also be important. For instance, the circulating helium purge system will contain trace amounts of molecular oxygen. Because of the temperature and the oxidation potential in this circuit, most of the molecular hydrogen in this environment is expected to react with the oxygen present to form water. Watson⁴⁴ and Johnson⁴⁵ have considered methods of

*Because B is expressed in weight-units (Table B-1), values of b for protium must be multiplied by $2/\sqrt{2} = 1.41$ and $3/\sqrt{3} = 1.73$ to obtain estimates for deuterium and tritium, respectively.

TABLE B-1

**PARAMETERS FOR THE SIEVERT'S LAW CONSTANTS,
K_s, OF H₂, D₂, and T₂ IN THE REFERENCE
DESIGN CONSTRUCTION MATERIALS**

$$K_s = K_s^0 \exp(Q_s/kT)$$

Material	K _s (wppm ^a /Torr ^{1/2})			Q _s (cal/mole)			Reference
	H ₂	D ₂	T ₂	H ₂	D ₂	T ₂	
304-SS	0.4	0.7	1.1	-1,680	-1,750	-1,830	(b)
Incoloy 800	0.4	0.7	1.1	-1,680	-1,750	-1,830	(b)
Copper	13.6	(27.2) ^c	(40.8) ^c	-13,911	(-13,911) ^c	(-13,911) ^c	37
Niobium	0.35	0.49	0.60	8,430	8,430	8,430	38
Lithium	7.9	13.0	17.9	12,284	12,284	12,284	39
Sodium	7.2	(14.4) ^d	(21.6) ^d	- 550	(- 550) ^d	(- 550) ^d	40

^awppm = weight parts per million H, D, or T.

^bCalculated by using the data of Ref. 36 for 347-SS and assuming the values of K_s and Q_s for 304-SS and Incoloy 800 would be essentially the same as for 347-SS. (Values of K_s for 304-SS and Incoloy 800 are not expected to be significantly different from those for 347-SS based on previous studies of the effect of composition on solubility of hydrogen in Fe-Cr-Ni alloys.)

^cApproximated by assuming the same parameters that are given for H₂ in Ref. 39.

^dApproximated by assuming the same parameters that are given for H₂ in Ref. 40.

TABLE B-2

**PARAMETERS FOR THE PERMEABILITY, φ, OF PROTIUM^a
THROUGH THE REFERENCE DESIGN CONSTRUCTION MATERIALS**

Material	(g cm/cm ² day Torr ^{1/2})	Q _p (cal/mole)	Reference
304-SS	0.65 × 10 ⁻²	16,100	41
Incoloy 800	0.65 × 10 ⁻²	16,100	(b)
Copper	0.86 × 10 ⁻²	18,562	(c)
Niobium	1.55 × 10 ⁻²	5,200	42

^aValues of φ for D₂ and T₂ were obtained by multiplying the values for H₂ by the $\sqrt{2}$ and the $\sqrt{3}$, respectively.

^bThe values of φ for hydrogen in Incoloy 800 are not expected to be significantly different from those for 304-SS based on previous studies (Ref. 36, p. 77) of the effect of composition on permeability of hydrogen in Fe-Cr-Ni alloys.

^cDetermined by least square refinement of the data presented in Ref. 43.

using this oxidation to advantage in creating barriers to tritium migration.

For purposes of the calculational model being developed the following equilibria between oxygen and hydrogen are assumed to be important.

The relationships between the partial pressures of the species involved in these equilibria and the equilibrium constants, k_1 , k_2 , and k_3 are given below.

$$\left. \begin{aligned} k_1 &= \frac{P_{\text{HT}}^2}{P_{\text{H}_2} \cdot P_{\text{T}_2}} \\ k_2 &= \frac{P_{\text{H}_2\text{O}}}{P_{\text{H}_2} \cdot (P_{\text{O}_2})^{1/2}} \\ k_3 &= \frac{P_{\text{HTO}} \cdot P_{\text{H}_2}}{P_{\text{H}_2\text{O}} \cdot P_{\text{HT}}} \end{aligned} \right\} \quad (\text{B-5})$$

The temperature dependence of k_1 , k_2 , and k_3 (Refs. 8, 46, 47) was analyzed to give a least-square fit to the following expression:

$$\ln(k_i) = A_i + B_i \ln(T) + C_i/T \quad (\text{B-6})$$

The values for the fitting parameters A_i , B_i , and C_i , obtained in this analysis are given in Table B-3.

Calculations using the constants k_1 , k_2 , and k_3 over typical operating temperature ranges for fusion reactors have revealed several interesting features.⁴⁵ For example, by rearranging Eqs. (B-5) the following relationships result:

$$\left. \begin{aligned} (P_{\text{O}_2})^{1/2} \cdot k_2 &= \frac{P_{\text{H}_2\text{O}}}{P_{\text{H}_2}} \\ (P_{\text{O}_2})^{1/2} \cdot k_2 \cdot k_3 &= \frac{P_{\text{HTO}}}{P_{\text{HT}}} \end{aligned} \right\} \quad (\text{B-7})$$

Equations (B-7) can be used to determine the relative amounts of water and molecular hydrogen as a function of the oxygen partial pressure. For temperatures between 300 and 930 K, the value of k_2 is greater than $10^{12} \text{ Torr}^{-1/2}$ and k_3 is a dimensionless number in the range from 1.0 to 5.0. For an oxygen partial pressure equal to 10^{-6} Torr in the helium purge system (if acceptable with respect to materials limitations), the predominant hydrogen-bearing species in the helium will be water (H_2O , HTO , or T_2O) by a factor of more than 10^9 if the reaction kinetics can be made favorable. This situation will exist regardless of the relative amounts of H_2 and T_2 entering the helium circuits. The H/T ratio in the helium affects only the relative amounts of H_2O , HTO , and HT formed.

The net result of these water-producing reactions is that the partial pressure of the molecular forms of hydrogen in the helium purge zones may be reduced to a value many orders of magnitude lower than in adjacent systems.^{44,45} Therefore, the helium purge flow system should prevent migration of tritium beyond the immediate containment shell of the equipment being purged. Since this purge system will undoubtedly operate at a lower temperature than the value used above, (930 K), the formation of water should be thermodynamically even more favorable.

To take advantage of the "gettering" capability of low levels of oxygen in the helium circuits, a processing scheme must be devised to remove and collect the HTO and H_2O , recover and separate the hydrogen isotopes, and recycle the tritium to the fueling system. Watson⁴⁴ and Johnson⁴⁵ considered methods for accomplishing these steps, and concluded that a drying agent (e.g., magnesium perchlorate) would probably be adequate for removing water from the helium in relatively simple apparatus. Molecular sieves could also be applied to these process steps.

TABLE B-3
PARAMETERS FOR THE EQUILIBRIUM CONSTANTS
 $k_1, k_2, \text{ and } k_3.$
 $\ln(k_i) = A_i + B_i \ln(T) + C_i/T$

Equilibrium Constant	Units	Reference	Fitting Parameters		
			A_i	B_i	C_i
k_1	Dimensionless	(a)	1.2787	0.0265	-141.34
k_2	Torr ^{-1/2}	(b)	-3.7675	0.	30,428
k_3	Dimensionless	8	-2.4292	0.2920	774.82

^aObtained by least-squares fitting of the data of Ref. 46.

^bObtained by least-squares fitting of the data of Ref. 47.

In summary, the solubility, diffusion, and oxidation characteristics of hydrogen (isotopes) have been

reviewed and existing data interpreted for use in a generalized tritium mapping model. This model is described in Appendix C.

APPENDIX C

C.1 DESCRIPTION OF MODEL USED TO MAP TRITIUM DISTRIBUTION IN RTPR POWER PLANT

Any tritium entering the steam system is presumed lost to the environment through blowdown paths in the steam generator. For these reasons the mapping model described herein has been applied solely to the lithium-processing heat-exchange zone of the RTPR, although this model is perfectly general and, in principle, can be applied to other subsystems when either design detail becomes available or the seriousness of a given tritium leakage path warrants a more detailed analysis. A leakage rate constant (volume percent per unit time) is assumed for the remaining tritium zones, and the helium purge system is assumed efficient enough to maintain a low tritium level within the helium. The validity of this assumption will depend on the leakage rate from specific equipment located within a given tritium zone and the efficiency with which tritiated water can be removed from the helium purge gas. Closer examination of these assumptions must await future design iterations and additional experimental data.

To apply the permeation-solubility equations developed in Appendix B in the determination of tritium leakage and inventory for a given subsystem(s), the subsystem(s) is divided into N isothermal regions. A ~ 40 K temperature difference across an "isothermal" zone is presumed adequate for most calculations. The fractional area of a particular subsystem in a given temperature zone is determined by the temperature profile across the entire subsystem. The hydrogen (isotope) activity gradient across a given interface is assumed to result in a hydrogen partial pressure ratio, P_H/P_L , greater than 10^2 . This assumption allows the average concentration of hydrogen within the structural material of an interfacial subsystem to be equal to roughly one-half that calculated from Eq. (B-1).*

$$* \langle S \rangle = 0.5 (\sqrt{kP_H} + \sqrt{kP_L}) K_s \cong 0.5 \sqrt{kP_H}$$

The amount of dissolved hydrogen, S_i , in the i^{th} temperature zone is calculated from

$$S_i = 0.5 (AF_i X_i D) (\sqrt{P_H} + \sqrt{P_L}) K_s e^{-Q_s/kT_i} \quad (C-1)$$

where

A = the total area,
 F_i = the fraction of the total area in the i^{th} zone,
 X_i = the thickness of the material in the i^{th} zone,
 D = density of the material,
 T_i = the absolute temperature of the i^{th} zone,
and K_s and Q_s are as given in Table B-1.

The rate of hydrogen permeation, ϕ_i , in the i^{th} temperature zone is determined by,

$$\phi_i = \frac{AF_i}{X_i} \sqrt{M} (\sqrt{P_H} - \sqrt{P_L}) B e^{-Q_p/kT_i} \quad (C-2)$$

The parameters on the right-hand-side of Eq. (C-2) are the same as those defined for Eqs. (B-3) and (C-1) and in Table B-2; the quantity M is the mass of the hydrogen isotope.

Once determined, the S_i and ϕ_i are summed over the N temperature zones to give the total concentration, S , and permeation rate ϕ for the subsystem being investigated.

$$S = \sum_{i=1}^N S_i$$

$$\phi = \sum_{i=1}^N \phi_i \quad (C-3)$$

Equations (C-1), (C-2), and (C-3) were programmed into a computer code for solution. The text of the computer program is given in Appendix D and examples of the output for an isobaric subsystem and an interfacial subsystem are given. The formalisms used in writing the code are summarized in Appendix D as are the procedures for reading in the necessary dimensional and operational parameters.

In applying this calculational model to the RTPR, several restrictions were necessary. For the blanket and heat transfer systems, the assumption was made that only protium and tritium are present in significant amounts. The predominant hydrogen-bearing species in the blanket are assumed to be H_2 and T_2 . The important sources of protium to the blanket are considered to be (n,p) reactions as well as diffusion from the steam system into the sodium and eventually into the lithium coolant. Because of isotopic swamping effects, which should occur in the steam system, only H_2 , H_2O and HTO are considered important in the steam. The primary fuel system is composed almost exclusively of D_2 , DT , and T_2 ; very little H_2 , HT , or HD is present because the steady-state protium level of the fuel stream is of necessity very low.

Several other restrictions were previously mentioned. The assumption has been made that the hydrogen isotope partial pressure drop across any interfacial barrier is at least a factor of 100 (i.e., $P_H/P_L \geq 10^2$). The driving forces to maintain

pressure ratios of this magnitude are as follows. As a result of the fueling and burning characteristics of the reference design, the partial pressures of deuterium and tritium species on the plasma side of the vacuum wall will fluctuate in the range 10^{-3} to 10^{-5} Torr; the partial pressures of protium-bearing species will be considerably lower. The lithium will be processed so that the partial pressure of the tritium-bearing species T_2 and HT is $\leq 10^{-10}$ Torr. Maintaining a partial pressure difference of this magnitude between the lithium and the plasma chamber may prove to be difficult, but the presence of the insulator (Al_2O_3) will help to mitigate this problem. The use of hydrogen activity meters on the exit manifold of the lithium coolant line for each module should prove to be a sensitive method for detection of failed, first-wall insulators. If the leaking module(s) unduly burdens the lithium processing system, it will be replaced.⁴⁶ The partial pressures of HT and T_2 in the helium purge gas could be reduced to low levels by maintaining an acceptable level of O_2 in the helium and removing the H_2O and HTO at a fast rate. For all of the subsystems included in the primary loop-processing system, the partial pressures of HT and T_2 just outside the primary containment of each subsystem can in principle be maintained at levels $\leq 10^{-15}$ Torr by the helium purge flow system. The results of the application of this code to the lithium-sodium and sodium-steam heat exchangers are given in the following section.

APPENDIX D

D.1 COMPUTER LISTING AND SAMPLE RESULTS FOR THE TRITIUM MAPPING CALCULATION

This appendix contains a listing of MAP3 together with MAP3 output for the heat exchanger network of the reference design model. MAP3 (a subroutine of HDTMAP) calculates the solubility and permeability of hydrogen isotopes in and through individual structural segments of a fusion reactor system or component. The program HDTMAP, which is currently being developed, contains a driver program and the subroutines MAP3, BMAPLM, and BMAPMS.

BMAPLM and BMAPMS give a detailed analysis of the hydrogen isotope migration and distribution profile in a liquid metal or molten salt blanket or coolant fluid. The driver program sorts input data, selects the appropriate subroutine for data analysis, and lists cumulative results of solubility and permeability calculations on all components and systems. Whereas MAP3 analyzes each component individually, it is intended that HDTMAP will provide a comprehensive analysis of an entire fusion power plant. The comments in the attached listing of MAP3 contain information pertinent to the use of this code.

MAP3 LISTING

MAP3 CALCULATES THE SOLUBILITY AND PERMEABILITY OF HYDROGEN ISOTOPES IN AND THROUGH INDIVIDUAL STRUCTURAL SEGMENTS OF A FUSION REACTOR SYSTEM OR COMPONENT

DIMENSION TITLE(1:20), T1, L2(20), T(20), C(20), A(20), T(20), TA(20)
DIMENSION S(20), G(20), GS(20), PERM(20), P/RMC(20), W(20)

DATA CARDS 1 AND 2 CONTAIN ALPHA NUMERIC INFORMATION ABOUT THE SYSTEM AND MATERIALS FOR WHICH THE CALCULATION IS BEING MADE. FORMAT = 20A4/20A4

100 READ(5,105)END = 999)TITLE1, TITLE2
105 FORMAT(20A4/20A4)

READ IN OF SOLUBILITY PARAMETERS, C AND QS *****
DATA CARD 3 CONTAINS (IN ORDER) VALUES FOR C IN WPPM H/SQRT(TORR) AND QS IN CAL/MOLE. FORMAT = 2F12.0

110 READ(5,110)QS
FORMAT(2F12.0)

READ IN OF PERMEABILITY PARAMETERS, B AND QP *****
DATA CARD 4 CONTAINS (IN ORDER) VALUES FOR B IN G-CM/SQ CM-DAY-SQRT(TORR) AND QP IN CAL/MOLE. FORMAT = 2F12.0

115 READ(5,115)B, QP
FORMAT(2F12.0)

DATA CARD 5 CONTAINS (IN ORDER) THE VALUES OF THE LOWEST AND HIGHEST TEMPERATURES IN THE SYSTEM UNDER STUDY IN DEG C. FORMAT = 2F12.0

120 READ(5,120)TL, TH
FORMAT(2F12.0)

DATA CARD 6 CONTAINS (IN ORDER) VALUES FOR THE TOTAL AREA OF THE SYSTEM IN SQ CM, THE THICKNESS OF THE SYSTEM STRUCTURE IN CM, AND THE DENSITY OF THE STRUCTURAL MATERIAL IN G/CU CM. FORMAT = 3F12.0

125 READ(5,125)AREA, H, G
FORMAT(3F12.0)

DATA CARD 7 CONTAINS THE VALUE FOR THE NUMBER OF ISOTHERMAL ZONES INTO WHICH THE SYSTEM IS TO BE DIVIDED BY THE CODE. FORMAT = I2

130 READ(5,130)N
FORMAT(I2)

DATA CARD 8 CONTAINS (IN ORDER) THE VALUES FOR THE HYDROGEN ISOTOPE PRESSURE ON THE DOWNSTREAM SIDE OF THE SYSTEM AND THE HYDROGEN ISOTOPE PRESSURE ON THE UPSTREAM SIDE OF THE SYSTEM IN TORR. FORMAT = 2E12.0

135 READ(5,135)P1, P2
FORMAT(2E12.0)

V = AREA * H
WT = D * V

140 WRITE(6,140)TITLE1, TITLE2
FORAT(11H)///20A4/20A4

145 WRITE(6,145)N, TL, TH, C, QS, B, QP, P1, AREA, H, V, D, WT
FORMAT(11//10X, NUMBER OF TEMPERATURE ZONES, I2/

110X, 'LOWEST TEMPERATURE, T1, ' DEG C /
210X, 'HIGHEST TEMPERATURE, T2, ' DEG C /
310X, 'SOLUBILITY CONSTANTS, C = 'IPE11 3, ' WPPM H/SQRT(TORR) /
410X, 'QS = 'OPF8 0, ' CAL/MOLE /
510X, 'PERMEABILITY CONSTANTS, B = 'IPE11 3, ' G-CM/SQ CM-DAY-SQRT(TORR) /
610X, 'QP = 'OPF8 0, ' CAL/MOLE /
710X, 'UPSTREAM HYDROGEN PARTIAL PRESSURE, IPE10 1, ' TORR /
810X, 'DOWNSTREAM HYDROGEN PARTIAL PRESSURE, IPE10 1, ' TORR /
910X, 'SURFACE AREA OF THE INTERFACE, IPE10 1, ' SQ CM /
110X, 'THICKNESS OF THE INTERFACE, OP = 8 3, ' CM /
210X, 'VOLUME OF THE INTERFACE STRUCTURE, IPE10 2, ' CU CM /
310X, 'DENSITY OF THE INTERFACE MATERIAL, OPF8 2, ' G CU CM /
410X, 'WEIGHT OF THE INTERFACE STRUCTURE, IPE10 2, ' G /

DATA CARDS 9 THROUGH 8 + N (N = THE NUMBER OF ISOTHERMAL ZONES) CONTAIN THE VALUES FOR THE FRACTIONAL AREA OF EACH ZONE STARTING WITH THE LOWEST TEMPERATURE ZONE (ONE VALUE PER CARD). FORMAT = 1F12.0

150 DO 155 I = 1, N
READ(5,150)F(I)

FORMAT(1F12.0)

155 A(I) = AREA * F(I)
W(I) = A(I) * H * D

CONTINUE

160 FC(I) = F(I)
DO 157 I = 2, N

FC(I) = FC(I-1) + F(I)

165 CONTINUE
T(I) = TL
NA = N + 1
DO 180 I = 2, NA

T(I) = TL + FC(I-1) * (TH - TL)

170 TA(I-1) = (T(I-1) + T(I))/2
CONTINUE
WRITE(6,165)

FORMAT(11//10X, 'ITEM 1: TEMPERATURE ZONING OF THE INTERFACE' /

175 111X, 'AREA FRACTION CUMULATIVE AREA OF AVERAGE' /
21X, 'ZONE IN EACH ZONE AREA FRACTION EACH ZONE TEMPERATURE' /
31X, '(SQ CM) (DEG C) /

DO 175 I = 1, N
WRITE(6,170)I, F(I), FC(I), A(I), F(A(I))

170 FORMAT(17, F11.2, = 15.2, IPE12.0, OPF11.5)
175 CONTINUE

SOLUBILITIES, S(I), ARE DETERMINED FROM THE RELATIONSHIP

$S(I) = W(I) * C * 0.5 * (SQRT(P2) + SQRT(P1)) * EXP(QS / (R * (T(I) + 273.15)))$

WHEREA W(I) = AREA * H * D

PERMEABILITIES, PERM(I), ARE DETERMINED BY THE RELATIONSHIP

$PERM(I) = B * A(I) * (SQRT(P2) - SQRT(P1)) * EXP(QP / (1.987 * (TA(I) + 273.15))) / H$

DO 180 I = 1, N

G(I) = C * 0.5 * (SQRT(P2) + SQRT(P1)) * EXP(QS / (1.987 * (TA(I) + 273.15)))

S(I) = G(I) * W(I)

PERM(I) = B * A(I) * (SQRT(P2) - SQRT(P1)) * EXP(QP / (1.987 * (TA(I) + 273.15)))

I/H

CONTINUE

GC(I) = S(I)

PERMC(I) = PERM(I)

DO 182 I = 2, N

GC(I) = GC(I-1) + S(I)

PERMC(I) = PERMC(I-1) + PERM(I)

CONTINUE

WRITE(6,185)TITLE1, TITLE2

FORMAT(11H)///20A4/20A4

WRITE(6,190)P2

FORMAT(11//10X, 'ITEM 2: HYDROGEN CONTENT OF THE INTERFACE STRUCTURE

1 AT: IPE11 1, ' TORR /

210X, 'AVERAGE WEIGHT OF HYDROGEN HYDROGEN CUMULATIVE' /

32X, 'ZONE TEMPERATURE EACH ZONE SOLUBILITY IN EACH ZONE AMOUNT

4 OF H/10X, ' (DEG C) (G) (WPPM H) (MICROGRAMS) (MICRO

5GRAMS) /

DO 200 I = 1, N

WRITE(6,195)I, TA(I), W(I), G(I), S(I), GC(I)

FORMAT(5, F11.1, IPE13 2, E12 2, E13 2, E13 2)

CONTINUE

IF (P2-F1) 100, 100, 201

WRITE(6, 205)

FORMAT(11//10X, 'ITEM 3: HYDROGEN PERMEATION THROUGH THE INTERFACE

1 ' /10X, 'AVERAGE AREA OF PERMEATION RATE CUMULATIVE

2 ' / ' ZONE TEMPERATURE EACH ZONE IN EACH ZONE PERM. RATE' /

32X, '(SQ CM) (G/DAY) (G/DAY) /

DO 210 I = 1, N

WRITE(6, 210)I, TA(I), A(I), PERM(I), PERMC(I)

FORMAT(5, F11.1, IPE13 2, E13 2)

CONTINUE

GO TO 100

STOP

END

TITANIUM MAPPING STUDY OF THE SODIUM/STEAM HEAT EXCHANGER FOR THE LASL THETA PINCH DESIGN. MATERIAL = 304-SS/COPPER/304-SS; 304-SS

NUMBER OF TEMPERATURE ZONES 8

LOWEST TEMPERATURE 349.0 DEG C

HIGHEST TEMPERATURE 521.0 DEG C

SOLUBILITY CONSTANTS, KS = 1.100E-02 WPPM H/SQRT(TORR)

QS = -1330. CAL/MOLE

PERMEABILITY CONSTANTS, B = 1.100E-02 G-CM/SQ CM-DAY-SQRT(TORR)

QP = -18100. CAL/MOLE

UPSTREAM HYDROGEN PARTIAL PRESSURE 1.0E-10 TORR

DOWNSTREAM HYDROGEN PARTIAL PRESSURE 1.0E-10 TORR

SURFACE AREA OF THE INTERFACE 4.0E-06 SQ CM

THICKNESS OF THE INTERFACE 0.100 CM

VOLUME OF THE INTERFACE STRUCTURE 4.01E-07 CU CM

DENSITY OF THE INTERFACE MATERIAL 8.00 G/CU CM

WEIGHT OF THE INTERFACE STRUCTURE 3.21E-06 G

ITEM 1. TEMPERATURE ZONING OF THE INTERFACE

Zone	Area Fraction in Each Zone	Cumulative Area Fraction	Area of Each Zone (sq cm)	Average Temperature (deg c)
1	0.06	0.06	2.01E-07	353.300
2	0.10	0.16	4.01E-07	366.700
3	0.20	0.36	8.02E-07	387.700
4	0.20	0.56	8.02E-07	417.800
5	0.20	0.76	8.02E-07	432.300
6	0.15	0.91	6.02E-07	442.300
7	0.10	1.01	4.01E-07	503.800
8	0.05	1.07	2.01E-07	516.700

ITEM 2. HYDROGEN CONTENT OF THE INTERFACE STRUCTURE AT 1.0E-10 TORR

Zone	Average Temperature (deg c)	Weight of Each Zone (g)	Hydrogen Solubility (wppm h)	Hydrogen in Each Zone (micrograms)	Cumulative Amount of H (micrograms)
1	353.3	1.61E-07	9.53E-03	4.08E-01	4.08E-01
2	366.2	3.21E-07	2.80E-02	8.36E-01	1.24E-02
3	387.7	6.42E-07	7.73E-02	1.31E-01	1.36E-02
4	417.8	6.42E-07	2.80E-02	1.86E-02	4.42E-02
5	432.3	6.42E-07	3.08E-02	1.98E-02	6.40E-02
6	442.3	6.42E-07	2.32E-02	1.57E-02	7.97E-02
7	503.8	2.01E-07	3.38E-02	1.08E-02	9.05E-02
8	516.7	1.61E-07	3.43E-02	5.50E-03	9.60E-02

TRITIUM MAPPING STUDY OF THE SODIUM/STEAM HEAT EXCHANGER FOR THE LAST THETA PINCH DESIGN MATERIAL = 304-SS/COPPER/304-SS; COPPER

NUMBER OF TEMPERATURE ZONES 8
 LOWEST TEMPERATURE 349.0 DEG C
 HIGHEST TEMPERATURE 521.0 DEG C
 SOLUBILITY CONSTANTS KS = 4.080E-01 WPPM H/SQRT(TORR)
 QS = -13911 CAL/MOLE
 PERMEABILITY CONSTANTS B = 1.490E-02 G-CM/SQ CM-DAY-SQRT(TORR)
 QP = 18562 CAL/MOLE
 UPSTREAM HYDROGEN PARTIAL PRESSURE 1.0E-10 TORR
 DOWNSTREAM HYDROGEN PARTIAL PRESSURE 8.0 TORR
 SURFACE AREA OF THE INTERFACE 4.0E 08 SQ CM
 THICKNESS OF THE INTERFACE 0.250 CM
 VOLUME OF THE INTERFACE STRUCTURE 1.00E 08 CU CM
 DENSITY OF THE INTERFACE MATERIAL 8.92 G/CM CM
 WEIGHT OF THE INTERFACE STRUCTURE 8.95E 08 G

ITEM 1. TEMPERATURE ZONING OF THE INTERFACE

Zone	Area Fraction in Each Zone	Cumulative Area Fraction	Area of Each Zone (sq cm)	Average Temperature (deg c)
1	0.05	0.05	2.01E 07	353.300
2	0.10	0.15	4.01E 07	366.200
3	0.15	0.30	6.02E 07	387.700
4	0.20	0.50	8.03E 07	417.800
5	0.20	0.70	8.03E 07	452.200
6	0.15	0.85	6.02E 07	482.300
7	0.10	0.95	4.01E 07	503.800
8	0.05	1.00	2.01E 07	516.700

ITEM 2. HYDROGEN CONTENT OF THE INTERFACE STRUCTURE AT 1.0E-10 TORR

Zone	Average Temperature (deg c)	Weight of Each Zone (g)	Hydrogen Solubility (wppm h)	Hydrogen in Each Zone (micrograms)	Cumulative Amount of H (micrograms)
1	353.3	4.47E 07	2.86E-09	1.28E-01	1.28E-01
2	366.2	8.95E 07	3.58E-09	3.20E-01	4.48E-01
3	387.7	1.34E 08	5.11E-09	6.86E-01	1.13E 00
4	417.8	1.79E 08	8.11E-09	1.45E 00	2.59E 00
5	452.2	1.79E 08	1.31E-08	2.35E 00	4.93E 00
6	482.3	1.34E 08	1.93E-08	2.59E 00	7.52E 00
7	503.8	8.95E 07	2.49E-08	2.23E 00	9.75E 00
8	516.7	4.47E 07	2.85E-08	1.28E 00	1.10E 01

ITEM 3. HYDROGEN PERMEATION THROUGH THE INTERFACE

Zone	Average Temperature (deg c)	Area of Each Zone (sq cm)	Permeation Rate in Each Zone (g/day)	Cumulative Perm. Rate (g/day)
1	353.3	2.01E 07	3.99E-06	3.99E-06
2	366.2	4.01E 07	1.48E-05	1.88E-05
3	387.7	6.02E 07	2.60E-05	4.48E-05
4	417.8	8.03E 07	1.22E-04	2.27E-04
5	452.2	8.03E 07	1.53E-04	3.80E-04
6	482.3	6.02E 07	1.44E-04	5.24E-04
7	503.8	4.01E 07	8.73E-05	6.11E-04
8	516.7	2.01E 07		

TRITIUM MAPPING STUDY OF THE SODIUM/STEAM HEAT EXCHANGER FOR THE LAST THETA PINCH DESIGN MATERIAL = 304-SS/COPPER/304-SS; COPPER

NUMBER OF TEMPERATURE ZONES 8
 LOWEST TEMPERATURE 349.0 DEG C
 HIGHEST TEMPERATURE 521.0 DEG C
 SOLUBILITY CONSTANTS KS = 1.36E-01 WPPM H/SQRT(TORR)
 QS = -13911 CAL/MOLE
 PERMEABILITY CONSTANTS B = 8.60E-03 G-CM/SQ CM-DAY-SQRT(TORR)
 QP = 18562 CAL/MOLE
 UPSTREAM HYDROGEN PARTIAL PRESSURE 1.0E-05 TORR
 DOWNSTREAM HYDROGEN PARTIAL PRESSURE 1.0E-05 TORR
 SURFACE AREA OF THE INTERFACE 4.0E-09 SQ CM
 THICKNESS OF THE INTERFACE 0.250 CM
 VOLUME OF THE INTERFACE STRUCTURE 1.00E-08 CU CM
 DENSITY OF THE INTERFACE MATERIAL 8.92 G/CM CM
 WEIGHT OF THE INTERFACE STRUCTURE 8.95E-08 G

ITEM 1. TEMPERATURE ZONING OF THE INTERFACE

Zone	Area Fraction in Each Zone	Cumulative Area Fraction	Area of Each Zone (sq cm)	Average Temperature (deg c)
1	0.05	0.05	2.01E 07	353.300
2	0.10	0.15	4.01E 07	366.200
3	0.15	0.30	6.02E 07	387.700
4	0.20	0.50	8.03E 07	417.800
5	0.20	0.70	8.03E 07	452.200
6	0.15	0.85	6.02E 07	482.300
7	0.10	0.95	4.01E 07	503.800
8	0.05	1.00	2.01E 07	516.700

ITEM 2. HYDROGEN CONTENT OF THE INTERFACE STRUCTURE AT 1.0E-05 TORR

Zone	Average Temperature (deg c)	Weight of Each Zone (g)	Hydrogen Solubility (wppm h)	Hydrogen in Each Zone (micrograms)	Cumulative Amount of H (micrograms)
1	353.3	4.47E 07	3.01E-07	1.35E-01	1.35E-01
2	366.2	8.95E 07	3.77E-07	3.38E-01	4.73E-01
3	387.7	1.34E 08	5.38E-07	7.24E-01	1.20E 00
4	417.8	1.79E 08	8.53E-07	1.53E 00	2.73E 00
5	452.2	1.79E 08	1.38E-06	2.47E 00	5.20E 00
6	482.3	1.34E 08	2.03E-06	2.73E 00	7.93E 00
7	503.8	8.95E 07	2.62E-06	2.35E 00	1.03E 01
8	516.7	4.47E 07	3.04E-06	1.36E 00	1.16E 01

ITEM 3. HYDROGEN PERMEATION THROUGH THE INTERFACE

Zone	Average Temperature (deg c)	Area of Each Zone (sq cm)	Permeation Rate in Each Zone (g/day)	Cumulative Perm. Rate (g/day)
1	353.3	2.01E 07	7.29E-04	7.29E-04
2	366.2	4.01E 07	1.87E-03	2.70E-03
3	387.7	6.02E 07	4.75E-03	7.45E-03
4	417.8	8.03E 07	1.17E-02	1.82E-02
5	452.2	8.03E 07	2.23E-02	4.15E-02
6	482.3	6.02E 07	2.79E-02	6.94E-02
7	503.8	4.01E 07	2.62E-02	9.56E-02
8	516.7	2.01E 07	1.58E-02	1.12E-01

TRITIUM MAPPING STUDY OF THE SODIUM/STEAM HEAT EXCHANGER FOR THE LAST THETA PINCH DESIGN MATERIAL = 304-SS/COPPER/304-SS; 304-SS

NUMBER OF TEMPERATURE ZONES 8
 LOWEST TEMPERATURE 360.0 DEG C
 HIGHEST TEMPERATURE 484.0 DEG C
 SOLUBILITY CONSTANTS KS = 4.130E-01 WPPM H/SQRT(TORR)
 QS = -1679 CAL/MOLE
 PERMEABILITY CONSTANTS B = 6.500E-03 G-CM/SQ CM-DAY-SQRT(TORR)
 QP = 16100 CAL/MOLE
 UPSTREAM HYDROGEN PARTIAL PRESSURE 1.0E-05 TORR
 DOWNSTREAM HYDROGEN PARTIAL PRESSURE 1.0E-05 TORR
 SURFACE AREA OF THE INTERFACE 3.7E 08 SQ CM
 THICKNESS OF THE INTERFACE 0.100 CM
 VOLUME OF THE INTERFACE STRUCTURE 3.72E 07 CU CM
 DENSITY OF THE INTERFACE MATERIAL 8.00 G/CM CM
 WEIGHT OF THE INTERFACE STRUCTURE 2.97E 08 G

ITEM 1. TEMPERATURE ZONING OF THE INTERFACE

Zone	Area Fraction in Each Zone	Cumulative Area Fraction	Area of Each Zone (sq cm)	Average Temperature (deg c)
1	0.05	0.05	1.86E 07	365.800
2	0.10	0.15	3.72E 07	382.400
3	0.15	0.30	5.57E 07	310.400
4	0.20	0.50	7.43E 07	349.800
5	0.20	0.70	7.43E 07	394.400
6	0.15	0.85	5.57E 07	433.600
7	0.10	0.95	3.72E 07	461.600
8	0.05	1.00	1.86E 07	478.400

ITEM 2. HYDROGEN CONTENT OF THE INTERFACE STRUCTURE AT 1.0E-05 TORR

Zone	Average Temperature (deg c)	Weight of Each Zone (g)	Hydrogen Solubility (wppm h)	Hydrogen in Each Zone (micrograms)	Cumulative Amount of H (micrograms)
1	365.8	1.49E 07	2.71E-04	4.04E 03	4.04E 03
2	382.4	2.97E 07	3.35E-04	8.46E 03	1.25E 04
3	310.4	4.45E 07	3.05E-04	1.37E 04	2.62E 04
4	349.8	5.55E 07	3.35E-04	1.95E 04	4.61E 04
5	394.4	5.55E 07	3.67E-04	2.18E 04	6.79E 04
6	433.6	4.45E 07	3.94E-04	1.76E 04	8.55E 04
7	461.6	2.97E 07	4.13E-04	1.23E 04	9.78E 04
8	478.4	1.49E 07	4.23E-04	6.29E 03	1.04E 05

TRITIUM MAPPING STUDY OF THE LITHIUM/SODIUM HEAT EXCHANGER FOR THE LAST THETA PINCH DESIGN MATERIAL = NIOBIUM

NUMBER OF TEMPERATURE ZONES 8
 LOWEST TEMPERATURE 410.0 DEG C
 HIGHEST TEMPERATURE 541.0 DEG C
 SOLUBILITY CONSTANTS KS = 6.000E-01 WPPM H/SQRT(TORR)
 QS = 8100 CAL/MOLE
 PERMEABILITY CONSTANTS B = 2.700E-02 G-CM/SQ CM-DAY-SQRT(TORR)
 QP = 3200 CAL/MOLE
 UPSTREAM HYDROGEN PARTIAL PRESSURE 1.0E-10 TORR
 DOWNSTREAM HYDROGEN PARTIAL PRESSURE 1.0E-10 TORR
 SURFACE AREA OF THE INTERFACE 3.7E 08 SQ CM
 THICKNESS OF THE INTERFACE 0.090 CM
 VOLUME OF THE INTERFACE STRUCTURE 3.34E 07 CU CM
 DENSITY OF THE INTERFACE MATERIAL 8.40 G/CM CM
 WEIGHT OF THE INTERFACE STRUCTURE 2.81E 08 G

ITEM 1. TEMPERATURE ZONING OF THE INTERFACE

Zone	Area Fraction in Each Zone	Cumulative Area Fraction	Area of Each Zone (sq cm)	Average Temperature (deg c)
1	0.05	0.05	1.96E 07	413.275
2	0.10	0.15	3.72E 07	423.100
3	0.15	0.30	5.57E 07	439.475
4	0.20	0.50	7.43E 07	462.400
5	0.20	0.70	7.43E 07	488.600
6	0.10	0.85	3.72E 07	527.900
8	0.05	1.00	1.96E 07	537.725

ITEM 2. HYDROGEN CONTENT OF THE INTERFACE STRUCTURE AT 1.0E-10 TORR

Zone	Average Temperature (deg c)	Weight of Each Zone (g)	Hydrogen Solubility (wppm h)	Hydrogen in Each Zone (micrograms)	Cumulative Amount of H (micrograms)
1	413.3	1.40E 07	2.90E 03	4.07E 04	4.07E 04
2	423.1	2.81E 07	2.86E 03	7.47E 04	1.15E 05
3	439.5	4.21E 07	2.31E 03	9.74E 04	2.13E 05
4	462.4	5.62E 07	1.92E 03	1.08E 05	3.21E 05
5	488.6	5.62E 07	1.57E 03	8.84E 04	4.08E 05
6	511.5	4.21E 07	1.34E 03	5.64E 04	4.63E 05
7	527.9	2.81E 07	1.20E 03	3.36E 04	4.96E 05
8	537.7	1.40E 07	1.12E 03	1.58E 04	5.13E 05

ITEM 1. TEMPERATURE ZONING OF THE INTERFACE

Zone	Area Fraction in Each Zone	Cumulative Area Fraction	Area of Each Zone (sq cm)	Average Temperature (deg C)
1	0.05	0.05	1.96E 07	353.300
2	0.10	0.15	3.72E 07	368.200
3	0.15	0.30	5.57E 07	387.700
4	0.20	0.50	7.43E 07	417.800
5	0.20	0.70	7.43E 07	452.200
6	0.15	0.85	5.57E 07	482.300
7	0.10	0.95	3.72E 07	503.800
8	0.05	1.00	1.96E 07	518.700

ITEM 2. HYDROGEN CONTENT OF THE INTERFACE STRUCTURE AT 1.0E-10 TORR

Zone	Average Temperature (deg c)	Weight of Each Zone (g)	Hydrogen Solubility (wppm h)	Hydrogen in Each Zone (micrograms)	Cumulative Amount of H (micrograms)
1	353.3	1.40E 07	2.53E 03	3.76E 04	3.76E 04
2	368.2	2.81E 07	2.60E 03	7.74E 04	1.15E 05
3	387.7	4.21E 07	2.73E 03	1.22E 05	2.37E 05
4	417.8	5.62E 07	2.90E 03	1.72E 05	4.09E 05
5	452.2	5.62E 07	3.09E 03	1.84E 05	5.93E 05
6	482.3	4.21E 07	3.25E 03	1.45E 05	7.38E 05
7	503.8	2.81E 07	3.36E 03	9.88E 04	8.38E 05
8	518.7	1.40E 07	3.43E 03	5.08E 04	8.93E 05

TRITIUM MAPPING STUDY OF THE LITHIUM/SODIUM HEAT EXCHANGER FOR THE LASL THETA PINCH DESIGN: MATERIAL = 304-SS

NUMBER OF TEMPERATURE ZONES 8
LOWEST TEMPERATURE 349.0 DEG C
HIGHEST TEMPERATURE 521.0 DEG C
SOLUBILITY CONSTANTS $K_S = 1.100E-02$ CO WPPM H/SQRT(TORR)
 $Q_S = -1830$ CAL/MOLE
PERMEABILITY CONSTANTS $B = 1.180E-02$ G-CM/SQ CM-DAY-SQRT(TORR)
 $Q_P = 18100$ CAL/MOLE
UPSTREAM HYDROGEN PARTIAL PRESSURE 1.0E-10 TORR
DOWNSTREAM HYDROGEN PARTIAL PRESSURE 1.0E-10 TORR
SURFACE AREA OF THE INTERFACE 3.7E 08 SQ CM
THICKNESS OF THE INTERFACE 0.10 CM
VOLUME OF THE INTERFACE STRUCTURE 3.72E 07 CU CM
DENSITY OF THE INTERFACE MATERIAL 8.00 G/CM CM
WEIGHT OF THE INTERFACE STRUCTURE 2.97E 09 G

APPENDIX E

RADIOACTIVITY AND AFTERHEAT CALCULATIONS

E.1 CALCULATIONAL MODEL

The evolution of a conceptual theta-pinch reactor design^{4, 7-51} has included successive neutronic analyses.^{52, 53} Except for the radial walls (1 mm niobium and 0.3 mm Al_2O_3) which separate the 100 segments around the minor circumference, the blanket can be represented by a series of concentric cylinders. Niobium-alumina walls were all taken into account either explicitly or by homogenization in the neutronic model, and the total blanket inventory of niobium structure, therefore, is realistically represented. Accounting for metal structure within regions such as the graphite moderator is especially important because a large proportion of thermal captures occur in moderating regions. No attempt was made to alter the structural design for V-20%Ti, rather, the alloy was simply substituted for niobium in activation calculations.

All transport calculations were performed with the DTF-IV discrete ordinates code,⁵⁴ using cross sections from the standard LASL CTR 100-group library.⁵⁵ Activation cross sections were obtained from various sources, including ENDF, the "barn book",⁵⁶ and nuclear model calculations.⁵⁵ In particular, multigroup cross sections for excitation of $^{93\text{m}}\text{Nb}$, the 12-year metastable state of ^{93}Nb , were derived from the work of Hegedus.⁵⁷ Because of its dominant contribution, production of ^{95}Nb by the $^{94}\text{Nb}(n, \gamma)^{95}\text{Nb}$ reaction introduces the greatest uncertainty in calculations of both radioactivity and afterheat from niobium. The cross section for this reaction has not been thoroughly measured; both the thermal cross section and the resonance integral have been found to be approximately 15 times the respective values for the $^{93}\text{Nb}(n, \gamma)^{94}\text{Nb}$ reaction.⁵⁸ Therefore, the assumption was made⁵⁵ that the $^{94}\text{Nb}(n, \gamma)^{95}\text{Nb}$ cross section is everywhere 15 times that for $^{93}\text{Nb}(n, \gamma)^{94}\text{Nb}$. This assumption should be conservative (i.e., overpredict ^{95}Nb and $^{95\text{m}}\text{Nb}$ activity in most cases) unless the ^{94}Nb resonances are predominantly at higher energies* than those in

^{93}Nb . Another effect tending to make the $^{95, 95\text{m}}\text{Nb}$ production calculations conservative is resonance self-shielding. Accounting for resonance self-shielding in ^{93}Nb has been estimated⁵³ to reduce the ^{94}Nb production by $\sim 20\%$, while this effect in ^{94}Nb resonance capture is fluence-dependent and indeterminate. Resonance self-shielding was ignored with the intention of providing conservative results for $^{95, 95\text{m}}\text{Nb}$. However, long-term ^{94}Nb activity may still be underpredicted because of overestimating burnout.

The branching ratio to $^{95\text{m}}\text{Nb}$ in the $^{94}\text{Nb}(n, \gamma)$ reaction was assumed⁵⁵ to be 0.2, and 0.5 was assumed⁵⁵ for the $^{93}\text{Nb}(n, \gamma)$ reaction to $^{94\text{m}}\text{Nb}$ and ^{94}Nb . As a conservative assumption all $^{93}\text{Nb}(n, \alpha)$ reactions were assumed to branch to $^{90\text{m}}\text{Y}$. Activation of the metastable state of ^{92}Nb has the cross section originally given (erroneously) in the ENDF-II data file for the $^{93}\text{Nb}(n, 2n)$ reaction. For $I_w = 2.0$ and 6.7 MW/m^2 the production of ^{92}Zr by $^{92\text{m}}\text{Nb}$ decay exceeded the naturally occurring ^{92}Zr in Nb-1%Zr by factors of 8 and 26 at $T = 5 \text{ y}$, indicating that the ^{89}Sr activity per unit operating power will be a strong function of both wall loading and operating time, which is similar to $^{95, 95\text{m}}\text{Nb}$ behavior. Likewise, ^{90}Sr , which results mostly from successive reactions via ^{93}Zr , has the same strong dependence. The contribution from $^{94}\text{Zr}(n, n' \alpha)$ reactions is minor. It is quickly apparent that the accuracy of radioactivity and afterheat calculations is limited almost entirely by the cross-section data uncertainties, especially for the $^{94}\text{Nb}(n, \gamma)$ reaction; errors due to angular quantization and other numerical approximations are relatively minor.

E.2 CALCULATIONAL RESULTS

E.2.1 Radioactivity and Biological Hazard Potential (BHP)

The radioactivity induced by neutron transmutations depends on the blanket structural material, the wall loading I_w , the operating time T ,

*Fragmentary data indicate the opposite behavior; i.e., low-lying resonances at $\sim 10\text{-}20 \text{ eV}$.

and the time after shutdown t_s . Although this radioactivity is conventionally expressed as either $a(Ci/Wt\ yr)$, or $A(Ci/Wt)$, these quantities have little technical significance, in that the induced radioactivity presents either a biological hazard (in the short- or long-run) or a cooling problem. The latter concern is best reflected in terms of the fraction, P/P_o , of the operating power represented by decay heat. The biological hazard is most conveniently measured in terms of a biological hazard potential, BHP, defined as the ratio of $A(Ci/Wt)$ to the maximum permissible concentration, MPC (Ci/km^3).^{*} The principle usefulness of the BHP lies in its comparative function, and a physical significance should not be given to the absolute value of the BHP, since important properties like volatility, chemical state, and source distribution are not specified. The radioactivity and afterheat calculations are based on pure alloys, and

^{*}MPC's are given for air and water according²⁴ to whether (1) the isotope is in a soluble or insoluble form and (2) the release is into a controlled (40 h/wk exposure) or an uncontrolled (168 h/wk exposure) area. These MPC's pertain to individual doses and must be reduced by a factor of 3 when applied to a suitable sample of the exposed population. To assure an unambiguous comparison, all MPC's used herein apply to individual exposures in uncontrolled areas, and the smaller between soluble and insoluble values is used. MPC's quoted are for air concentrations. MPC values not found in Ref. 24 were calculated by J. W. Healy (Los Alamos Scientific Laboratory, personal communication, 1974).

the effects of impurities at the levels found in commercial alloys may be nontrivial.

As indicated on Fig. 4.2-2, increased wall loading generally results in higher short-term activation (Ci/Wt) of the niobium structure as a result of second-order reactions which lead to $^{95,95m}Nb$. The long-term activation for a given value of T , however, is seen to decrease with increasing I_w , as a result of the burnout of the long-lived ^{94}Nb and its parent ^{93}Nb .

Because the copper compression coils are in close proximity to the blanket in the RTPR design, the neutron flux level and subsequent activation will be significant. Table E-1 summarizes the RTPR radioactivity and biological hazard potential for the copper compression coil at shutdown for the design wall loading ($2.0\ MW/m^2$) as a function of operating time. The isotopes ^{60}Co ($t_{1/2} = 5.272\ y$) and ^{63}Ni ($t_{1/2} = 100y$) represent the major long-term radwaste concern (0.5% of the total copper activity at shutdown for $T = 10\ y$). The ^{63}Ni activity, although long-lived, emits only a soft ($E_{max} = 0.07\ MeV$) beta. The total copper activity is 20% of the Nb-1%Zr activity ($I_w = 2.0\ MW/m^2$, $T = 5\ y$) at shutdown, although the short-lived nature of the dominant copper activity presents mainly an afterheat problem rather than a radiological hazard. A comparison of the BHPs for $I_w = 2.0\ MW/m^2$ in Tables E-1 and E-3 does show, however, that the copper coils have slightly higher values than a V-20%Ti blanket. In the case of a Nb-1%Zr blanket, the copper coils contribute only about 5% of the biological hazard potential.

TABLE E-1.

RTPR RADIOACTIVITY AT SHUTDOWN IN
COPPER VS OPERATING TIME
($I_w = 2.0\ MW/m^2$, the reference case)

Isotope	$t_{1/2}$	Activity (Ci/Wt)			BHP (km^3/Wt)
		$T = 5\ y$	$T = 10\ y$	$T = 20\ y$	$T = 20\ y$
^{60}Co	5.27 y	1.75×10^{-3}	2.66×10^{-3}	3.38×10^{-3}	1.13×10^{-2}
^{62}Co	13.9 m	1.63×10^{-3}	1.63×10^{-3}	1.63×10^{-3}	8.15×10^{-7}
^{63}Ni	100. y	7.48×10^{-4}	1.47×10^{-3}	2.84×10^{-3}	1.42×10^{-3}
^{65}Ni	2.520 h	9.54×10^{-4}	9.54×10^{-4}	9.54×10^{-4}	4.77×10^{-5}
^{62}Cu	9.78 m	0.0262	0.0262	0.0262	8.73×10^{-6}
^{64}Cu	12.74 h	0.649	0.649	0.648	1.62×10^{-2}
^{66}Cu	5.10 m	0.108	0.108	0.108	1.08×10^{-5}
TOTALS	- Ci/Wt	0.7883	0.7899	0.7910	BHP: 0.0290
	- $Ci/Wt\ yr$	0.1577	0.07899	0.03955	
	- $P/P_o(\%)$	0.245	0.246	0.247	

The dependence of the blanket radioactivity (and afterheat) on T and I_w was indicated by the results presented in Figs. 4.2-2 and 4.2-3. Tables E-2 and E-3 illustrate the effect of wall loading, I_w , and operating time, T, on the Nb-1%Zr and V-20%Ti radioactivity biological hazard potential, and afterheat. The strong dependence on I_w of the $^{95,95m}\text{Nb}$ activity is immediately evident from Table E-2 with ^{95}Nb dominating the activity at the higher values of I_w . The less dramatic effect of $^{93,94}\text{Nb}$ burnout is also observed. Figure E-1 summarizes the dependence of $A(\text{Ci/Wt})$, $\alpha(\text{Ci/Wt yr})$, and P/P_0 on the operating time, T, for various wall loadings, I_w , in the Nb-1%Zr RTPR. As discussed previously, calculations of the $^{95,95m}\text{Nb}$ activities are subject to considerable error caused by uncertainty in the ^{94}Nb radiative capture cross section. The high activities at large values of I_w may in fact be illusory if ^{94}Nb capture resonances are predominately low lying. Table E-2 clearly illustrates the strong effect of ^{95}Nb activity on the BHP. Even though the ^{94m}Nb activity (Ci/Wt) dominates for $I_w < 2.0 \text{ MW/m}^2$, its BHP is negligible relative to that of ^{95}Nb . Thus, the total BHP increases much more rapidly with wall loading than does $A(\text{Ci/Wt})$.

Studies of the variation of radioactivity from V-20%Ti with wall loading were also performed, with less pronounced differences (Table E-3). Variations in total Ci/Wt, BHP, and P/P_0 are not significant (at most 3%). Therefore, the second-order reactions on ^{45}Sc , ^{50}Ti , ^{50}V and ^{52}Cr (produced by first-order

reactions on stable nuclei of titanium or vanadium) are shown to be of minor importance, unlike the possible case with ^{94}Nb . Consecutive (n,2n) reaction on ^{46}Ti to produce ^{44}Ti (47 y) were assumed to be a negligible contributor to the radioactivity. The first six radioisotopes in Table E-3 are those considered by Steiner⁵⁹ for a pure vanadium structure; the remaining six are principally products of titanium activation. Approximately 75% of the ^{47}Sc production in the RTPR, however, comes from the $^{49}\text{Ti}(n,n'p)$ reactions, not from $^{50}\text{V}(n,\alpha)$ or $^{51}\text{V}(n,n'\alpha)$ reactions. As can be seen in Table E-3, the ^{45}Ca and ^{46}Sc will be the dominant contributors to BHP for several (6-10) years after shutdown, because of their high initial BHP at least two orders of magnitude lower than ^{45}Ca or ^{46}Sc for any wall loading considered.

On the basis of the foregoing results, the dependence of $\alpha(\text{Ci/Wt yr})$, $A(\text{Ci/Wt})$, and BHP (km^3/Wt) on I_w , T, and t_s for Nb-1%Zr is complicated by second-order production reactions, burnout of stable ^{93}Nb and radioactive isotopes, and the natural variations in half-life and MPCs. As evidenced from the data presented in Fig. E-1, different costs are accrued (i.e., total curies of activity) for the same apparent benefit (i.e., Wt-yr of energy received) depending on the individual values of wall loading (directly relatable to power) and operating time. For the case where the activity at shutdown is comprised primarily of short-lived isotopes (i.e., ^{94m}Nb and ^{92m}Nb), this situation is best

TABLE E-2
RTPR AFTERHEAT AND RADIOACTIVITY AT SHUTDOWN IN
Nb-1%Zr VS WALL LOADING (T = 5 yr)

Isotope	$I_w = 0.2$		$I_w = 0.5$		Reference Design $I_w = 2.0$		$I_w = 5.7$	
	Ci/Wt	BHP	Ci/Wt	BHP	Ci/Wt	BHP	Ci/Wt	BHP
$^{95}\text{Nb}(2 \times 10^7 \text{ y})$	7.42×10^{-8}	2.12×10^{-10}	7.41×10^{-8}	2.12×10^{-10}	7.36×10^{-8}	2.1×10^{-10}	7.19×10^{-8}	2.05×10^{-10}
$^{92m}\text{Nb}(10.13 \text{ d})$	0.288	7.78×10^{-4}	0.288	7.78×10^{-4}	0.287	7.76×10^{-4}	0.282	7.62×10^{-4}
$^{93m}\text{Nb}(13 \text{ y})$	0.0609	0.0152	0.0608	0.0152	0.0606	0.0152	0.0601	0.0150
$^{94}\text{Nb}(2.0 \times 10^4 \text{ y})$	5.98×10^{-4}	2.99×10^{-4}	5.68×10^{-4}	2.84×10^{-4}	4.45×10^{-4}	2.22×10^{-4}	2.39×10^{-4}	1.20×10^{-4}
$^{94m}\text{Nb}(6.26 \text{ m})$	1.78	8.90×10^{-6}	1.78	8.90×10^{-6}	1.75	8.79×10^{-6}	1.67	8.35×10^{-6}
$^{95}\text{Nb}(35.1 \text{ d})$	0.213	0.0710	0.503	0.168	1.55	0.517	2.61	0.870
$^{95m}\text{Nb}(3.61 \text{ d})$	0.0425	1.52×10^{-4}	0.101	3.61×10^{-4}	0.311	1.11×10^{-3}	0.521	1.86×10^{-3}
$^{88}\text{Zr}(50.5 \text{ d})$	...	...	...	...	1.04×10^{-4}	3.47×10^{-4}	3.25×10^{-4}	1.06×10^{-3}
$^{90}\text{Zr}(29 \text{ y})$	...	...	...	...	4.67×10^{-7}	1.56×10^{-5}	1.42×10^{-6}	4.73×10^{-5}
$^{87}\text{Y}(64 \text{ h})$	...	...	...	...	6.78×10^{-3}	2.28×10^{-3}	6.78×10^{-3}	2.26×10^{-3}
$^{90m}\text{Y}(3.19 \text{ h})$	...	...	...	...	$< 6.78 \times 10^{-3}$	$< 9.68 \times 10^{-6}$	$< 6.78 \times 10^{-3}$	$< 9.68 \times 10^{-6}$
$^{92}\text{Zr}(9.5 \times 10^5 \text{ y})$	...	...	...	...	6.97×10^{-8}	2.24×10^{-8}	8.96×10^{-8}	2.24×10^{-8}
TOTALS	2.38	0.0874	2.73	0.185	3.96	0.534	5.14	0.889
P/P ₀ (%)	0.312		0.459		0.990		1.52	

TABLE E-3
RTPR AFTERHEAT AND RADIOACTIVITY AT SHUTDOWN IN V-20%Ti VS
WALL LOADING ($T = 5 \text{ yr}$)

Isotope	$t_{1/2}$	Reference Design					
		$I_w = 0.5 \text{ MW/m}^2$		$I_w = 2.0 \text{ MW/m}^2$		$I_w = 6.7 \text{ MW/m}^2$	
		Ci/Wt	BHP	Ci/Wt	BHP	Ci/Wt	BHP
^{47}Sc	3.41 d	3.61×10^{-3}	1.80×10^{-4}	3.83×10^{-3}	1.92×10^{-4}	4.45×10^{-3}	2.22×10^{-4}
^{48}Sc	43.7 h	0.0228	4.56×10^{-3}	0.0226	4.52×10^{-3}	0.0226	4.52×10^{-3}
^{51}Ti	5.76 m	0.0467	5.49×10^{-5}	0.0467	5.49×10^{-5}	0.0466	5.48×10^{-5}
^{49}V	331. d	1.15×10^{-3}	4.60×10^{-6}	4.58×10^{-3}	1.83×10^{-5}	0.0147	5.88×10^{-5}
^{52}V	3.755 m	0.370	1.06×10^{-3}	0.369	1.05×10^{-3}	0.366	1.05×10^{-3}
^{51}Cr	27.71 d	9.03×10^{-5}	1.13×10^{-6}	3.62×10^{-4}	4.52×10^{-6}	1.20×10^{-3}	1.50×10^{-5}
SUBTOTAL		0.444	5.86×10^{-3}	0.447	5.84×10^{-3}	0.456	5.91×10^{-3}
^{45}Ca	163. d	4.66×10^{-3}	4.66×10^{-3}	4.66×10^{-3}	4.66×10^{-3}	4.65×10^{-3}	4.65×10^{-3}
^{47}Ca	4.54 d	6.14×10^{-5}	1.02×10^{-5}	6.56×10^{-5}	1.09×10^{-5}	7.88×10^{-5}	1.31×10^{-5}
^{46}Sc	83.8 d	5.07×10^{-3}	6.33×10^{-3}	5.10×10^{-3}	6.38×10^{-3}	5.15×10^{-3}	6.44×10^{-3}
^{49}Sc	57.3 m	4.44×10^{-4}	3.42×10^{-7}	4.44×10^{-4}	3.42×10^{-7}	4.43×10^{-4}	3.41×10^{-7}
^{50}Sc	1.71 m	1.54×10^{-4}	1.40×10^{-6}	1.64×10^{-4}	1.49×10^{-6}	1.96×10^{-4}	1.78×10^{-6}
^{45}Ti	3.078 h	2.08×10^{-4}	6.12×10^{-7}	2.08×10^{-4}	6.12×10^{-7}	2.08×10^{-4}	6.12×10^{-7}
TOTAL		0.455	0.0169	0.458	0.0169	0.466	0.0170
P/P₀ (%)		0.554		0.553		0.549

Fig. E-1.

Summary of the dependence of $A(\text{Ci/Wt})$, $\alpha(\text{Ci/Wt yr})$, and P/P_0 on operating time, T , at various wall loadings, I_w , for the Nb-1%Zr RTPR and $t_s = 0.0$.

described by plotting the "cost-benefit ratio" $A(\text{Ci/Wt})$ vs the total energy delivered GWt yr . Figure E-2 illustrates this correlation at $t_s = 0$, and this correlation is independent of wall loading in the range 0.2 to 6.7 MW/m^2 (360 to 12,000 MWt for the RTPR). The "activity cost" (Ci) per unit of thermal power (Wt) derived from a given blanket structure increases with increasing energy (GWt yr) as second-order reactions build in activity. As the derived energy increases, the benefit of activity burnout is realized and the curve shows a maximum. Specifically, this maximum is a result of the burnout of ^{93}Nb and ^{94}Nb and occurs at unrealistically large first-wall neutron fluences (corresponding to transmutation of $\sim 20\%$ of the original niobium), but not unrealistic for recycled niobium.

Figure E-2 corresponds to a "cost-benefit" ratio which is most applicable at shutdown ($t_s = 0$). Since the real cost of the induced radioactivity must often be attributed to long-term storage requirements, the dependence of the "cost-benefit" ratio, $\alpha(\text{Ci/Wt yr})$, on total delivered energy (GWt yr) at long shutdown times is of interest; this behavior is illustrated in Fig. E-3. The quantity $\alpha(\text{Ci/Wt yr})$ represents a "cost-benefit" ratio which is more useful for large values of

Fig. E-2.

Dependence of $A(\text{Ci}/\text{Wt})$ on $P_{TH}T$ (GWt yr) for $t_s = 0.0$.

Fig. E-3.

Dependence of $a(\text{Ci}/\text{Wt yr})$ on $P_{TH}T$ (GWt yr) for various values of t_s .

t_s and eventually becomes independent of wall loading for long times after shutdown. The isotope ^{94}Nb represents the major long-term activity, and the decrease of $a(\text{Ci}/\text{Wt yr})$ with delivered energy (GWt yr) for a given large value of t_s , reflects the burnout of this activity.

E.2.2 Nuclear Afterheat

Nuclear afterheat represents a concern for both fission and fusion power reactors in event of a loss-of-cooling accident. In assessing the nuclear afterheat problem, primary consideration must be given to the fraction of the operating thermal power which is represented by nuclear afterheat, P/P_o , as well as the specific power (MW/m^3) generated within the blanket by the afterheat. The dependence of P/P_o on wall loading, operating time, and material have been summarized for the RTPR in Tables E-2 and E-3. The lifetime of the copper coils is expected to be 2-4 times that of the blanket, but their afterheat power saturates in a few days and changes $<1\%$ for up to 20 years operation.

For the case of a loss-of-cooling accident interest focuses on the magnitude of P/P_o for approximately the first day after the accident. Although P/P_o for a fission reactor is of the same order as for the RTPR, the difference in *specific* afterheat power can be significant. The RTPR generates 3 600 MWt in a 718 m^3 blanket, which corresponds to an operating power density of 16.7 MW/m^3 (of blanket). The volume fraction of niobium in the RTPR blanket is 6.1 vol% (44 m^3), and all afterheat can be conservatively assumed to be deposited within the niobium. This assumption leads to an average afterheat power density of 4.4 MW/m^3 (of niobium) shortly after shutdown. The peak-to-average afterheat power ratio in the niobium is ~ 2.3 ,* with the maximum occurring at the first wall.

Detailed heat-transfer calculations must be made which account for post-accident core configurations before the significance of this difference in afterheat power density can be accurately evaluated. New data on the $^{94}\text{Nb}(n,\gamma)$ cross section will in all likelihood significantly reduce the calculated ^{95}Nb contribution to afterheat. Furthermore, more detailed

*This value of peak-to-average afterheat ratio is much less than might be inferred from the attenuation of the total neutron flux through the blanket. Most of the activation which contributes to afterheat results from successive neutron captures (^{93}Nb , ^{94}Nb , ^{95}Nb) and is therefore most dependent upon the low energy neutron flux.

calculations performed on fusion reactor designs which incorporate activities induced in the impurities found in commercially pure structural alloys may increase P/P_0 for fusion reactors. However, fusion

reactors as now envisioned are inherently low power density machines, and afterheat problems which portend significant meltdown in event of a loss-of-cooling accident do not appear significant.

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