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Fusion Fuel Cycle Solid Radioactive Wastes

**B. F. Gore
J. D. Kaser
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June 1978

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ABSTRACT

Eight conceptual deuterium-tritium fueled fusion power plant designs have been analyzed to identify waste sources, materials and quantities. All plant designs include the entire D-T fuel cycle within each plant. Wastes identified include radiation-damaged structural, moderating, and fertile materials; getter materials for removing corrosion products and other impurities from coolants; absorbents for removing tritium from ventilation air; getter materials for tritium recovery from fertile materials; vacuum pump oil and mercury sludge; failed equipment; decontamination wastes; and laundry waste. Radioactivity in these materials results primarily from neutron activation and from tritium contamination. Activation product isotope production is subject to control by plant designers through materials selection and location.

For the designs analyzed (which were prepared in part to identify needed research) annual radwaste volume was estimated to be 150 to 600 m³/GWe. This may be compared to 500 to 1300 m³/GWe estimated for the LMFBR fuel cycle. Major waste sources are replaced reactor structures and decontamination waste.

Research to improve the economic attractiveness of fusion power by reducing reactor materials requirements and by increasing lifetimes of structural materials may reduce estimated volumes of replaced structures significantly. Estimates of waste volumes from decontamination are extremely uncertain. More complete design information is required to improve them.

CONTENTS

ABSTRACT	iii
LIST OF TABLES	vi
LIST OF FIGURES	vi
INTRODUCTION	1
REACTOR REQUIREMENTS OF THE D-T FUEL CYCLE	2
FUSION POWER PLANT DESIGNS	11
WASTE SOURCES, MATERIALS AND QUANTITIES	13
REPLACED BLANKET STRUCTURES	13
CORROSION PRODUCTS AND SPENT GETTERS	17
CONTAINMENT AIR TRITIUM ABSORBENTS	19
BRED TRITIUM RECOVERY MATERIALS	20
FUEL RECYCLE PURIFICATION WASTES.	22
VACUUM SYSTEM WASTES.	23
FAILED EQUIPMENT	24
DECONTAMINATION AND LAUNDRY WASTE	24
SUMMARY AND CONCLUSIONS	27
REFERENCES.	29

LIST OF TABLES

1	Fusion Power Plant Designs Studied	11
2	Primary Fusion Reactor Radwastes.	13
3	Annual Waste Quantities--Blanket Replacement.	14
4	Annual Waste Quantities--Corrosion Product Cleanup.	19
5	Annual Waste Quantities--Bred Tritium Recovery	22
6	Annual Waste Quantities--Vacuum Systems	23
7	Annual Waste Quantities--Equipment Failure	24
8	Annual Waste Quantities--Decontamination and Laundry	25
9	Summary--Annual Fusion Fuel Cycle Solid Radwaste Quantities	27

LIST OF FIGURES

1	Deuterium-Tritium Fuel Cycle	2
2	Fusion Power Plant Systems and Functional Relations	4
3	Wisconsin Tokamak Fusion Reactor, UWMAK-I.	6
4	Mirror-LLL	7
5	Theta Pinch-LASL	8
6	Inertial Method of Plasma Confinement	9
7	Retrievable Storage Concepts for Waste Blanket Structure	16
8	Corrosion Product Recovery from Liquid Lithium	18
9	Containment Atmosphere Tritium Cleanup.	20
10	Bred Tritium Recovery	21
11	Fuel Purification	22

INTRODUCTION

Fusion is a potential energy source with a major advantage of having an almost unlimited fuel supply. Fusion is the process in which the nuclei of light elements are forced to combine and form new light elements with a net yield of energy. The development and deployment of fusion power plants will result in the production of a variety of solid radioactive wastes, some of which differ from those produced in the fission fuel cycle.

The first generation fusion reactors will be based on the deuterium-tritium (D-T) reaction since it is the easiest fusion reaction to achieve. Present conceptual designs include the entire fuel cycle within each power plant.

In this paper, the D-T fuel cycle and the basic requirements which it places upon reactor designs are reviewed. Existing fusion reactor design concepts are identified and briefly characterized. Reactor systems and structures necessary for the operation of fusion power plants are described. Finally, primary radwastes and their sources are described and annual quantities are estimated.

REACTOR REQUIREMENTS OF THE D-T FUEL CYCLE

First generation fusion reactors are expected to use the deuterium-tritium (D-T) fuel cycle because the plasma temperature required to ignite the D-T reaction is lower than for other fusion reactions. Plasma density and confinement time requirements are also lower. Naturally occurring tritium, however, is insufficient in quantity and concentration for use as a fuel. Consequently, a tritium breeding fuel cycle is required. The D-T fuel cycle is shown schematically in Figure 1. Present conceptual reactor designs include the entire fuel cycle within each power plant.

FIGURE 1. Deuterium-Tritium Fuel Cycle

The fusion of D-T, illustrated in the upper left of Figure 1, releases 17.6 MeV of energy, carried by a ⁴He nucleus and a 14 MeV neutron. That neutron can produce a triton (tritium atom) by inelastic scattering off a ⁷Li nucleus at energies above 2.25 MeV, or by fissioning a ⁶Li nucleus at thermal energies. Sequential reactions are possible, as are neutron losses to parasitic capture in structural materials. Reactor materials surrounding the plasma chamber must be so chosen and located that one or more triton is produced per D-T fusion reaction.

The helium "ash" of D-T fusion is not radioactive. However, large fluxes of energetic neutrons will activate reactor structures, coolants, and impurities present in structures, coolants and fuel materials. Activation and tritium breeding are the primary sources of radioactivity in D-T fuel cycle wastes.

The D-T fusion fuel cycle places certain basic requirements on conceptual reactor designs. The plasma must be confined and heated to ignition conditions. This is a matter of primary research importance at the present time. Plasma confinement methods strongly influence the details of fusion reactor design; the design in turn influences amounts of wastes generated. All fusion reactors must have a chamber in which the reacting plasma can be isolated from surrounding structures. This plasma chamber must be surrounded by a thick material "blanket" to stop and capture the energetic neutrons liberated in the fusion reaction. Lithium must be contained in this blanket for tritium breeding. In addition, because 80% of the energy released in D-T fusion is carried by the neutrons (and consequently can only be recovered as heat energy) fusion reactors must incorporate a thermodynamic cycle for electric power generation. In most designs this is a steam cycle.

Two recycle loops are required in fusion power plant designs which incorporate the entire D-T fuel cycle. First, fuel that has passed through the plasma and escaped without reacting must be recovered, purified, and returned to the plasma. (Generally only a quarter or less of the plasma reacts per pass.) And second, bred tritium must be recovered from the blanket and sent to the fuel system for purification and injection into the plasma.

All fusion power plant designs which incorporate the entire D-T fuel cycle require systems to perform the functions enumerated above. Necessary systems and structures, and their functional interactions are shown schematically in Figure 2. This representation is sufficiently general to represent all existing D-T fusion power plant designs.

FIGURE 2. Fusion Power Plant Systems and Functional Relations

Less abstract representations of D-T fusion reactors and power plants are shown in Figures 3-6. Each of these designs is based upon the confinement of fusing plasma by a different physical mechanism.

- The tokamak method of magnetic confinement of plasma is illustrated in Figure 3. It is typified by a large toroidal plasma chamber and blanket, enclosed by massive superconducting magnet coils.
- The mirror method of magnetic confinement is illustrated in Figure 4. In this concept plasma is confined between magnetic "mirrors" produced by two specially shaped superconducting magnet coils.
- The theta pinch method of plasma confinement is illustrated in Figure 5. It is characterized by a smaller diameter plasma chamber than tokamak confinement, which may be toroidal as shown, or linear.
- The inertial method of plasma confinement is illustrated in Figure 6. Intense laser beams focused on a pellet of solid fuel create, heat, and compress the plasma in this concept. Multiple plasma chambers are shown in the figure, serviced by a central laser facility.

The duration of thermonuclear burn in these fusion reactor designs varies from continuous (mirror) through hour-long (tokamak) through a fraction of a second (theta pinch) to a fraction of a microsecond (laser pellet).

The plasma confinement concept (or concepts) which may ultimately succeed in producing reactor grade plasma is not known. However, research effort presently is concentrated on the tokamak and inertial confinement concepts. The method of plasma confinement chosen for a fusion powerplant design strongly influences design details, including:

- reactor geometry
- blanket thickness and layout
- thermonuclear burn duration (power cycle)
- plasma heating and fueling requirements
- spent plasma recovery and recycle requirements
- support systems required.

From the standpoint of waste generation, however, the choice of materials within the reactor blanket is the controlling factor. For instance, the choice of blanket structural material controls radiation damage rates and activation products created. The chemical compatibility of blanket structure with fertile material and coolant controls corrosion rates, and limits the blanket operating temperature. Both plasma confinement method and blanket material composition significantly affect fusion power plant wastes.

1. PLASMA	6. Li PRIMARY SYSTEM (ONE FOR EACH OF 12 MODULES)	11. HOT CELL REPAIR AREA
2. TOROIDAL FIELD COIL (12)	7. Na SECONDARY SYSTEM (ONE FOR EACH OF 12 MODULES)	12. MAGNET SHIELD
3. RETRACTABLE DIVERTOR COIL (8)	8. DIVERTOR Li PRIMARY SYSTEM	13. BLANKET
4. TRANSFORMER COIL (10)	9. DIVERTOR Na SECONDARY SYSTEM	14. DIVERTOR COLLECTION AREA
5. EVACUATED PRIMARY CONTAINMENT BUILDING	10. TURBINE-GENERATOR BUILDING	15. MOTORIZED MODULE SUPPORT VEHICLE
		16. AUXILIARY EQUIPMENT AREA

FIGURE 3. Wisconsin Tokamak Fusion Reactor, UWMAK-I

FIGURE 4. Mirror-LLL

FIGURE 5. Theta Pinch-LASL

FIGURE 6. Inertial Method of Plasma Confinement

FUSION POWER PLANT DESIGNS

Data for this paper were obtained from analysis of the eight conceptual fusion power plant designs listed and characterized in Table 1. These were the current designs when the data for this paper were assembled (1975). In accordance with the research emphasis placed upon tokamak plasma confinement there are five designs based upon this concept, and one each for the other concepts. Figure 3 shows the first tokamak conceptual design listed, prepared by the University of Wisconsin and designated UWMAK-I.⁽¹⁾ Other designs based upon tokamak plasma confinement include an alternate blanket design for UWMAK-I produced by BNL,⁽²⁾ a second Wisconsin design using different materials, UWMAK-II,⁽³⁾ a design from Princeton Plasma Physics Laboratory⁽⁴⁾ and a design from ORNL.⁽⁵⁾ The LLL mirror fusion reactor conceptual design⁽⁶⁾ is shown in Figure 4. Figure 5 shows the LASL theta pinch conceptual power plant.⁽⁷⁾ The laser-pellet conceptual power plant designed by LASL⁽⁸⁾ is shown in Figure 6.

TABLE 1. Fusion Power Plant Designs Studied

<u>Confinement Method</u>	<u>Name or Originator</u>	<u>Fertile Material</u>	<u>Coolant</u>	<u>Blanket Structure</u>
Tokamak	UWMAK-I	Liquid Li Metal		316 SS
	BNL Blanket	LiAlO ₂ He		Al
	UWMAK-II	LiAlO ₂ He		316 SS
	Princeton	Flibe		PE-16
	ORNL	Liquid Li Metal		Nb
Mirror	LLL	Liquid Li Metal		SS
Theta Pinch	LASL	Liquid Li Metal		Nb
Inertial	LASL	Liquid Li Metal		Nb

These designs, which form the quantitative basis for this paper, are conceptual in nature (as opposed to fully engineered). They are based upon different assumed extrapolations of present technological capabilities, and therefore wide ranges of estimates are drawn from them.

The designs in many cases represent first attempts to integrate all aspects of fusion power plant performance and to define and scope subsystems to perform all necessary functions for plant operation. An objective of many of the design studies was to identify through the design attempt itself components which cannot be realistically designed at the present time. Necessary technology can then be developed while plasma confinement research is being carried out, to be ready when it is needed.

Fusion power plant designs are expected to evolve considerably as research proceeds. Consequently quantitative estimates of waste production may change radically. This process is already occurring. For instance, the confinement of reactor grade plasma by the methods postulated in the LLL mirror design and by the LASL theta pinch design is now considered unlikely. Mirror studies are turning to other concepts, including linked regions of mirror confinement ("bumpy torus") and "field reversed" mirrors in which plasma currents significantly contribute to confinement. Theta pinch studies are being directed away from the toroidal geometry assumed in the design study analyzed. Nevertheless, reactor designs based upon new plasma confinement schemes must include systems to perform the functions indicated in Figure 2, and those systems will be the waste sources. Consequently the information developed in early reactor designs is relevant to the assessment of the magnitude of the waste problem.

Assessment of the early fusion reactor designs considered here has indicated that the large, low power density tokamaks (such as UWMAK-I) would be impractical from an economic point of view. Consequently research emphasis is being placed on development of more compact and economically attractive design concepts. Many aspects of such concepts, while increasing economic viability, may also reduce radioactive waste production.

WASTE SOURCES, MATERIALS AND QUANTITIES

The systems and structures required to perform the functions indicated in Figure 2 are the sources of estimated fusion power plant radwaste. Analysis of power plant design documents yielded the primary radwastes identified in Table 2. These are treated sequentially in the remainder of this paper. In many cases the discussion is general because reactor design documents did not address the topic of waste production; because system design was conceptual and lacking engineering detail; and/or because operating experience with proposed systems does not exist. Consequently some estimates have been made on a generic basis.

TABLE 2. Primary Fusion Reactor Radwastes

- Replaced Blanket Structures
- Corrosion Products and Spent Getters
- Containment Air Tritium Absorbents
- Bred Tritium Recovery Materials
- Fuel Recycle Purification Wastes
- Vacuum System Materials
- Miscellaneous Failed Equipment
- Decontamination and Laundry Wastes

REPLACED BLANKET STRUCTURES

Per unit of energy released, fusion reactors produce about four times as many neutrons as fission reactors. These 14 MeV neutrons can induce radiation damage reactions which few-MeV fission neutrons cannot (helium production in particular). Consequently replacement of structures located near the plasma chamber wall is expected to be necessary, although estimates of material lifetimes vary. Estimated waste quantities per year per GWe of plant capacity are listed in Table 3 for the designs studied.

TABLE 3. Annual Waste Quantities--Blanket Replacement (t/GWe)

	<u>Stainless Steel</u>	<u>Niobium</u>	<u>Aluminum</u>	<u>Other Materials</u>
Tokamak				
UWMAK-I	490			
BNL			144	140
UWMAK-II	490			175
Princeton	110			
ORNL		27		212
Mirror-LLL	192			66
Theta Pinch-LASL		19		34
Laser-LASL	90			
<hr/>				
LWR Spent Fuel Rods--	32			

Based on these conceptual designs, replacement of damaged structures could yield from 90 to 490 metric tons of stainless steel, or from 20 to 30 metric tons of niobium per year per GWe. Reasons for the wide range and the differences between materials include higher operating temperature and thermal efficiencies possible with niobium structure; differing estimates of blanket lifetimes; differing structural complexity of various designs; and the use of graphite instead of stainless steel as a neutron moderator. Other materials requiring replacement include the aluminum structure in the BNL design, and the graphite moderator, solid lithium compound fertile materials, and beryllium used as a neutron multiplier in some designs. These waste quantities are quite significant compared to the 32 metric tons of fuel rods removed from a 1000 MWe LWR each year.

Present research efforts may lead to significant reductions in estimates of annual blanket structural waste. Development of more compact reactor structures would reduce the size of structures to be replaced. And development of design concepts having extended structural lifetimes would reduce

replacement frequency. These research efforts are being vigorously pursued in order to improve the economic attractiveness of fusion power plant designs by reducing construction costs and increasing plant availability.

The blanket materials removed from fusion reactors will be highly radioactive. Calculations indicate that the total Curie content of fusion reactors constructed of niobium or stainless steel could be comparable to that of fission reactors. Fusion reactor Curie content could be reduced significantly, however, if designs using low activation alloys of aluminum or vanadium can be developed. The radioisotopes produced by neutron activation are determined by reactor material choices, in contrast to the fission product spectrum over which the reactor designer has no control. Also, activation products produced in fusion reactor structures are dispersed and immobilized in large volumes of structural materials, and they do not include actinide elements. Although waste volumes could be large, management of the waste is eased, compared to fission reactor wastes, by the reduced biological hazard of the waste and by its relative immobility.

Waste blanket materials would have to be processed for storage or disposal. Baking to remove tritium would be required to minimize the possibility for tritium losses to the environment. Then, following possible recovery of materials for recycle, smearable contamination would be removed and the blanket materials might be compacted. This is especially attractive in reactor designs where the replaced materials is primarily thin flow ducting for liquid metal coolants. Compaction would be followed by packaging for storage, which will certainly be required if storage is to take place offsite.

Retrievable storage of replaced blanket materials (as opposed to ultimate disposition of the materials) appears attractive because the structural materials removed generally have a high recovery value. In addition, the radioactivity is fixed within the material, which makes it hard to remove by processing, but reduces the possibility of its dispersal. Finally, the afterheat generation per unit volume in this material is expected to be low, and air-cooling will probably be adequate during storage.

Various concepts have been proposed for the retrievable storage of waste blanket structures. Two possibilities are shown in Figure 7. On the left is a storage tunnel, similar to one existing on the Hanford Reservation.

FIGURE 7. Retrievable Storage Concepts for Waste Blanket Structure

Waste packages are loaded onto railroad flatcars which are then positioned inside the shielded tunnel and allowed to remain there until radioactivity levels have decayed appropriately, or until the need to retrieve the material and recycle it is realized.

The second storage concept shown is from the Brookhaven design, in which the blanket is made up of a large number of individual modules containing moderator and fertile material. In this concept the individual blanket modules would be removed from the reactor and, following decontamination, stored intact in cased wells in the ground. Easy retrievability for reprocessing and recycling, when that becomes desirable, is a feature of both these designs.

CORROSION PRODUCTS AND SPENT GETTERS

Liquid metals tend to be corrosive, and liquid lithium is more corrosive than the liquid sodium used in LMFBRs. Two conceptual methods of recovering corrosion products from liquid lithium are illustrated schematically in Figure 8. The method used in the UWMAK-I design is illustrated on the left. It uses gettering by zirconium foil, followed by cold trapping to remove lithium oxides (and possibly nitrides) which might be produced by air inleakage. The zirconium foil getter must be packaged and disposed of when it is inactivated by collected impurities.

The conceptual method used to recover corrosion products and other impurities in the theta pinch design is illustrated on the right of Figure 8. It differs greatly from the UWMAK-I method, and it also recovers tritium bred in the liquid lithium. The lithium is mixed with a molten salt having a higher affinity for the impurities than lithium, and then separated by centrifuging. The corrosion products travel with the molten salt (a eutectic mixture of LiCl and KCl) and are recovered by either cold trapping or electrolysis.

The annual quantities of waste per GWe which may be produced by corrosion product cleanup have been estimated from the conceptual designs and are listed in Table 4. They range from 75 metric tons of getter material plus two tons of corrosion products, through 6.4 tons of corrosion products recovered in

FIGURE 8. Corrosion Product Recovery from Liquid Lithium

the cold trapping of molten salt in the Princeton design, to 0.4 metric tons recovered through the molten salt contactor in the theta pinch design. The estimate of getter waste is derived from an assumed bed life of one year. The differences between amounts of corrosion products recovered stem from differences in the corrosivity of lithium and molten salt interacting with stainless steel and niobium.

TABLE 4. Annual Waste Quantities-Corrosion Product Cleanup (t/GWe)

	<u>Corrosion Products</u>	<u>Getter Material</u>
Tokamak		
UWMAK-I	1.9	75 (Zr Foil)
Princeton	6.4	(Cold Trap)
Theta Pinch-LASL	0.4	(Salt Contactor)

CONTAINMENT AIR TRITIUM ABSORBENTS

Maintaining MPC levels of tritium in reactor containment building atmospheres is expected to require large flows of air through cleanup systems which oxidize the tritium to water (HTO) and capture it by drying the air in beds of molecular sieve materials. This is indicated schematically in Figure 9. Heating the molecular sieve beds releases the water for recovery and regenerates the beds for reuse. The beds are long lived, but can be deactivated by contamination (by hydrocarbon vapors for instance). An order of magnitude estimate of waste from this process is 20 metric tons per year per GWe, composed of aluminosilicate molecular sieve material. This is based upon a postulated loss of one bed every ten years. The assumed mass of bed material is 200 metric tons, having a volume of 300 m³. Such a bed would be capable of processing an air flow of 60 m³/sec (100,000 cfm).

FIGURE 9. Containment Atmosphere Tritium Cleanup

Detailed designs for fusion reactor containment buildings, from which air inleakage rates may be calculated, have not been published. The amount of tritiated water which might be recovered by a containment atmosphere clean-up system is not estimated here, because it depends primarily on the rate of air inleakage and the humidity of the inleaking air.

BRED TRITIUM RECOVERY MATERIALS

Two methods of recovering tritium bred in lithium fertile material are indicated in Figure 10. The method used in the UWMAK-I design is illustrated on the left. Tritium bred in liquid lithium would be absorbed into a bed of yttrium metal foil for later evolution by heating. Surface contamination of the foils would eventually require that the yttrium be removed and replaced, packaged, and disposed of as solid waste.

FIGURE 10. Bred Tritium Recovery

The method used in the Princeton design is shown on the right of Figure 10. Tritium would be recovered in the form of tritium fluoride from the molten salt flibe in which it is bred. The TF gas would be evolved from the flibe in a spray chamber, collected as a liquid, and then introduced into an electrolytic cell. The TF would be absorbed into the potassium fluoride electrolyte and separated by electrolysis into T_2 and F_2 gases. Wastes from this process would result primarily from contamination of the potassium fluoride in the electrolytic cell, which would be recovered in a cold trap and packaged for disposal.

Annual quantities of waste which could be produced in the recovery of bred tritium are listed in Table 5. The estimate of yttrium foil waste is based upon a postulated bed life of one year. Tritium is recovered in the theta pinch design by the salt contactor process used to recover corrosion products. Wastes from this process are listed in Table 4.

TABLE 5. Annual Waste Quantities--Bred Tritium Recovery

	(t/GWe)	Material
Tokamak		
UWMAK-I	1.3	Yttrium Foil
Princeton	0.5	KF + Crud
Theta Pinch-LASL	(included with corrosion products)	

FUEL RECYCLE PURIFICATION WASTES

Isotopic separation of ordinary hydrogen (^1H) from deuterium and tritium by cryogenic distillation is the primary process required in fuel purification for recycle to the plasma. Additional steps include the cold trapping of condensable materials such as carbon dioxide, nitrogen, oxygen, and argon, plus the oxidation of tritiated hydrocarbons to water, followed by the electrolytic decomposition of the tritiated water and return of the hydrogen isotopes to the cryogenic distillation column. Process flows are indicated schematically in Figure 11.

FIGURE 11. Fuel Purification

Activation of air inleakage into the plasma chamber will produce some radioactivity in the gases rejected by this purification system. However, it is anticipated that it will be possible to collect and hold these gases until radioactivity levels have decayed sufficiently to allow their release. The primary wastes resulting from fuel purification are expected to be failed equipment.

VACUUM SYSTEM WASTES

Wastes produced in vacuum systems include mercury sludge, vacuum pump oil, and charcoal from cryopumps. Contaminated mercury from diffusion pumps could be distilled and the resulting sludge packaged for disposal. The sludge would contain primarily an amalgam of structural material sputtered from the walls of the plasma chamber. Inactivated charcoal from cryopumps would be used to absorb pump oil wastes when that is possible. Otherwise the oily wastes would be absorbed on an absorbent for packaging and disposal as solid waste.

Annual quantities of waste per GWe which could be produced by vacuum systems are listed in Table 6. Although the sludge quantities are similar, the charcoal/oil quantities differ considerably. The Princeton design uses mercury diffusion and ejector pumps, and does not use cryopumps.

TABLE 6. Annual Waste Quantities--
Vacuum Systems (t/GWe)

	Mercury Sludge Plus Erosion Product Amalgam	Cryopump Charcoal Plus Absorbed Pump Oil
UWMAK-I	1.1	27
Princeton	1.5	<1

FAILED EQUIPMENT

Failed equipment will be repaired where possible. Otherwise it will be compacted and packaged for storage or disposal. Major sources of failed equipment include vacuum systems, fuel systems and coolant cleanup systems. Annual quantities of failed equipment which might arise from these systems have been estimated and are listed in Table 7.

TABLE 7. Annual Waste Quantities--
Equipment Failure

<u>Source</u>	<u>m³/GWe</u>
Vacuum System Hardware	15
Fuel System	1-15
Accelerator Parts	
Pellet Injectors	
Purification Equipment	
Coolant Cleanup Hardware	<3

DECONTAMINATION AND LAUNDRY WASTE

Laundry waste differs from decontamination waste by its detergent content, which is difficult to process. Consequently, it is anticipated that laundry waste would be released through cooling tower blowdowns, with dilution to acceptable release levels. Nondetergent solutions used for equipment decontamination would be processed by settling and by evaporating as appropriate. Oily sludge resulting from this process would be solidified by mixing it with asphalt and then storing it in drums. Aqueous concentrates from water evaporation would be solidified by mixing with concrete and pouring into drums.

Annual waste quantities which might be produced by decontamination and laundry activities are listed in Table 8. They result from the assumptions that the large plants (with outputs ranging from the 1.5 GWe of UWMAC-I to the 4.1 GWe of the LASL theta pinch design) might each produce: 20 kg/day of oily liquids; five times the aqueous waste of light water fission reactors; and 100,000 gal/yr of laundry waste. The spread in quoted values results from the range of power outputs of the various plants.

TABLE 8. Annual Waste Quantities--
Decontamination and Laundry
(m³/GWe)

	<u>Before Solidification</u>	<u>Solidified</u>
Oily Sludges	2-7	7-20
Aqueous Concentrate	35-100	70-200
Detergent Liquids	100-260	--

The estimated quantities of solidified decontamination wastes are characterized by large magnitude, broad spread, and weak basis. Improvement of these estimates will require fusion power plant designs which are much more complete. The large magnitude and broad spread of these estimates cause estimates of total fusion fuel cycle solid radwastes to be extremely uncertain.

SUMMARY AND CONCLUSIONS

Volumes of solid radwaste production estimated for the conceptual D-T fusion power plant designs analyzed are listed in Table 9. Decontamination is seen to be the largest waste source. It is also the most uncertain. Improved power plant design information is required to reduce this uncertainty.

TABLE 9. Summary--Annual Fusion Fuel Cycle
Solid Radwaste Quantities (m^3/GWe)

Blanket Replacement	20-260
Molecular Sieve Material	30
Getters and Recovered Products	3-60
Failed Equipment	15-35
Decontamination	80-220
Total	150-600

Blanket replacement is the second largest waste source. However, present research efforts to reduce fusion reactor costs and materials usage by developing smaller structures having extended lifetimes could reduce wastes from this source considerably. Such research is being vigorously pursued in order to improve the economic attractiveness of fusion power. Recycle of replaced structures could also reduce waste quantities from this source.

Radioisotopes in fusion reactor wastes result from neutron activation and tritium contamination. Neutron activation of structures is subject to control by plant designers through materials selection and location. Low activation designs are being developed. Activation products are generally immobilized in structures and dispersed throughout large volumes of material. And, the biological hazard potential of fusion reactor contents has been calculated to be lower than for fission reactors.

The total annual volume of estimated solid radwaste production shown in Table 8 is seen to range from 150 to 600 m^3/GWe . Comparison with quantities produced by commercial light water fission reactors is not particularly meaningful because the introduction of commercial fusion reactors is not

expected before the year 2010. In that time frame, comparison with fission fuel cycles requires consideration of breeder reactors. Annual solid radwaste quantities expected from the LMFBR fuel cycle range from 500 to 1300 m³/GWe.^(9,10) Thus, solid radwaste quantities from the D-T fusion fuel cycle may be comparable to, or smaller than, those from the LMFBR fuel cycle. Management of these wastes is expected to be a nontrivial but solvable problem.

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