**AN EVALUATION OF RETENTION AND DISPOSAL OPTIONS
FOR TRITIUM IN FUEL REPROCESSING**

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CHAPTER 5. AN EVALUATION OF RETENTION AND DISPOSAL OPTIONS FOR TRITIUM IN FUEL REPROCESSING

1. INTRODUCTION

This chapter assesses the possible options for retention of tritium and its ultimate disposal during future reprocessing of irradiated oxide fuels discharged from light water reactors (LWRs) and liquid metal fast breeder reactors (LMFBRs). The assessment includes an appraisal of the state of the retention and disposal options, an estimate of the dose commitments to the general public, an estimation of the incremental costs of the several retention and disposal options, and the potential reduction of the dose commitments resulting from retention and disposal of the tritium. The assessment is based upon an extensive study of tritium retention in reprocessing completed in 1982 by Grimes et al.¹

Two plants were assumed, one to process LWR oxide fuel and the other to process LMFBR fuel. In each base case plant the tritium was vaporized to the atmosphere. Each of the hypothetical plants was assumed to be constructed during the 1990's and to operate for a 20-year lifetime beginning in the year 2000 at a rate of 1500 metric tons of heavy metal (MTHM) per 300-d year.

In addition to the base case (Case I), six other cases which included tritium retention options were examined. Although many of the features of the base-case plants remain unchanged in the tritium retention options, each case requires some additions, deletions, and modifications of portions of

the plants. The retained tritium must also be managed and disposed of in a manner that is environmentally acceptable. These additional cases, summarized in Figure 1, were:

- Case II The tritium is concentrated by recycling 90% of the water, in addition to the acid; 12% of the water and tritium is vaporized to the stack; and a sidestream from the recycle water stream is disposed of by one of several disposal options.
- Case III Tritium is concentrated by voloxidation and disposed of by one of several disposal options.
- Case IV Tritium is separated by isotopic separation from a 90% recycle stream and is disposed of by one of several disposal options.
- Case V Tritium is separated by isotopic separation from the discharge or nonrecycle stream (10%) and disposed of by one of several disposal options.
- Case VI The aqueous streams in the plants are isolated so that tritium can be confined and concentrated in the head-end portions of the plant. This concentrate can then be disposed of by one of the several disposal options.

The "disposal" options considered for this assessment range from storage for 50-100 years with subsequent dispersal of the undecayed HTO to the atmosphere, to isolation in a geologic repository. These options are very briefly described in Section 3.

Capital and operating costs incremental to the base-case plants were estimated by the same method for each of the retention options. The

TRITIUM RETENTION OPTIONS

Fig 1. Tritium Retention Options

base-case plants (for which the incremental costs were, of course, zero) were conceptually divided into about 30 functional modules. Those modules of the base-case plants that required additions, deletions, or modifications were identified for each of the retention options, and the incremental capital costs for design, fabrication, and installation of the required major in-cell equipment items were estimated.

For both the retention and disposal options, the capital costs were converted to an annual capitalized cost by assuming a 20-year plant lifetime with zero salvage value at end-of-life and include the cost of capital (interest). Combining these annual capitalized costs with the operating costs yielded uniform annual costs for each of the options.

The total tritium available for release from a 5 t/d LWR reprocessing plant is 8×10^5 Ci/year, which is 20% of the estimated four megacuries that are generated worldwide by natural processes each year. An LMFBR reprocessing facility would have about one-seventh of this quantity available for release; however, this quantity could increase the natural background radiation by only very small fractions of a percent on a world-wide basis.

Applicable Regulations and Dose Limits for Reprocessing Plants

The allowable concentration of tritium leaving the site as water has been set at 3 mCi/m^3 and of air at 0.2 Ci/m^3 . The limit for the concentration of tritium in air within the reprocessing plant is set at 5 Ci/m^3 (10 CFR 20).²

The Nuclear Regulatory Commission (NRC) of the United States subscribes fully to the principle that radiation exposures to the public from nuclear

facilities should be kept As Low As is Reasonably Achievable (ALARA). The NRC has given (Appendix I of 10 CFR 50)² a quantitative definition of ALARA dose levels for routine emissions from light-water-cooled nuclear reactors, however it has issued no such quantitative guidance for fuel reprocessing plants.

The Environmental Protection Agency (EPA) has adopted a rule-making action (40 CFR 190) that applies to several parts of the uranium fuel cycle associated with light-water reactors. This rule specifically includes fuel reprocessing as a part of this fuel cycle. By this rule, exposures to (1) planned discharges of radioactive materials (except radon and its daughters) and (2) direct radiation from uranium fuel cycle operations should not result in annual dose commitments in excess of 25 millirem to the whole body and to any organ except the thyroid or in excess of 75 millirem to the thyroid of any members of the public. These limits would, presumably, be the maximum allowable for the processing plant, assuming that the individuals receiving the maximum exposure were affected by no other nuclear facility. It is presumed that these rules will be extended to apply to the breeder reactor fuel cycle also.

The limits of occupational doses are set (10 CFR 20) at 5 rem/year, with additional restrictions for the allowable dose in any quarter of the year.

Origin and Quantities of Tritium to be Considered

Tritium is continuously generated by natural processes and occurs naturally in the earth's surface waters. Spontaneous fission of uranium and thorium contributes trivial quantities of this isotope. Interaction of

high-energy cosmic rays with oxygen and nitrogen atoms in the upper atmosphere is believed to be responsible for production of about 1.6×10^6 Ci/year; this production rate would lead to a steady-state inventory³ of 28×10^6 Ci. A very similar estimate (26×10^6 Ci) of the steady-state inventory caused by cosmic ray interactions has recently been presented.⁴ Measurements of tritium concentrations in the earth's waters, however, suggest a considerably larger inventory of tritium.⁵ Solar flares^{6,7} may be responsible for much of the earth's total natural tritium. It seems to be generally accepted⁴ that the earth's inventory of natural tritium is near 70×10^6 Ci; this corresponds to an appearance rate of about 4×10^6 Ci/year.

Tritium is formed within nuclear reactor fuels (both LWR and LMFBFR) by ternary fission and may also be formed there by neutron bombardment of light elements (such as boron or lithium) if they are present within the fuel. Additional tritium is generated in the reactor through interaction of neutrons with some materials (of which lithium and boron are the most important) used for control of excess reactivity and for control of pH. Such tritium, generated outside the fuel element of an LWR or LMFBFR, is of some environmental concern for the particular reactor, but it is not delivered to the reprocessing plant and does not concern the present assessment.

Little, if any, of the tritium generated within the fuel of the modern LWR is lost via diffusion through the zirconium cladding.⁸ Some small quantity (perhaps 0.1 to 1% of that produced) may be lost at such a reactor

by leakage through cladding defects.^{9,10} The ORIGEN code,¹¹ often used for estimating isotope generation in long-term fuel cycle projections, predicts that ternary fission will be responsible for the generation of 450 Ci of tritium per MTHM irradiated at an average specific power of 30 MW/MTHM to a burnup of 33,000 MWd/MTHM.

Although essentially no tritium escapes through the zircaloy cladding, a portion diffuses into the zircaloy and is immobilized as zirconium tritide. The quantity so immobilized, which is not available for release on dissolution of the oxide fuel, is a function of operating parameters and cannot be said to be surely known. Goode and Vaughn⁸ found 13% of the total tritium within the zirconium cladding of a commercial pressurized water reactor (PWR) fuel irradiated to about 40,000 MWd/MTHM. Recent experimental data¹² showed that for PWR fuel roughly 2/3 of the tritium was in the cladding and 1/3 in the unclad fuel. For BWR fuel, the tritium was equally distributed between the cladding and the fuel.

To assure conservative results, it is assumed that the LWR oxide fuel delivered to the reprocessing plant will contain 600 Ci of tritium per MTHM. Of this total, 10% (60 Ci/MTHM) is assumed to be retained within the cladding; the remainder (540 Ci/MTHM) is assumed to reside within the fuel oxide and to be available for release.

The amount of tritium that will be produced during operation of a large LMFBR is not well known both because the fast fission yields of tritium are not well established and because design details for large LMFBRs are not fully developed. Moreover, a large fraction of the tritium generated in the core fuel has been found to diffuse through the stainless steel fuel

cladding and be trapped as NaT in the sodium coolant circuits.¹⁵ Tritium trapped in this way is not of concern to the reprocessing plant. Because the blanket operates at a lower temperature, a lower percentage of tritium will diffuse through the cladding.

Tritium will arise in the LMFBR fuel both by ternary fission and by activation of impurities (boron and lithium) in the core and blanket oxides. The quantities expected to be produced have been estimated by several authors. The quantities expected to be delivered to the processing plant are addressed subsequently.

It seems conservative to assume that the tritium produced by ternary fission will total 1090 Ci/MTHM for the blended core and blanket oxides (the core material responsible for 1035 Ci/MTHM and the combined radial and axial blankets responsible for 55 Ci/MTHM). Adding the (trivial) contributions from the lithium and boron activation suggests 1040 Ci/MTHM generated in the core and 60 Ci/MTHM generated in the combined radial and axial blanket. These estimates would seem to be conservative (perhaps by as much as a factor of 2) unless the recent ~~indications~~^{14,15} of markedly higher fast fission yields of tritium from U and Pu prove to be correct. The

quantity of interest is, of course, the quantity of tritium that is delivered to the reprocessing plant. Again, there is no certain knowledge of the real situation, and it has been assumed that the core fuel retains 3% of the tritium generated therein and that the radial and axial blankets each retain 90% of that generated within them. As a consequence, the blend of core and blanket materials will contain 85 Ci of tritium per MTHM.

Accordingly, a plant processing 1500 MTHM/year would receive 1.28×10^5 Ci/year of tritium. All of this tritium would be contained in the oxide, and essentially none would reside in the stainless steel cladding. If all of the tritium were released to the atmosphere, some 30 such reprocessing plants (servicing some 120 LMFBs) would be required to equal the appearance rate of naturally produced tritium.

2. CHARACTERISTICS OF BASE-CASE REPROCESSING PLANTS

Two base-case plants, one to process LWR fuel (case LWR-I) and one to process LMFBR fuel (case LMFBR-I), were envisioned. Each of these plants was assumed (as with all other plants of this assessment) to use variants of the Purex process and to reprocess 1500 metric tons of heavy metal (MTHM) per 300-d year.

The somewhat simplified flowsheets, including process fluid flows and tritium balances, are based on such 300-d operation at 5 MTHM/d. It is important to note, however, that to ensure the capability for processing 1500 MTHM/year, each feature and equipment item of the base-case (and of all other) plants are assumed to be designed and sized to operate at an 8-MTHM/d rate.

All plants were assumed to have an operating lifetime of 20 years and to operate between the years 2000 and 2020. Each plant, including LWR-I and LMFBR-I, was assumed to discharge no radioactivity in aqueous effluents to the environment and to meet applicable regulations. All plants were assumed to include facilities for solidifying high-level waste, and headend aqueous waste (HAW) solutions were assumed to be evaporated and calcined promptly. All LWR processing plants (cases LWR-I through LWR-VI) were assumed to receive 600 Ci of tritium per MTHM, with 60 Ci/MTHM immobilized within the fuel cladding and 540 Ci/MTHM available for release. All LMFBR plants (cases LMFBR-I through LMFBR-VI) were assumed to receive 85 Ci of tritium per MTHM in the fuel, with all of this available for release.

The base-case plants (LWR-I and LMFBR-I) were assumed to recycle plant acid but to use no recycle of process water and to discharge all releasable tritium (as HTO) to the atmosphere by vaporization of process water through

a single stack 100 m high. The cost for tritium retention and disposal for these base-case plants was assumed to be zero; costs of the tritium retention and disposal options were evaluated (in mid-1986 dollars) for each other case based on equipment, maintenance, and operating costs incremental to those of the base-case plants.

Figure 2 shows a simplified flowsheet for the base-case plant for LWR fuels (case LWR-1) operating at 5 MTHM/d; water balances are shown in liters per hour and tritium balances (the circled numbers) in curies per day. The flowsheet assumes that all recycle process water is sent to a vaporizer and is discharged to the stack as water vapor. Daily tritium releases (2700 Ci) represent the total present in the 5 t of fuel processed each day, minus that (300 Ci) entrapped in the cladding hulls.

A simplified flowsheet for the base-case plant for processing LMFBR fuel (LMFBR-I) operating on 5 MTHM/d is shown in Fig. 3, with water balances in liters per hour and tritium balances (circled figures) in curies per day. The flowsheet assumes that none of the tritium (425 Ci/d) will be immobilized by the stainless steel fuel cladding, but instead, all will be released to the stack as HTO in water vapor. The LMFBR flowsheet shows that twice the amount of water is required for reprocessing LMFBR fuel than for LWR fuel. Because of criticality considerations, aqueous streams in both flowsheets (LMFBR and LWR) will consist of approximately the same fissile quantity (in grams per liter). The LMFBR flowsheet will require larger and more expensive equipment.

The base-case plants discharge larger quantities of tritium to the environment than do any of the others which have provisions for retaining

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Circled numbers: Curies of tritium per day.
Uncircled numbers: Liters of water per hour.

Fig. 1 - LWR reference case - no recycle.

Circled numbers: Curies of tritium per day.
Uncircled numbers: Liters of water per hour.

Fig. 1. Case I - LMFBR reference case.

this isotope. The base-case plants are the largest consumers of water, since all other plants in this study include recycle of a large fraction (about 90%) of the plant process water. However, the base-case plants use fresh makeup water and therefore send very little tritium or traces of fission products through the chemical makeup areas. Only the recycle acid will contain tritium or trace fission products. Because of their low-tritium and fission-product content, the base-case plants present the smallest opportunity for exposure of plant personnel.

3. OPTIONS AND COSTS FOR RETENTION, INTERIM STORAGE, AND DISPOSAL

Options for Tritium Retention

When light-water reactor (LWR) fuel is sent to be processed, a very large fraction of the generated tritium from the reactor will be delivered with the fuel to the reprocessing plant. The liquid-metal fast breeder reactor (LMFBR), whose fuel must be reprocessed if such reactors are to be viable, will send only a small fraction of its generated tritium to a fuel-reprocessing facility. These fuel reprocessing plants are the only fuel cycle facilities, except for the reactors themselves, that will handle tritium in appreciable quantities.

Table 1 identifies and details some general features of the tritium retention options examined in this assessment of separate plants processing oxide fuels from LWRs and LMFBRs. It is apparent from Table 1 that none of the retention options completely eliminates dispersal of tritium to the atmosphere.

None of these options, other than LWR-I, has been practiced in operating reprocessing plants; none of the several options has been optimized for the particular application; and, given the present state of the art, it is far from clear that such optimization is practical. Each option has been examined objectively and has been evaluated on a comparable basis for its feasibility, complexity, and relative incremental cost.

Cases LWR-II and LMFBR-II: High Water Recycle

These plants differ from the base-case plants in that process water is recycled in a manner to minimize to a practicable level the fresh water

TABLE 1. Some characteristics of tritium retention options^a

Case	Tritium to dissolver solution (% of total)	Tritium concentration methods		Disposition of tritium		Volume of concentrate (m ³ /year) ^b
		Water recycle (% of total)	Additional method for concentration	In concentrate in solution (% of total)	Vaporized to atmosphere (% of total)	
LWR-I and LMFBR-I	100	0	None	0	100	
LWR-II and LMFBR-II	100	90	None	87.5	12.5	3880
LWR-III and LMFBR-III	4	90	Fuel voloxidation ^c	95.3/93.4	4.7/6.6	14.4
LWR-IVA and LMFBR-IV	100	90	Isotope separation ^d	95.3/93.4	4.7	900
LWR-IVB	100	90	Isotope separation ^d	99.5	0.48	900
LWR-VA and LMFBR-V	100	90	Isotope separation ^e	95.3/93.4	4.7/6.6	86.4
LWR-VB	100	90	Isotope separation ^e	99.5	0.49	86.4
LWR-VI and LMFBR-VI	100	90	First-cycle con- finement ^f	97.8	2.2/4.0	2660

^aSee text for more detailed descriptions.

^bVolumes shown are, in all cases, those for plants processing LWR fuel. Those from processing LMFBR fuel differ substantially in a few cases; these values are presented in subsequent portions of this section.

^cA large fraction of the tritium is volatilized and collected during oxidation of fuel material before dissolution.

^dIsotope separation is applied to the entire water recycle stream.

^eIsotope separation is applied only to excess water for discharge to disposal and dispersal systems.

^fTritium is confined to the first (high-activity) cycle, with balance of plant waters containing tritium at low concentrations.

required for plant operations. The simplified flowsheets for these cases are shown in Figures 4 and 5. The flowsheets were prepared using the assumptions detailed for the base-case plants (cases LWR-I and LMFBF-I) in Section 2.

LWR and LMFBF plants employing this manner of recycle discharge only 349 and 53 curies of tritium per day, respectively, to the atmosphere, but they furnish relatively large volumes of tritiated water for management and disposal. Careful design of the plants is vital to ensure protection of the plant personnel. Such safe design and operation would seem feasible, and LWR reprocessing plants employing such recycle have been designed.¹⁶

Cases LWR-III and LMFBF-III: Voloxidation with Water Recycle

If tritium can be removed from the oxide fuel before it contacts the dissolver solution, a very small quantity of water with high tritium concentration can be obtained for disposal (NOTE: the specific activity of tritium is approximately 10^4 ci/g). A process by which such removal has been shown to be effective in relatively small-scale operations is known as voloxidation.¹⁷⁻²⁰

Voloxidation releases tritium by heating and oxidizing the oxide pellets of the sheared fuel. When LWR fuel is used, tritium is released with the disintegration of the UO_2 fuel matrix as it expands during its oxidation to U_3O_8 . Tritium in voids and along grain boundaries is released immediately, and tritium within the matrix diffuses quickly from the small grains. The exact form of the tritium in the fuel is not known, but most of it is in the oxidized form as it leaves the voloxidizer.

When LWR fuel, or other fuel containing a sufficiently high concentration of UO_2 , is employed, the voloxidation process has been shown in small-scale tests to remove >99% of the tritium.^{17,18} However, when fuel

Circled numbers: Curies of tritium per day.

Uncircled numbers: Liters of water per hour.

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Fig. 4. Case II - LWR high recycle.

Circled numbers: Curies of tritium per day.
Uncircled numbers: Liters of water per hour.

Fig. 4-2 Case II - LMFBR high recycle.

containing more than about 15% of PuO_2 are voloxidized, the conversion to U_3O_8 does not proceed, the fuel pellets do not disintegrate, and longer times or higher temperatures are necessary to achieve high removal efficiencies. It has been established²¹ that voloxidation of LMFBR fuel will require a voloxidizer kiln with holdup volume four times that used with LWR fuel.

It has been assumed, somewhat arbitrarily, that oxidation of UO_2 fuels (LWR fuels or LMFBR blanket materials) will remove 96% of the contained tritium on a 5 t/d scale. This assumption would result in a release of 4% of the tritium in the oxide matrix (180 Ci/d) to the LWR dissolver solution. It is also assumed that voloxidation of LMFBR fuel will be slightly less effective and that 25 Ci/d of tritium (5.8% of the total) will be delivered to the dissolver solution with the mixed fuel blanket oxides. The simplified flowsheets for plants using voloxidation of LWR and LMFBR fuel are shown in Figures 6 and 7.

Cases LWR-IVA, LWR-IVB, LWR-VA, LWR-VB, LMFBR-IV, and LMFBR-V: Isotope Separation in Aqueous Streams with Water Recycle

Without other prior treatment, the tritium contained in the oxide fuel matrix is diluted by the acidic dissolver solution, and some form of isotope separation must be employed in order to obtain a concentrated tritium solution for management or disposal.

Choice of Isotope Separation Techniques

Separation of hydrogen isotopes, primarily for production of deuterium or D_2O from normal water, has been studied for years. Lin,²² in his review of the art, considered six methods and concluded that two processes (the dual temperature water-- H_2S exchange process and the cryogenic hydrogen distillation process) had the highest potential for economic and

Circled numbers: Curies of tritium per day.
 Uncircled numbers: Liters of water per hour.

Fig. 7.3. Case III - LWR voloxidizer plus high recycle.

Circled numbers: Curies of tritium per day.
Uncircled numbers: Liters of water per hour.

Fig. 3/67
Fig. 3A. Case III - LMFBR voloxidizer plus high recycle.

technical feasibility. Burger and Trevorrow²³ concluded that electrolysis coupled with chemical exchange or molecular excitation by lasers might prove to be viable alternatives to those selected by Lin.

An attempt was made to establish the preferred isotope separation options from the published values of capital and operating costs of the several systems. In general, the published costs were for systems that differed in throughput, degree of separation, concentration in product streams, project and process contingency factors, and value of capital money. Many of the published estimates contained little or no information about some of these parameters. In short, the literature did not yield conclusive information about the relative process economics of the several systems.

As a consequence, the several separation methods were compared on a subjective (and somewhat arbitrary) effectiveness basis. The seven criteria considered in this comparison were separation factor, throughput, process readiness, operational simplicity, industrial hazards, maintainability, and energy requirements. Based upon this review, it was concluded that:

1. The laser-isotope separation process was not selected because this process has only been evaluated on a small scale, and thus, large-scale feasibility has not been established. This process needs to be reexamined in the future, and it may well become a competitor but is not considered further in this study.
2. Cryogenic distillation requires that large volumes of molecular hydrogen be generated, liquefied, stored, and distilled. It was rejected for this study because it seems very difficult to certify that it is sufficiently safe for remote or semi-remote operations.

3. Conventional electrolysis and bipolar electrolysis showed the lowest effectiveness rating of all the processes evaluated. Both of these options were rejected for this study, although further development may make bipolar electrolysis a competitor.

As a consequence of this evaluation, the two isotope separation processes with the highest ratings were selected for consideration in this study. These processes are the combined electrolysis catalytic exchange and the fractional distillation of H_2O . In addition, the modified or dual-temperature H_2S exchange process was included because it represents an improved version of a proven large-scale process. This modified process introduces pure water to the top of the cold towers, and, as a consequence, yields markedly larger volumes of tritium-depleted water than do the other two processes considered.

Requirements for the Isotope Separations Process

At least two options that use isotope separation processes on aqueous plant solutions are available. These options are detailed as LWR-IVA and IMFBR-IV in the following text. One of these would use the isotope separation system on the entire water recycle system of the reprocessing plants. The other alternative would use this isotope separation process only on the discharge stream to provide (with a considerably smaller volume for treatment) a concentrated tritium solution for management or disposal and a dilute solution of tritiated water for atmospheric dispersal. These alternatives are designated as LWR-VA and IMFBR-V. As a means of testing the sensitivity of removal efficiencies on the overall cost of the facility, two additional sub-cases were examined. In these isotope separation

options for LWR fuel, the retention efficiency was increased from 95 to 99.5% of the tritium, and their costs were estimated. These are listed as cases LWR-IVB and LWR-VB.

Where the combined electrolysis catalytic exchange (CECE) process or fractional distillation of water is used as the isotope separation process, the simplified flowsheets for LWR-IVA, LWR-IVB, and LMFBR-IV are shown as Figures 8 through 10. For these two isotope separations, the flowsheets for LWR-VA, LWR-VB, and LMFBR-V are Figures 11 through 13.

It should also be noted that, although the isotope separation systems are large facilities, they are designed to operate with tritium as the only significant radionuclide in the water, and, as a consequence, do not require containment within high-cost, heavily shielded cell areas.

Cases LWR-VI and LMFBR-VI (Tritium Confinement to First Cycle Solutions)

A proposed alternative method of isolating the tritium in fuel reprocessing plants is that of confining this isotope, insofar as possible, to the first cycle portion of the plant.²⁴⁻²⁶

The hydrogen in Purex solvent exchanges very slowly with tritium; as a consequence, the tritium carried in the organic stream of the first cycle is effectively limited to that in the (about 5%) entrained and dissolved nitric acid and water.²⁷ The tritium/protium ratio in the organic stream is effectively set by the tritium/protium ratio in the scrub stream and is subject to some control. Process parameters can be adjusted to limit tritium passage to the uranium and plutonium decontamination equipment to the order of 0.2% of the tritium processed through the first cycle components.

CASE IV A LWR
ISOTOPE SEPARATION
ON HIGH RECYCLE*

*Flow sheet applicable to CECE and fractional distillation processes.

Circled numbers: Curies of tritium per day.

Uncircled numbers: Liters of water per hour.

Fig. 4.6. Case IV A - LWR isotope separation on high recycle.

CASE IV B LWR
ISOTOPE SEPARATION
ON HIGH RECYCLE, LOW RELEASE

Circled numbers: Curies of tritium per day.
Uncircled numbers: Liters of water per hour.

Fig. 4.7. Case IVB - LWR isotope separation on high recycle low release.

*Flow sheet applicable to CECE and fractional distillation processes only.

Circled numbers: Curies of tritium per day.

Uncircled numbers: Liters of water per hour.

Fig. 4.8. Case IV - LMFBR isotope separation on high recycle.

*Feed to vaporizer shown for CECE and fractional distillation processes. See text.

Circled numbers: Curies of tritium per day.

Uncircled numbers: Liters of water per hour.

Fig. 49. Case VA - LWR isotope separation on recycle side stream.

*Feed to vaporizer shown for CECE and fractional distillation processes. See text.

Circled numbers: Curies of tritium per day.

Uncircled numbers: Liters of water per hour.

3 12

Fig. 4-10. Case VB - LWR isotope separation on recycle side stream, low release.

*Feed to vaporizer shown for CECE and fractional distillation processes. See text.

Circled numbers: Curies of tritium per day.

Uncircled numbers: Liters of water per hour.

Fig. 4-11. Case V - LMFBR isotope separation on recycle side stream.

The key to head-end confinement is to reduce the water additions to an absolute minimum through the use of high-tritium-level process water (i.e., chemical make-up solutions, high-level acid, stream jets, etc.). The acid must be made using NO_x produced with a denitrator using recycle acid from the low-level side. The low-level concentrator bottoms add to the water input to the high level side. Also, a small amount of water from decontamination solutions must be disposed of in this system. In addition, the transfer of water with high tritium concentrations to the low side of the plant must be reduced to a minimum. These precautions will keep the amount of tritiated water for disposal as small as possible. Simplified flowsheets for cases LWR-VI and LMFB-R-VI are shown in Figures 14 and 15.

It should be noted that these options yield large volumes of tritiated water for disposal and that the concentration of tritium in the process water returned to the chemical make-up area of the plant is the highest of any of the options considered. It also seems likely that plants of this design are more vulnerable to upset conditions than are most of the other options considered.

Estimated Costs of the Retention Options

A previous study²⁸ provided bases for much of the capital cost information developed in the present assessment. All capital costs presented below are reported in mid-1986 dollars.

For each case of the methods of tritium isolation, a detailed listing was prepared showing major in-cell process equipment required. The three isotope separation processes [modified dual-temperature Girdler-sulfide (GS), combined electrolysis catalytic exchange (CECE), and vacuum

**CASE VI - LWR
HEAD-END CONFINEMENT**

Circled numbers: Curies of tritium per day.
Uncircled numbers: Liters of water per hour.

Fig. 4.12. Case VI - LWR head-end confinement.

Fig. 7.19. Case VI - LMFBR head-end confinement.

distillation] were each sized to provide the separation required for cases LWR-IVA, LWR-IVB, LWR-VA, LWR-VB, LMFB-IV, and LMFB-V. Estimated costs of each item were derived. Tables 2 and 3 summarize the incremental costs [above those of the base-case plants (numbers LWR-I and LMFB-I)] for each case.

Several points, obvious from examination of these tables, include:

1. Incorporation of water recycle (cases LWR-II and LMFB-II) results in a modest decrease in capital cost as a result of requiring smaller final evaporators.
2. In the isotope separation systems that operate on the entire recycle water (cases LWR-IVA, LWR-IVB, and LMFB-IV), the modified GS process appears to be substantially less expensive than the CECE process and markedly less expensive than vacuum distillation. However, as previously noted, application of the modified GS process to these particular cases yields dilute tritiated water in volumes so large as to preclude its dispersal via evaporation through the plant stack; accordingly, no vaporization equipment is included for these cases. Moreover, it should be noted that for these cases all the isotope separation processes appear to be more expensive than voloxidation.
3. First cycle containment, although it requires an addition of several equipment items, is relatively inexpensive (cases LWR-VI and LMFB-VI).

Operating costs were estimated on a comparable basis for all options for processing LWR and LMFB fuels. LWR-I and LMFB-I again served as the

Table 2. Capital cost comparisons for options retaining tritium from LWR fuel

Case	Description	Incremental costs (\$1 000)		
		In-cell equipment	Total direct	Total project ^a
I	Reference case - no recycle	0	0	0
II	High recycle	-258	-851	-2 070
III	Voloxidation	6 474	36 253	91 310
IVA	Modified Girdler-sulfide process ^{b,c}	9 486	53 121	133 800
	Vacuum distillation ^b	59 954	334 622	842 830
	Combined electrolysis catalytic exchange ^b	15 149		84 835
IVB	Modified Girdler-sulfide process - low release ^{b,c}	10 183	57 024	143 632
	Vacuum distillation - low release ^b	62 238	348 532	877 868
	Combined electrolysis catalytic exchange ^b	15 263	85 479	215 300
VA	Modified Girdler-sulfide process ^d	2 775	15 541	39 144
	Vacuum distillation ^d	10 667	59 737	150 464
	Combined electrolysis catalytic exchange ^d	2 724		37 898
VB	Modified Girdler-sulfide process - low release ^d	3 525	19 745	49 732
	Vacuum distillation - low release ^d	13 565	75 966	191 340
	Combined electrolysis catalytic exchange - low release ^d	3 248	18 190	45 820
VI	First cycle confinement	1 654	7 943	23 338

^aTotal project includes: 30% field distributable costs, 25% home office costs, and 55% contingency; it totals a 2.52 multiplier on total direct costs before rounding

^bIsotope separation process is applied to entire (about 90%) water recycle.

^cThese options, which require very large additions of fresh water to the isotope separation operation, are impracticable unless on-site, deep-well disposal of very large volumes of water is feasible (see text).

^dWater recycle (about 90%) employed but isotope separation is applied to discharge (excess) water only.

Table 3. Capital cost comparisons for tritium retention options for LMFBR fuel

Case	Description	Delta costs (\$1 000)		
		In-cell equipment	Total direct	Total project ^a
I	Reference case - no recycle	0	0	0
II	High recycle	-435	-1 435	-3 450
III	Voloxidation	10 012	56 066	141 204
IV	Modified Girdler-sulfide process ^{b,c}	19 909	111 492	280 820
	Vacuum distillation ^b	126 587	708 889	1 785 515
	Combined electrolysis catalytic exchange process ^b	32 174	180 178	453 826
V	Modified Girdler-sulfide process ^d	4 528	25 356	63 865
	Vacuum distillation ^d	13 862	77 627	195 525
	Combined electrolysis catalytic exchange process ^d	4 500	25 193	63 455
VI	First cycle containment	1 793	10 040	25 285

^aTotal project includes: 30% field distributable costs, 25% home office costs, and 55% contingency; it totals a 2.52 multiplier on total direct costs before rounding

^bIsotope separation process is applied to entire (about 90%) water recycle.

^cThis option, which requires very large additions of fresh water to the isotope separation process, is impracticable unless on-site, deep-well disposal of very large volumes of tritiated water is feasible (see text).

^dWater recycle (about 90%) employed but isotope separation is applied to discharge (excess) water only.

base cases, and operating costs for the individual options incremental to those cases were estimated. Table 4 summarizes the incremental annual operating costs for several cases.

It should be noted that the options using water recycle only (cases LWR-II and LMFBR-II) show operating costs below those of the corresponding base-case plants.

When the isotope separation systems are applied to all the recycle water (cases LWR-IVA, LWR-IVB, and LMFBR-IV), the modified GS process shows the lowest operating costs, largely because no costs for evaporation of excess water are included. However, as noted previously, this process could meet the ground rules of this study only if on-site, deep-well disposal of very large volumes of tritiated water were feasible and permissible.

Tables 5 and 6 summarize the cost evaluations for LWR and LMFBR reprocessing plant options for retention of tritium with government financing assumed (20-year facility lifetime, zero salvage value, and cost of money at 7.5%) for the capital. Those cases involving isotope separations have assumed that the CECE process would be used.

Use of the isotope separation process on the total recycle stream (cases LWR-IVA, LWR-IVB, and LMFBR-IV) leads to the most expensive of the facilities and these cases are ones that furnish relatively large volumes of tritiated water to storage or disposal (see Tables 7 and 8). It has the virtue of sending a quite dilute tritiated water as recycle to the plant.

Operation of the isotope separation system on the discharge stream (cases LWR-VA, LWR-VB, and LMFBR-V) produces (see Tables 7 and 8) a much smaller volume of tritiated water for disposal and, because of the smaller

Table 4. Incremental annual operating cost summary
(mid 1986 dollars)

Case	Description	Incremental cost (\$1 000)
LWR-I		
LWR-II		-1 558
LWR-III		1 370
LWR-IVA	Modified Girdler-sulfide process ^a	2 780
	Distillation	9 375
	Combined electrolysis catalytic exchange	4 500
LWR-IVB	Modified Girdler-sulfide process ^a	2 990
	Distillation	9 910
	Combined electrolysis catalytic exchange	5 340
LWR-VA	Modified Girdler-sulfide process	5 400
	Distillation	2 420
	Combined electrolysis catalytic exchange	1 613
LWR-VB	Modified Girdler-sulfide process	5 590
	Distillation	2 760
	Combined electrolysis catalytic exchange	1 713
LWR-VI		1 783
LMFBR-I		
LMFBR-II		-3 243
LMFBR-III		1 606
LMFBR-IV	Modified Girdler-sulfide process ^a	4 833
	Distillation	18 816
	Combined electrolysis catalytic exchange	9 526
LMFBR-V	Modified Girdler-sulfide process	9 827
	Distillation	3 186
	Combined electrolysis catalytic exchange	2 240
LMFBR-VI		2 505

^aThese options are impracticable unless on-site, deep-well disposal of very large volumes of tritiated water is feasible (see text). No costs for evaporation of water are included.

Table 5. LWR retention cost summary - government financing
(7.5% interest, 20-year lifetime)

Costs	Case							
	I Reference case	II Water recycle	III Voloxi- dation	IVA Isotope separation ^a	IVB Isotope separation ^b	VA Isotope separation ^c	VB Isotope separation ^d	VI First cycle confinement
Capital costs. million \$	0	-2.1	91.4	213.6	215.3	34.2	49.7	23.3
Capitalized cost. million \$/year	0	-0.203	8.96	20.96	21.12	3.84	4.87	2.29
Operating costs. million \$	0	-1.56	1.29	5.00	5.34	1.61	1.71	1.78
Uniform annual costs. million \$	0	-1.76	10.25	25.96	26.46	5.45	6.58	4.07
Annual tritium reduction to environment, MCi	0	0.705	0.772	0.772	0.806	0.772	0.806	0.792
Annual cost. \$/Ci retained		-2.50	13.27	33.67	32.84	7.10	7.87	5.15

^aCombined electrolysis catalytic exchange process on total recycle stream.

^bCombined electrolysis catalytic exchange process on total recycle stream; very low release to atmosphere.

^cCombined electrolysis catalytic exchange process on discharge stream only.

^dCombined electrolysis catalytic exchange process on discharge stream only; very low release to atmosphere.

Table 6. LMFBR retention cost summary - government financing
(7.5% interest, 20-year lifetime)

Costs	Case					
	I Reference	II Water recycle	III Voloxi- dation	IV Isotope separation ^a	V Isotope separation ^b	VI First Cycle confinement
Capital costs. million \$	0	-3.5	141.2	453.9	63.5	25.3
Capitalized cost, million \$/year	0	-0.338	13.850	44.530	6.228	2.477
Operation costs. million \$		-3.243	1.606	9.526	2.240	2.505
Uniform annual costs. million \$	0	-3.59	15.46	54.05	8.47	4.98
Annual tritium reduction to environment, MCi		0.112	0.119	0.119	0.119	0.122
Annual costs. \$/Ci retained		-32.02	129.86	454.30	71.21	40.85

^aCombined electrolysis catalytic

^aCombined electrolysis catalytic exchange process on total water recycle.

^bCombined electrolysis catalytic exchange process on excess (discharge) water.

sizes of the separation facility, appears reasonably economical. The recycle process waters, however, are as concentrated in tritium as those from the operations (LWR-II and LMFBR-II) that employ water recycle only.

Confinement of the tritium to the first cycle (cases LWR-VI and LMFBR-VI) appears reasonably economical. However (see Tables 7 and 8), it produces a large volume of tritiated water for disposal, and it sends very concentrated tritium solutions to the chemical make-up areas.

Protection of Reprocessing Plant Personnel

The several retention options along with the base cases fall into two distinct categories with respect to concentration of tritium in water recycled to the chemical make-up areas of the plant as indicated and emphasized above.

As Table 7 shows, case LWR-IVA recycles water with 1.3 Ci/m^3 and case LWR-III recycles water at 7.4 Ci/m^3 . The base-case LWR-I uses fresh water with essentially no contained tritium. Cases LWR-II and LWR-V, on the other hand, recycle water with 182 Ci/m^3 , and case LWR-VI recycles water with 300 Ci/m^3 . The several options for LMFBR reprocessing (see Table 8) show a similar pair of categories, but at substantially lower tritium concentrations. The highest concentration in LMFBR recycle water, that of case LMFBR-VI, is 46 Ci/m^3 .

None of the recycle water streams will be completely free of radionuclides other than tritium; as a consequence, all will require precautions for protection of plant personnel. However, tritium will be virtually the only radionuclide capable of vaporization with the recycle water in the event of a leak or a spill. Tritiated water vapor in the

Table 7. Summary of water volumes and tritium concentrations for LWR fuel reprocessing

Case	Stack discharges			Discharge waste stream			Tritium		
	Water (L/h)	Tritium (Ci/d)	Tritium (Ci/L)	Water (L/h)	Tritium (Ci/d)	Tritium (Ci/L)	Recycle acid (Ci/L)	Process water (Ci/L)	Vessel off-gas (Ci/L)
I	5459	2700	0.0206	0	0	0	0.0212	0	0.0156
II	80	349	0.182	539	2351	0.182	0.182	0.182	0.182
III	376	127	0.0141	2	2573	53.604	0.00730	0.00739	0.00728
IVA	694	127	7.6×10^{-4}	125	2573	0.858	0.0221	0.00130	0.0171
IVB	294	13	1.8×10^{-3}	125	2687	0.86	0.0219	0.00104	0.0179
VA	611	127	8.7×10^{-3}	12	2573	8.93	0.182	0.182	0.181
VB	606	13	8.9×10^{-4}	12	2687	9.3	0.185	0.182	0.188
VI	2849	60	8.8×10^{-4}	369	2640	0.39	0.39	0.30	0.004

Table 8. Summary of water volumes and tritium concentrations for LMFBR fuel reprocessing

Case	Stack discharges			Discharge waste stream			Tritium		
	Water (L/h)	Tritium (Ci/d)	Tritium (Ci/L)	Water (L/h)	Tritium (Ci/d)	Tritium (Ci/L)	Recycle acid (Ci/L)	Process water (Ci/L)	Vessel off-gas (Ci/L)
I	11 351	425	0.0016	0	0	0	0.0016	0	0.0013
II	160	53	0.014	1 137	372	0.014	0.014	0.014	0.014
III	896	28	1.3×10^{-3}	2	397	8.3	0.0013	0.0013	0.0013
IV	1 787	28	6×10^{-4}	125	397	0.13	0.0017	0.00013	0.0015
V	1 141	28	1×10^{-3}	12	397	1.38	0.014	0.014	0.014
VI	5 698	17	1.2×10^{-4}	366	408	0.046	0.047	0.046	0.00066

atmosphere is taken into the human body both by inhalation and by absorption through the skin. Exposure of plant personnel, as a worst-case example, to a large spill or recycle water at 300 Ci/m^3 of tritium (from LWR-VI) would (at a normal breathing rate of $22 \text{ m}^3/\text{d}$ of air and with skin absorption at 50% that of inhalation) leads to the maximum permissible body burden of 1 mCi in about 12.5 minutes. In contrast, the recycle stream from LWR-III (voloxidation) would give the permissible body burden in about 8.5 h, and that from LMFBR-VI, which is the worst of the LMFBR retention cases from this point of view, would require 82 min. If the spill were small (1 ft^2 area), such as would result from a minor valve leak, then the concentration in the air five ft above the floor of a normally ventilated room would be less than the 10 CFR 20² limit for all concentrations shown except LWR-VI at 300 Ci/m^3 . It is presumed that leaks larger than this would be repaired and that these smaller leaks would be cleaned up or flushed to the suspect drain system.

It is obvious that for many areas in each of these plants, other radionuclides dominate personnel protection considerations, and in these areas, tritium contributes relatively little hazard to plant employees. Any plant that recycles acid (and water) will, as a minimum, have to treat this material as "suspect" or low activity.²⁹ In most cases, the tritium-containing streams also contain some other isotope of greater radiological hazard. Therefore, control of tritium usually is accomplished as a result of controlling something else and is not a direct result of a control measure directed mainly at tritium. However, for those few areas involving recycle water (where tritium represents the significant hazard), it is

equally obvious that special precautions must be taken against leaks and spills of tritiated water.

Options for Tritium Storage and Terminal Disposal

If retention of tritium by reprocessing plants is practised, it will be necessary to isolate this isotope from the biosphere for periods sufficient to permit a large fraction of it to decay. Such isolation could be accomplished by a relatively long-term storage or, in principle, by any of several methods for permanent disposal. The options considered in this assessment include: 50 year storage, 100 year storage, trench burial, monolith burial, deep-well disposal, hydrofracture, and deep-sea disposal.

10 CFR 61² permits ground burial of up to 40 Ci per cubic meter in shallow trenches (within 30 m of the ground surface) under reasonable restrictions as a Class A waste. Waste classified as Class B or C has higher restrictions with regard to depth of burial, immobilization, and containment. However, the curie concentration is governed only by heat generation or external radiation as long as the waste meets the other requirements.

Tank Storage Tank storage of high level wastes from military and from some commercial operations has been practiced since the beginning of spent fuel reprocessing in the United States and in Europe. Storage of alkaline high level wastes in first generation, single-walled steel tanks has led to leakage at Hanford and Savannah River and to losses to the subsurface soil at Hanford. However, subsequent experience with carefully constructed double-walled tanks has been excellent. Tanks for high level waste have recently been constructed at both Hanford and Savannah River, with an anticipated tank lifetime of 50 years. Storage of tritiated water in such

tanks, with essentially no other radioactivity, should present fewer problems. No cooling is required and no corrosive chemicals are present, although care must be taken (as with the high level waste) to recombine or otherwise control the radiolytic decomposition products of water. These double-walled, below-grade, steel tanks should be capable of storage of tritiated water for at least 50 years.

Storage for 50 years would reduce the tritium concentration to 6.0% of its initial value. For this assessment, it is assumed that the tritiated water is released to the atmosphere by vaporization from the original facility stack after the 50-year storage period. The added dose commitment to the affected population from this release must be combined with that from the tritium released during the original processing operation.

Nearly all the disposal options will require short-term storage of tritiated water. Such storage might be safely accommodated in somewhat less expensive tanks, but this assessment assumes that the uncooled, double-walled, below-grade, steel tanks anticipated to be for 50-year storage will be used for whatever short-term storage is required.

Storage as solid Tritiated water could be converted to a solid and given interim storage before final disposal. Many options for fixation of tritium that might be suitable for permanent disposal have been described.³¹⁻³⁴ Incorporation of tritiated water into cement provides one way of preparing tritium for permanent disposal. Additional water retaining solids, such as silica gel, clays, and molecular sieves can be incorporated in the mix. Asphalt coatings and polymer impregnation have been shown to reduce the rate of leaching of tritium from the cement.

Methods of packaging tritiated wastes have been described.³⁵

Rhinehammer and Mershad³⁶ would introduce tritiated solids into a thick-walled, 103-L (27-gal) polyethylene drum that is sealed and placed in a sealed asphalt-lined, 114-L (30-gal) stainless steel drum, and this in turn is contained in a sealed asphalt-lined, 209-L (55-gal) stainless steel drum. These operations may be carried out in enclosed systems and are therefore suitable for high levels of tritium concentration.

A recent review³⁷ gives the opinion:

For retention or immobilization, incorporation of tritiated water in cement blocks in containers is currently favoured unless the tritium has been captured by a high enrichment process. In this case, immobilization as zirconium hydride is favoured, although many other immobilizing matrices are being investigated.

For some of the final disposal options (deep-sea disposal, for example), interim storage as drummed solid would be necessary. It should be noted, however, that relatively short-term storage does not promise to simplify the ultimate disposal problem. Incorporation of even the most dilute (14 Ci/m^3 of tritium) concentrate in cement (yielding 8 Ci/m^3 of cement) would yield a waste that would have to be stored for 90 years to decay to the level ($5 \times 10^{-2} \text{ Ci/m}^3$) recommended for unrestricted release.³⁰

Options for Terminal Disposal

A variety of final disposal options for radioactive wastes have been practiced or are being investigated. Some classes of radioactive wastes are presently disposed of by trench burial, near-shore discharge to the sea,

pumping into deep wells, and shale fracturing (hydro-fracture). In addition, disposal by incorporation into concrete monoliths at intermediate depth and by deposition in geologic repositories are under active investigation, and disposal as solid waste to the deep oceans has been practiced. Nearly all of these options could, in principle, be used for ultimate disposal of tritiated water. The options are described briefly below:

Shallow trench burial Near-surface burial has been used routinely for several decades for the disposal of low- and intermediate-level radioactive wastes in the United Kingdom^{38,39} and Canada,^{40,41,42} as well as at several federal installations in the United States.⁴³⁻⁴⁸ Moreover, the disposal of low-level radioactive waste has been practiced at a total of seven commercial sites in the United States.^{43,49,50} It has been reported that some 645,000 Ci of tritium was buried in 1971 at one of the commercial sites.⁵⁰ Only three commercial burial sites are accepting wastes at present; these are those at Hanford (Washington), Beatty (Nevada), and Barnwell (South Carolina). The availability of any of these sites in the future for the burial of wastes containing moderate quantities of tritium appears questionable.

The most economical fixation technique for this disposal option would appear to be the incorporation of tritiated water into concrete that is then poured into steel drums. The concrete could hold some 20 to 25% of water by weight and provide a dilution factor of 1.8.

Concrete monoliths at intermediate depth The proposed guidelines of the Nuclear Regulatory Commission suggest that burial of the tritiated water

by fixation in concrete monoliths at intermediate depths may be an acceptable method of disposal. Such emplacement in concrete monoliths has been under examination for the disposal of partially decontaminated salt at Savannah River and at Hanford. For such burial to be employed, the top of the monolith must be more than 10 m below grade, and the water table must be well below the bottom of the monolith. These requirements may be met at many sites.

Such disposal might reasonably be done in the following manner.³⁰ The entire disposal area would be excavated to a depth of about 20 m; the bottom of the excavation must be above the water table. A layer (1.5 m thick) of compacted and essentially impermeable clay is placed in the bottom of the excavation, and a permeable (sand- or gravel-filled) layer is then emplaced to permit the collection and monitoring of any water that might penetrate the disposal area. Additional layers of compacted, impermeable clay would then be emplaced to a depth of 8 m above the permeable layer. The pit would then be backfilled with native soil to the original ground surface. For the disposal to proceed, the section to be used would be reexcavated to the top of the impermeable clay layer, and trenches (maximum 14 m in length, 6 m wide) would be excavated in the thick clay layer. The tritiated concrete would be poured into the trench; after the concrete has solidified, the trench would be covered with a compacted layer (1.5 m thick) of essentially impermeable clay. The used portions of the pit would then be backfilled with native soil to the original ground level.

It seems plausible that, given a site in which the water table is sufficiently below grade, such an emplacement would suitably immobilize

tritium and prevent its adverse interaction with the biosphere. The essential lack of moving water, the massive size of the monoliths, and, especially, the thick, protective, and essentially impermeable layer of clay would seem to preclude appreciable tritium migration. A considerable assessment of the extent and permanence of the tritium confinement would, however, seem necessary before such a procedure could be certain of acceptance.

Discharge to deep wells The Committee on Waste Disposal of the National Academy of Sciences generally endorsed the proposal that portions of deep sedimentary basins and coastal plain areas could be used as disposal reservoirs for liquid wastes containing radioactivity.⁵¹ Several reconnaissance studies of sedimentary basins were performed by the U.S. Geological Survey.⁵² In a subsequent review,⁵³ the National Academy of Sciences noted that truly satisfactory disposal sites would be permeable sedimentary rock layers at least 3000 ft deep in geologic basins confined above by relatively thick impermeable strata (i.e., shale or salt deposits) so that movement of liquid would be restricted vertically but not horizontally.

The U.S. Environmental Protection Agency is not categorically opposed to waste disposal by deep-well injection⁵⁴ but is on record as "being opposed to emplacement of materials by subsurface injection without strict controls and a clear demonstration that such emplacement will not interfere with present or potential use of the subsurface environment, contaminate groundwater resources, or otherwise damage the environment."

The disposal of chemical and petrochemical wastes in deep wells has increased markedly in the United States during the past 25 years,⁵⁴ however, this method has seldom been used in this country for the disposal of radioactive wastes.⁴⁹

Trevorrow et al.⁵⁵ examined the possibility of disposing of tritium-bearing aqueous waste in deep wells. They concluded that suitable sites were abundant in the United States; however, the only deep-well disposal of tritium to date has been done at the Idaho National Engineering Laboratory. There, low-level liquid wastes from the Idaho Chemical Processing Plant have been disposed of at a rate of $11,400 \text{ m}^3/\text{year}$ ($3 \times 10^6 \text{ gal/year}$) through a well about 600 ft deep into the Snake River aquifer.^{52,53,56,57} Nearly all the activity except tritium is removed from the wastes by distillation and ion exchange processes before discharge. Since 1952, approximately 22,000 Ci of tritium at an average concentration below $5 \times 10^{-4} \text{ Ci/m}^3$ has been discharged into the aquifer; the discharge rate since 1961 has been such that a quasi-steady-state inventory of 14,000 Ci of this isotope has been maintained in the aquifer.⁵⁷ The distribution and migration of radionuclides (along with stable sodium, chlorine, and chromium) in the aquifer have been monitored for many years by sampling from some 45 observational wells near and downstream from the discharges. The tritium plume (with boundaries taken as $2 \times 10^{-6} \text{ Ci/m}^3$) occupied an area of about 15 square miles in 1972.⁵⁷

Disposal of shale fracturing (Hydrofracture) Oak Ridge National Laboratory (ORNL) has used the hydrofracture technique for the disposal of intermediate-level wastes for several years. By this method, which is probably unique to ORNL, the waste solution is mixed with a solids blend of

cement and other additives and is injected into an impermeable and isolated shale formation at a depth of 215 to 300 m. The injection pressure is sufficient to propagate a thin, horizontal crack in the shale. As the injection of grout continues, this crack is filled by the grout and is further extended so that a thin, approximately horizontal grout sheet several hundred feet in diameter is formed. The grout sheet sets within a few days after completion of the injection and fixes the wastes in the shale formation. The preparation of a borehole for the injection consists of drilling a hole to and through the shale formation, lowering and cementing a casing to a total depth, and cutting the casing at the depth at which the fracture (and resulting grout sheet) is desired. A preliminary shale fracture is done by water pressure before injection of the grout. Subsequent cuts and fractures at increasingly shallower depths permit the formation of essentially parallel sheets a few feet apart.⁵⁸⁻⁶⁰

The well-head pressure (typically about 2500 psi to fracture and somewhat less to propagate the fracture) is carefully monitored during injection for any sudden drops that might indicate a vertical fracture. Such vertical fractures are not uncommon in the oil industry, but the 30 experimental and operational injections at the ORNL site appear to have each produced essentially horizontal grout sheets.⁶⁰

Geologic disposal The placement of radioactive wastes into stable geological formations can isolate these wastes from the biosphere for geologic time periods. Geologic storage is being examined both here and abroad for the disposal of high-level and long-lived radioactive wastes. It would not be economical to prepare a geological repository for relatively

short-lived radioactive species like tritium. If, however, geological repositories were opened for long-lived, high-level wastes, they might also be used for high-level, shorter half-life species.

The storage holes for high-level waste are spaced to reduce the thermal loading in the repository medium. Since the heat loading of even the highest tritium concentration is negligible, it should be possible to place tritium containers in positions interstitial to the high-level waste containers. The temperatures reached at such locations are not high enough to breach the tritium containers.

Disposal at sea The United States apparently began the practice of disposing of packaged, solid radioactive wastes into the deep oceans very early, but only relatively small amounts of radioactive materials have been given such disposal by this country.

A panel of experts, established by the International Atomic Energy Agency, examined the scientific and technical aspects of such disposals of radioactive materials. The panel reported⁶¹ that low- and intermediate-level radioactive wastes could be safely disposed of into the sea under carefully controlled and specified conditions; they also recommended criteria for selecting disposal sites and guidance for the design of sea disposal containers. A second panel⁴⁰ confirmed and extended the conclusion stated in 1961.

The general framework for the operational control of sea disposal was firmly established during eight sea-disposal campaigns carried out under the supervision of the Nuclear Energy Agency from 1967 to 1976. Since the first operation in 1967, up to eight European countries have taken part in these

campaigns.⁶² Three dumping sites have been used for these operations, following recommendations from a group of international experts from NEA countries. The sites were all located in the north-eastern Atlantic at an average depth of about 5 km.

The quantities of tritium adjudged to be safely releaseable to the North Atlantic Ocean are impressively large. In an assessment carried out under NEA supervision, the limiting capacity (Slansky's concept⁶³) of the North Atlantic was estimated to be 10^{15} Ci of tritium per year. The disposal of that quantity of tritium each year would lead (with tritium assumed quite conservatively to be released from its solidified state in the sealed drum at the moment of dumping) to dose levels equivalent to the appropriate ICRP dose limits for the most exposed members of the public. While it is clear that no one advocates the disposal of tritium in these quantities, it is obvious that the North Atlantic Ocean might safely contain very large quantities of this isotope. If the world were producing 5000 GW(e) years of power from LWRs, the reprocessing plants might collect as much as 1×10^8 Ci of tritium for disposal each year. The accumulated tritium from nuclear power programs by the year 2000 has been estimated to be about 5×10^8 Ci.^{62,64}

It seems apparent that sea bed disposal of tritium solidified in concrete and contained in steel drums should be a satisfactory disposal method.

Costs for the Disposal Options

Costs of the several disposal options are presented in mid-1986 dollars. The uniform annual cash flow method has been used, and all cash flows associated with the investment have been converted into an equivalent

uniform annual amount. Several disposal options effectively deal with the tritiated water produced by the retention system as it is produced. For these options, the disposal plant life is effectively 20 years, and the capital costs have been converted to the equivalent annual cost, assuming no salvage value at the end of 20 years. Other disposal options (such as long-term tank storage or disposal by intermediate depth burial) represent investments for a period far greater than the 20 years or entail surveillance and maintenance activities for a long period. For such disposal options, the uniform annual costs are computed for the longer period that the disposal or storage system must be used and manned.

Capital costs developed in this assessment include direct costs of material and labor as well as distributable field costs and home office costs. They do not include local taxes, insurance, interest during construction, and costs, if any, of facility decommissioning. Wherever possible, the estimated capital and operating costs are based on actual experience, with allowance for inflation. If, as is the case for some options, no applicable experience exists, estimates are based (with allowance for inflation) on available published results from other studies.

Table 9 presents a summary of the capital and operating costs of the several disposal options that might be considered for the large volumes of tritiated water resulting from retention option LWR-II employing recycle of 90% of the plant water. As expected, tank storage of these large volumes for 100, or even for 50 years, is expensive. Capital costs for the tanks for 50-year storage would amount to about 7% of the cost of the entire reprocessing facility, and longer term storage would be much more expensive.

Table 9. Summary of costs for disposal options for tritium from LWR fuel reprocessing with water recycle only (retention case LWR-II)

Disposal option		Capital	Capitalized	Operating	Uniform	Disposal cost
Number	Description	cost (million \$)	cost ^a (million \$/year)	cost (million \$/year)	annual cost (million \$/year)	
A-I	50-year storage	146	11.3	1.5	12.8	48.85
A-II	100-year storage	292	21.9	1.5	23.5	166.29
B-I	Trench burial	30	3.0	8.3	11.3	16.15
C-I	Monolith ^b	30	3.0	3.5	6.5	9.11
D-I	Deep-well disposal	0.7	0.07	0.1	0.16	0.23
E-I	Hydrofracture	13.8	1.35	0.7	2.0	2.86
G-I	Sea-bed disposal	36.4	3.57	9.7	15.7	22.36

^aCapitalized costs assume 7.5% interest rate in all cases

^bMonolith at intermediate depth in clay matrix; no long-term surveillance assumed.

None of the other disposal options would seem to entail capital costs of as much as 2% of the total facility, or to add more than \$10.00/kg to the cost of reprocessing heavy metal. It is clear, however, that the only disposal options that are relatively inexpensive for such large volumes of tritiated water are deep-well disposal or hydrofracture.

Tables 10 and 11 show similar cost data for cases LWR-III (voloxidation) and LWR-V (isotope separation on discharge water), both of which produce small volumes of tritiated water for disposal. Tank storage for 100 years would add approximately \$4.35/kg of heavy metal processed to the total reprocessing cost. The least expensive options, deep-well disposal and hydrofracture, would contribute about \$0.15 and \$0.62/kg to the cost of reprocessing heavy metal. For the intermediate (although still low) volume option LWR-V only, geologic disposal would add as much as \$8/kg of heavy metal to the reprocessing cost, although deep-well disposal and hydrofracture are the least expensive options.

In general, the costs of the disposal options for tritium from LMFBR fuel are similar fractions of the cost of the overall plant. However, primarily because far less tritium is delivered to the plant with the fuel, the costs per unit of tritium disposal or tritium stored are much higher than those from LWR fuel reprocessing.

Summary of Costs of Combined Retention and Disposal Cases

Tables 12 and 13 summarize the uniform annual costs and the annual cost in dollars per curie of confined tritium for the several combinations of retention and storage or disposal options for plants processing LWR and

Table 10. Summary of costs for disposal options for tritium from LWR fuel reprocessing with tritium retention by voloxidation (retention option LWR-III)

Disposal option		Capital	Capitalized	Operating	Uniform	Disposal cost
Number	Description	cost	cost ^a	cost	annual cost	(\$/Ci)
		(million \$)	(million \$/year)	(million \$/year)	(million \$/year)	
A-I	50-year storage	2.3	0.18	0.94	1.12	3.86
A-II	100-year storage	4.7	0.35	0.94	1.30	8.42
C-II	Monolith ^b	4.6	0.44	1.19	1.63	2.10
C-III	Monolith ^c	6.1		1.67	2.26	3.04
D-I	Deep-well disposal	0.7	0.07	0.09	0.17	0.23
E-I	Hydrofracture	9.5	0.92	0.10	0.94	1.23
F-I	Geologic disposal	3.5	0.33	2.50	2.83	3.65
G-I	Sea-bed disposal	4.1	0.40	1.52	1.92	2.48

^aCapitalized costs assume 7.5% interest rate in all cases

^bConcrete monolith poured in clay matrix; 150-year surveillance assumed.

^cDrummed concrete encased in monolith (ordinary concrete) poured in clay matrix; 150-year surveillance assumed.

Table 11. Summary of costs for disposal options for tritium from LWR fuel reprocessing with isotope separation on discharge water (retention case LWR-VA)^a

Number	Disposal option	Capital cost (million \$)	Capitalized cost ^a (million \$/year)	Operating cost (million \$/year)	Uniform annual cost (million \$/year)	Disposal cost (\$/Ci)
	Description					
A-I	50-year storage	7.8	0.61	0.88	1.49	5.11
A-II	100-year storage	15.9	1.19	0.88	2.07	13.46
C-II	Monolith ^b	4.6	0.44	1.19	1.63	2.10
C-III	Monolith ^c	10.9	1.06	2.79	3.85	5.00
D-I	Deep-well disposal	0.7	0.07	0.10	16.56	0.24
E-I	Hydrofracture	10.2	0.99	0.02	1.01	1.32
F-I	Geologic disposal	4.6	0.44	12.39	12.83	16.70
G-I	Sea-bed disposal	5.5	0.54	3.37	3.91	5.11

^aCapitalized costs assume 7.5% interest rate in all cases

^bConcrete monolith poured in clay matrix; 150-year surveillance assumed.

^cDrummed concrete encased in monolith (ordinary concrete) poured in clay matrix; 150-year surveillance assumed.

Table 12. Costs of combined retention and disposal options for LWR fuel reprocessing^a

Retention case	Disposal option								
	A-I 50-year storage	A-II 100-year storage	B-I Trench burial	C-I or C-II Monolith (concrete)	C-III Monolith (drums)	D-I Deep-well disposal	E-I Hydro- fracture	F-I Geologic disposal	G-I Sea-bed dispos
LWR-II water recycle									
Uniform annual cost (million \$/year)	11.12	15.72	9.56	4.66 ^b		-1.60	0.25		13.97
Cost (\$/Ci)	46.37	118.70	13.52	6.62 ^b		-2.21	0.41		19.87
LWR-III voloxidation									
Uniform annual cost (million \$/year)	11.36	8.40		11.87 ^c	12.59 ^d	10.42	11.19	13.08	12.17
Cost (\$/Ci)	17.11	15.70		15.46 ^c	16.28 ^d	13.52	14.49	16.97	15.73
LWR-IV-A ^e									
Uniform annual cost (million \$/year)	29.90	23.77		28.48 ^c		26.12	27.19		30.42
Cost (\$/Ci)	47.33	56.60		36.98 ^c		33.95	35.19		39.47
LWR-IV-B ^f									
Uniform annual cost (million \$/year)	30.40	24.10		28.98 ^c		26.62	27.68		30.91
Cost (\$/Ci)	46.51	56.00		36.16 ^c		33.12	34.50		38.64
LWR-VA ^g									
Uniform annual cost (million \$/year)	6.96	5.46		7.08 ^c	9.32 ^d	5.63	6.49	18.30	9.38
Cost (\$/Ci)	12.28	14.90		9.25 ^c	12.14 ^d	7.31	8.42	23.74	12.21
LWR-VB ^h									
Uniform annual cost (milliom \$/year)	7.84	6.10		7.96 ^c	10.20 ^d	6.51	7.37	19.18	10.27
cost (\$/Ci)	12.97	15.40		9.94 ^c	12.83 ^d	8.14	9.25	24.56	12.97
LWR-VI first cycle confinement									
Uniform annual cost (million \$/year)	13.81	15.80		8.93 ^c		4.23	5.78		14.88
Cost (\$/Ci)	37.81	85.00		11.32 ^c		5.38	7.31		18.77

^a7.5% interest on capital money in all cases^bOption C-I - poured concrete monolith in clay matrix; no surveillance assumed.^cOption C-II - poured concrete monolith in clay matrix; 150-year surveillance assumed.^dOption C-III - drummed concrete emplaced in concrete monolith in clay matrix; 150-year surveillance assumed.^eCombined electrolysis catalytic exchange process on total water recycle.^fSame as (e) but with very small atmospheric release.^gCombined electrolysis catalytic exchange process on excess (discharge) water only.^hSame as (g) but with very small atmospheric release.

Table 13. Costs of combined retention and disposal options for LMFBR fuel reprocessing^a

Retention case	Disposal option								
	A-I 50-year storage	A-II 100-year storage	B-I Trench burial	C-I or C-II Monolith (concrete)	C-III Monolith (drums)	D-I Deep-well disposal	E-I Hydro- fracture	F-I Geologic disposal	G-I Sea-bed dispos
LMFBR-II water recycle									
Uniform annual cost (million \$/year)	22.02	44.16	18.08	8.32 ^b		-3.42	-0.44		26.91
Cost (\$/Ci)	582.00	2042	163	74.66 ^b		-30.49	-3.86		242
LMFBR-III voloxidation									
Uniform annual cost (million \$/year)	16.57	16.75		17.07 ^c	17.17 ^d	15.62	16.39	18.29	17.37
Cost (\$/Ci)	154	185		144 ^c	148 ^d	131	138	153	146
LMFBR-IV ^e									
Uniform annual cost (million \$/year)	58.00	60.90	57.30	56.26 ^b		54.22	55.28		58.51
Cost (\$/Ci)	542	742	482	473 ^b		455	465		491
LMFBR-V ^f									
Uniform annual cost (million \$/year)	9.96	10.54		10.09 ^c	12.32 ^d	8.64	9.63	21.30	12.39
Cost (\$/Ci)	104	159		84.73 ^c	101 ^d	72.73	79.76	179	104
LMFBR-VI first cycle confinement									
Uniform annual cost (million \$/year)	14.72	22.72	12.60	9.49 ^b		5.15	6.69		15.81
Cost (\$/Ci)	255	766	103	77.69 ^b		42.37	54.79		129

^a7.5% interest on capital money in all cases

^bOption C-I - poured concrete monolith in clay matrix; no surveillance assumed.

^cOption C-II - poured concrete monolith in clay matrix; 150-year surveillance assumed.

^dOption C-III - drummed concrete emplaced in concrete monolith in clay matrix; 150-year surveillance assumed.

^eCombined electrolysis catalytic exchange process on total water recycle.

^fCombined electrolysis catalytic exchange process on excess (discharge) water only.

LMFBR fuels respectively. These values represent syntheses of costs from Tables 5 and 6 with those from Tables 9 through 11.

Examination of Table 12 indicates that the retention cases (LWR-IVA, LWR-IVB, and LWR-III) that provide the lowest concentrations of tritium in the plant recycle waters fare poorly in these cost comparisons. Cases LWR-IVA and LWR-IVB are the most expensive of the lot, and they remain so in all combinations with the disposal options other than long-term storage. Voloxidation (LWR-III), which also provides recycle water containing relatively little tritium, is also among the more expensive. Retention cases LWR-VA and LWR-VB, which provide recycle water at 182 Ci/m^3 , are by comparison less expensive. However, the least expensive of these (LWR-VA) can retain and immobilize tritium for less than \$6/Ci only by use of deep-well disposal and for \$7 or less per curie only by use of hydrofracture disposal, or (if feasible) by disposal as poured concrete monoliths.

First cycle confinement (case LWR-VI), which provides the most concentrated tritiated water as recycle, is also reasonably economical. However, the volume of tritiated water for disposal is relatively large, and its disposal in deep wells or by hydrofracture is required if the combined cost is to be below \$6/Ci or \$7/Ci respectively.

It is clear from consideration of Table 13 that if disposal to deep wells (perhaps unlikely) or by hydrofracture (likely) can be employed, the water recycle (case LWR-II) is the most economical case and that it remains so (but with little margin) should poured concrete monoliths in a clay matrix be employed for disposal. With its large volume of tritiated water for disposal, the water recycle retention cases (LWR-II and LMFBR-II) could

not compete if expensive disposal options (tank storage, geologic disposal, or sea-bed disposal) were mandated. Use of the water recycle case and of case LWR-VA or LWR-VB requires design to safely accommodate recycle water with 182 Ci/m³ of tritium.

Examination of Table 13 reveals the same general pattern, but with the important exception that the costs in dollars per curie of tritium confined are quite (and probably prohibitively) high. Indeed, only the water recycle (case LMFBR-II), when combined with deep-well disposal or hydrofracture, seems likely to result in costs below about \$40/Ci retained.

4. ENVIRONMENTAL IMPACT OF RELEASED TRITIUM

Tritiated water vapor released to the atmosphere through the stacks of reprocessing plants is dispersed in a manner determined first by the local weather and meteorological conditions and subsequently by those conditions of larger areas. Tritiated water is removed from the atmosphere by deposition and (primarily) by precipitation processes. The deposited tritium flows with surface waters and is subject to reevaporation and to run-off to the ocean. The tritiated water mixes with the waters of its hemisphere of origin and remains a part of the hydrological cycle until it decays. The tritium, accordingly, contributes to the radiological dose commitments to individuals, to the population near the plant, and to the population of the much larger area over which the released tritium is dispersed. As a component of the hydrological cycle, it also contributes (at very low dose rates to the average individual) to dose commitments to the population of the world during its decay.

Tritium Dosimetry

Tritium decays with a half-life of 12.33 years by emission of an electron⁶⁵(maximum energy 18 keV and average energy 5.88 keV) to form helium. External doses to man are inconsequential because of the low electron energy. However, tritium in the atmosphere and in surface waters equilibrates rapidly with the biosphere and, as a consequence, can affect man by several pathways.

Tritium inhaled as water vapor is essentially completely retained, and tritiated water vapor in air is readily absorbed through the skin. Tritium

is ingested in drinking water, as the water content of foods, and as organically bound tritium in foods. Tritium taken into the body seems to be nearly uniformly distributed throughout the body water.

Tritium is eliminated rapidly from the human body. A recent publication⁶⁶ assumes a three-compartment model of the human body for which the effective biological half-time for tritium elimination is 9.7 d. Several publications^{65,67-69} assume a biological half-time of 10 d; that value is used in the dose estimates presented here. A single uptake of 1 Ci of tritium uniformly distributed within the 52 kg of wet tissues of a 70-kg man would deliver an initial dose rate of 6.69×10^{-11} rad/s. If this tritium were eliminated with a biological half-life of 10 d, the integral dose delivered by the 1 Ci would be 8.3×10^{-5} rad. It should be noted that the specific activity of tritium is relatively high; approximately 10^4 curies per gram. Thus one Ci would be contained in 10^{-10} grams of tritium.

Dose commitments from the tritium released were estimated, using quite conservative assumptions, for four groups of people. These included: (1) individuals who could theoretically receive the maximum exposure (the individual who lived at the facility fence), (2) the average "local" individual who lived within 50 miles of the facility, (3) the average "regional" individual who lived in the United States outside the 50-mile range, and (4) global exposures for the rest of the world.

Estimated exposures for the three groups of personnel residing in the United States are summarized in Table 14. These doses are expressed as percentages of exposure that an individual would receive from the natural background radiation in the United States. As can be seen, only the LWR

Table 14. Theoretical dose commitments to individuals as a result of tritium releases
(Stated as percent of natural background radiation)

Tritium retention option	Maximally exposed hypothetical individual (% of background)	Average of local personnel - (50-mile radius; % of background)	Average of regional personnel - (outside 50 miles; % of background)
LWR			
Base	20	0.9	0.02
90% recycle	2.5	0.1	0.002
Voloxidation or isotope separation	1	0.04	0.0008
Head-end isolation	0.3	0.02	0.0004
LMFBR			
Base	3	0.1	0.003
90% recycle	0.3	0.02	0.0003
Voloxidation or isotope separation	0.2	0.01	0.0002
Head-end isolation	0.1	0.006	0.0001

and LMFBR base-case options (I) and the LWR recycle option (II) exceed 1% of natural background for the maximum theoretical fence individual. None of the local or regional doses is an appreciable percentage of natural background.

It is apparent that the LWR base-case plant (with by far the largest tritium release rate) might deliver (with the conservative assumptions used) an annual dose commitment of 20% of natural background or 24 millirem to the maximally exposed hypothetical individual. This maximum dose is below (but not comfortably below) the present Environmental Protection Agency (EPA) requirement that individuals receive less than 25 millirem/year from all portions of the nuclear fuel cycle. It is possible that such dose commitments to individuals may require that not all the tritium be released from the base-case LWR plant or that increasing the stack height and distance to the site boundary would reduce this dose to a lower value. Also site specific values may reduce the calculated dose relative to the conservative assumptions used.

It is, however, apparent that none of the other options appears to offer a significant threat to the maximally exposed individual.

It must be emphasized that even the largest of these population dose commitments are minuscule in comparison with the dose commitments from exposure to the natural background radiation. The local (50-mile) population would receive (in 20 years) a total body dose commitment of 2.9×10^6 man-rem from natural background compared with 2.6×10^4 man-rem from LWR base-case tritium; corresponding figures for the large regional (3.1×10^8) population are 1.35×10^5 man-rem from LWR base-case tritium and 8.0×10^8 man-rem from a natural background.

It is not apparent that a compelling argument can be made for retention of tritium during LWR fuel reprocessing for any personnel other than the "fence" individual, and then only if the very conservative assumptions are used. Retention of LMFBR tritium will prove more expensive and appears to be even less urgent.

5. SUMMARY AND CONCLUSIONS

All of the retention options markedly reduce the emission of tritium to the atmosphere relative to the base case. For each of the options, the water volumes, tritium quantities, and tritium concentrations dispersed to the atmosphere and retained for disposal, as well as tritium concentrations in recycle acid, recycle water, and vessel off-gas, are summarized in Table 15. The incremental costs for each of the retention options have been described in Section 3, and they are summarized in Tables 13 and 14. The results summarized in the three tables have been used to determine the incremental cost per curie of tritium retained relative to the base-case for the retention and disposal options. The results are displayed in Figures 16 and 17 for the LWR and LMFBF options, respectively. Several conclusions can be drawn from these figures:

- o Long-term tank storage is always the most expensive disposal option; the longer the storage, the higher the cost.
- o Voloxidizer costs are relatively independent of disposal methods.
- o Monolith, hydrofracture, and deep-well disposal of the tritium concentrate are essentially equal for all cases involving concentration of tritium (options III, IV, and V).
- o Sea-bed disposal is 10 to 100% more expensive than any of the items just above.
- o Deep-well and hydrofracture disposal costs are a small fraction of the retention costs.
- o LMFBF dollars per curie are always much higher than LWR dollars per curie for the same options because LMFBF fuel contains markedly less tritium.

ALARA

ORNL DWG. 81-10552R4

Fig 16. Cost Summaries LWR

Fig 17 Cost Summaries LMFBR

As a consequence of Appendix I to 10 CFR 50,² light-water cooled nuclear reactors are required to add equipment for control of radioactive emissions if the resulting decrease is at an annualized cost of \$1000 or less per total body man-rem and \$2000 or less per man-thyroid rem. Thus the maximum value considered would be \$2000 per man-rem. Release of one curie of tritium from a processing plant is conservatively estimated to deliver a dose commitment of 1.6×10^{-3} man-rem to the total body and an equal quantity to the thyroid of the 50-mile population. If this criteria is also applied to reprocessing plants, it might be argued that control of tritium could be considered cost effective if it could be achieved for \$3.20 per curie or less. This value is marked as ALARA in Figures 16 and 17. On this basis only the 90% water recycle option (II) combined with deep-well or hydrofracture disposal would be cost effective.

Two other factors that must be considered in evaluating the need for tritium collection and retention are (1) the effect of release on the world tritium inventory and (2) the effect of recycling high concentrations of tritium in the reprocessing plant.

The total tritium available for release from a 5 t/d LWR reprocessing plant is 8×10^5 Ci/year, which is 20% of the estimated four megacuries that are generated worldwide by natural processes each year. An LMFBR reprocessing facility would have about one-seventh of this quantity available for release; however, as implied in Table 14, this quantity could increase the natural background radiation by only very small fractions of a percent on a world-wide basis.

Table 15. Summary of water volumes and tritium concentrations for various retention options

Disposition	Retention options					
	I	II	III	IV	V	VI
LWR						
Dispersed through stack						
Tritium, Ci/d	2700	349	127	127	127	60
Retained for disposal						
Water, m ³ /d	0	12.94	0.048	3	0.288	8.86
Tritium, Ci/d	0	2351	2573	2573	2573	2640
Tritium, Ci/m ³	0	182	53 600	858	8930	350
Recycle acid						
Tritium, Ci/m ³	21.2	182	7.30	22.1	182	390
Recycle water						
Tritium, Ci/m ³	0	182	7.38	1.30	182	300
LMFBR						
Dispersed through stack						
Tritium, Ci/d	425	53	28	28	28	17
Retained for disposal						
Water, m ³ /d	0	27.3	0.048	3	0.288	8.78
Water, Ci/d	0	372	397	397	397	408
Tritium, Ci/m ³	0	13.6	8270	132	1378	46.4
Recycle acid						
Tritium, Ci/m ³	1.6	14.0	1.3	1.7	14.0	47.0
Recycle water						
Tritium, Ci/m ³	0	14.0	1.3	0.13	14.0	46.4

The concentration of tritium in the recycle acid and water for options II, V, and VI is more than an order-of-magnitude higher than for the base-case plants. However, this increased tritium concentration would require no additional radiation protection measures beyond those required because of trace fission products in the recycle water and acid.

In summary, there appears to be no compelling benefit of tritium retention and disposal for LMFBR reprocessing. The need for tritium retention and disposal for LWR fuel is marginal, except for conservative estimates for the fence exposure conditions.

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