MASTER

**Correlation of Radioactive Waste
Treatment Costs and the Environmental
Impact of Waste Effluents in the
Nuclear Fuel Cycle—Reprocessing
Light-Water Reactor Fuel**

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CORRELATION OF RADIOACTIVE WASTE TREATMENT COSTS AND THE
ENVIRONMENTAL IMPACT OF WASTE EFFLUENTS IN THE NUCLEAR FUEL
CYCLE - REPROCESSING LIGHT-WATER REACTOR FUEL

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CORRELATION OF RADIOACTIVE WASTE TREATMENT COSTS AND THE ENVIRONMENTAL IMPACT OF WASTE EFFLUENTS IN THE NUCLEAR FUEL CYCLE - REPROCESSING LIGHT-WATER REACTOR FUEL

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ABSTRACT

A cost/benefit study was made to determine the cost and effectiveness of radioactive waste (radwaste) treatment systems for decreasing the release of radioactive materials from a model nuclear fuel reprocessing plant which processes light-water reactor (LWR) fuels, and to determine the radiological impact (dose commitment) of the released materials on the environment. The study is designed to assist in defining the term "as low as reasonably achievable" in relation to limiting the release of radioactive materials from nuclear facilities. The base case model plant is representative of current plant technology and has an annual capacity of 1500 metric tons of LWR fuel. Additional radwaste treatment systems are added to the base case plant in a series of case studies to decrease the amounts of radioactive materials released and to reduce the radiological dose commitment to the population in the surrounding area. The cost for the added waste treatment operations and the corresponding dose commitments are calculated for each case. In the final analysis, radiological dose is plotted vs the annual cost for treatment of the radwastes. The status of the radwaste treatment methods used in the case studies is discussed. Much of the technology used in the advanced cases is in an early stage of development and is not suitable for immediate use. The methodology used in estimating the costs, and the radiological doses, detailed calculations, and tabulations are presented in Appendix A and ORNL-4992. This report is a revision of the original study (ORNL/TM-4901).

1.0 SUMMARY AND CONCLUSIONS

A study was made to determine the cost and effectiveness of radioactive (radwaste) treatment systems for decreasing the release of radioactive materials from a model nuclear fuel reprocessing plant, and to determine the radiological impact (50-year dose commitment) of the released materials on the environment. The model plant is representative of current plants that are in operation or are under construction and has an annual capacity for reprocessing 1500 metric tons of fuel. The fuel is irradiated to 33,000 MWd/metric tons at 30 kW/kg and cooled 160 days before reprocessing. The gaseous radwaste effluents are treated and released. All of the radioactive ^{14}C , ^{85}Kr , and tritium (in water vapor) are released in the gaseous effluent. The

high-level liquid radwastes and the miscellaneous low-level liquid radwastes are stored in tanks. No radioactive liquid waste is released to the environment. The products are shipped as liquid solutions of plutonium and uranium nitrates. The options for solidification of the wastes and the plutonium product and the conversion of the uranium product to uranium hexafluoride can be added to the base plant. However, these options are not included in the cost assigned to the base Case 1 or in the succeeding cases. The radiological impact of the model plant is evaluated at a midwestern and at a southeastern coastal site.

Several conceptual cases and their corresponding flowsheets were prepared for treating the liquid and gaseous effluents from the model plant. Case 1 is the base case, representing the lowest cost and current treatment technology. In each succeeding case, equipment is added to accomplish a specific objective. Cases 2 through 6 represent the use of advanced technology that might be applied (Sect. 4.3) in the future to plants which are constructed using the present design concepts for handling process gases and liquids. Case 7 represents an advanced design concept in which gaseous and liquid effluents are treated and recycled. Implementation of the concepts contained in Case 7 would extend over a period of about 2 decades and would require the design and construction of a new plant following the successful demonstration of the concepts in a pilot plant.

Advanced cases are in the early stage of development and are not suitable for immediate use in existing plants. The costs for each case are based on the preconstruction design of a new plant. Some of the equipment listed in Cases 2 through 6 can probably be backfitted to existing plants, but this must be considered on an individual basis. No attempt is made in this survey to determine backfitting costs for existing plants.

The costs are in early 1973 dollars, and the effect of inflation has not been considered; however, it is estimated that the 1975 costs would be about 30% higher. The efficiency of a treatment system or plant for retention of radioactive material is expressed as a decontamination factor (DF), (i.e., the ratio of the amount of material entering a plant to that released to the environment). Table 4.3 presents a summary of the general plan of the study, including the objectives, DFs, and treatment systems selected for each case.

The annual amounts of radioactive materials released (the source term), the capital and annual costs for radwaste treatment, the unit fuel reprocessing costs for radwaste treatment (\$/kg of uranium), the contribution to power costs (mills/kWhr), and the radiological impact (dose commitment) to the environment are estimated for each case. The capital costs include the costs for equipment, equipment installation, cell structure space, and indirect costs. The annual costs include the amortized capital costs and the operating costs. The 50-year dose commitments for each case are estimated for: (1) the annual population total-body (person-rem) and organ (person-organ-rem) doses out to a distance of 55 miles from the stack ground zero; (2) the incremental annual population total-body dose (person-rem), gastrointestinal (GI) tract, bone, and thyroid doses (person-organ-rem); (3) the maximum annual individual (adult) total-body and organ doses (millirem) at 0.5 and 1.5 miles from the stack ground zero; and (4) the incremental maximum individual (adult) total-body dose (millirem) at 1.5 miles from the stack ground zero. Factors are provided to calculate doses at other distances.

The incremental doses (population and individual) represent the difference in dose between Case 1 and a given case or cases. The costs and doses are summarized in Tables 6.1, 7.4, 7.5, 7.6, and 8.1-8.4. Additional correlations are presented in Sect. 8.0. The costs are the estimated

amounts required for each case beyond that required for the base Case 1, which in effect, is the incremental cost. The difference in dose commitment at the two sites is the result of differences in meteorology and population distribution. Internal exposure to radiation through inhalation and ingestion of radionuclides accounts for about 78% of the total body dose to individuals and the population living around a nuclear fuel reprocessing plant. Conservative assumptions are used in estimating doses which tend to maximize the doses. For example, it is assumed that 100% of the food consumed by an individual is produced at his reference location (Sect. 7.0).

1.1 Total-Body Dose

The principal radionuclides that contribute to total-body dose are tritium (^3H), carbon-14 (^{14}C), iodine (^{129}I), cesium ($^{134,137}\text{Cs}$), and krypton (^{85}Kr). The contributions to the population total-body dose within a 55-mile radius of the midwestern site in the Case 1 study are ^3H (55.7%), ^{14}C (17.2%), ^{129}I (6.5%), $^{134,137}\text{Cs}$ (9.9%), and ^{85}Kr (4.6%); at the east coastal site they are ^3H (52.5%), ^{14}C (16.2%), ^{129}I (8%), $^{134,137}\text{Cs}$ (12%), and ^{85}Kr (3.9%). The same radionuclides are the major contributors to the individual total-body dose at 1.5 miles, and the contributions are ^3H (52.3%), ^{14}C (16.2%), ^{129}I (7.7%), $^{134,137}\text{Cs}$ (12.7%), and ^{85}Kr (3.3%). The estimated annual population total-body doses for Case 1 are 1040 person-rem at the midwestern site and 299 person-rem at the east coastal site; at the two plants, the respective total-body doses are 14 millirem and 11 millirem at 1.5 miles. Only small reductions in the annual population and maximum individual total-body doses are achieved (7 to 15%) by decreasing the release of $^{129,131}\text{I}$, $^{103,106}\text{Ru}$, and ^{14}C in Cases 2a, 2b, 2c, and 5, since these radionuclides contribute a relatively small fraction of the total-body dose. Dose reductions of 13 to 58% are achieved in Cases 3, 4, and 6 where the amounts of ^{14}C and ^{85}Kr released are reduced by a factor of 100, particulates and $^{103,106}\text{Ru}$ by a factor of 10, and ^3H by a factor of 100.

The cost/benefit ratios of annual cost for radwaste treatment to the incremental reduction in annual population and incremental reduction in maximum individual total-body doses are listed in Tables 8.1 and 8.2 for the midwestern and east-coastal sites, respectively. The cost/benefit ratios vary from \$0.45 thousand/person-rem for Case 2a, to \$7.30 thousand/person-rem for Case 3, and \$43.71 thousand/person-rem for Case 7 for the midwestern site. The cost/benefit ratios at the east-coastal site for the same case studies are \$1.30 thousand/person-rem, \$27.25 thousand/person-rem, and \$152.17 thousand/person-rem. The maximum individual total-body incremental dose reduction cost/benefit ratios at 1.5 miles from the stack varies from \$26.32 thousand/millirem to \$3,241 thousand/millirem at the midwestern site and \$33.98 thousand/millirem to \$4,182 thousand/millirem at the east-coastal site for case studies 2a and 7, respectively. The most efficient system in terms of cost/benefit ratio (\$10³/person-rem) (i.e., the lowest cost/benefit ratio) is the ion exchange system for treating liquid wastes. The cost/benefit ratios increase in the following order: Case 2a (iodine retention), Case 5 (semivolatiles and ^{14}C retention), Case 6 (tritium and iodine retention), Case 4 (particulate and semivolatile retention), Case 3 (krypton and ^{14}C retention), Case 2b (iodine retention), Case 2c (iodine retention), and Case 7 (improved retention of all radionuclides).

The cost/benefit ratios for combining case studies in the order of decreasing efficiency (increasing annual cost/incremental reduction in annual population total-body dose) is presented in Table 8.3 for the midwestern site and in Table 8.4 for the east-coastal site, and graphically in Figs. 8.1 and 8.2. The most efficient combination of cases in terms of cost/benefit ratios (\$/person-rem) occurs by adding Cases 2a, 5, and 6 to the base case; this results in a reduction of annual population total-body dose from 1040 person-rem to 229 person-rem (78% of total) at the midwestern site, and from 299 person-rem to 71 person-rem (75% of total) at the east-coastal site for an annual cost of \$1,156,000. The addition of Cases 4, 3, 2b, and 2c results in a further reduction in total-body dose from 229 person-rem to 22 person-rem (20% of total) at the midwestern site, and from 71 person-rem to 7 person-rem at the east-coastal site at an additional cost of \$2,358,000.

1.2 Thyroid Dose

The annual cost of reducing the annual population thyroid dose out to a distance of 55 miles is shown in Tables 8.1-8.4, and graphically in Figs. 8.3 and 8.4, for Cases 1 through 7. Radioactive iodine contributes 93% of the dose to the thyroid. For Cases 2a, 2b, and 2c, all of the treatment cost can be assessed to reduction in thyroid dose, since these cases are specially designed to illustrate iodine retention. The addition of Case 2a decreases the thyroid dose from 12,286 person-thyroid-rem to 1237 person-rem (90% of total) at the midwestern site and from 4245 person-thyroid-rem to 375 person-thyroid-rem (91% of total) at the east coastal site. The addition of Cases 2b and 2c results in a small additional reduction in thyroid dose, 3% of the total at the midwestern and east-coastal sites.

The primary purpose of Case 6 is for tritium retention through use of the voloxidation process. However, some additional retention of iodine also occurs which results in a reduction of thyroid dose. Similarly, small reductions in thyroid dose occur in Cases 3, 4, 5, and 7. The total costs for these cases are assessed to the primary purpose of each case, and no effort is made to define the fractions of the total cost of these cases that apply to retention of iodine or reduction in thyroid dose.

The cost/benefit ratios for combining case studies in the order of decreasing efficiency (increasing annual cost/incremental reduction in thyroid dose) are presented graphically in Figs. 8.3 and 8.4 for the midwestern and east-coastal sites. The cost/benefit ratio for Case 2a is \$3.17/person-rem at the midwestern site and \$9.04/person-rem at the east-coastal site at an annual cost of \$35,000. The addition of Cases 5, 6, and 4 results in a cost/benefit ratio of \$1620/person-rem at the midwestern site and \$5540/person-rem at the east-coastal site at an additional annual cost of \$1,424,000. The cost/benefit ratio for the addition of Cases 3, 2b, and 2c (equivalent to composite Case 6c) is \$6740/person-rem at the midwestern site and \$24,760/person-rem at the east-coastal site at an additional annual cost of \$2,055,000.

1.3 GI-Tract Dose

The principal radionuclides that contribute to the GI tract are ^{106}Ru (64.4% of total) and ^3H (21% of total). The most effective treatment cases for the reduction of GI-tract dose are Case 4 (particulate and semivolatile retention), Case 5 (^{14}C and semivolatile retention), Case 6 (tritium and iodine retention), cumulative Case 6c (retention of all the radionuclides), and advanced Case 7 (further retention of radionuclides). The annual costs of reducing the annual population GI-tract dose are shown in Tables 8.1 and 8.3 for the midwestern site. The annual population GI-tract dose decreases from 1704 person-rem for Case 1 to 1043 person-rem for Case 4 (annual cost of \$303,000), to 1492 person-rem for Case 5 (annual cost of \$123,000), to 1090 person-rem for Case 6 (annual cost of \$998,000), to 79 person-rem for Case 6c (annual cost of \$3,514,000), and to 0.89 person-rem for Case 7 (annual cost of \$45,500,000). The most efficient system as measured in terms of ratio (cost/benefit) of annual cost to incremental reduction in GI-tract dose is Case 5. These ratios in annual cost per person-rem are \$203 for Case 4, \$83 for Case 5, \$1717 for Case 6, \$1470 for Case 6c, and \$18,747 for Case 7. The same pattern is obtained for these values at the east-coastal site.

1.4 Bone Dose

The major contributors to the bone dose are ^3H (28.4% of total), ^{90}Sr (17.4% of total), and ^{14}C (10.8% of total). The most effective treatment cases for reducing the bone dose are Case 6 (tritium and iodine retention), Case 4 (particulate and semivolatile retention), Case 3 (krypton and ^{14}C retention), Case 5 (semivolatile and ^{14}C retention), cumulative Case 6c (retention of all the radionuclides), and advanced Case 7 (further retention of all the radionuclides). The annual costs of reducing the bone dose are shown in Tables 8.1 and 8.3 for the midwestern site.

The annual population bone dose decreases from 1704 person-rem for Case 1 to 1090 person-rem for Case 6 (annual cost of \$998,000), to 1043 person-rem for Case 4 (annual cost of \$303,000), to 1328 person-rem for Case 3 (annual cost of \$1,635,000), to 1492 person-rem for Case 5 (annual cost of \$123,000), to 79 person-rem for Case 6c (annual cost of \$3,514,000), and to 0.89 person-rem for Case 7. The most efficient system as measured in terms of ratio (cost/benefit) of annual cost to incremental reduction in bone dose is Case 4. These ratios in annual cost per person-rem are \$1625/person-rem for Case 6, \$458/person-rem for Case 4, \$5923/person-rem for Case 3, \$580/person-rem for Case 5, \$2162/person-rem for Case 6c, and \$26,717/person-rem for Case 7. The same pattern is obtained for these values at the east-coastal site.

1.5 Cost Comparison

Radwaste treatment costs can be compared with fuel reprocessing costs, value of the spent fuel, power costs, and total capital investment in establishing the relative importance of alternative cost factors in a cost-benefit analysis. A comparison of the cost of reprocessing fuel

with the value of the fuel is particularly important, since the incentive for reprocessing fuel decreases as the cost of reprocessing approaches the value of the fuel.

The value of the spent fuel is estimated as \$75.35 per kilogram of contained uranium plus plutonium for PWR fuel and \$60.37 per kilogram of uranium plus plutonium for BWR fuel (Sect. 6.4, Table 6.4). The annual reprocessing cost in the base Case 1 is \$30.33 per kilogram of U (Table 6.1). In this case, the difference between the value of the spent fuel and the reprocessing cost is about \$45 per kilogram of uranium plus plutonium for PWR fuel and \$30 per kilogram of uranium plus plutonium for BWR fuel. In Cases 2a through 6 the costs for additional radwaste treatment systems increase from about 0.03 to 9% of the incremental difference between the reprocessing cost and the value of the PWR fuel and by 0.04 to 1.1% for BWR fuel. In the cumulative Case 6c, the increased treatment cost is about 3% of the incremental difference for PWR fuel and 3.8% for BWR fuel. An increased treatment cost of \$30.33 per kilogram of uranium, as illustrated for Case 7, would approach the incremental difference between the value of the fuels and the reprocessing cost under the economic conditions present in 1973.

The capital costs for Cases 2a through 6 range from \$92,000 to \$4,491,000, or up to about 3.6% of the \$125,000,000 capital cost of the base plant (Sect. 3.4), and a total of about 8% of the base plant cost for the cumulative Case 6c. Case 7 represents a 100% increase in the capital cost of the base plant. The annual costs for Cases 2a through 6 range from \$35,000 to \$1,635,000, which is equivalent to contributions to power cost of 9.2×10^{-5} to 4.24×10^{-3} mill/kWhr, respectively (Sect. 6.0, Table 6.1). Cumulative Cases 6c and 7 amount to 9.14×10^{-3} and 1.2×10^{-1} mill/kWhr, respectively. All of these values are less than about 2% of an estimated total generation cost of 7 to 10 mills/kWhr.

2.0 INTRODUCTION

This study was performed to determine the cost and the effectiveness of radioactive waste treatment systems that are used, or could be used, at nuclear fuel reprocessing plants to decrease the amounts of radioactive and nonradioactive materials released to the environment. A second objective was to determine the impact of the radioactive releases on the environment. The effectiveness of the alternative radioactive waste treatment systems that are considered is measured by comparing the amounts of radioactive materials released by the various systems and the impact of these releases on the environment. The amount of radioactive materials released in each case is called the "source term," since these values are the source of initial numbers used in evaluating the impact of radioactive releases on the environment. The impact on the environment is assessed and compared with the radioactive waste treatment costs as the basis for a cost/benefit analysis. The radioactive materials are formed in the nuclear fuels by the nuclear processes that occur while the fuel is used at the nuclear power station to produce heat and electricity. The spent nuclear fuels are transported to fuel reprocessing plants in massive, heavily shielded carriers.

The purpose of fuel reprocessing is to purify and recover the valuable, unused uranium and plutonium for reuse in new nuclear fuels. The radioactive waste materials, fission products, and transuranium nuclides are separated from the uranium and plutonium and stored in liquid concentrates in underground tanks. Ultimately, they will be solidified, sealed in stainless steel storage containers, and, after an interim storage period, could be shipped to a Federal repository for permanent storage or disposal. The recovered uranium can be shipped as a nitrate solution to a conversion plant, or can be converted to uranyl hexafluoride onsite prior to shipment to a gaseous diffusion plant for enrichment. The enriched product is subsequently transferred to a fuel fabrication plant for reuse in new fuels. Similarly, the recovered plutonium can be shipped as a nitrate solution or can be converted to solid plutonium dioxide for transfer to a fuel fabrication plant for reuse. However, at present (1976), the supply of recycle plutonium is limited and the fabrication of recycle plutonium fuel occurs only on an experimental basis. A small fraction of the radioactive materials is volatilized as gases or aerosols during the reprocessing operations, and treatment systems are used to minimize the release of these materials in the gaseous effluent from the plant. Liquid effluent treatment systems are used to prevent the release of any radioactive liquids from the plant.

A model plant that is typical of current designs for reprocessing plants is used as the base case for this study. However, the model plant does not represent the design for any particular existing facility. The radiological impact of the plant is considered at two typical sites, specifically, a midwestern and a southeastern coastal plain. Increasingly efficient radioactive waste treatment systems are added to the "base" plant, and the annual cost and environmental impact of each case is calculated as the basis for cost and benefit analysis. It was not feasible to include all possible variations of base plants and radioactive waste treatment systems, but sufficient information is provided in this study so that the costs and impacts can be estimated for other radioactive waste treatment systems by extrapolation or interpolation from the data provided. The base case illustrates the important features of current plants. The advanced cases use technology ranging from that which is being considered for installation in the near future

to the foreseeable limits of available technology based on expected typical operations over the next 30 years.

Some of the technology used in the advanced cases is in an early stage of development and is not suitable for immediate use in existing plants. However, it is necessary to use this technology in the study to predict cost/benefit relationships over the next few decades. In most cases, alternative technology to accomplish a given objective is not fully developed. In an earlier report¹ it was estimated that these processes could be "reduced to practice" by about 1983, if development work had been started in 1974 and a "hot" pilot plant or full-size plant demonstration were started about 1978. The development of these processes has not proceeded as rapidly as anticipated in 1974; consequently, estimates of dates "for reduction to practice" are not given since such estimates are beyond the scope of this report.

The original study, as represented by this survey, was completed in late 1973, and the report was issued in May 1975. The environmental impact and cost/benefit analyses were based on storing high-level liquid waste in tanks, the production of uranium and plutonium nitrates as liquid products, and the use of early 1973 costs for equipment. The purpose of the current revision of the original report¹ is to upgrade both the engineering assessment and the radiological impact based on current information (early 1976). Major revisions occurred in: (1) the selection of case studies and the calculation of source terms for iodine; (2) the plant decontamination factors achieved for semivolatiles (ruthenium); (3) the addition of ¹⁴C to the source term; and (4) the use of revised methods for calculating the dose from tritium, ¹⁴C, ⁸⁵Kr, and other minor changes in methodology. A complete description of the methodology used for calculating doses is given in ref. 2. The use of early 1973 dollars is retained so that this report will be consistent with the other reports in this series. It is estimated that the costs would be about 30% higher in 1975.

This report is one of a series of studies on the nuclear fuel cycle. Other reports in this series are concerned with milling uranium ores,^{3,4} fabricating LWR fuels containing enriched uranium,⁵ fabricating LWR fuels containing plutonium,⁶ fabricating HTGR fuels,⁷ reprocessing LWR fuel,¹ and reprocessing HTGR fuels.⁸

2.1 References for Section 2

1. B. C. Finney, R. E. Blanco, R. G. Dahlman, F. G. Kitts, and J. P. Witherspoon, *Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing "As Low As Practicable" Guides - Nuclear Fuel Reprocessing*, ORNL/TM-4901 (May 1975).
2. G. G. Killough and L. R. McKay (eds.), *A Methodology for Calculating Radiation Doses from Radioactivity Released to the Environment*, ORNL-4992 (March 1976); this report serves as Appendix B for this and the following reports.
3. M. B. Sears, R. E. Blanco, R. C. Dahlman, G. S. Hill, A. D. Ryon, and J. P. Witherspoon, *Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing "As Low As*

Practicable" Guides - Milling of Uranium Ores, ORNL/TM-4903, Vol. 1 (May 1975).

4. A. D. Ryon and R. E. Blanco, *Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing "As Low As Practicable" Guides - Appendix A. Preparation of Cost Estimates for Volume 1, Milling of Uranium Ores, ORNL/TM-4903, Vol. 2 (May 1975).*
5. W. H. Pechin, R. E. Blanco, R. C. Dahlman, B. C. Finney, R. B. Lindauer, and J. P. Witherspoon, *Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing "As Low As Practicable" Guides - Fabrication of Light-Water Reactor Fuel from Enriched Uranium Dioxide, ORNL/TM-4902 (May 1975).*
6. W. S. Groenier, R. E. Blanco, R. C. Dahlman, B. C. Finney, A. H. Kibbey, and J. P. Witherspoon, *Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing "As Low As Practicable" Guides - Fabrication of Light-Water Reactor Fuels Containing Plutonium, ORNL/TM-4904 (May 1975).*
7. J. W. Roddy, R. E. Blanco, G. S. Hill, R. E. Moore, R. D. Seagren, and J. P. Witherspoon, *Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle - Fabrication of High-Temperature Gas-Cooled Reactor Fuel Containing Uranium-233 and Thorium, ORNL/NUREG/TM-5 (September 1976).*
8. W. Davis, Jr., R. E. Blanco, B. C. Finney, G. S. Hill, R. E. Moore, and J. P. Witherspoon, *Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle - Reprocessing of High-Temperature Gas-Cooled Reactor Fuel Containing U-233 and Thorium, ORNL/NUREG/TM-4 (May 1976).*

3.0 OBJECTIVES AND ASSUMPTIONS

3.1 Objectives

The objectives of this study are: (1) to determine the cost (in dollars) that would be required to reduce the amount of radioactive materials released to the environment from plants using current treatment systems, to very low levels using advanced, complex treatment systems; and (2) to evaluate the radiological impact of the radioactive effluents released from these conceptual installations. The definition of the incremental value of additional radioactive waste treatment equipment is an important part of the basic objective and is emphasized in the study. Generally, these values will not change with the size of the plant. For example, the amount of waste effluent to be treated generally increases with the plant size and, thus, larger treatment systems are required. However, the fraction released is essentially the same for large and small systems; therefore, a larger total amount of radioactive material is released for the larger unit when operating on the same type, but larger volume, of radioactive effluent. The calculated total amounts of radioactive materials released are also defined but are less important in this study, since they are expected to vary with the plant size. Hence, the incremental and absolute values derived in this study for a single size of conceptual plant can be extrapolated to larger or smaller plants. The volumes of radioactive wastes were selected on the assumption that a careful internal waste management program has been followed.

Estimates are made of the average radioactive and nonradioactive releases and the cost of radioactive waste treatment operations over the lifetime of the reprocessing plant. In a similar study for nuclear power reactors,¹ great emphasis was placed on maintaining continuous operation of the power plant. Consequently, the more complex radioactive waste treatment systems contained redundant (parallel) treatment units to ensure continued operation in case one of the units should become inoperable. In the reprocessing study, less emphasis is placed on continuous operation since the plant could temporarily cease operations in the event that a major radioactive waste treatment unit failed. Only potential releases from normal operations, including anticipated operational occurrences, have been considered in this study.

3.2 Selection of the Model Plant

The model plant selected for the base case (Case 1) is similar to plants being designed or licensed in 1973 and is representative of the plants that will reprocess the major load of fuel throughout the next 2 decades.²⁻⁴ The plant will process 1500 metric tons of fuel per year (calculated as 1500 metric tons of uranium charged to the reactor) irradiated at an average specific power of 30 kW/kg to an exposure of 33,000 MWd/metric ton and decayed 160 days. A mechanical shear and nitric acid leach system is used to segment and dissolve the fuel in nitric acid. The fuel is then purified and recovered by solvent extraction using the Purex process. The high-level liquid radioactive wastes are stored in tanks for 5 years, and the miscellaneous low-level liquid radioactive wastes are stored for an indefinite period. No radioactive liquid wastes are released to the environment. The base case can also include the

options of solidification of the high-level liquid waste, conversion of the uranium product to UF_6 , and the conversion of the plutonium product to a solid form (i.e., PuO_2). These options can be included with the base-case plant and are designed so that no significant increase in the release of radioactive materials occurs relative to the base case.

3.3 Management of Radioactive Wastes

The most complex flowsheets in this study illustrate very low, but not "zero," release of radionuclides in the gaseous radioactive waste effluents. Radioactive liquid effluents are not released from the plant.

Gaseous effluents. Gaseous effluents from process vessels and the ventilation air from cell areas contain radioactive gases (iodine, noble gases, ^{14}C compounds, tritiated water vapor) and particulates or aerosols that contain a spectrum of all of the radionuclides in the process equipment. The gases are treated so that increasingly larger fractions of these materials are retained in the various case studies. The treated gases are released through a 100-m-high stack.

Liquid effluents. Liquid effluents are treated so that the dissolved radioactive and nonradioactive solids and a fraction of the water are retained in the plant. Excess water is not released as a liquid but is vaporized and released with the gaseous wastes. At equilibrium and in the absence of any removal mechanism, the amount of tritium leaving the plant in the water vapor will equal the amount entering the plant in the fuel.

The concentrated high-activity radwastes are stored in tanks for 5 years as permitted by government regulation.⁵ These wastes contain >99% of the radioactive materials entering the plant. Optionally, these wastes can be solidified and stored for a period of 10 years after their formation in the reprocessing operation, and subsequently shipped to a government repository. The concentrated miscellaneous low-level liquid radwastes are stored in tanks for an indefinite period. These wastes contain <1% of the radioactive materials and the bulk of the soluble nonradioactive materials. Optionally, these wastes can be solidified as indicated above for the high-level wastes.

Solid wastes. Solid wastes consisting of segments of fuel cladding, rags, clothing, floor sweepings, etc., are packaged in steel drums for burial in a licensed burial ground.

3.4 Cost Parameters

A base case is selected which is similar to plants being designed or licensed in 1973. The capital and annual costs are then estimated for waste effluent treatment segments added to the base case in a series of case studies. The calculation of these incremental annual costs is a primary objective of the study. They are correlated with the changes in environmental impact for each case study in Sect. 8.0. The estimated costs are based on a new plant design using remote maintenance in the highly radioactive head-end operations, and direct maintenance for the less radioactive sections of the plant. No attempt is made to estimate backfitting costs for present plants that are either being constructed or modified. The costs are based on add-ons to a new plant design prior to the beginning of construction.

The capital cost of the base plant was set at \$125 million in 1973. This does not include the options of solidification of the high-level radioactive liquid waste, the formation of a solid plutonium product (PuO_2), the conversion of uranium product to UF_6 , and the addition of safeguard systems that have subsequently been required. This is not a precise value since it will vary considerably with the type of facility constructed [i.e., direct maintenance, semiremote maintenance, or completely remote maintenance (canyon type)]. This cost is used for a qualitative comparison with the incremental capital costs of the cases studied. Complete details of the cost estimating procedure are given in Sect. 6.0.

3.5 Equipment Operation

It is assumed that all radioactive wastes will be treated by the radioactive waste equipment; that is, wastes will not bypass treatment systems and will be discharged even though the radioactive content of the waste is lower than "permissible" licensing levels. The equipment is adequately sized to ensure high operating flexibility and efficiency factors. For example, if the radioactive liquid waste is not decontaminated to the desired degree in a single evaporation, it may be recycled and reevaporated. This type of design provides extra assurance that radioactive releases will not exceed the calculated design levels.

3.6 Plant Siting

A model plant is located at each of two sites having environments that are characteristic of contemporary nuclear fuel reprocessing and fuel fabrication facilities. Site 1 is situated on a plain in a rural southeastern coastal area adjacent to a continuously flowing stream that empties into an estuary. Cities with moderate populations are established a short distance from the site. Site 2 is situated on a plain in a rural midwestern environment adjacent to a continuously flowing stream which empties into a large river. The survey area contains cities with moderate populations, as well as a large city. Meteorological data for Sites 1 and 2 are derived from first-order weather stations in the coastal southeastern and midwestern areas of the United States. The population distribution for the sites is determined by averaging the distributions around several nuclear installations in the southeastern and midwestern areas. Site selection is described in detail in Sect. 7.0.

3.7 Radiological Impact

Radiation doses to the population surrounding the model plant are estimated by using the procedures that are currently being used in environmental impact statements for light-water-cooled nuclear power reactor stations by the U.S. Nuclear Regulatory Commission.¹ Pathways for external radiation dose from sources outside the body and for internal dose from sources in the body are considered. Immersion in the gaseous effluents as they are diluted and dispersed leads to external exposure, and inhalation causes internal exposure. The deposition of

radioactive particulates on the land surface leads to direct external exposure and to internal exposure by the ingestion of food products through various food chains. Similarly, swimming in water containing radionuclides can lead to external exposure, whereas the harvesting of fish or drinking the water can lead to internal exposure. In this study, no radioactive materials are released in liquid effluents.

The estimated radiation doses to individuals and to the human population are calculated for annular distances out to 55 miles in 22.5° sectors using the site parameters listed in Sect. 3.6. Doses to individuals are calculated for the total body and individual organs. Population doses (person-rem) are the sum of the total body doses to all individuals in the population considered. Details of dose models, assumptions, and methods are given in Sect. 7.0.

3.8 References for Section 3

1. USAEC-Directorate of Regulatory Standards, *Final Environmental Statement Concerning Proposed Rule Making Action: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "as Low as Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents*, WASH-1258 (July 1973).
2. Nuclear Fuel Services, Inc., *Safety Analysis - Spent Fuel Reprocessing Plant*, Docket No. 50-201 (1962-1965).
3. Allied-Gulf Nuclear Services, *Barnwell Nuclear Fuel Plant - Environmental Report*, Docket No. 50-332 (November 1971).
4. General Electric Company, *Applicant's Environmental Report*, NEDO-14504 and NEDO-14504-2 (1971).
5. Code of Federal Regulations, Title 10, Part 50, Appendix F.

4.0 SOURCE TERM FOR RADIOACTIVE RELEASES

4.1 Origin of Radioactive Wastes in Reprocessing Plants

Nuclear reactor fuel elements must be replaced periodically as they begin to suffer from depletion of fissile fuel material, accumulation of fission products, and irradiation damage. Typically, the fuel is replaced after generating about 25,000 to 35,000 MWd of heat per metric ton of contained fuel. Each year, typical large power reactors discharge from 25 to 40 metric tons of spent fuel, contained in from 60 to 200 fuel elements. These spent fuel elements are sources of heat and intense radioactivity; the radioactive materials consist of fission products, hardware that has become radioactive from exposure to neutrons in the reactor, residual fuel material, and other elements such as americium and curium that are formed from neutron capture in the fuel materials. In this study, the model fuel is irradiated to 33,000 MWd/metric ton and the model plant will process 1500 metric tons of fuel annually, corresponding to the fuel from about 55 power reactors.

The function of a fuel reprocessing plant is to recover the residual fuel materials, uranium and plutonium, in a pure form suitable for recycle and to isolate radioactive wastes for storage or disposal. The spent fuel is transported from the reactor to the reprocessing plant in heavy, shielded casks by truck or rail after a normal period of storage at the reactor of 120 to 150 days, which allows for decay of greater than 95% of the sources of heat and radioactivity of the fuel. The cooling time before reprocessing is 160 days in this study.

A fuel reprocessing plant is typically a massive building which may be 200 ft wide by 115 ft high above the foundations by 300 ft long.¹ The main process operations are housed in a 3- to 5-1/2-ft-thick walled, heavily reinforced concrete structure which is approximately 60 ft wide by 175 ft long by 70 ft high, including the shielding wall thicknesses. The central structure is subdivided into smaller enclosures, called cells, that are isolated from each other by appropriate shielding and ventilation systems according to the radioactivity level, equipment maintenance system, and process function. A longitudinal cross section of a typical plant is shown in Fig. 4.1. A cross section of the plant building is shown in Fig. 4.2 and provides additional information on the arrangement of the process cells and supporting areas.

Fuel storage. The fuel is stored under 20 to 40 ft of water at the reprocessing plant. This storage area represents the first potential source of radioactive wastes where a defective fuel element could release small amounts of radioactive materials to the water or to the room ventilation system. Generally, there are provisions for canning and/or shrouding a defective fuel element to prevent the release of radioactive gases into the working area.

Fuel shearing and dissolution. The first step in fuel reprocessing is to shear the fuel element into approximately 1-in. pieces to expose the fuel for subsequent dissolution in nitric acid. This is accomplished by a hydraulic shear which exerts up to 320 tons of force on the shear blade. In practice, the sheared fuel may be collected in a stainless steel cylindrical basket for transfer to the fuel dissolver, or may be dropped directly into a basket in the dissolver. In the dissolver, the uranium, plutonium, and fission products are dissolved in nitric acid, leaving the hulls of zirconium fuel tube cladding as a residue in the basket. The basket and residual cladding hulls are withdrawn from the dissolver and examined to make sure that fuel

dissolution is complete. The waste residue is then placed in steel drums and transferred to the solid waste storage area. The fuel cladding hulls constitute a major fraction of the solid radioactive wastes at a reprocessing plant. The dissolver solution containing the uranium, plutonium, and fission products is transferred to the fuel recovery and purification system.

The dissolution system is a major source of radioactive effluents. Radioactive gases such as iodines, krypton, ^{14}C , tritium, and tritiated water vapor are released to the dissolver off-gas system. These gases also contain an aerosol of droplets of highly radioactive dissolver solution which, in turn, contains a complete spectrum of the radioactive materials in the solution. On drying, these droplets form radioactive particulates. Semivolatile elements such as ruthenium, which are partially volatilized from the dissolver, also constitute an important source of airborne particulate radioactive materials.

Fuel purification. The uranium and plutonium are recovered and purified by the solvent extraction process in which an immiscible solvent (tributyl phosphate dissolved in dodecane or kerosene) is vigorously mixed with the dissolver solution. The uranium and plutonium transfer into the solvent, and the fission products remain in the acidic waste water. The uranium and plutonium are then separated in a similar second step. The waste liquid containing the fission products is transferred to the waste treatment system. Extremely high separations are possible by this method such that less than one part in ten million of the fission products remains with the uranium or plutonium. The off-gas from the vessels in the extraction system contains an aerosol of radioactive liquid and organic vapors. The organic vapors can combine with iodines to form organic iodides that are more difficult to remove from the gaseous effluent than elemental iodine.

Fuel packaging. The purified uranium and plutonium are packaged in a shipping container and shipped to the fuel fabrication plant as aqueous solutions. In some reprocessing plants, the uranium may be converted to uranium hexafluoride for direct return to a uranium enrichment plant. Currently, commercial fuel reprocessors are considering the installation of facilities to convert the product solution of plutonium nitrate to solid plutonium oxide in order to facilitate shipment to a fuel fabrication plant. Methods considered for this conversion are (1) thermal denitration, (2) oxalate precipitation and calcination, and (3) preparation of sol-gel microspheres. All of these operations will be sources of airborne particulates. If the plutonium solidification step is added as an option to the model facility, additional scrubbers and HEPA (high efficiency) filters must be provided to ensure that the off-gas from this facility will not contribute a significant amount of radioactive material to the gaseous effluent from the separations plant.

Process cell ventilation. The process building is supplied with washed and conditioned air that is introduced into normal access zones at a positive pressure, flows to adjacent limited access zones, and then to restricted access zones. The flow is maintained in the direction of increasing contamination potential by providing progressively lower pressures with about 0.1- to 0.2-in. water pressure difference between zones. Radioactive solutions that may leak from process vessels and piping to the floor of the process cells can be vaporized and contribute radioactive materials to the ventilation air.

Treatment of liquid wastes. Modern reprocessing plants are designed to prevent the release of radioactive liquid effluent. Nonradioactive cooling water is discharged continuously

and, if radioactive materials should leak into the cooling water, the radiation detection and diversion system would divert the water to evaporators or to retention basins. These evaporators concentrate miscellaneous plant waters that contain low levels of radioactive materials. The miscellaneous wastes are neutralized with caustic prior to evaporation and the concentrate containing the radioactive materials is sent to a waste storage tank. The purified vapor is discharged up a 100-m-high stack. This vapor contains radioactive tritium as tritiated water. The highly radioactive acidic wastes from the solvent extraction system are concentrated by evaporation to decrease the volume to be stored and to recover the nitric acid for reuse.

Liquid waste storage. The concentrated liquid wastes from evaporators contain essentially all of the nonvolatile fission products from the spent fuel. It is general practice to store this liquid for an interim period. Wastes are stored as acidic solutions in stainless steel tanks. The need for constant surveillance and periodic replacement of tanks and equipment provides the incentive to convert these wastes into stable solid forms which can be handled and stored more safely and economically. Federal regulations require the solidification of the wastes within a 5-year period. One plant was designed for essentially immediate solidification of the wastes.²

Waste solidification. A system for solidification of wastes can be added to the model plant to conform with the licensing requirements³ for eventual solidification of high-level wastes. The system would be added in conjunction with the evaporation of the wastes such that the off-gas from the calcination unit could be scrubbed by the raffinate waste from the extraction system. The noncondensable vapors would pass into the off-gas treatment system and should not increase the net amount of radioactive material released from the plant in the gaseous effluent.

4.2 Composition and Amount of Radioactive Material Entering Model Plant

A list of the radionuclides selected as components of the source term for this study, along with their relative inhalation hazards, is presented in Table 4.1. The list was compiled from the ORIGEN⁴ computer code, which calculates the relative inhalation hazard for each nuclide by dividing the curies present in 1 metric ton of fuel at 160 days cooling by the Radiation Concentration Guide (RCG)⁵ for that nuclide. The criteria for selection of the nuclides to be used are as follows:

- (1) Gaseous nuclides tritium, krypton, ¹⁴C, ¹²⁹I, and ¹³¹I; actinide nuclides whose contribution to the actinide relative inhalation hazard is $\geq 0.02\%$ of the total ($1.01 \times 10^{17} \text{ m}^3$ of air at RCG); and fission products whose contribution to the relative inhalation hazard is $\geq 0.02\%$ of the total ($1.04 \times 10^{16} \text{ m}^3$ of air at RCG).
- (2) Uranium does not meet the above criteria but is included because of its importance in the fuel cycle.

Radionuclides that are excluded on this basis are examined to ensure that they would not contribute more than 0.02% of the total body dose for individuals in the Case 1 study (<0.01 millirem) as the result of bioaccumulation in the environment.

The amount of radioactive materials entering the plant is calculated with the ORIGEN⁴ computer code for a plant reprocessing 1500 metric tons of fuel per year of 3.3% enriched fuel irradiated to 33,000 MWd/metric ton at a specific power of 30 kW/kg and cooled 160 days (Table 4.2, Column 2).

4.3 Description of Waste Treatment Methods

Nuclear fuel reprocessing plant effluents are treated extensively by using a variety of unit process operations to minimize the release of radioactive or other noxious materials. Volatile, semivolatile, and particulate radioactive materials are removed from gaseous effluents by filters, adsorbers, absorbers, and scrubbers. The treated gaseous effluents are then released through a 100-m-high stack to achieve a high degree of dilution in the atmosphere. The primary methods for treating liquid radwastes are evaporation and ion exchange (demineralization). Treated liquid radwastes are not released but are vaporized and the water vapor is released through the 100-m stack. The sections that follow describe the radwaste treatment systems and, in addition, give an indication of the "state of the art" of each particular treatment method. The "state of the art" for the treatment methods varies from well-established technology that is used in present installations to advanced methods that are in the small engineering scale of development.

The advanced radwaste treatment systems used in this study include the iodine evolution, adsorption on resin beds, and the Iodex processes for retaining iodine, the voloxidation process for retaining tritium, and the selective absorption process for retaining krypton and ¹⁴C. Prior to 1976, these processes were not being developed at government laboratories for application to the reprocessing of LWR fuels. A program was initiated in 1976 at the Savannah River Laboratory (SRL) and ORNL to develop these processes in nonradioactive laboratory studies and small-scale studies with irradiated fuels in hot cells. No schedule has been developed as yet (early 1976) to predict the completion of these studies, nor for the demonstration of these processes in pilot or full-scale plants.

These processes are also being developed for reprocessing Liquid Metal Fast Breeder Reactor (LMFBR) fuels for demonstration in a "hot" pilot plant starting in 1986 and are expected to be "reduced to practice" in the following several years. It is expected that the work on LMFBR fuels will be applicable to processing LWR fuels. Thus, it is difficult to predict when these processes will be "reduced to practice." A process is considered to be "reduced to practice" when it has been successfully demonstrated in a "hot" pilot plant or a full-scale commercial plant. This definition implies that the design or operating objective (e.g., the retention of 99% of desired radioactive material) can be met in routine operations over long periods. Alternatively, an installation could be constructed and operated on a "best effort" basis so that lower removal efficiencies would be obtained, but without the assurance that the design value would ever be attained.

In an earlier report,⁶ we estimated that these processes could be "reduced to practice" by about 1983 if development work had been initiated in 1974 and a "hot" pilot plant or full-size plant demonstration were started about 1978. However, the development of these processes has not proceeded as rapidly as anticipated in 1974. As a result of these uncertainties, and, since the estimation of dates when these processes will be "reduced to practice" is beyond the scope

of this survey, revised estimates of dates for "reduction to practice" are not presented. The objective of this survey, as stated in Sect. 2.0, is limited to estimating the cost and radiological impact of adding more effective radwaste treatment systems to the base case reprocessing plant.

4.3.1 Filtration

High-efficiency particulate air (HEPA) filters. HEPA filters have been used for many years in the nuclear industry to remove radioactive particles from air streams. A standard HEPA filter has a cross section of 2 ft by 2 ft and a depth of 1 ft for an air capacity of about 1000 cfm. The filters are installed in banks to achieve the required system capacity. These filters are expendable (single use) pleated mats of fiberglass paper. They are specified to exhibit a minimum efficiency of 99.97% for 0.3- μ m particles and a maximum resistance (when clean) of 1.0-in. H₂O pressure when operated at rated airflow. Tests of filter efficiency are conducted in special facilities which ensure that no significant leakage occurs around the sides of the filter or through other bypasses. It is necessary to construct an equally tight filter enclosure in a field installation to achieve the rated filtration efficiency. The construction of large, tight filter enclosures is a difficult engineering task. Testing of the individual filter banks in place in the enclosure, both before and periodically during the service period, by the dioctyl phthalate (DOP) smoke test is required to ensure that no significant leaks are present in either the filter or the enclosure.

Variables that have been considered in HEPA filter performance analyses include the particle size distribution of the various plutonium aerosols encountered. A literature survey by Davis,⁷ however, does not indicate a gross variation in the range of reported particle sizes in field operations.

Numerous tests have been carried out with plutonium aerosols in small laboratory and large-scale field installations. In a detailed survey by Hetland and Russell,⁸ large-scale filter systems were found which produced overall mass removal efficiencies of 10⁷ or greater. One such system at Rocky Flats showed a removal efficiency of 99.999% across the first two banks of a system of four HEPA filter banks in series, 94% across the third filter bank, and 83% across the fourth filter bank. The low efficiency value for the fourth bank was attributed to probable bypassing of gases and was not a measure of filter media performance. This system, which is about 15 years old, does not represent those presently installed at Rocky Flats where most of the filter plenums have been replaced or modified within the last few years.^{9,10}

The newer plenums are designed to facilitate testing of individual filters and filter banks and to ensure that each stage of filtration can be certified to be at least 99.95% efficient with pneumatically generated DOP aerosol. Data presented by Linck and Geer⁹ for some four-stage systems at Rocky Flats show efficiencies of >99.99% to 99.998% for fourth stages and 99.997% to 99.999% for first stages. These efficiencies of production-scale equipment equal (or perhaps exceed) those obtained by Ettinger et al.¹¹⁻¹³ in laboratory tests using plutonium aerosols in small installations that are tightly sealed and tested periodically for leaks with DOP. They have observed removal efficiencies of at least 99.97% for each of three single-filter stages in series. USNRC Guide3.12¹⁴ for the design of plutonium ventilation systems indicates that removal efficiencies of >99.95% should be obtained for a single bank of HEPA filters if the installation

containing the filters is constructed according to the recommended guidelines and is tested for leaks after installation of the filters. Consequently, a value of 99.95% has been used in this study to represent the *rated* efficiency of each HEPA filter.

Several factors must be considered, however, in predicting the overall installed efficiency of multiple filters in series even though each bank is tested separately in place with DOP and shows an efficiency of 99.95% or greater. First, several tests show that the second and third filters are exposed to much lower concentrations of particles that are of a size distribution which is strongly biased toward the smaller sizes.¹¹ Second, filter efficiencies are sensitive to gas flow rate, and possibly all filters in a bank may not experience the same flow rate. Finally, the concentration of particles is different for each stage of filtration, and filter efficiency varies with particle concentration. Moeller¹⁵ has recently summarized performances of air cleaning systems in nuclear facilities during the years 1966 to 1974.

The present analysis assumes that performances of HEPA-filter installations after 1985 will not be subject to the design and operational errors of these earlier systems. For these reasons, Burchsted¹⁶ recommends the assignment of lower overall efficiencies to filter systems that use HEPA filters in series until more experimental information is available from large installations. Consequently, the overall filter system decontamination factors (DFs) selected for use in this study for HEPA filters in series are conservative in comparison to rated DF values; that is, the first filter is assigned an efficiency of 99.95% and the second 98%, corresponding to a total removal efficiency of 99.999% and a DF of 1×10^5 . These theoretical filter efficiencies were used to estimate the amounts of radioactive particulates expected to be released from the model plant. These estimated amounts were then compared to the amounts released from the Nuclear Fuel Services plant and to other estimates; a DF value of 5×10^8 was selected for use in this study (Sect. 4.5.1) of the retention of radioactive solid particulates across the total plant. This approach is consistent with the basic objective of the study in the use of conservative realistic costs and doses.

The potential exists for mechanical damage to the filters during their initial installation and during replacement in the enclosures. After operations have started, filter efficiency can be decreased through¹⁶ (1) attack by corrosive chemicals such as hydrogen fluoride;^a (2) degradation of the binder for the filter fibers by condensed moisture or by radiation; (3) matting of the retained particles, which decreases the resistance of the binder to moisture and causes an increase in pressure drop; (4) degradation by high temperatures;^a and (5) damage by sudden pressure surges. Thus, continuous monitoring of the pressure drop across the filter and periodic testing with DOP is required to ensure that the filters are operating satisfactorily.

Sand filters. Sand filters¹⁸⁻²¹ are constructed of graded layers of aggregate and sand. The aggregate is in layers of relatively coarse (2-1/2 in. by 1-1/2 in.) to fine particles (-4 +16 mesh). The sand layer is generally about 30 in. thick and is made up to -20 +50 mesh sand. The flow through the filter is upward. Sand filters have good resistance to heat, shock, and chemical attack, but have the disadvantages of higher cost, higher pressure drop, and lower aerosol collection efficiency when compared to other types of aerosol filters such as HEPA filters. Also, it would be difficult to recover a valuable material, such as plutonium, that is collected on a sand filter. For some applications, it may be advantageous to use combinations of sand and HEPA filters to eliminate some of the disadvantages of either type when used alone.

^aHowever, filter media that are very resistant to damage by hydrogen fluoride, water, and fire have already been produced¹⁷ although they are not used in commercial installations. Development of media to resist attack by other chemicals such as nitrogen oxides is also in progress.

Large, fixed-bed sand filters at Hanford have been operated continuously for ~20 years (in 1968) without maintenance or replacement, and at Savannah River for about 18 years²¹ (in 1974). Based on actual measurements at both installations, the collection efficiencies of the filters are >99%.¹⁹ Penetration rate tests have demonstrated that a 30-in.-deep sand bed is roughly equivalent to a single HEPA filter for the particle size remaining airborne after two stages of HEPA filtration.⁸ (The comparison is based on a superficial air velocity of about 5 ft/min^{17,18} at the face of the sand filter and 125 or 250 ft/min specification for a HEPA filter.) However, in this report a sand filter following two HEPA filters in the off-gas system and in the ventilation system was assigned a DF of 10 for particulate and semivolatile matter.

4.3.2 *Evaporation*

Evaporation is commonly used in the chemical industry to concentrate aqueous solutions by boiling off the water and leaving behind most of the dissolved solids and materials having vapor pressures lower than water.²² Similarly, evaporation is very effective in separating dissolved radioactive solids from waste water, and essentially all sizes and types of evaporators have been used in the nuclear industry. However, materials that have vapor pressures higher than water or that combine with water to form high-vapor-pressure materials are difficult to separate from water by evaporation. Because of these factors, iodine, ruthenium, and tritium are among the few radioactive materials that are poorly separated from waste water by evaporation. System DFs of 10^3 to 10^4 can be expected for nonvolatile radioactive contaminants treated in single-stage evaporators. Similar DFs can be expected for ruthenium under alkaline, but not oxidizing or acidic, conditions. The DFs for iodine can be expected to be a factor of 10 to 100 lower than those expected for nonvolatile species under alkaline, but not oxidizing or acidic, conditions.²² These values assume that the evaporator is well-designed, adequately sized, and operated with reasonable skill. An overall decontamination (separation) factor of more than 10,000 between condensate (distillate) and thick liquor (concentrate) is generally expected for nonvolatile radioactive contaminants treated in single-stage evaporators.

In evaporating radioactive waste, care must be taken to avoid too rapid boiling or foaming since each tends to cause the entrainment of minute particles of radioactive solids or liquid droplets in the vapor rising from the boiling liquid surface. Also, the velocity of the vapor must be kept low and the distance the vapor travels upward (disengaging space) must be made as great as practicable to encourage particles and droplets to fall back into the liquid rather than be carried over into the condenser with the vapor. A variety of devices to deentrain particles and droplets can also be incorporated into evaporators to improve DFs to as high as 100,000 or even a million. Such devices work by changing the direction of the vapor path, causing particles and droplets to impinge on and adhere to metal surfaces from which they can later be flushed back into the liquid. Wire mesh filters, sieve trays, bubble-cap trays, and centrifugal separators are among such devices.

Evaporators for radioactive waste can vary from simple pots with steam heating pipes coiled inside to elaborate devices having pumps to circulate the feed through outside heaters and compressors to squeeze more heat efficiency from the hot vapors (vapor compression evaporators). In general, less expensive maintenance and more satisfactory operation is obtained

from simple evaporators equipped with adequate auxiliaries to achieve the DF required. Depending on the amount of dissolved solids in the waste fed to an evaporator, a volume reduction of 10 to 50 can usually be achieved in the radioactive thick liquor (bottoms or concentrate) while maintaining the level of radioactive material in the condensate (overheads or distillate) 10,000 to a million times lower than that in the bottoms. To achieve such good separation, however, no foamovers can be permitted and entrainment must be kept to a minimum. Therefore, laundry wastes containing detergents or other foam-producing materials must be kept out of the evaporator. Liquid waste evaporators should be tested before use on actual waste streams. This is probably the only reliable method of demonstrating that the desired DF values can be achieved over the extremes of conditions expected. Stable isotopes and tracer levels of radioactivity can be used in these tests.

The behavior of iodine during liquid waste evaporation is complicated and poorly understood. Studies are needed to better define its vapor pressure as a function of pH value, redox potential, and other parameters which determine the physicochemical behavior of iodine. Changes in operating conditions suggested as a result of these studies should be confirmed in large-scale evaporator tests.

4.3.3 *Adsorption on silver zeolites*

Silver and other metal exchange zeolites have been investigated on a laboratory scale for possible use in removing iodine from gaseous effluents.²³⁻²⁵ The adsorbent is prepared by a partial replacement of sodium ions with silver (Ag) ions in the zeolite. Silver-exchanged zeolites have received the most laboratory testing; other cations such as Cd, Cu, Hg, Pb, Ti, and mixed rare earths have also been studied, but they are much less effective than silver.²³ The adsorption efficiency of the silver zeolites for both elemental iodine and methyl iodide is >99.9% (DF = 10^3 to 10^4) with fresh adsorbent and when the adsorbent is maintained at about 200°C. Several variables are important²⁶ in estimating the effective decontamination factors of silver zeolite beds for ^{129}I and ^{131}I . These include: (1) nuclide half-life; (2) the extent of isotopic exchange between fresh iodine and that which is already sorbed; and (3) the extent of recycle of iodine in the system.

Ackley and Davis²⁵ performed long-term (11 months) small-scale tests of iodine removal (with 60-day ^{125}I tracer) and of CH_3I removal (with 12.4-hr ^{130}I tracer) by silver zeolite beds. Their results can be interpreted²⁶ by including loss of short half-life isotopes by radioactive decay as part of the measured (or expected) DF. The DF due to decay can be much larger than that due to trapping by silver zeolite for isotopes with half-lives as short as the 12.4 hrs for ^{130}I . Correspondingly, the DF for ^{131}I , with a half-life of only 8.05 days, will be greater than that for ^{129}I , with a half-life of 1.6×10^6 years; the factor will depend on the extent of isotopic exchange and on the history of the zeolite, including the time during which the bed has been operating and whether nonradioactive iodine was deposited on it before "hot" startup.

Silver zeolite has the following advantages when compared with other adsorbents such as activated charcoal:

- (1) *Stability.* Iodine is not released at temperatures up to 910°C.²³ Iodine is removed from air streams containing methyl iodide at temperatures up to 400°C.

- (2) Nonflammable. Zeolites are inorganic aluminum silicates and are not flammable.
- (3) Low explosion hazard. Zeolites do not react explosively with the oxides of nitrogen.
- (4) Efficient in humid conditions. Iodine is removed from air under conditions of 90 to 100% relative humidity.
- (5) Resistant to poisoning. Zeolites are resistant to the poisons normally contained in air. However, they are readily poisoned by halides such as chloride.
- (6) Solid waste product. The adsorbed radioiodine species form a solid insoluble product with silver zeolite, which is desirable from the standpoint of waste management.

Although it has not been demonstrated on a plant scale, the efficiency of the silver zeolites appears to be one or two orders of magnitude higher than that of the older type of silver reactors used at Hanford and Savannah River.²⁷ The high cost of silver will probably preclude the use of silver zeolites for removal of the bulk of the iodine, and their use will be confined to that of polishing adsorbers after the bulk removal treatment steps. Silver zeolite beds will be used in two commercial fuel reprocessing plants as polishing units for the removal of iodine from gaseous effluents.^{28,29}

A conservatively low, average, removal efficiency of 99% (DF = 100) for both ¹²⁹I and ¹³¹I on a silver zeolite bed is used in this study pending the development of additional data in plant-scale usage. This is consistent with the estimates presented in current environmental analyses.^{1,30} The silver zeolite bed would be replaced when the average DF for ¹³¹I decreased to some value defined in the Technical Specifications of the reprocessing plant. The average DF for ¹³¹I will actually be greater than for ¹²⁹I, since the effective retention of the former is increased by decay due to its short half-life.²⁶

4.3.4 Iodine evolution

About 95 to 98% of the iodine in the dissolver solution can be evolved by steamstripping in the presence of excess N₂O₃ or a mixture of nitrogen oxides (NO_x). Additional iodine is evolved by (1) adding an iodine carrier (KI), (2) oxidizing the solution with ozone or H₂O₂, and (3) sparging with N₂O₃. Greater than 99% of the iodine can be evolved from a nitric acid solution by this method.³¹⁻³⁴ An iodine evolution efficiency of 99.5% is used in this study when these procedures are applied in properly designed supplemental equipment (Fig. A-1, Appendix A). In laboratory-scale work by D. O. Campbell et al.^a with LWR fuel irradiated to 31,360 MWd/MTU, greater than 99% of the iodine was evolved during the dissolution of the fuel and during the sparging of the dissolver solution with air. Thus, the addition of an iodine carrier and an oxidizing agent and sparging with N₂O₃ in specially designed supplemental equipment may not be required with highly irradiated fuel. However, the dissolver systems in large pilot plants or full-scale processing plants are complicated engineering systems. Consequently, the achievement of a high removal of iodine during the dissolution of highly irradiated fuel, without the use of the supplemental iodine evolution system, requires demonstration in a large-scale dissolver system.

The removal of greater than 99% of the iodine from the dissolver solution is desirable to prevent a large fraction of the iodine from entering the solvent extraction system. The iodine is volatilized and enters into the gaseous effluent treatment system; consequently, the amount of

^aB. L. Vondra et al., *LWR Fuel Reprocessing and Recycle Program Quarterly Report for Period July 1 to September 30, 1976*, ORNL/TM-5660 (1976).

iodine that must be removed in the liquid waste treatment system is reduced. Removal of the iodine from the aqueous stream prior to its entry into the solvent extraction system also minimizes the formation of organic iodides, which are produced by the reaction of the iodine with the organic solvent. Generally, organic iodides are more difficult to remove from gaseous effluents than elemental iodine. A further advantage of this process is that 99.5% of the mass of iodine is transferred to a single process stream from which it can be removed as a solid waste. Thus, iodine is effectively removed from the plant fluid streams and does not accumulate in the total plant in a mobile form.

The process is not complex, and conventional equipment would be used in a commercial reprocessing plant. The process has been successfully demonstrated on a laboratory scale using simulated dissolver solutions³³ and in hot-cell tests with dissolver solutions from irradiated fuel samples.³⁴ The amounts of iodine volatilized were in the high nineties and 99.5% was achieved with irradiated fuel. Consequently, an iodine evolution efficiency of 99.5% is assumed for this study. However, engineering development and a demonstration of the process in a pilot plant or a reprocessing plant with irradiated LWR fuel and dissolver solution are required.

4.3.5 *Mercuric nitrate-nitric acid scrubber*

The mercuric nitrate-nitric acid process for the removal of iodine species from gaseous effluents uses an 8 to 10 M HNO_3 -0.2 to 0.4 M mercuric nitrate solution as a scrubbing agent. Iodine species are scrubbed from the gas stream and converted to a nonvolatile, soluble mercury-iodine complex. The spent scrub solution can be handled (1) by storing the solution in the concentrated miscellaneous waste storage tank along with other liquid wastes (as in the Case 1 study, Sect. 4.5.1), or (2) by installing equipment to recover and reuse the mercury and to isolate the iodine as a solid, sodium iodate (as in Case 2b, Sect. 4.5.3). It is expected that the solid sodium iodate can be packaged in stainless steel cans for storage or shipment to a waste repository. The recycle procedure has several advantages, as follows:

- (1) Less mercury is used.
- (2) Less mercury is introduced into the stored wastes. Mercury will volatilize if these wastes are subsequently solidified by a high-temperature process such as calcination.
- (3) The 1.35 kg of ^{127}I and ^{129}I that enters the plant each day in the fuel is not accumulated in the waste tank.

The iodine in the waste tank will partially volatilize during sparging, which will be performed to provide uniform mixing of successive batches and to minimize the accumulation of sludges in the bottom of the tank. This will complicate the problem of retaining iodine to the plant (Sects. 4.5.1 - 4.5.3). Essentially complete volatilization of iodine will occur in the event that these wastes are subsequently solidified by a high-temperature process. The mercury recycle system consists of:

- (1) Evaporation of the scrub solution followed by cooling to precipitate mercuric iodate. The supernate is recycled to the scrubbing system.
- (2) Treatment of the mercuric iodate with caustic to form a precipitate of mercuric hydroxide and a solution of sodium iodate. The solids are separated by filtration and are dissolved in nitric acid and recycled to the scrubbing system.

(3) Solidification and packaging of the sodium iodate.

The mercuric nitrate-nitric acid system can be used either as a primary step for removing the bulk of the iodine from the dissolver off-gas stream or as a polishing unit for removal of small amounts of iodine from the combined process and cell off-gas streams. Both elemental iodine and methyl iodide, which are representative of organic iodides, are removed by the mercuric nitrate scrubbing system. However, the system is more efficient for the removal of elemental iodine than organic iodides. Decontamination factors of less than 10 and up to 10^4 to 10^5 have been obtained in experimental systems at ORNL under various operating conditions.^{35,36}

The mercuric nitrate-nitric acid scrubbing process has been demonstrated in plant installations using less-concentrated solutions of nitric acid and mercury.³⁷ Similar scrubbing systems are being installed on the dissolver off-gas and vessel off-gas streams at the Barnwell Nuclear Fuel Processing Plant and are proposed for installation at the Nuclear Fuel Services Plant. The advanced process (described above), which uses a more concentrated solution of mercuric nitrate and nitric acid, is used in this study. This process and the associated system for recycle of mercury and isolation of sodium iodate have been successfully developed and demonstrated at ORNL on a laboratory scale. A plant DF of 10 for ^{131}I and ^{129}I is used in this study for the scrubbing system in the Case 1 study, where the mercury recycle and iodine isolation systems are not used. This is consistent with the estimates presented in current environmental analyses.^{1,30} In the Case 2b study, the iodine evolution and iodine isolation systems are installed; consequently, the off-gas from the solvent extraction system is expected to contain a smaller concentration of organic iodides (Sects. 4.3.4 and 4.5.1 - 4.5.3). As a result of these additions and the decrease in organic iodide content, the DF in the off-gas system for all species of iodine will be greater in Case 2b than in Cases 1 and 2a. However, no credit is taken for the decrease in organic iodide content in the analysis of Case 2b, although an additional factor of 10 for iodine removal is assigned to the mercuric nitrate scrubber in the dissolver off-gas system. Finally, because the inventories of ^{127}I and ^{129}I are expected to correspond to about 8 days of input and ^{131}I will decay during its accumulation for 8 days, the DF of ^{131}I is assumed to be about twice that of ^{129}I .

4.3.6 Iodox

The Iodox process^{38,39} is an advanced method for removing elemental iodine and organic iodides, such as methyl iodide, from gaseous effluents. The process is suitable for use either as a primary step for removing the bulk of the iodine from the dissolver off-gas stream or as a polishing unit for removing small amounts of iodine from off-gas streams. The iodine can be isolated as a solid, anhydriodic acid (HI_3O_8), which is probably suitable for storage in stainless steel cans or for shipment to a waste repository.

The steps in the Iodox system are: (1) oxidation of the iodine species to the soluble, nonvolatile iodate form using 19 to 21 *M* HNO_3 in a bubble-cap or packed column; (2) concentration of the iodine-bearing nitric acid scrub solution in an evaporator; (3) recycle of iodine-free nitric acid condensate from the evaporator to the plant nitric acid system; and (4) transfer of the iodine-nitric acid concentrate from the bottom of the evaporator to a second

evaporator where it is evaporated to dryness to form HI_3O_8 . The condensate from the second evaporator is recycled to the first evaporator. The final evaporation to dryness could occur in the waste storage shipping can.

The Iodex process has been demonstrated successfully on a laboratory and small engineering scale using simulated off-gas containing small amounts of radioactive iodine. Decontamination factors for removal of both elemental iodine and methyl iodide of greater than 10^4 have been obtained using (1) a 2.54-cm-diam bubble-cap column containing six plates and with $>75\%$ HNO_3 at temperatures $>80^\circ\text{C}$, or (2) a 4-cm-diam by 117-cm-high packed column with boiling 70% HNO_3 or 80% HNO_3 at $>60^\circ\text{C}$. The high acid concentrations and temperatures are required to decompose organic iodides and to convert the contained iodine to a form that is readily retained (probably elemental iodine). Elemental iodine is retained efficiently at lower acidities and temperatures. In plants in which the high-activity wastes are stored in tanks, the concentrated acid required for the Iodex process would be purchased and the excess acid recycled to the dissolver. This type of operation is reflected in the cost estimates in Sect. 6.0 for the model plant. In plants where the high-level wastes are solidified, the acid is recovered at low concentrations and an extra fractionation system is required to produce the concentrated acid for recycle to the dissolver and Iodex systems. Very little fresh acid would be purchased for this type of plant.

The principal advantages of the Iodex process are: (1) the high removal of iodine achieved for either elemental or organic iodides, (2) the feasibility of handling large amounts (mass) of iodine, (3) that no new chemicals are introduced into the system, and (4) that conventional processing equipment is used. Disadvantages include: (1) corrosion problems that will require the use of titanium or zirconium equipment as materials of construction, and (2) the requirements for internal plant production of concentrated nitric acid in plants where the high-activity wastes are solidified and nitric acid is recycled.

The Iodex process has been successfully demonstrated in laboratory- and small-scale engineering equipment using synthetic solutions. However, engineering development and a demonstration of the process in a pilot plant or reprocessing plant with irradiated LWR fuel and dissolver solution are required.

4.3.7 Voloxidation

The voloxidation process is an advanced method for the removal of volatile fission products from the sheared fuel prior to dissolution.⁴⁰ The overall objective of the voloxidation step is to remove tritium, iodine, xenon, and krypton from the sheared fuel for isolation and storage as waste. In this process, the sheared fuel is heated to about 550°C in air or oxygen to release the volatile fission products via thermal evolution or oxidation. The major emphasis is centered on the removal and isolation of tritium to prevent the mixing of the tritium with natural water in the dissolver. It is economically impractical to separate tritium from natural water. The removal of radioiodine is not considered essential since effective techniques are being developed to remove iodine from the dissolver solution and off-gases (Sects. 4.3.3-4.3.6). However, the partial volatilization of iodine in the voloxidation step would be an advantage in the overall control of iodine. The same comment applies to the noble gases, xenon and

krypton, which are evolved during dissolution.

The voloxidation step in the reprocessing head-end facility would be located between the shear and dissolver. The process equipment would consist of a rotary kiln (approximately 3 ft in diameter by 24 ft long for a 5-metric ton/day capacity) in which the sheared LWR fuel is heated to oxidize the UO_2 to U_3O_8 to release the volatile fission products. The off-gas from the voloxidizer would be processed through (1) a recombiner to form tritiated water from hydrogen, tritium, and oxygen, and (2) a dryer to collect the tritiated water (a few gallons per day from a 5-metric ton/day plant) and separate the tritiated water from the other volatile fission products. The tritiated water can be stored as water or adsorbed on a desiccant, such as Drierite or molecular sieves, and packaged in a stainless steel can for storage as a solid or for shipment to a waste repository. Laboratory-scale tests with highly irradiated UO_2 and $\text{UO}_2\text{-PuO}_2$ show that greater than 99% of the tritium and up to 75% of the iodine and 45% of the krypton are volatilized in the voloxidation process. In some cases with highly irradiated and restructured fuel, up to 98% of the krypton is volatilized.

The voloxidation process has been demonstrated successfully on a laboratory scale for the removal of tritium from irradiated LWR and LMFBR fuel samples. Engineering development has been limited to rotary calciner tests with unirradiated UO_2 and design studies. Engineering development and a demonstration of the process in a pilot plant or reprocessing plant with irradiated LWR fuel are required.

4.3.8 *Selective absorption*

The selective absorption process is a method for removing the noble gases, krypton and xenon, and carbon dioxide (CO_2) from gaseous effluents. Carbon-14 will also be removed if it is present as $^{14}\text{CO}_2$. This process has progressed to the nonradioactive pilot plant stage of development.^{41,42} The retained gases are recovered in a form suitable for bottling in gas cylinders. The absorption process takes advantage of the relative solubilities of gases in a fluorocarbon solvent (chlorofluoromethane). Water is removed from the gas in an initial step to prevent ice formation in subsequent steps. The gas is then compressed to about 500 psia, cooled to about -4°F , and contacted countercurrently with the liquid solvent in a packed absorber column. The least-soluble gases (including nitrogen and oxygen) exit from the top of the column, exchange heat with the feed gas, and are discharged to the atmosphere. The solvent from this column, loaded with krypton, xenon, and other soluble gases, is routed into a packed fractionating column that is operated at about 30°F and 45 psia. At the lower pressure and higher temperature, solvent vapor is recycled in the column between the reboiler and condenser, driving the remaining slightly soluble gases in a recycle back to the feed stream and concentrating the more soluble gases, including krypton and xenon, in the liquid solvent flowing down the column. The enriched solvent is routed from the reboiler of the fractionator to a stripper column that is operated at a temperature of about 12°F and a pressure of about 30 psig. Krypton, xenon, and other soluble gases are vaporized in the stripper and may be collected as a concentrated product. Essentially pure solvent, suitable for recirculation to the absorber, is collected in the reboiler of the stripper. The product gases can be collected in pressurized cylinders in a form that is suitable for storage or shipment to a waste repository.

The existing pilot plant equipment is designed for an inlet air flow of 20 scfm. The absorber and fractionator columns have diameters of 3 in. and heights of 10 ft. The stripper column is 6 in. in diameter by 8 ft tall. In tests with various concentrations of natural krypton and xenon in air, greater than 99.9% of these noble gases were retained and transferred to a product stream that had less than 0.001 the flow rate of the feed stream. Short tests (~2 days) have been conducted in the pilot plant to evaluate the behavior of gaseous impurities such as nitric oxide, nitrous oxide, nitrogen dioxide, carbon dioxide, methyl iodide, and iodine. Refrigerant-12 was used as the fluorocarbon solvent in these tests. No operational problems were encountered. Impurity tests have verified that carbon dioxide removal in excess of 99.9% can be achieved with the fluorocarbon-based process.⁴³ The removal efficiency for CO is very low (essentially the same as that for oxygen), and it is assumed that a catalytic oxidizer would be installed upstream of the selective absorption process to assure that the CO and organics such as short chain alkanes are oxidized to CO₂. The efficiency for removal of krypton remained high (>99%) in the tests with the nitrogen oxides and carbon dioxide. However, there was no krypton in the gas stream in the tests with methyl iodide and iodine. Auxiliary systems consisting of feed gas preparation, solvent recovery, solvent purification, and product purification are being investigated.⁴⁴

The process is safe from explosions and fire and can be operated for sustained periods at high efficiency with low maintenance requirements. The system is versatile, continuous, and adaptable to scale-up, and is offered commercially for use on the gaseous effluents from nuclear power reactors.⁴⁵ The process has not been tested on the off-gas from a reprocessing system that could contain significant amounts of contaminants. Engineering development and a demonstration of the process in a pilot plant or reprocessing plant on the off-gas from the dissolution of irradiated LWR fuels are required.

4.3.9 Cryogenic distillation

The cryogenic process is designed to remove noble gases (krypton and xenon) from gaseous effluents.⁴⁶ Cryogenic distillation provides an effective, continuous, small-sized system for the separation of gases based on their relative volatility. This type of process is used commercially for isolation of the components of air and is being used intermittently to remove radioactive xenon and krypton from a 20-scfm off-gas stream at the Idaho Chemical Processing Plant (ICPP).^{47,48} The process is capable of recovering krypton and xenon in a relatively pure form suitable for bottling in gas cylinders. The equipment generally consists of a gas pretreatment train, a regenerative heat exchanger, the primary distillation column, and a batch distillation column that is used intermittently for product purification. The distillation column is operated at a temperature of about -300°F, at which nitrogen is liquid and relatively volatile. Hydrogen, if present, is more volatile and is vaporized out the top of the column with much of the nitrogen. Argon and oxygen, if present, concentrate in the bottom of the column as a liquid. Most other gases, including xenon, krypton, and hydrocarbons, would be dissolved and concentrated in the liquid nitrogen-argon-oxygen solution at the bottom of the column.

Water, carbon dioxide, nitrogen oxides, and hydrocarbons must be removed from the inlet gas to prevent plugging of the column. Carbon-14 as ¹⁴CO₂ would be removed from the

process off-gas during this pretreatment step. It is particularly important that hydrogen and solid forms of acetylene, other hydrocarbons, nitrogen oxides, and ozone not be allowed to accumulate in the still since these materials have been the source of violent explosions in commercial air liquefaction plants.

At the ICPP these hazards are minimized by a high-quality system for purification of the entering gas and frequent transfer to the batch still to minimize the accumulation of objectional species. However, the accumulation of potentially explosive concentrations of ozone from the irradiation of oxygen is a possibility. This problem could be eliminated by removal of the oxygen in a pretreatment step. A cryogenic system is offered commercially in which oxygen, hydrogen, nitrogen oxides, and hydrocarbons are removed in an initial step by catalytic units to reduce or eliminate these potential hazards.⁴⁹ However, this system has not been tested on the off-gases from a reprocessing plant that would contain significant amounts of contaminants.

Cryogenic distillation systems are used commercially to produce the components of air. Modifications of these systems will be installed at nuclear power stations to remove noble gases from gaseous effluents.^{50,51} However, these off-gases do not contain the same impurities, such as nitrogen oxides, as the off-gases from a reprocessing plant. A small cryogenic plant has been operated on reprocessing off-gases at the ICPP plant. The main problems associated with the latter operations have been associated with the removal of impurities from the feed stream. The equipment has been operated in campaigns not exceeding 1.5 months in duration, and the krypton-xenon recovery has generally been less than 90%.

Either the selective absorption (Sect. 4.3.8) or the cryogenic system could be used for the removal of noble gases and ^{14}C as $^{14}\text{CO}_2$ in this study, since the cost and the state of development of the two systems are estimated to be comparable and either system could achieve the designated removal efficiency. An overall plant removal efficiency of 99%, including bottling, is assumed for both krypton and ^{14}C . Selective absorption was chosen for use in this study because there are no known operating hazards associated with this method. If the cryogenic process is selected for use, further engineering development and demonstration in a pilot plant or reprocessing plant on the off-gas from the dissolution of irradiated LWR fuels will be required.⁵²

4.3.10 *Caustic scrubbers*

Caustic scrubbers have been studied in small-scale experiments and are used extensively in reprocessing plants and other nuclear installations to remove iodine, nitrogen oxides, and semivolatiles and particulates such as ruthenium from gaseous effluents.^{53, 54} Decontamination factors as high as 10^2 to 10^4 have been achieved for I_2 and HI . Caustic scrubbing can remove a large fraction of the ruthenium from gaseous effluents. About 99% of the ruthenium was removed from the off-gas leaving the scrubber prefilter in pilot plant tests of the solidification of high-level radioactive waste.⁵³ However, caustic scrubbers are probably less efficient for removing organic iodides or particulates. It has been suggested that the volatile species from alkaline solutions is HOI or a species in equilibrium with it,⁵⁵ but the chemistry and the performance of HOI are largely unknown. The reasons for the relatively poor performance of large scrubbers at Idaho and Hanford (i.e., ~70% and 90% removal of iodine, respectively) are

also unknown. The low concentrations of iodine in the gas phase, that is, $1 \times 10^{-3} \mu\text{g}$ of I_2 per cubic foot, could be a factor at Hanford. (Concentrations have not been reported at Idaho.) Tests of the caustic scrubber for the Oak Ridge Research Reactor (ORR) showed a removal efficiency of 99%, with an inlet iodine concentration of $15 \mu\text{g}/\text{ft}^3$. Other factors are the unknown amounts of particulates and organic iodides present.

No reports have been found in the literature that relate the efficiencies or kinetics of caustic scrubbing to irradiation rate, concentration of iodine in both phases, concentration of caustic, temperature, and contacting efficiency. The use of additives with the caustic to reduce iodine and iodate to iodide increases the removal efficiency for iodine. Thiosulfate is a commonly used reductant. The removal efficiency of the Idaho system increased from 90 to 97% when the caustic contained thiosulfate.⁵⁶

Caustic scrubbers have been studied^{57,58} for the removal of CO_2 from atmospheric air, and this work is applicable for the removal of ^{14}C -bearing CO_2 from the off-gas from a LWR fuel reprocessing plant. It is estimated that about 80% of the CO_2 could be removed in a caustic scrubber. The degree of removal could probably be enhanced by the addition of nonradioactive CO_2 to the off-gas. However, it is assumed in this report that 80% of the ^{14}C as CO_2 is removed in a caustic scrubber. It is also assumed that most of the carbon-bearing compounds are present as CO_2 and that a catalytic oxidation unit is not required to oxidize CO and other carbon-bearing compounds to CO_2 to achieve this limited efficiency.

4.3.11 *Absorption on charcoal or macroreticular resins*

Solid sorbents have been used in laboratory studies to remove radioactive iodine from water and acid solutions with high efficiency. More than 99.99% of the iodine was removed from water containing $1 \times 10^{-6} M$ I_2 by passing it through a 4-in.-deep bed of coconut charcoal.⁵⁹ Charcoal is not effective in removing iodine from acidic solutions.

Macroreticular resins absorb iodine efficiently from water or acidic solutions.^{60,61} These resins are hard, insoluble beads of porous polymer. They have discrete pores ranging from 50 to 200,000 Å and are available in a wide range of surface polarities. Macroreticular resin XAD-12 (Rohm and Haas Company) adsorbed ~99.4% of the iodine from water, a slightly lower efficiency than that obtained with charcoal under comparable conditions. However, the sorbed iodine could be removed from the resin with thiosulfate solution, and the resin could be reused while still maintaining the high iodine removal efficiency. Macroreticular resin XAD-4 is also effective for removing iodine from nitric acid solutions. Removal efficiencies of about 99% were obtained when treating 3 M HNO_3 solution that contained $5 \times 10^{-4} M$ I_2 .

4.4 Selection of Case Studies

Eight conceptual cases and the corresponding flowsheets were prepared for treating the radioactive liquid and gaseous effluents. Case 1 represents the base cost, current treatment case. In each succeeding case, equipment is added to accomplish a specific objective in reducing the release of radioactive material (Table 4.3). The efficiency of a treatment system or plant for

retention of radioactive material is expressed as a decontamination factor (DF), which is the ratio of the amount of material entering a plant to that released to the environment. Relative to Case 1, the retentions of radioactive material achieved in the succeeding cases are increased by the following factors: Case 2a, a DF of 25 for ^{129}I and 50 for ^{131}I ; Cases 2b and 2c, a DF of 500 for ^{129}I and 250 for ^{131}I . Consequently, the total additional DF for Case 2 relative to Case 1 is 500 for ^{129}I and 250 for ^{131}I ; Case 3, a DF of 100 for krypton and ^{14}C ; Case 4, a DF of 10 for particulates including and uranium and the semivolatile ruthenium; Case 5, a DF of 10 for the semivolatile material ruthenium and a DF of 5 for ^{14}C ; Case 6, a DF of 100 for tritium, 1.5 for ^{129}I , and 2.5 for ^{131}I ; Case 6c, an accumulation of Cases 2 through 6. In Case 7, the additional DFs relative to Case 1 are 2.5×10^5 for ^{129}I and 1.25×10^5 for ^{131}I , 1×10^4 for krypton, 1×10^3 for ^{14}C , 2×10^3 for particulates, 1×10^4 for semivolatiles, 5×10^3 for plutonium, 2×10^4 for uranium, and 1×10^3 for tritium.

Each case represents the probable limit of retention obtainable with existing and presently projected technology. With the exception of Case 2, additional parametric studies within a given case appear impractical since the cost for treatment systems to achieve lower retentions would be about the same as those cited. The treatment units contained in Cases 2 through 6 can be added to Case 1 independently or cumulatively for assessment of environmental impact or cost calculations. Generally, Case 1 represents current technology, and Cases 2 through 6 represent technology that may be applied in the next 2 decades to plants that are designed by using the present concepts for management of process and ventilation gases and process liquids. Case 7 represents an advanced design in which gases and liquids are treated and recycled, and where extensive changes in the present concepts for effluent management and plant design are required. Much of the technology used in the advanced cases is in an early stage of development and is not suitable for immediate use in existing plants. Some of the equipment listed in Cases 2 through 6 can probably be backfitted to existing plants, but this must be considered on an individual basis. Implementation of the concepts contained in Case 7 would require the construction of a new plant.

Most of the treatment systems used in this study are similar to those used, or are proposed for use, in the industry. The remaining systems are extrapolations from existing systems or are in the pilot plant or small engineering scale of development. The reasons for choice of these systems, along with technical descriptions of their functions and stage of development, are given in Sects. 4.3 and 4.5.

4.5 Description of Case Studies and Calculation of Source Terms

Descriptions of the case studies for decreasing the releases of radioactive materials from the model plant are presented in the following sections. The assumptions used in the case studies are presented in Table 4.3, the calculated source terms are in Table 4.2, and the flowsheets are in Figs. 4.3-4.11.

4.5.1 Case 1

Case 1 is the "base case" for the model fuel reprocessing plant, and it represents current practice in the industry (Fig. 4.3). The selection of assumptions used to calculate the source term in this case is particularly important since the improvements presented in succeeding cases are incremental with reference to the base case. The background information available to

substantiate the estimates of the amounts of various types of radioactive materials to be released from the model plant is somewhat limited because of the limited experience established in private industry. Estimates of the amounts of radioactive materials released are based on experience at the Nuclear Fuel Services (NFS) plant (Table 4.4); models developed from data obtained at ERDA facilities;⁶² safety analysis reports for the three commercial reprocessing plants, NFS, Midwest Fuel Recovery Plant (MFRP), and Barnwell Nuclear Fuel Plant (BNFP);⁶³⁻⁶⁵ environmental reports for MFRP and BNFP;^{66,67} and a current ERDA development program.⁶⁸

Iodine. Iodine-131 was not detected in the liquid or gaseous effluents from NFS, because it had decayed to nondetectably low levels during the 1 year that the fuel was cooled before processing. The ¹²⁹I release data for 1969 and 1970 show that up to about 25% of the iodine was released in the aqueous low-level waste, corresponding to a plant DF of about 4 for ¹²⁹I. This indicates that additional iodine removal equipment is required on the aqueous effluent stream to obtain a higher DF, regardless of whether the water is vaporized and released up the stack or is released as a liquid. In this study, equipment is used to treat both the gaseous effluent and the water before it is vaporized and released up the stack such that overall plant DFs of 20 and 40 are achieved for ¹²⁹I and ¹³¹I, respectively, in Case 1. The lower value was selected for ¹²⁹I because this isotope does not decay significantly during its lifetime in the plant and is expected to accumulate in the plant recycle streams.

Yarbro^{67,70} has summarized the retention of iodine as it applies to the BNFP, which was used as a model for some of the aqueous-phase processes for Case 1. His estimate indicated that 90% of the iodine would be evolved from the dissolver and that 10% would remain in the solution. This conservative estimate is also used in this report although recent data indicate that more iodine may be evolved from highly radioactive solutions than previously expected based on tests with low activity solutions (Sect. 4.3.4). Davis^{71,72} has made a detailed analysis of the movement of iodine in gaseous and liquid waste streams that is applicable to a LWR reprocessing plant.

Tritium, krypton, and ¹⁴C. Tritium, krypton, and ¹⁴C are released quantitatively to the atmosphere. Consequently, the plant DF for these nuclides is 1.

Particulates. The particulate release data for NFS (Table 4.4) for the years 1969 and 1970 indicate plant DFs of 5×10^8 and 3×10^8 , respectively, and an overall average of 10^8 for the years 1966 to 1971. It has been estimated in a theoretical analysis of particulate release⁶² that, if the radioactive solutions in the plant contain 300 g of fuel per liter (typical of the dissolver and accountability tanks, which contribute significantly to the off-gas) and have a specific gravity of about 1.2, the estimated concentration of fuel in the effluent gas from the filters is 0.3×10^{-12} metric ton of fuel per cubic meter of air. This corresponds to a DF of about 10^8 based on a filter efficiency of 99.98%, a plant capacity of 1 metric ton/day, and an off-gas rate of 1000 cfm. This value is consistent with the estimates for the new reprocessing plants,^{1,2} and 5×10^8 was selected for this study. Additional discussion of the particulate removal efficiencies estimated for filters in the model plant are presented in the section on *Gaseous effluent treatment*.

Semivolatiles. A DF of 1×10^8 was selected for semivolatiles based on the available information. The DF for semivolatiles is expected to be lower than that for nonvolatile particulate material. Only ruthenium is considered to be semivolatile in this report.

Uranium and plutonium. Uranium and plutonium will be released as particulates in the gaseous effluents. However, DFs of 5×10^7 and 2×10^8 , respectively, were chosen for these materials (as opposed to the higher DFs for the particulate materials) to reflect the additional processing steps in which concentrated solutions of uranium and plutonium are handled.

The above values for plant DFs and the calculated amounts entering the plant are used to calculate the source term for Case 1 (Tables 4.2 and 4.3).

Fuel dissolver system. The selection of the type of dissolver system to be used in the model plant is important, since the rate of dissolution of the fuel determines the rate at which the off-gases will be formed, and the volume of gas to be treated in a unit of time determines the capacity of the equipment required in the off-gas treatment system. Several dissolver systems have been studied for the dissolution of nuclear fuels, and the development of these systems has been reviewed by Groenier.⁷³ Dissolvers are generally classified as batch, semicontinuous, or continuous, depending on the manner (rate) in which the sheared fuel is added to the dissolver vessel (Sect. 4.1, *Fuel shearing and dissolution*). Semicontinuous and continuous dissolvers produce off-gases at a reasonably uniform rate during dissolution.

A semicontinuous dissolver system, which consists of three basket stations, is selected for use in this study. Each station handles up to 2 tons of fuel per day (a total of 5 tons/day). The fuel is added to a station over a period of 6 to 8 hr and thence to each station in succession. Nitric acid is circulated continuously through the basket station. About 90% of the fuel is dissolved during the first hour of contact with the acid, and about 99% is dissolved in 2 hr.⁷⁴ After the filling period, the fuel hulls are contacted with fresh acid for 8 hr to dissolve any remaining fuel. On these bases, it is estimated that:

- (1) Less than 0.04 metric ton of fuel remains undissolved at any one time.
- (2) Essentially all of the krypton is evolved continuously at the same rate that the fuel is dissolved, since the solubility of krypton in hot nitric acid is low.
- (3) Iodine is evolved continuously as the fuel dissolves, but a fraction of the iodine remains dissolved in the dissolver solution. In this study, it is estimated that about 90% of the iodine is evolved from the dissolver vessel (the major fraction) and from subsequent process vessels (minor amounts), and that about 10% of the iodine remains in the liquid entering the solvent extraction and the waste treatment systems.
- (4) Tritium remains with the dissolver solution.
- (5) All of the ^{14}C is evolved during shearing and dissolution of the fuel and from process vessels. It is assumed that the ^{14}C is volatilized from the dissolver as $^{14}\text{CO}_2$ or some other type of carbon-bearing gaseous compound. Little experimental data exist on the fate of ^{14}C in a fuel reprocessing plant. In laboratory-scale work⁷⁵ with LWR fuel irradiated to 31,360 MWd/MTU, a large fraction of the ^{14}C was evolved during the dissolution of the fuel. The gas was retained in a caustic scrubber indicating that it was probably $^{14}\text{CO}_2$. However, the actual fraction of the ^{14}C evolved was not determined, since the amount remaining in the dissolver solution was not measured.

These data indicate that less than 0.04 metric ton of fuel would continue to dissolve after the dissolver is shut down and that the amounts of krypton, ^{14}C , and iodine evolved thereafter are less than those contained in 0.04 metric ton of fuel. Shutdown of the dissolver consists of terminating the addition of sheared fuel and introducing cold water into the cooling coils.

These data are used in analyzing the limits that may be placed on the releases of radioactive materials if the advanced treatment systems added in the advanced case studies should fail, and if the dissolver is shut down as an emergency control measure (Sects. 4.5.1-4.5.9).

Gaseous effluent treatment. In Case 1 (Fig. 4.3), gases from the shear and noncondensables from dissolution steps are combined and passed directly to a NO_x absorber for the recovery of acid values. This composite stream contains a small percentage of the tritium (in the entrained droplets of liquid), some particulates, all of the krypton, all of the ^{14}C , and about 90% of the iodine initially present in the fuel. These materials are evolved during dissolution. Some of the particulates and semivolatiles are removed from the off-gas in the NO_x absorber. The combined DOG and VOG gaseous effluents pass through the mercuric nitrate-nitric acid scrubber, which removes 90% of the total iodine. The process off-gas is then heated and filtered before being passed through the silver zeolite bed, which achieves an additional ~99% removal of both organic and elemental iodine. Subsequently, the stream is passed through two banks of HEPA filters in series. These filters can be tested "in place" after installation and periodically thereafter. The two filter banks in series would have a rated efficiency of 99.95% for each bank for the removal of particulates, which corresponds to a rated total DF of 4×10^6 . However, in this study it is assumed that the filters in a commercial plant are subject to some degree of impairment of efficiency and that the DF across the two filter banks is reduced to 1×10^5 (Sect. 4.3.1). This degree of removal, coupled with the assumption that the plant DF for particulates is $>5 \times 10^8$, indicates that up to 0.02% of the fuel entering the plant could be dispersed into the off-gas system without decreasing the plant DF. The value of 0.02% of the fuel (~1 kg/day) is conservative, and the entrainment of this large amount of fuel is not expected. Cell and laboratory ventilation air is also passed through roughing and two HEPA filters in series, mixed with the purified process off-gas stream, and discharged through the 100-m stack.

The overall plant DFs for both ^{129}I and ^{131}I in the gaseous effluent are conservatively estimated⁷¹ to be 10^3 for the combination of mercuric nitrate-nitric acid scrubber and silver zeolite bed based on 90% of the iodine being vaporized from the dissolvers into the off-gas. It is estimated that dissolver solutions entering the solvent extraction systems will contain up to 10% of the iodine from which significant amounts of organic iodides could be formed. Essentially all of the organic iodides entering the off-gas system will be removed by the mercuric nitrate-nitric acid scrubber and by the silver zeolite sorbant.²⁵ However, such iodides passing into the aqueous waste would be expected to limit the efficiency of iodine removal in this waste.

The feasibility of maintaining the plant DFs has been analyzed in terms of the reliability of the equipment and the consequences of anticipated operational occurrences such as equipment failures. The plant would be shut down when any significant equipment failure occurred. The HEPA filters will be tested on a scheduled basis to ensure that adequate performance is maintained. Spare, parallel filters will also be provided. These parallel units will be brought into service when tests indicate that the efficiency of the "in service" filters has ceased to satisfy Technical Specifications. The mercuric nitrate-nitric acid scrubber system is a standard engineering unit that is expected to operate indefinitely without interruption. Spare pumps will be included in the installation to ensure the continuous movement of the scrub solution. The silver zeolite absorbers will be present as additional subsequent treatment systems, and since they do not have movable components, they represent very reliable operating units. Dissolving operations would be shut down if a problem should arise in the performance of these iodine retention units. In this event, less than 0.4 metric ton of fuel would continue to

dissolve, as described under "Fuel dissolver system" in Sect. 4.5.1. This small amount of fuel contains less than 0.026% of the iodine processed each year, and all of it could be released without exceeding the limit placed by the plant DF of 40 for ^{131}I , which is equivalent to a release of 2.5%. Actually, up to about 100 similar incidents could be tolerated annually on this basis, so that the iodine in a total of about 40 metric tons of fuel could be released.

The anticipated operational occurrence in which there is an unexpected leakage of dissolver solution, or other process solutions, onto the floor of the operating cells must also be considered in defining the limits of retaining a plant DF of 40 for ^{131}I . A fraction of the iodine could be volatilized and released through the cell ventilation system. The loss of a large volume of solution before the leak is discovered by the sensitive instrumentation in the cells and ventilation air stream, and operations are terminated is not credible. Actually, a significant leakage of dissolver solution (the most radioactive solution) could be tolerated without exceeding the release limit for iodine of 0.1% ($\approx$ to a plant DF of 10^3). This would amount to about 20 liters/day in cases where the dissolver solution has a uranium concentration of 320 g/liter. This assumes that all of the iodine is volatilized from the leaked solution.

Similarly, up to 0.02% of the fuel (approximately 1 kg/day) could be tolerated as an aerosol (particulates) dispersed into the ventilation air from the leaked solution without exceeding the specified plant DF for particulate retention of $>5 \times 10^8$. This assumes that the DF for particulates across the roughing filter and two HEPA filters is 1×10^5 . Up to 0.0004% of the fuel could be tolerated as an aerosol if only one HEPA filter is used. However, the leakage of these large volumes of highly radioactive dissolver solution would represent an intolerable situation and operations would be terminated immediately.

Liquid effluent treatment. The aqueous waste from the solvent extraction step is expected to contain up to 10% of the iodine, none of the krypton, none of the ^{14}C , all of the tritium (as tritiated water), 0.5% of the uranium and plutonium, and essentially all of the other fission products and transplutonium elements initially present in the irradiated fuel. These solutions are concentrated by evaporation, sampled, and stored indefinitely in large underground tanks. The condensate from the evaporators is mixed with the low-activity liquid waste (LALW) and reevaporated, and the bottoms are recycled to the high-activity waste evaporator. The overhead from the LALW evaporator is fed to a nitric acid fractionator. The bottom product from the fractionator is recycled to nitric acid storage for reuse in dissolution. The overhead product is excess water, which is to be purified before discharge to the environment. At this point, the overhead liquid potentially could contain most of the $\sim 10\%$ of the iodine that remained in the liquid phase from solvent extraction.

The iodine removal partial evaporator shown in Fig. 4.3 receives most of the iodine that passes through the solvent extraction system (up to 10% of the iodine that enters the plant) plus a portion of the iodine that is trapped in the mercuric nitrate-nitric acid scrubber; the latter iodine includes the 90% of the plant input that reports to the vapor phase during dissolving operations plus iodine recycled from the concentrated miscellaneous waste tank due to sparging. About 15% of the liquid and about 50% of the iodine entering the iodine removal partial evaporator will be vaporized, condensed, and routed to the miscellaneous liquid waste evaporator. The liquid stream from the iodine removal partial evaporator, containing only about 5% of the iodine (including 2.5% of the ^{131}I) that enters the plant, will then be vaporized, and the vapor will be discharged up the stack. This vapor will be superheated to prevent condensation until it can become mixed with the large volume of ventilation air.

The miscellaneous liquid waste (MLW) will be neutralized and concentrated in the MLW evaporator; about 1% of the iodine will be vaporized with the overhead vapor. Neutralization increases the efficiency of retention of iodine during evaporation. The overhead condensate will

flow to the iodine removal partial evaporator. The hot MLW evaporator concentrate will be discharged to the miscellaneous waste storage tank where, upon cooling, solids crystallize from the solution. A fraction of the iodine is expected to be immobilized in the solids.

The plant DFs for iodine in the liquid waste treatment system in this study are estimated by using the following assumptions:

- (1) Essentially all of the ^{131}I and ^{129}I that enters the plant accumulates in the miscellaneous waste storage tank. Approximately 90% comes from the mercuric nitrate-nitric acid scrubber and 10% from the solvent extraction raffinates.
- (2) The inventory of ^{131}I in the waste tank (MWS) becomes constant after about 80 days, and equal to 11.6 (from $8.05 \ln 2$) times the daily input, because of the decay of the short-lived ^{131}I (half-life, 8.05 days). However, the amount of ^{129}I (half-life, 1.6×10^7 years) in the tank increases continuously during the time that the waste is not encapsulated.
- (3) About 10% of the iodine that enters the plant each day plus about 1% of the iodine (including recycle iodine) that is collected in the mercuric nitrate-nitric acid scrubber reaches the iodine removal partial evaporator in the overhead from the fractionator. The efficiency of the partial evaporator for removal of iodine is 50%. On these bases, and including an additional DF of ^{131}I for decay, a plant DF of 40 is estimated for ^{131}I for the liquid waste treatment system in this study.
- (4) The estimation of the plant DF for ^{129}I must include an assumption concerning the quantity of waste that will accumulate in the plant before solidification. As described earlier in this section, sparging of the concentrated miscellaneous waste tank (with air), transfer of the sparge gas to the off-gas treatment system, and return of the contained iodine to the MLW evaporator will reduce the DF for ^{129}I . Thus, a DF of 20 was chosen for this nuclide.

Summary. The considerations listed in Sect. 4.5.1 indicate that the overall plant DFs for ^{131}I and ^{129}I in Case 1 are limited by the DFs achieved in the liquid waste treatment system; it is feasible to maintain plant DFs of 40 for ^{131}I , 20 for ^{129}I , and 5×10^8 for particulates in the model plant in Case 1. The radioactive materials released to the atmosphere in Case 1 are ^{131}I , 34.6 Ci/year; ^{129}I , 2.5 Ci/year; tritium, 1.0×10^5 Ci/year; ^{85}Kr , 1.4×10^7 Ci/year; ^{14}C , 8.31×10^2 Ci/year; uranium, 4.2×10^{-5} Ci/year; plutonium, 8.0×10^{-1} Ci/year; semivolatiles, 7.0 Ci/year; and particulates, 7.0 Ci/year.

4.5.2 Case 2a

Case 2a, as shown in Table 4.3, provides an increase in the overall DF for iodine, both ^{129}I and ^{131}I , by the addition of either an ion exchange or macroreticular resin sorption column downstream from the iodine removal partial evaporator (Fig. 4.4). Such a bed will further reduce the iodine concentration in the water before it is vaporized and discharged to the atmosphere. Moore and Howerton^{59,60} have demonstrated that macroreticular resins can provide DFs for iodine in aqueous solutions of 10^2 . However, such values were obtained with laboratory-size equipment; no engineering-scale measurements have yet been made. As a result, the increase in iodine DF ascribed to this unit has been estimated to be 50. Thus, the net effect of this unit is to increase the overall plant DFs for ^{129}I and ^{131}I to 5×10^2 and 2×10^3 , respectively (Table 4.3).

Gaseous effluent treatment. The off-gas treatment is the same as in Case 1, in which the fractional release of both ^{129}I and ^{131}I is 1×10^{-3} of the iodine entering the plant.

Liquid effluent treatment. As stated above, the difference between Case 2a and Case 1 is the resin sorption bed in the former that provides an additional DF of 50 for iodine. The fractional release of ^{129}I to the atmosphere via the liquid route is 1×10^{-3} (approximately 0.1/100); the corresponding number for ^{131}I is 5×10^{-4} (approximately 0.1/200). Even with this increased DF, the limiting path for iodine discharge is still the aqueous path.

Summary. This case represents an improvement in the plant DF for ^{129}I to 5×10^2 and for ^{131}I to 2×10^3 , both with respect to Case 1, by installation of a single iodine sorption unit upstream from the excess water vaporizer. Releases of ^{129}I and ^{131}I will be limited to 1.0×10^{-1} and 6.92×10^{-1} Ci/year, respectively, as contrasted with values 25 and 50 times higher in Case 1.

4.5.3 Case 2b

In Case 2b, the overall plant DF for iodine will be improved by an additional factor, with respect to Case 2a, for both ^{131}I and ^{129}I (Fig. 4.5). Iodine evolution equipment will be installed to vaporize iodine from the dissolver solution so that 99.5% of the iodine will enter the primary off-gas treatment system and about 0.5% will enter the liquid radwaste treatment system.

In order to avoid sending iodine collected in the gaseous effluent mercuric nitrate-nitric acid scrubber system to aqueous waste, equipment is added to recycle the mercury and to convert the retained iodine to a nonvolatile solid, sodium iodate, which is stored as a radioactive waste or packaged for disposal (Sect. 4.3.5). Thus, twenty-fold less iodine enters the liquid waste treatment system than in Case 1. This results in the release of less iodine from the aqueous treatment system to the vessel off-gas system and provides overall plant DFs in excess of 1×10^4 for ^{129}I and 1×10^4 for ^{131}I , as shown in Table 4.3.

Gaseous effluent treatment. The off-gas treatment system is the same as in Case 1 except for the addition of a second mercuric nitrate-nitric acid scrubber to remove iodine from the dissolver off-gas and iodine evolution tanks (Fig. 4.5). The vessel off-gas from the solvent extraction system contains 20 times less organic iodides than in Case 1; consequently, the DF for ^{129}I and ^{131}I will be at least 10^4 .

The reliability of the mercuric nitrate-nitric acid scrubber and silver zeolite systems is discussed in Sect. 4.5.1 for Case 1, and similar considerations apply to Case 2b. The iodine evolution and iodine isolation systems, when fully developed, will represent standard engineering operations, and continuous long-term service is expected.

Liquid effluent treatment. The treatment of the liquid effluents is the same as in Case 2a (Sect. 4.5.2). In this case, however, only about 0.5%, instead of 10% as in Cases 1 and 2a, of the iodine entering the plant each day will reach the iodine removal partial evaporator and only 0.5% instead of 100% of the iodine will reach the miscellaneous waste storage tank. The remainder of the iodine will be isolated and stored or packaged as sodium iodate in the mercury recycle system. Consequently, the DFs for ^{129}I and ^{131}I across the liquid waste treatment system are increased to values in excess of 1×10^4 and 1×10^4 , respectively.

Summary. The addition of three unit operations (iodine evolution, mercury recycle, and a second mercuric nitrate-nitric acid scrubber) to the process described in Case 2a, will provide overall plant DFs for ^{129}I and ^{131}I of 1×10^4 and 1×10^4 , respectively. In this case, the plant DFs will be limited by release of iodine from the gaseous effluent treatment system; that is, the fractional releases of ^{129}I and ^{131}I shown in Table 4.2 are 0.99×10^{-4} in the gaseous effluent and 0.5×10^{-4} and 0.25×10^{-4} , respectively, in the liquid effluent. The radioactive materials released to the atmosphere include ^{129}I , 5.0×10^{-3} Ci/year, and ^{131}I , 1.4×10^{-1} Ci/year, as compared with 2.5 and 3.46 Ci/year, respectively, for Case 1. All other nuclides will be the same as in Case 1.

4.5.4 Case 2c

Case 2c (Fig. 4.6) is similar to Case 2b except for the replacement of the mercuric nitrate-nitric acid scrubbers of the latter by the Iodex system in the former (Sect. 4.3.7). In this case the overall plant DFs for both ^{129}I and ^{131}I will also be 1×10^4 (Fig. 4.6 and Table 4.3). The Iodex system contains equipment for removing iodine from the off-gases and for converting the iodine to a solid that is stored or packaged for disposal (Sect. 4.3.6). The iodine evolution system installed in Case 2b and silver zeolite traps are retained.

Gaseous effluent treatment. The combined off-gas streams from the dissolver, iodine evolution, and various vessels will be passed through the Iodex system, where $>99.9\%$ of the iodine will be removed. Except for eliminating the mercuric nitrate-nitric acid scrubbers and the mercury recycle and iodine isolation systems, the remainder of the off-gas treatment system will be the same as in Case 2b. The iodine retained in the Iodex system will be converted to a solid (HI_3O_8) which will be stored or packaged for disposal. Most of the mass of the iodine, approximately 1.4 kg/day, will be retained in the Iodex system. This will reduce the load on the silver zeolite system and increase the life of the zeolite bed. The overall DF for ^{131}I and ^{129}I for the total gaseous effluent treatment system will be $>100,000$. The iodine evolution and Iodex systems will represent standard engineering operations when they are fully developed, and continuous long-term service is expected.

Liquid effluent treatment. The liquid waste treatment system will be the same as that described for Case 2b in Sect. 4.5.3. The DFs for ^{129}I and ^{131}I in this system are both 1×10^4 .

Summary. The considerations listed in Sect. 4.5.1 indicate that the overall plant DFs for ^{131}I and ^{129}I in Case 2c will be limited by the DFs achieved in the liquid waste treatment system and by leakage of process solutions into ventilated spaces, such as the processing cells; it will be feasible to maintain overall plant DFs of 1×10^4 for both ^{131}I and ^{129}I . Case 2c will reduce the amount of iodine released to the atmosphere to 5.0×10^{-3} Ci/year for ^{129}I and 1.4×10^{-1} Ci/year for ^{131}I as compared with 2.5 and 34.6 Ci/year, respectively, for Case 1. All other nuclides are the same as in Case 1.

4.5.5 Case 3

In Case 3, the amount of ^{14}C and ^{85}Kr released to the atmosphere is decreased by a factor ≥ 100 (Fig. 4.7). This is accomplished by installing selective absorption equipment in the primary off-gas system downstream of the NO_x absorber. A catalytic oxidizer is installed upstream of the selective absorption equipment to assure that the ^{14}C is in the form of $^{14}\text{CO}_2$. The noble gases and carbon dioxide are absorbed in a fluorocarbon at a low temperature ($< 0^\circ\text{C}$), stripped, compressed, and bottled in cylinders. It is assumed that the noble gases and carbon dioxide are separated prior to bottling. The cylinders can be stored for about 1 year in existing shielded space or in the fuel storage canal (Table 6.3). The amount of radioactive krypton and carbon released to the atmosphere annually is 1.4×10^5 and 8.31 Ci as compared to 1.4×10^7 and 8.31×10^2 Ci for Case 1.

In pilot-plant tests, 99.9% of the krypton and carbon dioxide⁷⁶ in the feed gas were removed by the selective absorption process; it is expected that a similar removal efficiency can be obtained in a commercial reprocessing plant (Sect. 4.3.8). However, for this study, a conservative removal efficiency of 99% (DF = 100) is selected. Spare (parallel) equipment components are provided for the critical items in the absorption system to ensure continuous operation whenever the shear and dissolver are in operation. If the operation of the absorption system should fail, the dissolver would be shut down and less than 0.04 metric ton of fuel would continue to dissolve (Sect. 4.5.1, *Fuel dissolver system*). Actually, the krypton and carbon dioxide in 15 metric tons of fuel could be released on an annual basis without exceeding the release limit of 1%. These considerations indicate that it will be feasible to maintain a plant DF of 100 for krypton and ^{14}C in the model plant after the process is fully developed and has been demonstrated with irradiated fuels in a pilot plant or commercial plant.

4.5.6 Case 4

In Case 4, the release of uranium, plutonium, and other nonvolatile fission products (particulates) and semivolatile ruthenium is decreased by a factor ≥ 10 by the addition of a sand filter to the off-gas system upstream of the 100-m stack (Fig. 4.8). All noncondensable and condensable off-gas is passed through the sand filter before discharge through the 100-m stack.

Gaseous effluent treatment. The process off-gas (DOG-VOG) treatment system is the same as for Case 1 except that a sand filter is added in series with the two HEPA filters. The off-gas passes upward through the sand filter (30-in.-high sand bed), which is equivalent to an additional HEPA filter (Sect. 4.3.1). This provides for a third stage of filtration for the DOG and VOG as well as a third stage of filtration for the cell and laboratory ventilation off-gas.

Liquid effluent treatment. The system for treatment of liquid effluent is the same as for Case 1 except that, after final vaporization, superheating, and dilution with cell and laboratory ventilation off-gas, the excess water to be discharged to the atmosphere passes through the sand filter. Filtration was not provided for the vapor stream in Case 1.

The radioactive materials released to the atmosphere in Case 4 are uranium, 4.2×10^{-6} Ci/year; plutonium, 8.0×10^{-2} Ci/year; ruthenium, 7.0×10^{-1} ; and other particulates, 7.0×10^{-1} Ci/year as compared to 4.2×10^{-5} , 8.0×10^{-1} , 7.0, and 7.0 Ci/year, respectively, for Case 1. The release of all other nuclides is the same as for Case 1.

4.5.7 Case 5

In Case 5 (Fig. 4.9), the release of semivolatile fission product ruthenium is decreased by a factor ≥ 10 and the release of ^{14}C is decreased by a factor of 5. A caustic scrubber is added to the secondary off-gas treatment system downstream of the mercuric nitrate-nitric acid scrubber to remove $\geq 90\%$ of the semivolatile fission product ruthenium in the combined (DOG-VOG) off-gas, and the same scrubber will remove about 80% of the ^{14}C as sodium carbonate (Sect. 4.3.10). The release of semivolatile fission products and ^{14}C to the atmosphere are 7.0×10^{-1} and 1.66×10^2 Ci/year as compared to 7.0 and 8.31×10^2 Ci/year for Case 1.

4.5.8 Case 6

In Case 6, the amount of tritium released to the atmosphere is decreased by a factor of 100, the amount of ^{129}I by a factor of 1.5, and the amount of ^{131}I by a factor of 2.5 (Fig. 4.10). This is accomplished by adding a voloxidation treatment step to the head-end system to remove tritium and iodine (Sect. 4.3.7). Laboratory development studies indicate that $\sim 99\%$ of the tritium and up to 75% of the iodine is evolved from irradiated UO_2 when the fuel is oxidized in air at temperatures in excess of 350°C . Using the voloxidation process, it is expected that in a commercial plant the same evolution efficiency can be achieved for tritium and about 85% of the laboratory-scale removal efficiency for iodine; the tritium can be collected as a small volume of tritiated water for permanent storage. It is assumed that the iodine will pass into the process off-gas system. The tritium and iodine evolution would be carried out in a heated rotary kiln (continuous voloxidizer) located between the shear and dissolver. It is estimated that the inventory of sheared fuel in the voloxidizer could be equivalent to about 1.5 metric tons of uranium; consequently, a malfunction in the voloxidation process could result in the release to the atmosphere of up to 770 Ci of tritium (514 Ci per metric ton of uranium) before the voloxidizer is cooled enough to end the volatilization of tritium. However, a DF of 100 (99% retention) allows for the release of the tritium in approximately 15 metric tons of fuel on an annual basis (about 7000 Ci per year). These considerations indicate that it will be feasible to retain 99% (DF = 100) of the tritium after the voloxidation process is fully developed and has been demonstrated with radioactive fuels in a pilot plant or commercial plant. It is expected that, when a fully developed voloxidation process is used, a tritium release rate of $< 9,000$ Ci/year can be achieved.

The volatilization of about 64% of the total iodine in the fuel into the off-gas treatment system will result in an improved iodine retention. The iodine entering the dissolver will split in the same way as in Case 1, with 90% going to the off-gas and 10% going to the liquid. Consequently, the total iodine entering the off-gas will be about 96%, and the estimated overall plant DFs for ^{129}I and ^{131}I are 30 and 100 as compared to 20 and 40 for Case 1.

Gaseous effluent treatment. The gas stream from the shear, consisting of a small amount of elemental tritium and tritiated water vapor, is combined with the off-gas from the voloxidation step, which contains >99% of the tritium as tritiated water vapor and about 64% of the iodine. The tritiated water is separated from the air and volatile fission products and packaged for storage and disposal.

Liquid effluent treatment. The liquid effluent treatment system is the same as that used for Case 1. The tritium released to the atmosphere is 7.7×10^3 Ci/year as compared with 7.7×10^5 Ci/year for Case 1. The iodine released to the atmosphere is 1.7 Ci/year of ^{129}I and 13.8 Ci/year of ^{131}I as compared to 2.5 and 34.6 Ci/year, respectively, for Case 1. The release of all other nuclides is the same as for Case 1.

4.5.9 Case 6c

Case 6c, which is a composite of Cases 2a through 6, represents an advanced fuel reprocessing plant based on current design concepts for controlling the release of gaseous and liquid effluents (Fig. 4.11). Technology is used that is commercially available or in various stages of development. The advanced radwaste treatment systems, such as selective absorption, voloxidation and Iodox, are not now commercially available; thus, Case 6c is a projection of advanced technology. No attempt has been made to optimize the combination of two or more of the radwaste treatment cases. The cases can be independently combined with the exception of Cases 2b and 2c. Either Case 2b or 2c would be used, but not a combination of both, since in Case 2c the mercuric nitrate-nitric acid scrubbers and the mercury recycle system are replaced with the Iodox process. For Cases 2b and 2c to be most effective, they must include Case 2a.

Gaseous effluent treatment. The primary off-gas treatment (DOG) system consists of a voloxidation step for the isolation of tritium; iodine evolution; an NO_x absorption step; and selective absorption equipment including a catalytic oxidizer for the removal of noble gases and ^{14}C . The secondary off-gas treatment system (DOG and VOG) consists of an Iodox unit, including equipment for the isolation and packaging of iodine; a caustic scrubber; a silver zeolite adsorber; two HEPA filters; and a sand filter. The cell and laboratory ventilation off-gases pass through two HEPA filters and are combined with the dissolver and vessel off-gas upstream of the sand filter. All the gaseous effluent is discharged from the 100-m stack.

Liquid effluent treatment. The liquid effluent treatment system is the same as that for Case 4. The radioactive materials released to the atmosphere in Case 6c are ^{129}I , 5.0×10^{-3} Ci/year; ^{131}I , 1.4×10^{-1} Ci/year; tritium, 7.7×10^3 Ci/year; ^{85}Kr , 1.4×10^5 Ci/year; ^{14}C , 8.31 Ci/year; semivolatiles, 7.0×10^{-2} Ci/year; uranium, 4.2×10^{-6} Ci/year; plutonium, 8.0×10^{-2} Ci/year; and particulates, 7.0×10^{-1} Ci/year. These releases compare with 2.5, 34.6, 7.7×10^5 , 1.4×10^7 , 8.31×10^2 , 7.0, 4.2×10^{-5} , 8.0×10^{-1} , and 7.0 Ci/year, respectively, for Case 1.

4.5.10 Case 7

Case 7 represents an advanced concept of a reprocessing plant that is designed to reduce the release of radioactive materials to a "near zero" level (Fig. 4.12). In contrast to Cases 2 through 6c, where additional equipment is added to the Case 1 plant to produce the desired objectives, the use of the "zero release" concept requires the construction of a new plant on a significantly different basis. This concept is based on a study in which extrapolations of current and developing technology were used to minimize the release of radioactive materials.³¹ The study indicates that significant reductions in the release of radioactive materials can be achieved by integrating advanced effluent control systems with new concepts of containment and ventilation that would (1) reduce net leakage of air to the process enclosures, (2) reduce net input of liquids into the process streams, and (3) provide for extensive recycle of liquids and gases. As a result of these measures, the volumes of waste effluents to be treated or stored are greatly reduced and the efficiency and the variety of effluent treatment methods to be considered are increased. With this concept, a high degree of overall containment must be maintained during all phases of plant functions, including routine operation, maintenance, and the decommissioning phase at the end of a plant's useful life.

The reduction of gaseous effluent volumes by orders of magnitude requires new approaches to cell containment and ventilation. The equipment is housed in sealed-cell enclosures, which limit net leakage of air to minimal volumes. In the ideal case, the volume of the leakage gas would be sufficiently small (<100 cfm) to flow into the equipment off-gas system. To further minimize effluent volumes, process liquid and air requirements are supplied by recycle systems. Only the small volume of gas representing the net leakage to the cell enclosure is given final treatment and released to the atmosphere. The small volume of liquid, representing the net input to the plant, is treated to remove essentially all of the radioactive materials other than tritium and is then permanently stored.

The principal problem concerning recycle of process gases and cell atmosphere is related to the lower practical limit of leakage to the cells, and from the cells into the process equipment. Aqueous reprocessing facilities have traditionally operated with large net flows of air into cells and certain process vessels. A large shielded fuel examination facility (the High-Level Fuel Examination Facility at the National Reactor Testing Station, Arco, Idaho) is operating with an air leakage rate of 0.004 cfm; this facility was constructed at a cost of about \$10 million. A practical leakage rate for a reprocessing facility built with the intent of obtaining a low air leakage rate appears to be about 100 cfm or less. It is possible to design, construct, and operate process equipment, including the shear and dissolver, so that the total leakage to the head-end process equipment will not exceed 100 cfm. Much lower rates may be attainable. A fuel reprocessing plant effluent-control flowsheet is shown in Fig. 4.13.

Iodine. The iodine control systems proposed for "near zero" release fuel reprocessing plants include five major areas of treatment:

- (1) evolution of iodine from the dissolver solution (Sect. 4.3.4);
- (2) primary iodine removal from head-end off-gas (Sect. 4.3.6);
- (3) secondary iodine removal from total cell and equipment off-gas (Sect. 4.3.6);
- (4) final iodine removal from plant gaseous effluents (Sect. 4.3.3);
- (5) iodine removal from recycle process water and acid (Sect. 4.3.11).

Specific case studies indicate overall tritium DFs in the range of 10^5 to 10^8 . A DF of 10^1 is selected for Case 7.

Krypton and ^{14}C . The retention of krypton and ^{14}C is limited by the ability to route all krypton-containing streams and ^{14}C to the primary krypton and ^{14}C removal system. The major sources of krypton and ^{14}C include the dissolver off-gas and the gas streams existing from the tritium removal system. Other minor sources include the off-gases from the processing cells and the storage and handling cell. In Case 7, the plant is designed so that these gases are recycled and the net off-gas passes through the primary krypton removal unit. An overall DF for krypton and ^{14}C of 10^4 is selected for Case 7.

Particulates. Removal of particulate activity from effluents to levels four to five orders of magnitude below current practice should be possible by an extrapolation of current technology to small gaseous streams. The off-gas from the vessel off-gas treatment system for the "near zero" release concept should be similar in particulate content to the off-gas from existing plants. Due to the small volumes, extensive treatment for particulate removal in the final off-gas treatment system should yield the desired results. An overall DF for particulates (nonvolatile and semivolatile fission products, uranium, and plutonium) of 1×10^{12} is selected for Case 7.

The release of radioactive materials for Case 7 as compared to Case 1 and Case 6c can be summarized as follows:

Nuclide	Case 7 (Ci/year)	Case 1 (Ci/year)	Case 6c (Ci/year)
Tritium	7.7×10^2	7.7×10^5	7.7×10^3
Krypton	1.4×10^3	1.4×10^7	1.4×10^5
Carbon-14	8.3×10^{-2}	8.31×10^2	8.31×10^0
Iodine-129	1.0×10^{-5}	2.5	5.0×10^{-3}
Iodine-131	2.8×10^{-4}	34.6	1.4×10^{-1}
Semivolatiles	7.0×10^{-4}	7.0	7.0×10^{-1}
Uranium	2.0×10^{-9}	4.2×10^{-5}	4.2×10^{-6}
Plutonium	1.6×10^{-4}	8.0×10^{-1}	8.0×10^{-2}
Particulates	3.5×10^{-3}	7.0	7.0×10^{-1}

The application of these systems to the overall retention of iodine is indicated in Fig. 4.14. The iodine treatment methods listed above are described in the sections indicated and are used in Cases 2a, 2b, 2c, and 6c. The difference in Case 7 is the recycle of the cell off-gas and the removal of the excess cell off-gas through the process off-gas system. An overall iodine DF of 5×10^6 is assigned to this system for Case 7.

Tritium. The containment of tritium in a fuel reprocessing plant can be achieved by evolution and retention of tritium in an initial step (voloxidation), total water containment, or a combination of both methods (Fig. 4.14). The overall plant containment factor for tritium that can be achieved by voloxidation is limited by the inability to totally evolve tritium from the fuel as well as to the escape of small quantities of tritium from failed fuel elements during

storage and handling. As in Case 6, a practical upper limit on tritium containment factors based on this method is 100. If water containment is used to control tritium release, an overall plant containment factor in excess of 10^4 appears to be achievable, assuming that the total plant off-gas rates are about 500 cfm and the total water input to the plant is kept sufficiently low so that the volume for long-term accumulation or permanent disposal is not excessive.

4.6 References for Section 4

- 1 Allied-Gulf Nuclear Services, *Barnwell Nuclear Fuel Plant - Environmental Report*, Docket No. 50-332 (November 1971)
- 2 General Electric Company, *Applicant's Environmental Report*, NEDO-14504 and NEDO-14504-2 (1971)
- 3 Code of Federal Regulations, Title 10, Part 50, Appendix F
- 4 Code of Federal Regulations, Title 10, Part 20, Appendix B, Table 2, Column 1
- 5 M. J. Bell, *ORIGEN - The ORNL Isotope Generation and Depletion Code*, ORNL-4628 (May 1973)
- 6 B. C. Finney, R. E. Blanco, R. C. Dahlman, F. G. Kitts, and J. P. Witherspoon, *Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing "as Low as Practicable" Guides - Nuclear Fuel Reprocessing*, ORNL/TM-4901 (May 1975)
- 7 W. Davis, Jr., *High Efficiency Particulate Air Filters - State of the Art Summary Pertaining to Plutonium Aerosols*, ORNL/TM-4463 (April 1974)
- 8 Nils Hetland and J. L. Russell, Jr., "Adequacy of Ventilation Exhaust Filtering System for New Plutonium Facilities," *Proceedings of the 12th AEC Air Cleaning Conference Held in Oak Ridge, Tennessee, Aug. 28-31, 1972*, CONF-720823 (January 1973)
- 9 F. J. Linck and J. A. Geer, "In-Place Testing of Multiple Stage HEPA Filters," *Proceedings of the 13th AEC Air Cleaning Conference Held in San Francisco, Calif., Aug. 10-15, 1974*, CONF-740807
- 10 A. J. Oliver and C. L. Amos, "Ventilation Design for New Plutonium Recovery Facility," *Proceedings of the 13th AEC Air Cleaning Conference Held in San Francisco, Calif., August 10-15, 1974*, CONF-740807
- 11 H. J. Ettinger *et al.*, "Size Characteristics of Plutonium Aerosols," p. 740 in *Proceedings of the 12th AEC Air Cleaning Conference Held in Oak Ridge, Tennessee, Aug. 28-31, 1972*, CONF-720823 (January 1973)

- 12 H J Ettinger, J C Elder, and M Gonzales, *Performance of Multiple HEPA Filters Against Plutonium Aerosols, Progress Report for Period January 1 through June 30, 1973*, LA-5349-PR (July 1973)
- 13 M Gonzales, J C Elder, and H J Ettinger, "Performance of Multiple HEPA Filters Against Plutonium Aerosols" in *Proceedings of the 13th AEC Air Cleaning Conference Held in San Francisco, Calif, August 10-15, 1974*, CONF-740807
- 14 USAEC Regulatory Guide, August 1973, Regulatory Guide 3.12, "General Design for Ventilation Systems of Plutonium Processing and Fuel Fabrication Plants"
- 15 D W Moeller, "Performance of Air Cleaning Systems in Nuclear Facilities," in *Semiannual Progress Report, March 1, 1974-August 31, 1974, Harvard Air Cleaning Laboratory*, COO-3049-5 (December 1974)
- 16 C A Burchsted, Oak Ridge National Laboratory, personal communication, January 1974
- 17 "Development of New and Improved Filter Medium Resistant to Fire, Water, and Fluorides," Final Report (No. 31) by the Herty Foundation for the USAEC on Contract No. AT(38-1)-411, Project 586 (Oct. 31, 1973)
- 18 G H Sykes and J A Harper, "Design and Operation of a Large Sand Bed for Air Filtration," IAEA Symposium on Treatment of Airborne Radioactive Wastes, Vienna, August 26-30, 1968
- 19 R A Juvinall, R W Kessie, and M J Steindler, *Sand-Bed Filtration of Aerosols: A Review of Published Information on Their Use in Industrial and Atomic Energy Facilities*, ANL-7683 (June 1970)
- 20 G A Schurr, D B Zippler, and D C Guyton, "Deep-Bed Filter Performances," in *Proceedings of the 12th AEC Air Cleaning Conference Held in Oak Ridge, Tennessee, August 28-31, 1972*, CONF-720823 (January 1973)
- 21 D B Zippler, "Evaluation of Multistage Filtration to Reduce Sand Filter Exhaust Activity," in *Proceedings of the 13th AEC Air Cleaning Conference Held in San Francisco, Calif, August 10-15, 1974*, CONF-740807
- 22 H W Godbee, *Use of Evaporation for the Treatment of Liquids in the Nuclear Industry*, ORNL-4790 (September 1973)
- 23 D T Pence, F A Duce, and W J Maeck, "Developments in the Removal of Airborne Iodine Species with Metal-Substituted Zeolites," *Proceedings of the 12th AEC Air Cleaning Conference Held in Oak Ridge, Tennessee, Aug. 28-31, 1972*, CONF-720823, Vol. 1 (January 1973)

- 24 R D Ackley and Zell Combs, *Applicability of Inorganic Sorbents for Trapping Radioiodine from LMFBF Fuel Reprocessing Off-Gas*, ORNL/TM-4227 (May 1973)
- 25 R D Ackley and R J Davis, *Effect of Extended Exposure to Simulated LMFBF Fuel Reprocessing Off-Gas on Radioiodine Trapping Performance of Sorbents*, ORNL/TM-4529 (August 1974)
- 26 W Davis, Jr , *Isotopic Exchange, Radioactive Decay and Recycle Effects in Removing Iodine from Gas and Liquid Streams*, ORNL-5060 (August 1975)
- 27 K J Schneider, *Effluent Technology Progress Letter No 7, June and July 1972*, DRAFT No 12314 (Aug 10, 1972) (OFFICIAL USE ONLY)
- 28 Allied-Gulf Nuclear Services, *Barnwell Nuclear Fuel Plant - Safety Analysis Report*, Vol 1, Docket No 50-332 (Apr 10, 1969)
- 29 *Final Detailed Statement by Fuels and Materials, Directorate of Licensing, U S Atomic Energy Commission on the Environmental Considerations Related to the Proposed Operation of the Midwest Fuel Recovery Plant, Morris, Illinois, by the General Electric Company*, Docket No 50-268 (December 1972)
- 30 *Draft Environmental Statement by Fuels and Materials, Directorate of Licensing, U S Atomic Energy Commission Related to the Barnwell Nuclear Fuel Plant, Allied-Gulf Nuclear Services*, Docket No 50-332 (January 1973)
- 31 O O Yarbrow, F E Harrington, and D S Joy, *Effluent Control in Fuel Reprocessing Plants*, ORNL/TM-3899 (March 1974)
- 32 W E Unger et al , *Aqueous Fuel Reprocessing Quarterly Report for Period Ending December 31, 1972*, ORNL/TM-4141 (April 1973)
- 33 W E Unger et al , *Aqueous Fuel Reprocessing Quarterly Report for Period Ending September 30, 1973*, ORNL/TM-4394 (February 1974)
- 34 W E Unger et al , *Aqueous Fuel Recycle Program Progress Report for Period April 1-December 31, 1973*, ORNL/TM-4488 (June 1974)
- 35 W E Unger et al , *Aqueous Fuel Reprocessing Quarterly Report for Period Ending March 31, 1973*, ORNL/TM-4240 (June 1973)
- 36 Staff of the Chemical Technology Division, *Chem Technol Div Ann Progr Rept Period Ending May 31, 1970*, ORNL-4572, p 69
- 37 J R Bower and J A Buckham, "Control of Fission Product Activity During Short-Cooled Fuel Reprocessing Connected with the ICPP RALA Process," International Atomic Energy Agency Panel on Reprocessing of Highly Irradiated Fuels, Vienna, Austria, May 27-30, 1969

38. W. S. Groenier, *An Engineering Evaluation of the Iodex Process: Removal of Iodine from Air Using a Nitric Acid Scrub in a Packed Column*, ORNL/TM-4125 (August 1973).
39. W. E. Unger et al., *Aqueous Fuel Reprocessing Quarterly Report for Period Ending June 30, 1973*, ORNL/TM-4301 (August 1973).
40. J. H. Goode (ed.), *Voloxidation - Removal of Volatile Fission Products from Spent LMFBR Fuels*, ORNL/TM-3723 (January 1973).
41. M. J. Stephenson et al., "Experimental Demonstration of the Selective Absorption Process for Krypton-Xenon Removal," *Proceedings of the 12th Air Cleaning Conference Held in Oak Ridge, Tennessee, Aug. 28-31, 1972*, CONF-720823, Vol. 1 (January 1973).
42. M. J. Stephenson, D. I. Dunthorn, W. D. Reed, and J. H. Pashley, "Absorption Process for Removing Krypton from the Off-Gas of an LMFBR Fuel Reprocessing Plant," *Proceedings of the 13th AEC Air Cleaning Conference, San Francisco, Calif., August 12-15, 1975*.
43. M. J. Stephenson, R. S. Eby, and J. H. Pashley, *Fluorocarbon Absorption Process for the Recovery of Krypton from the Off-Gas of Fuel Reprocessing Plants*, K-GD-1390 (January 1976).
44. J. R. Merriman, *Supplemental Testimony on Kr-85 Removal and Storage in the Matter of Barnwell Nuclear Fuel Plant*, U.S. Nuclear Regulatory Commission, Docket No. 50-332.
45. R. M. Hogg, "New Radwaste Retention System," *Nucl. Eng. Intern.* 17(189), 98-99 (1972).
46. J. P. Nichols and F. T. Binford, *Status of Noble Gas Removal and Disposal*, ORNL/TM-3515 (August 1971).
47. C. L. Bendixsen and G. F. Offutt, *Rare Gas Recovery Facility at the Idaho Chemical Processing Plant*, IN-1221 (April 1969).
48. C. L. Bendixsen and K. L. Rohde, "Operational Performance and Safety of a Cryogenic System for Krypton Recovery," *Trans. Am. Nucl. Soc.* 15(1), 96 (1972).
49. J. S. Davis and J. R. Martin, "A Cryogenic Approach to Fuel Reprocessing Gaseous Radwaste Treatment," *Trans. Am. Nucl. Soc.* 16, 176-77 (1973).
50. *Draft Environmental Statement, Limerick Generating Station*, Docket 50-352 and 50-353.

51. *Draft Environmental Statement, Susquehanna Steam Electric Station*, Docket 50-387 and 50-388.
52. J. A. Buckham, *Supplemental Testimony on Kr-85 Removal and Storage, in the Matter of Barnwell Nuclear Fuel Plant*, U.S. Nuclear Regulatory Commission, Docket No. 50-332.
53. Staff of the Chemical Technology Division, *Aqueous Processing of LMFBR Fuels - Technical Assessment and Experimental Program Definition*, ORNL-4436 (June 1970), pp. 173-86.
54. J. L. McElroy et al., *Waste Solidification Program Summary Report, Vol. 11, Evaluation of WSEP High-Level Waste Solidification Processes*, BNWL-1667 (July 1972).
55. F. O. Cartan et al., "Evidence for the Existence of Hypoiodous Acid as a Volatile Iodine Species Produced in Water-Air Mixtures," *Proceedings of the 15th Conference on Remote Systems Technology, 1967*, pp. 342-49.
56. G. K. Cederberg and D. K. MacQueen, *Containment of Iodine-131 Released by the RALA Process*, IDO-14566 (Oct. 30, 1961).
57. J. B. Tepe and B. F. Dodge, "Absorption of Carbon Dioxide by Sodium Hydroxide Solutions in a Packed Column," *Trans. AIChE*, 39, 255 (1943).
58. N. A. Spector and B. F. Dodge, "Removal of Carbon Dioxide from Atmospheric Air," *Trans. AIChE*, 42, 827 (1946).
59. W. E. Unger et al., *LMFBR Fuel Cycle Studies Progress Report for December 1970*, No. 22, ORNL/TM-3281, p. 27.
60. W. E. Unger et al., *LMFBR Fuel Cycle Studies Progress Report for August 1970*, No. 18, ORNL/TM-3127, p. 33.
61. W. E. Unger et al., *LMFBR Fuel Cycle Studies Progress Report for November 1970*, No. 21, ORNL/TM-3250, p. 34.
62. Staff of the Oak Ridge National Laboratory, *Siting of Fuel Reprocessing Plants and Waste Management Facilities*, ORNL-4451 (July 1970).
63. Nuclear Fuel Services, Inc., *Safety Analysis - Spent Fuel Processing Plant*, Docket No. 50-201 (1962-1965).
64. General Electric Company, *Design and Analysis - Midwest Fuel Recovery Plant*, Docket No. 50-268 (1966-1967).

65. Allied Chemical Nuclear Products, Inc., *Barnwell Nuclear Fuel Plant - Safety Analysis Report*, Docket No. 50-332 (1969-1970).
66. *Draft Detailed Statement by the Division of Radiological and Environmental Protection, U.S. Atomic Energy Commission, on the Environmental Considerations Related to the Proposed Operation of the Midwest Fuel Recovery Plant, Morris, Illinois, by the General Electric Company*, Docket No. 50-268 (March 1972).
67. Allied-Gulf Nuclear Services, *Environmental Report - UF₆ Facility*, Docket No. 70-1327 (June 1972).
68. "Effluent Control in Fuel Reprocessing Plants," *Chem. Technol. Div. Ann. Progr. Rept. Mar. 31, 1972*, ORNL-4794, pp. 24-25.
69. O. O. Yarbrow, *Supplementary Testimony Related to Fission Product Source Terms and Effluents from Barnwell Nuclear Fuel Reprocessing Plant*, U.S. Nuclear Regulatory Commission, Barnwell Nuclear Fuel Plant, Docket No. 50-332 (October 1974).
70. O. O. Yarbrow, *Supplementary Testimony Regarding the State of Technology for and Practicability of Control and Retention of Iodine in a Nuclear Fuel Reprocessing Plant*, U.S. Nuclear Regulatory Commission, Barnwell Nuclear Fuel Plant, Docket No. 50-332 (October 1974).
71. W. Davis, Jr., *Models for Calculating the Effects of Isotopic Exchange, Radioactive Decay, and of Recycle in Removing Iodine from Gas and Liquid Streams*, ORNL-5060 (September 1975).
72. W. Davis, Jr. et al., *Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle - Reprocessing of High-Temperature Gas-Cooled Reactor Fuel Containing U-233 and Thorium*, ORNL/NUREG/TM-4 (in press).
73. W. S. Groenier, *Equipment for the Dissolution of Core Material from Sheared Power Reactor Fuels*, ORNL/TM-3194 (April 1971).
74. B. C. Finney et al., *Shear Leach Process: Semicontinuous and Batch Leaching of Sheared, Unirradiated, Stainless-Steel-Clad and Zircaloy-2-Clad UO₂ and UO₂-ThO₂*, ORNL-3984 (July 1969).
75. B. L. Vondra et al., *LWR Fuel Reprocessing and Recycle Program Quarterly Report for Period July 1 to September 30, 1976*, ORNL/TM-5660 (1976).
76. *LMFBR Fuel Recycle Program for Period January 1 to June 30, 1975*, ORNL/TM-5107 (March 1976).

5.0 MISCELLANEOUS RADIOACTIVE AND NONRADIOACTIVE WASTES

The operation of a nuclear fuel reprocessing plant will generate various radioactive and nonradioactive wastes, such as fuel element parts, discarded equipment, laboratory waste, sanitary waste, process cooling water, and combustion products. In addition, significant volumes of high- and low-activity liquid wastes are accumulated in storage tanks. Stainless steel cylinders filled with solid high-activity wastes will be produced at plants which have facilities for solidification of the liquid waste. Estimates of the amounts of these wastes and a discussion of methods for their management at the model reprocessing plant that handles 1500 metric tons of fuel per year are presented in the following sections. The information is derived from environmental reports describing a reprocessing plant now under construction and from survey reports from Oak Ridge National Laboratory.¹⁻⁵

5.1 Solid Radioactive Waste

Miscellaneous. The miscellaneous solid radioactive waste consists primarily of fuel element parts, discarded equipment, and laboratory wastes such as gloves, clothing, etc. Such waste is buried in a retrievable manner onsite above the groundwater level. The burial areas are prepared in a manner to minimize the percolation of water down over the waste containers. The packaging of the waste and preparation of the burial area will eliminate the leaching by water and migration of the radionuclides in the ground. If it is required at some later date, the waste could be placed in a shipping cask and transferred to a permanent disposal site. An estimate of the amounts of radioactive solid waste to be handled at a reprocessing plant now under construction is presented in Table 5.1.² Another estimate based on the experience at the nuclear Fuel Services reprocessing plant indicates that the total volume of wastes may approach 300,000 ft³/year in comparison to the maximum of 83,000 ft³ listed in Table 5.1.⁴

Fuel cladding (hulls) waste. The amount of fuel cladding (hulls) waste has been estimated as 15,000 to 30,000 ft³ annually for a 1500 metric ton/year plant.² It has also been estimated that the neutron-induced radioactivity in the hulls, such as ⁶⁰Co in stainless steel and ⁹³Zr and ⁹⁵Zr-Nb in the Zircaloy, and some unleached fuel and fission products could result in a radioactivity level of about 7000 Ci per cubic foot of waste.³ A separate estimate indicates a volume of 21,000 ft³ for uncompacted waste (14 ft³ per metric ton of fuel) for a 1500 metric ton/year plant or 3150 ft³, if the waste is compacted to 2.1 ft³ per metric ton of fuel (70% of theoretical density). The radioactivity in this waste is estimated to be about 10,000 Ci per metric ton of fuel produced after 1 year of decay.⁴ These wastes are expected to contain up to 0.1 wt % of the plutonium and other transuranium nuclides that enter the plant.

Iodide waste. The isolation of iodine in the mercury recycle or Iodex systems (Sects. 4.3.5 and 4.3.6) will produce about 5 lb per day of NaIO₃ or 4 lb per day of HI₃O₈, which is packaged in stainless steel cylinders and retained in storage prior to transfer to a waste repository.

Solid high-activity wastes. Estimates indicate that about 2 ft³ of solid high-activity waste will be produced from the solidification of the liquid waste residues from reprocessing 1 metric

ton of fuel.⁵ This amounts to an annual accumulation of 3000 ft³/year in 478 stainless steel cylinders that are 12 in. diam by 10 ft high (filled to the 8-ft level). Federal regulations state that solidified wastes can be stored on the reprocessing site up to 10 years prior to shipment to a waste repository.⁶ Each cylinder would contain about 970,000 Ci of alpha and beta activity and 3.35 kW of heat after 10 years of decay.

5.2 Liquid Wastes

The annual rate of production of liquid wastes is estimated as 450,000 gal for high-activity waste and 300,000 gal for miscellaneous low-activity wastes, assuming 100 gal of high-activity, acidic waste concentrate per 10,000 MWd(t), an irradiation level of 33,000 MWd per metric ton of fuel, and 200 gal of miscellaneous waste per metric ton.⁴ Under these assumptions, the volume of high-level waste is 330 gal per metric ton of fuel processed. A separate estimate lists 60 gal of high-activity waste per metric ton of fuel (35,000 MWd/ton) or an annual rate of 90,000 gal.¹ These variations are the result of different operating conditions that introduce varying amounts of solids into the waste solutions and that limit the degree of concentration achieved.

5.3 Chemical and Sanitary Waste

Approximately 2300 gpm of primary and secondary cooling water would be required for a 5-metric ton/day reprocessing plant. This liquid is discharged to the environment. In-line radiation monitors sound alarms if radioactivity is detected in the primary cooling loop. A settling pond and cooling tower are included in the model plant cooling water system, and both the secondary cooling system and the cooling tower basin will be monitored. The entry of radioactive materials into the primary cooling and secondary cooling loops would require the failure of both of the heat exchangers at the same time and is an extremely remote possibility. The amounts of chemicals used for boiler treatment and cooling tower treatment are estimated as follows:

Chemical added	Amount (lb/day)
Polyacrylate	6
Sodium sulfite	1
Sodium sulfate	5
Trisodium phosphate	4
Calcium and magnesium phosphates	2

The sanitary waste treatment system handles about 15,000 gpd. The effluent from the sanitary waste treatment facility is chlorinated and discharged to a pond that provides several days retention, after which it is discharged to the environment. The estimated composition of the discharged sewage is presented (Table 5.2).

5 4 Nonradioactive Gaseous Effluents

In addition to the gaseous effluent released from the 100-m stack, nonradioactive gaseous effluents will be released from other stacks within the plant complex, such as utility boiler stacks and vessel vents from the cold chemical makeup area. The only nonradioactive chemical gaseous effluent of any consequence is comprised of the oxides of nitrogen in the service equipment off-gas (assumed to be NO₂) and a mixture of NO, NO₂, and N₂O in the process off-gas. The annual amount of NO₂ released from the service equipment, assuming operation for 300 days per year, is 240 metric tons. The annual amount of NO_x released in the process off-gas is 337 metric tons (Table 5 3).

5 5 References for Section 5

- 1 *Barnwell Nuclear Fuel Plant Environmental Report*, Allied-Gulf Nuclear Services, Barnwell, S C , Docket 50-332-22 (November 1971)
- 2 *Draft Environmental Statement by Fuels and Materials Directorate of Licensing, U S Atomic Energy Commission Related to the Barnwell Nuclear Fuel Plant, Allied-Gulf Nuclear Services*, Docket No 50-332 (January 1973)
- 3 *Safety Evaluation by the Directorate of Licensing - Fuels and Materials - U S Atomic Energy Commission in the Matter of Midwest Fuel Recovery Plant - General Electric Company*, Docket No 50-268 (November 1972)
- 4 Staff of the Oak Ridge National Laboratory, *Siting of Fuel Reprocessing Plants and Waste Management Facilities*, ORNL-4451 (July 1970)
- 5 J O Blomeke and J P Nichols, *Commercial High-Level Waste Projections*, ORNL/TM-4224 (May 1973)
- 6 Code of Federal Regulations, Title 10, Part 50, Appendix F

6.0 COSTS FOR RADWASTE TREATMENT

Costs for the radwaste treatment cases for the 1500-metric ton/year model fuel reprocessing plant are estimated as additions to the base case model plant. The capital costs, annual fixed charges, annual operating cost, total annual cost, reprocessing cost, and contribution to the cost of power for the radwaste treatment cases are summarized in Table 6.1. The incremental costs and the corresponding calculated amounts of radioactive materials released (source terms) are presented in Table 6.2. Annual fixed charges are estimated at 26% of total capital investment; this is typical of cost estimates for investor-owned reprocessing plants.¹ The basis for calculation of the fixed charge rate and the operating cost is presented in Sect. 6.2. The installed equipment costs are listed in Table 6.3. The annual operating expense is added to the annual fixed charge on capital to obtain the total annual cost for each radwaste treatment case. This cost is then divided by the annual amount of fuel reprocessed, or by the annual amount of electricity that was produced by the reprocessed fuel, to obtain the cost of radwaste treatment per weight of fuel reprocessed or the total contribution to the cost of power for each radwaste case. A fuel reprocessing plant with a nominal production rate of 1500 metric tons/year can service approximately fifty-five 1000-MW(electrical) LWRs (based on a burnup of 33,000 MWd/metric ton, 80% load factor, and 32.5% thermal efficiency). Costs are estimated in terms of 1973 dollars. No attempt is made to include the effect of inflation; however, based on the Marshall and Swift Cost Index² for chemical equipment, the 1975 costs would be about 30% higher than the 1973 costs. The cost estimates are expected to have an accuracy of about $\pm 30\%$. The details of the cost estimate are provided in Appendix A.

6.1 Capital Cost

The capital cost of the radwaste treatment cases is the sum of the direct cost and the indirect cost. The interest during construction and the contingency allowance are included as indirect costs to simplify the calculations.

6.1.1 *Direct costs*

The size and initial costs of the major equipment components are based on vendor bids, using early-1973 dollars, obtained by Burns and Roe, Incorporated.³ Appropriate costs based on experience in the nuclear industry³⁻⁶ are then added to the initial costs to allow for: (1) installation of the components, including piping, instrumentation, and controls; (2) modifications to provide for remote maintenance; and (3) fabrication upgrading (where necessary) to provide the required quality assurance.

Cell space requirements are estimated based on equipment size and the requirements for auxiliary equipment (pumps, condensers, etc.). The costs for the cells are estimated as Class 1 structures. The costs of a warehouse and other related facilities are not included. The total

direct cost for each radwaste treatment case is the complete, installed, equipment cost (material and labor), including the structure

6.1.2 Indirect costs

Indirect costs are estimated as follows

	Percentage of direct cost
Engineering and supervision	15
Construction expense and contractor's fee	20
Engineering design (A-E)	19
Quality assurance ^a	6
Other owner's cost	10
Contingency	40
Interest during construction ^b	39
	149

6.2 Annual Fixed Charges and Operating Costs

The annual fixed charges on invested capital are based on the Fuel Recycle Task Force⁷ annual fixed charge rate of 24%, which in turn is, based on the following assumptions

Plant lifetime, years	15
Capital investment in bonds, %	30
Capital investment in equity, %	70
Interest rate on bonds, %	5
Rate of return on equity (after taxes), %	16
Federal income tax rate, %	50
State income tax rate, %	3
Local property tax rate, %	3.2
Annual cost of replacements, %	0.35
Annual property insurance rate, %	0.25

^aQuality assurance costs based on the vendor's equipment bid prices obtained by Burns and Roe, Incorporated, do not reflect quality assurance costs anticipated for a nuclear fuel processing plant being built in the near future. An additional cost of 6% of the direct costs has been assumed to reflect costs for more stringent quality assurance.

^bInterest is applied to the cumulative total cost at a rate of 8% per year over a 5-year cash flow expenditure period.

By present-day standards, the 5% bond interest rate is probably low. Increasing it to 8% would increase the fixed charge rate to about 26%, and for this study a fixed charge rate on invested capital of 26% is used.

No attempt was made to perform a detailed analysis of each radwaste treatment case to estimate the annual operating and maintenance cost; however, since radwaste treatment is a part of a complete reprocessing plant, an annual operating cost of 40% of the annual fixed charge is used. This is in agreement with a previous fuel reprocessing cost estimate.¹ The total annual cost of each radwaste treatment case is the sum of the annual fixed charge and the annual operating cost.

6.3 Installed Equipment Costs

The estimated direct and capital costs for equipment in radwaste Cases 2a through 6c are presented in Table 6.3. The direct cost is the estimated installation (material and labor) cost of the equipment; the capital cost includes direct costs and indirect costs.

6.4 Determination of the Value of Spent BWR and PWR Fuels

The value of spent LWR fuel is the sum of the value of the uranium and the indifference value of plutonium contained in the spent fuels. The indifference value of plutonium is that value of plutonium which makes the total fuel cycle cost for an enriched uranium LWR (EUR) reactor equal to that of a plutonium recycle LWR (PRR) reactor. At this value, a reactor operator would be willing to use plutonium in place of ^{235}U .

The indifference value of plutonium⁸ can be determined by plotting the fuel cycle cost (\$/10⁶ Btu) vs the value of plutonium (\$/g fissile) according to the following procedure:

- (1) The curves are generated by calculating the fuel cycle cost for a ^{235}U -fueled reactor and for a ^{235}U -Pu-fueled reactor with arbitrarily assigned values of plutonium. The costs are levelized over 23 years in these calculations.

- (2) The EUR curve has a negative slope because the increase in value of the plutonium causes a decrease in the fuel cycle cost. Thus, it is more economical to consume ^{235}U (and a fraction of the plutonium) and to produce excess plutonium as the value of the plutonium increases. The costs of the processing steps within the EUR fuel cycle, such as shipping, reprocessing, conversion, and enrichment remain constant, and the only variable is the increasing value of plutonium.
- (3) The PRR curve has a positive slope because the fuel cycle costs increase as the value of the plutonium increases. In this reactor, plutonium is consumed in place of a fraction of the ^{235}U that is normally in an EUR. Thus, it is more economical to consume less plutonium as the cost of plutonium increases. Again, the costs of the other steps in the fuel cycle remain constant.
- (4) The point at which the two curves intersect is defined as the indifference value of plutonium. If the fuel cycle costs increase or decrease due to changes other than the cost of the plutonium the curves shift upward or downward.

Estimates of the fuel cycle costs and the indifference value for plutonium have been published by the Babcock & Wilcox Company (B&W).⁸ These estimates have been revised to use the parameters given below.⁹

- (1) A total of 2 years is allowed between the time of discharge of the fuel from the reactor until the recovered plutonium is returned to the reactor in refabricated fuel. The plutonium is treated as a discharge material for the first 10 months, and as a recycle material for the next 14 months. The costs are levelized over a 23-year period.
- (2) The individual costs are as follows:

U_3O_8	\$8/lb
Shipping	\$5/kg
Conversion	\$2.50/kg
Reprocessing	\$30/kg
Separative work	\$38/unit
U fabrication	\$65/kg
Pu fabrication	\$130/kg

- (3) The plutonium is recycled to the reactor in which it was produced.

The indifference value for plutonium calculated by B&W using the above costs is \$8.13 per gram of fissile plutonium for a PWR.⁹ A similar study performed by the General Electric Company for the Edison Electric Institute gave an indifference value of \$7.50 per gram of fissile plutonium for a BWR.⁹ Based on these estimates, an indifference value for plutonium for spent PWR and BWR reactor fuel of \$8 per gram of fissile plutonium is used in our current study.

The values of the fuels that are reprocessed in the model plant are estimated using the factors listed in Table 6.4. The estimated value of the spent BWR fuel is \$60.37 per kilogram of uranium charged to the reactor, the value of the spent PWR fuel is \$75.35 per kilogram of uranium charged to the reactor.

6.5 References for Section 6

1. Staff of the Oak Ridge National Laboratory, *Siting of Fuel Reprocessing Plants and Waste Management Facilities*, ORNL-4451 (July 1970).
2. *Chemical Engineering*, McGraw-Hill, New York, Jan. 5, 1976.
3. *Engineering Evaluation and Cost Estimate of Radwaste Treatment Facilities for Nuclear Fuel Reprocessing Plants* for Nuclear Division, Union Carbide Corporation, Subcontract No. 3894, Burns and Roe, Inc., W. O. 3048-01, December 14, 1973 (unpublished).
4. E. D. North and R. L. Booth, *Fission Product Gas Retention Study Final Report*, ORNL/TM-4409 (August 1973).
5. E. W. Murbach, W. H. Carr, and J. H. Gray, III, *Fission Product Gas Retention Process and Equipment Design Study*, ORNL/TM-4560 (May 1974).
6. *AEC Conceptual High-Temperature Gas-Cooled Reactor (HTGR) Fuel Processing Plant*, WASH-1152 (October 1970).
7. *Reactor Fuel Cycle Costs for Nuclear Power Evaluation*, WASH-1099 (December 1971).
8. E. L. Wright and J. S. Tulenko, "Plutonium Indifference-Value Sensitivity for a B&W Pressurized Water Reactor," *Trans. Am. Nucl. Soc.* 16, 169-70 (1973).
9. E. L. Wright, Babcock and Wilcox Company, personal communication, Aug. 3 and 6, 1973.

7.0 ENVIRONMENTAL IMPACT

The radiological impact of the model LWR fuel reprocessing plant is assessed by calculating radiation doses to individuals and populations for each site and radwaste treatment case. Potential pathways for radiation exposure to man from radionuclides originating in a nuclear facility are presented schematically in Fig. 7.1. Those shown in the figure are not exhaustive, but they illustrate the principal pathways of exposure based on experience.

Estimates of dose per year of plant operation to both individuals and to the population within a 55-mile radius, which may result from the expected radionuclide discharges during normal operation, are presented below. A dose commitment calculated for 1 year of radionuclide intake (internal exposure pathways) is an estimate of the dose an individual will accrue within the next 50 years as a result of that 1 year of intake (i.e., 50-year dose commitment). All of the doses estimated in this report represent dose commitments. Assumptions, models, and codes used to estimate radiation doses are presented in ref. 1.

Organ doses may vary considerably for internal exposure from ingested or inhaled materials because some radionuclides concentrate in certain organs of the body. For this reason, estimates of radiation dose to the total body and major organs are considered for all pathways of internal exposure based on parameters applicable to an average adult.

Radiation doses to the internal organs of children in the population vary from those received by an average adult because of differences in metabolism, organ size, and diet. Differences between the organ doses of a child and those of an average adult by more than a factor of 3 would be unusual for all pathways of internal exposure except the atmosphere-pasture-cow-milk pathway. For this pathway, the estimated dose to the thyroid of a 1-year-old child from radioactive iodine in milk is several times that for an average adult.^{2,3}

The population dose estimates for the total body are the sums of the total-body doses of the individuals within 55 miles of the plant. Since radiation doses to the total body are relatively independent of age,⁴ the person-rem estimates are based on total-body doses calculated for adults. Similarly, the population dose estimates for the various organs are the sums of the specific organ doses of the individuals within 55 miles of the plant. The person-organ-rem estimates are based on the adult organ doses.

Estimates of doses to the total population from airborne effluents are limited to distances of 55 miles from the model plant. Even at a 20-mile distance, the doses have decreased to about 5% of their values at 0.5 to 1 mile from the plant, and the population density and meteorological factors merge with the average for the United States rather than remaining as distinct functions of the model LWR fuel reprocessing plant. For example, a recent study⁵ suggests that a plant which processes 5 metric tons of fuel per day and releases all of the krypton and tritium could lead to an annual total-body dose to the population of the United States of 520 person-rem from ⁸⁵Kr and 3700 person-rem from tritium. The sum of these amounts to less than 0.02% of the 2.7×10^7 person-rem dose received by the 2.1×10^8 residents from the natural, annual, background dose of about 130 millirem. The annual dose to the world population ascribable to the LWR fuel reprocessing plant is estimated to be 8620 person-rem from ⁸⁵Kr and 4800 from tritium. Similarly, those doses are very small fractions of the natural background. The estimated doses to the population from other radionuclides are smaller fractions of the background radiation.

7.1 Radiological Impact of Airborne Effluents During Operations

The release of radioactive materials to the atmosphere is the mode of environmental contamination from the LWR fuel reprocessing facility.

7.1.1 *Models and assumptions*

AIRDOS,⁶ a FORTRAN IV computer code, is used to estimate individual and population doses resulting from the continuous simultaneous atmospheric release of airborne radioactive materials from the model LWR fuel reprocessing plant. Pathways to man include (1) inhalation of radionuclides in air, (2) immersion in air containing radionuclides, (3) exposure to ground surfaces contaminated by deposited radionuclides, (4) ingestion of food produced in the area, and (5) immersion (swimming) in water subjected to surface deposition from plumes. Doses are estimated for the total body and these organs: GI tract, bone, thyroid, lung, muscle, kidney, liver, spleen, testes, and ovaries.

The area surrounding the nuclear facility is divided into 16 sectors. Each sector is bounded by radial distances of 0.5, 1.0, 2.0, 3.0, 4.0, 5.0, 10, 15, 25, 35, 45, and 55 miles from the point of release. There are 176 areas lying outside the plant boundary within 55 miles of the plant. Human population, numbers of beef and dairy cattle, and specification as to whether each of the 176 areas is used for producing vegetable crops or is a water area are required as input data.

The first part of AIRDOS⁶ is an atmospheric dispersion model (AIRM0D) which estimates concentrations of radionuclides in air at ground level and their rates of deposition on ground surfaces as a function of distance and direction from the point of release. Annual average meteorological data for the area are supplied as input for AIRM0D.

AIRM0D is interfaced with experimental models within AIRDOS to estimate doses to man through the five pathways previously mentioned. The most complex environmental model is a terrestrial model (TERMOD).⁷ This model estimates radionuclide intakes via ingestion of radionuclides deposited on crops, soil, and pastures. The intakes result from eating beef and vegetable crops and drinking milk.

Population doses are summarized in the output tables of AIRDOS in all possible ways – by nuclide, pathway, and organs. The highest individual doses in the area for each organ are tabulated for each radionuclide and the highest organ doses from all radionuclides in the source term are listed. The highest individual dose is specified.

7.1.1.1 Meteorology. Release of airborne effluents to the atmosphere is the major pathway for environmental contamination from a fuel reprocessing facility. There are no planned releases of radioactive waste in the liquid effluents of the model plant. The LWR fuel reprocessing plant does not require large quantities of water (i.e., for cooling or dilution) as compared to power plants, and thus does not need to be located at a site close to large aquatic environments. Atmospheric transport is the principal mode of delivery of radioactive materials to terrestrial environments associated with the LWR fuel reprocessing facility.

The basic equations used to estimate atmospheric transport to the terrestrial environment is Pasquill's equation⁸ as modified by Gifford.⁹ For particulate releases, the meteorological χ/Q values are used in conjunction with dry deposition velocities and scavenging coefficients to estimate air concentrations and steady-state ground concentrations. Radioactive decay during plume travel is considered in AIRDOS. Daughters produced during plume travel must be added to the AIRDOS source term. Concentrations in air for each sector are used to calculate dose via inhalation and submersion in air. Ground surface concentrations are used for external radiation exposure. The ground deposits are also assimilated into food, which, when ingested, results in additional dose via the food chain pathway.

The meteorologic data required for the calculations are joint frequency distributions of wind velocity and direction summarized by stability class. Meteorologic data from representative midwestern and southeastern coastal regions¹⁰ are used to calculate average values of χ/Q (sec m^3), that is, these factors are used to calculate the concentrations of radioactive materials at a reference point per unit of source strength. The χ/Q values are calculated for sectors in the 16 principal compass directions bounded by radial distances of 0.5, 1.0, 2.0, 3.0, 4.0, 5.0, 10, 15, 25, 35, 45, and 55 miles from the point of release.

Radioactive particulates are removed from the atmosphere and deposited on the ground through mechanisms of dry deposition and scavenging (washout). Dry deposition, as used in this analysis, represents an integrated deposition of radioactive materials by processes of gravitational settling, adsorption, particle interception, diffusion and chemical-electrostatic effects; it is calculated from deposition velocity, V_d , for a 1-year time interval. Deposition velocity values for particles and reactive gases commonly range from 0.1 to 1.0 cm sec^{-1} .^{11,12} A value of 1.0 cm sec^{-1} is used for calculation of ground concentrations of radioactive particles for this facility. Scavenging of radionuclides in a plume is the process through which rain or snow washes out particles or dissolves gases and deposits them on the ground or water surfaces. A scavenging coefficient¹³ for particulates, tritium, ^{14}C , and radioiodines of $2.0 \times 10^{-5} \text{ sec}^{-1}$ is used for both the midwestern and southeastern coastal sites.

The χ/Q values are based on a release from a 100-m stack and a plume rise resulting from momentum of stack gases by assuming a $1.2 \times 10^5 \text{ cfm}$ flow rate through a stack 7.0 ft in diameter. Magnitude of χ/Q values are somewhat similar at the two sites, but directions at which maximum values are obtained are different. Buoyancy of the plume due to its temperature being higher than atmospheric ambient temperature was not included in calculating dose distributions. However, this buoyancy is discussed briefly in Sect. 7.3.

Maximum χ/Q values are predicted over the range of 0.5 to 2 miles, beyond which the χ/Q values decrease more than tenfold out to a distance of 55 miles. The appropriate χ/Q value¹ for each of the 16 quadrants and for the distance from the stack is multiplied by the release rate at the stack to obtain the concentration at the desired point. Maximum χ/Q values for radioiodine and particulates are 6.67×10^{-8} and $3.61 \times 10^{-8} \text{ sec m}^3$ at the 1.5-mile plant boundary for the midwestern and coastal sites, respectively.

7.1.1.2 Population Population distributions were derived that are representative of southeastern coastal and midwestern environments. They are the average of distributions around two fuel fabrication plants and one reprocessing plant for each area, specifically, the midwestern and southeastern coastal sites. Distributions for sites near St. Louis, Missouri, and

Wilmington, North Carolina, are included in the averaging, because the meteorological data used for atmospheric transport of radioactive substances are based on these areas. The Wilmington site also represents the half-annulus distribution that is representative of areas adjacent to the ocean.

Average population distributions are calculated from data sets for areas determined by the latitude-longitude coordinates specified in Table 7.1. Actual population distributions from these locations were summarized from 1970 Census Bureau tape records to obtain representative distributions for midwestern and southeastern coastal regions (Tables 7.2 and 7.3). The computer code, PANS,¹⁴ provides sector summaries for annuli bounded by distances of 0.5, 1.0, 2.0, 3.0, 4.0, 5.0, 10, 15, 25, 35, 45, and 55 miles. The sector summaries correspond to the same sectors in the 16 compass directions for which χ/Q values are calculated. The computer code summaries of population data from census tapes are accurate beyond a 5-mile radius. Within 5 miles, where sectors represent relatively small areas, distributions are somewhat disconnected because census enumeration districts encompass several sectors although the population record is reported in a single sector. Averaging data from three locations smooths the major discontinuities and results in cumulative totals that are somewhat similar to those reported for actual fuel reprocessing facilities.^{15,16}

Population distributions for the two sites of the model fuel reprocessing facilities have somewhat different characteristics (Tables 7.2 and 7.3). Average density within the 55-mile radial distance was 50 to 60 individuals per square mile for the southeastern coastal site; however, there was a factor of 5 increase to 289 individuals per square mile, representing a small city, in the 5- to 10-mile annulus. The approximately 9500-sq-mile area encircling the coastal site is distinctly rural (58 individuals per square mile) in terms of population density. By comparison, the population density of the midwestern site within the 5-mile radius is nearly twice as great (95 vs 55) as the coastal site. Beyond 5 miles, the density increases to 126 persons per square mile at 10 miles, and to 440 persons per square mile at the 25- to 55-mile annulus. A large city is included in a portion of the 55-mile area encircling the model fuel reprocessing plant. Cumulative population in the midwestern site is approximately six times greater than for the coastal site.

7.1.1.3 Radiation doses from airborne effluents. Concentrations of radionuclides in the air and on the soil surface are used to estimate the radiation dose to individuals at various distances and directions from the model plant. The dose conversion factors for submersion in the airborne effluent, exposure to contaminated ground surface, and intake of radionuclides through inhalation and ingestion are calculated with computer codes^{17,18} which use dosimetric criteria of the International Commission on Radiological Protection and other recognized authorities. The dose conversion factors for most radionuclides are based on ICRP-2.¹⁹ Estimates of intake of radionuclides by man through terrestrial food chains are made with a model and computer code,⁷ incorporated with AIRDOS,⁶ which considers transfer of all radionuclides to man via ingestion of crop plants, beef, and milk. A reference handbook on the methods used in estimating radiation doses has been prepared as ORNL-4992.¹

Many of the basic incremental parameters used in this model are conservative; that is, values are chosen to maximize intake by man. Many factors which would reduce the radiation dose, such as shielding provided by dwellings and time spent away from the reference location, are not considered. It is assumed that an individual lives outdoors at the reference location

100% of the time. Moreover, in estimating the doses to individuals via ingestion of plants, meat, and milk, all of the food consumed by the individual is assumed to be produced at the reference location specified in the calculation. This situation is not impossible, but is extremely unlikely. Thus, individual dose estimates calculated by these methods are maximized.

7.1.2 *Radiation dose commitments from airborne effluents*

7.1.2.1 *Individual and population doses.* The maximum annual total-body doses and organ doses to individuals from all airborne effluents at 0.5 mile and 1.5 miles from the operating model fuel reprocessing plant are summarized in Tables 7.4 and 7.5, respectively. The 1.5-mile distance is considered to be representative of the boundary distance from a 1500-metric-ton-per-year LWR fuel reprocessing plant. The doses for the 0.5-mile distance from the plant are shown for comparative purposes. Doses at a distance of 1.0 mile are approximately 7% higher than doses at 1.5 miles at the midwestern site and ~17% higher at the coastal site. The absolute maximum annual doses occur at a distance of 0.5 mile for the midwestern site (Table 7.4). The maximum doses at the coastal site occurs at a distance of ~0.6 mile and are ~8% higher than the values for 0.5 mile presented in Table 7.4. The population doses out to a distance of 55 miles are presented in Table 7.6. The contributions to total-body dose through various exposure modes are listed in Table 7.7.

Approximately 23% of the estimated total-body dose to individuals living within 55 miles of the model fuel reprocessing plant is the result of external exposure from submersion in air (3%) and to contaminated ground (20%). Internal exposure from inhalation accounts for approximately 11% and ingestion contributes the remaining 66% of the total-body dose. The ingestion dose is based on conservative parameters that tend to maximize the dose estimates such as assuming that 100% of the diet of the individual is grown and consumed at the area of residence. Approximate dose reduction factors can be applied when the food production and consumption patterns are known.

At the midwestern site, the maximum total-body dose of 14 millirem/year (Case 1, Table 7.5) at 1.5 miles is approximately 20% higher than the comparable dose at the southeastern coastal site due to meteorological differences. Population doses (Table 7.6) are also higher around the midwestern site due in part to meteorological differences but primarily because six times more people live within 55 miles of this site as compared with the coastal site. It is doubtful that either individuals or population would, in fact, receive these estimated doses. The conservative assumptions listed in Sect. 7.1.1.3 tend to maximize the estimated doses.

The principal radionuclides in the gaseous effluent that contribute to the total-body and organ doses of individuals are listed in Table 7.8. The major contributions to the total-body dose in the Case 1 study are from tritium (52%), ^{14}C (16%), $^{134,137}\text{Cs}$ (13%), ^{129}I (7.7%), and ^{85}Kr (3.3%).

The contribution of radionuclides through various pathways are shown for the total-body dose in Table 7.9. Of the approximately 52.3% of the total-body dose due to tritium, 43% is by way of the ingestion pathway and 9% is by the route of inhalation exposure. Similarly, essentially all of the 16% of the total-body dose due to ^{14}C is by way of ingestion. Other

major contributors and their exposure mode are $^{129,131}\text{I}$, 3% by ingestion, and 7% from exposure to contaminated surfaces; and $^{134,137}\text{Cs}$, 3% by ingestion, and 10% from contaminated surfaces.

The annual dose to the population for the midwestern site based on a population within 55 miles of the model plant is 1.04×10^3 person-rem for Case 1 (Table 7.6). The similar dose for the eastern coastal site is 2.99×10^2 person-rem. Some of the differences in dose at the two sites are due to meteorological differences; however, the principal reason for the difference is that there are only 5.5×10^5 persons (Table 7.3) living within the 55-mile radius of the east-coastal site as compared to 3.56×10^6 (Table 7.2) at the midwestern site. The contribution of major radionuclides to total-body and organ population doses are shown in Tables 7.10 and 7.11 for the midwestern and coastal sites, respectively.

7.1.2.2 Doses to organs of individuals. The maximum annual doses to organs of individuals at 1.5 miles from the model fuel reprocessing plant located on both sites are given in Table 7.5 for all radwaste treatment cases. The principal radionuclides in the airborne effluent which contribute to the organ doses of individuals are listed in Table 7.8. The fractional contribution of inhaled and ingested radionuclides to the dose of certain organs is shown in Table 7.12. Radiation doses to organs are largely dependent on the specificity for certain radionuclides to accumulate in certain organs. Therefore, a radwaste treatment case which reduces the presence of a given radionuclide in the environment will also reduce the dose to the organ that is exposed to that radionuclide via inhalation or ingestion pathways. Ingestion is the major pathway of exposure to internal body organs (Table 7.7). In the case of skin, submersion in the airborne effluent is the major pathway and ^{85}Kr contributes approximately 100% of the dose (Table 7.9). Skin is not included in the organs listed in Tables 7.5 and 7.8 because essentially all of the dose to the skin is caused by ^{85}Kr (47 millirem/year for the midwestern site and 36 millirem/year for the east-coastal site at 1.5 miles from the plant).

The highest organ dose of 290 millirem/year (Case 1, midwestern site) is received by the thyroid (Table 7.5) and, as shown in Table 7.8, is due primarily to ^{129}I (51%) and ^{131}I (45%). The maximum dose to the bone is 26 millirem/year, with tritium accounting for 34%, and ^{14}C and ^{90}Sr for 13% each. The dose to the GI tract is due primarily to ^{106}Ru (61%) and to a lesser extent to tritium (24%).

The presence of ^{129}I in the airborne effluent from the reprocessing plant is particularly significant in terms of dose to the thyroid, since iodine concentrates in the thyroid; the long-lived ^{129}I (half-life 1.7×10^7 years) can accumulate in the earth and foods, thereby contributing to exposure from both ingestion of food and irradiation from contaminated earth. The shorter-lived ^{131}I (half-life, 8.05 days) accumulates to a lesser degree and, hence, contributes a smaller fraction of the thyroid dose. For example, in the Case 1 study where almost 14 times more ^{131}I than ^{129}I is released, the dose from ^{129}I is still somewhat higher than that for ^{131}I (i.e., a maximum annual adult thyroid dose of 147 millirem for ^{129}I vs 129 millirem for ^{131}I). Maximum annual adult thyroid doses for all nuclides are shown in Table 7.5. Similarly, ^{129}I accounts for 7.7% of the total body dose in Case 1 vs 2.1% for ^{131}I (Table 7.8). Ingestion is the major internal exposure pathway for both radionuclides. Iodine-131, because of its relatively short half-life, is primarily ingested in milk and to a lesser extent in vegetables. Although ^{129}I also is ingested primarily in milk, it can accumulate in beef; under steady-state conditions,

almost 22% of the thyroid dose from this radionuclide is due to the ingestion of beef. The estimates of the amounts of ^{129}I contained in beef and milk at the 1.5 mile distance in this study (see terrestrial model TERMOD⁷) are similar to the amounts of ^{129}I found in beef and milk samples taken 1 to 2 miles from a fuel reprocessing plant in New York.²⁰

The maximum annual dose to the thyroid of a 1-year-old child from ^{129}I and ^{131}I at a distance of 1.5 miles in the Case 1 study is 998 millirem, or approximately three times the adult dose. Similarly, the ^{129}I dose of 212 millirem is about 1.5 times the adult dose and the ^{131}I dose of 786 millirem is 6.1 times the comparable adult dose due to differences in the diet.

As discussed in Sect. 7.1.2.1, these doses are estimated by using the very conservative assumption that all of the food consumed by the individuals is produced at their site of residence.

7.2 Post-Operational Exposures from Long-Lived Radionuclides Released into the Environment from a LWR Fuel Reprocessing Plant

Potential releases of radionuclides during plant operation and estimates of resulting radiation doses to individuals and the population are discussed in Sect. 7.1.2.1. In this section, estimates are presented of future potential radiation doses to individuals and populations after the LWR fuel reprocessing plant has ceased to operate. It is assumed that the plant has operated for 30 years and that long-lived radionuclides are deposited on the land during this operating period.

These estimates involve many complex considerations. In the absence of complete information, estimates are made on the basis of the best current knowledge. Conservative assumptions are used in areas where deficiencies of knowledge exist. These assumptions make it likely that the estimates of health consequences are well above the probable effects. A more detailed assessment of the radiation exposure to future generations from long-lived elements has been included in a recent environmental analysis of the LMFR program.²¹

7.2.1 Post-operational source terms

Prior to the shutdown and over the 30-year operating period, the model LWR fuel reprocessing plant (Case 1) releases approximately $1.5\text{E}+7$ Ci per year of operation to which ^{85}Kr contributes approximately 94.7% and tritium 5.2%. During this time, solid radionuclides are deposited on the earth, and gaseous radionuclides such as ^{14}C bearing carbon dioxide and tritiated water, exchange from the atmosphere into the biosphere, after the plant is closed. People will be exposed to decreasing quantities of radiation from long-lived radionuclides for many years after the plant has ceased operation through inhalation of resuspended contaminated material, through direct radiation from contaminated ground, and from ingestion of contaminated food. Table 7.13 lists these radionuclides and the total quantities released from the model LWR fuel reprocessing facility. The longest-lived radionuclides, ^{14}C , ^{129}I , ^{238}Pu , ^{239}Pu , ^{240}Pu , ^{241}Pu , ^{242}Pu , ^{241}Am , ^{243}Am , ^{242}Cm , and ^{244}Cm will remain in the environment for generations.

The distribution of these radionuclides around the reprocessing plant must be estimated in order to define the radiation dose to the population. For this assessment, it is estimated that all of the radioactive materials are deposited within a 55-mile radius of the plant. Estimates of the deposition of particulates indicate that as much as 70% of the materials are deposited within 50 miles, even though the release point is the top of a 100-m-high stack.¹² On the other hand, the ^{14}C has a more complex mechanism of dispersal into and accumulation in the biosphere.²² Dose estimates from ^{14}C were made by assuming that man reaches equilibrium with the ^{14}C in the atmosphere. Using the dose commitment model for atmospheric releases from UNSCEAR,¹ this dose may be estimated by:

$$D(\infty) = \gamma_0 W/B$$

where

- $D(\infty)$ = the infinite dose commitment (millirems),
- γ_0 = the dose rate (1.3 millirem/year) from natural ^{14}C ,
- W = the amount (Ci) of ^{14}C released from the facility,
- B = the amount of natural ^{14}C produced annually (0.03 mCi).

7.2.2 Post-operational pathways of exposure

7.2.2.1 Resuspended air activity. After airborne particulates are removed from the atmosphere and reach the ground by deposition and washout, they may again enter the atmosphere by resuspension processes. If they do, they may be inhaled. There is presently no general model which may be used to predict the levels of resuspended air activity with due regard to the geometrical configuration of the land surface, the particle characteristics of the deposited radioactive particulates, and the parameters of most soil, the vegetation cover, and the meteorological conditions. These highly variable factors and others related to land use, such as the disturbance of soil surfaces by human activity, must be considered in preparing a precise estimate of resuspended radioactivity.

A resuspension factor can be estimated from measurements made above aged contaminated soil and from consideration of natural radioactive materials such as ^{238}U . Resuspension factors of 10^{-9} and 10^{-10} m^{-1} were obtained from recent measurements of ^{239}Pu made at the Nevada test site in an area which had been contaminated for 17 years.²¹ Measurements of ^{239}Pu in the vicinity of the Rocky Flats plant several years after deposition indicated a resuspension factor of 10^{-9} m^{-1} .²¹ Discounting airborne material of industrial origin, the data concerning movement of natural ^{238}U indicate that a realistic estimate of the resuspension of aged radioactive material in surface soil lies between 10^{-8} and 10^{-10} m^{-1} .¹² This is in agreement with the field measurements for ^{239}Pu . An intermediate value of 1×10^{-9} is used in this survey to estimate the amounts of radioactive materials inhaled over a long post-shutdown period for the relatively large, well-vegetated regions around a fuel reprocessing plant. It is assumed that this value remains constant even though the deposited radionuclides may not remain on or near the surface of the soil. Actually, a continual reduction in the availability of these materials beyond the current measurement experience of 20 years can be expected. Thus, the use of a constant

resuspension factor is a conservative assumption which will maximize the estimated dose. Resuspended radionuclides are also assumed to enter terrestrial food pathways (vegetables, milk, and beef) via redeposition on foliage of crops and pastures. For estimating intake via inhalation of resuspended radionuclides the expression is:

$$Ci \text{ intake year}^{-1} = Ci \text{ m}^{-2} \times 10^{-9} \text{ m}^{-1} \times 7200 \text{ m}^3 \text{ inhaled year}^{-1} .$$

7.2.2.2 Ingestion. The radionuclides that are not inhaled by man remain in the environment for times proportional to their radiological half-lives. During this time they may be ingested by man. Plants may be contaminated by direct deposition of airborne particles onto foliar parts and by root uptake of radionuclides leached from or exchanged with particles deposited in soil. Plant uptake studies show that heavy elements are strongly excluded from plant uptake and are poorly translocated by plant systems.

The fraction of long-lived radionuclides that enters man during their long existence in the environment will depend on their distribution, their chemical and physical behavior in the environment for thousands of years, and the climatological conditions and land use patterns specific to the area. Sufficiently detailed and accurate knowledge regarding the many factors influencing the movement of these elements through the environment over the periods of hundreds to tens of thousands of years, during which time they may enter man through the ingestion pathways, is not available to permit a precise estimate of the dose to man. It is appropriate, therefore, to estimate potential human ingestion using conservative parameters and assumptions. In preparing the estimate for this survey, it is assumed that plant material accumulates a concentration of plutonium equal to 1×10^{-3} , americium equal to 2.5×10^{-4} , curium equal to 2.5×10^{-3} and iodine equal to 2.0×10^{-2} of the concentration in the soil in which the plants grow (concentration factor, CF); that there is no downward movement of the radionuclide in the soil beyond the root zone (15 cm); and that the radionuclide is not lost by drainage of water. With a soil density of 1.5 g cm^{-3} , the radionuclides deposited on a square meter of earth are contained in $2.25 \times 10^5 \text{ g}$ of soil. Thus, the following expression is used to estimate the intake via ingestion of plants:

$$Ci \text{ year}^{-1} \text{ ingested} = Ci \text{ m}^{-2} / 2.25 \times 10^5 \text{ g m}^{-2} \times CF \times 9.12 \times 10^4 \text{ g plant ingested year}^{-1} .$$

Additional intake from the ingestion of plants contaminated via resuspended radionuclides was calculated using the TERMOD code.⁷

7.2.2.3 Contaminated ground. Exposure via contaminated ground is also estimated. It is assumed that there is no loss of deposited radionuclides from the soil surfaces except through radioactive decay.

7.2.3 Estimates of post-operational doses

The post-operational radiation dose to an individual residing within the uniformly contaminated area of $9.5 \times 10^3 \text{ sq miles}$ ($2.46 \times 10^{10} \text{ m}^2$) is estimated for total-body and for organs that are known to accumulate the long-lived radionuclides. Population doses are

expressed as person-rem per 3.6×10^6 persons, the population within 55 miles of the midwestern plant. No assumptions for population change are included. All radiation doses from ingestion and inhalation are 50-year dose commitments from 1 year of exposure: that is, the dose an individual will accrue over a 50-year period (essentially a lifetime dose) from 1 year of intake of radionuclide. External doses (exposure to contaminated ground) are actual annual doses from 1 year of exposure. It is conservative to call a dose commitment an annual dose in the case of a single year's intake of long-lived radionuclides. However, for assessing a situation where people are continually exposed over long periods of time and radionuclides have reached steady-state conditions in the environment, dose commitments approximate annual doses.

7.2.3.1 Individual and organ dose. As a result of the deposition of long-lived radionuclides such as the actinides, persons living within a 55-mile radius of the model LWR fuel reprocessing plant will continue to receive some radiation dose above background long after the plant operation has been terminated, or actually until the ultimate decay of all the radionuclides occurs. The average annual doses to the individual out to 55 miles for the various radionuclides and exposure modes are shown in Table 7.14. Fifty-three percent of the total-body dose (7.7×10^{-2} millirem/year in Case 1) results from exposure to contaminated ground and about 47% from the ingestion pathway. Essentially all of the exposure from the contaminated ground is due to the ^{129}I radionuclide, and nearly all of the ingestion dose is due to ^{14}C . These doses are the average doses out to 55 miles, and the dose range, as a function of distance, will vary considerably over the 55-mile span.

The average annual doses to the organs resulting from the various radionuclides and the major internal pathways are shown in Table 7.15. Due primarily to ^{129}I (93%), the thyroid receives the highest organ dose (2.8×10^{-1} millirem/year in Case 1), which is more than 10 times the dose to the bone and 17 times the lung dose. Approximately 90% of the dose to the bone, 100% of the liver and lung doses, and 6% of the thyroid dose is due to ^{14}C .

7.2.3.2 Population doses. The annual population doses to the total body, bone, thyroid, liver, and lung, given in person-rem per 3.6×10^6 persons, are shown in Table 7.16 for the Case 1 study. The total-body dose, 2.8×10^2 person-rem/ 3.6×10^6 persons, is based on the population of a representative midwestern site. The doses (total body and organs) are due primarily to ^{129}I and ^{14}C , similar to that of the individual doses (Sect. 7.2.3.1).

7.3 Effect of Buoyancy of Hot Stack Gases on Plume Rise and Doses

The plume of stack gases released from the plant at temperatures above the ambient air temperature will rise because of a density-deficiency buoyancy; the plume rise due to this effect can exceed that due to momentum alone. This additional plume rise can reduce radiation doses at the plant boundary. A brief parametric study of this effect was performed for a release of ^{14}C assuming a gas flow of 120,000 cfm at the midwestern site. Table 7.17 lists fractional reductions in dose at 1.5 miles for stack gas temperatures ranging from 125° to 300°F as compared with an ambient temperature stack gas release in which only momentum is considered in computing plume rise. It is apparent that very significant reductions in the 1.5 mile doses are produced by discharging heated gases from the plant stack. This reduction

amounts to 85% when the stack-gas temperature is 300°F. A brief study of heat losses from a 100-m stack showed that at 300°F the temperature of air drops only a few fahrenheit degrees on passing from the bottom to the top of the stack.

Buoyancy of the plume due to its temperature being higher than atmospheric ambient temperature was not included as a factor in calculating dose distributions in this study.

7.4 Nocturnal Versus Daytime Release of Carbon Dioxide

In this report the calculations of maximum doses from ^{14}C are based on daytime releases from the stack. However, a detailed analysis²³ of the local radiological impact of $^{14}\text{CO}_2$ releases indicates that the timing of the release will strongly influence the maximum dose. Because of the limitation of photosynthesis to daylight hours, daytime release during the growing season will result in increased uptake of ^{14}C by local vegetation, and, hence, an increased transfer into the food-chain pathway, which ordinarily accounts for more than 99% of the ^{14}C dose to man. Therefore, this analysis suggests that significant local-dose benefits can be obtained from daytime holdup and nocturnal release of $^{14}\text{CO}_2$.

7.5 References for Section 7

1. G. G. Killough and L. R. McKay (eds.), *A Methodology for Calculating Radiation Doses from Radioactivity Released to the Environment*, ORNL-4992 (March 1976).
2. J. K. Soldat, "Environmental Evaluation of an Acute Release of ^{131}I to the Atmosphere," *Health Phys.* 11, 1009 (1965).
3. B. Shleian, "An Evaluation of Internal Radiation Exposure Based on Dose Commitment from Radionuclides in Milk, Food, and Air," *Health Phys.* 18, 267 (1970).
4. Federal Radiation Council, *Estimates and Evaluation of Fallout in the United States from Nuclear Weapons Testing Conducted Through 1962*, Report No. 4 (May 1963).
5. U.S. Environmental Protection Agency, *Environmental Analysis of the Uranium Fuel Cycle. Part III. Nuclear Fuel Reprocessing*, EPA-520/9-73-003-D (October 1973).
6. R. E. Moore, *AIRDOS - A Computer Code for Estimating Population and Individual Doses Resulting from Atmospheric Releases of Radionuclides from Nuclear Facilities*, ORNL/TM-4687 (January 1975).
7. R. S. Booth, S. V. Kaye, and P. S. Rohwer, "A Systems Analysis Methodology for Predicting Dose to Man from a Radioactively Contaminated Terrestrial Environment," *Proceedings of the Third National Symposium on Radioecology*, CONF-710501, pp. 877-93, D. J. Nelson (ed.), Oak Ridge, Tenn. (May 10-12, 1971).

8. F. Pasquill, *Meteorog. Mag.* 90, 1063 (1961).
9. F. A. Gifford, Jr., "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," *Nucl. Safety* 2(4), 47 (1961).
10. U. S. Dept. of Commerce, NOAA, *Seasonal Annual Wind Distribution by Pasquill Stability Classes (6) STAR Tabulation*, St. Louis, Mo., and Wilmington, N. C.
11. F. A. Gifford, Jr., and D. H. Pack, "Surface Deposition of Airborne Materials," *Nucl. Safety* 4(4), 76-80 (1962).
12. K. E. Cowser, M. Reeves, P. G. Fowler, and W. M. Culkowski, "Radiological Impact of Gaseous Effluents on Local Environments," *Proc. Int. Symp. Radioecology Applied to the Protection of Man and His Environment*, EUR-4800 (1972), pp. 923-38.
13. D. H. Slade, *Meteorology and Atomic Energy*, TID-24190 (1968).
14. P. R. Coleman and A. A. Brooks, *PANS: A Program to Tally Population by Annuli and Sectors*, ORNL/TM-3923 (October 1972).
15. *USAEC Final Environmental Statement Related to the Operation of the Midwest Fuel Recovery Plant*, Docket No. 50-268 (December 1972), p. 9.
16. *USAEC Draft Environmental Statement Related to the Barnwell Nuclear Fuel Plant*, Docket No. 50-332 (January 1973) pp. II-5 to II-7. Additional population data appear in the *Allied-Gulf Nuclear Services Environmental Report - Barnwell Nuclear Fuel Plant*, (November 1971).
17. D. K. Trubey and S. V. Kaye, *The EXREM Computer Code for Estimating External Radiation Doses to Populations from Environmental Releases*, ORNL/TM-4322 (December 1973).
18. G. G. Killough, P. S. Rohwer, and W. D. Turner, *INREM - A FORTRAN Code Which Implements ICRP 2 Models of Internal Radiation Dose to Man*, ORNL-5003 (February 1975).
19. International Commission on Radiological Protection, *Report of Committee 2 on Permissible Dose for Internal Radiation*, ICRP Publication 2, Pergamon, New York, London, Oxford (1959).
20. P. J. Magno, T. C. Reavey, and J. C. Apidianakis, *Iodine-129 in the Environment Around a Nuclear Fuel Reprocessing Plant*, U. S. Environmental Protection Agency (October 1973).

21. *Proposed Final Environmental Statement, Liquid Metal Fast Breeder Reactor Program, Vol. II. Environmental Impact of the LMFBR*, WASH-1535 (December 1974).
22. United Nations, "Ionizing Radiations: Levels and Effects," *Report of United Nations Scientific Committee on the Effects of Atomic Radiation, Vol. I, Levels*, p. 29, New York, 1972.
23. G. G. Killough, K. R. Dixon, N. T. Edwards, B. D. Murphy, P. S. Rohwer, W. R. Harris, and S. V. Kaye, *Progress Report of Evaluation of Potential Impact of ^{14}C Releases from an HTGR Reprocessing Facility*, ORNL/TM-5284 (July 1976).

8.0 CORRELATION OF ENVIRONMENTAL IMPACT WITH COST OF WASTE TREATMENT

8.1 Comparison of Radwaste Treatment Costs with Radiological Dose

The relationships between the annual costs of the radwaste treatment systems described in Sects. 4.5 and 6.0 and the impact of radioactive releases, (i.e., the 50-year dose commitments) from these systems described in Sect. 7.0 are presented in this section. The accuracy of the cost estimates is about $\pm 30\%$, and the dose commitments represent maximum values. Many of the treatment systems are in an early stage of development, and their technical feasibility has not been verified in plant installations. Similarly, many of the models for the movement and concentration of the radionuclides in the environment are receiving additional study to increase their accuracy. In all cases, realistically conservative assumptions are made in selecting treatment efficiency ratings for equipment, in estimating costs, in defining the movement of radionuclides in the environment, and in selecting food and liquid consumption patterns.

The annual costs and dose commitments for the base case (Case 1) and succeeding case studies (Cases 2 to 7) at the midwestern and east-coastal sites are summarized in Tables 8.1-8.4 and Figs. 8.1-8.4. The costs are the estimated total annual costs required for the additional radwaste treatment system for a given case beyond that required for the base Case 1, i.e., in effect, the added incremental cost. The costs are also presented as a unit fuel reprocessing cost for additional radwaste treatment in terms of dollars per kilogram of uranium. Cases 2b and 2c include the cost of Case 2a, and Case 6c is a cumulative case which includes the costs of Cases 2 through Case 6. The dose commitments are reported on several bases: (1) annual population total-body, GI tract, bone, and thyroid doses (person-rem) out to 55 miles from the plant, which are the total doses from all radioactive materials released from the plant in a specified case; (2) incremental decreases in these annual values, with respect to Case 1; and (3) maximum annual individual (adult) total-body doses (millirem) at a distance of 1.5 miles from the stack. The distance of absolute maximum dose and factors to convert doses at 0.5 miles to doses at 1.0 mile are given in Sect. 7.0 (7.1.2.1). Doses at 0.5 miles and to additional organs are listed in Tables 7.4 and 7.5. However, only the the annual population doses to total body, GI tract, bone, and thyroid, and the maximum annual individual (adult) total-body dose are used to illustrate the cost/benefit relationships in this section of the report. The maximum, rather than the average doses, are used to maintain the principle of the selection of maximum effects. The total-body dose is selected because of its obvious importance; GI tract, bone, and thyroid are selected because the principal radionuclides (^3H , ^{14}C , and ^{129}I) that contribute to total-body dose also contribute to the GI tract, bone, and thyroid doses. These organs are also selected because of the interest in defining the effect of the release of iodine on the thyroid, ruthenium on the GI tract, and plutonium and curium on the bone. Internal exposure to radiation through inhalation and ingestion of radionuclides accounts for about 77% of the total body dose to individuals and population. The major contributors to this dose are ^3H , ^{14}C , ^{129}I , $^{134,137}\text{Cs}$, and ^{85}Kr .

Annual population total-body dose. The annual cost of additional radwaste treatment equipment and the incremental decreases in doses to total body, GI tract, bone, and thyroid (in

person-rem) out to a distance of 55 miles from ground zero of the plant stack at the midwestern and east-coastal sites are presented in Tables 8.1 and 8.2; some of the data from Tables 8.3 and 8.4 are given in Figs. 8.1 and 8.2. Small reductions (7 to 15%) in total-body dose are achieved by decreasing the release of $^{129,131}\text{I}$, particulates, and semivolatiles in Cases 2a-2c, Case 4, and Case 5 at annual costs varying from \$35,000 to \$455,000. The iodine removal Cases 2a-2c decrease thyroid doses by 60 to 93% as shown in Tables 8.1 and 8.2 and Figs. 8.3 and 8.4; particulate removal in Case 4 decreases the dose to the GI tract by 16 to 20%. A 56 to 58% reduction in the total-body dose is achieved in Case 6 at an annual cost of \$998,000 by retaining tritium in the plant. Installing the selective absorption process (Case 3) at an annual cost of \$1,635,000 decreases the total-body dose by about 20% by retaining 99% of the krypton and ^{14}C .

The cost/benefit varies from \$0.45 thousand/person-rem for Case 2a to \$43.71 thousand/person-rem for Case 7 at the midwestern site, and from \$1.30 thousand/person-rem for Case 2a to \$152 thousand/person-rem for Case 7 at the east-coastal site. The most efficient system in terms of cost/benefit ratio ($\$10^3/\text{person-rem}$), (i.e., the lowest cost/benefit ratio) is the ion exchange system in Case 2a. The cost/benefit ratios increase in the following order: Case 2a, iodine retention; Case 5, semivolatile and carbon-14 retention; Case 6, tritium and iodine retention; Case 4, particulate and semivolatile retention; Case 3, krypton and ^{14}C retention; Cases 2b and 2c, iodine retention; and Case 7, improved retention of all radionuclides. Cases 2a, 2b, and 2c rank low in relative efficiency for iodine retention, since iodine and the dose to the thyroid contribute a small fraction of the total-body dose.

The cumulative reduction in population total-body dose out to 55 miles as a function of cumulative increase in waste treatment efficiencies and cost is shown in Figs. 8.1 and 8.2, for the midwestern and east-coastal sites, respectively. The treatment systems are added to the base case in the order of increasing cost/benefit ratio (i.e., decreasing efficiency). Cost/benefit ratios from Tables 8.1 and 8.2 are used to obtain the combined treatment values listed in Tables 8.3 and 8.4. As a result of the cumulative addition of the case studies, the cost/benefit relationship after the addition of the final case, i.e., Case 2c, represents a summation of all of the cases and is equivalent to the cumulative Case 6c. The most efficient combination of cases in terms of cost/benefit ratio ($\$/\text{person-rem}$) occurs by adding Cases 2a, 5, and 6 to the base case, resulting in a reduction of annual population total-body dose from 1,042 person-rem to 229 person-rem (78%) at the midwestern site and from 299 person-rem to 71 person-rem (76%) at the east-coastal site for a total increase in annual cost of \$1,156,000. The addition of Cases 4, 3, 2b, and 2c results in a further smaller reduction of total-body dose from 229 person-rem to 22 person-rem (20% of total) at the midwestern site and from 71 person-rem to 7 person-rem (21%) of the total at the east-coastal site at an additional increase in annual cost of \$2,358,000. The resultant cost/benefit ratio for the addition of Cases 2a, 5, and 6 is \$1421/person-rem at the midwestern site and \$5070/person-rem at the east-coastal site and for the further addition of Cases 4, 3, 2b, and 2c is \$11,282/person-rem at the midwestern site and \$36,843/person-rem at the east-coastal site. The overall cost/benefit ratio for Case 6c (98% reduction in total-body dose) is \$3,445/person-rem at the midwestern site and \$12,034/person-rem at the east-coastal site. The cost/benefit for Case 7 is \$43.7 thousand/person-rem at the midwestern site and \$152 thousand/person-rem at the east-coastal site.

Thyroid dose. The annual costs of reducing the annual population thyroid dose out to a distance of 55 miles from the plant are presented in Tables 8.1-8.4, and Figs. 8.3 and 8.4. All of the treatment costs can be assessed to reduction in thyroid dose in Cases 2a, 2b, and 2c, since these cases are specifically designed to illustrate iodine retention. However, the total costs for the additional cases (3-6) should not be assessed to reduction in iodine releases. The equipment in each of these additional cases is designed to remove specific types of radioactive materials but, in fact, will also remove small amounts of iodine. However, these systems represent poor efficiencies in terms of retention of iodine and no effort is made to define the fractions of the total cost of Cases 3-6c and 7 that apply to retention of iodine or reduction in thyroid dose. The ion exchange unit provides the greatest reduction in thyroid dose (about 90%) at a cost/benefit at the midwestern site of \$3.17/person-thyroid-rem and \$9.04/person-thyroid-rem at the east-coastal site. Further small reductions in thyroid doses are produced by the additional treatment methods employed in Case 2b and 2c. Cases 2b and 2c include Case 2a and the cost/benefit for these cases with reference to Case 1 are \$35.48 and \$39.95/person-thyroid-rem at the midwestern site and \$101.28 and \$114.06/person-thyroid-rem at the east-coastal site. The incremental cost/benefit for Cases 2b and 2c with reference to Case 2a, excluding the ion exchange unit, is \$1088 and \$1239/person-thyroid-rem at the midwestern site, and \$3100 and \$3823/person-thyroid-rem at the east-coastal site.

The iodine isotopes $^{129,131}\text{I}$ account for 96% of the total thyroid dose (Table 7.8); most of the remaining 4% is from tritium, which contributes 2.6%. For this reason, some reduction in thyroid dose is achieved by Case 6 (voloxidation) both by the retention of tritium and the volatilization of about 64% of the iodine into the off-gas treatment system, which results in an improved retention of iodine with reference to Case 1.

GI-tract dose. Tritium, ^{14}C , and ^{106}Ru contribute about 24, 6, and 61%, respectively, of the GI-tract dose at the midwestern site, and 20, 5, and 66% of the dose at the east-coastal site (Table 7.10). The remaining 9% of the dose is due primarily to particulate radionuclides released from the plant. In the case studies (Sect. 4.5), it is assumed that 90% of the particulates and 90% of the ruthenium (ruthenium is assumed to be a particulate at this point in the off-gas stream) are removed on the sand filter in Case 4 and 90% of the semivolatile ruthenium and 80% of the ^{14}C (as CO_2) are removed in the caustic scrubber in Case 5.

The cost/benefit for Case 4 (sand filter) is \$203/person-GI tract-rem at the midwestern site and \$583/person-GI tract-rem at the east-coastal site. The cost/benefit for Case 5 (caustic scrubber) is \$83.39/person-GI tract-rem at the midwestern site and \$243/person-GI tract-dose at the east-coastal site.

Bone dose. The dose commitment to the bone is due primarily to tritium, ^{90}Sr , and ^{14}C , in decreasing importance (Table 7.10). The effectiveness of the radwaste treatment methods for reducing the bone dose is 39% for Case 4 (sand filter), 36% for Case 6 (voloxidation), 16% for Case 3 (selective absorption), and 12% for Case 5 (caustic scrubber). The cost/benefit for the radwaste treatment cases at the midwestern and east-coastal sites are \$458 and \$1377/person-bone-rem for Case 4, \$1625 and \$5884/person-bone-rem for Case 6, \$5923 and \$22,305/person-bone-rem for Case 3, and \$580 and \$2043/person-bone-rem for Case 5.

Lung dose. The lung dose is due primarily to tritium (about 60%), ^{85}Kr (about 9%), and ^{14}C (about 8%) (Table 7.8). Except for $^{129,131}\text{I}$, ^{106}Ru , ^{134}Cs , and ^{137}Cs , which contribute about 5%, 2%, 3%, 4%, and 6%, respectively, to the lung dose, all other radioactive nuclides contribute less than 1% to this dose. Thus, the lung dose is most effectively reduced by the

retention of tritium (Case 6, voloxidation), krypton, and ^{14}C (Case 3, selective absorption), and particulates and semivolatiles (Case 4, sand filter). The cost/benefit at the midwestern and east-coastal sites for the retention of tritium (Case 6) is \$1678 and \$6137/person-lung-dose, for the retention of krypton and ^{14}C (Case 3) it is \$9370 and \$36,577/person-lung-dose; and for the retention of particulates and semivolatiles (Case 4) it is \$2393 and \$6933/person-lung-dose.

Maximum annual individual (adult) total-body dose. The annual costs and reductions of maximum adult doses at 1.5 miles at the two sites are presented in Tables 8.1 and 8.2 for individual case studies; Tables 8.3 and 8.4 summarize costs and benefits of combining the case studies. Tritium and ^{14}C releases are responsible for 52% and 16%, respectively, of the total body doses (Table 7.8). Reduction of the maximum individual (adult) total-body dose is achieved by the same equipment as that for the reduction of population-total-body dose, namely, equipment to retain tritium and ^{14}C (see above); smaller reductions in total-body dose are achieved by retaining ^{129}I , ^{137}Cs , ^{134}Cs , and ^{85}Kr in reducing order, as shown in Table 7.8.

8.2 Comparison of Radwaste Treatment Costs with Fuel Reprocessing Costs, Value of Spent Fuel, Power Costs, and Total Capital Investment

Radwaste treatment costs can be compared with fuel reprocessing costs, value of the spent fuel, power costs, and total capital investment in establishing the relative importance of alternative cost factors in a cost/benefit analysis. A comparison of the cost of reprocessing fuel with the value of the fuel is particularly important since the incentive for reprocessing fuel decreases as the cost of reprocessing approaches the value of the fuel. The cost for treatment of radwaste in Case 7 is a rough estimate of very advanced technology that is presented for illustrative purposes. A formal cost estimate was not prepared.

Fuel reprocessing costs and the value of spent fuel. The value of the spent fuel is estimated as \$75.35 per kilogram of contained uranium plus plutonium^a for PWR fuel, and \$60.37 per kilogram uranium plus plutonium^a for BWR fuel (Sect. 6.4, Table 6.4). The annual reprocessing cost in the base Case 1 is \$30.33 per kilogram of uranium (Sect. 6.0, Table 6.1). In this case, the difference between the value of the spent fuel and the reprocessing cost is about \$45 per kilogram of uranium plus plutonium for PWR fuel and \$30 per kg of uranium plus plutonium for BWR fuel. In Cases 2a through 6, the costs for additional radwaste treatment systems increase from about 0.05 to 1.5% of the incremental difference between the reprocessing cost and the value of the PWR fuel, and by 0.08 to 2% for BWR fuel. In the cumulative Case 6c, the increased treatment cost is about 5% of the incremental difference for PWR fuel and 8% for BWR fuel. An increased treatment cost of \$30.33 per kilogram of uranium, as illustrated for Case 7, would approach the incremental difference between the value of the fuels and the reprocessing cost under current economic conditions.

^aThe production rate at the reprocessing plant is based on an annual charge of 1500 metric tons of uranium to the reactor.

Capital and power costs. The capital costs for Cases 2a through 6 range from \$0.092 million to \$4,491 million, or up to about 3.6% of the \$125 million capital cost of the base plant, and a total of about 8% of the base plant cost for the cumulative Case 6c. Case 7 represents a 100% increase in the capital cost of the base plant. The annual costs for Cases 2a through 6 range from about \$0.035 million to \$1.635 million, which is equivalent to contributions to the power cost of 9.1×10^{-5} to 4.2×10^{-3} mill/kWhr, respectively (Sect. 6.0, Table 6.1). Cumulative Case 6c and Case 7 amount to 2.1×10^{-2} and 1.2×10^{-1} mill/kWhr, respectively. All of these values are less than 2% of an estimated total generation cost of 7 to 10 mills/kWhr.

Table 4.1. Relative inhalation hazard for radionuclides in the fuel charged to the model fuel reprocessing plant^a

Nuclide	Half-life	Air volume required to reduce activity to RCG (m ³ /MTHM)
<u>Gaseous</u>		
H-3	12.33 y	2.57E+9
C-14	5730 y	5.54E+6
Kr-85	10.73 y	3.09E+10
I-129	1.59E+7 y	1.67E+9
I-131	8.04 d	9.23E+9
<u>Semivolatile</u>		
Ru-103	39.6 d	4.37E+12
Ru-106	365 d	1.27E+14
<u>Particulates</u>		
Sr-89	50.5 d	2.84E+14
Sr-90	28 y	2.56E+15
Y-90	64 h	2.56E+13
Y-91	58.6 d	1.43E+14
Zr-95	65.5 d	6.35E+13
Nb-95	35.1 d	1.58E+14
Ag-110m	249 d	7.92E+12
Sb-125	2.7 y	8.77E+12
Te-127m	109 d	1.02E+12
Te-129m	33.4 d	9.13E+11
Cs-134	2.06 y	2.21E+14
Cs-137	30.1 y	2.13E+14
Ce-141	32.5 d	9.11E+12
Ce-144	284.4 d	2.45E+15
Pm-147	2.7 y	4.71E+13
Eu-154	16 y	1.17E+14
Eu-155	1.7 y	8.13E+11
U-234	2.44E+5 y	3.77E+10
U-235	7.13E+8 y	8.55E+8
U-236	2.39E+7 y	1.44E+10
U-238	4.51E+9 y	1.05E+11
Pu-238	87.8 y	4.03E+16
Pu-239	2.44E+4 y	5.38E+15
Pu-240	6540 y	7.96E+15
Pu-241	15 y	3.43E+16
Pu-242	3.85E+5 y	2.30E+13
Am-241	433 y	7.63E+14
Am-243	7370 y	9.08E+13
Cm-242	163 d	4.25E+15
Cm-244	17.9 y	8.01E+15

^aAir dilution volumes pertain to the curies of each isotope per metric ton of uranium charged to the reactor after 160 days cooling. Radiation Concentration Guides (RCG) were taken from 10 CFR 20, Appendix B, Table 2, Column 1.

Table 4.2 Calculated amounts of radioactive materials entering the model nuclear fuel reprocessing plant^a and released in the gaseous effluent

Nuclide	Initial activity ^b (Ci/metric ton)	Case 1		Case 2a (Ci/yr)	Case 2b (Ci/yr)	Case 2c (Ci/yr)	Case 3 (Ci/yr)	Case 4 (Ci/yr)	Case 5 (Ci/yr)	Case 6 (Ci/yr)	Case 6c (Ci/yr)	Case 7	
		(DF)	(Ci/yr)									(DF)	(Ci/yr)
Volatile													
H-3	5 14E+2 ^c	1	7 7E+5	7 7E+5	7 7E+5	7 7E+5	7 7E+5	Same as Case 1	7 7E+5	7 7E+3	7 7E+3	1E+3	7 7E+2
Kr-85	9 26E+3	1	1 4E+7	1 4E+7	1 4E+7	1 4E+7	1 4E+5	↓	1 4E+7	1 4E+7	1 4E+5	1E+4	1 4E+3
C-14	5 54E-1 ^d	1	8 31E+2	8 31E+2	8 31E+2	8 31E+2	8 31E00	↓	1 66E+2	8 31E+2	8 31E00	1E+3	8 3E-2
I-129	3 33E-2	20	2 5E00	1 00E-1	5 0E-3	5 0E-3	2 5E00	↓	2 5E00	1 7E00	5 0E-3	5E+6	1 0E-5
I-131	9 23E-1	40	3 46E+1	6 92E-1	1 4E-1	1 4E-1	3 46E+1	↓	3 46E+1	1 38E+1	1 4E-1	5E+6	2 8E-4
Semi-Volatile													
Ru-103	8 74E+4	1E+8	1 3E00	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	1 3E-1	1 3E-1	Same as Case 1	1 3E-2	1E+12	1 3E-4
Ru-106	3 80E+5	1E+8	5 7E00	↓	↓	↓	↓	5 7E-1	5 7E-1	↓	5 7E-2	1E+12	5 7E-4
Uranium													
U-234	7 55E-1	5E+7	2 3E-5	↓	↓	↓	↓	2 3E-6	Same as Case 1	↓	2 3E-6	1E+12	1 1E-9
U-235	1 71E-2	5E+7	5 1E-7	↓	↓	↓	↓	5 1E-8	↓	↓	5 1E-8	1E+12	2 6E-11
U-236	2 88E-1	5E+7	8 6E-6	↓	↓	↓	↓	8 6E-7	↓	↓	8 6E-7	1E+12	4 3E-10
U-238	3 14E-1	5E+7	9 4E-6	↓	↓	↓	↓	9 4E-7	↓	↓	9 4E-7	1E+12	4 7E-10
Plutonium													
Pu-238	2 82E+3	2E+8	2 1E-2	↓	↓	↓	↓	2 1E-3	↓	↓	2 1E-3	1E+12	4 2E-6
Pu-239	3 23E+2	2E+8	2 4E-3	↓	↓	↓	↓	2 4E-4	↓	↓	2 4E-4	1E+12	4 8E-7
Pu-240	4 75E+2	2E+8	3 6E-3	↓	↓	↓	↓	3 6E-4	↓	↓	3 6E-4	1E+12	7 1E-7
Pu-241	1 02E+5	2E+8	7 7E-1	↓	↓	↓	↓	7 7E-2	↓	↓	7 7E-2	1E+12	1 5E-4
Pu-242	1 37E00	2E+8	1 0E-5	↓	↓	↓	↓	1 0E-6	↓	↓	1 0E-6	1E+12	2 1E-9
Particulate													
Sr-89	8 51E+4	5E+8	2 6E-1	↓	↓	↓	↓	2 6E-2	↓	↓	2 6E-2	1E+12	1 3E-4
Sr-90	7 68E+4	5E+8	2 3E-1	↓	↓	↓	↓	2 3E-2	↓	↓	2 3E-2	1E+12	1 2E-4
Y-90	7 68E+4	5E+8	2 3E-1	↓	↓	↓	↓	2 3E-2	↓	↓	2 3E-2	1E+12	1 2E-4
Y-91	1 43E+5	5E+8	4 3E-1	↓	↓	↓	↓	4 3E-2	↓	↓	4 3E-2	1E+12	2 1E-4
Zr-95	2 54E+5	5E+8	7 6E-1	↓	↓	↓	↓	7 6E-2	↓	↓	7 6E-2	1E+12	3 8E-4
Nb-95	4 73E+5	5E+8	1 4E00	↓	↓	↓	↓	1 4E-1	↓	↓	1 4E-1	1E+12	7 1E-4
Ag-110m	2 38E+3	5E+8	7 1E-3	↓	↓	↓	↓	7 1E-4	↓	↓	7 1E-4	1E+12	3 6E-6
Sb-125	7 89E+3	5E+8	2 4E-2	↓	↓	↓	↓	2 4E-3	↓	↓	2 4E-3	1E+12	1 2E-5
Te-127m	5 08E+3	5E+8	1 5E-2	↓	↓	↓	↓	1 5E-3	↓	↓	1 5E-3	1E+12	7 6E-6
Te-129m	2 74E+3	5E+8	8 2E-3	↓	↓	↓	↓	8 2E-4	↓	↓	8 2E-4	1E+12	4 1E-6
Cs-134	2 21E+5	5E+8	6 6E-1	↓	↓	↓	↓	6 6E-2	↓	↓	6 6E-2	1E+12	3 3E-4
Cs-137	1 07E+5	5E+8	3 2E-1	↓	↓	↓	↓	3 2E-2	↓	↓	3 2E-2	1E+12	1 6E-4
Ce-141	4 56E+4	5E+8	1 4E-1	↓	↓	↓	↓	1 4E-2	↓	↓	1 4E-2	1E+12	6 8E-5
Ce-144	7 36E+5	5E+8	2 2E00	↓	↓	↓	↓	2 2E-1	↓	↓	2 2E-1	1E+12	1 1E-3
Pm-147	9 42E+4	5E+8	2 8E-1	↓	↓	↓	↓	2 8E-2	↓	↓	2 8E-2	1E+12	1 4E-4
Eu-154	1 17E+4	5E+8	3 5E-2	↓	↓	↓	↓	3 5E-3	↓	↓	3 5E-3	1E+12	1 8E-5
Eu-155	2 44E+3	5E+8	7 3E-3	↓	↓	↓	↓	7 3E-4	↓	↓	7 3E-4	1E+12	3 7E-6
Am-241	1 58E+2	5E+8	4 7E-4	↓	↓	↓	↓	4 7E-5	↓	↓	4 7E-5	1E+12	2 4E-7
Am-243	1 80E+1	5E+8	5 4E-5	↓	↓	↓	↓	5 4E-6	↓	↓	5 4E-6	1E+12	2 7E-8
Cm-242	1 69E+4	5E+8	5 1E-2	↓	↓	↓	↓	5 1E-3	↓	↓	5 1E-3	1E+12	2 5E-5
Cm-244	2 38E+3	5E+8	7 1E-3	↓	↓	↓	↓	7 1E-4	↓	↓	7 1E-4	1E+12	3 6E-6

^aThe model plant processes 1,500 metric tons of uranium per year^bThe fuel is irradiated to 33,000 MWd/metric ton at 30 kW/kg and decayed 160 days (computer code ORIGEN)^cBased on PWR fuel with a fresh fuel Li content of 0 ppm. A Li content of 2 ppm in PWR fresh fuel will add an additional 512 Ci/metric ton (total initial activity 1026 Ci/metric ton)^dBased on PWR fuel with a fresh fuel ¹⁴N content of 25 ppm. On the same basis, the initial activity for BWR fuel will be 0.532 Ci/metric ton

Table 4.3. Summary of variables for model nuclear fuel reprocessing plant^a, gaseous and liquid radwaste treatment systems

Treatment objective	Radwaste treatment case No									
	1 ^b	2 ^c	2 ^d	2 ^e	3 ^f	4 ^g	5 ^h	6 ⁱ	6 ^j	7 ^k
	Base case	Reduce iodine ^c content of vapor from aqueous-waste evaporator	Reduce iodine release by evolving I ₂ from feed solution. Recycle mercury	Reduce iodine release by use of Iodex process, evolve I ₂ from feed solution	Reduce krypton and carbon-14 release by 100	Reduce particulate, uranium, plutonium and semi-volatile release by 10	Reduce semi-volatile release by 10 and carbon-14 release by 5	Reduce tritium release by 100, reduce iodine release by vaporization from voloxidizer	Cumulative reduction of releases, summation of Cases 2c through 6	Further reduction in release, overall DFs: Iodine - 5 x 10 ⁶ Krypton and carbon-14 - 10 ⁴ Particulates and semi-volatiles - 10 ¹² Tritium - 10 ³
					Overall plant decontamination factor ¹					
Iodine-131	40	2 x 10 ³	1 x 10 ⁴	1 x 10 ⁴	40	40	40	100	1 x 10 ⁴	5 x 10 ⁶
Iodine-129	20	5 x 10 ²	1 x 10 ⁴	1 x 10 ⁴	20	20	20	30	1 x 10 ⁴	5 x 10 ⁶
Krypton-85	1	1	1	1	100	1	1	1	100	1 x 10 ⁴
Carbon-14	1	1	1	1	100	1	5	1	100	1 x 10 ¹²
Particulates	5 x 10 ⁸	5 x 10 ⁸	5 x 10 ⁸	5 x 10 ⁸	5 x 10 ⁸	5 x 10 ⁹	5 x 10 ⁸	5 x 10 ⁸	5 x 10 ⁹	1 x 10 ¹²
Plutonium	2 x 10 ⁸	2 x 10 ⁸	2 x 10 ⁸	2 x 10 ⁸	2 x 10 ⁸	2 x 10 ⁹	2 x 10 ⁸	2 x 10 ⁸	2 x 10 ⁹	1 x 10 ¹²
Uranium	5 x 10 ⁷	5 x 10 ⁷	5 x 10 ⁷	5 x 10 ⁷	5 x 10 ⁷	5 x 10 ⁸	5 x 10 ⁷	5 x 10 ⁷	5 x 10 ⁸	1 x 10 ¹²
Semi-volatiles	1 x 10 ⁸	1 x 10 ⁸	1 x 10 ⁸	1 x 10 ⁸	1 x 10 ⁸	1 x 10 ⁹	1 x 10 ⁹	1 x 10 ⁹	1 x 10 ¹⁰	1 x 10 ¹²
Tritium	1	1	1	1	1	1	1	100	100	1 x 10 ³
Equipment unit or function and flowsheet reference										
Iodine										
Gaseous	1 - Hg(NO ₃) ₂ -HNO ₃ scrubber, AgZnO adsorber, HEPA filters	Same as Case 1	Same as Case 1 except add second Hg(NO ₃) ₂ -HNO ₃ scrubber and mercury recycle	Same as Case 1 except replace Hg(NO ₃) ₂ -HNO ₃ scrubber by Iodex process	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 2c	Reduced volume net off-gas release, recycle process and cell off-gas, 2 stages of Iodex, 2 stages of AgZnO adsorbers, HEPA filters
Liquid	Iodine removal partial evaporator	Ion exchange down-stream from evaporator	Same as Case 2a plus evolving iodine from feed solution	Same as Case 2b	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 2c	Evaporation, iodine removal on resin bed, evaporation and recycle, storage for excess water, no release of water or water vapor
Solid	None	Same as Case 1	Recover and store I ₂ as NaIO ₃	Recover and store I ₂ as HI ₂ O ₃	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 2c	Isolate NaI for storage
Krypton	None	Same as Case 1	Same as Case 1	Same as Case 1	Selective absorption	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 3	Reduced volume net off-gas release, recycle process and cell off-gas, recycle cell off-gas through selective absorption unit
Carbon-14	None	Same as Case 1	Same as Case 1	Same as Case 1	Selective absorption	Same as Case 1	Caustic scrubber	Same as Case 1	Same as Cases 3 and 5	Reduced volume net off-gas release, recycle process and cell off-gas, recycle cell off-gas through selective absorption unit
Particulates, uranium and plutonium	HEPA filters	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	HEPA filters, sand filter	Same as Case 1	Same as Case 1	Same as Case 4	Reduced volume net off-gas release, recycle process and cell off-gas, improved filtration efficiency on a small volume of final off-gas
Semi-volatiles	Acid scrubbers, HEPA filters	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Acid scrubbers, HEPA filters, sand filter	Acid scrubbers, caustic scrubber, HEPA filters	Same as Case 1	Same as Case 5	Reduced volume net off-gas release, recycle process and cell off-gas, caustic scrubber on process off-gas, improved adsorption efficiency on small volume of final off-gas
Tritium	None	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Voloxidation, recombiner, Drierite, molecular sieve, cold trap, storage	Same as Case 6	Reduced volume net off-gas release, recycle process and cell off-gas, same as Case 6, dry off-gas by cooling to -100°F
Cell ventilation	HEPA filters	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	HEPA filters, sand filter	Same as Case 1	Same as Case 1	Same as Case 4	Recycle, no release
Lab ventilation	HEPA filters	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	HEPA filters, sand filter	Same as Case 1	Same as Case 1	Same as Case 4	HEPA filter, roof vents

^aModel nuclear fuel reprocessing plant has a nominal capacity of 1500 metric tons per year, reference fuel is 3 % enriched uranium irradiated at an average specific power of 30 MW/metric ton to an exposure of 33,000 MWd/metric ton and decayed 160 days. All gaseous and water vapor releases are discharged through a 100-meter stack

^bFigure 4.3.

^cFigure 4.4.

^dFigure 4.5.

^eFigure 4.6.

^fFigure 4.7.

^gFigure 4.8.

^hFigure 4.9.

ⁱFigure 4.10.

^jFigure 4.11.

^kFigure 4.12.

¹Decontamination factor (DF) is amount entering plant/amount released in waste effluents.

Table 4.4. Liquid and gaseous effluent operating experience at the NFS fuel processing plant

	1966	1967	1968	1969	1970	1971
<u>Fuel Processed^a</u>						
Reactor exposure, MWd(th)	150,000	1,000,000	510,000	950,000	720,000	790,000
Gross beta, curies	10,000,000	67,000,000	34,000,000	64,000,000	48,000,000	53,000,000
⁸⁵ Kr, curies	48,000	320,000	160,000	300,000	230,000	250,000
³ H, curies	2,600	18,000	9,000	17,000	13,000	14,000
¹²⁹ I, curies	0.13	0.90	0.46	0.85	0.65	0.71
⁹⁰ Sr, curies	410,000	2,700,000	1,400,000	2,600,000	2,000,000	2,200,000
<u>Liquid Effluents^b</u>						
Gross alpha, curies	0.038	0.056	0.14	0.38	0.1	0.06
Gross beta, curies	8.3	31	46	140	87	77
³ H, curies	290	4,200	2,600	6,000	4,500	3,800
¹²⁹ I, curies		0.07	0.028	0.22	0.34	0.21
⁹⁰ Sr, curies		4.4	5.0	10.1	14.2	6.6
Percent of limit ^c	6	11	9	19	22	13
<u>Gaseous Effluents^b</u>						
Annual release, curies						
Particulates	0.15	0.45	1.1	0.12	0.18	0.01
⁸⁵ Kr	77,000	330,000	190,000	300,000	180,000	220,000
¹³¹ I	<0.06	<0.06	<0.06	<0.06	<0.06	<0.06
Percent of limits ^d						
Particulates	4.6	14	35	3.8	5.7	0.32
⁸⁵ Kr	1.7	7.2	4.2	6.5	3.9	4.8
¹³¹ I	<1	<1	<1	<1	<1	<1

^aRadionuclide content of fuel based upon reactor power data and properties of typical LWR fuel at a postirradiation decay time of 1 year. A measure of the consistency of the data may be obtained by comparing the computed ⁸⁵Kr processed with the ⁸⁵Kr that was measured in the gaseous effluent.

^bMeasured effluent data taken from Environmental Effects of Producing Electric Power, Part 2 (Vol. 1), pp. 1711-1716, and NFS Quarterly Reports.

^cPercent of 10 CFR 20 concentration guides in Cattaraugus Creek.

^dPercent of release limits imposed by AEC operating license.

Table 5.1. Solid radioactive waste^a

Waste type	Radioactivity level		Annual amounts (ft ³)
	Beta-gamma	Alpha	
Undissolved fuel element hulls ^b	High ^c	Low	15,000-30,000
Other fuel element parts and discarded equipment ^d	Low	Low	6,000-10,000
Laboratory waste, small tools, gloves, clothing, etc.	Low	Medium	8,000-43,000

^a1500 metric tons of fuel per year. Data taken from ref. 2.

^bNot compacted.

^cThe high beta-gamma activity is due primarily to the neutron-induced activity in the fuel cladding and hardware.

^dAll discarded equipment will be decontaminated prior to disposal.

Table 5.2. Sewage discharged^a

Total dissolved solids	560 ppm
Total suspended solids	40 ppm
Total volatile solids	20 ppm
Total solids	600 ppm
Five-day biochemical oxygen demand (BOD)	15
Ultimate BOD	25
Alkalinity	350
Calcium	20
Dissolved oxygen	6.0
Hardness	30.0
Magnesium	5.0
Nitrogen-ammonia	7.5
Nitrogen-nitrate	10.5
Nitrogen-organic	18.0
Phosphorus-ortho	32
pH	8
Potassium	2
Sodium	10
Sulfate	10
Chemical oxygen demand (COD)	33
Total organic carbon	60
<u>E. coli</u>	2300, most prob. colonies
Fecal <u>E. coli</u>	2300, most prob. colonies
Fecal streptococcus	230, most prob. colonies

^aData taken from ref. 2.

Table 5.3. Estimated average nonradioactive gaseous effluents

	H ₂ O (lb/hr)	NO ₂ ^a (mg/sec)	SO ₂ (mg/sec)	CO (mg/sec)
Main stack (100 m)	14,000	13,000	nil	nil
Service concentrator	4,560	nil	nil	nil
Cold off-gas	nil	580	nil	nil
Process boilers (gas) ^b	13,900	2,280	4.1	270
Process boilers (oil) ^b	9,800	6,270	3,290	270
Administration boiler (oil)	70	73.8	123	0.62
Furnaces (oil)	62	56.1	92	0.45

^aThe estimated total annual release of NO_x (assumed to be NO₂) normalized to a 7200-hr year is 577 metric tons, 337 metric tons from the process (dissolver and vessel off-gas), and 240 metric tons from service facilities.

^bNormal fuel is natural gas.

Table 6.1. Estimated annual costs, reprocessing cost, and contribution to power cost for the 1500-metric ton/year model fuel reprocessing plant and radwaste treatment cases

Radwaste treatment case	Capital cost ^a (\$1000)	Annual fixed charges (\$1000)	Annual operating cost ^b (\$1000)	Total annual cost (\$1000)	Reprocessing cost ^c [\$ /kg(U)]	Contribution to power cost ^d (mills/kWhr)
<u>Total Cost - Base Plant</u>						
1 ^e	125,000	32,500	13,000	45,500	30.33	0.12
<u>Additional Cost for Radwaste Treatment Systems</u>						
2a	92 (125,092)	24 (32,524)	11 (13,011)	35 (45,535)	0.02 (30.35)	0.000092 (0.120092)
2b	1,102 (126,102)	287 (32,787)	115 (13,115)	402 (45,902)	0.27 (30.60)	0.00107 (0.12107)
2c	1,247 (126,247)	324 (32,824)	131 (13,131)	455 (45,955)	0.30 (30.63)	0.00120 (0.12120)
3	4,491 (129,491)	1,168 (33,668)	467 (13,467)	1,635 (47,135)	1.09 (31.42)	0.00424 (0.12424)
4	1,059 (126,059)	275 (32,775)	28 (13,028)	303 (45,803)	0.20 (30.53)	0.000786 (0.120786)
5	340 (125,340)	88 (32,588)	35 (13,035)	123 (45,623)	0.08 (30.41)	0.000319 (0.120319)
6	2,744 (127,744)	713 (33,213)	285 (13,285)	998 (46,498)	0.67 (31.00)	0.00259 (0.12259)
6c	9,881 (134,881)	2,569 (35,069)	946 (13,946)	3,514 (49,014)	2.35 (32.68)	0.00914 (0.12914)
7 ^f	125,000 (250,000)	32,500 (65,000)	13,000 (26,000)	45,500 (91,000)	30.33 (60.66)	0.12 (0.24)

^aSystem and structural capital cost consists of direct and indirect cost. The interest during construction is included as an indirect cost.

^bAnnual operating costs are estimated at 40% of annual fixed charges with the exception of the sand filter in Case 4 which is estimated at 10%.

^cThe reprocessing cost equals the annual cost divided by the 1.5×10^6 kg per year of uranium charged to the reactor.

^dThe contribution to power cost is computed on the basis of a 1500-metric ton/year reprocessing plant servicing a nuclear economy of fifty-five 1000-MW(e) LWRs (irradiation level, 33,000 MWd/metric ton; load factor, 80%; thermal efficiency, 32.5%). The costs include the direct charges but do not include the effect of carrying charges on fuel working capital.

^eCase 1, the base case, represents a complete model nuclear fuel reprocessing plant which produces uranium nitrate and plutonium nitrate products and stores high-level liquid waste. The capital cost of the plant is \$125,000,000. Radwaste treatment Cases 2a-6 are additions to the base case, and consequently the total capital cost for Cases 2a through 6 would be \$125,000,000 plus the capital cost of the radwaste treatment case. The total capital cost for Case 6c is the cost of Case 1 plus the cost of Cases 2c through 6. The numbers in parentheses are total cost, i.e., base case plus added radwaste treatment cost.

^fCase 7 represents the advanced concept of a "near zero release" plant. The capital cost is taken as twice the base case, i.e., \$250,000,000 for comparative purposes. The difference in the capital cost between the base case and Case 7, \$125,000,000, is taken as the added cost of radwaste treatment for Case 7 as compared to Case 1.

Table 6.2. Estimated incremental costs and amounts of radioactive materials released in the gaseous effluent discharged from the 100-m stack for the radwaste treatment cases

	Radwaste treatment case									
	1	2a	2b	2c	3	4	5	6	6c	7
Capital cost, ^a \$1000	(125,000)	92	1,106 (1,014) ^b	1,247 (141) ^c	4,491	1,059	340	2,744	9,881	125,000
Annual cost, ^a \$1000	(45,500)	35	404 (369) ^b	455 (41) ^c	1,635	303	123	998	3,514	45,500
Calculated radioactive materials released, ^d Ci/year										
H-3	7.7×10^5	7.7×10^5	7.7×10^5	7.7×10^5	7.7×10^5	7.7×10^5	7.7×10^5	7.7×10^3	7.7×10^3	7.7×10^2
Kr-85	1.4×10^7	1.4×10^7	1.4×10^7	1.4×10^7	<u>1.4×10^5</u>	1.4×10^7	1.4×10^7	1.4×10^7	<u>1.4×10^5</u>	<u>1.4×10^3</u>
C-14	8.3×10^2	8.3×10^2	8.3×10^2	8.3×10^2	<u>8.3</u>	8.3×10^2	<u>1.66×10^2</u>	8.3×10^2	<u>8.3</u>	<u>8.3×10^{-1}</u>
I-129	2.5	<u>1.0×10^{-1}</u>	<u>5.0×10^{-3}</u>	<u>5.0×10^{-3}</u>	2.5	2.5	2.5	<u>1.7</u>	<u>5.0×10^{-3}</u>	<u>1.0×10^{-5}</u>
I-131	34.6	<u>6.9×10^{-1}</u>	<u>1.4×10^{-1}</u>	<u>1.4×10^{-1}</u>	34.6	34.6	34.6	<u>13.8</u>	<u>1.4×10^{-1}</u>	<u>2.8×10^{-4}</u>
Semivolatiles	7.0	7.0	7.0	7.0	7.0	0.70	<u>0.70</u>	7.0	<u>0.70</u>	<u>7.0×10^{-4}</u>
Uranium	4.2×10^{-5}	4.2×10^{-5}	4.2×10^{-5}	4.2×10^{-5}	4.2×10^{-5}	<u>4.2×10^{-6}</u>	4.2×10^{-5}	4.2×10^{-5}	<u>4.2×10^{-6}</u>	<u>2.0×10^{-9}</u>
Plutonium	0.8	0.8	0.8	0.8	0.8	<u>0.08</u>	0.8	0.8	<u>0.08</u>	<u>1.6×10^{-4}</u>
Particulates	7.0	7.0	7.0	7.0	7.0	<u>0.70</u>	7.0	7.0	<u>0.70</u>	<u>3.5×10^{-3}</u>

^aThe cost is in addition to the base Case 1, Cases 2b and 2c include Case 2a.

^bThe difference in cost between Cases 2a and 2b.

^cThe difference in cost between Cases 2b and 2c.

^dThe nuclide(s) affected by each radwaste treatment case are underlined.

Table 6.3. Installed costs^a of equipment for model fuel reprocessing plant radwaste treatment Cases 2a-6c

Radwaste treatment case		Costs without structure (\$1000)	
		Direct ^b	Capital ^c
2a	Ion exchange	10	25
2b	Ion exchange, mercury recycle, second $\text{Hg}(\text{NO}_3)_2$ - HNO_3 scrubber, iodine evolution	201	500
2c	Ion exchange, iodine evolution, Iodox	914	2,276
3	Selective adsorption system including catalytic oxidizer	1,289 ^d	3,209 ^d
4	Sand filter	425	1,059
5	Caustic scrubber	94	235
6	Voloxidation	472	1,176
6c	Case 2c through Case 6	2,274	5,662

^aDetails of the cost estimates are presented in Appendix A.

^bCost for 1973. Direct cost includes purchase cost and complete installation cost.

^cCapital costs are calculated by multiplying the direct cost by 2.49. Capital costs include direct costs and indirect costs.

^dIncludes the cost for a station for bottling the noble gases and carbon dioxide in pressurized cylinders. The costs do not include funds for additional shielded space for the long-term storage of the cylinders of gases. Temporary storage for about a year is provided in the model plant in existing shielded areas and in the fuel receiving and storage canal. Shipment of the cylinders to a permanent storage facility is expected as presented in Management of Noble-Gas Fission-Product Wastes From Reprocessing Spent fuels, ORNL-TM-2677.

Table 6.4. Factors used to determine the value of spent light water reactor fuel irradiated to 33,000 MWd/metric ton

	PWR ^a	BWR ^b
Discharge enrichment	0.84	0.63
Grams fissile plutonium per kg U discharged	6.51	6.05
Tails enrichment	0.20	0.20
Separative work units per kg U discharged	0.155	-0.081
Cost per unit, \$/kg U discharged	38.50	38.50
Separative work cost	5.97	-3.12
Value of U (based on \$8/lb U ₃ O ₈), \$/kg U discharged	20.80	17.89
Total value of U, \$/kg U discharged	26.77	14.77
Indifference value of Pu, \$/g fissile Pu	8.00	8.00
Value of Pu, \$/kg U discharged	52.08	48.40
Total U+Pu value, \$/kg U discharged	78.85	63.17
Total U+Pu value, \$/kg U charged ^c	75.35	60.37

^aThe characteristics of the spent PWR fuel was calculated with computer code ORIGIN (information on ORIGIN is available in ref. 3, Sect. 4.6).

^bThe source for the characteristics of the spent BWR fuel was WASH-1082, pp. 5-57.

^cThe capacity of the reprocessing plant is expressed in terms of metric tons of uranium charged to the reactor, although the weight of fuel discharged from the reactor and reprocessed is about 5% less than the amount charged. Therefore, the value of the U+Pu in fuel to be reprocessed is also expressed in terms of the amount of uranium charged to the reactor.

Table 7.1. Latitude-longitude coordinates used to derive
data sets for population distribution

Site	Latitude (N)	Longitude (W)
Midwestern	35° 52' 50"	97° 35' 00"
	38° 12' 18"	90° 28' 28"
	41° 22' 43"	88° 16' 36"
Coastal	33° 15' 00"	81° 29' 20"
	33° 53' 13"	80° 55' 58"
	34° 19' 19"	77° 56' 12"

Table 7.2. Representative population distribution at successive distances for midwestern site

Sector	Radial distance (miles)											
	0-0.5	0.5-1	1-2	2-3	3-4	4-5	5-10	10-15	15-25	25-35	35-45	45-55
N	0	0	0	0	0	252	2007	1037	19193	108738	96229	46889
NNE	0	0	0	0	0	816	847	7688	40643	347330	300030	300804
NE	0	0	0	0	0	709	936	23608	22601	77981	625661	575054
ENE	0	0	0	0	652	1197	1906	1377	8737	85826	192983	110272
E	0	0	0	365	0	452	3506	254	1824	10629	14875	24482
ESE	0	0	0	0	69	2	799	972	3323	4470	8449	4378
SE	0	0	0	13	537	482	1022	696	3241	23827	5080	15453
SSE	0	0	0	0	0	0	1796	706	10056	41868	4461	7339
S	0	0	0	87	0	72	1498	908	30234	100668	10935	17328
SSW	0	0	0	0	0	98	626	586	3588	6416	7425	3933
SW	0	0	146	0	0	0	2233	428	2614	6862	1717	3257
WSW	0	0	0	0	526	0	907	202	1380	8621	2690	4601
W	0	0	0	0	0	0	3128	655	4400	8192	14438	8317
WNW	0	0	0	0	132	77	505	402	1424	6379	4908	3646
NW	0	260	0	0	0	0	346	1083	8288	5991	6200	4146
NNW	0	0	0	0	544	0	579	829	5823	5027	28615	20359
Total (by distance)	0	260 ±449 ^a	146 ±220	465 ±804	2460 ±1453	4157 ±4280	22641 ±8469	40498 ±49447	167369 ±42111	848825 ±378192	1324696 ±1536279	1150618 ±1698458
Cumulative	0	260	406	871	3331	7488	30129	70627	237996	1086821	2411517	3562135
Density (ind./mile ²)	<div> <div><----- 95 -----></div> <div>96</div> <div><----- 126 -----></div> <div><----- 440 -----></div> </div>											

^aStandard deviation of the mean (total).

Table 7.3. Representative population distribution at successive distances for coastal site

Sector	Radial distance (miles)											
	0-0.5	0.5-1	1-2	2-3	3-4	4-5	5-10	10-15	15-25	25-35	35-45	45-55
N	0	0	0	151	0	46	10358	7761	3512	4060	4835	9942
NNE	0	0	0	0	0	0	965	1147	1978	3115	5985	17515
NE	0	0	0	0	0	0	438	284	1139	6646	27892	7382
ENE	0	0	0	0	443	0	847	1119	4112	6321	12413	9022
E	0	0	0	0	0	239	2539	801	1553	17556	4215	5544
ESE	0	0	0	0	0	0	1726	420	660	2463	4700	6466
SE	0	0	0	0	246	213	1710	933	1453	3261	2909	4130
SSE	0	0	0	35	282	0	5954	1780	3546	2991	3247	3380
S	0	0	0	0	250	570	12327	1095	2803	9367	2829	2744
SSW	0	0	0	0	0	0	0	318	1518	2978	5556	4590
SW	0	0	0	0	0	0	710	990	1620	3953	4320	4846
WSW	0	1112	0	0	0	0	0	470	732	3309	2833	13724
W	0	0	0	0	0	0	1313	669	1975	5684	7106	10573
WNW	0	0	0	0	0	0	1568	4341	5456	42402	24875	7668
NW	0	0	0	0	0	7	7970	11817	8353	13856	4110	7239
NNW	0	0	0	0	421	310	15334	22775	4024	8447	5564	9189
Total (by distance)	0	1112 ±1926 ^a	0 -	186 ±237	1642 ±927	1385 ±1555	63759 ±54948	56720 ±79376	44434 ±17548	136409 ±93262	123389 ±30247	123954 ±29498
Cumulative	0	1112	1112	1298	2940	4325	68084	12804	169238	305631	42902	552974
Density (ind./mile ²)	<div> <div>← 55 →</div> <div>289</div> <div>← 61 →</div> <div>← 51 →</div> </div>											

^aStandard deviation of the mean (total).

Table 7.4. Maximum annual total-body and organ doses^a to individuals^b from gaseous effluents at 0.5 mile from the model LWR fuel reprocessing plant

Radwaste treatment case	Maximum total-body dose (millirem)	Maximum adult organ dose (millirem)									
		GI tract	Bone	Thyroid	Lungs	Muscle	Kidney	Liver	Spleen	Testes	Ovaries
Midwestern site											
1	2.2E+1	5.3E+1	3.9E+1	4.5E+2	2.2E+1	2.0E+1	2.1E+1	2.0E+1	1.9E+1	2.1E+1	1.5E+1
2a	1.9E+1	5.3E+1	3.7E+1	3.1E+1	2.1E+1	1.8E+1	1.9E+1	1.9E+1	1.8E+1	1.8E+1	1.4E+1
2b	1.9E+1	5.3E+1	3.6E+1	1.9E+1	2.1E+1	1.8E+1	1.9E+1	1.9E+1	1.7E+1	1.8E+1	1.4E+1
2c	1.9E+1	5.3E+1	3.6E+1	1.9E+1	2.1E+1	1.8E+1	1.9E+1	1.9E+1	1.7E+1	1.8E+1	1.4E+1
3	1.5E+1	4.9E+1	3.1E+1	4.4E+2	1.4E+1	1.5E+1	1.6E+1	1.5E+1	1.4E+1	1.6E+1	1.2E+1
4	1.8E+1	1.8E+1	2.2E+1	4.4E+2	1.9E+1	1.6E+1	1.6E+1	1.5E+1	1.4E+1	1.7E+1	1.3E+1
5	1.9E+1	1.8E+1	3.5E+1	4.5E+2	2.1E+1	1.7E+1	1.8E+1	1.8E+1	1.7E+1	1.9E+1	1.4E+1
6	1.2E+1	4.5E+1	3.0E+1	2.4E+2	1.3E+1	1.0E+1	1.2E+1	9.6E00	9.6E00	1.1E+1	5.7E00
6c	5.2E-1	7.0E-1	2.0E00	1.8E00	4.7E-1	5.5E-1	5.9E-1	5.4E-1	5.4E-1	5.2E-1	3.4E-1
7	1.4E-2	1.6E-2	2.1E-2	1.5E-2	1.2E-2	1.4E-2	1.3E-2	1.3E-2	1.3E-2	1.2E-2	1.1E-2
Coastal site											
1	1.6E+1	4.3E+1	2.9E+1	3.7E+2	1.6E+1	1.5E+1	1.5E+1	1.5E+1	1.4E+1	1.5E+1	1.1E+1
2a	1.4E+1	4.2E+1	2.7E+1	2.4E+1	1.5E+1	1.3E+1	1.4E+1	1.4E+1	1.3E+1	1.4E+1	9.9E00
2b	1.4E+1	4.2E+1	2.6E+1	1.4E+1	1.5E+1	1.3E+1	1.4E+1	1.4E+1	1.3E+1	1.3E+1	9.9E00
2c	1.4E+1	4.2E+1	2.6E+1	1.4E+1	1.5E+1	1.3E+1	1.4E+1	1.4E+1	1.3E+1	1.3E+1	9.9E00
3	1.2E+1	4.0E+1	2.4E+1	3.7E+2	1.0E+1	1.2E+1	1.2E+1	1.2E+1	1.1E+1	1.2E+1	9.4E00
4	1.3E+1	1.3E+1	1.5E+1	3.7E+2	1.3E+1	1.1E+1	1.1E+1	1.1E+1	1.1E+1	1.2E+1	9.1E00
5	1.1E+1	1.3E+1	2.6E+1	3.7E+2	1.5E+1	1.3E+1	1.3E+1	1.3E+1	1.3E+1	1.4E+1	1.0E+1
6	8.7E00	3.6E+1	2.2E+1	2.0E+2	9.0E00	7.6E00	8.4E00	8.0E00	7.1E00	8.2E00	4.2E00
6c	4.1E-1	5.8E-1	1.5E00	1.4E00	3.6E-1	4.3E-1	4.5E-1	4.9E-1	4.2E-1	4.1E-1	2.6E-1
7	1.0E-2	1.3E-2	1.5E-2	1.1E-2	8.9E-3	1.0E-2	9.5E-3	9.7E-3	9.2E-3	8.9E-3	8.1E-3

^a A 50-year dose commitment from exposure to plant effluents from 1 year of model plant operation.

^b Dose to individuals located downwind of the prevailing wind direction. Daily food intake is assumed to be 0.25 kg of vegetables, 0.3 kg of meat, and 1 liter of milk. It is assumed that all food consumed is produced or grown at the location of residence.

Table 7.5. Maximum annual total-body and organ doses^a to individuals^b from gaseous effluents at 1.5 miles from the model LWR fuel reprocessing plant

Radwaste treatment case	Maximum total-body dose (millirem)	Maximum adult organ dose (millirem)									
		GI tract	Bone	Thyroid	Lungs	Muscle	Kidney	Liver	Spleen	Testes	Ovaries
Midwestern site											
1	1.4E+1	3.5E+1	2.6E+1	2.9E+2	1.2E+1	1.4E+1	1.6E+1	1.4E+1	1.3E+1	1.3E+1	1.1E+1
2a	1.3E+1	3.5E+1	2.4E+1	2.0E+1	1.1E+1	1.3E+1	1.3E+1	1.3E+1	1.2E+1	1.2E+1	1.0E+1
2b	1.3E+1	3.5E+1	2.4E+1	1.2E+1	1.1E+1	1.3E+1	1.3E+1	1.3E+1	1.2E+1	1.1E+1	1.0E+1
2c	1.3E+1	3.5E+1	2.4E+1	1.2E+1	1.1E+1	1.3E+1	1.3E+1	1.3E+1	1.2E+1	1.1E+1	1.0E+1
3	1.1E+1	3.3E+1	2.3E+1	2.9E+2	1.0E+1	1.2E+1	1.2E+1	1.2E+1	1.1E+1	1.2E+1	9.7E00
4	1.2E+1	1.2E+1	1.4E+1	2.9E+2	1.0E+1	1.1E+1	1.2E+1	1.0E+1	9.9E00	1.1E+1	9.3E00
5	1.2E+1	1.3E+1	2.3E+1	2.9E+2	1.1E+1	1.2E+1	1.2E+1	1.2E+1	1.1E+1	1.2E+1	9.9E00
6	6.2E00	2.8E+1	1.8E+1	1.5E+2	4.6E00	6.3E00	6.1E00	5.9E00	4.9E00	5.3E00	3.2E00
6c	3.4E-1	5.6E-1	1.4E00	1.1E00	2.7E-1	3.7E-1	4.0E-1	4.3E-1	3.5E-1	3.4E-1	2.3E-1
7	1.1E-2	1.2E-2	1.6E-2	1.1E-2	9.4E-3	1.1E-2	1.0E-2	1.1E-2	9.8E-3	9.5E-3	9.0E-3
Coastal site											
1	1.1E+1	2.7E+1	2.0E+1	2.2E+2	9.4E00	1.1E+1	1.0E+1	1.0E+1	9.5E00	1.0E+1	8.5E00
2a	9.9E00	2.7E+1	1.9E+1	1.5E+1	8.8E00	9.7E00	9.7E00	9.6E00	8.9E00	8.7E00	7.6E00
2b	9.8E00	2.7E+1	1.9E+1	9.0E00	8.8E00	9.6E00	9.7E00	9.6E00	8.9E00	8.7E00	7.6E00
2c	9.8E00	2.7E+1	1.9E+1	9.0E00	8.8E00	9.6E00	9.7E00	9.6E00	8.9E00	8.7E00	7.6E00
3	8.8E00	2.5E+1	1.8E+1	2.2E+2	8.1E00	8.8E00	9.2E00	8.9E00	8.4E00	8.9E00	7.4E00
4	9.1E00	9.6E00	1.1E+1	2.2E+2	8.1E00	8.6E00	7.9E00	7.8E00	7.5E00	8.1E00	7.0E00
5	9.3E00	1.0E00	1.8E+1	2.2E+2	8.7E00	9.0E00	9.1E00	9.2E00	8.6E00	9.2E00	7.5E00
6	4.8E00	2.1E+1	1.4E+1	1.2E+2	3.6E00	4.8E00	4.7E00	4.5E00	3.7E00	4.0E00	2.4E00
6c	2.6E-1	4.3E-1	1.1E00	8.7E-1	2.1E-1	2.8E-1	3.0E-1	3.2E-1	2.7E-1	2.5E-1	1.8E-1
7	8.2E-3	9.3E-3	1.2E-2	8.5E-3	7.1E-3	8.3E-3	7.7E-3	7.9E-3	7.4E-3	7.2E-3	6.8E-3

^a A 50-year dose commitment from exposure to plant effluents from 1 year of model plant operation.

^b Dose to individuals located downwind of the prevailing wind direction. Daily food intake is assumed to be 0.25 kg of vegetables, 0.3 kg of meat, and 1 liter of milk. It is assumed that all food consumed is produced or grown at the location of residence.

Table 7.6. Annual total-body and organ doses^a to the population^b from gaseous effluents out to 55 miles from the model LWR fuel reprocessing plant

Radwaste treatment case	Population total-body dose (person-rem)	Population organ doses (person-organ-rem) ^a									
		GI tract	Bone	Thyroid	Lungs	Muscle	Kidney	Liver	Spleen	Testes	Ovaries
Midwestern site											
1	1.04E+3	2.43E+3	1.71E+3	1.23E+4	9.49E+2	1.02E+3	1.01E+3	9.69E+2	9.12E+2	9.66E+2	7.94E+2
2a	9.65E+2	2.41E+3	1.60E+3	1.24E+3	8.99E+2	9.45E+2	9.54E+2	9.27E+2	8.72E+2	8.68E+2	7.55E+2
2b	9.62E+2	2.41E+3	1.59E+3	8.98E+2	8.97E+2	9.42E+2	9.52E+2	9.26E+2	8.71E+2	8.65E+2	7.54E+2
2c	9.62E+2	2.41E+3	1.59E+3	8.98E+2	8.97E+2	9.42E+2	9.52E+2	9.26E+2	8.71E+2	8.65E+2	7.54E+2
3	8.18E+2	2.25E+3	1.43E+3	1.22E+4	7.74E+2	8.23E+2	8.64E+2	8.18E+2	7.88E+2	8.45E+2	7.08E+2
4	9.02E+2	9.39E+2	1.04E+3	1.21E+4	8.22E+2	8.68E+2	8.00E+2	7.95E+2	7.64E+2	8.20E+2	7.16E+2
5	8.81E+2	9.53E+2	1.49E+3	1.22E+4	8.64E+2	8.56E+2	8.76E+2	8.63E+2	8.24E+2	8.91E+2	7.30E+2
6	4.38E+2	1.85E+3	1.09E+3	6.76E+3	3.54E+2	4.16E+2	4.21E+2	3.78E+2	3.22E+2	3.56E+2	2.04E+2
6c	2.20E+1	3.72E+1	7.88E+1	5.22E+1	1.96E+1	2.27E+1	2.53E+1	2.52E+1	2.18E+1	2.15E+1	1.46E+1
7	8.33E-1	9.49E-1	1.09E00	7.98E-1	7.28E-1	8.36E-1	7.78E-1	7.83E-1	7.45E-1	7.28E-1	6.91E-1
Coastal site											
1	2.99E+2	7.89E+2	5.15E+2	4.25E+3	2.69E+2	2.93E+2	2.91E+2	2.78E+2	2.62E+2	2.80E+2	2.54E+2
2a	2.73E+2	7.84E+2	4.78E+2	3.75E+2	2.51E+2	2.68E+2	2.73E+2	2.63E+2	2.48E+2	2.46E+2	2.11E+2
2b	2.71E+2	7.84E+2	4.76E+2	2.56E+2	2.51E+2	2.67E+2	2.72E+2	2.63E+2	2.47E+2	2.45E+2	2.10E+2
2c	2.71E+2	7.84E+2	4.76E+2	2.56E+2	2.51E+2	2.67E+2	2.72E+2	2.63E+2	2.47E+2	2.45E+2	2.10E+2
3	2.39E+2	7.42E+2	4.42E+2	4.21E+3	2.24E+2	2.41E+2	2.54E+2	2.38E+2	2.29E+2	2.49E+2	2.01E+2
4	2.51E+2	2.67E+2	2.94E+2	4.20E+3	2.25E+2	2.37E+2	2.21E+2	2.19E+2	2.10E+2	2.30E+2	1.97E+2
5	2.54E+2	2.82E+2	4.55E+2	4.22E+3	2.44E+2	2.48E+2	2.52E+2	2.48E+2	2.36E+2	2.58E+2	2.06E+2
6	1.34E+2	6.31E+2	3.45E+2	2.35E+3	1.06E+2	1.28E+2	1.29E+2	1.16E+2	1.01E+2	1.12E+2	6.33E00
6c	6.99E00	1.20E+1	2.57E+1	1.77E+1	6.18E00	7.25E00	8.03E00	8.00E00	7.01E00	6.90E00	4.56E00
7	2.31E-1	2.75E-1	3.15E-1	2.26E-1	2.01E-1	2.32E-1	2.17E-1	2.19E-1	2.07E-1	2.02E-1	1.90E-1

^aA 50-year dose commitment from exposure to effluents from 1 year of model plant operation.

^bTo entire population within 55 miles of the model plant; daily intake assumed to be 0.25 kg of vegetables, 300 ml of milk, and 0.3 kg of meat. It is assumed that 100% of the food consumed is produced or grown at the location of residence.

Table 7.7. Contribution of exposure modes to the maximum, annual total-body dose from gaseous effluents at 1.5 miles from the model LWR fuel reprocessing plant^a

Exposure mode	Maximum annual individual total-body dose (millirem)	Percent of total dose
Immersion in air ^b	0.46	3.3
Contaminated ground ^b	2.81	19.9
Inhalation ^c	1.50	10.6
Ingestion ^d	9.34	66.2

^aRadwaste treatment Case 1 at 1.5 miles from plant in the prevailing wind direction; midwestern site.

^bExposure for 100% of the time; no shielding.

^cDaily intake assumed to be 20 m³ of air.

^dDaily intake assumed to be 0.25 kg of vegetables, 1 liter of milk, and 0.3 kg of meat. It is assumed that 100% of the food consumed is produced or grown at the reference location.

Table 7.8. Contribution of major radionuclides^a to the maximum annual total body and organ doses^b to individuals from gaseous effluents at 1.5 miles from the model LWR fuel reprocessing plant

Radionuclide	Percent of total-body and organ dose ^{c,d,e}				
	Total body	GI tract	Bone	Thyroid	Lung
H-3	52.3	21.0	28.4	2.6	59.6
C-14	16.2	5.2	10.8	0.4	8.0
Kr-85	3.3	0.9	2.1	0.2	7.5
I-129	7.7	0.3	5.9	51.1	5.3
I-131	2.1	0.5	1.2	44.9	2.5
Ru-103	0.1	0.9	0.1	<0.1	0.1
Ru-106	2.0	64.4	2.2	0.1	3.0
Sr-89	<0.1	0.1	0.6	<0.1	<0.1
Sr-90	0.6	0.1	17.4	<0.1	0.7
Zr-95	0.3	0.2	0.2	<0.1	0.3
Nb-95	0.2	0.4	0.1	<0.1	0.2
Cs-134	6.0	1.6	3.1	0.3	4.4
Cs-137	6.7	2.1	4.7	0.3	6.7
Ce-144	0.2	1.9	0.3	<0.1	<0.1
Eu-154	0.6	0.2	0.4	<0.1	<0.1
Cm-242	1.4	0.1	7.2	0.1	0.9
Pu-238	0.3	<0.1	7.2	<0.1	0.5
Pu-239	<0.1	<0.1	0.9	<0.1	0.1
Pu-240	<0.1	<0.1	1.4	<0.1	0.1
Pu-241	<0.2	<0.1	5.8	<0.1	<0.1

^aRadionuclides contributing <0.1% not listed.

^bRadwaste treatment Case 1.

^cExposure for 100% of the time; no shielding.

^dAll food is produced and consumed at the reference location. Daily intakes are 1 liter of milk, 0.25 kg of vegetables, and 0.3 kg of beef.

^eInhalation rate of 20 m³ of air per day.

Table 7.9. Contribution of major radionuclides^a to the maximum annual total-body dose^b to individuals by pathway from gaseous effluents at 1.5 miles from the model LWR fuel reprocessing plant

Radionuclide	Percent of total-body dose by pathway			
	Contaminated ^c ground	Submersion ^c in air	Ingestion ^d	Inhalation ^e
H-3	<0.1	<0.1	43.0	9.3
C-14	<0.1	<0.1	16.1	<0.1
Kr-85	<0.1	3.3	<0.1	<0.1
Sr-90	<0.1	<0.1	0.6	<0.1
Zr-95	0.3	<0.1	<0.1	<0.1
Nb-95	0.2	<0.1	<0.1	<0.1
Ru-103	0.1	<0.1	<0.1	<0.1
Ru-106	1.7	<0.1	0.3	<0.1
I-129	6.3	<0.1	1.4	<0.1
I-131	0.4	<0.1	1.7	<0.1
Cs-134	3.6	<0.1	2.3	<0.1
Cs-137	6.0	<0.1	0.7	<0.1
Ce-144	0.1	<0.1	<0.1	<0.1
Eu-154	0.6	<0.1	<0.1	<0.1
Pu-238	<0.1	<0.1	<0.1	0.3
Pu-240	<0.1	<0.1	<0.1	0.1
Pu-241	<0.1	<0.1	<0.1	0.2
Cm-242	0.6	<0.1	<0.1	<0.1

^aRadionuclides contributing <0.1% not listed.

^bRadwaste treatment Case 1.

^cExposure for 100% of the time; no shielding.

^dAll food is produced and consumed at the reference location. Daily intakes are 1 liter of milk, 0.25 kg of vegetables, and 0.3 kg of beef.

^eInhalation rate of 20 m³ of air per day.

Table 7.10. Contribution of major radionuclides^a to the annual total-body and organ doses^b to the population from gaseous effluents out to 55 miles from the model LWR fuel reprocessing plant at the midwestern site

Radionuclide	Percent of total-body and organ dose				
	Total body	GI tract	Bone	Thyroid	Lung
H-3	55.7	23.9	34.0	4.7	61.1
C-14	17.2	5.9	13.0	0.7	8.3
Kr-85	4.6	1.4	3.4	0.4	10.3
I-129	6.5	0.3	5.7	57.9	4.1
I-131	1.2	0.3	0.8	35.1	1.3
Ru-103	0.1	0.9	0.1	<0.1	0.1
Ru-106	1.7	61.4	2.2	0.2	2.4
Sr-89	<0.1	0.1	0.4	<0.1	<0.1
Sr-90	0.4	0.1	13.2	<0.1	0.5
Zr-95	0.3	0.2	0.2	<0.1	0.3
Nb-95	0.1	0.3	0.1	<0.1	0.2
Cs-134	4.2	1.2	2.7	0.3	3.6
Cs-137	5.7	1.9	4.3	0.5	5.6
Ce-144	0.1	1.8	0.2	<0.1	0.4
Eu-154	0.5	0.2	0.4	<0.1	0.6
Cm-242	1.2	0.1	6.2	0.1	0.8
Pu-238	0.3	<0.1	6.1	<0.1	0.4
Pu-239	<0.1	<0.1	0.8	<0.1	<0.1
Pu-240	<0.1	<0.1	1.2	<0.1	0.1
Pu-241	0.2	<0.1	4.9	<0.1	<0.1

^aRadionuclides contributing <0.1% to neither the total-body nor to an organ dose are not included; radwaste treatment Case 1.

^b50-year dose commitment from exposure to the effluents from 1 year of model plant operation.

Table 7.11. Contribution of major radionuclides^a to the annual total-body and organ doses^b to the population from gaseous effluents out to 55 miles from the model LWR fuel reprocessing plant at the southeastern coastal site

Radionuclide	Percent of total-body and organ dose				
	Total body	GI tract	Bone	Thyroid	Lung
H-3	52.5	19.9	30.5	3.7	58.4
C-14	16.2	5.0	11.6	0.5	7.9
Kr-85	3.9	1.1	2.8	0.3	8.9
I-129	8.0	0.3	6.6	58.4	5.1
I-131	1.4	0.3	0.9	35.8	1.6
Ru-103	0.1	0.9	0.1	<0.1	0.1
Ru-106	2.1	66.2	2.6	0.2	2.9
Sr-89	<0.1	0.1	0.5	<0.1	<0.1
Sr-90	0.5	0.1	15.2	<0.1	0.6
Zr-95	0.3	0.2	0.2	<0.1	0.4
Nb-95	0.2	0.3	0.1	<0.1	0.2
Cs-134	5.1	1.3	3.1	0.3	4.4
Cs-137	6.9	1.9	5.0	0.5	7.0
Ce-144	0.2	2.0	0.3	<0.1	0.4
Eu-154	0.6	0.2	0.4	<0.1	0.7
Cm-242	1.3	0.1	6.5	0.1	0.9
Pu-238	0.3	<0.1	6.3	<0.1	0.4
Pu-239	<0.1	<0.1	0.8	<0.1	<0.1
Pu-240	<0.1	<0.1	1.2	<0.1	0.1
Pu-241	0.2	<0.1	5.1	<0.1	<0.1

^aRadionuclides which contribute <1% of the dose to neither the total body nor to an organ are not listed; radwaste treatment Case 1.

^b50-year dose commitment from exposure to the effluents from 1 year of model plant operation.

Table 7.12. Contribution of the major radionuclides^a to the maximum annual organ doses of individuals for the inhalation and ingestion pathways from gaseous effluents at 1.5 miles from the model LWR fuel reprocessing plant^b

Radionuclide	GI tract		Bone		Thyroid		Lung	
	Inhaled	Ingested	Inhaled	Ingested	Inhaled	Ingested	Inhaled	Ingested
H-3	3.7	17.3	5.0	23.3	0.5	2.1	10.4	48.5
C-14	<0.1	5.2	<0.1	10.8	<0.1	0.4	<0.1	7.9
Sr-89	<0.1	0.1	<0.1	0.5	<0.1	<0.1	<0.1	<0.1
Sr-90	<0.1	0.1	0.1	17.2	<0.1	<0.1	<0.1	0.7
Ru-103	<0.1	0.9	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1
Ru-106	<0.1	63.6	<0.1	1.1	<0.1	<0.1	0.8	0.5
I-129	<0.1	<0.1	<0.1	0.8	0.1	50.8	<0.1	1.6
I-131	<0.1	0.4	<0.1	0.9	0.3	44.5	0.1	1.9
Cs-134	<0.1	0.6	<0.1	0.8	<0.1	0.1	<0.1	0.5
Cs-137	<0.1	0.4	<0.1	0.8	<0.1	<0.1	<0.1	0.2
Ce-144	<0.1	1.9	0.2	<0.1	<0.1	<0.1	0.3	<0.1
Pu-238	<0.1	<0.1	7.1	0.1	<0.1	<0.1	0.5	<0.1
Pu-239	<0.1	<0.1	0.9	<0.1	<0.1	<0.1	<0.1	<0.1
Pu-240	<0.1	<0.1	1.4	<0.1	<0.1	<0.1	0.1	<0.1
Pu-241	<0.1	<0.1	5.7	0.1	<0.1	<0.1	<0.1	<0.1
Cm-242	<0.1	<0.1	6.3	0.2	<0.1	<0.1	0.4	<0.1

^aRadionuclides contributing <0.1% are not listed.

^bRadwaste treatment Case 1.

^cDaily intake of 20 m³ of air is assumed.

^dDaily intake of 0.25 kg of vegetables, 0.3 kg of meat, and 1 liter of milk is assumed. It is also assumed that 100% of the food consumed is produced or grown at the reference location.

Table 7.13. Curies and concentration of long-lived radionuclides released during the 30-year life of the model LWR fuel reprocessing plant

Radionuclide	Total released ^a during 30-yr plant life (Ci)	Activity concentration ^b (Ci/m ²)
C-14	2.5E+4	-
I-129	7.5E+1	3.0E-9
Pu-238	6.3E-1	2.6E-11
Pu-239	7.2E-2	2.9E-12
Pu-240	1.1E-1	4.5E-12
Pu-241	2.3E+1	9.3E-10
Pu-242	3.0E-4	1.2E-14
Am-241	1.4E-2	5.7E-13
Am-243	1.6E-3	6.5E-14
Cm-242	1.5E00	6.1E-11
Cm-244	2.1E-1	8.5E-12

^aRadwaste treatment Case 1. These values divided by $2.46 \times 10^{10} \text{ m}^2$ equal the assumed deposition rate; midwestern site.

^bDeposition is assumed to occur uniformly out to a distance of 55 miles and are dispersed over an area of $2.46 \times 10^{10} \text{ m}^2$.

Table 7.14. Contribution of radionuclides and exposure modes to the annual total-body dose^a to individuals after the model LWR fuel reprocessing plant is closed^b until the significant decay of all radionuclides occurs

Radionuclide ^c	Exposure mode			
	Contaminated ground (millirem)	Inhalation (millirem)	Ingestion (millirem)	Total (millirem)
C-14	-	-	3.6E-2	3.6E-2
I-129	4.1E-2	2.1E-7	3.3E-4	4.1E-2
Pu-238	7.0E-5	2.7E-5	3.6E-7	9.7E-5
Pu-239	3.0E-6	3.6E-6	4.6E-8	6.6E-6
Pu-240	1.1E-5	5.4E-6	5.1E-8	1.6E-5
Pu-241	0	1.7E-5	2.3E-7	1.7E-5
Pu-242	2.8E-8	1.4E-8	1.8E-10	4.2E-8
Am-241	7.0E-6	5.9E-7	1.6E-8	7.6E-6
Am-243	2.7E-6	6.5E-8	1.9E-9	2.8E-6
Cm-242	1.5E-4	1.5E-6	6.3E-8	1.5E-4
Cm-244	2.2E-5	4.6E-6	2.1E-7	2.7E-5
Total	4.1E-2	6.0E-5	3.6E-2	7.7E-2

^aDose is the average total-body dose of the individuals out to a distance of 55 miles from the plant; radwaste treatment Case 1.

^bA 30-year lifetime for the plant is assumed; midwestern site.

^cRadionuclides which contribute <0.02% to the total dose are not included.

Table 7.15. Annual dose^a to individuals after the model LWR fuel reprocessing plant^b is closed until significant decay of all radionuclides occurs

Radionuclide ^c	Organ dose per exposure mode (millirem)							
	Bone		Thyroid		Liver		Lung	
	Inhalation	Ingestion	Inhalation	Ingestion	Inhalation	Ingestion	Inhalation	Ingestion
C-14	-	2.3E-2	-	1.7E-2	-	2.3E-2	-	1.6E-2
I-129	2.1E-7	3.3E-4	8.9E-5	2.6E-1	2.1E-7	2.8E-4	2.0E-6	3.1E-4
Pu-238	1.1E-3	1.4E-5	2.7E-5	4.3E-7	1.5E-4	2.0E-6	3.6E-5	3.6E-7
Pu-239	1.4E-4	1.9E-6	3.6E-6	5.6E-8	1.9E-5	2.4E-7	3.9E-6	4.6E-8
Pu-240	2.2E-4	2.9E-6	5.4E-6	8.6E-8	2.9E-5	3.6E-7	6.1E-6	7.1E-8
Pu-241	8.6E-4	1.1E-5	1.7E-5	2.3E-7	4.3E-5	5.6E-7	1.1E-6	2.3E-7
Pu-242	5.8E-7	7.4E-9	1.4E-8	2.2E-10	7.6E-8	9.9E-10	1.6E-8	1.8E-10
Am-241	9.2E-6	2.4E-7	2.6E-7	1.6E-8	3.0E-6	1.3E-7	5.9E-7	1.9E-8
Am-243	1.0E-6	2.7E-8	6.5E-8	2.3E-9	3.3E-7	7.9E-9	2.9E-8	1.9E-9
Cm-242	2.3E-5	9.6E-7	1.5E-6	6.3E-8	2.4E-5	6.6E-7	1.9E-7	6.3E-8
Cm-244	7.9E-5	4.3E-6	4.6E-6	2.1E-7	3.3E-5	1.5E-7	4.0E-6	2.1E-7
Total	2.4E-3	2.3E-2	1.5E-4	2.8E-1	3.0E-4	2.3E-2	5.4E-5	1.6E-2

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^aDose is the average organ dose for individuals out to a distance of 55 miles; radwaste treatment Case 1; midwestern site.

^bAn operating lifetime of 30 years is assumed.

^cRadionuclides which contribute <0.02% of the dose are not included.

Table 7.16. Annual dose^a to the population after the model LWR fuel reprocessing plant^b is closed until significant decay of all radionuclides occurs

Radionuclide ^d	Dose (person-rem per 3.6×10^6 persons ^c)				
	Total body	Bone	Thyroid	Liver	Lung
C-14	1.3E+2	8.1E+1	6.1E+1	8.1E+1	5.7E+1
I-129	1.5E+2	1.1E00	7.7E+2	9.8E-1	1.3E00
Pu-238	3.5E-1	3.8E00	1.0E-1	5.5E-1	1.3E-1
Pu-239	2.4E-2	5.2E-1	1.3E-2	6.9E-2	1.4E-2
Pu-240	5.8E-2	8.1E-1	2.0E-2	1.1E-1	2.2E-2
Pu-241	6.1E-2	3.1E00	6.2E-2	1.6E-1	4.8E-3
Pu-242	1.5E-4	2.1E-3	5.2E-5	2.8E-4	5.8E-5
Am-241	2.7E-2	3.3E-2	2.2E-3	1.1E-2	1.1E-3
Am-243	1.0E-2	3.7E-3	1.1E-4	1.2E-3	1.1E-4
Cm-242	5.4E-1	8.6E-2	5.6E-3	8.9E-2	9.1E-4
Cm-244	9.7E-2	3.0E-1	1.7E-2	1.2E-1	1.5E-2
Total	2.8E+2	9.1E+1	8.3E+2	8.3E+1	5.8E+1

^aDose to the population is the sum of the individual doses out to 55 miles from the plant; radwaste treatment Case 1; midwestern site.

^bA lifetime of 30 years is assumed for plant operation.

^cActual population within a 55-mile radius of the midwestern plant site.

^dRadionuclides which contribute <0.02% of the dose are not included.

Table 7.17. Effect of buoyancy of hot stack gases on plume rise and doses from carbon-14 at 1.5 miles from the model LWR fuel reprocessing plant

Temperature of stack gases (°F)	Plume rise (meters)	Fractional residue dose ^d
55 ^a	14 ^b	1.00
125	134 ^c	0.24
150	149 ^c	0.21
200	156 ^c	0.19
250	189 ^c	0.14
300	204 ^c	0.12

^aMidwestern site average air temperature = 55°F.

^bPlume rise resulting only from the momentum of the stack gases in 100-m stack.

^cPlume rise (100-m stack) resulting only from buoyancy. Stack gas momentum would add only a small amount to these values.

^dFactors are different for other radionuclides.

Table 8 1. Cost/benefit analysis for the model LWR fuel reprocessing plant at the midwestern site

Radwaste treatment case	Annual cost ^a increment (\$10 ³)	Unit cost increment (\$/kg U)	Annual population dose ^b and overall dose reduction ^c (person-rem)				Maximum individual total body dose ^d and overall dose reduction (mrem)	Cost/benefit ratio, total body	
			Total body	GI tract	Bone	Thyroid		$\left(\frac{\$10^3}{\text{annual population person-rem}} \right)$	$\left(\frac{\$10^3}{\text{maximum individual mrem}} \right)$
1	0	0.00	1,042	2,428	1,704	12,286	14.05	0	0
2a	35	0.023	965 (77)	2,414 (14)	1,598 (106)	1,237 (11,049)	12.72 (1.33)	0.45	26.32
2b	404	0.27	962 (80)	2,413 (15)	1,594 (110)	898 (11,388)	12.68 (1.37)	5.05	294.89
2c	455	0.30	962 (80)	2,413 (15)	1,594 (110)	898 (11,388)	12.68 (1.37)	5.69	332.12
3	1,635	1.09	818 (224)	2,252 (176)	1,428 (276)	12,155 (131)	11.35 (2.70)	7.30	605.56
4	303	0.20	902 (140)	938 (1,490)	1,043 (661)	12,148 (138)	11.72 (2.33)	2.16	130.04
5	123	0.08	881 (161)	953 (1,475)	1,492 (212)	12,202 (84)	11.97 (2.08)	0.76	59.13
6	998	0.67	438 (604)	1,847 (581)	1,090 (614)	6,756 (5,530)	6.24 (7.81)	1.65	127.78
6c	3,514	2.34	22 (1,020)	37 (2,391)	79 (1,625)	52 (12,234)	0.34 (13.71)	3.45	256.31
7	45,500	30.33	0.67 (1,041)	0.82 (2,427)	0.89 (1,703)	0.72 (12,285)	0.01 (14.04)	43.71	3,241

^aCost of additional equipment required for each case with reference to Case 1.^bThe average annual population dose out to a distance of 55 miles from the stack ground zero.^cThe number in parentheses is the reduction in dose with reference to Case 1.^dThe maximum annual individual (adult) total body dose at 1.5 miles from the stack ground zero.

Table 8 2 Cost/benefit analysis for the model LWR fuel reprocessing plant at the east-coastal site

Radwaste treatment case	Annual cost ^a increment (\$10 ³)	Unit cost increment (\$/kg U)	Annual population dose ^b and overall dose reduction ^c (person-rem)				Maximum individual total body dose ^d and overall dose reduction (mrem)	Cost/benefit ratio total body	
			Total body	GI tract	Bone	Thyroid		$\left(\frac{\$10^3}{\text{annual population person-rem}} \right)$	$\left(\frac{\$10^3}{\text{maximum individual mrem}} \right)$
1	0	0 00	299	787	515	4,245	10 89	0	0
2a	35	0 023	272 (27)	782 (6)	478 (37)	375 (3,870)	9 86 (1 03)	1 30	2 98
2b	404	0 27	271 (28)	782 (6)	476 (39)	256 (3,989)	9 80 (1 07)	14 42	7 27
2c	455	0 30	271 (28)	782 (6)	476 (39)	256 (3,989)	9 82 (1 07)	16 25	425 2
3	1,635	1 09	239 (60)	740 (47)	442 (73)	4,211 (34)	8 80 (2 09)	27 25	782 0
4	303	0 20	251 (48)	267 (520)	294 (221)	4 197 (48)	9 08 (1 81)	0 21	167 40
5	123	0 08	254 (45)	280 (507)	455 (60)	4 221 (24)	9 28 (1 61)	73	76 40
6	998	0 67	133 (166)	629 (158)	245 (170)	2 351 (1 894)	4 83 (6 06)	6 01	164 63
6c	3,514	2 34	7 (292)	12 (775)	26 (489)	18 (4 227)	0 26 (10 6)	12 0	30 57
7	45,500	30 33	0 19 (299)	0 24 (787)	0 26 (515)	0 21 (4 245)	0 01 (10 88)	15 17	4 1

^aCost of additional equipment required for each case with reference to Case 1^bThe average annual population dose out to a distance of 55 miles from the stack ground zero^cThe number in parentheses is the reduction in dose with reference to Case 1^dThe maximum annual individual (adult) total body dose at 1.5 miles from the stack ground zero

Table 8.3. Cumulative annual cost increments, dose decrements, and cost/benefits for additional radwaste treatments at the model LWR fuel reprocessing plant at the midwestern site

Radwaste treatment	Cumulative annual cost increment ^a (\$10 ³)	Cumulative unit cost increment (\$/kg U)	Annual population and overall dose reduction ^b (person-rem)				Maximum individual ^c total body dose and overall dose reduction (mrem)	Cost/benefit ratio, total body	
			Total body	GI tract	Bone	Thyroid		$\frac{\$10^3}{\text{annual population person-rem}}$	$\frac{\$10^3}{\text{maximum individual mrem}}$
Case 1	0	0.00	1,042	2,428	1,704	12,286	14.05	0	0
Add ion exchange in liquid stream to remove iodine (Case 2a)	35	0.023	965 (77)	2,414 (14)	1,598 (106)	1,237 (11,049)	12.72 (1.33)	0.45	26
Add caustic scrubber to remove semi-volatiles and carbon-14 (Case 5)	158	0.105	804 (238)	929 (1,489)	1,386 (318)	1,153 (11,133)	10.64 (3.41)	0.66	46
Add voloxidation to remove tritium and iodine (Case 6)	1,156	0.77	229 (813)	365 (2,063)	811 (893)	480 (11,806)	3.36 (10.69)	1.42	108.14
Add sand filter to remove particulates and semi-volatiles (Case 4)	1,459	0.97	105 (937)	99 (2,329)	181 (1,523)	357 (11,929)	1.26 (12.79)	1.56	114.07
Add selective absorption to remove krypton and carbon-14 (Case 3)	3,094	2.06	24 (1,018)	37 (2,391)	82 (1,622)	292 (11,994)	0.27 (13.68)	3.04	226.17
Add iodine evolution, mercury recycle, and DOG Hg(NO ₃) ₂ -HNO ₃ scrubber to remove iodine (Case 2b)	3,463	2.31	22 (1,020)	37 (2,391)	79 (1,625)	52 (12,234)	0.34 (13.71)	3.40	252.59
Replace Hg(NO ₃) ₂ -HNO ₃ scrubbers and mercury recycle with Iodex to remove iodine (Case 2c, same as Case 6c)	3,514	2.34	22 (1,020)	37 (2,391)	79 (1,625)	52 (12,234)	0.34 (13.71)	3.45	256.21
Add advanced Case 7	45,500	30.33	0.67 (1,041)	0.82 (2,427)	0.89 (1,703)	0.72 (12,285)	0.01 (14.04)	43.71	2,241

^aThe radwaste treatment cases (2a through 7) are added to the base plant (Case 1) in the best order of decreasing cost benefit, increasing ratio of annual cost/reduction in annual population total body dose out to a distance of 55 miles from the stack ground zero.

^bThe number in parentheses is the reduction in dose with reference to Case 1

^cThe maximum individual total body dose is the adult dose at a distance of 1.5 miles from the stack ground zero

Table 8 4 Cumulative annual cost increments, dose decrements and cost/benefits for additional radwaste treatments at the model LWR fuel reprocessing plant at the east-coastal site

Radwaste treatment	Cumulative annual cost increment ^a (\$10 ³)	Cumulative unit cost increment (\$/kg U)	Annual population and overall dose reduction ^b (person-rem)				Maximum individual ^c total body dose and overall dose reduction (mrem)	Cost/benefit ratio, total body	
			Total body	GI tract	Bone	Thyroid		$\left(\frac{\$10^3}{\text{annual population person-rem}}\right)$	$\left(\frac{\$10^3}{\text{maximum individual mrem}}\right)$
Case 1	0	0 00	299	787	515	4,245	10 89	0	0
Add ion exchange in liquid stream to remove iodine (Case 2a)	35	0 023	272 (27)	782 (5)	478 (37)	375 (3,870)	9 86 (1 03)	1 30	33 98
Add caustic scrubber to remove semi-volatiles and carbon-14 (Case 5)	158	0 105	227 (72)	275 (512)	417 (98)	351 (3,894)	8 25 (2 64)	2 19	59 85
Add voloxidation to remove tritium and iodine (Case 6)	1,156	0 77	71 (228)	120 (667)	262 (253)	160 (4,085)	2 60 (8 29)	5 07	139 45
Add sand filter to remove particulates and semi-volatiles (Case 4)	1,459	0 97	29 (270)	28 (759)	52 (463)	118 (4,127)	0 98 (9 91)	5 40	147 23
Add selective absorption to remove krypton and carbon-14 (Case 3)	3,094	2 06	8 (291)	12 (775)	27 (488)	101 (4,144)	0 29 (10 60)	10 63	291 87
Add iodine evolution, mercury recycle, and DOG Hg(NO ₃) ₂ -HNO ₃ scrubber to remove iodine (Case 2b)	3,463	2 31	7 (292)	12 (775)	26 (489)	18 (4,227)	0 26 (10 63)	11 86	325 78
Replace Hg(NO ₃) ₂ -HNO ₃ scrubbers and mercury recycle with Iodox to remove iodine (Case 2c, same as Case 6c)	3,514	2 34	7 (292)	12 (775)	26 (489)	18 (4,227)	0 26 (10 63)	12 03	330 57
Add advanced Case 7	45,500	30 33	0 19	0 24	0 26	0 21	0 01	152 17	4,182

^aThe radwaste treatment cases (2a through 7) are added to the base plant (Case 1) in the best order of decreasing cost benefit, increasing ratio of annual cost/reduction in annual population total-body dose out to a distance of 55 miles from the stack ground zero

^bThe number in parentheses is the reduction in dose with reference to Case 1

^cThe maximum individual total body dose is the adult dose at a distance of 1.5 miles from the stack ground zero

Fig. 4.1. Longitudinal cross section of a typical fuel reprocessing plant (ref. 1, Sect. 4.6).

Fig. 4.2. Cross section of a typical fuel reprocessing plant (ref. 1, Sect. 4.6).

Fig. 4.4. Model fuel reprocessing plant: Case 2a - reduction in release of iodine in liquid effluent using ion exchange.

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Fig. 4.6. Model fuel reprocessing plant: Case 2c - reduction in release of iodine in liquid and gaseous effluents using ion exchange, iodine evolution, and the Iodex process.

Fig. 4.7. Model fuel reprocessing plant: Case 3 – reduction in release of krypton and ^{14}C using selective absorption including a catalytic oxidizer.

Fig. 4.8. Model fuel reprocessing plant: Case 4 - reduction in release of particulates and semivolatiles in gaseous effluent using a sand filter.

Fig. 4.9. Model fuel reprocessing plant: Case 5 – reduction in release of semivolatiles and ¹⁴C in the gaseous effluent using a caustic scrubber.

Fig. 4.11. Model fuel reprocessing plant: Case 6c - reduction in all releases using a combination of Cases 2c through 6.

Fig. 4.12. Model fuel reprocessing plant: Case 7 - minimum release of radioactive materials.

Fig. 4.13. Model fuel reprocessing plant: Case 7 - minimum release of radioactive materials; flowsheet for control and treatment of radwaste effluents.

Fig. 4.14. Model fuel reprocessing plant: Case 7 - minimum release of radioactive materials; flowsheet for retention of krypton, ^{14}C , and iodine.

Fig. 4.15. Model fuel reprocessing plant: Case 7 - minimum release of radioactive materials; flowsheet for retention of tritium.

Fig. 7.1. Pathways for external and internal exposure of man.

Fig. 8.1. Cumulative annual cost for reduction of total-body dose from the model LWR fuel reprocessing plant at the midwestern site.

Fig. 8.2. Cumulative annual cost for reduction of total-body dose from the model LWR fuel reprocessing plant at the east-coastal site.

Fig. 8.3. Cumulative annual cost for reduction of thyroid dose from the model LWR fuel reprocessing plant at the midwestern site.

Fig. 8.4. Cumulative annual cost for reduction of thyroid dose from the model LWR fuel reprocessing plant at the east-coastal site.

APPENDIX A. PREPARATION OF COST ESTIMATES

B. C. Finney, W. G. Stockdale and R. E. Blanco

1.0 INTRODUCTION

This Appendix presents the details of the methods used to estimate the capital costs of the installations required for treating the radwastes at the model LWR fuel reprocessing plant. The details of the methods used for estimating the annual fixed charges, annual operating costs, total annual costs, and contribution to power costs are presented in Sect. 6.0 of the survey report. In summary, the total annual cost is obtained as the sum of the annual fixed charge (26% of the capital cost) and the annual operating cost (40% of the annual fixed charge).

1.1 Capital Cost

The capital cost of the radwaste treatment cases is the sum of the direct cost and the indirect cost. The methods used for estimating the direct and indirect costs are presented in the following sections.

1.1.1 *Direct costs*

The initial cost of the major equipment components are primarily based on vendor bids, using early-1973 dollars, obtained by Burns and Roe, Incorporated, under subcontract from Union Carbide Corporation, Nuclear Division.¹ Appropriate costs based on experience in the nuclear industry are then added to the initial cost to allow for (1) installation of the components, including piping, instrumentation, and controls; (2) modifications to provide for remote maintenance; and (3) fabrication upgrading (where necessary) to provide the required quality assurance. Where necessary, a "N Stamp Required" designation is included in the specifications for equipment. Applicable ASME codes are also required. In the equipment lists presented in the tables in this Appendix, "A" indicates a vendors price, "AB" indicates a vendor's price plus a Burns and Roe estimate, "B" indicates a Burns and Roe estimate, and "ORNL" indicates an ORNL estimate.

The costs prepared by Burns and Roe are based on conceptual flowsheets and process technology provided by ORNL and vendor bid prices for equipment. These costs were revised and the cost of additional equipment was estimated at ORNL to reflect subsequent changes in flowsheets. Equipment costs were transferred to the appropriate case study, and the necessary adjustments were made for piping, instrument, and installation costs. As described below, further adjustments were made in installation costs for equipment which is installed in remotely operated cells and for cell structure costs. Cell structure costs were increased substantially in Cases 2a, 2b, 2c, 3, 5, 6, and 7 based on background knowledge gained from construction

costs at AEC facilities. The costs for Case 4 are not affected by this revision, since the sand filter is housed in a concrete shell outside the plant building and, thus, does not require cell space. The cost of a warehouse and other related facilities are not included in the costs. The total direct cost for each radwaste treatment case is the complete, installed, equipment cost (material and labor), including the structure.

Cell space requirements for each case are determined by estimating the space required for major pieces of equipment and the requirements for auxiliary equipment, i.e., pumps, condensers, etc. This amount of space is then considered to be an addition to the length of cell that is required in the base case. The specifications assumed for the cell in the base case are: (1) cells are 25 ft wide, (2) remote maintenance cells are 60 ft high, (3) contact maintenance cells are 30 ft high, and (4) cells are lined with stainless steel. Operating, maintenance, and crane bay areas (contact maintenance cells) that are adjacent to external cell walls are lengthened in proportion to the increase in length of the cell. The length of remote and direct crane rails is also increased proportionally to service the additional equipment. For both remote and contact maintenance cells, additional pipe sleeves are cast into the concrete walls. The pipe sleeves in the remote maintenance cells are positioned accurately to ensure that piping and instrument jumpers from cell walls to the equipment can be placed by remote cranes and operated through the use of television cameras or through shielded windows. The cost estimates for the cells are based on Class I construction. Based on these considerations, cell costs of \$35 and \$25 per cubic foot of internal cell volume are estimated for remote and contact maintenance cells, respectively.

It is assumed that the equipment is installed in the remote cells by the remote crane to ensure that the equipment could be removed after operations are initiated. Thus, all piping, instrument lines, etc., must be prefabricated to exacting tolerances. Equipment and piping are installed in contact maintenance cells by labor operating within the cell. Some increase in installation and piping costs are made to cover these types of installation. Instrument and control units are those provided by Burns and Roe with appropriate changes as the result of flowsheet revisions.

All costs are based on new construction costs in which all of the equipment that is added for each case study is included in an integrated plant. Backfitting costs for existing plants are not considered. Backfitting costs for the installation of equipment for retaining krypton and tritium have been prepared by Nuclear Fuel Services² and Gulf General Nuclear Services.³

A summary of the capital costs (direct and indirect) for radwastes Cases 2a through 6, and cell space requirements and costs for radwaste treatment Cases 2a, 2b, 2c, 3, 5, and 6 are presented in Tables A-1 and A-2, respectively. The information presented is grouped according to radwaste treatment case and consists of detailed equipment lists and equipment costs (Tables A-1 through A-8) and equipment flowsheets (Figs. A-2 through A-8).

1.1.2 *Indirect costs*

Indirect costs are estimated as follows:

	Percentage of direct cost
Engineering and supervision	15
Construction expense and contractor's fee	20
Engineering design (A-E)	19
Quality assurance ^a	6
Other owner's cost	10
Contingency	40
Interest during construction ^b	39
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The interest during construction and the contingency allowance are included as indirect costs to simplify the calculations.

1.2 References for Appendix A

1. *Engineering Evaluation and Cost Estimate of Radwaste Treatment Facilities for Nuclear Fuel Reprocessing Plants* for Nuclear Division, Union Carbide Corporation, Subcontract No. 3894, Dec. 14, 1973 (unpublished).
2. E. D. North and R. L. Booth, *Fission Product Gas Retention Study Final Report*, ORNL/TM-4409 (August 1973).
3. E. W. Murbach, W. H. Carr, and J. H. Gray, III, *Fission Product Gas Retention Process and Equipment Design Study*, ORNL/TM-4560 (May 1974).

^aQuality assurance costs based on the vendor's equipment bid prices obtained by Burns and Roe, Incorporated, do not reflect quality assurance costs anticipated for a nuclear fuel processing plant being built in the near future. An additional cost of 6% of the direct costs has been assumed to reflect costs for more stringent quality assurance.

^bInterest is applied to the cumulative total cost at a rate of 8% per year over a 5-year cash flow expenditure period.

Table A-1. Summary of capital cost for model fuel reprocessing plant
radwaste treatment Cases 2a-6

	Case 2a	Case 2b ^a	Case 2c ^{a,b}	Case 3	Case 4	Case 5	Case 6
Equipment purchase	\$ 6,700	\$ 88,500	\$ 146,900	\$ 907,780	\$ 20,200	\$ 46,840	\$ 425,000
Installation	3,300	29,200	32,200	66,360	1,100	5,200	8,900
Piping	-	57,700	76,800	188,400	20,800	26,460	32,000
Instrument and controls	-	25,600	59,000	63,400	8,000	15,900	5,600
Cell and adjacent building	27,000	243,000	352,500	577,500	-	41,300	630,000
External structure					-	1,000	-
Sand filter					375,300	-	-
Direct construction cost	\$37,000	\$ 440,000	\$ 667,900	\$1,803,440	\$ 425,400	\$136,700	\$1,102,000
Indirect construction cost, ~149% ^c	55,000	662,000	994,000	2,687,125	633,800	203,700	1,642,000
Total construction cost	\$92,000	\$1,102,000	\$1,661,000	\$4,491,000	\$1,059,000	\$340,000	\$2,744,000

^aCases 2b and 2c include Case 2a.

^b Does not include a capital cost credit of \$414,000 for the Case 1 Hg(NO₃)₂-HNO₃ scrubber.

^c Indirect cost	Direct cost (%)
Engineering and supervision	15
Construction expense and contractor's fee	20
Engineering design (A-E)	19
Quality assurance	6
Other owner's cost	10
Contingency	40
Interest during construction	39
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Table A-2. Summary of cell space requirements and costs for model fuel reprocessing plant
radwaste treatment Cases 2a-6

	Case 2a	Case 2b	Case 2c	Case 3	Case 4	Case 5	Case 6
Remote maintenance cell, ft ²	-	25	25	-	-	-	250
Contact maintenance cell, ft ²	36	254	400	700	-	55	140
<u>Allowance for cost of cell structure and adjacent areas</u>							
Remote maintenance cell	-	\$ 52,500	\$ 52,500	-	-	-	\$525,000
Contact maintenance cell	<u>\$27,000</u>	<u>190,500</u>	<u>300,000</u>	<u>\$577,500</u>	<u>-</u>	<u>\$41,300</u>	<u>105,000</u>
Total cell and adjacent building	\$27,000	\$243,000	\$352,500	\$577,500	-	\$41,300	\$630,000

Remote maintenance cells are 25 ft wide, 60 ft high, and are costed at \$35/ft².

Direct maintenance cells are 25 ft wide, 30 ft high, and are costed at \$25/ft².

All cells are lined with stainless steel.

The dashes indicate that cell space is not required; the sand filter is housed in an external concrete shell.

Table A-3. Equipment list and cost for radwaste treatment Case 2b -
reduction in release of iodine - ion exchange, iodine evolution,
mercury recycle, and DOG $\text{Hg}(\text{NO}_3)_2$ - HNO_3 scrubber

Item No.	Descriptive title	Number of items	Unit cost (\$)	Total cost (\$)	Reference and/or source
<u>Remote in-cell equipment</u>					
T-2A-1	Stripper, S.S.	1	13,420	13,420	AB
E-2A-1	Stripper reboiler, S.S.	1	5,000	5,000	A
E-2A-2	Stripper condenser, S.S.	1	4,700	4,700	A
E-2A-3	Feed preheater, S.S.	1	1,000	1,000	B
P-2A-3	Diodized solution pump	1	4,740	4,740	A
Subtotal				28,860	
<u>Contact in-cell equipment</u>					
	Ion exchange column	1	6,700	6,700	ORNL
	Scrubber	1	27,200	27,200	ORNL
T-2B-4	Fractionator	1	5,580	5,580	AB
E-2B-4	Fractionator condenser	1	590	590	A
E-2B-5	Flash vaporizer reboiler	1	700	700	A
T-2B-6	Decanters	2	990	1,980	A
A&B					
D-2B-1	Filtrate receiver	1	1,790	1,790	A
D-2B-2	Surge tank	1	1,790	1,790	A
P-2B-1	Filtrate pump	1	2,330	2,330	A
P-2B-2	Fractionator reboiler circ. pump	1	4,740	4,740	A
P-2B-3	Solution recycle pump	1	2,020	2,020	A
F-2B-1	Filter and S.S. support	1	650	650	AB
C-2B-1	Ejector	1	230	230	A
Subtotal				56,300	
<u>Chemical preparation area</u>					
V-2A-1	KI solution tank	1	700	700	A
P-2A-1	KI solution pump	1	350	350	A
P-2A-2	H_2O_2 pump	1	460	460	A
M-2A-1	Mixer	1	320	320	A
T-2B-5	Solution make-up tank	1	720	720	A
M-2B-5	Make-up tank mixer	1	320	320	A
P-2B-4	Make-up solution pump	1	430	430	A
Subtotal				3,300	
Grand total				88,460	

Table A-4. Equipment list and cost for radwaste treatment Case 2c -
reduction in release of iodine - ion exchange, iodine
evolution, and Iodex

Item No.	Descriptive title	Number of items	Unit cost (\$)	Total cost (\$)	Reference and/or source
<u>Remote in-cell equipment</u>					
T-2A-1	Stripper, S.S.	1	13,420	13,420	AB
E-2A-1	Stripper reboiler, S.S.	1	5,000	5,000	A
E-2A-2	Stripper condenser, S.S.	1	4,700	4,700	A
E-2A-3	Feed preheater, S.S.	1	1,000	1,000	B
P-2A-3	Deiodized solution pump	1	4,740	4,740	A
Subtotal				28,860	
<u>Contact in-cell equipment</u>					
	Ion exchange column	1	3,600	3,600	ORNL
T-2B-1	Absorber, Ti-lined	1	82,200	82,200	AB, ORNL
T-2B-2	Evaporator #1, Ti	1	22,800	22,800	A, ORNL
E-2B-1	Evaporator #2, heater, S.S.	1	3,100	3,100	A, ORNL
E-2B-2	Evaporator condenser, Ti and S.S.	1	1,485	1,485	A, ORNL
Subtotal				113,185	
<u>Chemical preparation area</u>					
V-2A-1	KI solution tank	1	700	700	A
P-2A-1	KI solution pump	1	350	350	A
P-2A-2	H ₂ O ₂ pump	1	460	460	A
M-2A-1	Mixer	1	320	320	A
Subtotal				1,830	
Grand total				143,875	

Table A-5. Equipment list and cost for radwaste treatment Case 3 - reduction in release of krypton and ^{14}C - selective absorption

Item No.	Descriptive title	Number of items	Unit cost (\$)	Total cost (\$)	Reference and/or source
<u>Contact in-cell equipment</u>					
T-3-1	Absorber and packaging	1	28,590	28,590	AB
T-3-2	Fractionator and packing	1	15,300	15,300	AB
T-3-3	Stripper and packing	1	27,100	27,100	AB
D-3-1	Flash drum	1	4,790	4,790	B
D-3-4	Solvent storage drum	1	13,470	13,470	B
X-3-1	Gas drier	1	4,790	4,790	AB
X-3-2	Solvent recovery absorbers	3	55,200	165,600	AB
A,B, C					
E-3-1	Stripper secondary condenser	1	1,090	1,090	A
E-3-3	Fractionator condenser	1	3,000	3,000	A
E-3-4	Fractionator reboiler	1	2,750	2,750	B
E-3-5	Stripper condenser	1	8,200	8,200	B
E-3-6	Stripper reboiler	1	6,250	6,250	B
E-3-8	Gas chiller	2	3,500	7,000	A
E-3-9	Final cooler	1	3,500	3,500	A
E-3-10	Solvent recovery condenser	1	7,520	7,520	A
E-3-11	Regeneration heaters	4	3,500	14,000	A
A,B,C,D					
E-3-12	Desupersaturater	1	1,820	1,820	A
C-3-1	1st stage compressor package	2	39,040	78,080	A
C-3-2	2nd stage compressor package	2	32,000	64,000	A
P-3-3	Recycle solvent pump	2	3,720	7,440	A
P-3-4	Solvent make-up pump	1	4,330	4,330	A
P-3-5	Desuperheater pump	1	350	350	AB
	Catalytic oxidizer system	1	93,600	93,600	ORNL
	Bottling station	1	36,600	36,600	ORNL
	Subtotal			599,170	
<u>Chemical preparation area</u>					
D-3-2	Freon-12 accumulator	1	16,400	16,400	B
D-3-3	Freon-22 accumulator	1	11,790	11,790	B
P-3-1	Freon-12 circ. pump	2	2,550	5,100	A
P-3-2	Freon-22 circ. pump	2	2,660	5,320	A
R-3-1	Freon-12 refrigeration pkg	2			A
R-3-2	Freon-12 refrigeration pkg	1	135,000	270,000	A
	Subtotal			308,610	
	Grand total			907,780	

Table A-6. Equipment and material list and cost for radwaste treatment Case 4 -
reduction in release of particulates - sand filter

Item No.	Descriptive title	Number of items	Unit cost (\$)	Total cost (\$)	Reference and/or source
B-4-1	Blower	1	20,200	20,200	A
	<u>Sand filter</u>				
F-4-1	Filter structure (I&M)	1	303,180	303,180	B
	Gravel, 2" x 1"	500 T	13.60 + freight	11,915	A
	Gravel, 1" x 1/2"	500 T	10.60 + freight	10,415	A
	Gravel, 1/2" x 1/4"	500 T	8.60 + freight	9,415	A
	Filter sand, 4x8 mesh	750 T	8.10 + freight	13,748	A
	Filter sand, 8x20 mesh	500 T	7.85 + freight	9,040	A
	Filter sand, 20x40 mesh	1000 T	7.35 + freight	<u>17,580</u>	A
	Grand total (sand filter)			375,300	

Bulk source of supply - sand and gravel: Cape May, New Jersey.

Freight is assumed at \$10.23 per ton. Bulk freight an average distance from the source of supply and Aiken, S. C., and Joliet, Ill.

Table A-7. Equipment list and cost for radwaste treatment Case 5 -
reduction in release of semi-volatiles and carbon-14 - caustic scrubber

Item No.	Descriptive title	Number of items	Unit cost (\$)	Total cost (\$)	Reference and/or source
<u>Contact In-Cell Equipment</u>					
T-5-4	Absorber	1	22,900	22,900	A
P-5-7	Spent caustic pump	1	660	<u>660</u>	A
	Subtotal			23,560	
<u>Chemical Storage Area</u>					
V-5-1	Caustic storage tank	1	19,040	19,040	A
P-5-6	Caustic transfer pump	1	830	830	A
P-5-8	Caustic makeup pump	1	1,520	1,520	AB
B-5-1	Booster blower	1	1,890	<u>1,890</u>	A
	Subtotal			<u>23,280</u>	
	Grand Total			46,840	

Table A-8. Equipment list and cost for radwaste treatment Case 6 -
reduction in release of tritium and iodine - voloxidation

Item No.	Descriptive title	Number of items	Unit cost (\$)	Total cost (\$)	Reference and/or source
<u>Remote In-Cell Equipment</u>					
K-6-1	Voloxidizer	1	281,500	281,500	A
F-6-1	Filters	5	2,000	<u>10,000</u>	A
	Subtotal			291,500	
			(Adjust for balance alignment of nozzles, etc.)		
<u>Contact In-Cell Equipment</u>					
X-6-1	Catalytic recombiner	1	75,000	75,000	A
E-6-1	Air cooler	2	3,000	6,000	B
X-6-2	Drier package	1	50,000	50,000	A
E-6-2	Recombiner preheater	1	3,000	<u>3,000</u>	
	Subtotal			<u>134,000</u>	
	Grand total			425,500

Fig. A-1. Equipment flowsheet No. 1 for model nuclear fuel reprocessing plant radwaste treatment Cases 2b and 2c - reduction in release of iodine: iodine evolution.

Fig. A-2. Equipment flowsheet No. 2 for model nuclear fuel reprocessing plant radwaste treatment Case 2b - reduction in release of iodine: mercury recycle.

Fig. A-3. Equipment flowsheet for model nuclear fuel reprocessing plant radwaste treatment Case 2c: reduction in release of iodine - Iodex.

Fig. A-4. Equipment flowsheet No. 1 for model nuclear fuel reprocessing plant radwaste treatment Case 3 - reduction in release of krypton and ¹⁴C: selective absorption.

Fig. A-5. Equipment flowsheet No. 2 for model nuclear fuel reprocessing plant radwaste treatment Case 3 - reduction in release of krypton and ^{14}C : selective absorption.

Fig. A-6. Equipment flowsheet for model nuclear fuel reprocessing plant radwaste treatment Case 4: reduction in release of particulates and semivolatiles - sand filter.

Fig. A-7. Equipment flowsheet for model nuclear fuel reprocessing plant radwaste treatment Case 5: reduction in release of semivolatiles and ^{14}C - caustic scrubber.

Fig. A-8. Equipment flowsheet for nuclear fuel reprocessing plant radwaste treatment
Case 6: reduction in the release of tritium and iodine - voloxidation.

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