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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

B. C. Finney
R. E. Blanco
R. C. Dahlman*
F. G. Kitts
J. P. Witherspoon*

Project Director: R. E. Blanco

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*Environmental Sciences Division

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
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PUBLICATION NOTICE

This engineering survey report was prepared for the Nuclear Regulatory Commission (NRC) - Office of Standards Development (formerly the Regulatory Office of the Atomic Energy Commission). It is one of a series of draft reports on segments of the nuclear fuel cycle that were prepared in 1973 and 1974 and were made available to the public in December 1974. These draft reports are subject to revision prior to, and subsequent to, their publication by the NRC in conjunction with draft environmental statements for comment by the public and government agencies.

It is expected that revisions to the report on fuel reprocessing will include (1) the addition of a new subcase 1, (2) the addition of ^{14}C to the source term, (3) the deletion of the mercuric nitrate scrubber from Case 2b, and (4) the recalculation of the doses from ^{85}Kr to allow for the inhomogeneous closed effect at distances out to about 1.5 miles (for example, the total body dose increases by a factor of 1.36 at 0.5 mile at the midwestern plant).

The reports in this series are:

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).

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).

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).

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).

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).

L. R. McKay (Ed.), A Methodology for Calculating Radiation Doses from Radioactivity Released to the Environment, ORNL-4992 (1975). (This report serves as Appendix B for all of the above reports.)

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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 practicable" 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.

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

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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 (dose commitment) of the released materials on the environment. The model plant is representative of current plants which 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/ton at 30 kW/kg and cooled 160 days before reprocessing. The gaseous radwaste effluents are treated and released. All of the radioactive krypton-85 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 liquid radioactive 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 by about 1983 to plants which are constructed using the present design concepts for handling process gases and liquids. This time scale is predicated on the assumption that the advanced processes are developed and demonstrated successfully over the ensuing years (Sect. 4.3). 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 two 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. 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. 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 uranium), the contribution to power costs, and the radiological impact (dose commitment) to the environment are estimated for each case. The dose commitments for each case were estimated for (1) the maximum annual individual total body, thyroid, and bone doses (millirem) at 0.5 mile from the plant (factors are provided to project the maximum dose to greater distances); (2) the incremental maximum annual individual dose (millirem) at 0.5 mile, which represents the difference in dose between Case 1 and a given case; (3) the average individual total body dose (millirem) out to a distance of 55 miles; and (4) the average annual population total body dose (man-rem) out to a distance of 55 miles. The costs and doses are summarized in Tables 8.1, 8.2, 6.1, and 7.4. 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, in effect, the incremental cost. The difference in dose commitment at the two sites is the result of differences in meteorology and population distribution at the two sites. Internal exposure to radiation through inhalation and ingestion of radionuclides accounts for about 62% of the total body dose to individuals and the population living around a nuclear fuel reprocessing plant.

Total Body Dose. — The principal radionuclides that contribute to total body dose are tritium (^{3}H), cesium (^{134}Cs, ^{137}Cs), and krypton (^{85}Kr). The contributions to total body dose in the Case 1 study are ^{3}H (43%), $^{134-137}$Cs (21%), and ^{85}Kr (17%). The estimated maximum annual individual total body doses do not exceed 8.0 millirem at 0.5 mile from the plant or 3.0 millirem at a distance of 1.5 miles. The average annual individual total body doses do not exceed 2.7 millirem for individuals living within one mile of the plant. Only small reductions in maximum total body dose at 0.5 mile (1.3 to 6.8%) are achieved by decreasing the release of ^{129,131}I and ^{103,106}Ru in Cases 2a, 2b, and 5, since these radionuclides contribute only a small fraction of the total body dose. Dose reductions of 18 to 43% are achieved in Cases 3, 4, and 6 where the amounts of ^{85}Kr, particulates, and ^{3}H released are reduced by factors of 100, 10, and 100, respectively. These values are reflected in the comparison of the ratios of annual costs for radwaste treatment to the incremental reductions in maximum total body dose at 0.5 mile (cost-benefit ratios), which are listed in Table 8.3 and shown as bar graphs in Figs. 8.3 and 8.4. The cost-benefit ratios vary from \$0.15 million/millirem for Case 4 to \$5.76 million/millirem for Case 7 at the midwestern site and from \$0.28 million/millirem for Case 4 to \$10.3 million/millirem for Case 7 at the coastal site. The most efficient system in terms of cost-benefit ratio (\$$10^{6}$/millirem), i.e., the lowest cost-benefit ratio, is the filter system in Case 4. The cost-benefit ratios increase in the following order: Case 4 (particulate retention), Case 5 (semivolatile retention), Case 6 (tritium retention), Case 6c (cumulative case), Case 3 (krypton retention), Case 2a and 2b (iodine retention), and Case 7 (improved retention of all radionuclides).

The cumulative effect of adding the radwaste systems in the case studies to the base case is illustrated in Table 8.3 and Figs. 8.5 and 8.6 for the midwestern and coastal sites, respectively. The treatment systems are added in the order of increasing cost-benefit ratio (i.e., decreasing efficiency), and the cumulative annual maximum individual total body dose at 0.5 mile is compared with both the cumulative total annual cost and the cumulative reprocessing cost ($/kg uranium) for radwaste treatment. The greatest decrease in dose with increasing expenditure of money occurs by adding Cases 4, 5, and 6 to the base case, resulting in a reduction of total body dose from 4.4 millirem to 1.1 millirem (75% of total) at the coastal site and from 7.9 millirem to 2.0 millirem (75% of total) at the midwestern site for a total increase in annual cost of $1.424 million. The addition of Cases 3, 2a, and 2b results in a further reduction of total body dose from 1.1 millirem to 0.19 millirem (~21% of total) at the coastal site and 2.0 millirem to 0.33 millirem (~21% of total) at the midwestern site at an additional increase in reprocessing cost of $2.247 million. The change in slope of the curve is illustrated more graphically in Fig. 8.7, where the dose vs the cumulative annual cost is presented on rectangular coordinates for the coastal site. In this comparison, the cases are cumulated on three different bases, i.e., in the order of (1) increasing cost-benefit ratio (annual cost/incremental reduction in dose), (2) decreasing incremental reduction in dose, or (3) increasing cost. The cost-benefit plot, of course, represents the most efficient use of money in reducing dose since the cases are selected on that basis.

Thyroid Dose. — The annual cost of reducing the maximum annual individual adult thyroid dose at a distance of 0.5 mile is shown in Tables 8.1 and 8.2 and Fig. 8.8 for Cases 1, 2a, 2b, 6c, and 7. These cases are designed to illustrate the effect of decreasing the release of radioactive iodine which contributes more than 95% of the dose to the thyroid. For Cases 2a and 2b, all of the treatment costs can be assessed to reduction in thyroid dose since these cases are specifically designed to illustrate iodine retention. However, the total costs for Case 6c and Case 7 should not be assessed to reduction in iodine releases, as these cases are designed for the retention of all types of radioactive materials. No effort is made to define the fractions of the total cost of Cases 6c and 7 that apply exclusively to retention of iodine or reduction in thyroid dose.

The maximum annual thyroid dose decreases from 103.2 millirem for Case 1 to 16.4 millirem for Case 2a (annual cost, $274,000) to 8.0 millirem for Case 2b (annual cost, $753,000), to 1.6 millirem for Case 6c (annual cost, $3,671,000), to 0.006 millirem for Case 7 (annual cost, $45,500,000) at the midwestern site, and to dose values about 44% lower at the coastal site. These ratios in annual dollars per millirem are $5,592 for Case 2a, $13,944 for Case 2b, $63,955 for Case 6c, and $780,486 for Case 7 at the coastal site. The same pattern is obtained for these values at the midwestern site.

Bone Dose. — The annual cost of reducing the maximum annual individual adult bone dose at a distance of 0.5 mile is shown in Tables 8.1 and 8.2 and Fig. 8.9 for Cases 1, 4, 6c, and 7. These cases are specially designed to illustrate the effect of decreasing the release of particulates (which includes the transuranium nuclides) on bone dose. No effort is made in Cases 6c and 7 to define the fractions of the total cost that apply exclusively to the retention of radioactive materials that concentrate in the bones. The maximum annual dose decreases from 20.5 millirem at the midwestern site for Case 1 to 6.4 millirem for Case 4 (annual cost of $303,000) to 2.4 millirem for Case 6c (annual cost of $3,671,000), and to 0.011 millirem for Case 7 (annual cost of $45,500,000). The dose values are about 43% lower for the coastal site. The most efficient system as measured in terms of ratio of annual cost to incremental reduction in bone dose is Case 4 at the coastal site. These ratios in annual dollars per millirem are $39,351 for Case 4, $356,408 for Case 6c, and $3,924,443 for Case 7. The same pattern is obtained for these values at the midwestern site.

Population Dose. — The annual cost of reducing the average total body dose (man-rem) for the general population out to a distance of 55 miles is presented in Tables 8.1 and 8.2 and Figs. 8.10 and 8.11. Reductions in population dose are small for Cases 2a, 2b, and 5, i.e., up to a 3.3% reduction in the 485.5 man-rem and 173.1 man-rem doses listed for Case 1 at the midwestern and coastal sites, respectively. Larger reductions are obtained in the other cases, i.e., ~15% for Case 3, ~22% for Case 4, ~54% for Case 6, ~97% for Case 6c, and about 1500-fold for Case 7. Case 6 (tritium removal) is the most effective at the midwestern site on a cost basis for reducing the population total body dose, i.e., $3,787/man-rem, whereas Case 4 (particulate removal) is the most effective at the coastal site, i.e., $10,100/man-rem (Figs. 8.10 and 8.11).

Cost Comparisons. — 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/kg of contained U+Pu for PWR fuel and $60.37/kg U+Pu for BWR fuel (Sect. 6.4, Table 6.4). The annual reprocessing cost in the base Case 1 is $30.33/kg U (Table 6.1). In this case, the difference between the value of the spent fuel and the reprocessing cost is about $45/kg U+Pu for PWR fuel and $30/kg U+Pu for BWR fuel. In Cases 2 through 6, the costs for additional radwaste treatment systems increase from about 0.2 to 2% of the incremental difference between the reprocessing cost and the value of the PWR fuel and by 0.3 to 3% 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/kg U, as illustrated for Case 7, would approach the incremental difference between the value of the fuels and the reprocessing cost under current economic conditions.

The capital costs for Cases 2a through 6 range from $0.340 million to $3.871 million, or up to about 3% 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 $0.123 million to $1.409 million, which is equivalent to contributions to power cost of 3.19×10^{-4} to 3.66×10^{-3} mill/kWhr, respectively (Sect. 6.0, Table 6.1). Cumulative Cases 6c and Case 7 amount to 9.54×10^{-3} and 1.2×10^{-1} mill/kWhr, respectively. All of these values are less than about 3% 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 is 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 or 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 reenrichment. 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 (1973), 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 which 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, i.e., 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 on the basis of 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 nonexistent. It is expected that the advanced technology used in the study could be "reduced to practice" by about 1983 if engineering development is initiated in 1974. The bases for this estimate are presented in detail in Sect. 4.3.

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 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. Thus, 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 two decades.[2-4] The plant will process 1500 metric tons of fuel per year (calculated as 1500 tons of uranium charged to the reactor) irradiated at an average specific power of 30 MW/metric ton 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 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 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 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. Liquid radioactive 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, tritiated water vapor) and particulates or aerosols that contain a spectrum of all of the radionuclides in the process equipment. The gases are treated such 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 such 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 evaporated 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-activity 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 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. The capital cost of the base plant, not including the options of solidification of the high-level radioactive waste, the formation of a solid plutonium product (PuO_2), and the conversion of uranium product to UF_6, was set at $125 million in 1973. 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, i.e., wastes will not bypass treatment systems and 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 liquid radioactive 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 which are characteristic of contemporary operations at 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 and biota surrounding the model plant are estimated using the procedures that have been standardized for environmental impact statements for light-water-cooled nuclear power reactor stations by the USAEC-Regulatory.[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 harvest of fish or drinking from the water can lead to internal exposure. In this study, no radioactive materials are released in liquid effluents. However, the effects of contamination of the water from gaseous radioiodines, tritiated water vapor, and the fallout of radioactive particulates are considered.

The estimated radiation doses to individuals, to the human population, and to the biota 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 (man-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

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 ton of contained fuel. Each year, typical large power reactors discharge from 25 to 40 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/ton and the model plant will process 1500 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-ft- to 5-1/2-ft-thick-walled, heavily reinforced concrete central 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 may release small amounts of radioactive materials to the water or to the room ventilation system.

Fuel Shearing and Dissolution. — The first step in fuel reprocessing is to shear the fuel 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 that arise 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, and tritiated water vapor are released to the vessel off-gas system. These gases also contain an aerosol of droplets of highly radioactive dissolver solution which contain 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. H_2O 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 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 is 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 requirement[a] 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 would be scrubbed by the raffinate waste from

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[a] Code of Federal Regulations, Title 10, Part 50, Appendix F.

the extraction system. The non-condensable 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 ORIGIN[3] computer code, which calculates the relative inhalation hazard for each nuclide by dividing the curies present in one metric ton of fuel at 160 days cooling by the Radiation Concentration Guide[a] for that nuclide. The criteria for selection of the nuclides to be used are as follows:

1. Gaseous nuclides tritium, krypton, ^{129}I, and ^{131}I; actinide nuclides whose contribution to the actinide relative inhalation hazard is $\geq$0.02% of the total (1.01×10^{17} m^3 of air at RCG); and fission products whose contribution to the relative inhalation hazard is $\geq$0.02% of the total (1.04×10^{16} 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 ORIGIN computer code[3] 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).

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[a] Code of Federal Regulations, Title 10, Part 20, Appendix B, Table 2, Column 1.

4.3 Description of Waste Treatment Methods

Nuclear fuel reprocessing plant effluents are extensively treated 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 which are in the small engineering scale of development. The advanced radwaste treatment systems used in this study include the iodine evolution and the Iodox[a] processes for retaining iodine, the voloxidation process for retaining tritium, and the selective absorption process for retaining krypton. Currently, these processes are not being developed for Light Water Reactor (LWR) fuels on an experimental engineering basis. However, they are being developed for Liquid Metal Fast Breeder Reactor fuels at ORNL. It is expected that engineering development will be completed within five years and that the processes will be applicable to LWR fuels. Following engineering development, the processes will be demonstrated in a pilot plant with radioactive fuels. Construction and demonstration will require 4 to 5 years. The equipment could also be installed in a reprocessing plant, instead of the pilot plant, and the processes demonstrated "in place" with radioactive LWR fuel. Also, the processes could be developed in non-AEC installations for LWR fuels on the same time schedule. On these bases, the processes can be expected to be "reduced to practice" for LWR fuels by about 1983.

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[a] Formerly called Iodex.

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-μm particles and a maximum resistance (when clean) of 1.0 in. H_2O 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, however, does not indicate a gross variation in the range of reported particle sizes in field operations.[4]

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, large-scale filter systems were found which produced overall mass removal efficiencies of 10^7 or greater.[5] 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 the most recent design practice for HEPA installations. Ettinger *et al.* have performed laboratory tests using plutonium aerosols in small installations that are tightly sealed and tested periodically for leaks with DOP.[6,7] They have observed removal efficiencies of at least 99.97% for each of three single filter stages in series. AEC Regulatory Guide 3.12 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.[8] Consequently, a value of 99.95% has been used in this study to represent the <u>rated</u> 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 to 99.99%. 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.[6] 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.[9] For these reasons, Burchsted 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.[9] 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, i.e., 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. This approach is consistent with the basic objective of the study 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 through[9] (1) attack by corrosive chemicals, such as fluorides; (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; 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 filters[10] are constructed of graded layers of aggregate and sand. The aggregate is in layers of relatively coarse (2-1/2 in. x 1-1/2 in.) to fine particles (#4 - #16 mesh). The sand layer is generally about 30 in. thick and made up of #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. 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.

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 13 years before being shut down because of water inleakage. Based on actual measurements at both installations, the collection efficiencies of the filters are >99%.[10] 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.[5]

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.[11] 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.[11] 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

Several metal exchange zeolites have been investigated on a laboratory scale for possible use in removing iodine from gaseous effluents.[12,13] The adsorbent is prepared by a partial replacement of sodium ions with silver (Ag) ions in the zeolite. 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. The adsorption efficiency of the silver zeolites for both elemental iodine and methyl iodide is >99.9% (DF = 10^3 - 10^4) with fresh adsorbent and when the adsorbent is maintained at about 200°C. However, in long-term (11 months) tests of silver zeolite beds in small-scale studies, the DF for removal of iodine species decreased more rapidly for the longer-lived species than for the shorter-lived species.[14] No rigorous explanation of this phenomenon has been proven.

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. 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.[15] 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.[16,17]

A conservatively low, average, removal efficiency of 99% (DF = 100) for ^{131}I 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.[1,18] The silver zeolite bed would be replaced when the average DF for ^{131}I reaches 100. The available data[14] indicate that the average DF for long-lived ^{129}I will be lower than for ^{131}I, and a DF of 10 is used in this study.

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 N_2O_3 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 H_2O_2, and (3) sparging with N_2O_3. Greater than 99% of the iodine can be evolved from a nitric acid solution by this method[19,20] and a removal efficiency of 99.5% is used in this study.

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 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 recovered 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 nitric acid solutions. 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. It is estimated that the process could be "reduced to practice" by about 1983 (Sect. 4.3).

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 Cases 2a and 2b, Sects. 4.5.2 and 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 evaporation of the waste solution and will complicate the problem of retaining iodine in the plant (Sects. 4.5.1, 4.5.2, and 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 is 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.[21,22]

The mercuric nitrate--nitric acid scrubbing process has been demonstrated in plant installations using less-concentrated solutions of nitric acid and mercury.[23] Similar scrubbing systems are being installed on the dissolver off-gas and vessel off-gas streams at the Barnwell Nuclear Fuel Processing 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.[1,18] In the Case 2a and Case 2b studies, 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, 4.5.1, 4.5.2, and 4.5.3). In these cases, a DF of 75 for iodine is used for the mercuric nitrate--nitric acid scrubbing system.

4.3.6 Iodox[a]

The Iodox process[24,25] is an advanced method for removing elemental iodine and organic iodides, such as methyl iodide, from gaseous effluents.

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[a] Formerly called Iodex.

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, iodine pentoxide, 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 19-21 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 $I_2O_5 \cdot 1/3\ H_2O$. 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 Iodox process has been successfully demonstrated 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°C, or (2) a 4-cm-diam by 117-cm-high packed column with boiling 70% HNO_3 or 80% HNO_3 at >60°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 Iodox 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 Iodox systems. Very little fresh acid would be purchased for this type of plant.

The principal advantages of the Iodox 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 requirement for internal plant production of concentrated nitric acid in plants where the high-activity wastes are solidified and nitric acid is recycled.

The Iodox 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. It is estimated that the process could be "reduced to practice" by about 1983 (Sect. 4.3).

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.[26] 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, efficient removal 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 drier 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 UO_2-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 successfully demonstrated 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. It is estimated that the process could be "reduced to practice" by about 1983 (Sect. 4.3).

4.3.8 Selective Absorption

The selective absorption process, which is a method for removing the noble gases, krypton and xenon, from gaseous effluents, has progressed to the nonradioactive pilot plant stage of development.[27] 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. 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. Over the next two years, auxiliary systems consisting of feed gas preparation, solvent recovery, solvent purification, and product purification will be developed and the effect of impurities will be investigated further.[27]

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.[28] 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. It is estimated that the process could be "reduced to practice" by about 1983 (Sect. 4.3).

4.3.9 Cryogenic Distillation

The cryogenic process is designed to remove noble gases (krypton and xenon) from gaseous effluents.[29] 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.[30,31] 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. 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.[32] However, this system has not been tested on the off-gases from a reprocessing plant which 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.[33,34] 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 about 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 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. 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 are required.

It is estimated that the process could be "reduced to practice" by about 1983 (Sect. 4.3).

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 (Ru), from gaseous effluents.[35,36] 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.[35] 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.[37] However, 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 unknown. The low concentrations of iodine in the gas phase, that is, 1×10^{-3} μg 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 showed a removal efficiency of 99%, with an inlet iodine concentration of 15 $\mu g/ft^3$. Other factors are the unknown amounts of particulates and organic iodides present. No reports which relate the efficiencies or kinetics of caustic scrubbing to irradiation rate, concentration of iodine in both phases, concentration of caustic, temperature, and contacting efficiency have been found in the literature. 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.[38]

4.3.11 Adsorption 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 1×10^{-6} *M* I_2 by passing it through a 4-in.-deep bed of coconut charcoal.[39] Charcoal is not effective in removing iodine from acidic solutions.

Macroreticular resins absorb iodine efficiently from water or acidic solutions.[40,41] 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 5×10^{-4} *M* I_2.

4.4 Selection of Case Studies

Seven 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), i.e., the ratio of 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 10 for ^{129}I and ^{131}I; Case 2b, a DF of 13.3 for ^{129}I and a DF of 1.33 for ^{131}I; the total additional DF for Case 2 relative to Case 1 is 133 for ^{129}I and 13.3 for ^{131}I; Case 3, a DF of 100 for krypton; Case 4, a DF of 10 for particulates including plutonium and uranium; Case 5, a DF of 10 for semivolatile materials; Case 6, a DF of 100 for tritium; Case 6c, a cumulation of Cases 2 through 6. In Case 7, the additional DFs relative to Case 1 are 6.7×10^3 for ^{131}I and 6.7×10^4 for ^{129}I, 1×10^4 for krypton, 2×10^3 for particulates, 1×10^4 for semivolatiles, 5×10^3 for plutonium, 2×10^4 for uranium, and 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 cost 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 two decades to plants which are designed 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 in Table 4.2, and the flowsheets 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 USAEC facilities;[42] safety analysis reports for the three commercial reprocessing plants, NFS, MFRP, and BNFP;[43-45] environmental reports for MFRP and BNFP;[46,47] and a current USAEC development program.[48]

Iodine. — Iodine-131 was not detected in the liquid or gaseous effluents from NFS because it had decayed to nondetectably low levels during the one year that the fuel was cooled before processing. The ^{129}I release data for 1969 and 1970 show that up to about 25% of the iodine was released in the aqueous low-level waste. This corresponds to a plant DF of about 4 for ^{129}I. 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 750 and 75 are achieved for ^{131}I and ^{129}I, respectively, in Case 1. The lower value was selected for ^{129}I because this isotope does not decay significantly during its lifetime in the plant and is expected to accumulate in the plant recycle streams.

Tritium and Krypton. — Both tritium and krypton 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 release[42] 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 a 1000-cfm off-gas rate. This value is consistent with the estimates for the new reprocessing plants,[1,2] and 5×10^8 was selected for this study.

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.

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 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.[49] 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 per 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 through the basket station continuously. About 90% of the fuel is dissolved during the first hour of contact with the acid, and about 99% in 2 hr.[50] 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 large fraction of the iodine remains dissolved in the dissolver solution. In this study, it is estimated that up to 75% of the iodine is evolved from the dissolver vessel (the major fraction) and from subsequent process vessels (minor amounts), and that about 25% of the iodine remains in the liquid entering the solvent extraction and the waste treatment systems.
4. Tritium remains with the dissolver solution.

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 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 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, and greater than 75% 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 $Hg(NO_3)_2$-HNO_3 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 ($\approx$ 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 HEPA filters, mixed with the purified process off-gas stream, and discharged through the 100-m stack.

The overall plant DF for ^{131}I in the gaseous effluent is 1300 based on a DF of 10 for the mercuric nitrate--nitric acid scrubber and 100 for the silver zeolite bed, and on the volatilization of 75% of the iodine from the dissolver into the off-gas. Similarly, the overall plant DF for ^{129}I in the gaseous effluent is 130 based on a DF of 10 for the scrubber and 10 for the adsorber (Sects. 4.3.3 and 4.3.5). The dissolver solution which enters the solvent extraction system is expected to contain up to about 25% of the iodine. Consequently, a significant amount of organic iodides could be formed by reaction with the organic solvent which would pass through the off-gas system to the mercuric nitrate--nitric acid scrubber. The presence of the organic iodides is expected to limit the efficiency of the scrubber. Consequently, a DF of 10 is used based on current data (Sect. 4.3.5).

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 occurs. The HEPA filters are tested periodically, as noted above, to ensure that adequate performance is maintained. Spare, parallel filters are also provided. These parallel units are brought into service when tests indicate that the "in service" filters have decreased in efficiency below the design values. The $Hg(NO_3)_2$-HNO_3 scrubber system is a standard engineering unit which is expected to operate indefinitely without interruption. Spare pumps are included in the installation to ensure the continuous movement of the scrub solution. The silver zeolite absorbers are present as additional subsequent treatment systems, and since they do not have movable components, they represent very reliable operating units. The dissolver would be shut down if a problem should arise in operation of these iodine retention units. In this event, less than 0.04 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 0.0026% of the iodine processed each year, and all of it could be released without exceeding the limit placed by the plant DF of 750 for ^{131}I, which is equivalent to a release of 0.13%. Actually, up to about 48 similar incidents could be tolerated annually on this basis such that the iodine in a total of about 2 metric tons of fuel would 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 750 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. However, the leakage of this large volume 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 25% of the iodine, none of the krypton, most 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 ~25% of the iodine that remained in the liquid phase from solvent extraction. In the Iodine Removal Partial Evaporator, about 15% of the liquid and about 95% of the iodine are volatilized and condensed and routed to the miscellaneous waste storage tank. The residual liquid in the bottom of the evaporator is passed through a bed of macroreticular resin, where about 90% of the residual iodine is removed. The purified liquid stream is then vaporized, and the vapor is discharged up the stack. This vapor is superheated to prevent condensation until it can become mixed with the large volume of ventilation air.

The miscellaneous liquid waste (MLW) is first neutralized and then concentrated in the MLW evaporator, where a small amount of the iodine is volatilized with the overhead vapor. Neutralization increases the efficiency of retention of iodine during evaporation. The overhead condensate flows to the Iodine Removal Partial Evaporator. The hot MLW evaporator concentrate is recycled 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 supernate from the miscellaneous storage tank is recycled to the MLW evaporator.

The plant DFs for iodine in the liquid waste treatment system in this study are estimated 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 (MWST).
2. The inventory of ^{131}I in the waste tank (MWST) becomes constant after about 80 days because of the decay of the short-lived ^{131}I (half-life, 8 days). However, the amount of ^{129}I (half-life, 1.6×10^7 years) in the tank increases continuously over the operating lifetime of the plant.
3. About 25% of the iodine that enters the plant each day reaches the Iodine Removal Partial Evaporator in the overhead from the fractionator plus a negligible amount of ^{131}I from the MLW evaporator. The efficiency of the Partial Evaporator for removal of ^{131}I is 95%, and the efficiency of the ion exchange bed is 90%. On these bases, a plant DF of 750 is estimated for ^{131}I in this study for the liquid waste treatment system.
4. The estimation of the plant DF for ^{129}I must include an assumption for the operating lifetime of the plant. It is assumed

that the operating lifetime of the plant is 30 years, that the accumulation of ^{129}I is linear with time, and that the average release of ^{129}I will occur during the fifteenth year. About 25% of the ^{129}I that enters the plant each day reaches the Partial Evaporator, as in the case of ^{131}I. However, the amount of ^{129}I that is in the inventory of the MWST is about 4500 times (300 days/year for 15 years) the amount entering the plant each day. If as much as 0.05% of the ^{129}I that is in the inventory of the MWST in the fifteenth year also enters the Partial Evaporator, i.e., about 225% of the amount of ^{129}I entering the plant each day, the DF for ^{129}I across the liquid waste treatment system will be reduced by a factor of about 10. On this basis, an average plant DF of 75 is used for ^{129}I across the liquid waste treatment system. The evaporators and the ion exchange units used in the liquid waste treatment system are standard engineering units which are expected to operate indefinitely without interruption. The liquid wastes can be stored temporarily if maintenance is required. Waste liquids can be reprocessed if the iodine retentions do not meet the required levels.

In 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 and that it is feasible to maintain plant DFs of 750 for ^{131}I, 75 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, 1.8 Ci/year; ^{129}I, 7.5×10^{-1} Ci/year; tritium, 1.0×10^6 Ci/year; ^{85}Kr, 1.6×10^7 Ci/year; uranium, 4.2×10^{-5} Ci/year; plutonium, 8.0×10^{-1} Ci/year; semivolatiles, 7.1 Ci/year; and particulates, 7.2 Ci/year.

4.5.2 Case 2a

In Case 2a, the overall plant DF for iodine is improved by a factor of 10 for both ^{131}I and ^{129}I (Fig. 4.4). Iodine evolution equipment is installed to volatilize iodine from the dissolver solution such that ~99.5%

of the iodine will enter the primary off-gas treatment system (DOG) and only about 0.5% will enter the liquid radwaste treatment system.

Equipment is added to recycle the mercury in the $Hg(NO_3)_2$-HNO_3 scrubber system 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, 50-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 of 7500 for ^{131}I and 750 for ^{129}I.

Gaseous Effluent Treatment. — The off-gas treatment system is the same as in Case 1 with the exception that the gas stream from iodine evolution enters the primary off-gas (DOG) treatment system downstream of the NO_x absorber. The vessel off-gas from the solvent extraction system contains up to 50 times less organic iodides than in Case 1, and consequently the DF for ^{129}I and ^{131}I for the mercuric nitrate--nitric acid scrubber is increased to 75 (Sect. 4.3.5). The DFs for the silver zeolite bed remain at 100 for ^{131}I and 10 for ^{129}I. On this basis, the overall plant DFs for the gaseous effluent are 7500 for ^{131}I and 750 for ^{129}I.

The reliability of the mercuric nitrate--nitric acid scrubber and silver zeolite systems is discussed for Case 1 in Sect. 4.5.1, and similar considerations apply to Case 2. The iodine evolution and iodine isolation systems, when fully developed, will represent standard engineering operations, and continuous long-term service is expected. In case of equipment failure, the iodine contained in about 0.2 ton of fuel could be released annually without exceeding the specified plant DFs.

Liquid Effluent Treatment. — The treatment of the liquid effluents is the same as in Case 1 (Sect. 4.5.1). In this case, however, only about 0.5% of the iodine entering the plant each day will reach the Iodine Removal Partial Evaporator and the miscellaneous waste storage tank. The remainder of the iodine is isolated and stored or packaged as sodium iodate in the mercury recycle system. Consequently, the DF for ^{131}I and ^{129}I across the liquid waste treatment system is increased to about 10,000.

In summary, the considerations listed in Sect. 4.5.1 and 4.5.2 indicate that the overall plant DFs for ^{131}I and ^{129}I in Case 2a are limited by the DFs achieved in the gaseous waste effluent system and that it is feasible to maintain plant DFs of 7500 for ^{131}I and 750 for ^{129}I in the model plant in Case 2a. The radioactive materials released to the atmosphere are ^{131}I, 1.9×10^{-1} Ci/year, and ^{129}I, 7.6×10^{-2} Ci/year, as compared with 1.9 and 7.6×10^{-1} Ci/year, respectively, for Case 1. All other nuclides are the same as in Case 1.

4.5.3 Case 2b

In this case the overall plant DF for ^{131}I and ^{129}I is increased to 10^4 (Fig. 4.5). This is accomplished by adding an Iodox system to the primary dissolver off-gas treatment system (DOG). The Iodox system contains equipment for removing iodine from the off-gases and for converting the iodine to a solid which is stored or packaged for disposal (Sect. 4.3.6). The iodine evolution, mercuric nitrate--nitric acid scrubber, mercury recycle, iodine isolation, and silver zeolite systems installed in Case 2a are retained.

Gaseous Effluent Treatment. — The combined gas streams from the shear, dissolver, and iodine evolution systems are passed through the Iodox system, where >99.9% of the iodine is removed prior to entering the NO_x absorber. The remainder of the off-gas treatment system is the same as in Cases 1 and 2a. The iodine retained in the Iodox system is converted to a solid (I_2O_5) which is stored or packaged for disposal. Most of the mass of the iodine, ~1.34 kg/day, is retained in the Iodox system. This relieves the load on the mercury scrubber and silver zeolite systems and increases the lifetime of the scrubber solution and the zeolite bed. The overall DF for ^{131}I and ^{129}I for the total gaseous effluent treatment system is >100,000.

The iodine evolution and Iodox systems will represent standard engineering operations when they are fully developed, and continuous long-term service is expected. However, if the operation of these units is interrupted, the dissolver would be shut down and only about 0.04 metric ton of fuel would continue to dissolve. The $Hg(NO_3)_2$-HNO_3 scrubbers and the silver zeolite units would remain active and ensure a plant DF of at least 750 until the plant is shut down. Actually, the iodine contained in 0.15 metric ton of fuel can be released on an annual basis without exceeding the release limit of 0.01% of the iodine ($\approx$ plant DF of 10^4). These considerations indicate that it is feasible to maintain a plant DF of 10^4 for iodine in Case 2b.

Liquid Effluent Treatment. — The liquid waste treatment system is the same as that described for Case 2a in Sect. 4.5.2. The DF for this system is ~10,000.

In summary, the considerations listed in Sect. 4.5.1 indicate that the overall plant DFs for ^{131}I and ^{129}I in Case 2b are limited by the DFs achieved in the liquid waste treatment system and that it is feasible to maintain an overall plant DF of 10^4 for ^{131}I and ^{129}I. Case 2b reduces the amount of iodine released to the atmosphere to 1.4×10^{-1} Ci/year for ^{131}I and 5.6×10^{-3} Ci/year for ^{129}I as compared with 1.8 and 7.5×10^{-1} Ci/year, respectively, for Case 1. All other nuclides are the same as in Case 1.

4.5.4 Case 3

In Case 3, the amount of ^{85}Kr released to the atmosphere is decreased by a factor ≥100 (Fig. 4.6). This is accomplished by installing selective absorption equipment in the primary off-gas system downstream of the NO_x absorber. The noble gases are absorbed in a fluorocarbon at a low temperature (<0°C), stripped, compressed, and bottled in cylinders. The cylinders can be stored for about one year in existing shielded space or in the fuel storage canal (Table 6.3). The amount of radioactive krypton released to the atmosphere annually is 1.6×10^5 Ci as compared with 1.5×10^7 Ci for Case 1. The release of all other nuclides is the same as in Case 1.

In pilot-plant tests, 99.9% of the krypton in the feed gas was removed by the selective absorption process and 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 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 is feasible to maintain a plant DF of 10^2 for krypton in the model plant in Case 3.

4.5.5 Case 4

In Case 4, the release of uranium, plutonium, and other nonvolatile fission products (particulates) 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. 4.7). 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 (Sect. 4.3.1). This provides for a third stage of filtration for the DOG and VOG as well as a second 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; and other particulates, 7.2×10^{-1} Ci/year as compared to 4.2×10^{-5}, 8.0×10^{-1}, and 7.2 Ci/year, respectively, for Case 1. The release of all other nuclides is the same as for Case 1.

4.5.6 Case 5

In Case 5, the release of semivolatile fission products is decreased by a factor ≥10 (Fig. 4.8). A caustic scrubber is added to the secondary off-gas treatment system downstream of the $Hg(NO_3)_2$-HNO_3 scrubber to remove ≥90% of the semivolatile fission products in the combined (DOG-VOG) off-gas.

The release of semivolatile fission products to the atmosphere is 7.1×10^{-1} Ci/year as compared with 7.1 Ci/year for Case 1. The release of all other nuclides is the same as for Case 1.

4.5.7 Case 6

In Case 6, the amount of tritium released to the atmosphere is decreased by a factor of 100 (Fig. 4.9). This is accomplished by adding a voloxidation treatment step to the head-end system to remove the tritium (Sect. 4.3.7). Laboratory development studies indicate that ~99% of the tritium is evolved from irradiated UO_2 when the fuel is oxidized in air at temperatures in excess of 350°C. It is expected that in a commercial plant the same evolution efficiency can be achieved using the voloxidation process and that the tritium can be collected as a small volume of tritiated water for permanent storage. The tritium 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 1050 Ci of tritium (691 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 9000 Ci per year). These considerations indicate that it is feasible to retain 99% (DF = 100) of the tritium in Case 6. It is expected that, when a fully developed voloxidation process is used, a tritium release rate of <9,000 Ci/year can be achieved.

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. 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 1.0×10^4 Ci/year as compared with 1.0×10^6 Ci/year for Case 1. The release of all other nuclides is the same as for Case 1.

4.5.8 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. 4.10). Technology that is commercially available or in various stages of development is used. The advanced radwaste treatment systems, such as voloxidation and Iodox, are not commercially available at present; 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 Case 2b. For Case 2b to be most effective, it 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 equipment; an Iodox step, including the isolation and packaging of iodine; an NO_x absorption step; and selective absorption equipment for the removal of noble gases. The secondary off-gas treatment system (DOG and VOG) consists of a $Hg(NO_3)_2$--HNO_3 scrubber, including equipment for mercury recycle and 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 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 ^{131}I, 1.4×10^{-1} Ci/year; ^{129}I, 5.7×10^{-3} Ci/year; tritium, 1.0×10^{4} Ci/year; ^{85}Kr, 1.5×10^{5} Ci/year; semivolatiles, 7.1×10^{-1} Ci/year; uranium, 4.2×10^{-6} Ci/year; plutonium, 8.0×10^{-2} Ci/year; and particulates, 7.2×10^{-1} Ci/year. These releases compare with 1.9, 7.6×10^{-1}, 1.0×10^{6}, 1.6×10^{7}, 7.1, 4.2×10^{-5}, 8.0×10^{-1}, and 7.2 Ci/year, respectively, for Case 1.

4.5.9 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. 4.11). 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.[15] 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 inleakage 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 inleakage of air to minimal volumes. In the ideal case, the volume of the inleakage 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 inleakage 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 on inleakage 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 inleakage rate of 0.004 cfm; this facility was constructed at a cost of about $10 million. A practical inleakage rate for a reprocessing facility built with the intent of obtaining a low air inleakage rate appears to be about 100 cfm or less. It is possible to design, construct, and operate process equipment, including the shear and dissolver, such that the total inleakage 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. 4.12.

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

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).

The application of these systems to the overall retention of iodine is indicated in Fig. 4.13. The iodine treatment methods listed above are described in the sections indicated and are used in Cases 2a, 2b, 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. A practical upper limit on tritium containment factors based on this method is 100 as in Case 6. 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 of the order of 500 cfm and the total water input to the plant is kept sufficiently low that the volume for long-term accumulation or permanent disposal is not excessive. Specific case studies indicate overall tritium DFs in the range of 10^5 to 10^8. A DF of 10^3 is selected for Case 7.

Krypton. — The retention of krypton is limited by the ability to route all krypton-containing streams to the primary krypton removal system. The major sources of krypton 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 such that these gases are recycled and the net off-gas passes through the primary krypton removal unit. An overall DF for krypton 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	1.0×10^{3}	1.0×10^{6}	1.0×10^{4}
Krypton	1.5×10^{3}	1.5×10^{7}	1.5×10^{5}
Iodine-129	1.1×10^{-5}	$7\ 6 \times 10^{-1}$	5.7×10^{-3}
Iodine-131	2.8×10^{-4}	1.9	1.4×10^{-1}
Semivolatiles	7.1×10^{-4}	7.1	7.1×10^{-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}	7.2	7.2×10^{-1}

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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, combustion products, etc. 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. At some later date if it is required, 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 are 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 ft^3/year in comparison to the maximum of 83,000 ft^3 listed in Table 5.1.[4]

Fuel Cladding (Hulls) Waste. — It is estimated that the neutron-induced radioactivity in the hulls, such as ^{60}Co in stainless steel and

^{93}Zr and ^{95}Zr-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 is in substantial agreement with these values and indicates a volume of 21,000 ft^3 and 10,000 Ci (after one-year decay) for uncompacted cladding waste. If the waste is compacted, the annual volume would be reduced to about 3000 ft^3. 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 Iodox systems (Sects. 4.3.5 and 4.3.6) will produce about 3 lb/day of $NaIO_3$ or $I_2O_5 \cdot 1/3\ H_2O$, 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 ft^3 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 ft^3/year in 478 12-in.-diam by 10-ft-high (filled to the 8-ft level) stainless steel cylinders. Federal regulations state that solidified wastes can be stored up to 10 years on the reprocessing site 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 (Table 5.2). The estimated composition of the discharged sewage is presented.

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 make-up area. The only nonradioactive chemical gaseous effluent of any consequence is comprised of the oxides of nitrogen and is postulated to be 100% NO_2 (Table 5.3).

5.5 References

1. Barnwell Nuclear Fuel Plant Environmental Report, Docket 50-332-22, Allied-Gulf Nuclear Services, Barnwell, S. C. (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. The cost estimates are expected to have an accuracy of about ±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 late-1973 dollars, obtained by Burns and Roe, Inc.[2] Appropriate costs based on experience in the nuclear industry[2-4] 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

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[a] Quality assurance costs based on the vendor's equipment bid prices obtained by Burns and Roe, Inc., 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.

[b] Interest is applied to the cumulative total cost at a rate of 8% per year over a 5-year cash flow expenditure period.

6.2 Annual Fixed Charges and Operating Costs

The annual fixed charges on invested capital are based on the Fuel Recycle Task Force[5] annual fixed charge rate of 24%, which is, in turn, 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

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

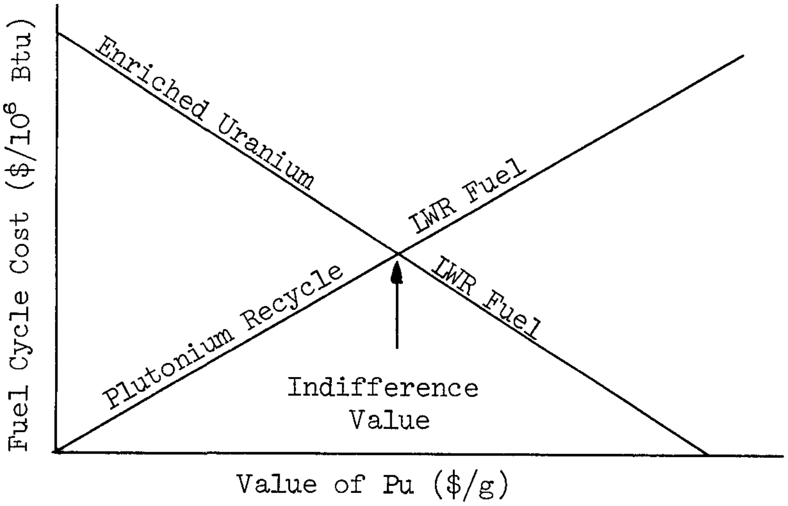
The estimated direct and capital costs for equipment in radwaste Cases 2a through 6c are presented in Table 6.3. The direct cost is the estimated installation (material and labor) cost of the equipment; the capital cost includes direct costs and indirect costs.

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 plutonium[6] can be determined by plotting the fuel cycle cost ($/10^6 Btu) versus the value of plutonium ($/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 Pu. 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, enrichment, etc., 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.[6] These estimates have been subsequently revised to use the parameters given below.[7]

1. A total of two 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/g fissile Pu for a PWR.[7] A similar study performed by the General Electric Company for the Edison Electric Institute gave an indifference value of \$7.50/g fissile Pu for a BWR.[7] Based on these estimates, an indifference value for plutonium for spent PWR and BWR reactor fuel of \$8/kg fissile Pu 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

1. Staff of the Oak Ridge National Laboratory, Siting of Fuel Reprocessing Plants and Waste Management Facilities, ORNL-4451 (July 1970).
2. 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).
3. E. D. North and R. L. Booth, Fission Product Gas Retention Study Final Report, ORNL-TM-4409 (August 1973).
4. AEC Conceptual High-Temperature Gas-Cooled Reactor (HTGR) Fuel Processing Plant, WASH-1152 (October 1970).
5. Reactor Fuel Cycle Costs for Nuclear Power Evaluation, WASH-1099 (December 1971).
6. 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).
7. E. L. Wright, Babcock and Wilcox Company, personal communication, August 3 and 6, 1973.

7.0 ENVIRONMENTAL IMPACT

The radiological impact of the model fuel reprocessing plant is assessed by calculating radiation doses to individuals, populations, and selected biota 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 average dose per year of plant operation to both individuals and to the population within 55 miles which may result from the expected radionuclide discharges during normal operation are discussed below. A dose calculated for 1 year of radionuclide intake (internal-exposure pathways) is an estimate of the total dose an individual will accrue within his lifetime as a result of that 1 year of exposure (i.e., dose commitment). All of the doses estimated in this report represent dose commitments.

The radiation doses to the total body and internal organs from exposure to penetrating radiation from external sources are approximately equal. However, they 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 for the atmosphere-pasture-cow-milk pathway. For this pathway, the estimated dose to the thyroid of a one-year-old child from radioactive iodine in milk is several times that for an average adult.[1,2]

The population dose estimates are the sums of the total body doses to individuals within 55 miles of the plant. Total body doses from gamma exposures are approximately the same as the doses to gonads and, therefore, are used in the man-rem estimates because gonads have the most restrictive dose limits.[3,4] Since radiation doses to the total body are relatively independent of age,[5] the man-rem estimates are based on total body doses calculated for adults.

Estimates of dose to the total population are limited to distances of 55 miles from the model plant. At greater distances, the doses become low as compared to the natural radiation background, and the population density and meteorological factors merge with the average for the United States rather than remaining as distinct functions of the model plants. For example, a recent study[6] suggests that a plant which processes 5 metric tons of fuel per day and releases all of the krypton and tritium (similar to Case 1 in this study) could lead to an annual total body dose to the population of the United States of 520 man-rem from ^{85}Kr and 3,700 man-rem from tritium. This amounts to less than 0.02% of the dose from the natural, annual, background dose of about 130 mrem. The annual dose to the world population is estimated at 8,520 man-rem from krypton and 4,800 from tritium.[6] Similarly, these doses are small fractions of the natural background. The estimated doses to the population from other radionuclides are lower fractions of the background radiation.

7.1 Meteorology

Release of gaseous effluents to the atmosphere is the major pathway for environmental contamination from fuel reprocessing facilities. No radioactive liquids are released from the model plant. Fuel reprocessing facilities have no special requirements for large quantities of water, i.e., for cooling or dilution as compared to power plants, and thus may be located at a site that is remote from aquatic environments. Atmospheric transport is the principal mode of delivery of radioactive materials to terrestrial environments associated with the fuel reprocessing facility.

Atmospheric transport of radioactive substances is calculated according to the Gaussian plume model.[7] A computer code[8] has been modified to calculate the approximate annual average concentrations in air for short- and long-lived radionuclides in the atmosphere at various distances from the source. The meteorologic data required for the calculations are joint frequency distributions of velocity and direction summarized by stability class. Meteorologic data from representative midwestern and southeastern coastal regions[9] are used to calculate average values of X/Q' (sec·m^{-3}),i.e., factors that are used to calculate the concentration of radioactive material at a reference point per unit of source strength. The X/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, 10.0, 15.0, 25.0, 35.0, 45.0, and 55.0 miles from the point of release. The X/Q' values are based on a release from a 100-m stack and a 1.5 x 10^6 cfm flow rate. Maximum and minimum annual X/Q' values in sectors at successive distances from the release point are given in Figs. 7.2 and 7.3 for the midwestern and southeastern coastal sites, respectively. All values, irrespective of direction, range between the maximum-minimum values at a given distance. Magnitudes of X/Q' values are somewhat similar at the two sites, but directions at which maximum-minimum values are attained are different.

For both locations, the maximum concentration of radioactive substances in air (largest X/Q', least dilution) occurs at approximately 0.7 mile from the point of release. Maximum X/Q' values are predicted over the range of 0.5 to 2 miles, beyond which the X/Q' values decrease about tenfold out to a distance of 55 miles (Figs. 7.2 and 7.3). The appropriate X/Q' value 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. X/Q' values of 5.3 x 10^{-8} and 3.7 x 10^{-8} are used at the midwestern and coastal locations, respectively, to calculate the maximum doses at a distance of 0.5 mile. Details of the use of X/Q' values to calculate doses are given in ORNL-4992. Concentrations in air for each sector are used to calculate dose via inhalation and submersion in air. Air concentrations in various sectors are also used in conjunction with deposition velocities to estimate a steady-state ground concentration for annual exposures.

Accumulation of radioactive materials on the ground surface is represented with an infinite plane source model for external radiation exposure. The ground deposits are also assimilated into food which, when ingested, results in additional dose via the food chain pathway. Radioactive materials from the atmosphere are deposited on the ground surfaces through the mechanisms of dry deposition and 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, and is calculated from deposition velocity,[10] Vg, for a one-year time interval. Deposition velocity values for particles and reactive gases such as iodine commonly range from 0.1 to 1.0 cm/sec;[11] for micron-sized particles, Vg's may approach 10 cm/sec. A value of 1.0 cm/sec is used for calculation of ground concentrations of radioactive particles, iodine, and semivolatile substances.

Although many variables influence the washout of radioactivity from the atmosphere with rain, dew, etc.,[12] Cowser et al.[13] showed that washout would cause only a negligible decrease in annual air concentration based on a washout weight of 0.038 (Oak Ridge, Tennessee) and a washout coefficient of 10^{-4} $\sec^{-1}$. The annual increase in ground concentration from washout would likewise be nominal. Thus, for model fuel reprocessing sites, total transfer of radioactive materials from the atmosphere to the ground surface is included in the dry deposition rate term.

7.2 Population

Population distributions were derived which would be representative of southeastern coastal and midwestern environments. The population distributions are the average of population distributions around two fuel fabrication plants and one reprocessing plant for each area, i.e., the midwestern and southeastern coastal sites. Distributions for sites near St. Louis and Wilmington were included in the averaging because the meteorologic data used for atmospheric transport of radioactive substances are based on these areas. The Wilmington site also represents the half-annulus distribution which 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,[13] provides sector summaries for annuli bounded by distances of 0.0, 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 atmospheric X/Q' values are calculated. The computer code summaries of population data from census tapes are accurate beyond a five-mile radius. Within five miles, where sectors represent relatively small areas, distributions are somewhat disconnected because census enumeration districts encompass several sectors while the population record is reported in a single sector. Averaging data from three locations smooths the major discontinuities and results in cumulative totals which are somewhat similar to those reported for actual fuel reprocessing facilities.[14,15]

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 coastal plain site except for a factor of 5 increase to 289 individuals per square mile, representing a small city, in the 5- to 10-mile annulus. The 9500-square-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 five miles, the density increases to 126 individuals per square mile at 10 miles, and to 440 individuals per square mile in the 25- to 50-mile annulus. A large city is included in a portion of the 55-mile area encircling the model fuel reprocessing facility. Cumulative population in the midwestern site is approximately six times greater than for the coastal site.

7.3 Radiation Dose from Gaseous Effluents

Concentrations of radionuclides in air and on the surface of the soil are used to estimate the radiation dose to individuals at various distances and directions from the model fuel reprocessing plant. The doses resulting from submersion in the gaseous effluent, exposure to contaminated ground surface, and intake of radionuclides through inhalation and ingestion are calculated with computer codes[16] which use dosimetric criteria of the International Commission on Radiological Protection and other recognized authorities.

Estimates of intake of radionuclides by man through terrestrial food chains were made with a model and computer code[17] which considers transfers of radioactivity to man via ingestion of food crops, beef, and milk. Many basic environmental parameters used in this model are conservative, i.e., values are chosen to maximize intake by man. Reducing factors, such as shielding provided by dwellings and time spent away from the calculation location, are not considered. Moreover, in estimating the dose to individuals via ingestion of plants, meat, and milk, an individual is assumed to obtain all of his food at the reference location specified in the calculation. This event is not impossible, but extremely unlikely. Thus, individual dose estimates calculated by these methods are higher than actually expected.

Assumptions, models, and codes used to estimate radiation doses are given in ORNL-4992.

7.3.1 Individual and Population Dose

Approximately 38% 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 and exposure to contaminated ground. Internal exposure from inhalation and ingestion contributes the remaining 62% of the total body dose.

The maximum annual total body dose and maximum organ doses to individuals at 0.5 mile from the model plant are summarized in Table 7.4 for all radwaste treatment cases and for the coastal and midwestern sites. The total population dose out to a distance of 55 miles is also presented. The doses to individuals at 0.5 mile can be multiplied by the following factors to obtain the dose at increased distances: 0.73 for 1 mile, 0.65 for 1.5 miles, and 0.56 for 2 miles at the coastal site; and 0.46 for 1 mile, 0.37 for 1.5 miles, and 0.28 for 2 miles at the midwestern site. Maximum individual and organ doses at 1.5 miles from the model plant are given in Table 7.5. This distance is probably more representative of the boundary distance from a 5-metric ton/day reprocessing plant. The contributions to total body dose through the various exposure modes are listed in Table 7.6.

Maximum total body doses at 0.5 mile for the midwestern site are approximately 1.8 times higher than those estimated for the coastal site due to meteorological differences. The average annual total body doses (Table 7.7) for individuals living within 55 miles of the two sites are generally two to three times higher for the coastal site. Population doses, however, are higher around the midwestern site because six times more people live within 55 miles of this site as compared with the coastal site. Cumulative population doses (total body man-rem) out to a distance of 55 miles from both sites are given in Table 7.8. It is doubtful that either individuals or populations would, in fact, receive these estimated doses. The conservative assumptions listed in Sect. 7.3 tend to maximize the estimated doses.

The principal radionuclides in the gaseous effluent which contribute to the total body doses of individuals are listed in Table 7.9. The major contributions in the Case 1 study are from tritium (43%), $^{134-137}$Cs (21%), and ^{85}Kr (17%). The fractional contributions of the radionuclides through various pathways are listed in Table 7.10. Tritium contributes about 94% of the dose through inhalation, ^{85}Kr about 99% through submersion, and $^{134-137}$Cs about 54 and 59% of the doses via contaminated ground and ingestion of contaminated food, respectively. The model plants release large amounts of tritium in Cases 1 through 5, and since tritium is a major contributor to total body dose, the reduction in release of tritium in Case 6 by a factor of 100 reduces the total body dose at the coastal plant from 4.4 millirem in Case 1 to 2.5 millirem in Case 6. A reduction in the release of ^{85}Kr in Case 3 by a factor of 100 decreases the total body dose to only 3.6 millirem, since the contribution from submersion to total body dose is less than that for inhalation. A reduction in release of $^{134-137}$Cs and other nonvolatile particulates by a factor of 10 in Case 4 decreases the total body dose to 3.3 millirem, a proportionally larger reduction than for either tritium or krypton.

7.3.2 Dose to Organs of Individuals

Maximum annual doses to organs of individuals at 0.5 mile from the model fuel reprocessing plant located on both sites are given in Table 7.4 for all radwaste treatment cases. Average doses to organs would be approximately 30 and 67% less than maximum values for the coastal and midwestern sites, respectively.

The principal radionuclides that contribute to the organ doses through the ingestion pathway are given in Table 7.11. Radiation dose to organs is largely dependent on the specificity for certain radionuclides to accumulate in certain organs. Therefore, a radwaste treatment case which greatly reduces the presence of a given radionuclide in the environment will reduce the dose to the organ that is exposed to the radionuclide via inhalation or ingestion pathways. Inhalation is the major pathway of exposure to internal body organs (Table 7.6). In the case of skin, submersion in the gaseous effluent, where ^{85}Kr contributes 99.9% of the submersion dose (Table 7.10), is the major exposure pathway. Skin is not included in the organs listed in Tables 7.4 and 7.11 because essentially all of the dose to skin is caused by ^{85}Kr.

The annual maximum dose to skin (0.04 mm depth) from ^{85}Kr at a distance of 0.5 mile is 58.4 and 104.2 millirem for the coastal and midwestern sites, respectively. This dose is effectively reduced by factors of 10^2 in treatment Cases 3 and 6c and 10^4 in Case 7 where the corresponding amounts of krypton are retained in the plant (Table 7.4). Similarly, the dose to the thyroid is decreased in cases where the releases of ^{129}I and ^{131}I are reduced; however, total body doses are only decreased slightly. Radwaste treatments that reduce the release of semivolatile materials cause a reduction in dose to the GI tract from

^{106}Ru. Treatments to reduce the release of particulates in the gaseous effluent cause a reduction in dose to bone from ^{90}Sr, and to the liver, spleen, muscle, lungs, and gonads from ^{134}Cs and ^{137}Cs.

The presence of ^{129}I in the gaseous effluent from a reprocessing plant is particularly significant in terms of dose to the thyroid since iodine concentrates in the thyroid and the long-lived ^{129}I (half-life, 1.6 x 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 than ^{129}I and hence contributes a smaller fraction of the thyroid dose. For example, in the Case 1 study, the dose from ^{129}I is about ten times higher than that for ^{131}I, i.e., a maximum, annual adult thyroid dose of 94.6 millirem for ^{129}I vs 8.7 millirem for ^{131}I (Tables 7.12 and 7.13). Similarly, ^{129}I accounts for 4.4% of the total body dose in Case 1 vs ~0.4% for ^{131}I (Table 7.9). Tables 7.12 and 7.13 show the relative importance of various pathways leading to adult thyroid doses from ^{129}I and ^{131}I, respectively. Ingestion is the major exposure pathway for both radionuclides. Iodine-131, because of its relatively short radioactive half-life is primarily ingested in milk and to a lesser extent in vegetables. Iodine-129 also is ingested primarily in milk but can accumulate in beef such that, under steady-state conditions, almost 22% of the thyroid dose from this radionuclide is due to the ingestion of beef. The estimates of ^{129}I intake via ingestion of beef[17] used in this study 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.[18] The average annual doses to the adult thyroid are given in Table 7.14 for individuals at selected distances from the coastal and midwestern sites.

The maximum annual dose to thyroid of a 1-year-old child from ^{129}I and ^{131}I at a distance of 0.5 mile in the Case 1 study is presented in Table 7.15. The dose is 136.3 millirem for ^{129}I and 53 millirem for ^{131}I, a total of 189.3 millirem. This total dose is 1.8 times higher than for the adult dose, and the doses from ^{131}I and from ^{129}I are about 1.5 and 6.1 times higher than for the adult. The major pathway, as with the adult, is by ingestion. However, milk is a much more important source, accounting for 96% of the dose in the case of both radionuclides; on the other hand, the dose due to the ingestion of vegetables and beef indicates that these sources play a much less important role in the dose to the thyroid of a one-year-old child than in the dose for the adult (Table 7.15).

7.4 Radiation Dose from Liquid Effluents

The model fuel reprocessing plant has no radioactive liquid effluent. However, to evaluate aquatic pathways leading to a potential radiation dose to man and other biota, it is assumed that radioactive materials from the gaseous effluent would be deposited in an estuary at the coastal site and in a fresh-water river at the midwestern site at the same rates and amounts as on a similar area of land.

For the coastal site, it is assumed that an estuary, 1 mile long by 0.5 mile wide by 2 m deep, is located 0.5 mile from the model plant in the direction of the prevailing wind. Gaseous effluents are deposited in this estuary for 1 year. All radionuclides remain in the water, with no further dilution due to tidal influences or settling out.

For the midwestern site, it is assumed that a segment of a river, 1 mile long by 0.1 mile wide by 3 m deep, is located 0.5 mile from the model plant in the direction of the prevailing wind. Gaseous effluents are deposited in this river segment for 1 year. All radionuclides remain in the water with no further dilution by volume flow or settling out.

7.4.1 Radiation Doses from Aquatic Pathways

The annual total body doses estimated for exposure by aquatic pathways are given in Table 7.16. These doses are a small fraction of the dose to individuals as estimated for exposure by terrestrial pathways. These estimates were made for treatment Case 1 (the base case plant). Eating fish from either the estuary or river results in the largest portion of the total dose from aquatic pathways. Table 7.17 gives the principal radionuclides that lead to total body exposures from aquatic pathways. In the case of swimming, most of the dose is from ^{134}Cs, ^{137}Cs, ^{3}H, and ^{95}Nb. Treatment cases involving reduction of semivolatiles and gases would be most effective in reducing this exposure mode. In the case of eating fish, the cesium radionuclides contribute the greatest portion of the dose. For the drinking pathway, ^{134}Cs, ^{137}Cs and ^{3}H contribute most of the dose to the total body. It is significant to note that the long-lived radionuclides such as ^{129}I and the isotopes of uranium and plutonium contribute very little to total body dose from aquatic pathways. Dose to the adult thyroid, however, is estimated to be 0.8 millirem/year from drinking the river water.

7.5 Radiation Doses to Organisms Other Than Man

Radiation doses to aquatic plants, invertebrates, fish, and waterfowl are estimated for the coastal estuary and freshwater river. Bioaccumulation factors for saline[19] and freshwater[20] organisms are used in these calculations. Tables 7.18 and 7.19 give annual doses to organisms living in the estuary and the river, respectively. The concentrations of radionuclides in these two aquatic environments are also given. Doses to organisms in the freshwater river are higher than those for the estuary because radionuclide bioaccumulation factors are higher for freshwater organisms.

In general, doses to algae and invertebrates (saline and freshwater) are due primarily to radionuclides of Cm, Ru, Cs, and Y. The dose to fish is heavily influenced by radionuclides of Cs, Cm, and Nb. For waterfowl, over 90% of the total dose from saline water is from the iodine radionuclides, while over 90% of the total dose from freshwater is from the cesium radionuclides. Therefore, radwaste treatment cases that reduce gases, semivolatiles, or particulates would decrease the doses to some major organisms of aquatic habitats.

7.6 Estimates of Error for Atmospheric Dilution and Population Parameters

The atmospheric concentration of radioactive substances and the population distribution are parameters which determine the radiation dose commitment to the human population. These parameters are used with dose conversion factors in the calculation of total body and organ doses for each sector. Variability of X/Q' values among direction sectors at a given distance is less for the coastal area than for the midwestern area (Table 7.20). However, the standard deviation for X/Q' ranges from 25 to 50% of the mean at both areas.

The variation in the cumulative population distribution is characterized by standard deviations ranging from 30 to 100% of the mean for coastal and midwestern regions, respectively (Table 7.20). Population distributions for certain annuli, e.g., a 10- to 15-mile increment (Table 7.2), exhibit standard deviations which often exceed the mean. Standard deviations for some sectors with relatively sparse population are twice the mean value.

Results of this limited error analysis of X/Q' values and population distribution indicate that the variability of these parameters would influence estimates of dose to individuals and population groups by factors of 2 to 4. This is based on the approximate assumption that 95% of the X/Q' factors and population distributions would fall within two standard deviations of the mean. Dose to an individual at any distance would vary by as much as a factor of 2 (CV = 0.5 for midwestern X/Q'), while dose commitment to the population would vary by as much as a factor of 4 (CV = 1.0 for midwestern population). This analysis considers error sources independently; no attempt is made to estimate cumulative or multiplicative sources of error.

7.7 References

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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 dose commitments, from these systems described in Sect. 7.0 are presented in this section. The accuracy of the cost estimates is about ±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, 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 such that the costs and doses are maximized.

The annual costs and dose commitments for the base case (Case 1) and succeeding case studies (Cases 2 to 7) at the midwestern and coastal sites are summarized in Tables 8.1-8.3 and Figs. 8.1-8.11. 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. Case 2b includes the cost of Case 2a, and Case 6c is a cumulative case which includes the costs of Cases 2b through Case 6. The dose commitments are reported on several bases, i.e., (1) maximum annual individual total body, thyroid, and bone dose (millirem) at 0.5 mile from the plant, which represents the total dose from the radioactive materials released from the plant in a given case; (2) incremental maximum annual individual total body dose (millirem) at 0.5 mile, which represents the difference in dose between Case 1 and a given case; and (3) annual average total body dose (man-rem) received by the general population out to a distance of 55 miles. Factors for calculating maximum doses at distances greater than 0.5 mile are given in Sect. 7.0. Average individual total body doses out to a distance of 55 miles are also listed in Sect. 7.0. The maximum, rather than the average individual doses to total body, thyroid, and bone, and the average population dose are used to illustrate the cost-benefit relationships in this section of the report and thus maintain the principle of the selection of maximum effect in this study. The total body dose is selected because of its obvious importance, and thyroid and bone are selected because the principal radionuclides that contribute to total body dose are also contributors for thyroid (^{129}I) and bone (^{244}Cm) doses. The latter two are also selected because of the interest in defining the effects of releases of radioactive iodine and transuranium nuclides. Internal exposure to radiation through inhalation and ingestion of radionuclides accounts for about 62% of the total body dose to individuals and population. The major contributors to this dose are ^{3}H, ^{85}Kr, and ^{134,137}Cs.

Individual Total Body Dose. — The annual cost of reducing the maximum individual total body dose (millirem) at 0.5 mile is presented in Tables 8.1 and 8.2 and Figs. 8.1 and 8.2. Small reductions (1.3 to 6.8%) in maximum total body dose are achieved by decreasing the release of ^{129,131}I and ^{103,106}Ru in Cases 2a, 2b, and 5 at annual costs varying from $123,000 to $753,000. Reductions of about 18 to 43% in the maximum total body dose are achieved in Cases 3, 4, and 6 at annual costs of $303,000 to $1,409,000, about a 96% reduction for Case 6c at an annual cost of $3,671,000, and a 1500-fold reduction for Case 7 for $45,500,000. The reductions for Cases 3, 4, and 6 are the result of decreasing the release of ^{85}Kr, particulates, and ^{3}H by factors of 100, 10, and 100, respectively. These values are reflected in the comparison of the ratios of annual costs for radwaste treatment to the incremental reductions in maximum total body dose at 0.5 mile (cost-benefit ratios), which are listed in Table 8.3 and shown as bar graphs in Figs. 8.3 and 8.4.

The cost-benefit ratios vary from $0.15 million/millirem for Case 4 to $5.76 million/millirem for Case 7 at the midwestern site, and from

\$0.28 million/millirem for Case 4 to \$10.3 million/millirem for Case 7 at the coastal site. The most efficient system in terms of cost-benefit ratio ($\$10^6$/millirem), i.e., the lowest cost-benefit ratio, is the filter system in Case 4. The cost-benefit ratios increase in the following order: Case 4 (particulate retention), Case 5 (semivolatile retention), Case 6 (tritium retention), Case 6c (cumulative case), Case 3 (krypton retention), Case 2a and 2b (iodine retention), and Case 7 (improved retention of all radionuclides). Cases 2a and 2b for iodine retention rank low in relative efficiency since iodine and the dose to the thyroid contribute a small fraction of the total body dose.

The cumulative effect of adding the radwaste systems in the case studies to the base case is illustrated in Table 8.3 and Figs. 8.5 and 8.6 for the midwestern and coastal sites, respectively. The treatment systems are added in the order of increasing cost-benefit ratio (i.e., decreasing efficiency), and the cumulative annual maximum individual total body dose at 0.5 mile is compared with both the cumulative total annual reprocessing cost and the unit reprocessing cost (\$/kg uranium) for radwaste treatment. The incremental reductions in dose and the incremental cost-benefit ratios between the case studies are also listed for each data point. As a result of the cumulative additions of the case studies, Case 2b represents the total effect of Cases 2a through 6 and is thus equivalent to the cumulative Case 6c. The greatest decrease in dose with increasing expenditure of money occurs by adding Cases 4, 5, and 6 to the base case, resulting in a reduction of total body dose from 4.4 millirem to 1.1 millirem (75% of total) at the coastal site and from 7.9 millirem to 2.0 millirem (75% of total) at the midwestern site for a total increase in annual cost of \$1.424 million. The further addition of Cases 3, 2a, and 2b results in a further reduction of total body dose from 1.1 millirem to 0.19 millirem (~21% of total) at the coastal site and from 2.0 millirem to 0.33 millirem (~21% of total) at the midwestern site at an additional increase in reprocessing cost of \$2.247 million. The change in slope of the curve is also illustrated by the cost-benefit ratios, which increase slowly from 0.28 to 0.53 for the coastal site and from 0.15 to 0.29 for the midwestern site for the addition of Cases 4, 5, and 6. For these additions, the cost-benefit ratio is about 1.3 times higher than the preceding case. However, Case 3 is about 3 times higher than Case 6, illustrating the change in the curve (Table 8.3). The change in slope of the curve is illustrated more graphically in Fig. 8.7, where the dose vs the cumulative annual cost is presented in rectangular coordinates for the coastal site. In this comparison, the cases are cumulated on three different bases, i.e., (1) increasing cost-benefit ratio (annual cost/incremental reduction in dose), (2) decreasing incremental reduction in dose, or (3) increasing cost. The cost-benefit plot, of course, represents the most efficient use of money in reducing dose since the cases are selected on that basis.

Thyroid Dose. — The annual cost of reducing the annual maximum individual adult thyroid dose at a distance of 0.5 mile is shown in Tables 8.1 and 8.2 and Fig. 8.8 for Cases 1, 2a, 2b, 6c, and 7. These cases are designed to illustrate the effect of decreasing the release of radioactive iodine, which contributes more than 95% of the dose to the thyroid. For Cases 2a and 2b, all of the treatment costs can be assessed to reduction in thyroid dose since these cases are specifically designed to illustrate iodine retention. However, the total costs for Cases 6c and Case 7 should not be assessed to reduction in iodine releases, as these cases are designed for the retention of all types of radioactive materials. Strictly speaking, the costs can be assessed to the reduction of thyroid dose since all of the radioactive materials have some effect on thyroid dose. However, the retention of materials other than iodine is an inefficient method for reducing thyroid dose. No effort is made to define the fractions of the total cost of Cases 6c and 7 that apply exclusively to retention of iodine or reduction in thyroid dose. The difference in the dose for a given case at the two sites (displacement of curves in Fig. 8.8) is the result of the difference in meteorology at the two sites.

The maximum annual dose decreases from 103.2 millirem for Case 1 to 16.4 millirem for Case 2a (annual cost, $274,000), to 8.0 millirem for Case 2b (annual cost, $753,000), to 1.6 millirem for Case 6c (annual cost, $3,671,000), to 0.006 millirem for Case 7 (annual cost, $45,500,000) at the midwestern site, and to dose values about 44% lower at the coastal site. These ratios in annual dollars per millirem are $5,592 for Case 2a, $13,944 for Case 2b, $63,955 for Case 6c, and $780,486 for Case 7 at the coastal site. The same pattern is obtained for these values at the midwestern site.

Bone Dose. — The annual cost of reducing the annual maximum individual adult bone dose at a distance of 0.5 mile is shown in Tables 8.1 and 8.2 and Fig. 8.9 for Cases 1, 4, 6c, and 7. These cases are specially designed to illustrate the effect of decreasing the release of particulates (which includes the transuranium nuclides) on bone dose. No effort is made in Cases 6c and 7 to define the fractions of the total cost that apply exclusively to the retention of radioactive materials that concentrate in the bones. The difference in the dose for a given case at the two sites (displacement of curves) is due to the difference in meteorology. The maximum annual dose decreases from 20.5 millirem at the midwestern site for Case 1 to 6.4 millirem for Case 4 (annual cost of $303,000), to 2.4 millirem for Case 6c (annual cost of $3,671,000) and to 0.011 millirem for Case 7 (annual cost of $45,500,000). The dose values are about 43% lower for the coastal site. The most efficient system as measured in terms of ratio of annual cost to incremental reduction bone dose is Case 4 at the coastal site. These ratios in annual dollars per millirem are $39,351 for Case 4, $356,408 for Case 6c, and $3,924,443 for Case 7. The same pattern is obtained for these values at the midwestern site.

Population Dose. — The annual cost of reducing the average total body dose (man-rem) for the general population out to a distance of 55 miles is presented in Tables 8.1 and 8.2 and Figs. 8.10 and 8.11. Reductions in population dose are small for Cases 2a, 2b, and 5, up to a 3.3% reduction in the 485.5 man-rem and 173.1 man-rem doses listed for Case 1 at the midwestern and coastal sites, respectively. Larger reductions are obtained in the other cases, i.e., ~15% for Case 3, ~22% for Case 4, ~54% for Case 6, ~97% for Case 6c, and about 1500-fold for Case 7. The difference in dose at the two sites is the result of a difference in population densities. Case 6 (tritium removal) is the most effective at the midwestern site on a cost basis for reducing the population total body dose, i.e., \$3787/man-rem, whereas Case 4 (particulate removal) is the most effective at the coastal site, i.e., \$10,100/man-rem (Fig. 8.10 and 8.11).

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/kg of contained U+Pu[a] for PWR fuel and \$60.37/kg U+Pu[a] for BWR fuel (Sect. 6.4, Table 6.4). The annual reprocessing cost in the base Case 1 is \$30.33/kg U (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/kg U+Pu for PWR fuel and \$30/kg U+Pu for BWR fuel. In Cases 2 through 6, the costs for additional radwaste treatment systems increase from about 0.2 to 2% of the incremental difference between the reprocessing cost and the value of the PWR fuel, and by 0.3 to 3% 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/kg U, as illustrated for Case 7, would approach the incremental difference between the value of the fuels and the reprocessing cost under current economic conditions.

Capital and Power Costs. — The capital costs for Cases 2a through 6 range from \$0.340 million to \$3.871 million, or up to about 3% of the

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[a] The production rate at the reprocessing plant is based on an annual charge of 1500 metric tons of uranium to the reactor.

$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 $0.123 million to $1.409 million, which is equivalent to contributions to the power cost of 3.19×10^{-4} to 3.66×10^{-3} mill/kWhr, respectively (Sect. 6.0, Table 6.1). Cumulative Case 6c and Case 7 amount to 9.54×10^{-3} and 1.2×10^{-1} mill/kWhr, respectively. All of these values are less than about 3% 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 a Reprocessing Plant

Nuclide	Half-Life	Cubic Meters of Air at RCG[a]
Gaseous		
H-3	12 y	3.46E+9
Kr-85	10.4 y	3.67E+10
I-129	1.6×10^7 y	1.87E+9
I-131	8.05 d	9.23E+9
Semi-Volatiles		
Ru-103	40 d	2.47E+13
Ru-106	1.0 y	2.01E+15
Particulates		
Sr-89	50.4 d	2.84E+14
Sr-90	28 y	2.56E+15
Y-90	64 h	2.56E+13
Y-91	59 d	1.43E+14
Zr-95	65 d	2.49E+14
Nb-95	35 d	1.58E+14
Ag-110 m	249 d	7.92E+12
Sb-125	2.7 y	8.77E+12
Te-127m	105 d	5.77E+12
Te-129m	33 d	2.21E+12
Cs-134	2.1 y	5.31E+14
Cs-137	30 y	2.13E+14
Ce-141	32.5 d	9.11E+12
Ce-144	285 d	3.76E+15
Pm-147	2.7 y	4.87E+13
Eu-154	16 y	6.68E+13
Eu-155	1.7 y	2.11E+12
U-234	2.48×10^5 y	3.77E+10
U-235	7.13×10^8 y	8.55E+8
U-236	2.39×10^7 y	1.44E+10
U-238	4.51×10^9 y	1.05E+11
Pu-238	86 y	4.03E+16
Pu-239	2.4×10^4 y	5.38E+15
Pu-240	6.6×10^3 y	7.96E+15
Pu-241	13 y	3.43E+16
Pu-242	3.87×10^5 y	2.30E+13
Am-241	458 y	7.63E+14
Am-243	7.5×10^3 y	9.08E+13
Cm-242	162 d	4.25E+15
Cm-244	18 y	8.01E+15

[a] Total for actinides is 1.01×10^{17} and for all others is 1.04×10^{16} at 160 days cooling; curies of each isotope per ton of uranium charged to reactor divided by the Radiation Concentration Guide to the isotope as listed in 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 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	6.91E+2	1	1.0E+6	1.0E+6	1.0E+6	1.0E+6	Same as Case 1	Same as Case 1	1.0E+4	1.0E+4	1E+3	1.0E+3
Kr-85	1.0E+4	1	1.5E+7	1.5E+7	1.5E+7	1.5E+5			1.5E+7	1.5E+5	1E+4	1.5E+3
I-129	3.74E-2	75	7.5E-1	7.5E-2	5.6E-3	7.5E-1			7.5E-1	5.6E-3	5E+6	1.1E-5
I-131	9.23E-1	750	1.8E00	1.8E-1	1.4E-1	1.8E00			1.8E00	1.4E-1	5E+6	2.8E-4
Semi-Volatile												
Ru-103	7.41E+4	1E+8	1.1E00	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	1.1E-1	Same as Case 1	1.1E-1	1E+12	1.1E-4
Ru-106	4.02E+5	1E+8	6.0E00					6.0E-1		6.0E-1	1E+12	6.0E-4
Uranium												
U-234	7.55E-1	5E+7	2.3E-5	Same as Case 1	Same as Case 1	Same as Case 1	2.3E-6	Same as Case 1	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	Same as Case 1	Same as Case 1	Same as Case 1	2.1E-3	Same as Case 1	Same as Case 1	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	Same as Case 1	Same as Case 1	Same as Case 1	2.6E-2	Same as Case 1	Same as Case 1	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	2.49E+5	5E+8	7.5E-1				7.5E-2			7.5E-2	1E+12	3.7E-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	5.77E+3	5E+8	1.7E-2				1.7E-3			1.7E-3	1E+12	8.7E-6
Te-129m	2.21E+3	5E+8	6.6E-3				6.6E-4			6.6E-4	1E+12	3.3E-6
Cs-134	2.15E+5	5E+8	6.5E-1				6.5E-2			6.5E-2	1E+12	3.2E-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	7.52E+5	5E+8	2.3E00				2.3E-1			2.3E-1	1E+12	1.1E-3
Pm-147	9.73E+4	5E+8	2.9E-1				2.9E-2			2.9E-2	1E+12	1.5E-4
Eu-154	6.86E+3	5E+8	2.1E-2				2.1E-3			2.1E-3	1E+12	1.0E-5
Eu-155	6.33E+3	5E+8	1.9E-2				1.9E-3			1.9E-3	1E+12	9.5E-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

[a] The model plant processes 1,500 metric tons of uranium per year.

[b] The fuel is irradiated to 33,000 MWd/metric ton at 30 kW/kg and decayed 160 days (computer code ORIGEN).

Table 4.3. Summary of Variables for Model Nuclear Fuel Reprocessing Plant(a) Gaseous and Liquid Radwaste Treatment Systems

	Radwaste Treatment Case No.								
	1	2a	2b	3	4	5	6	6c	7
Treatment Objective	Base case(b)	Reduce iodine(c) release by 10	Reduce iodine(d) release, DF - 10^4	Reduce krypton(e) release by 100	Reduce particulate,(f) uranium and plutonium release by 10	Reduce semi-volatile(g) release by 10	Reduce tritium(h) release by 100	Cumulative reduction(i) of releases, summation of Cases 2 through 6	Further reduction in(j) releases, overall DFs: Iodine - 5×10^6 Krypton - 10^4 Particulates and Semi-volatiles - 10^{12} Tritium - 10^3
					Overall Plant Decontamination Factor(k)				
Iodine-131	750	7500	1×10^4	750	750	750	750	1×10^4	5×10^6
Iodine-129	75	750	1×10^4	75	75	75	75	1×10^4	5×10^6
Krypton-85	1	1	1	100	1	1	1	100	1×10^4
Particulates	5×10^8	5×10^8	5×10^8	5×10^8	5×10^9	5×10^8	5×10^8	5×10^9	1×10^{12}
Plutonium	2×10^8	2×10^8	2×10^8	2×10^8	2×10^9	2×10^8	2×10^8	2×10^9	1×10^{12}
Uranium	5×10^7	5×10^7	5×10^7	5×10^7	5×10^8	5×10^7	5×10^7	5×10^8	1×10^{12}
Semi-Volatiles	1×10^8	1×10^8	1×10^8	1×10^8	1×10^8	1×10^9	1×10^8	1×10^9	1×10^{12}
Tritium	1	1	1	1	1	1	100	100	1×10^3
					Equipment Unit or Function and Flowsheet Reference				
Iodine									
Gaseous	$Hg(NO_3)_2$-HNO_3 scrubber, AgZeO adsorber, HEPA filters	Iodine evolution,(c) $Hg(NO_3)_2$-HNO_3 scrubber, AgZeO adsorber, HEPA filters	Iodine evolution,(d) Iodox, $Hg(NO_3)_2$-HNO_3 scrubber, AgZeO adsorber, HEPA filters	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 2b(i)	Reduced volume net off-gas(j) release, recycle process and cell off-gas, 2 stages of Iodox, 2 stages of AgZeO adsorbers, HEPA filters
Liquid	Evaporation, iodine removal on resin bed, evaporation and vaporization	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 2b	Evaporation, iodine removal on resin bed, evaporation and recycle, storage for excess water, no release of water or water vapor
Solid	None	Hg recycle, $NaIO_3$ recovery and storage	Same as Case 2a	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 2b	Isolate NaI for storage
Krypton	None(a)	Same as Case 1	Same as Case 1	Selective absorption(e)	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
Particulates, Uranium and Plutonium	HEPA filters	Same as Case 1	Same as Case 1	Same as Case 1	HEPA filters,(f) 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,(g) 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(a)	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Same as Case 1	Voloxidation,(h) 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 filter	Same as Case 1	Same as Case 1	Same as Case 1	HEPA filter, sand filter	Same as Case 1	Same as Case 1	Same as Case 4	Recycle, no release
Lab Ventilation	HEPA filter	Same as Case 1	Same as Case 1	Same as Case 1	HEPA filter, sand filter	Same as Case 1	Same as Case 1	Same as Case 4	HEPA filter, roof vents

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[a] Model nuclear fuel reprocessing plant has a nominal capacity of 1500 metric tons per year; reference fuel is 3.3% 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. No credit is taken for decay during processing. All gaseous and water vapor releases are discharged through a 100-meter stack.

[b] Figure 1.

[c] Figure 2.

[d] Figure 3.

[e] Figure 4.

[f] Figure 5.

[g] Figure 6.

[h] Figure 7.

[i] Figure 8.

[j] Figure 9.

[k] Decontamination 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
Fuel Processed[a]						
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
Liquid Effluents[b]						
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
Gaseous Effluents[b]						
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

[a] Radionuclide 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 ^{85}Kr processed with the ^{85}Kr that was measured in the gaseous effluent.

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

[c] Percent of 10 CFR 20 concentration guides in Cattaraugus Creek.

[d] Percent of release limits imposed by AEC operating license.

Table 5.1. Solid Radioactive Waste[a]

Waste Type	Radioactivity Level Beta-Gamma	 Alpha	Annual Amounts (ft^3)
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

[a] 1500 metric tons of fuel per year. Data taken from ref. 2.

[b] Not compacted.

[c] The high beta-gamma activity is due primarily to the neutron-induced activity in the fuel cladding and hardware.

[d] All 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
E. coli	2300 Most Prob. Colonies
Fecal *E. coli*	2300 Most Prob. Colonies
Fecal Streptococcus	230 Most Prob. Colonies

[a] Data taken from ref. 2.

Table 5.3. Estimated Average Nonradioactive Gaseous Effluents

	H_2O (lb/hr)	NO_2 (mg/sec)	SO_2 (mg/sec)	CO (mg/sec)
Main	14,000	13,000	nil	nil
Service concentrator	4,560	nil	nil	nil
Cold off-gas	nil	580	nil	nil
Process boilers (gas)[a]	13,900	2,280	4.1	270
Process boilers (oil)[a]	9,800	6,270	3,290	270
Administration boiler (oil)	70	73.8	123	0.62
Furnaces (oil)	62	56.1	92	0.45

[a] Normal 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)
			Total Cost - Base Plant			
1[e]	125,000	32,500	13,000	45,500	30.33	0.12
			Additional Cost for Radwaste Treatment Systems			
2a	753 (125,753)	196 (32,696)	78 (13,078)	274 (45,774)	0.18 (30.51)	0.000711 (0.120711)
2b	2,071 (127,071)	538 (33,038)	215 (13,215)	753 (46,253)	0.50 (30.83)	0.00196 (0.12196)
3	3,871 (128,871)	1,006 (33,506)	403 (13,403)	1,409 (46,909)	0.94 (31.27)	0.00366 (0.12366)
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	10,085 (135,085)	2,622 (35,122)	1,049 (14,049)	3,671 (49,171)	2.45 (32.78)	0.00954 (0.12954)
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)

[a] System and structure capital cost consists of direct and indirect cost. The interest during construction is included as an indirect cost.

[b] Annual 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%.

[c] The reprocessing cost equals the annual cost divided by the 1.5 x 10^6 kg per year of uranium charged to the reactor.

[d] The 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.

[e] Case 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 2b through 6. The numbers in parentheses are total cost, i.e., base case plus added radwaste treatment cost.

[f] Case 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	3	4	5	6	6c	7
Capital cost,[a] $1000	(125,000)	753	2,071 (1,318)[b]	3,871	1,059	340	2,744	10,085	125,000
Annual cost,[a] $1000	(45,500)	274	753 (479)[b]	1,409	303	123	998	3,671	45,500
Calculated radioactive materials released,[c] Ci/year									
H-3	1.0×10^6	1.0×10^6	1.0×10^6	1.0×10^6	1.0×10^6	1.0×10^6	1.0×10^4	1.0×10^4	1.0×10^3
Kr-85	1.6×10^7	1.6×10^7	1.6×10^7	1.6×10^5	1.6×10^7	1.6×10^7	1.6×10^7	1.6×10^5	1.6×10^3
I-129	0.76	0.076	0.0057	0.76	0.76	0.76	0.76	0.0057	1.1×10^{-5}
I-131	1.9	0.19	0.14	1.9	1.9	1.9	1.9	0.14	2.8×10^{-4}
Semivolatiles	7.1	7.1	7.1	7.1	7.1	0.71	7.1	0.71	7.1×10^{-4}
Uranium	4.2×10^{-5}	4.2×10^{-5}	4.2×10^{-5}	4.2×10^{-5}	4.2×10^{-6}	4.2×10^{-5}	4.2×10^{-5}	4.2×10^{-6}	2.0×10^{-9}
Plutonium	0.8	0.8	0.8	0.8	0.08	0.8	0.8	0.08	1.6×10^{-4}
Particulates	7.2	7.2	7.2	7.2	0.72	7.2	7.2	0.72	3.5×10^{-3}

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[a] The cost is in addition to the base Case 1; Case 2b includes Case 2a.

[b] The difference in cost between Cases 2a and 2b.

[c] The 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	Iodine evolution and mercury recycle	123	306
2b	Iodox	147	366
3	Selective absorption system	1,012[d]	2,519[d]
4	Sand filter	425	1,059
5	Caustic scrubber	94	235
6	Voloxidation	472	1,176
6c	Case 2a through Case 6	2,274	5,662

[a] Details of the cost estimates are presented in Appendix A.

[b] Cost for 1973. Direct cost includes purchase cost and complete installation cost.

[c] Capital costs are calculated by multiplying the direct cost by 2.49. Capital costs include direct costs and indirect costs.

[d] Includes the cost for a station for bottling the noble gases 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_3O_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

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

[b] The source for the characteristics of the spent BWR fuel was WASH-1082, p. 5-57.

[c] The 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° 76' 12"

Table 7.2. Representative Population Distribution at Successive Distances for Midwestern Site

	Radial Distance (miles)											
Sector	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)	<— 95 —>						96	<— 126 —>		<— 440 —>		

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[a] Standard deviation of the mean (total).

Table 7.3. Representative Population Distribution at Successive Distances for Coastal Plain Site

	Radial Distance (miles)											
Sector	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)	<——— 55 ———>						289	<——— 61 ———>		<——— 51 ———>		

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[a] Standard deviation of the mean (total).

Table 7 4 Summary of Annual Doses to Individuals[a] and Population[b] from Gaseous Effluent of a Model Fuel Reprocessing Plant

Site	Radwaste Treatment Case	Maximum Total Body Dose[c] (millirem)	Maximum Adult Organ Doses[c] (millirem)										Population Total Body Dose (man-rem)
			Thyroid	GI Tract	Bone	Liver	Kidney	Muscle	Lungs	Testes	Ovaries	Spleen	
Coastal	1	4.4	58.3	55.6	11.6	3.7	4.3	1.1	2.3	0.76	0.75	0.98	173.1
	2a	4.3	9.4	55.4	11.5	3.6	4.2	1.0	2.2	0.66	0.65	0.88	168.9
	2b	4.2	4.3	55.4	11.5	3.6	4.2	1.0	2.2	0.66	0.65	0.88	168.5
	3	3.6	58.3	55.6	11.6	3.7	4.3	1.1	2.3	0.76	0.75	0.98	135.2
	4	3.3	57.8	54.8	3.9	2.3	3.4	0.32	2.3	0.29	0.29	0.31	143.1
	5	4.1	58.2	6.6	11.0	3.6	3.1	1.0	2.1	0.68	0.67	0.90	167.4
	6	2.5	56.7	53.8	9.7	1.8	2.4	1.1	0.39	0.72	0.71	0.94	79.3
	6c	0.19	0.90	5.4	1.3	0.20	0.25	0.10	0.09	0.07	0.07	0.10	5.7
	7	0.003	0.003	0.004	0.006	0.003	0.002	0.0005	0.002	0.0004	0.0003	0.0005	0.11
Midwestern	1	7.9	103.2	98.4	20.5	6.6	7.6	1.9	4.1	1.4	1.3	1.7	485.5
	2a	7.8	16.4	98.4	20.5	6.4	7.4	1.8	3.9	1.2	1.2	1.6	472.9
	2b	7.7	8.0	98.4	20.5	6.4	7.4	1.8	3.9	1.2	1.2	1.6	471.8
	3	6.5	103.2	98.4	20.5	6.6	7.6	1.9	4.1	1.4	1.3	1.7	378.6
	4	5.9	102.4	97.0	6.4	4.0	6.1	0.56	4.0	0.50	0.50	0.54	414.1
	5	7.4	103.1	12.0	19.4	6.5	5.4	1.8	3.8	1.3	1.2	1.6	468.8
	6	4.5	100.5	97.7	17.1	3.2	4.2	1.8	0.71	1.3	1.2	1.6	222.0
	6c	0.33	1.6	9.5	2.4	0.22	0.33	0.18	0.15	0.13	0.12	0.16	14.2
	7	0.005	0.006	0.006	0.011	0.005	0.005	0.0009	0.004	0.0006	0.0006	0.0008	0.32

[a] Dose to individuals is at 805 meters (0.5 mile) and downwind of the prevailing wind direction. Doses at 1.5 miles are 0.65 and 0.37 times these values for the coastal and midwestern sites, respectively.

[b] Dose to the population is average total body dose to the population out to a distance of 55 miles.

[c] Average total body and organ doses at 0.5 mile are 0.70 and 0.37 times these values for the coastal and midwestern sites, respectively.

Table 7.5. Maximum Annual Doses to Individuals at 1.5 Miles[a] from a Model Fuel Reprocessing Plant

Site	Radwaste Treatment Case	Maximum Total Body Dose (millirem)	Organ Dose (millirem) Thyroid	Organ Dose (millirem) GI Tract	Organ Dose (millirem) Bone
Coastal	1	2.9	37.9	36.1	7.5
	2a	2.8	6.0	36.0	7.5
	2b	2.8	2.8	36.0	7.5
	3	2.3	37.9	36.1	7.5
	4	2.1	37.6	35.6	2.5
	5	2.7	37.8	4.3	7.2
	6	1.6	36.8	34.9	6.3
	6c	0.12	0.6	3.5	0.86
	7	0.002	0.002	0.003	0.004
Midwestern	1	2.9	38.2	36.4	7.6
	2a	2.9	6.1	36.4	7.6
	2b	2.8	3.0	36.4	7.6
	3	2.4	38.2	36.4	7.6
	4	2.2	37.9	35.6	2.4
	5	2.7	38.1	4.4	7.2
	6	1.7	37.2	36.1	6.3
	6c	0.12	0.6	3.5	0.9
	7	0.002	0.002	0.002	0.004

[a] Individuals are assumed to be located in the prevailing wind direction.

Table 7.6. Contribution of Exposure Modes to Total Body Dose from the Gaseous Effluent of a Fuel Reprocessing Plant[a]

Exposure Mode	Annual Dose (millirem)	Percent of Total Dose
Submersion in air	0.74	16.7
Contaminated ground	0.94	21.3
Inhalation	1.99	45.0
Ingestion	0.75	17.0

[a] Maximum total body dose at 0.5 mile, coastal site treatment Case 1.

Table 7.7. Average Annual Total Body Dose (millirem) to Individuals from Gaseous Effluents as a Function of Distance from a Model Fuel Reprocessing Plant[a] on a Coastal or a Midwestern Site

Distance (miles)	Site	
	Coastal	Midwestern
0-0.5	3.10E+00	2.92E+00
0-1	2.70E+00	1.35E+00
0-2	1.73E+00	8.2E-01
0-3	1.43E+00	6.1E-01
0-4	1.20E+00	4.8E-01
0-5	1.05E+00	4.1E-01
0-10	6.4E-01	1.7E-01
0-15	4.4E-01	1.4E-01
0-25	2.5E-01	7.8E-02
0-35	1.7E-01	5.7E-02
0-45	1.3E-01	3.7E-02
0-55	1.0E-01	3.0E-02

[a] Treatment Case 1.

Table 7.8. Cumulative Population and Dose[a] (man-rem) as a Function of Distance from a Model Fuel Reprocessing Plant on a Coastal or a Midwestern Site

Distance		Coastal Site		Midwestern Site	
(meters)	(miles)	Population	Dose	Population	Dose
805	0.5	0	0	0	0
1,609	1	1,112	2.8	260	0.60
3,218	2	1,112	2.8	406	0.78
4,829	3	1,298	3.1	871	2.1
6,436	4	2,940	5.4	3,371	5.0
8,045	5	4,325	7.3	7,488	11.5
16,090	10	68,080	64.6	30,130	32.7
24,135	15	124,900	90.7	71,560	51.8
40,225	25	169,300	106.4	238,900	97.8
56,315	35	305,700	136.2	1,088,000	229.4
72,405	45	429,100	157.6	2,412,000	385.9
88,495	55	552,974	173.1	3,562,135	485.5

[a] Total body dose, treatment Case 1.

Table 7.9. Major Contributors to Total Body Dose to Individuals at 0.5 Mile from a Model Fuel Reprocessing Plant[a]

Radionuclide	Percent of Total Body Dose
^{3}H	43.3
$^{134-137}$Cs	21.3
^{85}Kr	17.5
^{106}Ru	6.2
^{129}I	4.4
^{90}Sr	1.7
Others[b]	<1.0

[a] Treatment Case 1.

[b] Iodine-131 is ~0.4%.

Table 7.10. Percent Contribution of Radionuclides to Individual Total Body Dose from the Gaseous Effluent of a Model Fuel Reprocessing Plant[a]

	Exposure Pathway			
Radionuclide	Submersion	Contaminated Ground	Ingestion	Inhalation
^{3}H			6.0	93.8
^{85}Kr	99.9	4.3		
^{90}Sr			9.6	
^{106}Ru		20.2	11.6	
^{129}I		11.7	11.4	
^{134}Cs		36.4	44.8	
^{137}Cs		17.3	14.3	
^{154}Eu		3.5		
^{238}Pu				2.6
^{103}Ru		1.1		
^{241}Pu				1.7
^{95}Zr		1.7		
^{95}Nb		1.8		
Others (<1.0)				

[a] At a distance of 0.5 mile for treatment Case 1.

Table 7.11. Percent Contribution of Ingested Radionuclides from the Gaseous Effluent of a Fuel Reprocessing Plant to Individual Organ Dose[a]

	Organ Dose (millirem)									
Radionuclide	Thyroid	GI Tract	Bone	Liver	Spleen	Kidney	Muscle	Lungs	Testes	Ovaries
^{3}H								11.1		
^{90}Sr			72.0	5.9	7.4		6.6	17.1	9.4	9.6
^{106}Ru		97.8	13.6	7.2	8.9	69.9	8.0	21.5	11.4	11.6
^{129}I	89.2			7.0	8.8		7.9	21.1	11.3	11.4
^{131}I	7.0									
^{134}Cs				51.6	45.4	10.7	53.5	16.6	44.1	44.8
^{137}Cs				22.5	22.3	5.2	18.3	7.6	14.1	14.3
Others (<5.0)										

[a] Treatment Case 1, distance of 0.5 mile.

Table 7.12. Pathways Leading to Maximum Annual Adult Thyroid Dose from ^{129}I[a]

Source	Daily Intake[b] (μCi)	Dose[c] (millirem)	Percent of Total Dose
Food crops	3.23E-06	6.4	6.8
Milk	3.42E-05	67.7	71.6
Beef	1.03E-05	20.4	21.7
Inhalation	6.24E-08	0.1	0.001
Total		94.6	

[a] Distance is 0.5 mile in the prevailing wind direction from the midwestern plant site (Case 1).

[b] Daily intake assumed to be 0.25 kg of vegetables, 1 liter of milk, 0.3 kg of meat, and 20 m^3 of air.

[c] Dose rate factor for ingestion is 5.428 rem/μCi intake; for inhalation, 4.16 rem/μCi intake. Intake is for 365 days.

Table 7.13. Pathways Leading to Maximum Annual Adult Thyroid Dose from ^{131}I[a]

Source	Daily Intake[b] (μCi)	Dose[c] (millirem)	Percent of Total Dose
Food crops	2.91E-06	2.0	23.0
Milk	8.98E-06	6.3	72.4
Beef	4.01E-07	0.3	3.4
Inhalation	1.49E-07	0.1	1.1
Total		8.7	

[a] Distance is 0.5 mile in the prevailing wind direction from the midwestern plant site (Case 1).

[b] Daily intake assumed to be 0.25 kg of vegetables, 1 liter of milk, 0.3 kg of meat, and 20 m^3 of air.

[c] Dose rate factor for ingestion is 1.922 rem/μCi intake; for inhalation, 1.473 rem/μCi intake. Intake is for 365 days.

Table 7.14. Average Annual Thyroid Dose (millirem) to Individuals Around a Model Fuel Reprocessing Plant[a]

Distance (miles)	Exposure Pathway: Inhalation	Exposure Pathway: Ingestion	Total Dose
	Coastal Site		
0.5	1.20	39.6	40.8
10.0	0.31	8.6	8.9
55.0	0.06	0.42	0.48
	Midwestern Site		
0.5	1.13	37.1	38.2
10.0	0.09	2.4	2.5
55.0	0.02	0.13	0.15

[a] Treatment Case 1.

Table 7.15. Pathways Leading to Maximum Annual Thyroid Doses from ^{129}I and ^{131}I for the 1-year-old Child[a]

Source[b]	^{129}I Dose (mrem)	^{129}I Percent of Total Dose	^{131}I Dose (mrem)	^{131}I Percent of Total Dose
Food Crops	1.24	0.91	1.62	3.1
Milk	131.1	96.2	51.0	96.2
Beef	3.95	2.9	0.24	0.45
Inhalation	0.05	<0.01	0.12	0.23
Total	136.3		53.0	

[a] Distance is 0.5 mile in the prevailing wind direction from the midwestern plant site (Case 1).

[b] Daily intake assumed to be 0.025 kg of vegetables, 1 liter of milk, 0.03 kg of meat, and 5.6 m^3 of air.

Table 7.16. Annual Total Body Dose (millirem) from Aquatic Pathways

Exposure Pathway	Coastal Estuary	Midwestern River
Submersion in water[a]	4.7E-04	4.3E-04
Eating fish[b]	1.3E-02	1.5E-01
Drinking water[c]	-	3.1E-02

[a] Swimming for 1% of the year.

[b] Consumption of 20 g of fish per day.

[c] Consumption of 1.2 liters of water per day.

Table 7.17. Aquatic Pathways and Principal Radionuclides

Exposure Pathway	Radionuclide and Percent Contribution to Dose
Submersion in water	$^{134+137}$Cs (39.3), ^{3}H (24.7), ^{95}Nb (8.4)
Eating fish	$^{134+137}$Cs (95.0 for estuary, 99.5 for river)
Drinking water	$^{134+137}$Cs (71.0), ^{3}H (14.8), ^{90}Sr (8.9)

Table 7.18. Annual Dose to Biota Living in an Estuary[a] near a Fuel Reprocessing Plant[b] on a Coastal Site

Radionuclide	Concentration (μCi/ml)	Annual Dose (millirad)			
		Algae	Invertebrates	Fish	Waterfowl
^{3}H	2.0E-07	3.7E-02	3.7E-02	3.7E-02	3.7E-02
^{129}I	1.4E-10	2.4E+00	2.4E-02	4.8E-03	4.3E+01
^{131}I	3.5E-10	2.9E+01	2.9E-01	5.8E-02	2.2E+01
^{103}Ru	2.1E-10	1.7E+00	1.7E-01	5.1E-03	2.8E-02
^{106}Ru	1.1E-09	3.0E+01	3.0E+00	8.9E-02	9.2E-01
^{134}Cs	5.9E-10	1.2E-01	6.1E-01	3.7E-01	5.9E-01
^{137}Cs	3.0E-10	3.3E-02	1.6E-01	9.8E-02	2.1E-01
^{127m}Te	1.6E-11	9.5E-02	9.5E-03	9.5E-04	4.5E-02
^{129m}Te	6.6E-12	2.7E-02	1.3E-02	1.3E-03	3.7E-02
^{89}Sr	4.8E-11	9.9E-03	5.0E-04	5.0E-04	2.1E-02
^{90}Sr	4.3E-11	1.8E-02	8.9E-04	8.9E-04	5.4E-02
^{90}Y	4.3E-11	1.4E-02	7.2E-04	7.2E-04	5.6E-07
^{91}Y	8.3E-11	2.8E-01	9.2E-02	2.8E-02	2.3E-04
^{95}Zr	1.4E-10	2.8E+00	2.9E-01	8.8E-02	1.2E-03
^{95}Nb	2.7E-10	2.5E-01	5.1E-01	2.5E-01	4.3E-04
^{110m}Ag	1.4E-12	4.4E-02	2.2E-01	4.4E-02	1.5E-04
^{125}Sb	4.5E-12	3.0E-01	3.0E-02	3.0E-02	4.7E-02
^{141}Ce	2.6E-11	3.0E-02	1.0E-02	3.0E-03	1.1E-05
^{144}Ce	4.3E-11	3.1E-01	1.0E-01	3.1E-02	8.6E-04
^{147}Pm	5.5E-11	7.1E-02	7.1E-02	7.1E-03	3.9E-04
^{154}Eu	3.9E-12	9.4E-02	9.4E-02	9.4E-03	5.1E-04
^{155}Eu	3.6E-12	1.1E-02	1.1E-02	1.1E-03	2.9E-05
^{234}U	4.3E-15	2.6E-04	3.9E-05	3.9E-05	3.9E-07
^{235}U	9.5E-17	5.5E-06	8.2E-07	8.2E-07	8.2E-09
^{236}U	1.6E-15	9.4E-05	1.4E-05	1.4E-05	1.4E-07
^{238}U	1.8E-15	9.5E-05	1.4E-05	1.4E-05	1.4E-07
^{238}Pu	4.0E-12	1.5E+00	4.2E-01	1.5E-02	1.4E-01
^{239}Pu	4.6E-13	1.6E-01	4.5E-02	1.6E-03	1.5E-02
^{240}Pu	6.6E-13	2.3E-01	6.5E-02	2.3E-03	6.2E-02
^{241}Pu	1.4E-10	2.2E+00	6.2E-01	2.2E-02	9.6E-04
^{241}Am	8.9E-15	4.8E-02	9.5E-03	2.4E-04	1.2E-02
^{243}Am	1.0E-14	5.1E-02	1.0E-02	2.5E-04	1.5E-02
^{242}Cm	9.5E-12	7.1E+00	1.4E+01	3.6E-01	1.3E-02
^{244}Cm	1.4E-12	7.7E-01	1.5E+00	3.8E-02	5.7E-02
Total Dose		7.9E+01	2.3E+01	1.6E+00	6.7E+01

[a] Estuary is 1 mile long by 0.5 mile wide and located 0.5 mile from the plant in the direction of the prevailing wind.

[b] Treatment Case 1.

Table 7.19. Annual Dose to Biota Living in a Freshwater River[a] near a Fuel Reprocessing Plant[b] on a Midwestern Site

Radionuclide	Concentration (μCi/ml)	Annual Dose (millirad)			
		Algae	Invertebrates	Fish	Waterfow
^{3}H	1.7E-07	3.2E-02	3.2E-02	3.2E-02	3.2E-02
^{129}I	1.2E-10	8.2E-03	1.0E-03	3.1E-03	1.6E-01
^{131}I	3.0E-10	9.9E-02	1.2E-03	3.7E-02	7.4E-02
^{103}Ru	1.8E-10	3.0E+00	4.4E-01	1.5E-02	1.8E-01
^{106}Ru	9.8E-10	5.1E+01	7.7E+00	2.5E-01	1.6E+00
^{134}Cs	5.2E-10	5.3E+00	1.1E+00	4.2E+00	2.6E+01
^{137}Cs	2.6E-10	1.4E+00	2.8E-01	1.3E+00	2.2E+01
^{127m}Te	1.4E-11	8.2E-02	8.2E-03	8.3E-04	3.9E-02
^{129m}Te	5.7E-12	2.3E-02	1.2E-02	1.2E-03	3.2E-02
^{89}Sr	3.9E-11	2.0E-01	4.0E-02	2.0E-03	3.4E-01
^{90}Sr	3.7E-11	3.8E-01	7.7E-02	3.8E-03	9.2E-01
^{90}Y	3.7E-11	3.1E+00	6.2E-01	1.6E-02	1.2E-04
^{91}Y	7.2E-11	4.0E+00	7.9E-01	2.0E-02	3.3E-03
^{95}Zn	1.2E-10	2.5E+00	1.7E-02	8.4E-03	9.6E-04
^{95}Nb	2.2E-10	1.7E+00	2.1E-01	6.4E+00	2.9E-03
^{110m}Ag	1.2E-12	7.5E-03	2.9E-02	6.7E-05	2.6E-05
^{125}Sb	3.9E-12	3.7E-02	2.4E-04	2.4E-05	4.4E-03
^{141}Ce	2.2E-11	3.4E-01	8.6E-02	2.2E-03	1.3E-04
^{144}Ce	3.7E-11	3.6E+00	9.0E-01	2.2E-02	9.9E-03
^{142}Pm	4.8E-11	3.1E-01	6.2E-02	1.5E-03	1.7E-03
^{154}Eu	3.3E-12	4.1E-01	8.1E-02	2.0E-03	2.2E-03
^{155}Eu	3.1E-12	4.6E-03	9.2E-04	2.3E-05	1.3E-03
^{234}U	3.7E-15	1.7E-06	2.0E-04	3.4E-05	3.4E-02
^{235}U	8.2E-17	3.5E-08	4.2E-06	7.1E-07	7.1E-09
^{236}U	1.4E-15	6.1E-07	7.4E-05	1.2E-05	1.2E-07
^{238}U	1.5E-15	6.2E-07	7.4E-05	1.2E-05	1.2E-07
^{238}Pu	3.4E-12	1.3E00	3.7E-01	1.3E-02	1.2E-01
^{239}Pu	4.0E-13	1.4E-01	3.9E-02	1.4E-03	3.8E-02
^{240}Pu	5.7E-13	2.0E-01	5.6E-02	2.0E-03	5.3E-02
^{241}Pu	1.2E-10	1.8E00	5.3E-01	1.8E-02	8.3E-04
^{241}Am	7.7E-15	4.1E-03	8.2E-04	2.0E-05	1.1E-02
^{243}Am	8.7E-15	4.4E-02	8.3E-03	2.2E-04	1.3E-02
^{242}Cm	8.2E-12	6.1E+01	1.2E+01	3.1E-01	1.1E-01
^{244}Cm	1.2E-12	6.7E00	1.3E00	3.3E-02	5.0E-01
Total Dose		1.5E+02	2.7E+01	1.2E+01	5.2E+01

[a] River segment 1 mile long by 0.1 mile wide by 3 m deep, located 0.5 mile from the plant in the prevailing wind direction.

[b] Treatment Case 1.

Table 7.20. Typical Variability of Atmospheric X/Q' Values and Population Data at Midwestern and Coastal Sites

		Midwestern	Coastal
X/Q'[a]	Mean	2.67×10^{-8} sec·m^{-1}	2.93×10^{-8} sec·m^{-1}
	Standard deviation	1.24×10^{-8} sec·m^{-1}	0.77×10^{-8} sec·m^{-1}
	Coefficient of variation, CV	0.46	0.26
Population[b]	Mean	3.56×10^{6}	5.53×10^{5}
	Standard deviation	3.34×10^{6}	1.86×10^{5}
	Coefficient of variation, CV	0.93	0.33

[a] Based on maximum X/Q' values at 0.7 mile from point of release. Represents directional variability at a given distance.

[b] Based on cumulative population for area with a 55-mile radius.

Table 8.1. Cost-Benefit Analysis for Model Nuclear Fuel Reprocessing Plant at the Midwestern Site

Radwaste Treatment Case	Total Annual Cost[a] ($1000)	Reprocessing Cost[a] ($/kg U)	Maximum Annual Individual Dose at 0.5 Mile[b]			Average Population Total Body Dose out to a Distance of 55 Miles (man-rem)
			Total Body (millirem)	Thyroid (millirem)	Bone (millirem)	
1	0	0	7.9	103.2	20.5	485.5
2a	274	0.18	7.8 (0.1)	16.4 (86.8)	20.5 (0.0)	472.9 (12.6)
2b	753	0.50	7.7 (0.2)	8.0 (95.2)	20.5 (0.0)	471.8 (13.7)
3	1,409	0.94	6.5 (1.4)	103.2 (0.0)	20.5 (0.0)	378.6 (106.9)
4	303	0.20	5.9 (2.0)	102.4 (0.8)	6.4 (14.1)	414.1 (71.4)
5	123	0.08	7.4 (0.5)	103.1 (0.1)	19.4 (1.1)	468.8 (16.7)
6	998	0.67	4.5 (3.4)	100.5 (2.7)	17.1 (3.4)	222.0 (263.5)
6c	3,671	2.45	0.33 (7.57)	1.6 (101.6)	2.4 (18.1)	14.2 (471.3)
7	45,500	30.33	0.005 (7.895)	0.006 (103.194)	0.011 (20.489)	0.32 (485.18)

[a] Cost of the additional equipment required for each case with reference to Case 1.

[b] The number in parentheses is the reduction in dose with reference to Case 1.

Table 8.2. Cost-Benefit Analysis for Model Nuclear Fuel Reprocessing Plant at the Coastal Site

Radwaste Treatment Case	Total Annual Cost[a] ($1000)	Reprocessing Cost[a] ($/kg U)	Maximum Annual Individual Dose at 0.5 Mile[b]			Average Population Total Body Dose out to a Distance of 55 Miles (man-rem)
			Total Body (millirem)	Thyroid (millirem)	Bone (millirem)	
1	0	0	4.4	58.3	11.6	173.1
2a	274	0.18	4.3 (0.1)	9.3 (49.0)	11.5 (0.1)	168.9 (4.2)
2b	753	0.50	4.2 (0.2)	4.3 (54.0)	11.5 (0.1)	168.5 (4.6)
3	1,409	0.94	3.6 (0.8)	58.3 (0.0)	11.6 (0.0)	135.2 (37.9)
4	303	0.20	3.3 (1.1)	57.8 (0.5)	3.9 (7.7)	143.1 (30.0)
5	123	0.08	4.1 (0.3)	58.2 (0.1)	11.0 (0.6)	167.4 (5.7)
6	998	0.67	2.5 (1.9)	56.7 (1.6)	9.7 (1.9)	79.3 (93.8)
6c	3,671	2.45	0.19 (4.21)	0.9 (57.4)	1.3 (10.3)	5.7 (167.4)
7	45,500	30.33	0.003 (4.397)	0.003 (58.297)	0.006 (11.594)	0.11 (172.99)

[a] Cost of additional equipment required for each case with reference to Case 1.

[b] The number in parentheses is the reduction in dose with reference to Case 1.

Table 8.3. Cumulative Annual and Reprocessing Costs for Reduction of Total Body Dose from Gaseous Effluent of the Model Fuel Reprocessing Plant at the Coastal and Midwestern Sites

Radwaste Treatment	Cumulative Annual Cost ($1000)	Cumulative Reprocessing Cost ($/kg U)[a]	Maximum Individual Total Body Dose		Incremental Reduction in Dose		Cost-Benefit Ratio, Annual Cost/Incremental Reduction in Dose	
			Coastal Site (millirem)	Midwestern Site (millirem)	Coastal Site (millirem)	Midwestern Site (millirem)	Coastal Site ($$10^6$/millirem)	Midwestern Site ($$10^6$/millirem)
Case 1	0	0	4.4	7.9	0	0	0	0
Add Case 4[b]	303	0.20	3.3	5.9	1.1	2.0	0.28	0.15
Add Case 5	426	0.28	3.0	5.4	0.3	0.5	0.41	0.25
Add Case 6	1,424	0.95	1.1	2.0	1.9	3.4	0.53	0.29
Add Case 3	2,833	1.89	0.3	0.6	0.8	1.4	1.76	1.0
Add Case 2a	3,107	2.07	0.2	0.5	0.1	0.1	2.74	2.74
Add Case 2b[c]	3,671	2.45	0.19	0.33	0.2	0.2	3.77	3.77
Case 6c	3,671	2.45	0.19	0.33	4.21	7.57	0.87	0.48
Case 7	45,500	30.33	0.003	0.005	4.397	7.895	10.3	5.76

[a] The reprocessing cost is expressed as dollars per kilogram of uranium charged to the reactor, i.e., the annual cost/1500 metric tons per year.

[b] Cases 2a through 6 are added to the base case in the order of increasing the annual cost of reducing the dose.

[c] Case 2b includes 2a.

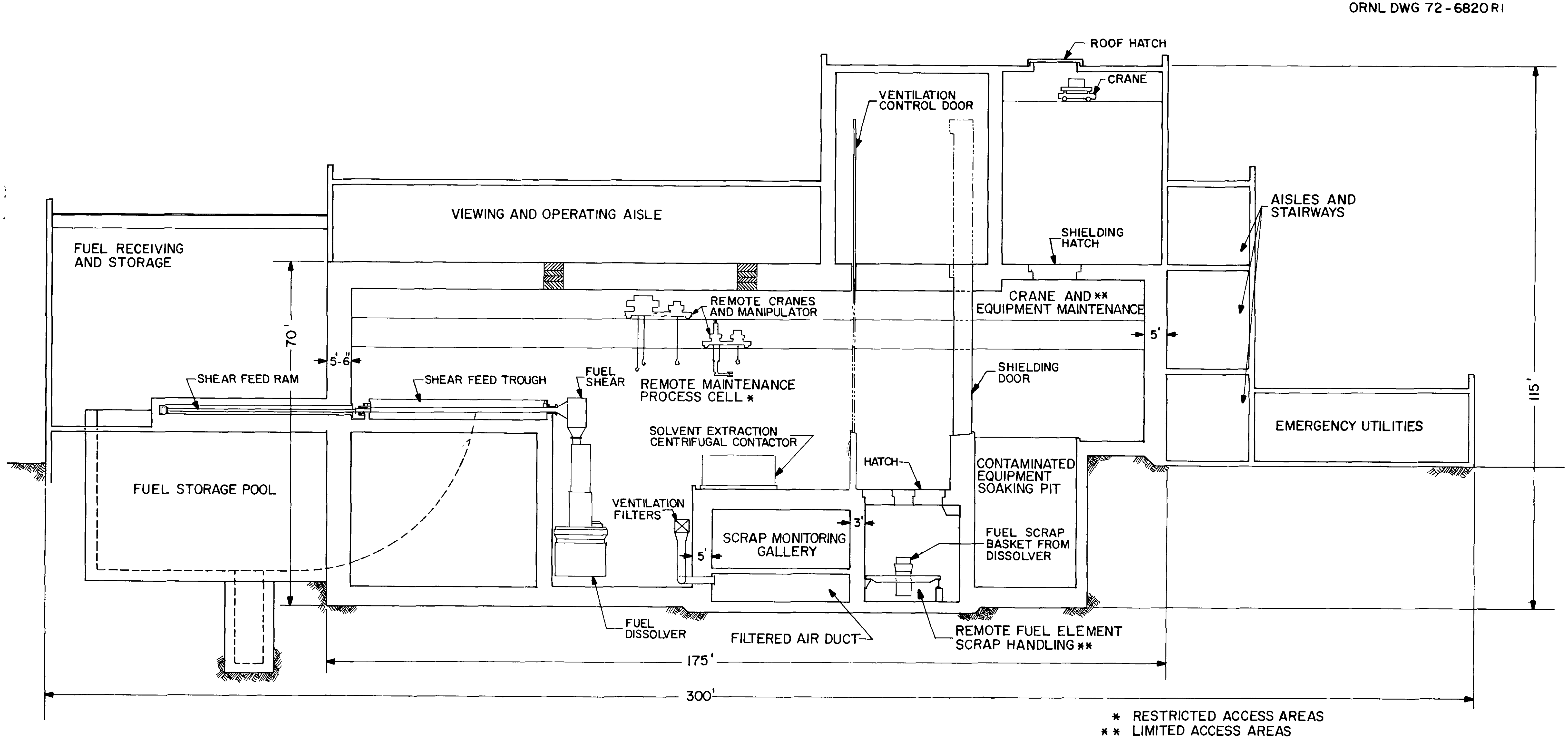


Fig. 4.1. Longitudinal Cross Section of a Typical Fuel Reprocessing Plant. (Ref. 1, Sect. 4.6)

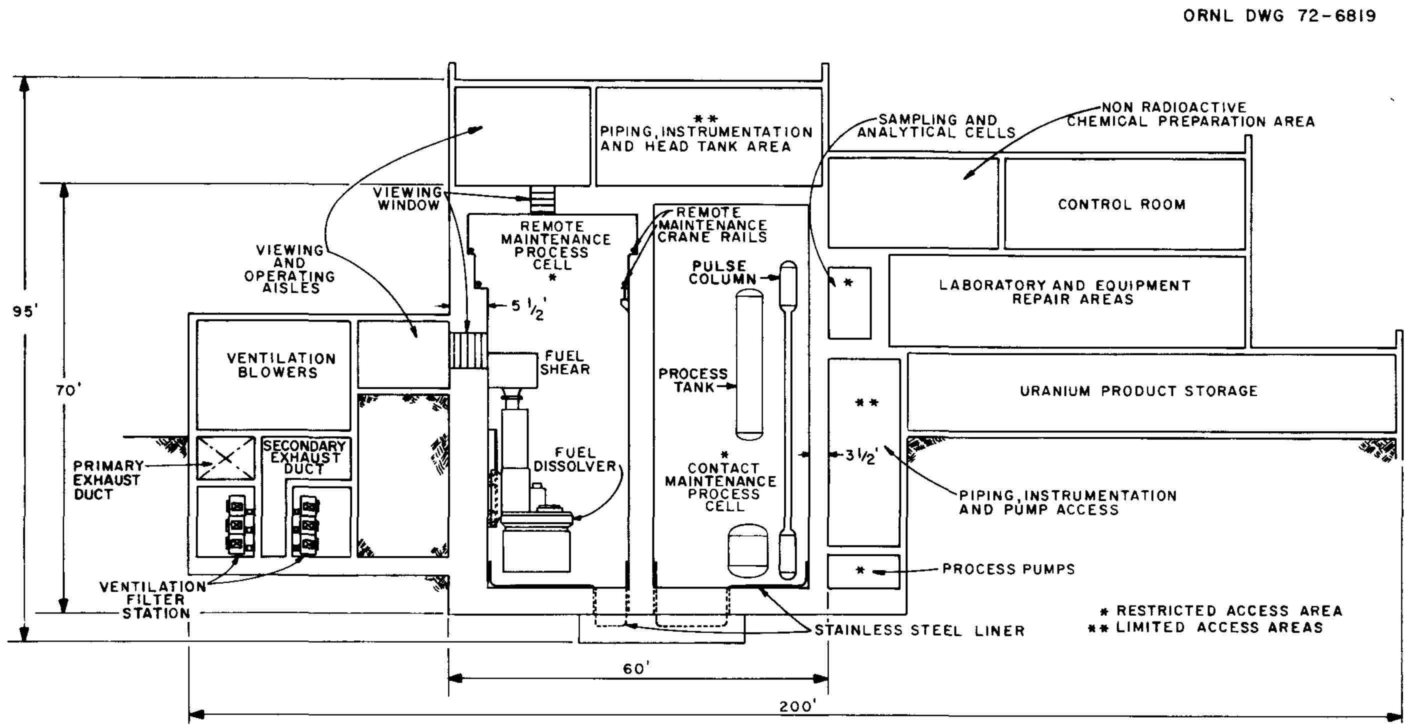
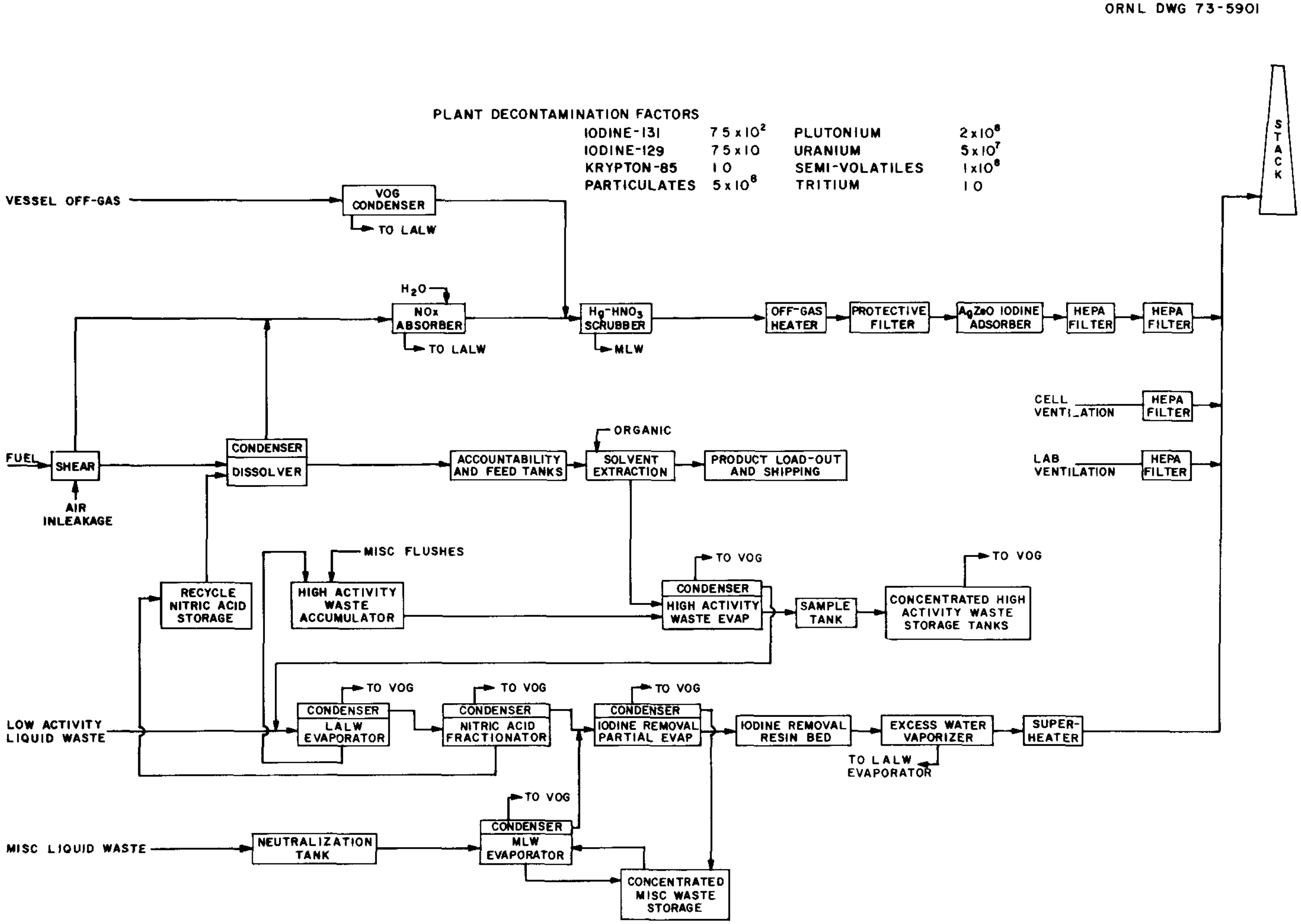


Fig. 4.2. Cross Section of a Typical Fuel Reprocessing Plant. (Ref. 1, Sect. 4.6)



PLANT DECONTAMINATION FACTORS

IODINE-131	7 5 x 10^2	PLUTONIUM	2 x 10^8
IODINE-129	7 5 x 10	URANIUM	5 x 10^7
KRYPTON-85	1 0	SEMI-VOLATILES	1 x 10^6
PARTICULATES	5 x 10^8	TRITIUM	1 0

Fig. 4.3. Model Fuel Reprocessing Plant: Case 1 - Base Case, Current Practice.

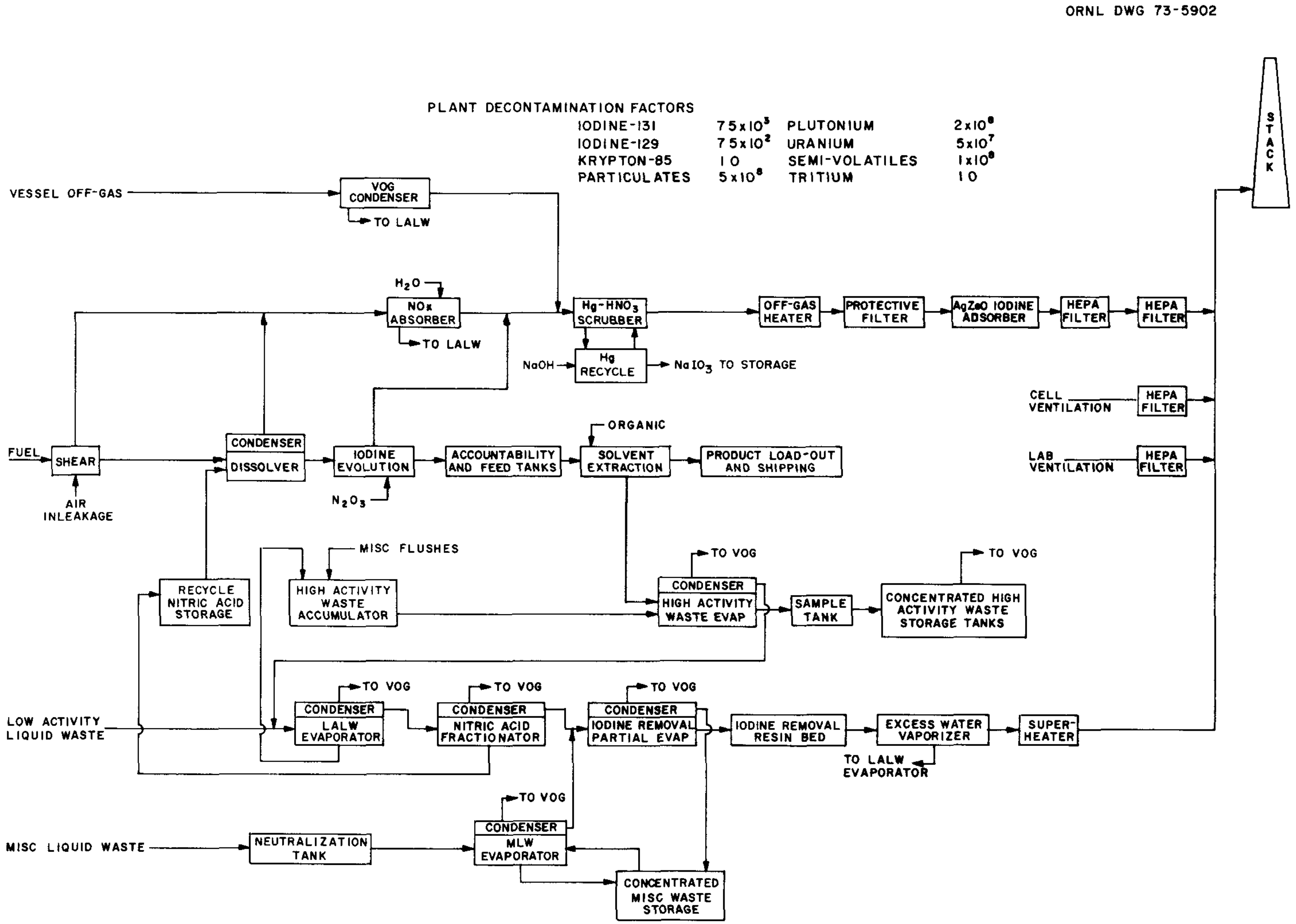


Fig. 4.4. Model Fuel Reprocessing Plant: Case 2a - Reduction in Release of Iodine.

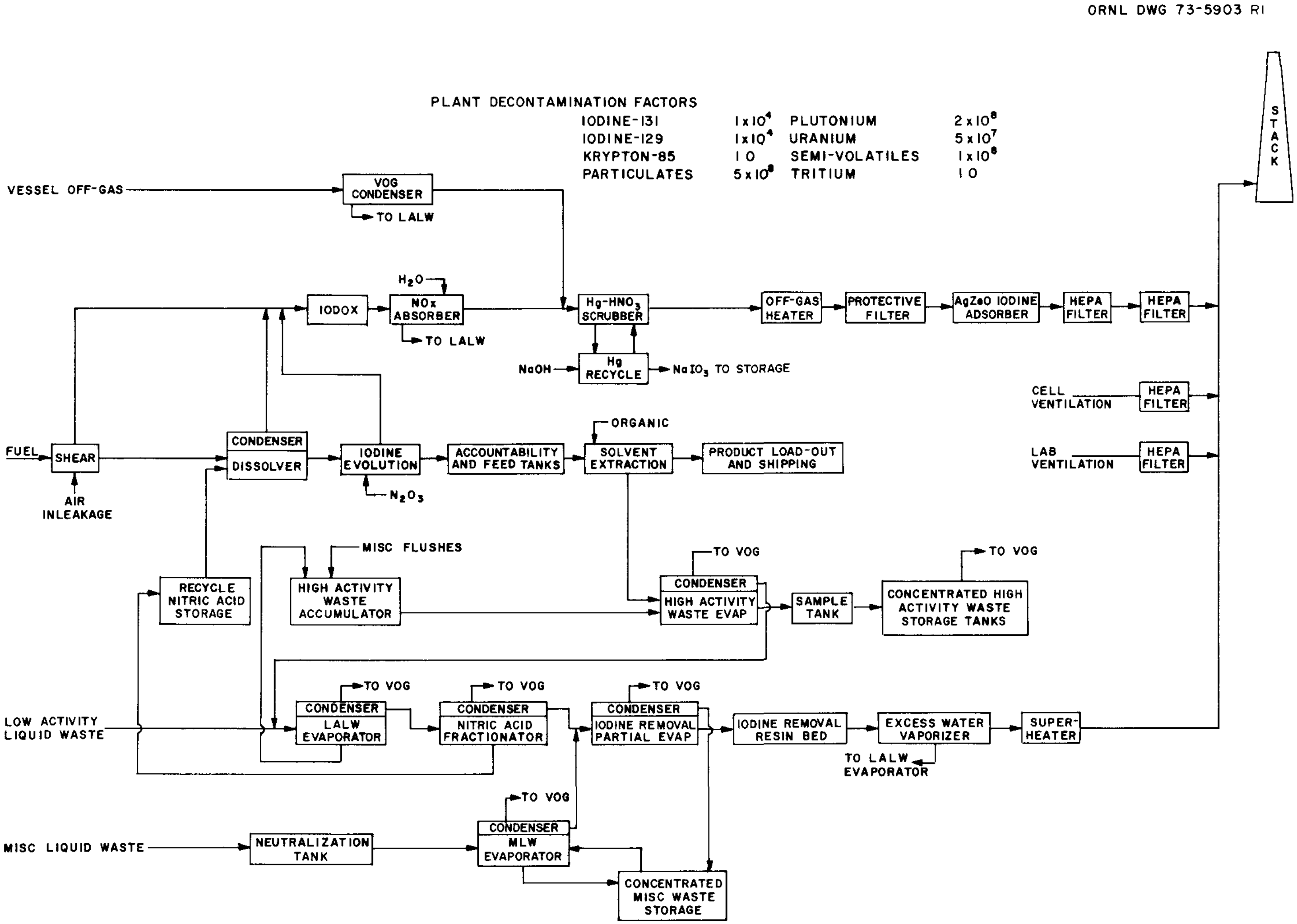


Fig. 4.5. Model Fuel Reprocessing Plant: Case 2b - Further Reduction in Release of Iodine.

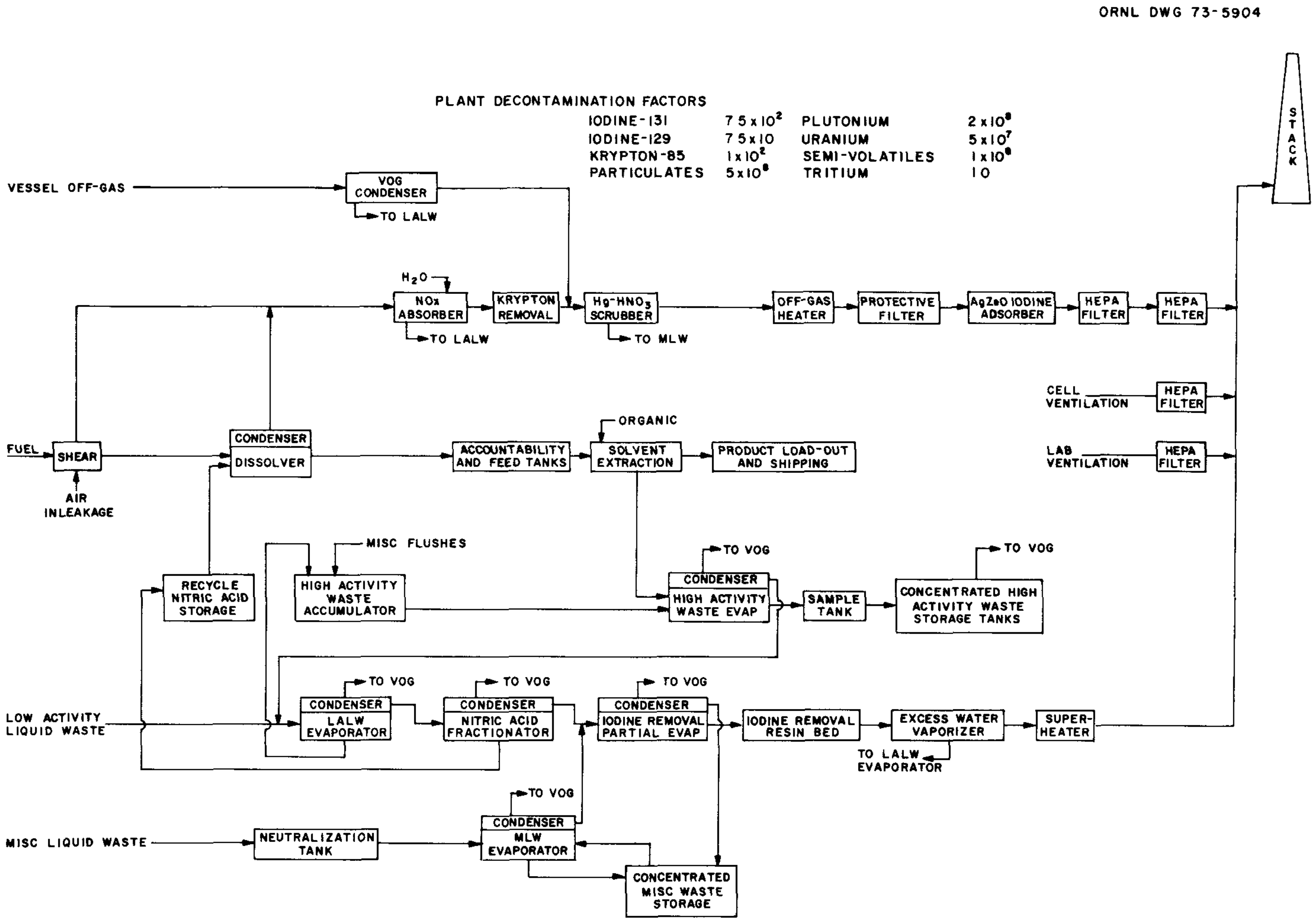


Fig. 4.6. Model Fuel Reprocessing Plant: Case 3 - Reduction in Release of Krypton.

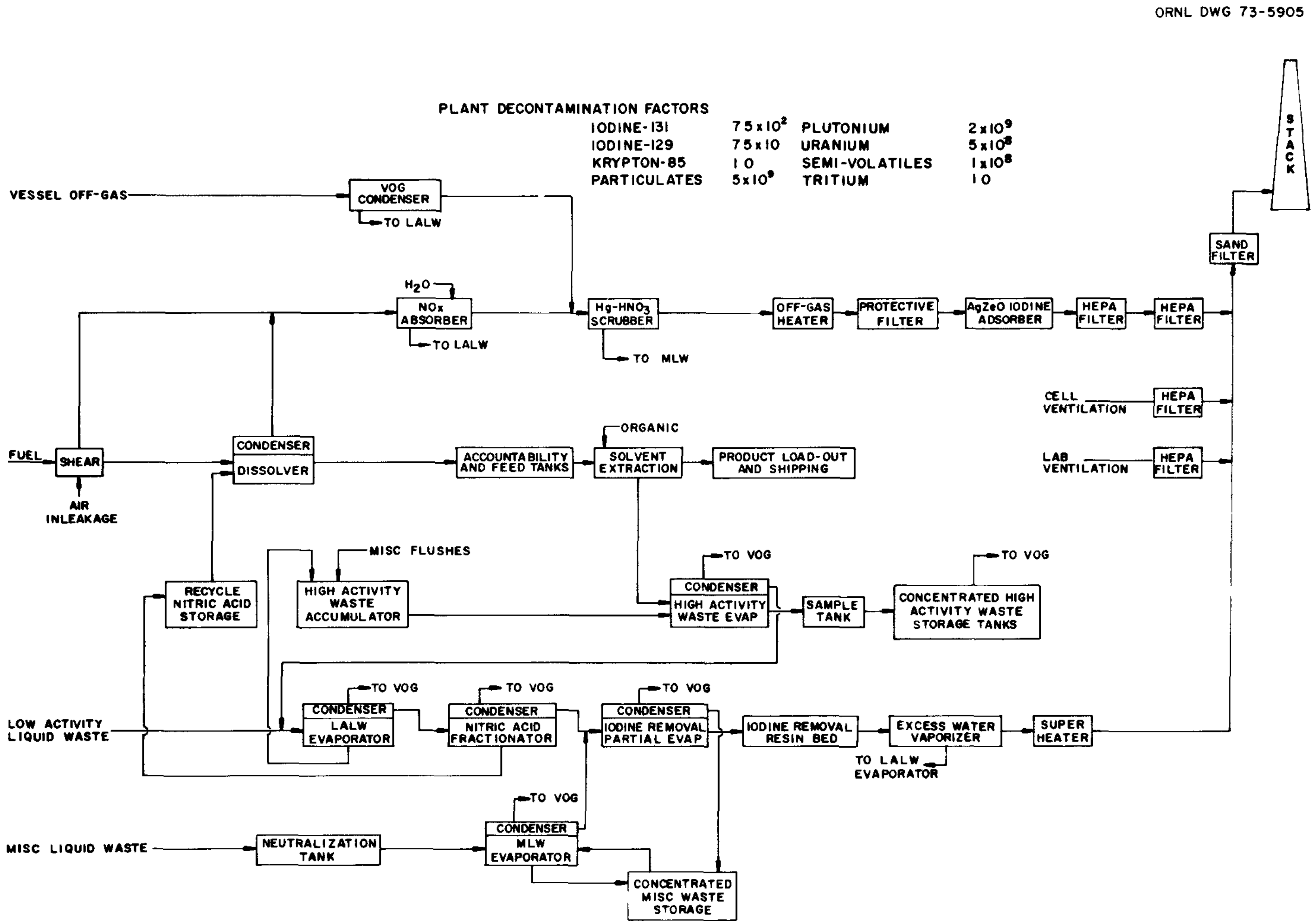


Fig. 4.7. Model Fuel Reprocessing Plant: Case 4 - Reduction in Release of Particulates.

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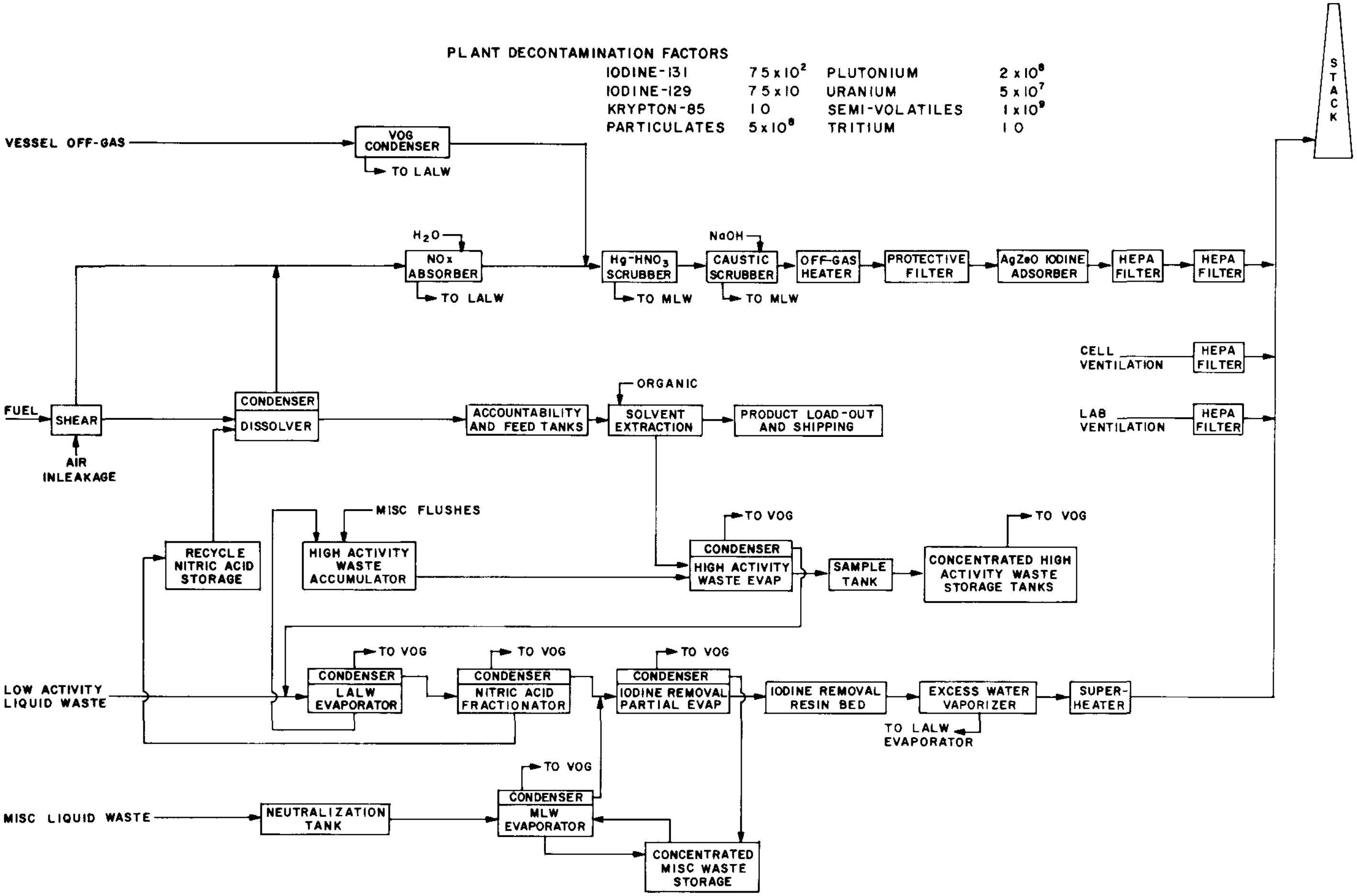

PLANT DECONTAMINATION FACTORS

IODINE-131	7 5 x 10^2	PLUTONIUM	2 x 10^6
IODINE-129	7 5 x 10	URANIUM	5 x 10^7
KRYPTON-85	1 0	SEMI-VOLATILES	1 x 10^9
PARTICULATES	5 x 10^8	TRITIUM	1 0

Fig. 4.8. Model Fuel Reprocessing Plant: Case 5 - Reduction in Release of Semivolatiles.

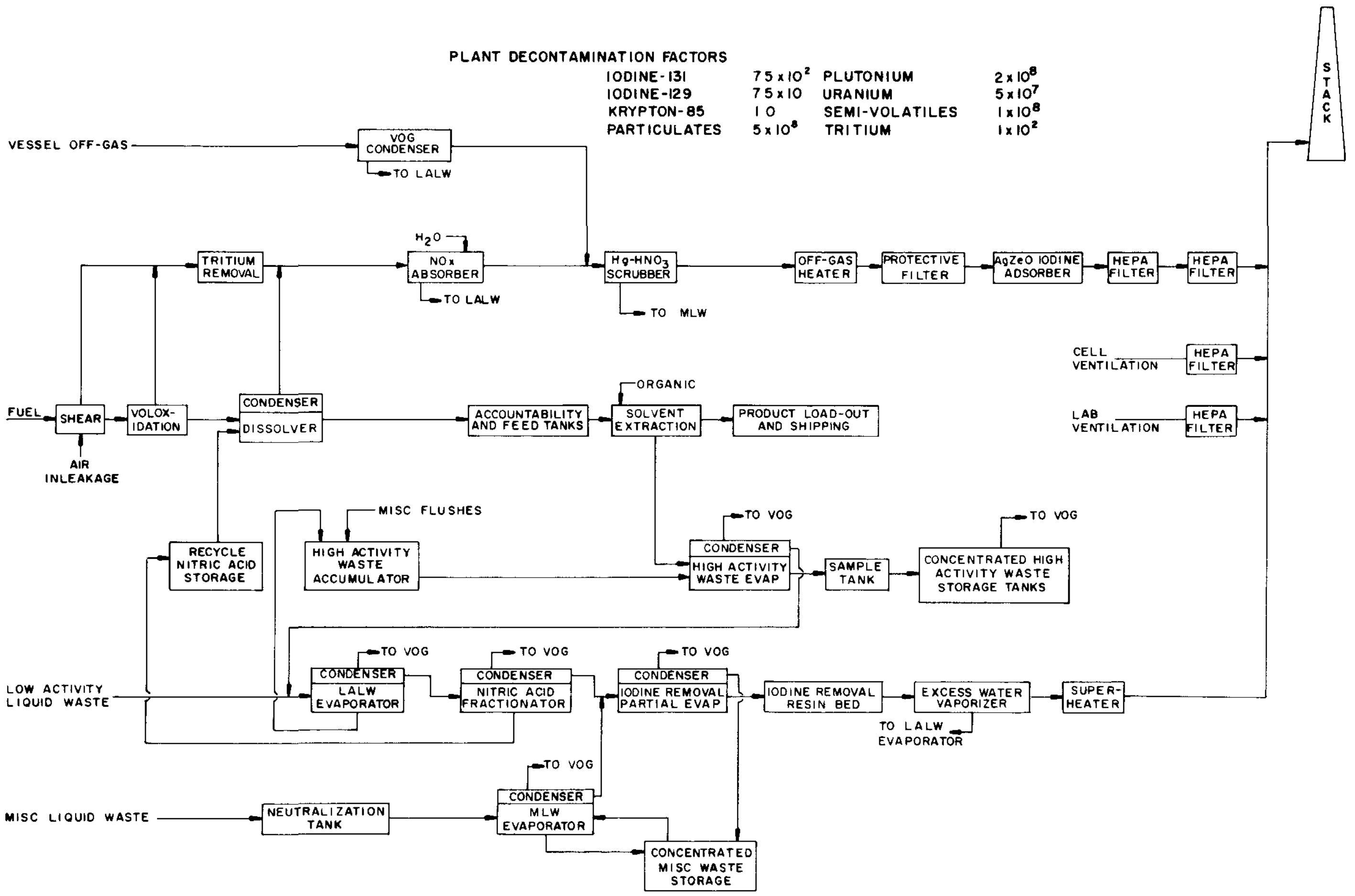

Fig. 4.9. Model Fuel Reprocessing Plant: Case 6 - Reduction in Release of Tritium.

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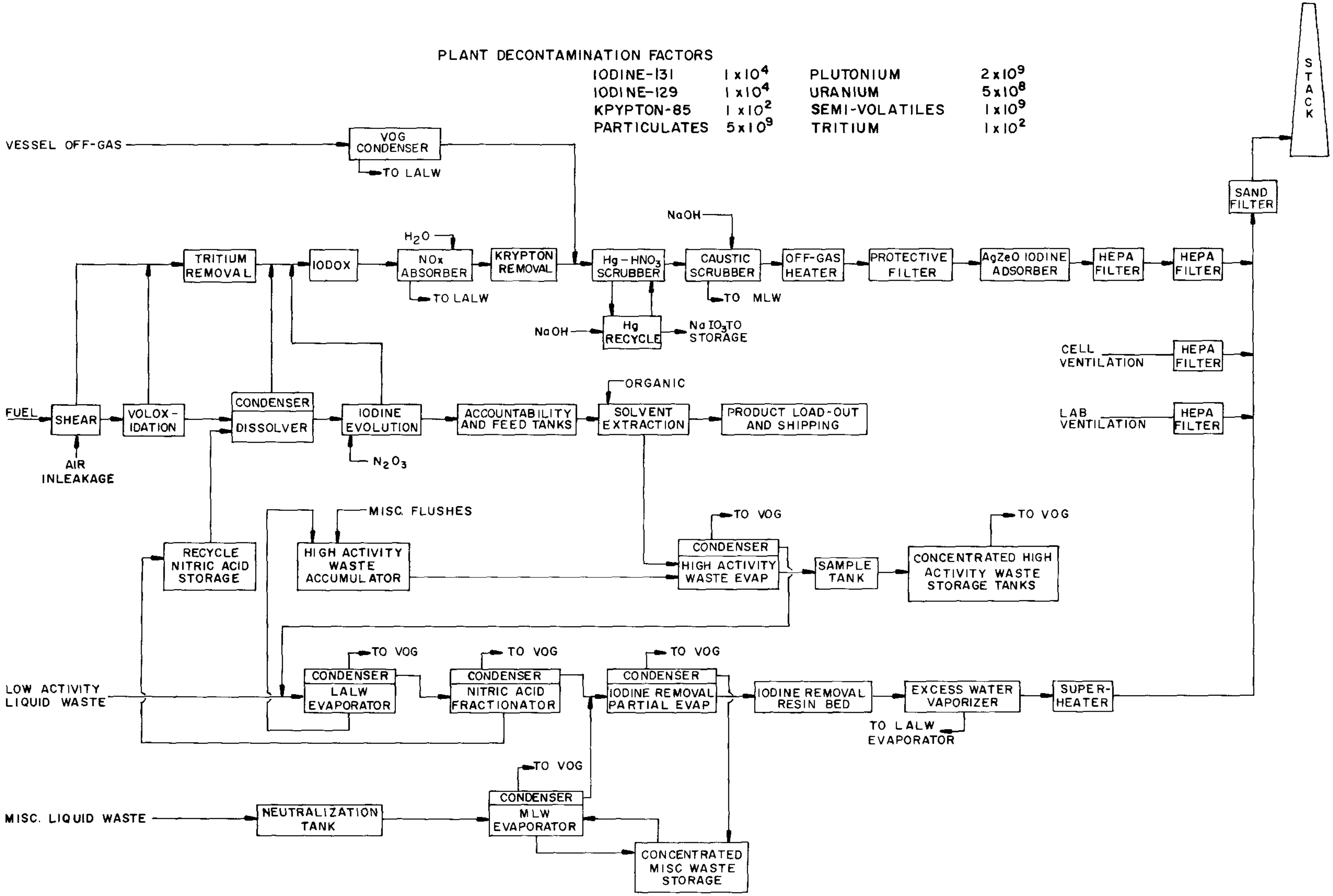

Fig. 4.10. Model Fuel Reprocessing Plant: Case 6c - Cumulative Reduction in Releases for Cases 2a to 6.

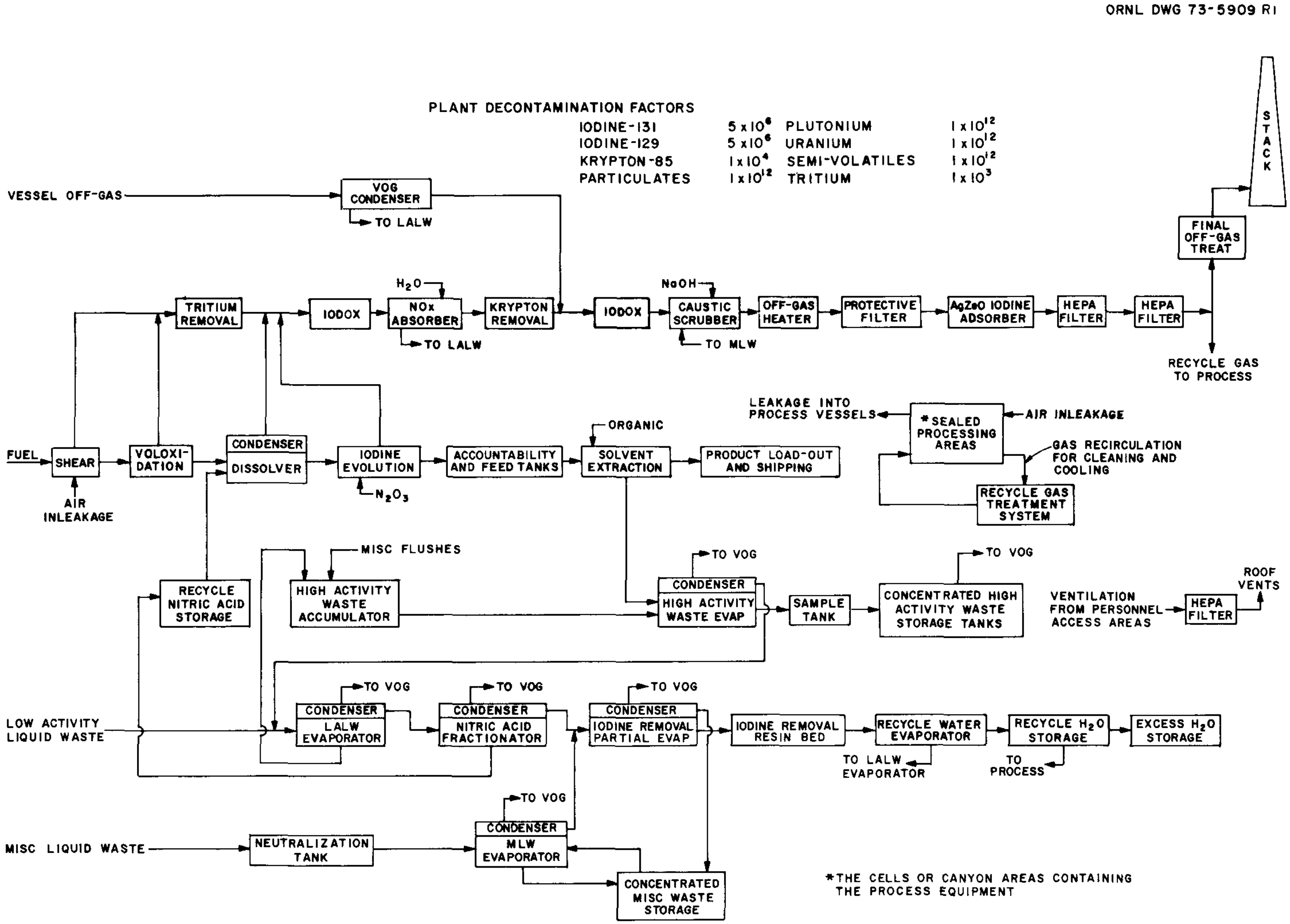

Fig. 4.11. Model Fuel Reprocessing Plant: Case 7 - Minimum Release of Radioactive Materials.

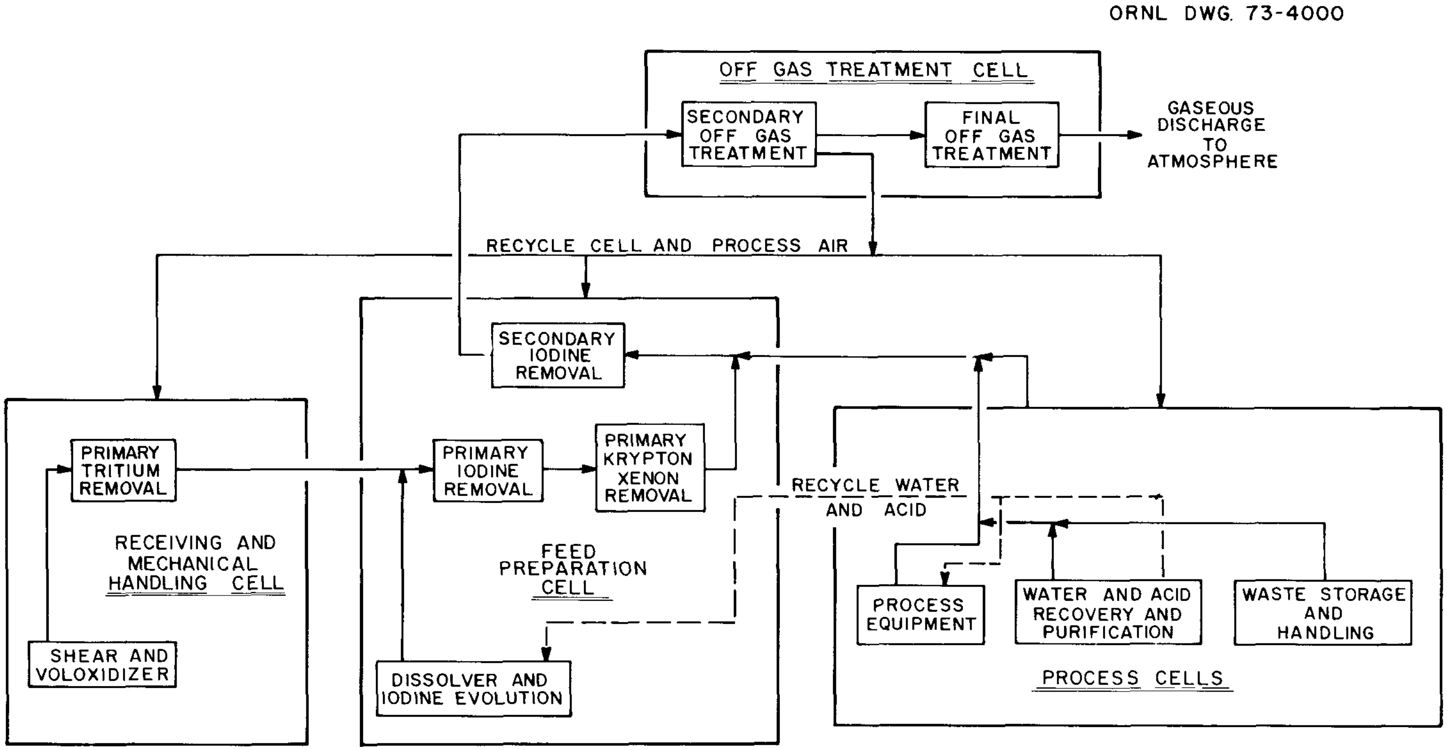

Fig. 4.12. Model Fuel Reprocessing Plant: Case 7 - Minimum Release of Radioactive Materials; Flowsheet for Control and Treatment of Radwaste Effluents.

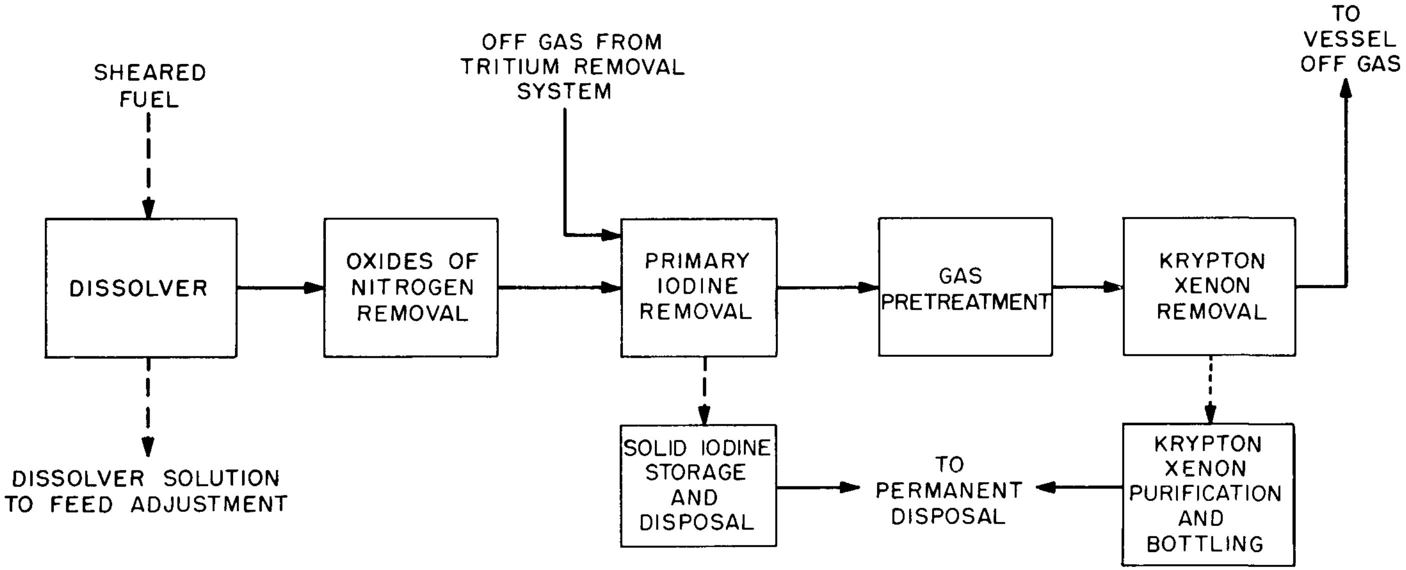

Fig. 4.13. Model Fuel Reprocessing Plant: Case 7 - Minimum Release of Radioactive Materials; Flowsheet for Retention of Krypton and Iodine.

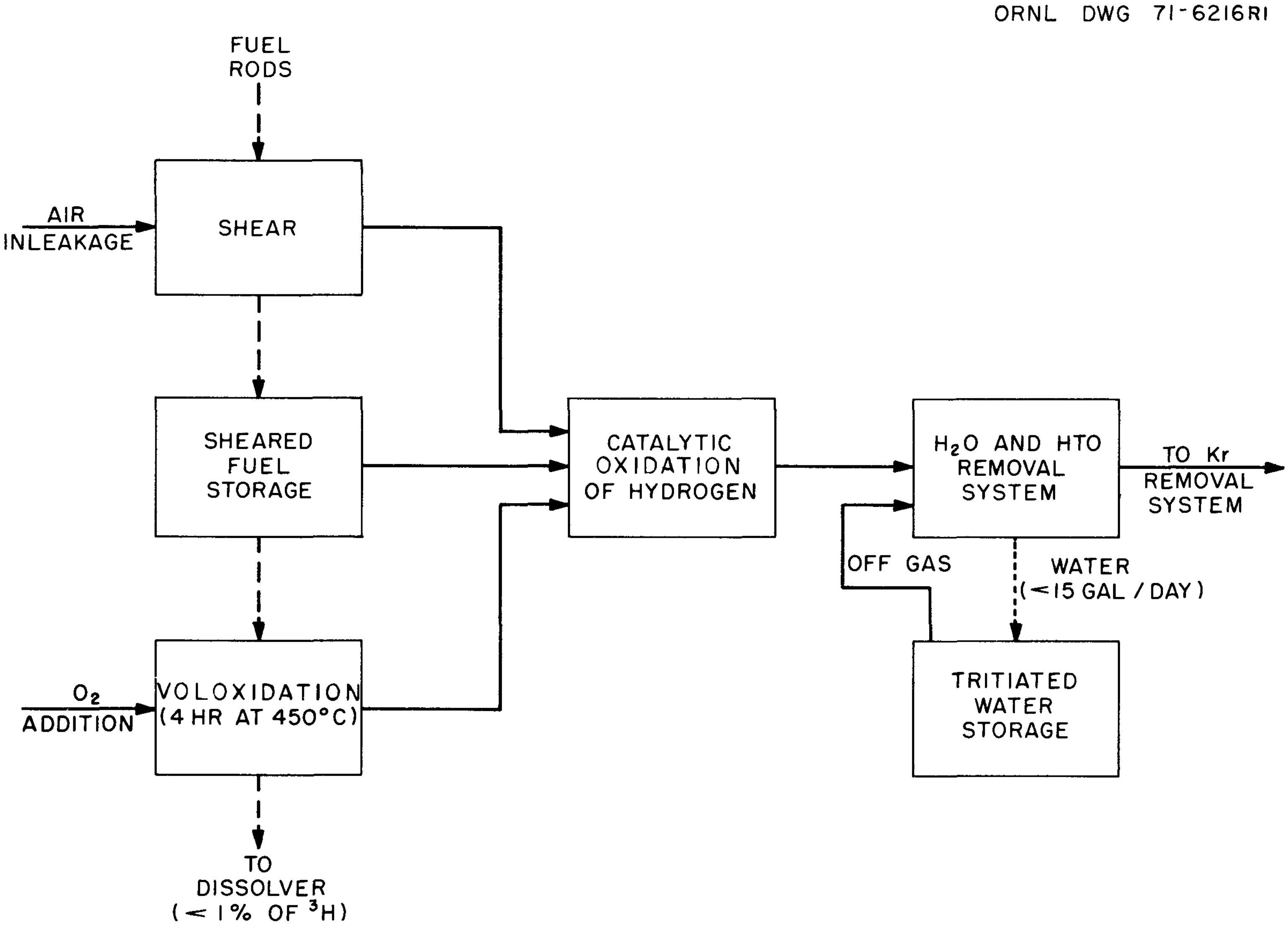

Fig. 4.14. Model Fuel Reprocessing Plant: Case 7 - Minimum Release of Radioactive Materials; Flowsheet for Retention of Tritium.

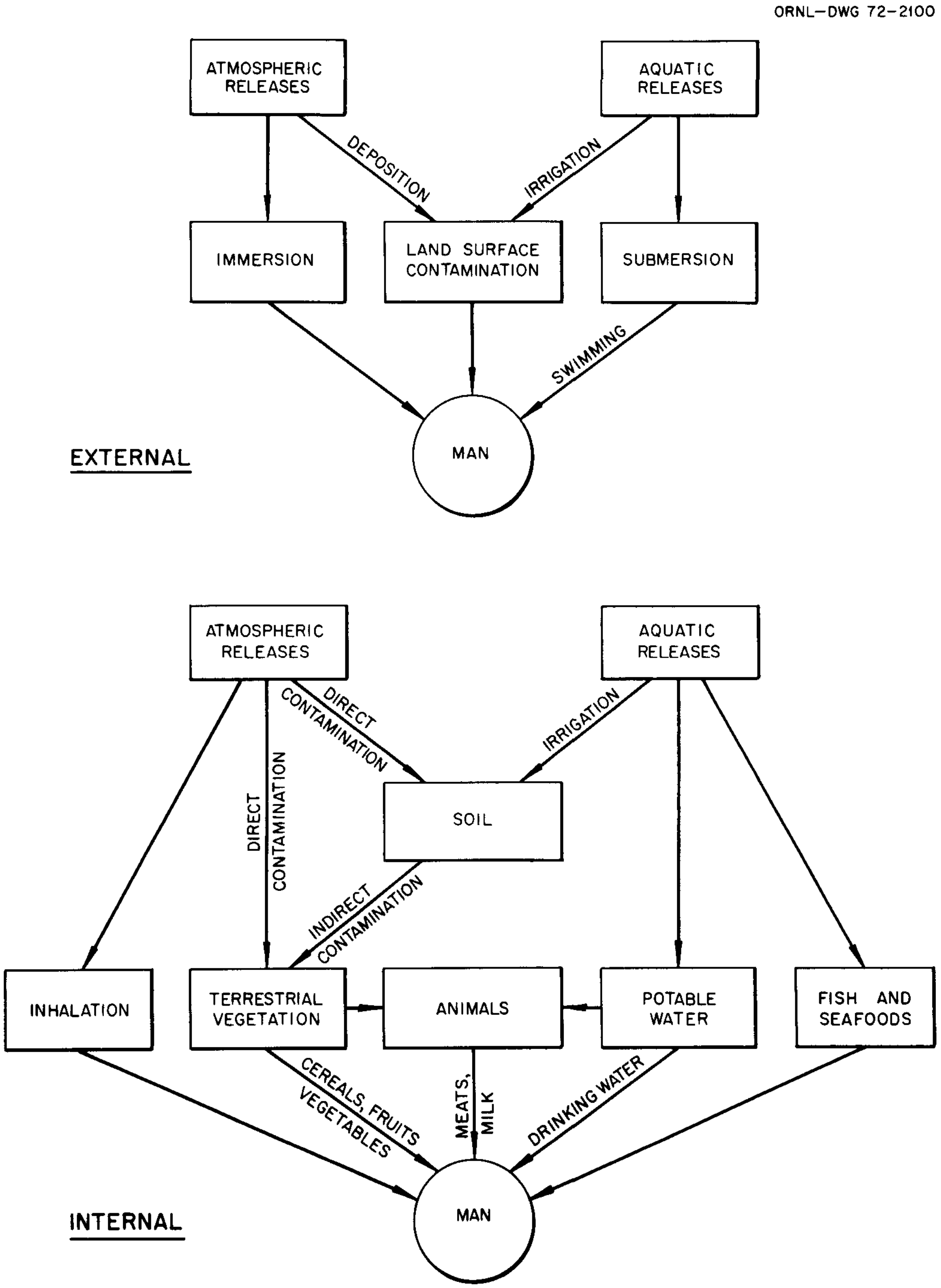

Fig. 7.1. Pathways for External and Internal Exposure of Man.

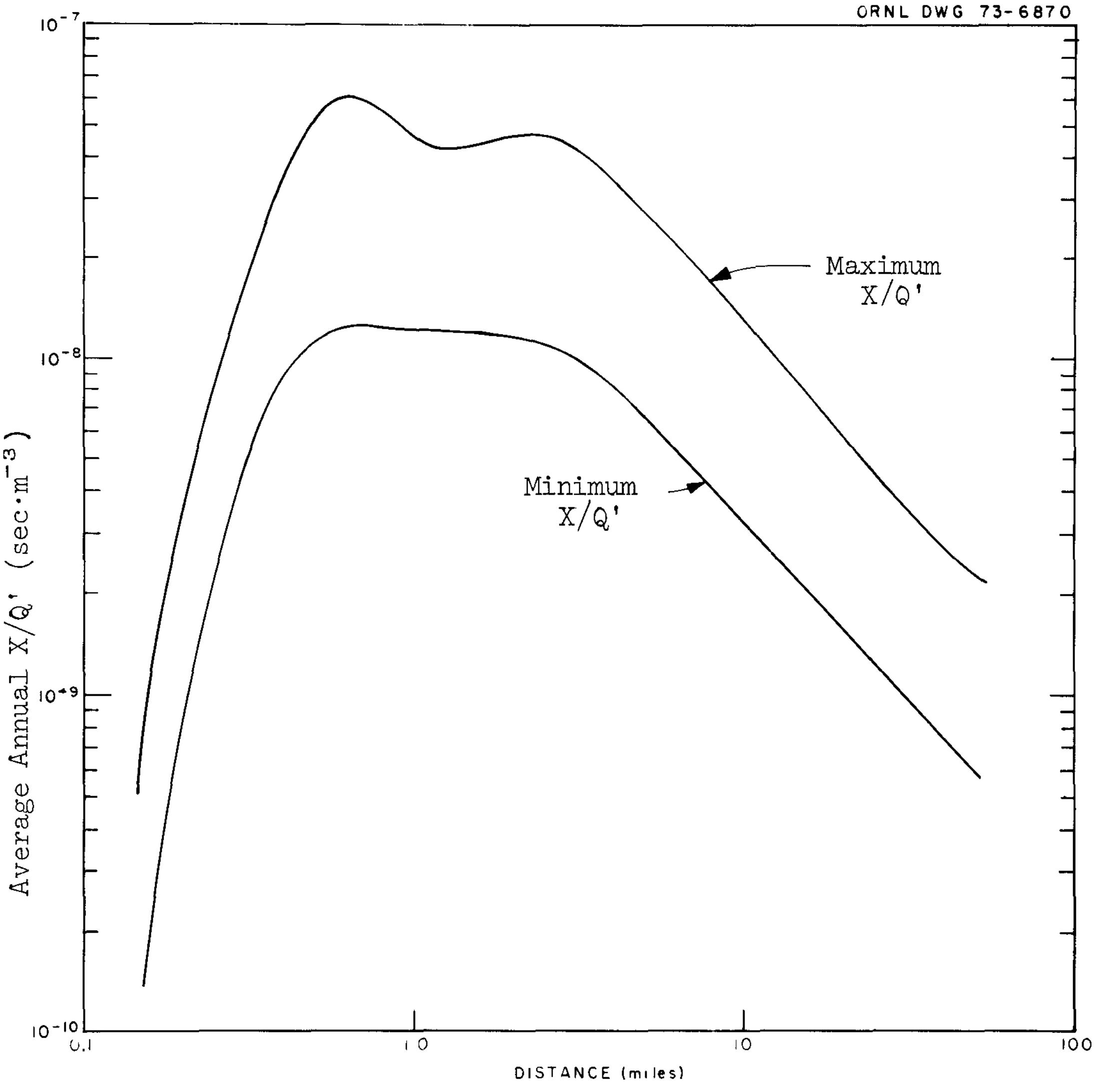

Fig. 7.2. Minimum and Maximum X/Q' Values for Elevated Release (100-m stack) for Midwestern Site. All values of X/Q' for 16 sectors fall between maximum-minimum limits at respective distances.

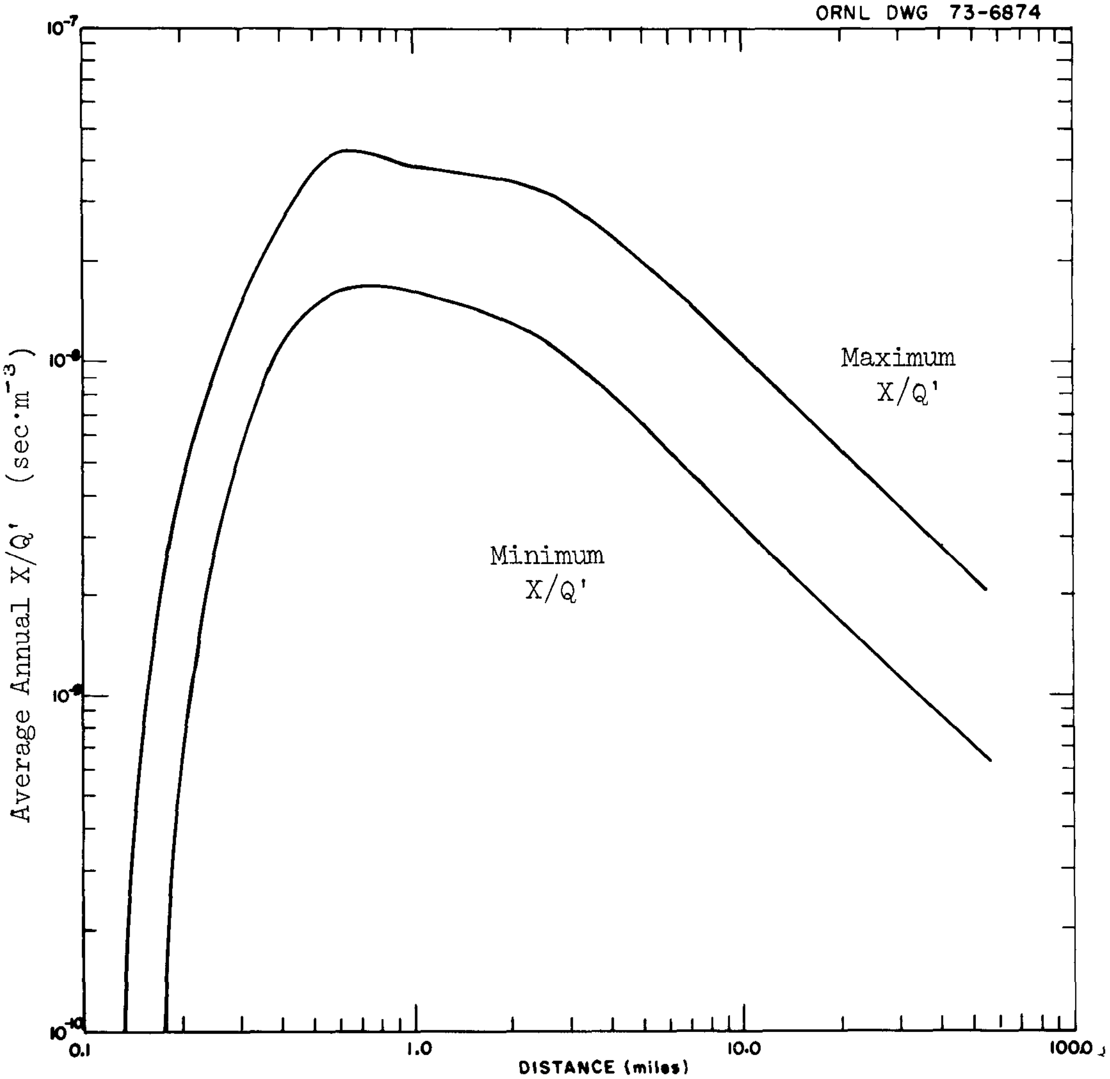

Fig. 7.3. Minimum and Maximum X/Q' Values for Elevated Release (100-m stack) for Coastal Plain Site. All values of X/Q' for 16 sectors fall between maximum-minimum limits at respective distances.

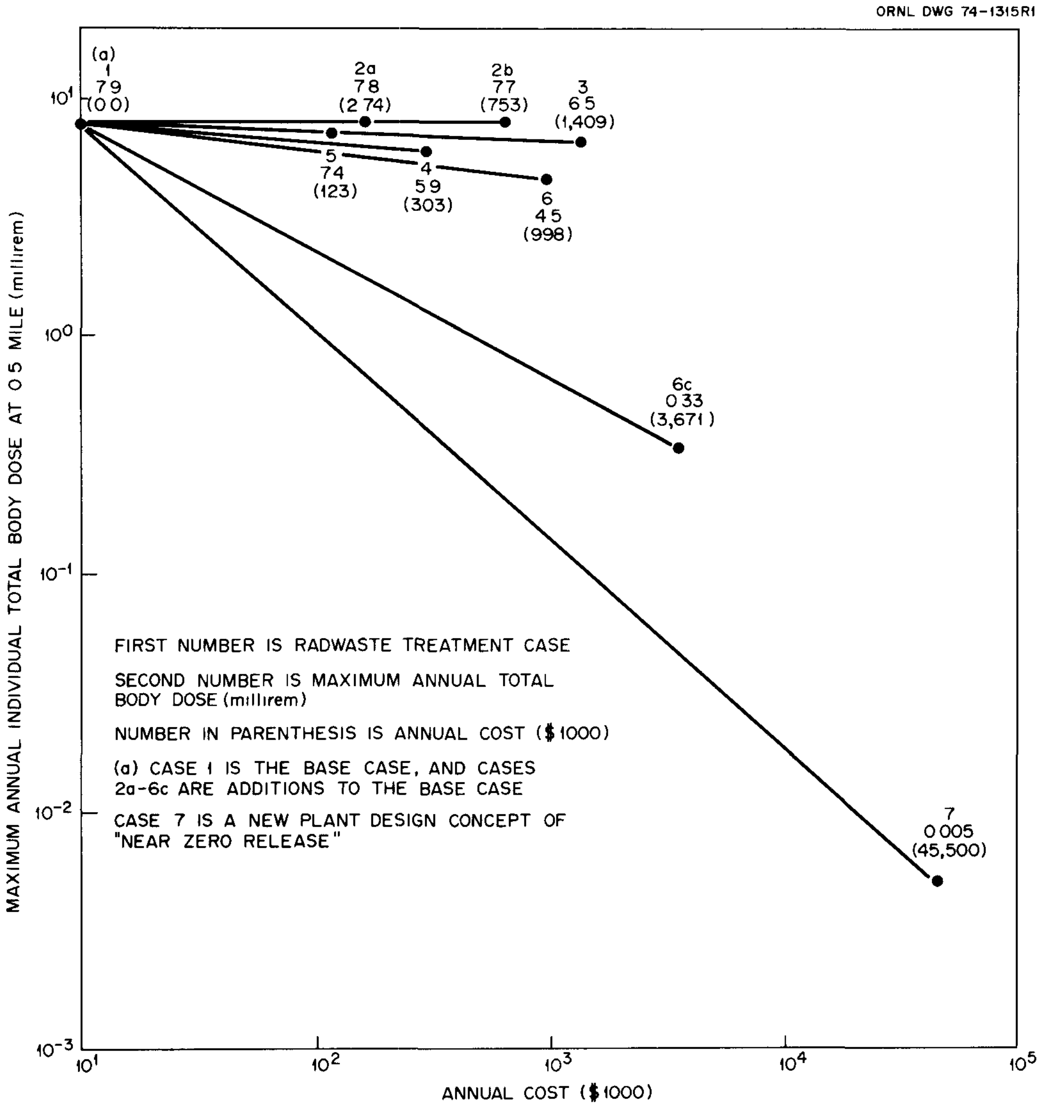

Fig. 8.1. Annual Cost for Reduction of Total Body Dose at 0.5 Mile From the Gaseous Effluent of the Model Fuel Reprocessing Plant at the Midwestern Site. Case 1 is the base case.

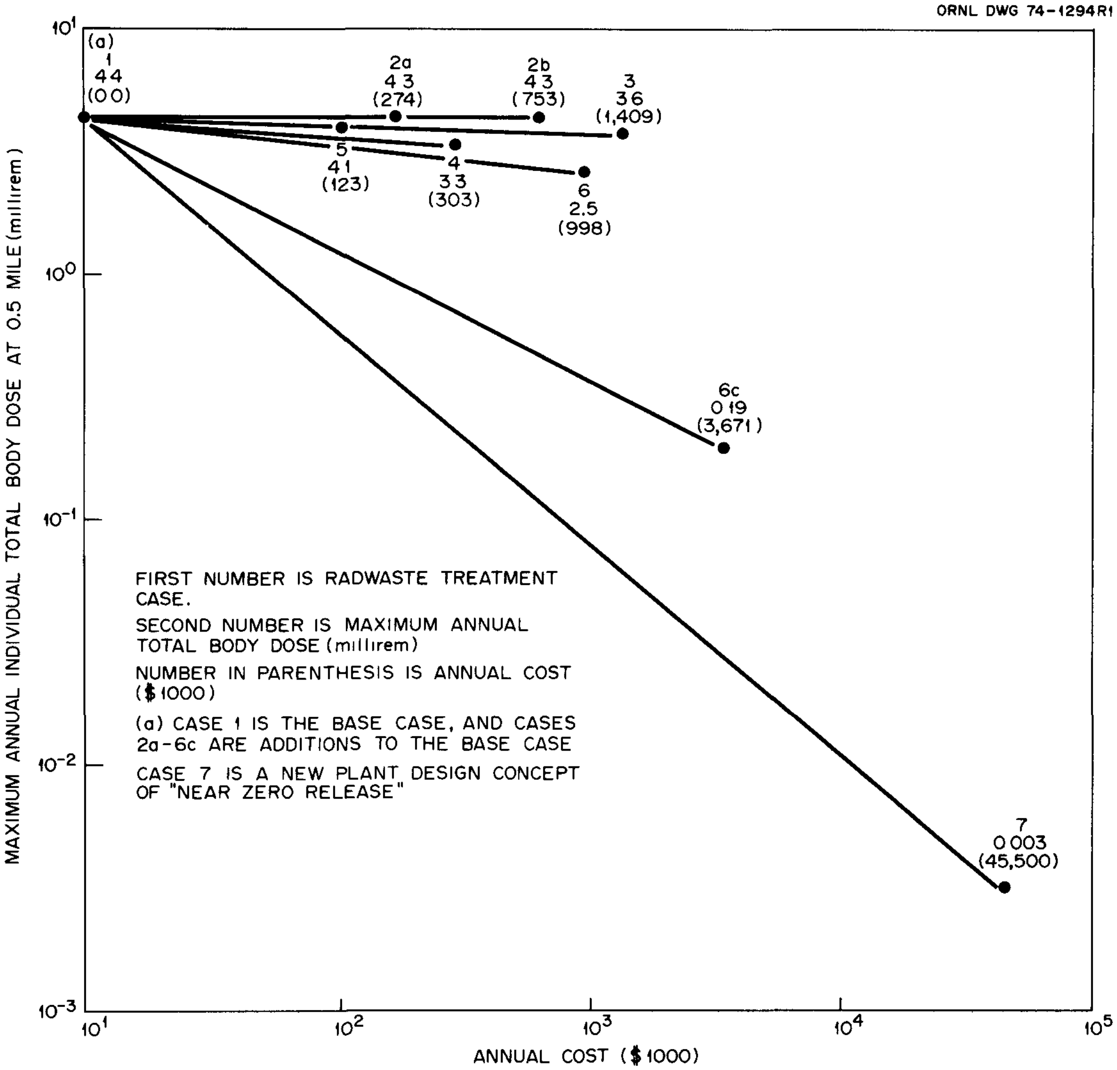

Fig. 8.2. Annual Cost for Reduction of Total Body Dose at 0.5 Mile From the Gaseous Effluent of the Model Fuel Reprocessing Plant at the Coastal Site. Case 1 is the base case.

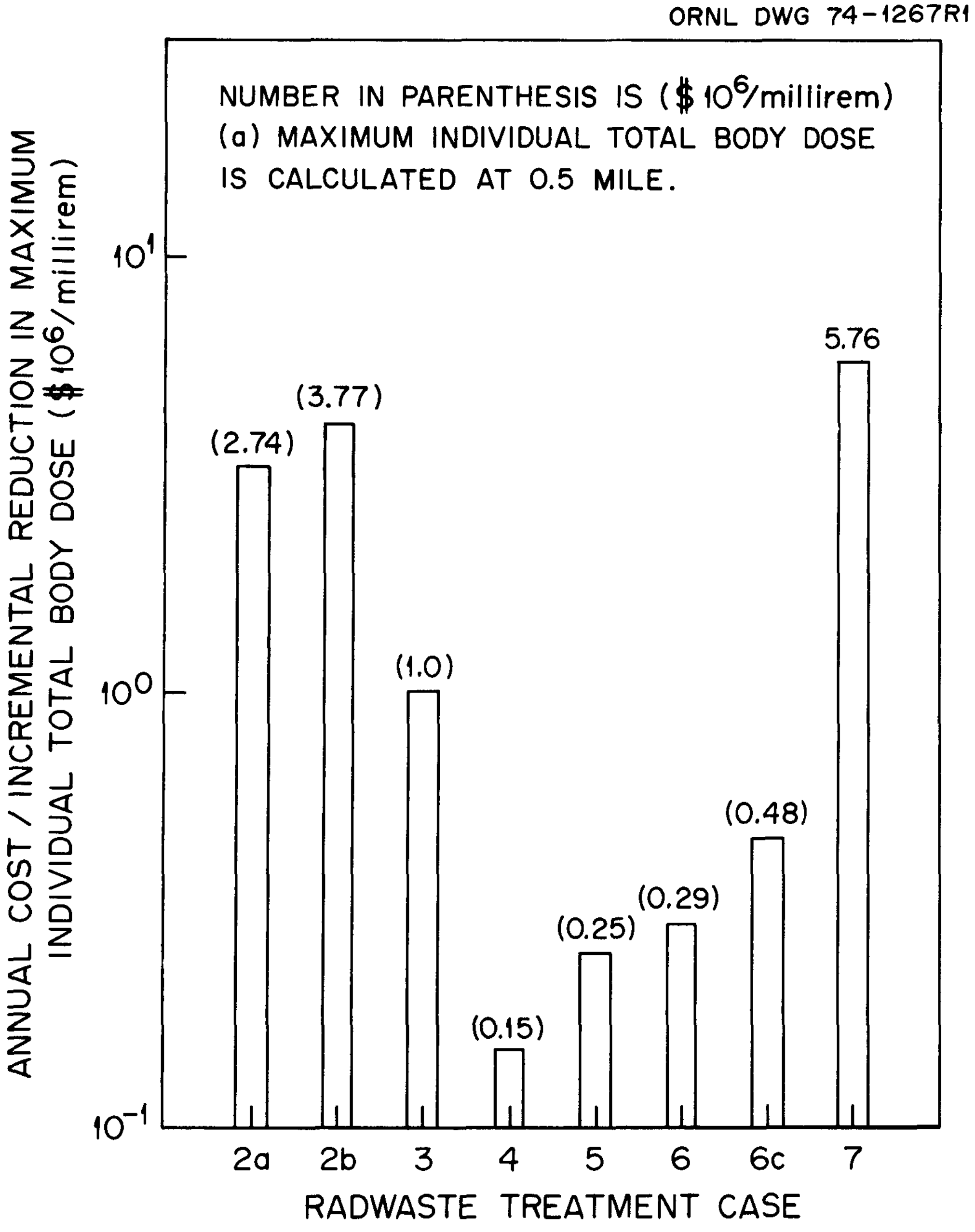

Fig. 8.3. Ratio of Annual Cost to Incremental Reduction in Maximum Individual Total Body Dose[a] From the Gaseous Effluent of the Model Fuel Reprocessing Plant at the Midwestern Site. Case 1 is the base case.

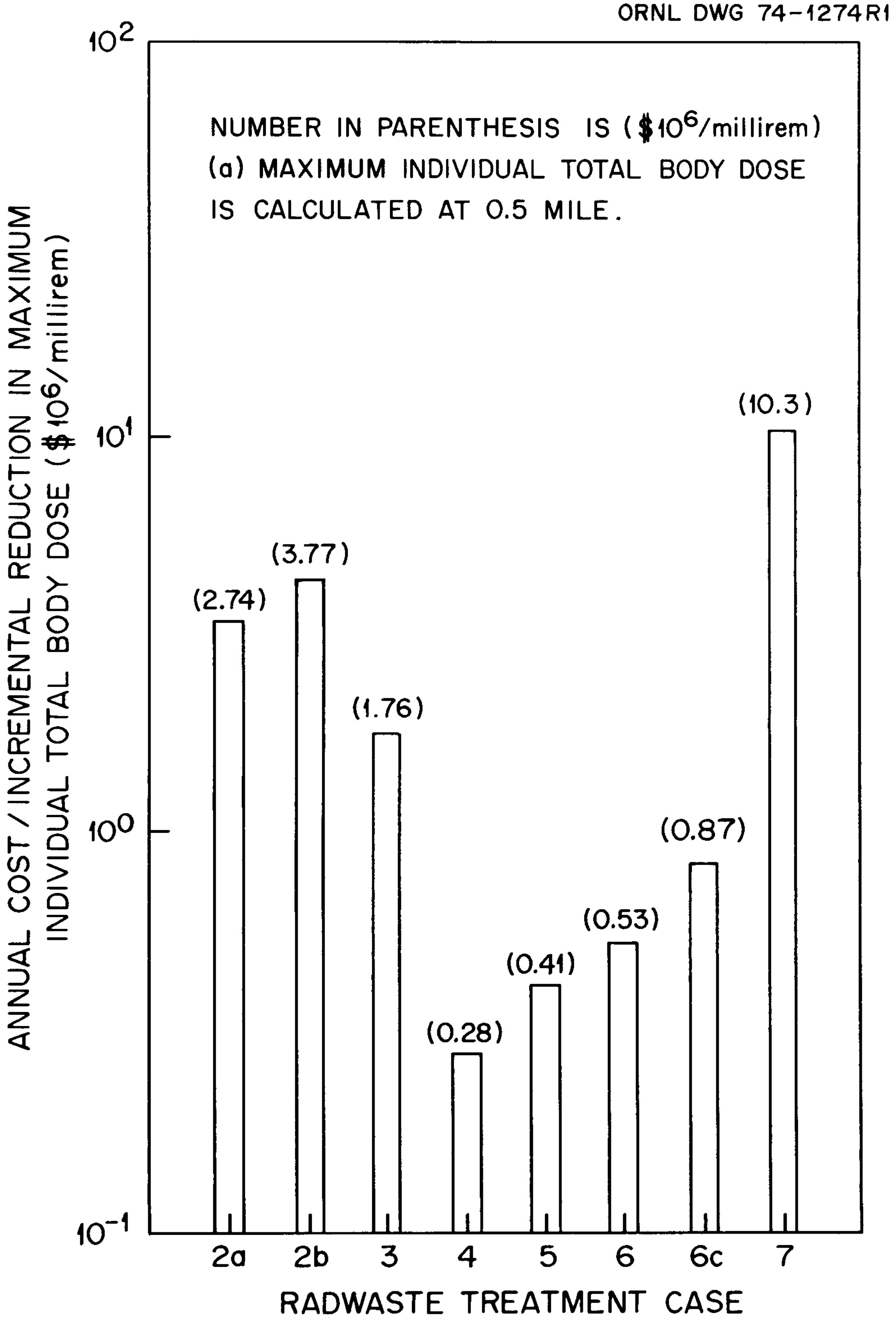

Fig. 8.4. Ratio of Annual Cost to Incremental Reduction in Maximum Individual Total Body Dose[a] From the Gaseous Effluent of the Model Fuel Reprocessing Plant at the Coastal Site. Case 1 is the base case.

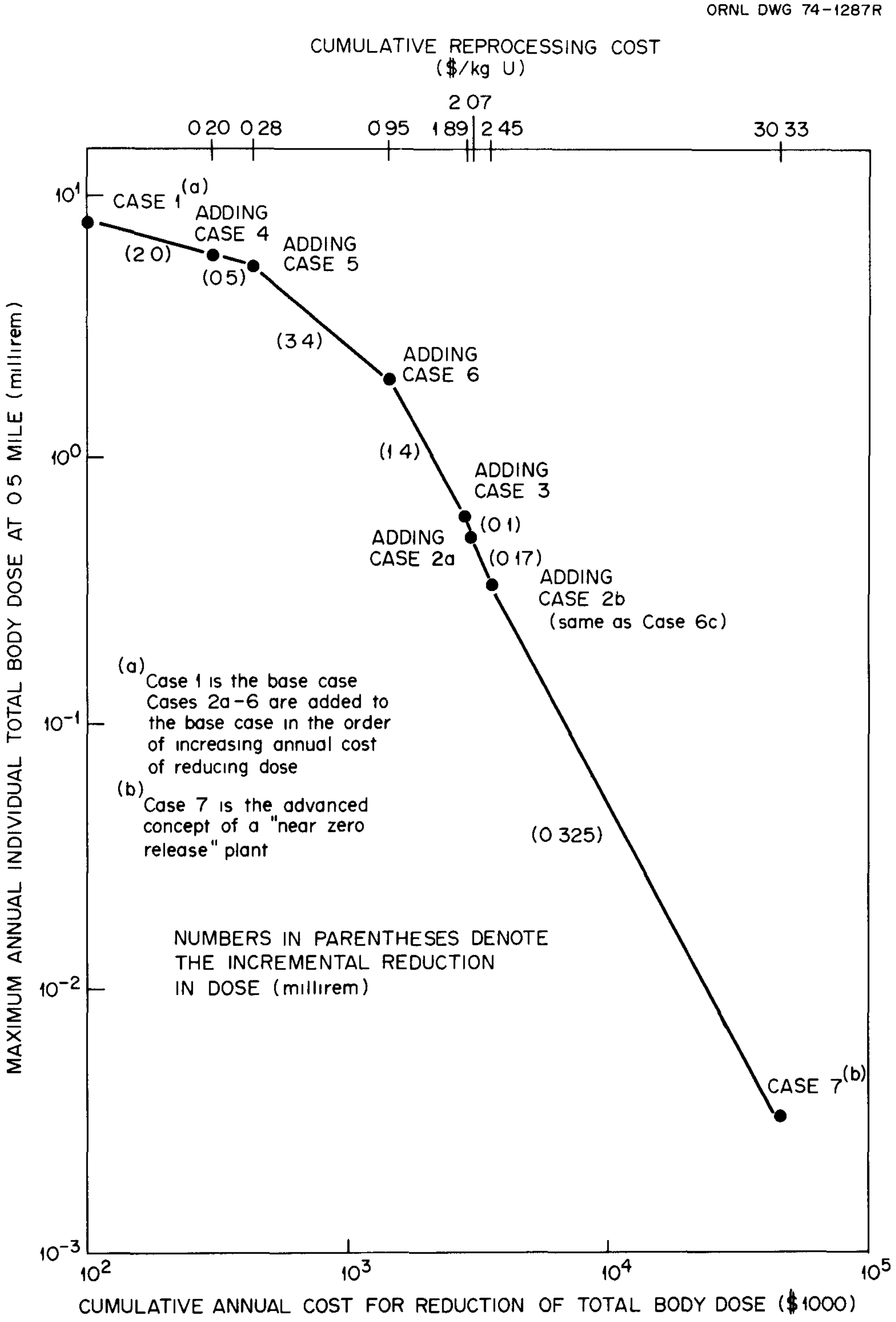

Fig. 8.5. Cumulative Annual and Reprocessing Costs for Reduction of Total Body Dose From the Gaseous Effluent of the Model Fuel Reprocessing Plant at the Midwestern Site.

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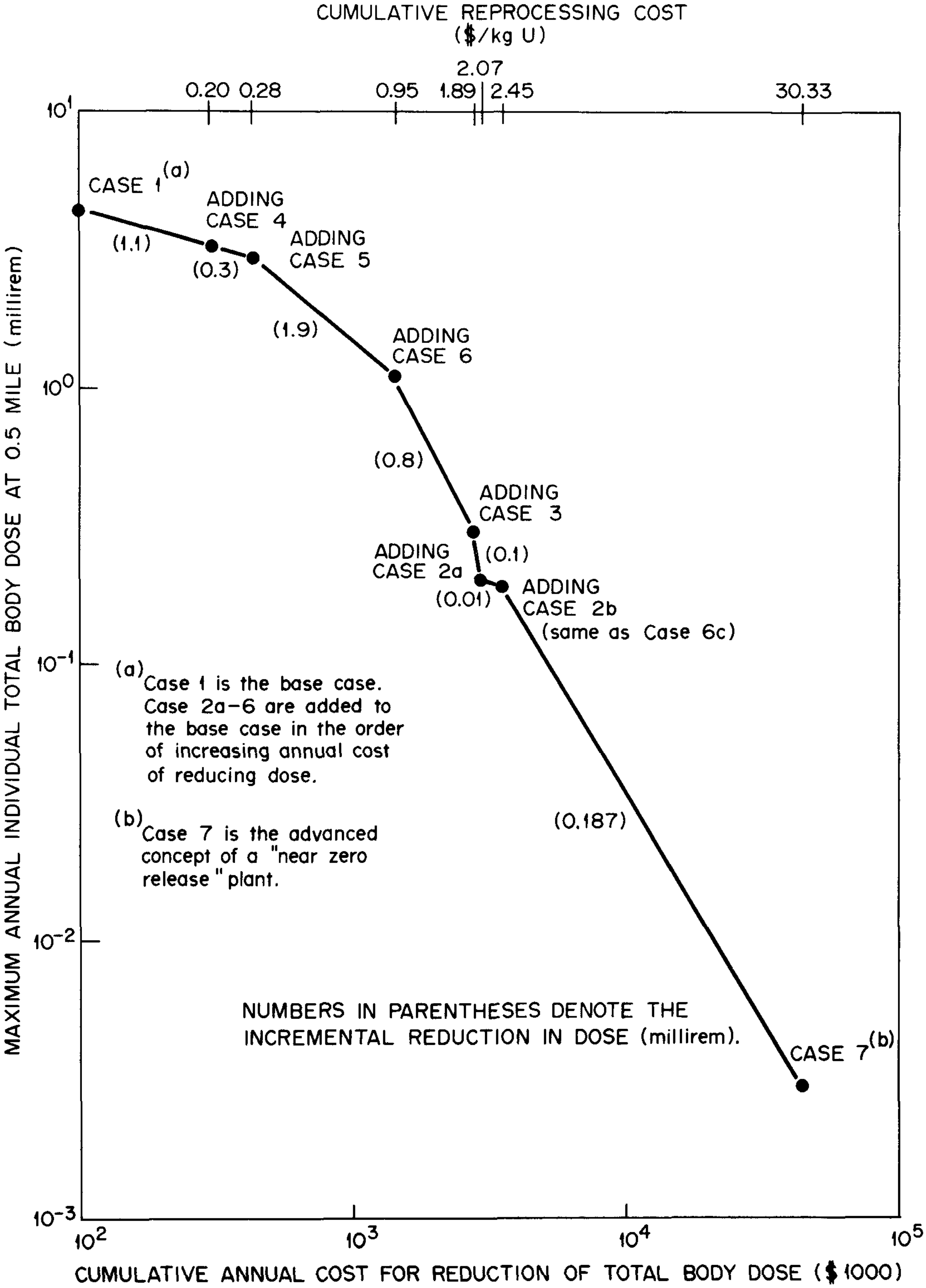

Fig. 8.6. Cumulative Annual and Reprocessing Costs for Reduction of Total Body Dose From the Gaseous Effluent of the Model Fuel Reprocessing Plant at the Coastal Site.

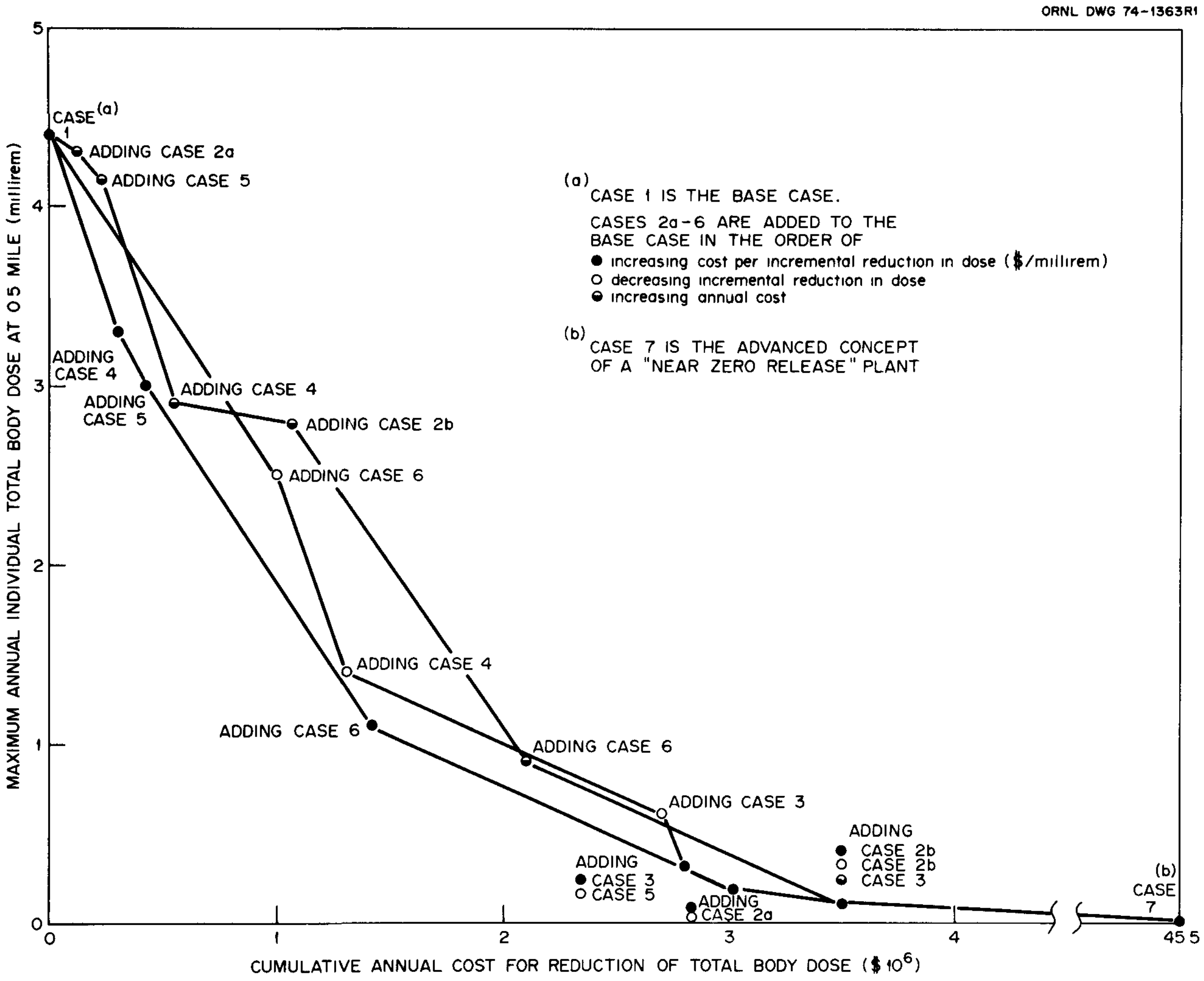

Fig. 8.7. Cumulative Annual Cost for Reduction of Total Body Dose From the Gaseous Effluent of the Model Fuel Reprocessing Plant at the Coastal Site Based on (1) Increasing Cost per Incremental Reduction in Dose ($/millirem), (2) Decreasing Incremental Reduction in Dose, and (3) Increasing Annual Cost.

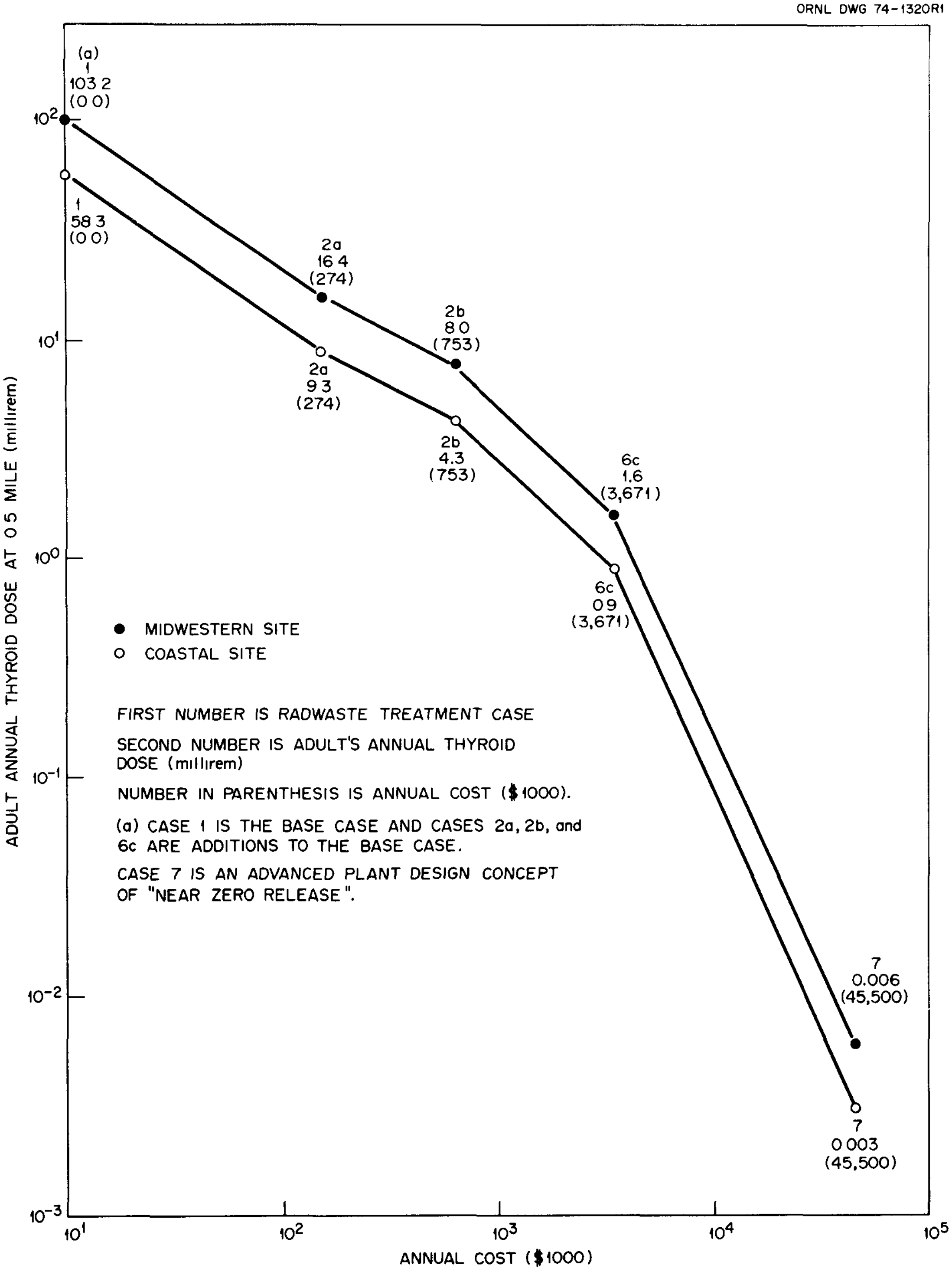

Fig. 8.8. Annual Cost for Reduction of Adult Thyroid Dose at 0.5 Mile From the Gaseous Effluent of the Model Fuel Reprocessing Plant at the Midwestern and Coastal Sites. Case 1 is the base case.

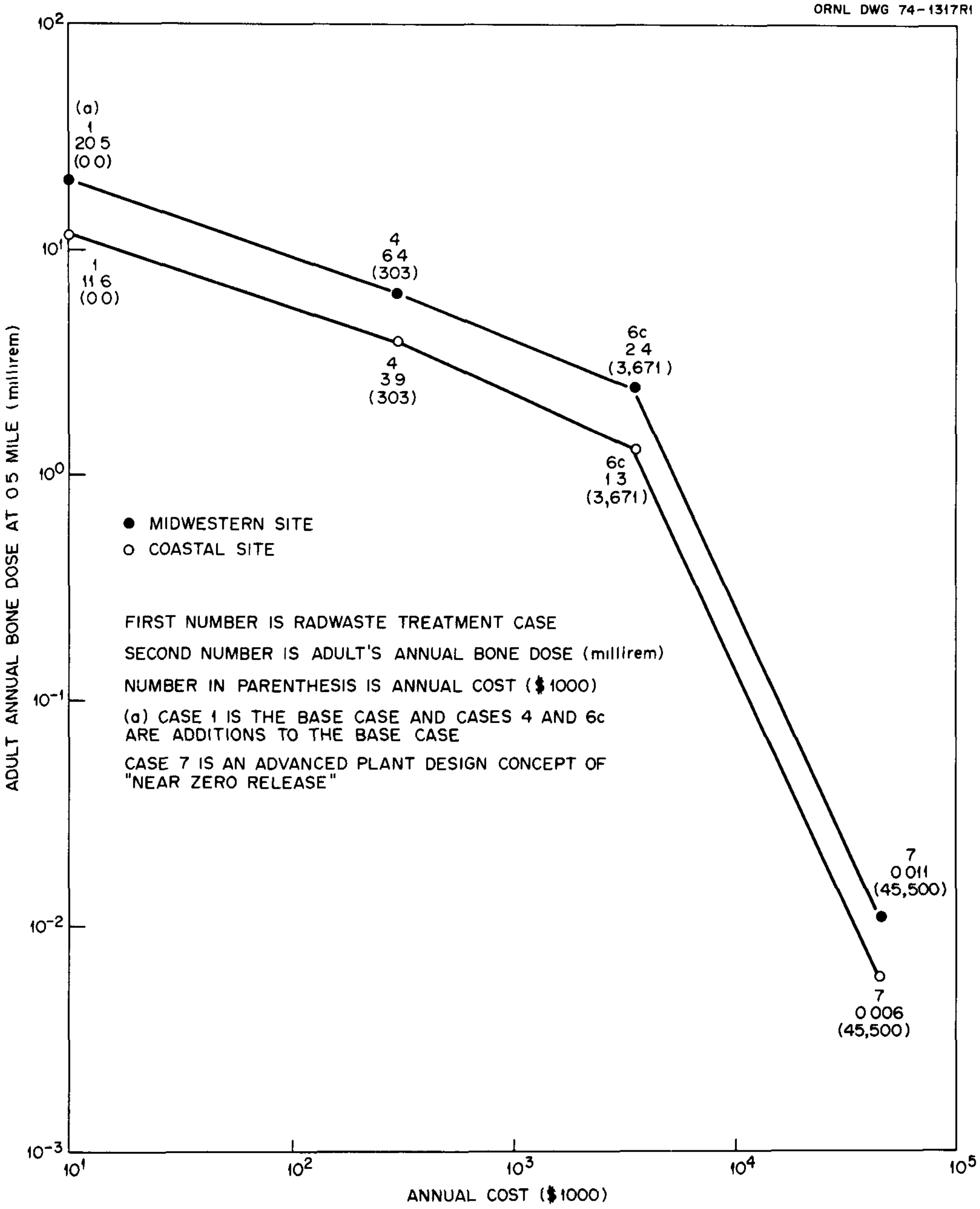

Fig. 8.9. Annual Cost for Reduction of Adult Bone Dose at 0.5 Mile From the Gaseous Effluent of the Model Fuel Reprocessing Plant at the Midwestern Site and Coastal Site. Case 1 is the base case.

ORNL DWG 74-1276R2

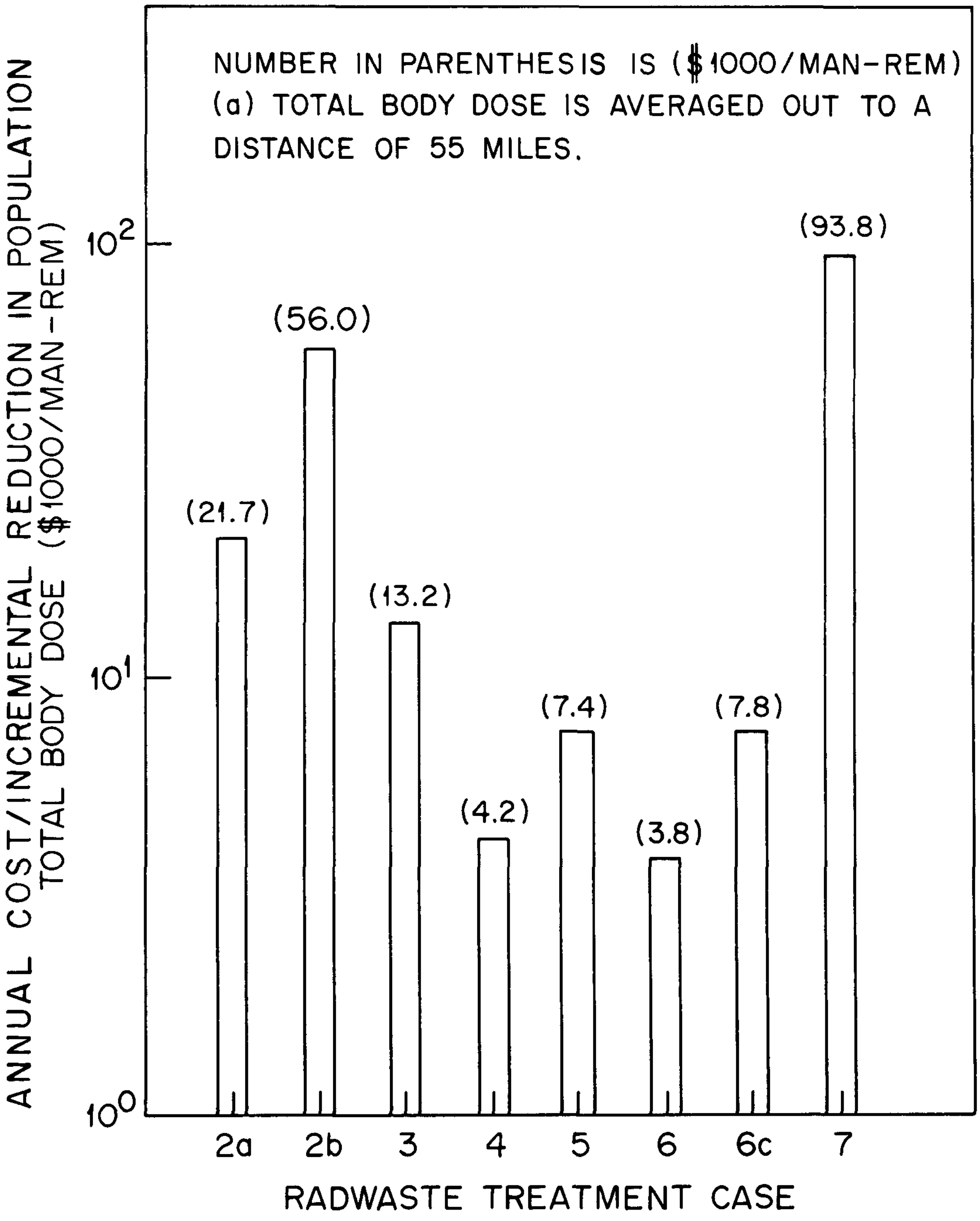

Fig. 8.10. Ratio of Annual Cost to Incremental Reduction in Population Total Body Dose[a] From the Gaseous Effluent of the Model Fuel Reprocessing Plant at the Midwestern Site. Case 1 is the base case.

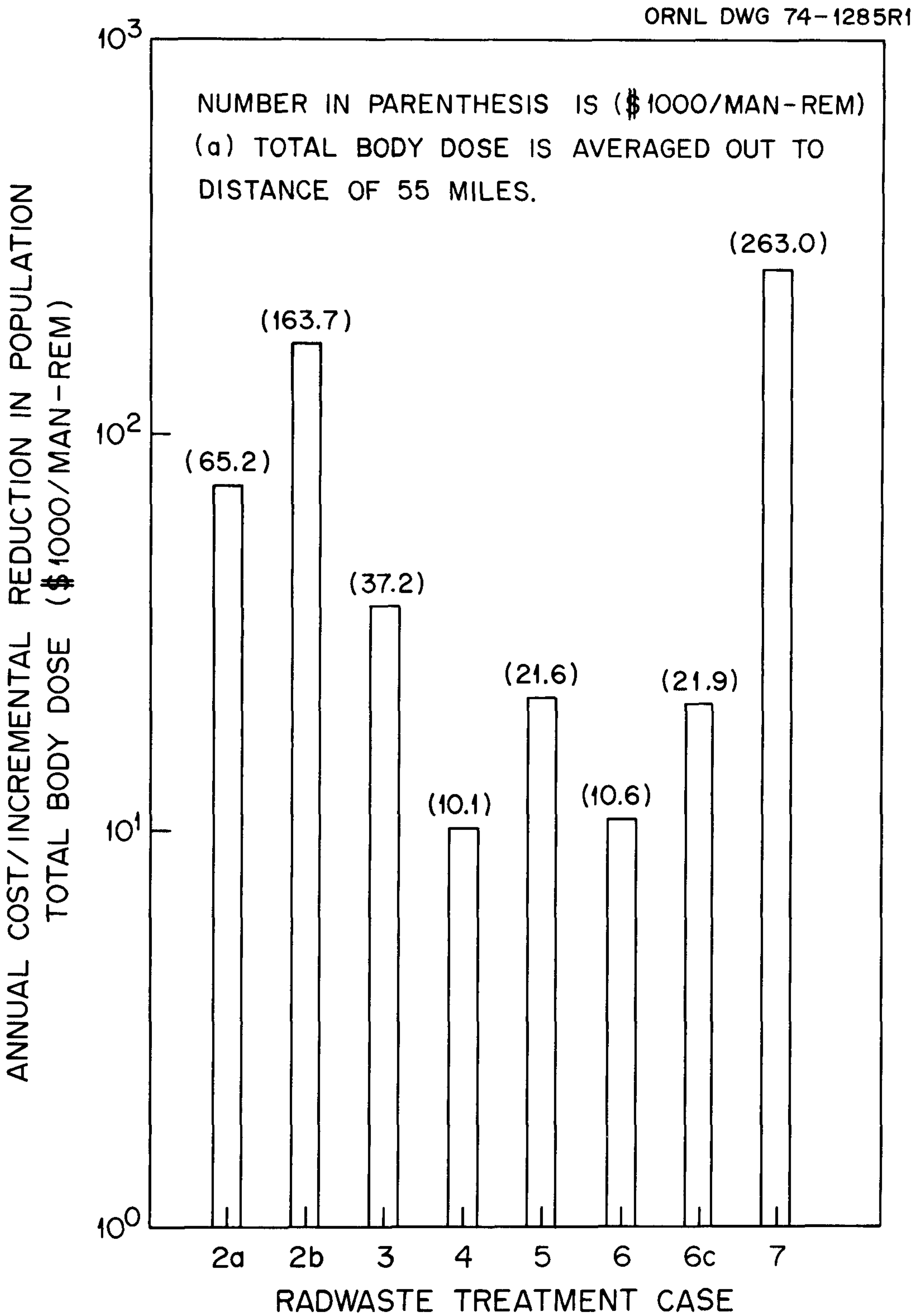

Fig. 8.11. Ratio of Annual Cost to Incremental Reduction in Population Total Body Dose[a] From the Gaseous Effluent of the Model Fuel Reprocessing Plant at the Coastal Site. Case 1 is the base case.

APPENDIX A. PREPARATION OF COST ESTIMATES

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

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B. C. Finney W. G. Stockdale
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 late-1973 dollars, obtained by Burns and Roe, Inc., 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, an "N Stamp Required" designation is included in the specifications for equipment. Applicable ASME codes are also included. In the equipment lists presented in the tables in this Appendix, "A" indicates a vendor's price, "AB" indicates a vendor's price plus a Burns and Roe estimate, and "B" indicates a Burns and Roe estimate.

The costs prepared by Burns and Roe are based on conceptual flowsheets and process technology provided by Oak Ridge National Laboratory and vendor bid prices for equipment. These costs were revised 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, 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 is 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 feet wide, (2) remote maintenance cells are 60 feet 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 accurately positioned to insure 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 ft^3 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 where 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 Services[2] 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, 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 equipment flowsheets (Figs. A-1 through A-8) and detailed equipment lists and equipment costs (Tables A-1 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

	Percentage of Direct Cost
Quality assurance[a]	6
Other owner's cost	10
Contingency	40
Interest during construction[b]	39
	149

The interest during construction and the contingency allowance are included as indirect costs to simplify the calculations.

1.2 References

1. 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).
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).

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[a] Quality assurance costs based on the vendor's equipment bid prices obtained by Burns and Roe, Inc., 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.

[b] Interest 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 3	Case 4	Case 5	Case 6
Equipment purchase	$ 54,560	$126,470	$777,580	$ 20,200	$ 46,840	$ 425,500
Installation	16,100	26,900	26,420	1,100	5,200	8,900
Piping	36,140	70,340	158,600	20,800	26,460	32,000
Instrument and controls	16,200	46,360	50,500	8,000	15,900	5,600
Cell and adjacent building	179,300	561,500	541,500	-	41,300	630,000
External structure	-	-	-	-	1,000	-
Sand filter	-	-	-	375,300	-	-
Direct Const. Cost	$302,300	$ 831,600	$1,554,600	$425,400	$136,700	$1,102,000
Indirect Cost, ~149%[b]	450,400	1,239,000	2,316,400	633,800	203,700	1,642,000
Total Const. Cost	$753,000	$2,071,000	$3,871,000	$1,059,000	$340,000	$2,744,000

[a] Case 2b includes Case 2a.

[b]

Indirect Cost	Percentage of 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
	149

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 3	Case 4	Case 5	Case 6
Remote maintenance cell, sq. ft.	25	207	-	-	-	250
Contact maintenance cell, sq. ft.	169	169	722	-	55	140
ALLOWANCE FOR COST OF CELL STRUCTURE AND ADJACENT AREAS						
Remote maintenance cell	$52,500	$434,700	-	-	-	$525,000
Contact maintenance cell	126,800	126,800	541,500	-	41,300	105,000
Total cell and adjacent building	$179,300	$561,500	$541,500	-	$41,300	$630,000

Remote maintenance cells are 25 feet wide, 60 feet high, and are costed at $35 per cubic foot.

Direct maintenance cells are 25 feet wide, 30 feet high, and are costed at $25 per cubic foot.

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.

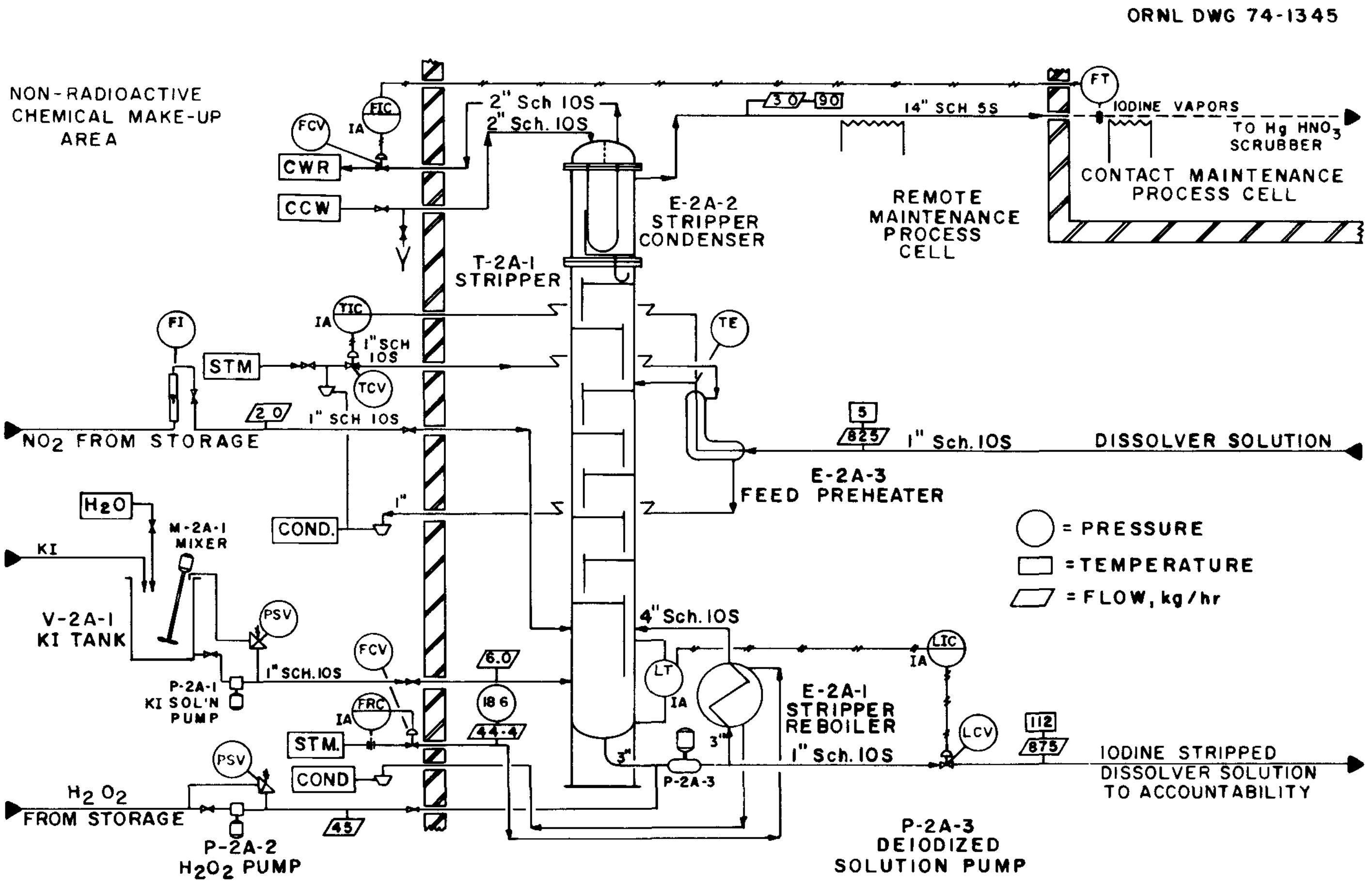

Fig. A-1. Equipment Flowsheet No. 1 for Model Nuclear Fuel Reprocessing Plant Radwaste Treatment Case 2a - Reduction in Release of Iodine — Iodine Evolution.

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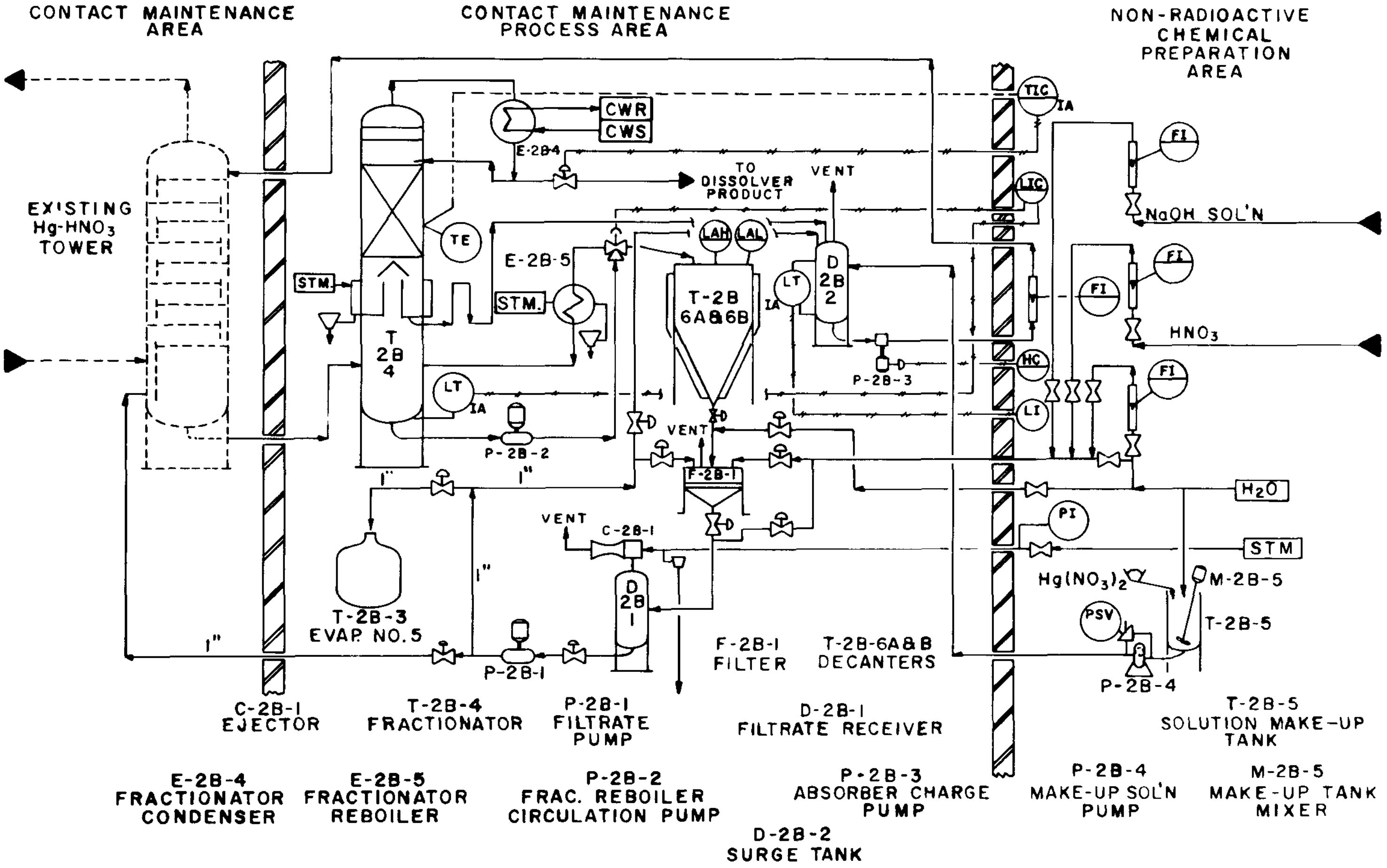

Fig. A-2. Equipment Flowsheet No. 2 for Model Nuclear Fuel Reprocessing Plant Radwaste Treatment Case 2a - Reduction in Release of Iodine — Mercury Recycle.

Table A-3. Equipment List and Cost for Radwaste Treatment Case 2a - Reduction in Release of Iodine— Iodine Evolution, and Mercury Recycle

Item No.	Descriptive Title	Number of Items	Unit Cost ($)	Total Cost ($)	Reference and/or Source
Remote In-Cell Equipment					
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	
Contact In-Cell Equipment					
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 A&B	Decanters	2	990	1,980	A
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			22,400	
Chemical Preparation Area					
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			$54,560	

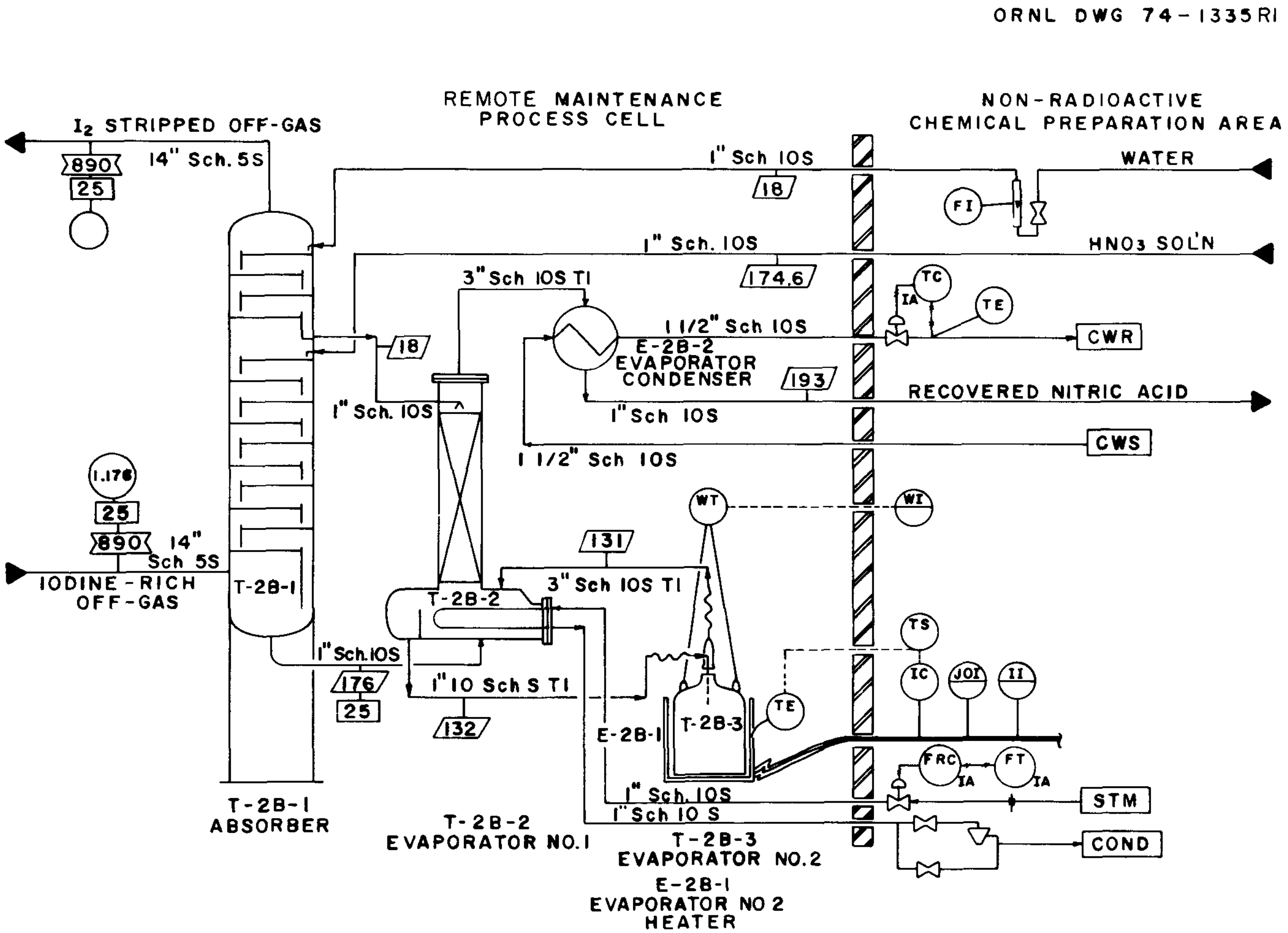

Fig. A-3. Equipment Flowsheet for Model Nuclear Fuel Reprocessing Plant Radwaste Treatment Case 2b - Reduction in Release of Iodine — Iodox.

Table A-4. Equipment List and Cost for Radwaste Treatment Case 2b - Reduction in Release of Iodine— Iodine Evolution, Mercury Recycle, and Iodox

Item No.	Descriptive Title	Number of Items	Unit Cost ($)	Total Cost ($)	Reference and/or Source
Remote In-Cell Equipments					
T-2B-1	Absorber, Ti-lined	1	54,000	54,000	AB
T-2B-2	Evaporator #1, Ti	1	15,000	15,000	A
E-2B-1	Evaporator #2, heater, S.S.	1	910	910	A
E-2B-2	Evaporator condenser, Ti and S.S.	1	2,000	2,000	A
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			100,770	
Contact In-Cell Equipment					
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 A&B	Decanters	2	990	1,980	A
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			22,400	
Chemical Preparation Area					
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			$126,470	

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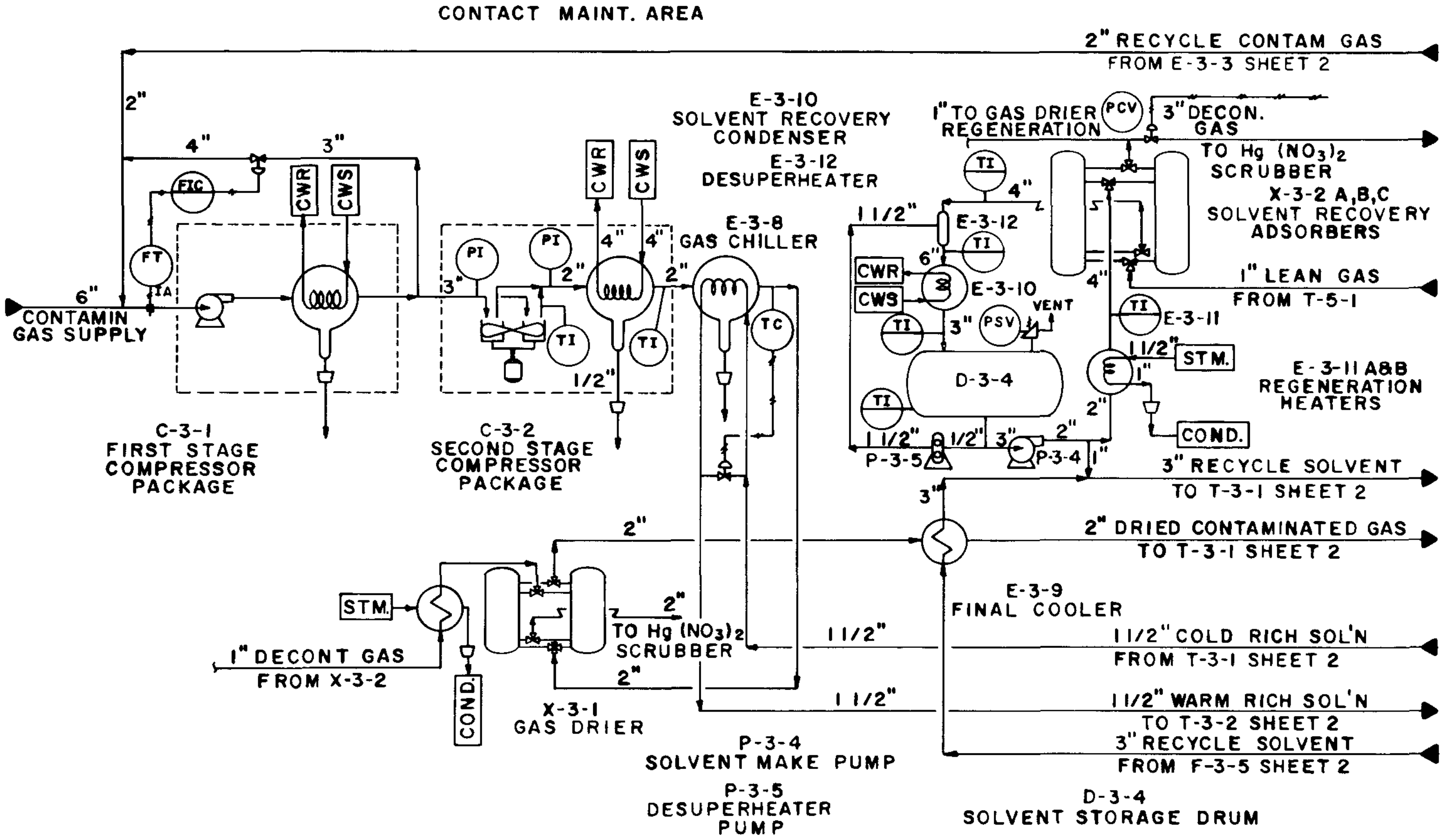

Fig. A-4. Equipment Flowsheet No. 1 for Model Nuclear Fuel Reprocessing Plant Radwaste Treatment Case 3 - Reduction in Release of Krypton — Selective Absorption.

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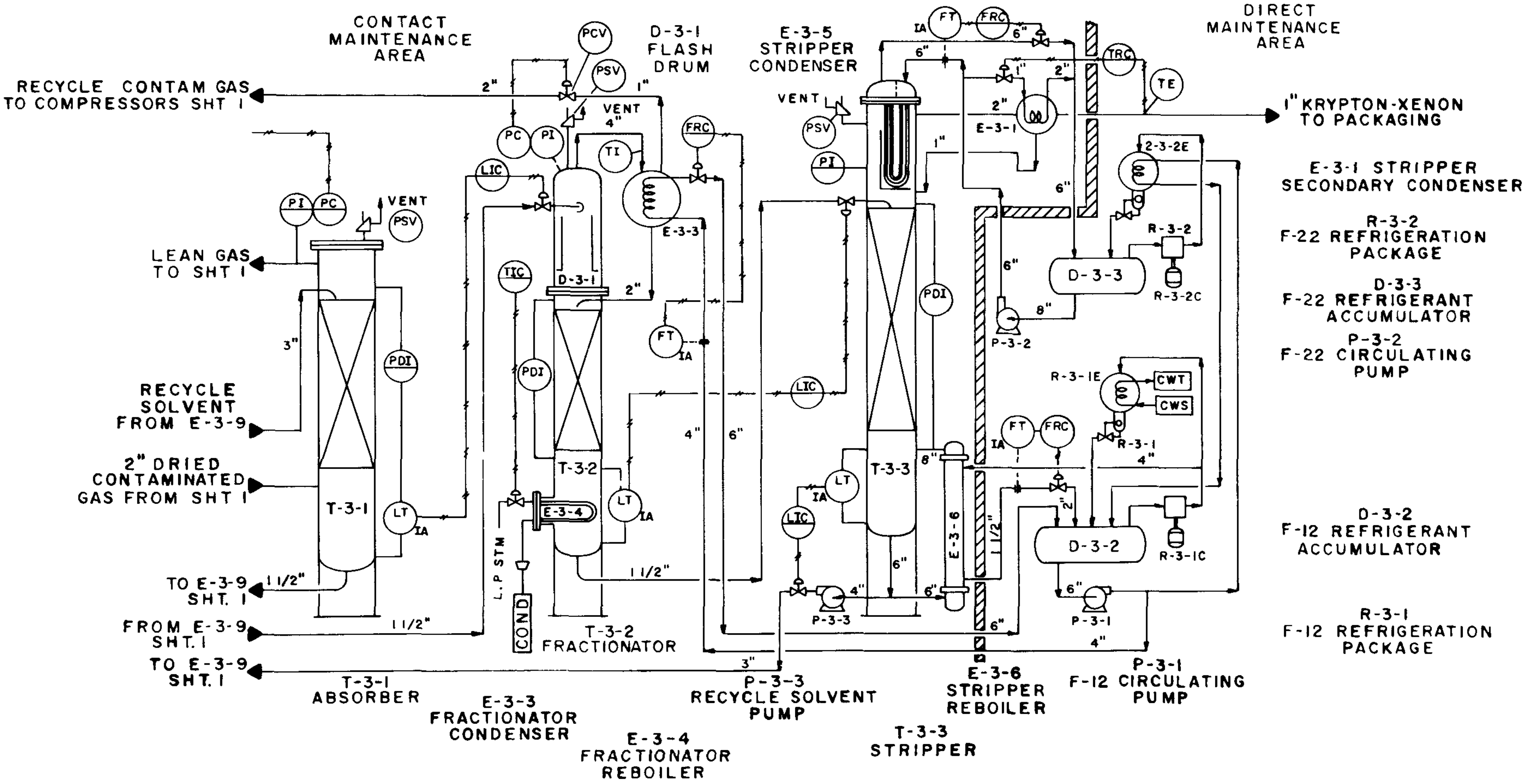

Fig. A-5. Equipment Flowsheet No. 2 for Model Nuclear Fuel Reprocessing Plant Radwaste Treatment Case 3 - Reduction in Release of Krypton — Selective Absorption.

Table A-5. Equipment List and Cost for Radwaste Treatment Case 3 - Reduction in Release of Krypton-Selective Absorption

Item No.	Descriptive Title	Number of Items	Unit Cost ($)	Total Cost ($)	Reference and/or Source
Contact In-Cell Equipment					
T-3-1	Absorber and packing	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 A,B,&C	Solvent recovery absorbers	3	55,200	165,600	AB
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 A,B,C,&D	Regeneration heaters	4	3,500	14,000	A
E-3-12	Desupersaturater	1	1,820	1,820	A
C-3-1	1st stage compressor pkg	2	39,040	78,080	A
C-3-2	2nd stage compressor pkg	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
	Subtotal			468,970	
Chemical Preparation Area					
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	135,000	270,000	A
R-3-2	Freon-22 refrigeration pkg.	1			A
	Subtotal			308,610	
	Grand Total			$777,580	

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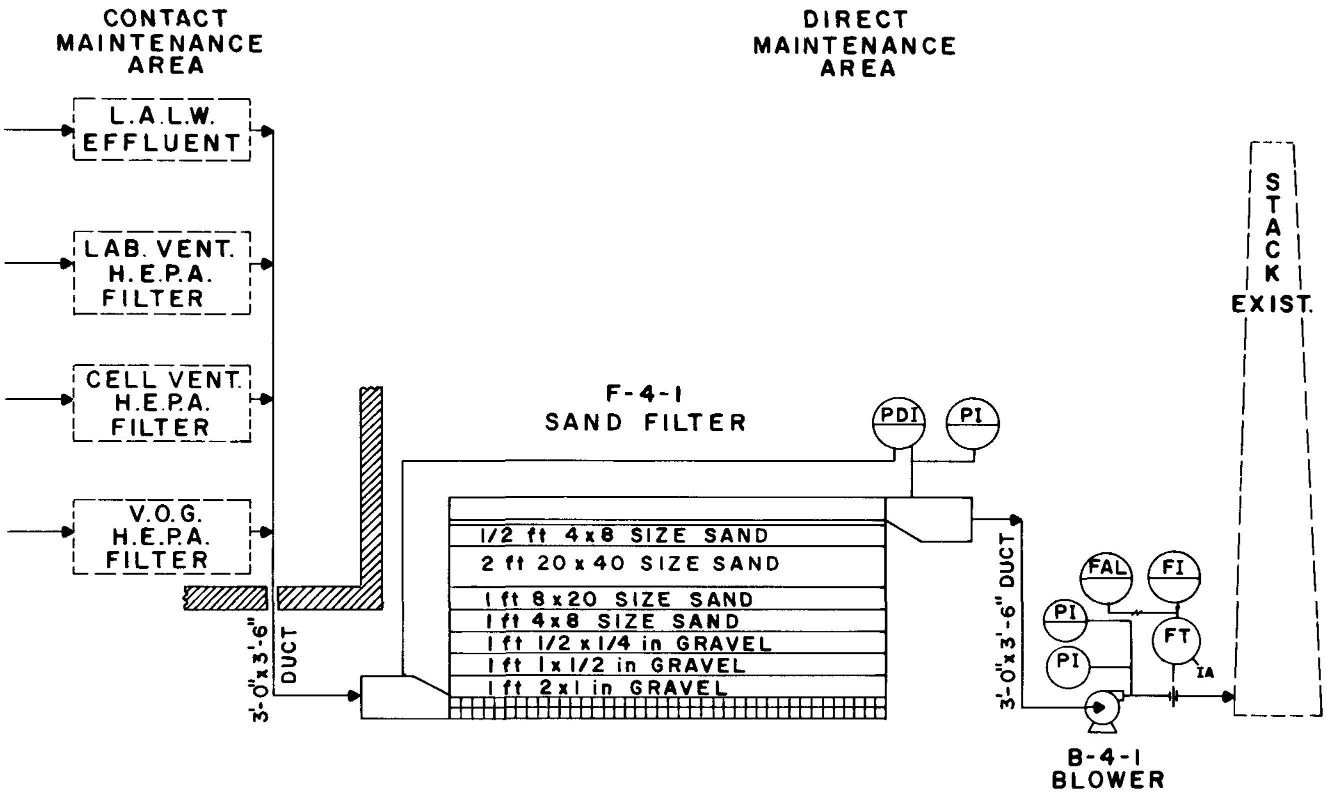

Fig. A-6. Equipment Flowsheet for Model Nuclear Fuel Reprocessing Plant Radwaste Treatment Case 4 - Reduction in Release of Particulates — Sand Filter.

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	No. of Items	Unit Cost ($)	Total Cost ($)	Reference and/or Source
B-4-1	Blower	1	20,200	20,200	A
		Sand Filter			
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	17,580	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.

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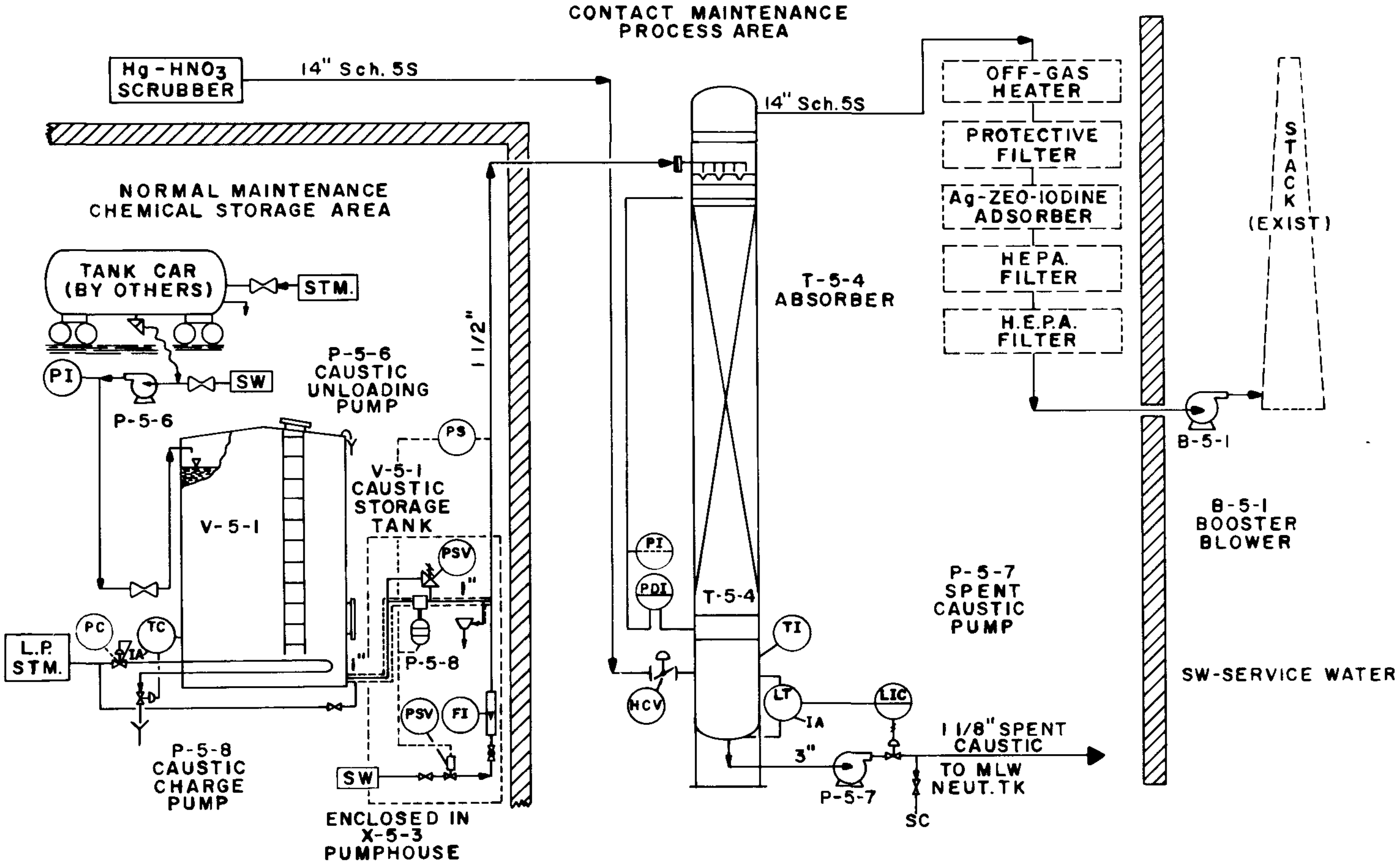

Fig. A-7. Equipment Flowsheet for Model Nuclear Fuel Reprocessing Plant Radwaste Treatment Case 5 - Reduction in Release of Semivolatiles — Caustic Scrubber.

Table A-7. Equipment List and Cost for Radwaste Treatment Case 5 - Reduction in Release of Semi-Volatiles - Caustic Scrubber

Item No.	Descriptive Title	Number of Items	Unit Cost ($)	Total Cost ($)	Reference and/or Source
Contact In-Cell Equipment					
T-5-4	Absorber	1	22,900	22,900	A
P-5-7	Spent caustic pump	1	660	660	A
	Subtotal			23,560	
Chemical Storage Area					
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	1,890	A
	Subtotal			23,280	
	Grand Total			$46,840	

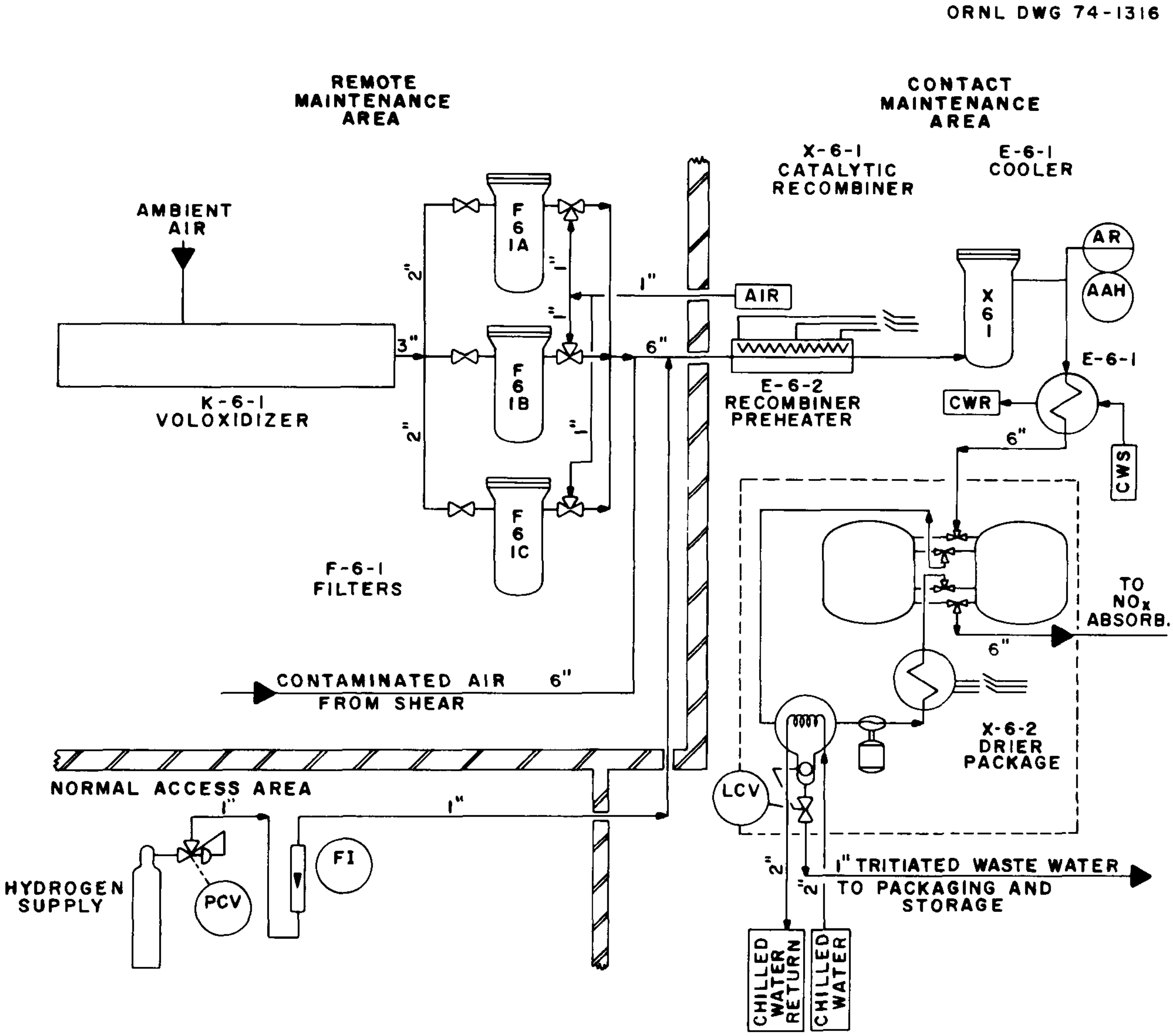

Fig. A-8. Equipment Flowsheet for Nuclear Fuel Reprocessing Plant Radwaste Treatment Case 6 - Reduction in the Release of Tritium - Voloxidation.

Table A-8. Equipment List and Cost for Radwaste Treatment Case 6 - Reduction in Release of Tritium - Voloxidation

Item No.	Descriptive Title	Number of Items	Unit Cost ($)	Total Cost ($)	Reference and/or Source
Remote In-Cell Equipment					
K-6-1	Voloxidizer	1	281,500	281,500	A
F-6-1	Filters	5	2,000	10,000	A
	Subtotal			291,500 (Adjust for balance alignment of nozzles, etc.)	
Contact In-Cell Equipment					
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	3,000	B
	Subtotal			134,000	
	Grand Total			$425,500	