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ENVIRONMENTAL ANALYSIS OF THE URANIUM FUEL CYCLE

PART I - Fuel Supply

U.S. ENVIRONMENTAL PROTECTION AGENCY
Office of Radiation Programs

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Field Operations Division
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FOREWORD

The generation of electricity by light-water-cooled nuclear power reactors using enriched uranium for fuel is experiencing rapid growth in the United States. This increase in nuclear power reactors will require similar growth in the other activities that must exist to support these reactors. These activities, the sum total of which comprises the uranium fuel cycle, can be conveniently separated into three parts: 1) the operations of milling, conversion, enrichment, fuel fabrication and transportation that convert mined uranium ore into reactor fuel, 2) the light-water-cooled reactor that burns this fuel, and 3) the reprocessing of spent fuel after it leaves the reactor.

This report is one part of a three-part analysis of the impact of the various operations within the uranium fuel cycle. The complete analysis comprises three reports: The Fuel Supply (Part I), Light-Water Reactors (Part II), and Fuel Reprocessing (Part III). High-level waste disposal operations have not been included in this analysis since these have no planned discharges to the environment. Similarly, accidents, although of potential environmental risk significance, have also not been included. Other fuel cycles such as plutonium recycle, plutonium, and thorium have been excluded. Insofar as uranium may be used in high-temperature gas-cooled reactors, this use has also been excluded.

The principal purposes of the analysis are to project what effects the total uranium fuel cycle may have on public health and to indicate where, when, and how standards limiting environmental releases could be effectively applied to mitigate these effects. The growth of nuclear energy has been managed so that environmental contamination is minimal at the present time; however, the projected growth of this industry and its anticipated releases of radioactivity to the environment warrant a careful examination of potential health effects. Considerable emphasis has been placed on the long-term health consequences of radioactivity releases from the various operations, especially in terms of expected persistence in the environment and for any regional, national or world-wide migration that may occur. It is believed that these perspectives are important in judging the potential impact of radiation-related activities and should be used in public policy decisions for their control.

Comments on this analysis would be appreciated. These should be sent to the Director, Criteria and Standards Division of the Office of Radiation Programs.

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PART I. FUEL SUPPLY

1.0 Overview of the Uranium Fuel Industry

1.1 Introduction

Because of the rapid increase in the use of light-water-cooled nuclear reactors to generate electricity, there is parallel growth in the basic industry that provides enriched uranium fuel for these operations. This industry includes various operations broadly classified as: (1) milling, (2) conversion of uranium oxide (U_3O_8) to uranium hexafluoride (UF_6), (3) enrichment, (4) fuel fabrication, and (5) radioactive material transportation between these facilities. Radioactive waste products are associated with each of the above activities. This report examines the predominant facilities and operations within these five categories which have the highest potential for environmental impact. Fuel reprocessing also relates to fuel supply; however, this activity has been analyzed separately in part 3 of the environmental analysis of the uranium fuel cycle.

Natural uranium (0.71% uranium-235) ore is mined and milled to a concentrate containing about 85% U_3O_8. The conversion step purifies and converts U_3O_8 to UF_6, the chemical form in which uranium is fed to the enrichment plants. This is followed by isotopic enrichment where the uranium-235 concentration of the uranium feed is increased to the design specification (usually 2 to 4% uranium-235) of the power reactor by a gaseous diffusion

process. The greatest portion of uranium becomes a plant tail impoverished in uranium-235 and is stored in cylinders as UF_6. The enriched UF_6 portion is processed into UO_2 pellets, loaded into alloy tubing, and finally fabricated into individual fuel element bundles. The tube bundles fuel the reactor. These processes are shown in simplified form in figure 1-1 as they relate to a 1,000 megawatt electrical (1000 MW(e)) power reactor. This figure also includes the basic parameters for assumed model plants for each of these operations. Such models are important to a consistent analysis of the environmental impact of the various operations.

The "model" facilities described herein represent the better features of current practice; as such, they are not exemplary facilities and health hazards from their operation are not necessarily acceptable. The model plant sizes are generally similar to those which have been described by Pigford (1) and the Atomic Energy Commission (2). Expressing the model operations in terms of 1,000 MW(e) equivalents gives a common base for the comparison of environmental dose and risk commitments as a function of radioactive waste control technology and cost. Table 1-1 indicates the relation between the various components in terms of 1,000 MW(e) equivalents.

Each step in the fuel supply generates radioactive wastes. Most of this material is controlled becoming solid waste of vari-

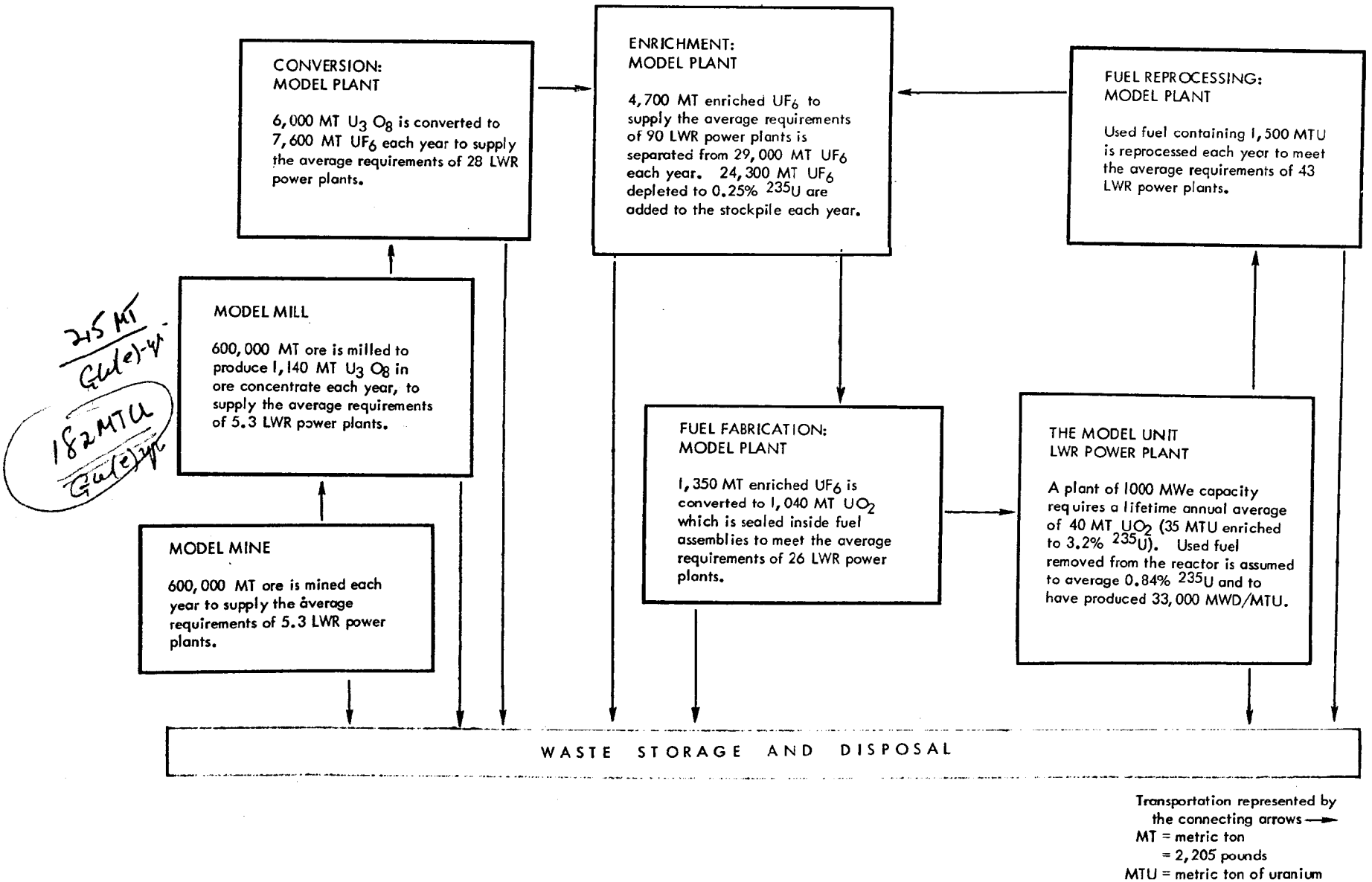

Figure 1-1. Model facility relationships in the uranium fuel cycle for LWR power plants

Table 1-1

Number of LWR's supported by model fuel supply facilities

Type of model fuel supply facility	Number of on-line LWR's Supported by model facility
Mill	5.3
Conversion	28
Enrichment	90
Fabrication	26

ous kinds; a small amount is released under controlled conditions to air and water in most steps in the cycle. The limited number of radionuclides involved in these releases are naturally occurring radionuclides which make up part of the radiation background to which all people are exposed. Mills and conversion facilities will release uranium-238 and its daughters including uranium -234, thorium-230, radium-226, and radon-222. By the time the uranium leaves the conversion facility, it is purified to the point where only uranium-238 uranium-235, and uranium-234 are present.

A projection of the growth of fuel supply facilities from 1980 up to the year 2000 is shown in figure 1-2. The number of such facilities up to about 1980 will be about the same as current industry capacity, which is discussed below. The actual total number of LWR's installed can be estimated by dividing the number of on-line plants by the fraction of time that they are on line (load factor). For the purpose of this report, no electrical power generation based on plutonium fuel was assumed.

1.2 Industry Operations

1.2.1 Mills

The purpose of milling is to obtain U_3O_8 in such a form that it can be converted, enriched, and eventually fabricated into reactor fuel. Milling of uranium ore must be done to separate

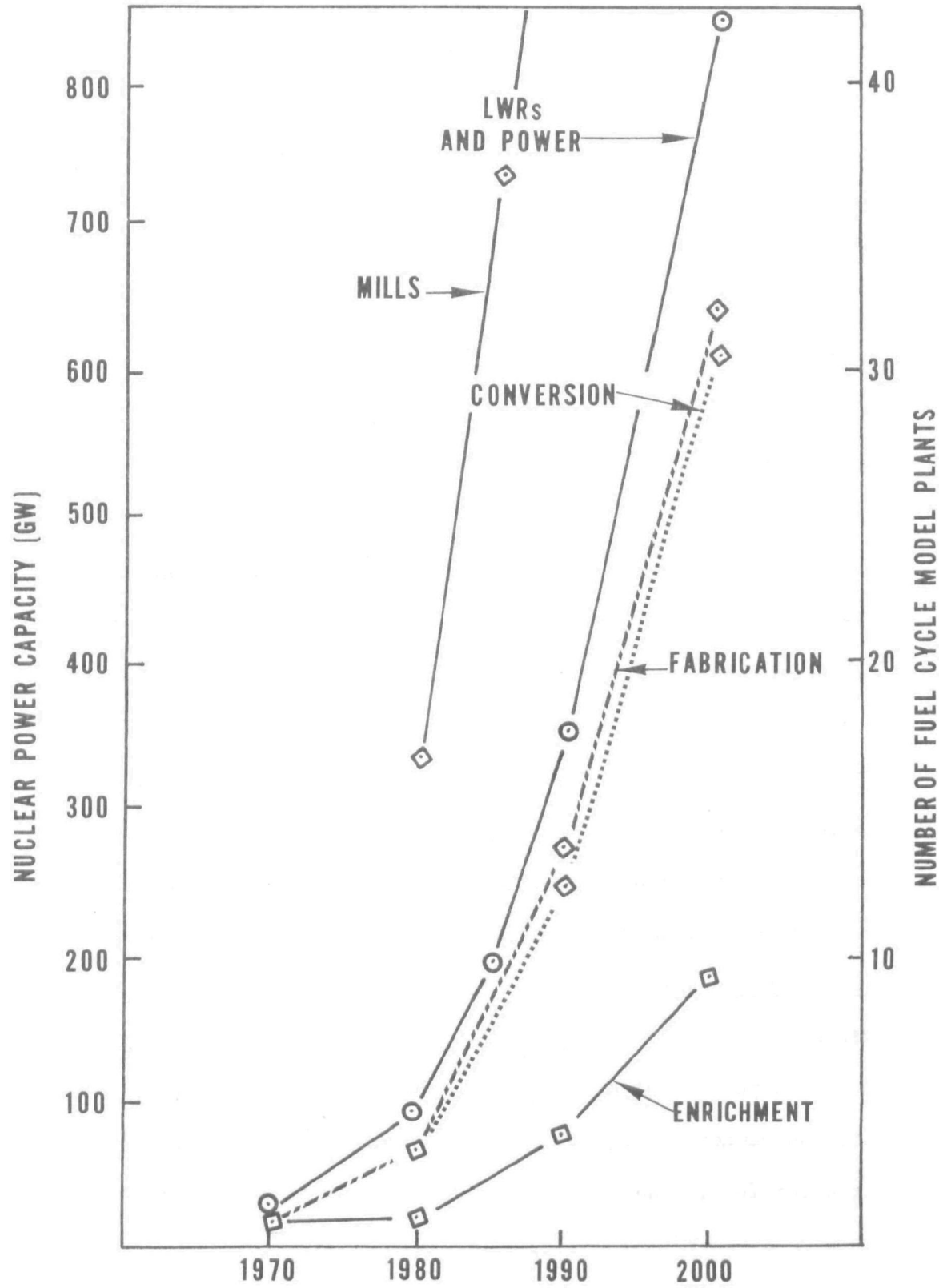

Figure 1-2. Projected nuclear fuel cycle needs

uranium from extraneous rock. This process is accomplished by mechanical crushing of the ore so that it can be dissolved by an acid leach, solvent extraction process. Uranium is purified and concentrated in solvent extraction steps, separated by thickening and centrifuging, and finally calcined and pulverized for packaging in 55 gallon drums for shipment. An alternate process for uranium milling is to use a carbonate leach process, however, there are no particular environmental advantages over the acid leach process.

1.2.2 Conversion Facilities

A typical conversion facility converts uranium to uranium hexafluoride (UF_6) prior to enrichment in sufficient quantities to support 28 1,000 MW(e) reactor equivalents. The two facilities in operation at the present time have capacity considerably in excess of that needed. It is expected that about 30 such facilities will be operational by the year 2000.

Plans are underway at the present time to start the recycling of uranium recovered from the fuel reprocessing step. It appears that each of the fuel reprocessing facilities may have a UF_6 production capability for recycle of recovered uranium. This approach would result in UF_6 production facilities being located at fuel reprocessing sites. A total of 3 or 4 such facilities could be operational by about 1980.

At this time, no significant technological advances can be projected that will affect environmental considerations from these plants. Accordingly, it is projected that UF_6 production plants in 1980 will resemble in most respects the plants of today, but, may be about double their size.

1.2.3 Enrichment Facilities

Enrichment services for the nuclear industry are supplied by the three AEC-owned and contractor-operated gaseous diffusion plants. Essentially any one of the existing plants operated independently would be adequate to meet the current demand for nuclear power plant fuel of about 5,000 metric tons of separative work units (SWU). However, since all three installations are operated in a combined fashion, it is difficult to isolate the environmental consideration of an individual plant.

It is anticipated that the capability of the existing three-plant complex will be increased from about 10,000 MT SWU to 22-27,000 MT SWU by 1980. This increased capability, enough to support about 230 on-line reactors, will be achieved without the construction of major new gaseous diffusion facilities and will be accomplished primarily by improving and upgrading present units. Thus, no new processing plants are required to meet the projected 1980 industry demands. The power demand in the year 2000 will require about 4 such upgraded enrichment plant complexes.

1.2.4 Fuel Fabrication Facilities

One model fuel fabrication plant is sufficient to support 26 on-line 1,000 MW(e) reactors. There are currently 10 commercial plants which are capable of performing all or part of the current fuel fabrication process. According to projections (3), a substantial expansion of production capacity is anticipated in the next 5 years. Existing plants with some shutdowns and some new additions will produce fuel assemblies from an enriched UF_6 feed.

The industry is expected to comprise some 20 to 30 plants by the year 2000. Most of the chemical effluents will be eliminated by future process improvements.

1.2.5 Transportation

Transportation of fuel and waste products will increase substantially over the next several years as the number of reactors and supporting facilities increases. Both rail and truck shipments will be made. The primary mode of exposure to the population will be from direct radiation resulting from the passage of such shipments along shipments routes.

1.3 Environmental Impact of Fuel Supply

With the exception of transportation, the various components of the uranium fuel industry introduce naturally-occurring radioactive materials into the environment through discharges to both air and water. The results are long-term radiation exposure to the skeleton and other organs of the body, especially the lung.

The doses and resultant effects were calculated using the techniques described in appendix A.

The impacts of the various representative facilities are shown in table 1-2 in terms of discharges to the general environment, the resultant individual and population doses committed, and potential health effects. The doses were assumed to be delivered in two principal ways: (1) directly from the effluent as it dispersed in the surrounding air or water environment, and (2) through entrainment in environmental pathways such as ingestion through food chains. Dose commitments through environmental pathways after the original deposition appear to be small in comparision to immediate exposure from the effluents directly, therefore, the environmental considerations for these radionuclides were based on the plant operations themselves; the dose commitments and resultant effects from environmental buildup were found to be minimal by comparison.

The data in table 1-2 indicate that, with the exception of radon, discharges of naturally-occurring materials for all model facilities controlled to current levels of good practice are on the order of 4 Ci/yr or less and that most individual organ doses are grouped below 70 mrem/yr. With the exception of milling, bone doses range from 0.6 to 3 mrem/yr; lung doses from 1 to 70 mrem/yr. Bone doses from milling were calculated to be as high as 13 mrem/yr and lung doses as high as 450 mrem/yr in a case with a short site boundary distance, a situation that occurs

Table 1-2

Radiological impact of model fuel supply facilities - current effluent control procedures

Facility	Uranium and daughters discharged		Maximum radiation dose at boundary (mrem/yr)		Immediate health effects committed (effects/facility-30 yr)	100 yr health effects committed[a] (effects/facility-30 yr)	Total health effects committed (effects/facility-yr)
	Air (Ci/yr)	Water (Ci/yr)	Air	Water			
Mill[b]	0.2	4	450 (lung)	13 (bone)	0.10	0.006	0.1
Conversion wet solvent	0.02	2	30 (lung)	2 (bone)	0.4	0.002	0.4
hydrofluor	0.06	0.8	70 (lung)	1 (bone)	0.2	0.005	0.2
Enrichment	0.05	0.6	1 (bone)	0.7 (bone)	0.1	0.004	0.1
Fabrication	0.005	0.5	10 (lung)	0.6 (bone)	0.1	0.0005	0.1
Transportation					0.02[c]	0	0.02

[a]Calculated assuming the dose is that which follows long term chronic exposure - but only to that fraction of intake resulting from 30 years of plant operations; uranium effluents only.

[b]Radon-222 is not included.

[c]Total health effects associated with 30 years of operation of 1 GW(e) power reactor.

infrequently because of relatively remote siting of mills. The resultant health effects committed for 30 years operation of such facilities range as high as 0.4 effects taking into account 100-year exposures from environmentally deposited uranium and daughters (radon excluded).

The data in table 1-3 indicates that, with the exception of radon, discharges of naturally-occurring materials from model fuel facilities using optimum effluent control technologies are held to less than 1 curie/yr and that individual organ doses are below 15 mrem/yr. Additional effluent controls were required on uranium mills and conversion facilities.

1.4 Control Technology Effectiveness

A number of technologies are available to influence the environmental impact of discharges of uranium and daughter products. Several of these were analyzed in other sections of this report for an optimum control level based on current practice and, wherever possible, a cost-effectiveness of alternative control options. Table 1-4 shows the results of this analysis in terms of the total fuel supply industry. The data were determined for present worth costs of averting annual health effects from operations of representative facilities.

The greatest effectiveness of reducing health effects occurs from control options applied to atmospheric discharges from conversion and fabrication facilities. Holding pond treatment for liquid discharges at conversion facilities and enrichment opera-

Table 1-3

Radiological impact of model fuel supply facilities - optium effluent control procedures

Facility	Optimum level of controls	Uranium and daughters discharged air (Ci/yr)	Uranium and daughters discharged water (Ci/yr)	Maximum radiation dose at boundary air (mrem/yr)	Maximum radiation dose at boundary water (mrem/yr)	total health effects committed (effects/facility-30 yr)
Mill	Additional bag filters and HEPA filters; catch basin and pumps, to present controls	0.004	0	11 (lung)	0	0.002
Conversion wet solvent	Additional bag filters and water treatment to present controls	0.01	0.2	3 (lung)	0.3 (bone)	0.03
Hydrofluor	Additional bag filters to present controls	0.02	0.8	7 (lung)	1 (bone)	0.2
Enrichment	Present controls	0.05	0.06	1 (bone)	0.7 (bone)	0.1
Fabrication	Present controls	0.05	0.5	10 (bone)	0.6 (bone)	0.1
Transportation	Present controls	0	0	------	------	0.02

Table 1-4

Cost effectiveness of control technology for the fuel supply

Operation	System control option	Health effects before control (effects/facility-30 yr)	Health effects after control (effects/facility-30 yr)	Present worth (1970 $)
Milling	Wet dust collector (yellow cake drying)	>1	8×10^{-2}	52,000
	Wet dust collector (ore crushing)	8×10^{-2}	1×10^{-2}	280,000
	HEPA System (yellow cake drying)	1×10^{-2}	3×10^{-3}	10,000
	Bag filter (ore crushing)	3×10^{-3}	2×10^{-4}	280,000
	Clay core dam	>9	0.09	4,000,000
	Seepage return added to clay core dam	0.09	0	600,000
Conversion - Wet solvent	Bag filters	<1.5	0.015	11,000
	2nd bag filters in series	0.015	0.001	11,000
	Settling ponds	>4	0.4	240,000
	Additional chemical treatment	0.4	0.04	14,000,000
Conversion - Hydrofluor	Bag filters	>4	0.04	190,000
	2nd bag filters in series	0.04	0.004	190,000
	Settling ponds	>1	0.15	240,000
	Additional chemical treatment	0.15	0.01	14,000,000
Enrichment	(Airborne releases)[a]	----	-----	----------
	Holding pond	>10	0.13	240,000
Fabrication	Scrubber and prefilter	----	>500	----------
	1 HEPA	>500	0.5	530,000
	2nd HEPA in series	0.5	0.005	530,000
	3rd HEPA in series	0.005	0.0002	530,000
	Settling tank	>0.9	0.09	1,000,000
	Precipitation and flocculation	0.09	0.002	4,300,000

a - Effects not critical compared to the water pathway

tions are next in order of effectiveness. The control options chosen are already operational in most facilities; therefore, even though the effectiveness of reducing health effects for other options is low, industry practice has already achieved these levels of control. In many cases such controls have probably resulted from attempts to recover economically valuable uranium as part of the process involved.

1.4.1 Airborne Discharge Control

Mills. Gaseous and particulate effluents are controlled at mills in three primary places: the ore crushing area, the fine ore bins, and the yellow cake packaging and drying area. Wet dust control systems are generally used in the ore crushing area and the fine ore storage bins; wet scrubbers and bag filters in various combinations are used in the drying and packaging area.

Conversion Plants. The major treatment of gaseous effluents for conversion facilities is a wet scrubber system combined with HF recovery and a H_2 burner. Bag filter systems are used to control uranium dusts in both processes (4,5). The wet solvent extraction system also uses absorption towers for scrubbing out oxides of nitrogen (2). Gaseous effluents escaping from the scrubber and bag filter systems are released through stacks without further treatment.

The scrubber system configuration consists of a venturi section, a liquid-gas separator (demister), fans and associated motors and controls. The venturi section is a vertical conver-

gent-divergent section connected to the separator system by a horizontal elbow. Waste gas from the process systems enters from the top into the converging system. A concurrent flow of scrubber-liquid is introduced into the converging venturi where liquid-gas mixing takes place. The gas and liquid flow to the separator with gaseous flow upward and the liquid exiting downward through a port. The fan draft moves the gas through a final demister after which it is exhausted (6).

Bag filter systems or fabric collectors use woven or felted fabric bags. Dust deposits in the porous fabric and the pressure increases until the dust is removed by manual or automatic means. Rapping, shaking gear, or automatic flow reversal mechanisms are used to remove and collect the dust (7).

Enrichment Facilities. Effluent treatment is part of the recovery of uranium from gaseous streams because of dollar value (8). Gaseous and airborne materials are removed from streams with cold traps and aluminum traps. The efficiency of uranium removal is unknown (8). Descriptions of the trap systems used are not presently available.

Fabrication Plants. The system for conversion of UF_6 to UO_2 is equipped with a scrubber-demister and one high efficiency particulate air (HEPA) filter for dust removal (2). Scrap recovery operations exhaust chemical systems through a scrubber-demister. Each system is equipped with a HEPA filter for uranium dust removal. The process systems handling UO_2 are assumed

to use two HEPA filters independent of those in the conversion and scrap recovery systems.

The effectiveness of high efficiency particulate air filters for uranium was assumed to be that for measurements made on plutonium entering and leaving two banks of HEPA filters (9). The measured fractional removal of plutonium from air passing through two HEPA filters was 0.99999 (9). Assuming uranium aerosols have the same characteristics as plutonium and the first filter is the most effective, the fractional removal was apportioned as 0.999 for the first filter and 0.99 for the second filter. A third HEPA filter in series was 94% efficient for removal of particulate plutonium passing through the first and second filters (9). The usual testing procedures for HEPA filters state removal efficiencies of 99.97% for 0.3 μm dioctyl phthalate (DOP) as a monodispersed aerosol (10). This measurement is a quality assurance test rather than an in-place filter performance test.

Transportation. There are no planned releases of radioactive materials from transportation activities supporting the uranium fuel cycle.

1.4.2 Liquid Discharge Control

Liquid waste streams are treated both to recover uranium and to reduce concentrations of entrained pollutants.

A basic scheme of liquid waste treatment is to mechanically separate solid waste particles from the water of the waste

stream; if necessary, the waste is changed chemically so that it will become particles which can be separated. Residual chemicals in the waste stream must then be neutralized, so that the effluent is neither too acidic nor basic.

Processes used are high efficiency centrifuges, chemical treatments for flocculation, precipitation and neutralization, filtration and settling.

Waste matter which is dissolved is usually in the form of ions which may be separated from the waste stream by passing it through beds of small spheres of chemically treated resins on which the ions are absorbed. Chemicals may be added to the waste stream to change its characteristics (e.g., neutralization) to cause the dissolved waste to become particles which may settle out or be separated from the water by filters or centrifuges. Flocculation is the addition of chemicals which either causes particles to be formed which are large enough to settle rapidly or causes the formation of large, quick-settling particles to which small, slow-settling particles become attached.

Settling basins or ponds are a preferred treatment for liquid waste streams because, once constructed, their use requires little power and maintenance. They are usually an excavation about 4 feet deep and may be several acres in size. Losses of the waste liquid into the ground are usually considered undesirable; if the soil is such that much seepage into the ground is likely, the ponds may be lined with special soils

or artificial liners of plastic or chemically-sealed fabric.

The effectiveness of individual processes varies widely, depending upon the nature of the waste. At one conversion facility, neutralization and settling of liquid waste reduced its uranium content by a factor of 500, its radium content by a factor of 4, and its thorium content by a factor of 180 (11). Waste concentrations are commonly reduced further before release by simple dilution.

1.5 Summary and Conclusions

This analysis of the potential environmental impact of the uranium fuel supply industry, and of the feasibility of minimizing it in the various operations comprising the industry, has involved consideration of:

1. projections of the growth of milling, conversion, enrichment, fabrication and transportation operations through the year 2000,
2. the technology for influencing discharges of naturally-occurring uranium and daughters, including estimates of relative costs based on initial cost and operating expense,
3. the distribution throughout the environment of the radionuclides released during normal operation,
4. estimates of radiation doses to the affected organs of individuals and populations for sites with assumed atmospheric parameters and typical aquatic environments, and

5. health effects expected to be associated with the estimated population doses.

The major conclusions derived from these considerations are as follows.

1. The various operations in the fuel supply industry are, in combination, an integral part of the entire nuclear power industry. Because most of the radionuclides involved in all parts of the industry are naturally-occurring uranium and daughters, the industry can be treated as a combined operation for purposes of evaluating its contribution to the overall discharge of radionuclides to the environment and their effects.

2. The quantities of uranium and daughter products discharged, with the exception of radon and radon daughters, can be maintained to less than 1 curie per year for representative plants with currently available and commonly used control technologies. Consequently, if projections of nuclear power are substantially correct, the overall discharges of such materials would not be expected to be large through the year 2000 since the various facilities support several reactor requirements.

3. Control technology exists to avert health effects from discharges of uranium and daughter products although many such technologies are a part of uranium recovery processes. The most cost-effective system to apply for risk reduction

appears to be HEPA filters at fabrication plants followed by bag filters at conversion facilities and holding ponds for liquid effluent treatment at various facilities.

4. The consequences of the environmental buildup of uranium and daughter products, although many have long half-lives, do not appear large. The major route of exposure is direct interaction of populations with the effluents immediately following discharge. Doses to individuals and organs (with the exception of doses from radon-222) have been estimated to be less than about 70 mrem/yr, with the exception of lung dose from airborne discharges from a uranium mill, which for short site boundary distances can go up to about 450 mrem/yr. These doses can be held to less than 15 mrem/yr by additional commonly used control technologies.

2.0 Uranium Milling

2.1 General Description of the Milling Process

A uranium mill extracts uranium from ore. The product is a semirefined uranium compound (U_3O_8) called "yellowcake" which is the feed material for the production of uranium hexafloride (UF_6). About 20 mills are currently operating in the United States with average outputs ranging from 400 to 7,000 MT/yr. These mills are characteristically located in arid, isolated regions of the west.

Eighty percent of the yellowcake is produced by a sulfuric acid leach process; the remainder by a sodium carbonate, alka-

line leach process. The principal steps in the acid-leach, solvent-extraction process diagrammed in figure 2-1, are:

a. Ore is blended and crushed to pass through a 2.5 cm (1 inch) screen. The crushed ore is then wet ground in a rod or ball mill and is transferred as a slurry to leaching tanks.

b. The ore is contacted with sulfuric acid solution and an oxidizing reagent to leach uranium from the ore. The product liquor is pumped to the solvent-extraction circuit while the washed residues (tailings) are sent to the tailings pond or pile.

c. Solvent extraction is used to purify and concentrate the uranium.

d. The uranium is precipitated with ammonia and transferred as a slurry.

e. Thickening and centifuging are used to separate the uranium concentrate from residual liquids.

f. The concentrate is calcined and pulverized.

g. The concentrate or yellowcake is packaged in 208 liter (55 gallon) drums for shipment.

Large amounts of solid waste tailings remain following the removal of the uranium from the ore. A mill may generate 1,800 metric tons per day of tailings solids slurried in 2,500 metric tons of waste milling solutions. Over the life time of the mill, about 100 hectares (250 acres) may permanently be committed to

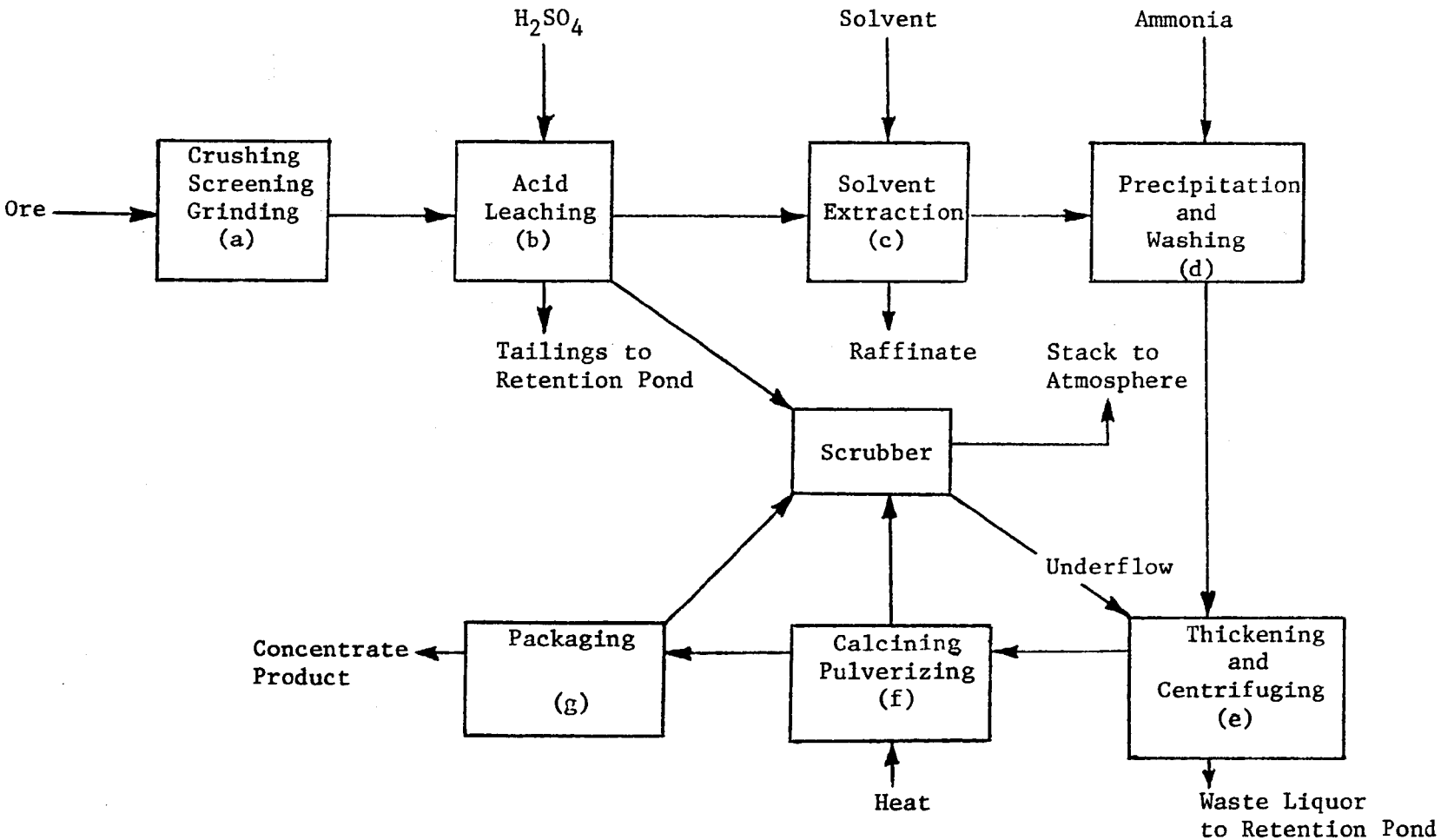

Figure 2-1 Uranium mill process diagram

store this material. These "tailings piles" will have a radiological impact on the environment through the air pathway by continuous discharge of radon-222 gas (a daughter of radium-226), through gamma rays given off by radon-222 and daughters which undergo radioactive decay, and finally through air and water pathways as radium-226 and thorium-230 are blown off the pile by wind and leached from the pile into surface waters.

The radiological impact of these piles requires special considerations; therefore, they will be treated separately in section 2.7.

2.2 The Model Mill

A system of model plants has been assumed for each segment of the nuclear fuel cycle in order to achieve a common base for the comparison of radiation doses, committed health effects, and radioactive effluent control technology.

The model mill is defined in terms of contribution to the nuclear fuel cycle that is consistent with current and projected commercial industry practice (1). Characteristics of the model mill are assumed to be:

1. 600,000 MT ore milled per year,
2. 1,140 MT U_3O_8 as yellowcake produced per year,
3. use of the acid leach process,
4. tailings retention pond system which uses a clay core earth dam and local topographic features of the area to form the impoundment,

5. seepage through the dam into a river with a flow rate of 14 m^3/s,
6. production to supply the requirements of 5.3 LWR power plants on line, i.e., 5.3 GW(e) power/yr, and
7. location in a western state in an arid, low-populated density region.

Radiation dose rates and health effects that might result from the discharges of radioactive effluents from the model mill were calculated using standard (χ/Q) values, dose conversion factors, model pathways, and health effect conversion factors that are similar to those for other facilities in this discussion of the fuel supply cycle. The factors and assumptions are discussed in appendix A.

The operating lifetime of a uranium mill is commonly from 12 to 15 years, depending upon the local ore supply and the demand for uranium. In a few instances, the operating lifetime may be longer and allowances are sometimes made for that possibility if it appears feasible. For this model mill, an operating lifetime of 20 years has been selected. Discussion of health effects from 30 years of plant operation is only for convenience in comparison with other operations in the fuel supply system.

2.3 Releases of Radioactive Effluent from Uranium Mills

The radioactivity associated with uranium mill effluents comes from the natural uranium and its daughter products present in the ore. During the milling process, the bulk of the natural

uranium is separated and concentrated while most of the radioactive daughter products of uranium remain in the uranium-depleted residues that are pumped to the tailings retention system. Liquid and solid wastes from the milling operation will contain low level concentrations of these radioactive materials. Airborne radioactive releases include radon gas and particles of the ore and the product uranium oxide. External radiation levels associated with uranium milling activities are low, rarely exceeding a few mrem/h even at surfaces of process vessels.

2.3.1 Airborne Releases

The radiological releases from uranium milling operations include airborne particulates and vapor. Dusts containing uranium and uranium daughter products (thorium-230 and radium-226) are released from ore piles, the tailing retention system, and the ore crushing and grinding ventilation system. Natural uranium is released from the yellowcake drying and packaging operations as entrained solids.

Radon gas is released from the leach tank vents, ore piles, tailings retention system and the ore crushing and grinding ventilation system. There is no practical method presently available to prevent the release of radon gas from uranium mills.

Table 2-1 gives the estimated maximum release rates and conservative estimates of site boundary concentrations considering all potential sources of airborne dust fumes and mists as predicted for the Highland Uranium Mill in Wyoming (2,3). The

Table 2-1

Predicted airborne releases of radioactive materials from the Highland Uranium Mill

Radionuclide	Release rate (Ci/yr)	Site boundary A [a] Air concentration (pCi/m^3)	Site boundary B [b] Air concentration (pCi/m^3)
Uranium-natural	0.1	0.003	0.0004
Thorium-230 (insoluble)	.06	.001	.0001
Radium-226 (insoluble)	.06	.001	.0001

[a]Distance to site boundary A assumed to be 800 m (2,600 ft) west of mill.

[b]Distance to site boundary B assumed to be 5,200 m (12,700 ft) east of mill.

capacity of the Highland Mill is about 1,200 MT/yr of yellowcake.

2.3.2 Waterborne Releases

The following discussion refers to the best of current procedures of handling mill liquid wastes where these wastes plus tailings are stored in a tailings retention pond system which uses an impervious clay cored earth dam and local topographic features of the area to form the impoundment.

The liquid effluent from a mill (acid-leach process) consists of waste solutions, from the leaching, grinding, extraction, and washing circuits of the mill. The solutions, which have an initial pH of 1.5 to 2, contain the unreacted portion of the sulfuric acid used as the leaching agent in the mill process, and sulfates and some silica as the primary dissolved solids with trace quantities of soluble metals and organic solvents. This liquid is discharged with the solids into the tailings pond.

Concentrations of radioactive materials predicted in the 2,500 MT/day of waste liquor from the Highland milling plant are shown in table 2-2. Radioactive products of radon decay may also be present in small concentrations. Since the concentrations of radium-226 and thorium-230 are about an order of magnitude above the specified limits in 10 CFR 20, considerable effort must be exerted to prevent any releases of this material from the site. The waste liquor is, therefore, stored in the tailings retention pond which is constructed to prevent discharge into the surface

Table 2-2

Concentrations of radioactive effluents in waste liquor from a uranium mill

Radionuclide	Concentration (pCi/l)
Uranium-natural	800[a]
Radium-226	350
Thorium-230	22,000

[a]About 0.001 g/ml.

water system and to minimize percolation into the ground. This is a continuing potential problem requiring monitoring programs to insure that there is no significant movement of contaminated liquids into the environment.

If an earth-fill, clay-cored dam retention system serves as a collection and storage system for the liquid and solid process wastes generated in the mill, it will permit the evaporation of most of the contained waste liquids and serve as a permanent receptacle for the residual solid tailings. However, after the initial construction of the retention system, it is to be expected that there will be some seepage of radionuclides through and around the dam (2,3). It has been estimated that this seepage will diminish over a period of about 2 years because of the sealing effect from accumulation of finer particles between the sandstone grains. On the other hand, sealing may not occur. Examples of the total quantities of radionuclides that are estimated to be released under such conditions are shown in table 2-3. Radium-226 is a radionuclide of concern in this case. Radium-226 levels as high as 32 pCi/l (4) have been found in seepage from current operating mills. Assuming a seepage rate of 300 liters per minute (80 gpm), the concentration of radium-226 seeping into a stream of 140 liters per second (5 cubic feet per second) is approximately 1 pCi/l which is 33% of the current drinking water standards. In the applicants environmental report for the Highland Uranium Mill, a concentration of 350 pCi/l was

Table 2-3

Estimates of quantities of radionuclides seeping through the impoundment dam of a uranium mill initially and at 2 1/4 years

Radionuclide	Initial seepage per day	Seepage per day after 2 1/4 years
Uranium	350 μCi	35 μCi to 3.5 μCi
Thorium-230	9,600 μCi	960 μCi to 96 μCi
Radium-226	150 μCi	15 μCi to 1.5 μCi

assumed bringing the concentration of radium-226 in such a stream up to 12 pCi/l. The Highland Uranium Mill is also estimated to release to the tailings pond 22,000 pCi/l thorium-230 and trace quantities of short-lived radon daughter products.

As an additional example, the analysis of plant tailings effluents for the Humeca Uranium Mill is given in table 2-4 (5).

The radiological significance of seepage from tailings ponds will depend on the location of the pond. In arid regions, the seepage may evaporate before leaving the site, leaving the radioactivity entrained and absorbed on soil. Should the tailings pond be located near a river, minor leakage might be diluted sufficiently by the additional river water to meet relevant drinking water standards. Discharge of pond seepage into streams providing insufficient dilution and not under the control of the licensee would not be acceptable. In such a case, a secondary dam may be built below the primary dam to catch the seepage which may then be pumped back into the tailings ponds.

2.3.3 Radioactive Effluents from a Model Uranium Mill

Because regulations have not required uranium mills to report the total amounts of each radionuclide discharged per year, the source terms chosen for the model mill are based on limited information (2,3,5). Source terms listed in table 2-5 for the model mill are believed to be reasonably accurate estimates of the quantities of radioactive materials discharged to air and water path-

Table 2-4

Analysis of plant tailings effluents
from the Humeca Uranium Mill
(alkaline leach process)

Radionuclide	pCi/l
Radium-226	10 to 2,000
Thorium-230	0.1
Uranium-238	4,000

Table 2-5

Discharge of radionuclides to the environment from a model uranium mill (acid leach process)

Radionuclide	Pathway	Possible chemical states	Source term (Ci/yr)
Uranium	Air	oxides	0.1
Radium-226	Air	a	0.06
Thorium-230	Air	a	0.06
Uranium	Water	UO_2^{++}	0.1
Thorium-230	Water	Th^{++}	3.5
Radium-226	Water	Ra^{++}	0.06

[a]Not known.

ways by real facilities but experimental confirmation of these values is not yet available. Because uranium is discharged to the air pathway as ore dust and as calcinated yellowcake, it will be considered as an insoluble aerosol. Radium-226 and thorium-230 discharged as ore dust will also be considered insoluble aerosols. Seepage of process liquors from tailings ponds will be assumed to be discharged directly into a river (appendix A).

2.4 Radiological Impact of a Model Mill

Estimates of the radiation doses received by individuals in the vicinity of a model mill from routine effluents are given in tables 2-6 and 2-7, for doses through the air and water pathways, respectively. The estimated aggregate doses to the population in the vicinity of a mill are given in table 2-8. The models for the dispersion and dose calculations are discussed in appendix A.

2.5 Health Effects Impact of a Model Mill

The expected cost in health effects to members of the general population in the vicinity of a model mill are presented in table 2-9. The models used for the calculation of health effects are given in appendix A.

2.6 Control Technology

2.6.1 Airborne Effluent Control Technology

Hazardous airborne gaseous and particulate wastes are generated in the milling operation from a number of different sources. The major areas of the milling operations in which gaseous and

Table 2-6

Radiation doses to individuals in the general population in the vicinity of a model mill, from inhalation

Source term (Ci/yr)	Critical organ	Maximum dose to critical organ[a,b]	
		Individual at plant boundary (mrem/yr per facility-yr)	Individuals within 80 km (mrem/yr per facility-yr)
0.1 (uranium)	Lung	190	0.04
0.06 (radium-226)	Lung	130	0.03
0.06 (thorium-230)	Lung	130	0.03
	Total	450	0.10

[a]Each facility supports 5.3 on-line 1 GW(e) power plants.

[b]Listed mrem/yr radiation dose will result from each year of facility operations.

Table 2-7

Radiation doses to individuals in the general population in the vicinity of a model mill, from drinking water

Source term (Ci/yr)	Critical organ	Maximum dose to critical organ[a,b]	
		Individual at plant boundary (mrem/yr per facility-yr)	Individual within 300 km (mrem/yr per facility-yr)
0.1 (uranium)	Bone	2	0.2
	Soft tissue[c]	0.2	0.02
0.06 (radium-226)	Bone	2	0.2
	Soft tissue[c]	0.06	0.006
3.5 (thorium-230)	Bone	9	0.9
Total dose --- bone		13	1.3
Total dose -soft tissue[c]		0.3	0.03

[a]Each facility supports 5.3 on-line 1 GW(e) power plants.

[b]Listed mrem/yr radiation dose will result from each year of facility operations.

[c]Average radiation dose to all organs except bone.

Table 2-8

Aggregate dose to the general population
in the vicinity of a model mill

Source (Ci/yr)	Pathway	Critical organ	Aggregate dose to population (dose to critical organs) (rem/yr per facility-yr)
0.1 (uranium)	Air	Lung	2
0.06 (radium-226)	Air	Lung	2
0.06 (thorium-230)	Air	Lung	2
	Total dose to lung		6
0.1 (uranium)	Water	Bone	10
		Soft tissue	1.0
0.06 (radium-226)	Water	Bone	8.1
		Soft tissue	0.3
3.5 (thorium-230)	Water	Bone	39
		Soft tissue	---
	Total dose to bone		57
	Total dose to soft tissue		1

Table 2-9

Committed health effects to the general population
in the vicinity of a model mill
(acid leach process)

Pathway	Critical organ	Mortality[b] (H.E./facility-yr)	Nonfatal effects[b] (H.E./facility-yr)	Genetic effects (H.E./facility-yr)
Air	Lung	0.0003	0	0
Water	Bone	0.0015	0.0009	0
	Soft tissue	0.0002	0.0002	0.0002
	Totals	0.002	0.001	0.0002

Total health effects for 30 years of plant operations is 0.1 effects/facility-30 yr.

[a]Each plant will support 5.3 on-line 1 GW(e) power reactors.

[b]Listed health effects will result from each year of facility operations.

H.E. - health effect

particulate effluents must be controlled are the ore crushing area, the fine ore bins, and the yellowcake packaging and drying area. Current practice involves the use of wet dust control systems for the ore crushing area and fine ore storage bins and wet scrubbers with bag filters for the yellowcake packaging and drying areas. A 95% efficient wet dust control system for the crusher costs approximately $14,000. Including the cost of installation, the total cost could range from $30,000 to $50,000. The fine ore bin can be controlled with a similar device except that it is smaller because the air flow rate from the bins is less. The purchase price of such a wet collecting device is $3,000 to $7,000. The total cost of the unit including installation would range from $6,000 to $11,000.

Control of effluents produced in the yellowcake packaging and drying system is currently achieved by using wet scrubbers or wet scrubber-bag filter units. A wet scrubber-bag filter unit currently in use in a modern mill is 99% efficient and can handle a flow rate of 1 m^3/s (2,250 cfm) of effluent gases (2). The purchase price of such a unit is approximately $4,000 to which an installation cost of from $8,000 to $12,000 is added. The total cost is about $15,000.

Older mills use only wet scrubbers for control of off-gases from the yellowcake drying furnace and a bag filter at the yellowcake drumming station. The scrubbers used on furnace off-gases cost from $8,000 to $12,000 including installation and

have an efficiency of 98.5%. Bag filters currently in use range from $5,000 up, including installation. A bag filter effectively controls 99.8% of the fugitive yellowcake. The total cost for scrubbing and filtering of the yellowcake system ranges from $13,000 to $17,000 and up and gives an effective system control of about 99%. Dust collection and control efficiencies can vary depending upon the type of equipment and the power input. By increasing the power applied to a given control device, the pressure drop across the collector increases resulting in an increased efficiency of particulate removal. The increase in power use results in an increased cost of operation and may also result in increased maintenance costs. Assuming that a given unit is designed for such efficiencies, the capital cost for a control unit would be similar whether the unit is to be operated at 90% efficiency or 99%, but operating costs would vary depending upon the cost of power, the desired efficiency and the frequency of maintenance. Additional control efficiencies and costs can be calculated for the model mill and these are found in table 2-10.

Other sources of gas and dust which can be controlled are the open pit mine haul roads and the ore storage and blending piles. In some instances the liquid content of the ore as mined has been said to be sufficiently high to eliminate most dust formation in the ore storage and blending area; due to insufficient information, this case will not be considered at present beyond stating that the problem appears significant, and it can

Table 2-10

Cost and reduction of effluents for control technology for mills

Control Method	Capital cost (1970 dollars)	Annual operating cost (1970 dollars)	Effluent percent reduction
A. Liquids and solids			
1. Diked solids retention (total liquid discharge)	800,000		94.30
2. (1.) + neutralization		400,000	99.50
3. (2.) + $BaCl_2$ treatment		510,000	99.90
4. Clay core dam retention system	1,800,000	90,000	99.20
5. (4.) + neutralization		490,000	99.92
6. (5.) + $BaCl_2$ treatment		600,000	99.99
7. (4.) + seepage return			
a. wells & pump	9,400		100.00
b. basin & pump	8,000		100.00
8. (4.) + pond lining	3,200,000		100.00
B. Gaseous (Crusher-fine ore bins)			
1. 95% control	50,000	3,500	95.00
2. 99% control	50,000	3,900	99.00
C. Gaseous (Yellowcake packaging & drying)			
1. 98.5% control (drying only)	10,000	350	99.00
2. 99.8% control (packaging)	5,000	280	
3. 99.30% control	48,000	1,000	
D. Gaseous (Tailing stabilization)[b]			
1. No stabilization	0		0.00
2. 0.6 meters (2 feet) of cover	250,000[c]		5.0-10.0
3. 6 meters (20 feet) of cover	2,400,000[c]		90.00

Table 2-10 (continued)

Cost and reduction of effluents for control technology for mills

[a]This figure is based on a total amount of radium discharged per day that is available for dissolution to the water. This assumes 3% of initial radium in the ore will be dissolved in milling operation. The dissolved 3% gives a starting figure from which reductions can be made.

[b]Stabilization of the tailings pile is being considered from the standpoint of radon reduction. A major advantage of stabilization which is not being considered is the elimination of wind erosion and the decreasing of water erosion from the tailings pile.

[c]The figures listed are for stabilization of a depression-fill tailings pile and would have to be increased for a surface pile because of additional costs for contouring for side slope reduction.

be controlled in principle by sprinkling. Dust generation on the ore haul road can also be controlled by sprinkling. The model mill is assumed to utilize two sprinkling trucks at a cost of approximately $15,000 per item or a cost of $1,500 per year plus maintenance.

2.6.2 Waterborne Effluent Control Technology and Solid Waste Control Technology

New mills in the Rocky Mountain area are using impoundment technology to try to reach zero liquid discharge levels. Recent practice for treatment of solid and liquid wastes is to select a natural ravine which has three basic qualifications for waste storage: (1) limited runoff, (2) dammable downstream openings, and (3) an underlying impermeable geologic formation. Diversion systems (dams and canals) are used to limit the runoff area emptying into the storage basin to prevent flooding of the ravine during a 50-100 year maximum rainfall occurrence. The tailings dam, which should be clay cored, is keyed into the underlying impermeable formation, which, in one example, is a low porosity shale. Tailings solids slurried in waste process liquids are pumped to the impoundment reservoir for storage and liquid reduction. Liquid reduction is accomplished primarily by evaporation, but also by seepage through the dam, the reservoir walls and floor. By filling a dammed natural depression with tailings a relatively flat, stable contour is achieved.

Impoundment of solids is being accomplished in older mills

merely by construction of a dike with natural materials and filling the diked area with slurried tailings. When full, the height of the dike is increased with dried tailings to accommodate even more waste material. Process liquids which overflow the tailings dike or seep through the dike are sometimes routed through a treatment system and discharged to the environment. The diking procedure, which is less costly initially, creates an above-ground pile of tailings which is difficult and costly to stabilize.

Stabilization of a tailings pile involves grading the tailings area to lessen side slopes, establishing drainage diversion, covering with nonradioactive material, and revegetating the area. In semiarid regions it may be necessary to irrigate the pile to achieve initial vegetation growth. Other types of stabilization may also be feasible. One method involves the covering of the tailings with large aggregate gravel from a river bottom. Silt fines which accompany the river gravel will blow away in a short time leaving what is effectively a wind proof rip rap, thus significantly reducing or eliminating migration of the tailings outside the controlled area. Assuming costs of 38¢ per cubic meter (29¢ per cubic yard) of cover material 0.6 m thick and $185 per hectare ($75 per acre) to seed, the cost of stabilizing a depression-fill pile would be about $2,500 per hectare ($1,000 per acre). The total cost for stabilization of a 100 hectare pile would be $250,000. The cost associated with stabilizing a diked surface pile is significantly higher and probably less effective because

of difficulties faced in grading, covering, and revegetating the potentially steep side slopes (3). The maintenance associated with perpetual care of a stabilized dike system would also be higher than that for the depression fill system if there is collapse of side slopes and possibly inadequate drainage of precipition from the pile. A rough calculation, performed by the Oak Ridge National Laboratory, indicates that covering a tailings pile with 6 meters (20 feet) of dirt will reduce the radon emanations by about 90%. A 6 meter cover and seeding would cost approximately $23,500 per hectare ($9,500 per acre) or $2,375,000 for a 100 hectare pile. Two waste control options are thus available for liquid and solid waste control from a mill. Either the liquids are treated to remove all but insignificant amounts of radioactivity before being discharged into a river or the liquids can be totally impounded in ponds so that further treatment is not necessary. Solids must be impounded in both procedures.

A mill operator can route the tailings, which are slurried in process liquids, into a settling pond. Following settling of the solids, the process liquids will overflow into a collection basin below the settling pond. Waste liquids then can be treated with lime to neutralize the acid picked up in the leaching operation. (Process liquids from an alkaline leach mill can be neutralized with $FeSO_4$-H_2O). Assuming an average dosage of 136 kg (300 pounds) of sulfuric acid to leach each metric ton of ore milled, and a lime cost of 0.44¢ per kilogram of sulfuric acid

applied, the cost of neutralization chemicals is calculated to be $360,000 per year for the model mill (7,8). (A dosage of 18 kg of sulfuric acid/metric ton of ore would have an associate neutralization cost of $48,000 per year). The total costs of neutralization include purchasing, hauling, and applying the lime to the process liquids is estimated to cost around $400,000 per year. Neutralization will effectively decrease the thorium concentration in the process liquids by 100% and the radium by 90% since both are insoluble in neutral or alkaline carbonate solutions.

If additional reduction of dissolved radium levels in the process liquids from the mill is desired the waste can be further treated with barium chloride ($BaCl_2$). This treatment will cause the radium to be precipitated as barium-radium sulfate. Assuming an average liquid waste production of 1,250 liters per metric ton of ore processed, a barium chloride dosage of 300 mg/l of waste, and a cost of $23 per 50 kg, the yearly cost of barium chloride would be $104,000. The total cost of barium treatment would be around $110,000 or effectively equal to the purchase price of the barium chloride since hauling and application equipment cost is estimated to be only a few thousand dollars. This treatment removes 90% of the dissolved radium in the liquids so that when coupled with neutralization, a 99% radium reduction is achieved. If the concentrations of radioactivity are small enough, the process liquids are then discharged to a nearby river.

A second option involves the construction of a complete tail-

ings retention system. This system with its clay core dam and impervious underlying geological formation is considered to be effectively a 100% liquid holdup system, although seepage can be expected. The retention system, however, alleviates the need for neutralization and barium chloride treatment except to the extent that the radioactivity concentrations of the seepage would be lessened if this treatment were carried out. Assuming that the clay-cored retention dam was 460 meters long, 30 meters high, and 6 meters at the crest, has face slopes of 2 to 1 and 2.5 to 1, and a cost of about $2.00 per cubic meter to construct, the total construction cost would be approximately $1,750,000.[1] This control would eliminate all effluents and therefore all exposures to uncontrolled areas with the exception of those exposures due to radon-222 and its daughter products.

Two methods for seepage collection and return are being considered for new mills. Seepage has been estimated to occur from a clay-core retention dam at a rate of 300 liters per minute. In that situation where an impermeable geological formation underlies the retention system, seepage can be collected in a catch basin located at the foot of the dam. The collected seepage can be pumped back into the retention pond thus eliminating release to the offsite environment. Assuming $0.38 per cubic meter of earth moved, the collection basin would have an approximate

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[1]Personal communication, Richard Bock, U.S. Bureau of Reclamation, Department of the Interior, Washington, D.C. (April 1973).

cost of $800 (2). This cost would vary depending upon whether it was necessary to line the basin. If necessary, lining could be either clay or synthetic in nature. Synthetic lining, at an approximate cost of $3.20 per square meter including instalation would be $32,000 per hectare.(about $13,000 per acre). Total cost with lining for a 460 m x 6 m x 1.5 m catch basin would be about $6,000. In that situation where either an underlying impermeable geological formation is not existent or is not continuous, vertical seepage may occur to the underlying ground water formation. Wells may be drilled downstream of the retention system into the subsurface formations where seepage would collect and this water is pumped back to the retention system. Such a system requires specific favorable subsurface conditions. If four wells were drilled to an average depth of 15 m (50 feet), the cost per well would be about $500 or $2,000 total for drilling.

Pumps for the collection systems would be deployed in the basin and in each well. Two pumps of about 200 liter per minute capacity would be utilized in the basin and one pump of about 100 liter per minute capacity could be utilized in each well. Pump costs are approximately $1,700 for a 200 liter per minute (50 gpm) unit and $1,500 for 100 liter per minute (25 gpm) units. It is necessary that the pump be able to withstand acid corrosion and, therefore, must be made of stainless steel.

The total cost of seepage collection would then be $9,400

for the basin system or $8,000 for the well system or $470 a year and $400 a year, respectively.

Another method to be considered to eliminate all liquid effluents would involve the complete lining of the tailings retention system. For the model mill, the area to be utilized for the tailings retention system is 100 hectares (250 acres). Assuming the lining of the total surface area at a cost of $3.20 per square meter ($0.30 per square foot) including installation, the total expenditure for lining would be $3.2 million (9). Exercise of this option would completely alleviate the need for neutralization, barium chloride treatment or any other liquid waste treatment, and would result in total containment of the liquid with the exception of evaporation. The dose attributable to process liquids received by a person in the uncontrolled area would be zero.

2.6.3 Effluent Controls for the Model Mill

Current effluent control systems for the model mill were assumed to be:

1. ore crushing area and fine ore storage - wet dust control system (95% control)
2. yellow cake packaging and drying areas - wet scrubbers and bag filters (99.3% control)
3. liquid and solid wastes - clay core dam retention system (99.2% control)

The additional add-on controls for airborne releases were assumed to be bag filters rated at 95% effectiveness and HEPA filters rated at 99% effectiveness. Liquid releases seeping through the dam were assumed collected in a basin and returned to the tailings pond by pumps.

The radiological impact of radioactive effluents versus controls for the model mill is shown in tables 2-11 and 2-12.

2.7 Uranium Mill Tailings Piles

2.7.1 Introduction

Large scale milling of uranium ore began in the United States in the late 1940's and will continue indefinitely. The average time of operation of a uranium mill is about 12 to 15 years. As a result there are currently as many inactive mills as active mills. When the uranium is extracted from the ore, more than 99 percent of the ore material becomes the mill wastes or tailings, a slurry of sandlike material in waste solutions. The tailings are pumped to a nearby location where the solids settle out and soon accumulate to form a tailings pile. Each location where a mill is operating or has operated has an accumulation of tailings. As of 1970, there were more than 80 million metric tons of tailings occupying more than 850 hectares (2,100 acres) of land.

More than 97 percent of the radioactive decay products of uranium and about 4 percent of the uranium from the ore remain in these tailings. The concentration of radium-226 in the

Table 2-11

Radiological impact of airborne effluents vs controls for a model uranium mill

Controls	Source term (Ci/yr)	Max. dose to the individual (mrem/yr, lung)	Total health effects 30 yr plant operations (H.E./facility-30 yr)	Capital cost (1970 $/fac.)	Annual operating cost (1970 $/fac.)	Present worth (1970 $/fac.)
None	>15	>30,000	>1	0	0	0
Wet dust collector on yellowcake packaging and drying[a]	1.6	3,600	8×10^{-2}	15,000	630	52,000
Wet dust collector on crusher and ore bins[a]	0.2	450	1×10^{-2}	100,000	7,400	280,000
Additional HEPA System on yellowcake packaging and drying area	0.07	150	3×10^{-3}	1,400	600	10,000
Additional bag filter on crusher and ore bins	0.004	10	2×10^{-4}	100,000	7,400	280,000

[a] Current levels of controls.
H.E. - health effects

Table 2-12

Radiological impact of waterborne effluents vs controls for a model uranium mill

Controls	Source term (Ci/yr)	Max. dose to the individual (mrem/yr, bone)	Total health effects 30 plant operations (H.E./facility-30 yr)	Capital cost (1970 $/fac.)	Annual operating cost (1970 $/fac.)	Present worth (1970 $/fac.)
None	> 400	>1,300	>9	0	0	0
Clay core dam[a] retention system	4	13	0.09	1,800,000	90,000	4,000,000
Seepage return system with clay core dam[b]	0	0	0	1,809,000	90,000	4,600,000
Lined clay core dam retention system	0	0	0	5,100,000	90,000	11,000,000

[a]Current level of control if seepage is to a river.

[b]Current level of control if seepage would be to offsite stream.

H.E. - health effects

tailings averages about 700 pCi/g, indicating an inventory of about 56,000 Ci of radium-226 in these piles. The radon-222 decay product of radium-226 emanates from these piles at an average rate of about 600 pCi/m^2-s, representing a total release of radon gas of more than 150,000 Ci/yr. The tailings piles release radioactive material to the air as radon gas, as airborne particulates, and as waterborne radionuclides leached out by precipitation, surface runoff, and the waste solutions. Sufficient radioactivity is in the tailings to create a weak field of gamma radiation in the immediate vicinity of the tailings. Because of the presence in the tailings of thorium-230, which by its decay maintains the radium inventory, the radioactivity in the tailings will remain almost constant for thousands of years.

2.7.2 Source Term

Uranium is widely distributed in nature, but generally in concentrations too small for economic recovery. Although uranium is found in a number of different types of rocks, the ores most commonly mined in the United States are ground water deposited uranium minerals in sandstone. The uranium is frequently accompanied by vanadium in these ores. The only thorium commonly found in these ores is that resulting from the radioactive decay of uranium.

Most of the uranium ores being milled in this country are sandstones consisting of silica grains poorly cemented together with materials such as calcium carbonate and containing some

clay minerals. In the milling process, the sandstone is broken down and after the uranium values (and sometimes vanadium) are extracted, essentially the whole original mass remains as tails. The tails consist of the original sand grains plus slimes made up of the clay minerals and other materials from between the sandstone grains. The slimes may constitute 15 to 60 percent by weight of the tailings solids; commonly about one quarter. The slimes may have concentrations of radioactivity 3 to 20 times those in the sands, and commonly have about three quarters of the total radioactivity in the tailings.

The radioactivity in uranium ores is not uniform from ore to ore. It depends, of course, on the amount of uranium in the ore. It also depends upon the length of time the uranium has been in the ore; for example, if the uranium has been in place ten million years, secular equilibrium may have been reached. This would be so if no processes of removal or addition, other than radioactive decay, were changing the quantities of radionuclides present. Secular equilibrium means, in this case, that one curie of uranium-238 is accompanied by one curie of each decay product in its decay chain, i.e., one curie of thorium-230, one curie of radium-226, etc. If the uranium has been in place only 100,000 years, secular equilibrium will not have been reached. Examples of removal processes which may upset equilibrium are leaching by ground water, and diffusion of the radon gas away from the uranium,for instance, in highly permeable sandstones. In the

majority of the uranium ores being mined in the United States, the distribution of radioactivity approximates the condition of secular equilibrium. The principal decay chain is that beginning with uranium-238 and ending with lead-206. The other decay chain, beginning with uranium-235 and ending with lead-207, contributes about 5 percent of the total radioactivity.

Most of the ores being mined are 0.1% or more uranium oxide. The ores are usually stockpiled by the mill and blended to provide a uniform mill feed of about 0.2% uranium oxide. The radioactivity concentrations are those associated with the 0.2% uranium. Twenty to eighty percent of the radon gas in the ore and small fractions of the other radionuclides are released to the environment during milling. Small fractions of some radionuclides are also shipped out in the yellowcake and vanadium pentoxide mill products. The radium-226 in the yellowcake varies from 0.001% to 0.2% of that passing through the mill for the acid-leach process, and from 1.5% to 2.2% for the alkaline-leach process. The radium in the vanadium pentoxide may be 1% of that passing through the mill. It is likely that similar fractions of the other radionuclides are also in the mill product. The problem of radium contamination and the radon it produces is, of course, present in other industries, e.g., the vanadium processing industry. In any case, almost all the radioactive decay products of the uranium in the ore find their way to the tailings, mostly in the tailings solids, but with small percentages in solution.

Ore that is 0.2% by weight of uranium oxide (U_3O_8) contains 0.00056 grams radium-226 per metric ton of ore if secular equilibrium of the radionuclides is assumed. Assuming that essentially all the radium remains with the tailings there will be 560 pCi radium-226 per gram of tailings. Reported values range from 100 to 1,000 pCi/g with values from 700 to 800 common. The higher values probably represent tailings from richer ores. Culot, et al. (10) have measured the sand fraction of tailings to determine that portion of radium which releases radon to the atmosphere. The tailings sand they used contained 125 pCi radium per gram. They determined that 23% of the radium in the tailings sand could release radon to the atmosphere. This will be assumed to hold for the slimes also, although Pearson (11) indicates that the smaller particles may release a greater fraction.

The net density of dry tailings is about 1.6 g/ml, implying a possible release rate of 430 pCi/m^3-s of tailings. However, these piles may be over 10 m thick and average over 5 m. Not all radon produced deep in the pile will escape to the surface before undergoing radioactive decay. A measure of the effect of the depth is the "relaxation length," the depth from which the fraction (1-1/e) or 63% of the radon escapes. For an average relaxation length of 1.15 m for sand (10), the release rate of radon is calculated to be about 500 pCi/m^2-s from the surface of a pile containing 560 pCi ^{226}Ra/g. Approximately 3 meters depth of tailings are required to produce close to the maximum radon

flux.

While mills are in operation, most of the tailings may be saturated with water or under water in tailings ponds. The water will tend to prevent the escape of the radon gas. There do not seem to be any measurements of radon releases from saturated tailings or tailings ponds, but theoretical calculations (3) indicate that saturating a dry tailings pile with water will reduce its radon emissions by a factor of about 25. This indicates a release rate of about 20 pCi/m^2-s for a radium concentration of 560 pCi/g. For comparison, the natural background release of radon from the ground is about 1 pCi/m^2-s in most of the country; it may be 100 times greater over uranium deposits (11).

Applying these values to 5×10^5 m^2 of tailings (about 125 acres) containing 560 pCi ^{226}Ra/g gives for a years release:

1. 7,900 Ci/yr - 5×10^5 m^2 - Dry tailings
2. 320 Ci/yr - 5×10^5 m^2 - Tailings pond
3. 16 Ci/yr - 5×10^5 m^2 - Natural background

At operating mills where the tailings pond and pile are still receiving tailings, most of the tailings solids will be quite wet; perhaps two-thirds of the area will be saturated to the surface or under water and the remaining third may have a depth of only about a meter that is relatively dry. When the milling operation is discontinued, the sands will dry quickly. However, if the slimes have been segregated and not mixed with the sands,

they may retain moisture longer.

The radon release rate at any one location is known to vary over a factor of 10 due to effects of weather, i.e., wind speed, barometric pressure, atmospheric stability, rain fall and snow cover.

The gamma radiation from tailings is low intensity and the majority of the photons are low energy. Net average exposure rates have been measured with thermoluminscent dosimeters on the face of four piles and ranged from 0.2 to 1.1 mrem/hour (12). These measurements also indicated that the external gamma radiation dose 50 meters from the tailings piles is at normal background level.

2.7.3 Radiation Dose and Health Effects to Members of the General Population in the Vicinity of a Uranium Mill Tailings Pile

A plume from a tailings pile 1,000 meters square (250 acres) is more diffuse at 1 km from the pile edge than if the same amount of curies were discharged from a point source at the center of the pile. The $(\bar{\chi}/Q)_a$ for an area source is therefore smaller by a factor of about 4 compared to the value for a point source located at the center of the pile. For a model pile 1,000 meters square, a $(\bar{\chi}/Q)_a$ of 4.5×10^{-7} s/m^3 was assumed to calculate air concentrations of radon-222 1 km from the pile edge. Calculations and pathway assumptions are otherwise the same as for the model mill; the dose conversion factor for radon-222 was assumed to be 4 mrem/yr per pCi/m^3 of radon-222

(section 2.7.5).

For a model dry pile 1,000 meters square containing 560 pCi radium-226 per gram of tailings, the maximum exposed individual living downwind 1 km from the pile edge receives a dose of 900 mrem/yr. Persons living within 80 km receive 2.2 mrem/yr giving an aggregate dose of 120 rem/yr to members of the general population. The aggregate dose predicts 6×10^{-3} mortality events/pile-yr and 0.2 mortality events/pile-30 yr.

Table 2-13 gives the radiological impact of a model uranium mill tailings pile.

2.7.4 Experimental Measurement of Radon around Tailings Piles

Under a joint agreement between the U.S. Public Health Service and the U.S. Atomic Energy Commission and in cooperation with the Colorado Department of Public Health and the Utah State Division of Health, a project was begun in 1967 to evaluate the public health aspects of radon-222 in the vicinity of certain tailings piles. Data from the study can be used to confirm radiation exposure predictions from radon emmanating from tailings piles.

Sampling stations for the radon were operated to provide an estimation of yearly average radon concentrations. These values include both the natural radon-222 background and the radon-222 from the pile. Stations not directly downwind were assumed to receive a portion of the radon from the pile.

Table 2-14 gives the results of the experimental measurements as well as predicted values for the two piles for which the

Table 2-13

Radiological impact of airborne effluents vs controls for a model uranium mill tailings pile

(250 acres) (local population)

Controls	Source term (Ci/yr)	Max. dose to the individual (mrem/yr, lung)	Total health effects per 30 yr plant operations (H.E./facility-30 yr)	Present worth (1970 dollars/facility)
None	20,000	1,300	0.25	0
Tailings pond[a] (2 ft water)	800	50	0.01	0
2 ft cover[b]	15,000	980	0.19	150,000
20 ft cover	2,000	130	0.025	1,400,000
100 ft cover[c]	0	0	0	Unknown

[a]Present level of control while operational for recent mills.

[b]Present level of control for recent mills - post operational.

[c]Tails used as back fill in strip mine; original layer of overburden is replaced. Ground-water effects must be considered in this option.

H.E. - health effect

Table 2-14

Experimental and predicted values of radon-222 emanating from tailings piles

Pile	Amount of tailings (tons)	Area (m^3)	Radium-226 concentration (pCi/g)	Predicted concentration of radon, 1 km from pile (pCi/m^3)	Measured concentration of radon 1 km from pile (pCi/m^3)	Radiation dose calculated from measured concentration (mrem/yr)
Grand Junction[a]	$2 \times 10^{+6}$	$2.2 \times 10^{+5}$	900	80	600	3,200
Salt Lake City[b]	$1 \times 10^{+6}$	$4.3 \times 10^{+5}$	1,100	200	300	1,100

[a]Natural radiation background from radon-222 - 2,400 mrem/yr.

[b]Natural radiation background from radon-222 - 600 mrem/yr.

greatest amounts of experimental data were available. For the Grand Junction pile the difference between predicted and measured values could be partly explained by the fact that the wind is blowing lengthwise down a long, narrow pile. The source term is behaving more as a point source than as a diffuse source and the concentration of radon-222 downwind would, therefore, tend to be higher. Both piles have since been stabilized by the addition of an earth cover. While such a cover would tend to eliminate the loss of radium and thorium from the piles by wind erosion and water leaching, it would not be expected to reduce the amount of radon-222 emanating by more than 25%.

2.7.5 Radon-222 Dosimetry

Radon-222 dosimetry is a complex and difficult subject. Radon-222 (and daughters) is one of the few radionuclides known to have caused significant numbers of fatalities in occupationally exposed workers (12). It is, therefore, unfortunately true that a certain limited amount of clinical evidence is available concerning the consequences to humans from exposure to radon. A biological effect of excess exposure to radon is a form of lung cancer, considered to be nearly 100% fatal. One concerned with radon exposure is aware of past failures to provide adequate protection and has reason to be conservative in any value judgements and assumptions he is forced to make concerning radon dosimetry. Some of the major elements of lung dosimetry for radon are described below as they are related to the problem of

radon emanations from tailings piles. A general review is given in reference 14. An estimate of the dose conversion factor for radon will then be made following a brief discussion of the more relevant opinions concerning this problem.

Table 2-15 lists the decay scheme and daughter products of radon-222. Examination of this table indicates that pure radon-222 of itself is not as hazardous as is radon and its daughters together. Being a noble gas, radon does not remain in the lungs and in addition, radon is not in intimate contact with the tissue. In the event of a disintegration of radon, the alpha energy is likely to be expended into a noncritical area of the lung. Radon daughters, in particular polonium-218 and polonium-214 which are not noble gas elements, have chemical and physical properties that cause them to be deposited on the mucus layer covering the bronchial epithelium of the lung leading to high dose rates to a region of the lung where tumors are most likely to arise. Radiation dose caused by radon is therefore a function of the state of equilibrium between radon and its daughters at the time of exposure, and therefore it is not enough simply to know the radon-222 concentration alone.

Given the case of radon emanating from tailings piles when a 5 m/s wind is blowing, it is likely that the dose near the pile will be quite low because emanating radon is free of daughters. By the time the wind has carried the radon 1 or 2 km (3 to 6 min.) the polonium-218 daughter has grown in and the dose

Table 2-15

Radon-222 decay scheme

Radionuclide	Half life	Emission	Energy
^{222}Rn	3 day	alpha	5.5 MeV
^{218}Po	3.05 min	alpha	6.00 MeV
^{214}Pb	26.8 min	beta	0.6 MeV[a]
^{214}Bi	19.7 min	beta	>1 MeV[a]
^{214}Po	1.6×10^{-4} s	beta	7.68 MeV
^{210}Pb	22 yr	beta	Low[a]
^{210}Bi	5 day	beta	1.2 MeV[a]
^{210}Po	138 day	alpha	5.3 MeV

[a]Maximum energy of most intense beta

delivered to individuals exposed to this air increases sharply. The same effect is achieved if the radon-loaded air is held up or delayed until the radon daughters can grow in. A house may have as few as two turnovers of air per hour and can delay the radon in its passage more than 30 minutes. For this reason, dose measurements made inside a house in the plume of a tailing pile should show higher exposure levels than similar measurements made outside the house.

When an atom of radon-222 decays in air, it becomes an atom of polonium-218. This polonium atom exists as an ion, i.e., it has an electrical charge. This charge makes it highly susceptible to absorption into an airborne aerosol particle; if inhaled, the ion is more likely to be captured by the lung than is the particle to which it might have become attached. The radiation dose delivered by radon daughters is assumed to be very much a function of the fraction of these daughters remaining as ions rather than absorbed into particles. The cleaner the air from particulate matter, the higher the dose rate will be from radon daughters because more daughters will be present as ions rather than absorbed into particles. Therefore, in western states where tailings piles are located, and where air is presumably lowest in aerosol concentrations, radiation dose rates delivered by exposure to a given concentration of radon-222 should be higher than in more industrial areas. Air conditioning may also increase doses by removing significant numbers of particles from

the air.

The critical biological target in the lung is assumed to be the nuclei of the basal cells of the bronchial epithelium. Alpha particles have a limited range in tissue. To reach the basal cells they must penetrate first a mucus layer, then parts of certain other cells. The bronchial epithelium shows extreme variations in thickness because the wall of the bronchus tends to fold upon itself. In addition, there is to be expected natural variation in thickness of both the mucus layer and the intervening cells if a population of all age groups is considered. It is, therefore, a matter of some controversy exactly to what depth the alpha particle must penetrate to deliver a critical exposure.

Given the above difficulties, which are by no means a complete review of the problem, it should be clear why radon dosimetry does not give a simple correlation between the radon-222 concentration in air and the radiation dose delivered to the lung. The several approaches considered in this analysis are described below.

ICRP Publication No. 2 states that for occupational exposures:

> "... Recent studies have indicated that when radon and its daughters are present in ordinary air the free ions of RaA constitute only about 10 percent of the total number of RaA atoms that would be present at equilibrium and these unattached atoms deliver all but a small frac-

tion of the dose to the bronchi. Based on these measured dose rates the $(MPC)_a$ for exposure to radon and daughter products is found to be $3 \times 10^6/(1 + 1000f)$ where f is the fraction of the equilibrium amount of RaA ions which are unattached to nuclei."

If f is taken to be 0.1 for "ordinary" air and assumed to range to 0.5 for "clean" air, the concentration of required to give 15 rem to the lung is 3×10^{-8} $\mu Ci/cm^3$ to 0.6×10^{-8} $\mu Ci/cm^3$, respectively, for a 40 hour week. These figures must be divided by 2.92 for continuous exposure (168 h week). The result is that 1 pCi/m^3 radon gives a dose rate of 1.5 to 7.5 mrem/yr depending on the assumptions concerning the number of free ions of polonium-218 present. It is notable with this model that in clean air where the uptake of ions by particles may be slow, that it would require only 3 minutes time for 50% of equilibrium of RaA ions to occur thereby yielding the higher dose rate to the lung.

A recent UNSCEAR report (15) has reviewed the literature on radon dosimetry and has recommended using a more complex method of estimating dose rate (table 2-16) from radon-222 concentrations. Calculations based on values presented in table 2-15 and using QF of 10 to convert from rads to rems yields for various penetration depths to cells at risk:

1. For adequate ventilation, 60 μm depth

$$1\ pCi/m^3 \text{ radon-222} = 3.6 \text{ mrem/yr}$$

Table 2-16

Calculated alpha dose rates in mrad/yr from inhalation of short-lived ^{222}Rn daughter products to the basal cell nuclei of segmental bronchi (14)

	Depth (μm)				
	15	30	45	60	70
Living accommodation: adequate ventilation[a]	550	280	100	40	1.5
Living accommodation: inadequate ventilation[b]	1,490	790	330	120	5
Industrial premises[c]	840	445	190	75	3
Air-conditioned sites[d]	280	140	50	15	0.6

a ^{222}Rn, ^{218}Po, ^{214}Pb and ^{214}Bi concentrations: 0.164, 0.148, 0.083 and 0.057 pCi/l, respectively. Annual dose (6,000 h) in millirads.

b ^{222}Rn, ^{218}Po, ^{214}Pb and ^{214}Bi concentrations: 0.37, 0.35, 0.26 and 0.21 pCi/l, respectively. Annual dose (6,000 h) in millirads.

c ^{222}Rn, ^{218}Po, ^{214}Pb and ^{214}Bi concentrations: 0.32, 0.31, 0.27, 0.25 pCi/l, respectively. Annual dose (2,000 h) in millirads.

d ^{222}Rn, ^{218}Po, ^{214}Pb and ^{214}Bi concentrations: 0.17, 0.15, 0.074, 0.060 pCi/l, respectively. Annual dose (2,000 h) in millirads.

2. For inadequate ventilation, 60 μm depth

$$1\ pCi/m^3 \text{ radon-222} = 11 \text{ mrem/yr}$$

3. For inadequate ventilation, 30 μm depth

$$1\ pCi/m^3 \text{ radon-222} = 71 \text{ mrem/yr}$$

Experimental measurements of radon daughter abundance levels in living quarters have been made by Yeats, et al. (16) and converted to dose rates through the format of:

1 working-level-month = 7 rads to the bronchial epithelium.

They yielded an overall conversion factor for radon exposure as experimentally measured in dwellings of:

$$1\ pCi/m^3 \text{ radon-222} = 35 \text{ mrem/yr}$$

The 10 CFR 20 $(MPC)_a$ occupational limit of 1×10^{-7} μCi/ml, which is assumed to be equivalent to 15 rem/yr (40 hour week), yields if calculated for a 168 hour week:

$$1\ pCi/m^3 = 0.5 \text{ mrem/yr}$$

A value of $1\ pCi/m^3$ radon-222 = 4 mrem/yr has been used to calculate dose rates resulting from exposure to radon-222 and daughters from tailings piles. This represents an acceptance of the UNSCEAR recommendations and assumes adequate ventilation and 60 μm penetration depth. It does not represent the worst conceivable case; but because of the uncertainties, it is considered an acceptable conversion factor to estimate the average dose to large numbers of members of the general population and for the setting of generally applicable environmental standards.

1 WL ⇒ 100 pCi ^{222}Rn/ℓ.

+ daughters ≅ 300 pCi/ℓ total.

$1\ pCi/m^3 = 0.001\ pCi/\ell$

∴ 0.01 WL = 4000 mrem/yr with ventilation

= 8000 " " w/o " "

2.7.6 100-year Dose Commitment from Radon-222 Emanations from Tailings Piles

The 100-year dose committment is an attempt to estimate the long term radiation dose and health effects that will result from radon emanating from the tailings piles of uranium mills. These health effects are in addition to the effects that occur from the immediate exposure of individuals within 80 km of the plant during the 30 year operating lifetime of the plant. Long term exposure occurs because the tailings from the mill contain long half-life radium-226, the parent of radon, so that radon will be continually produced and will emanate from the pile indefinitely. After release, it was assumed that the radon will distribute over the eastern United States and into the northern hemisphere causing health effects. While the dose to any individual is extremely small, the number of people exposed is large so that because of the linear, nonthreshold health effects model, the number of predicted health effects is significant.

The health effects committed as the result of each year's plant operations (effluents) exposing members of the general population to radiation for the following 100 years are calculated. The committed health effects for each year of plant operations are then summed over the lifetime of the plant to give the health effects resulting from the 100-year dose committment for 30 years of plant operations. The calculations are

similar in concept to those given in appendix A, section 7.

Results of the 100-year dose committment calculated for a uranium mill tailings pile are given in table 2-17.

2.8 Summary and Conclusion

Both theoretical predictions and experimental evidence indicate that individuals in the general population may be receiving very high levels of radiation exposure to the lung caused by the release of radon from uranium mill tailings piles.

As examples: For the Grand Junction pile, the value is 3 rem/yr to certain residential locations downwind of the pile, and for the Salt Lake City pile, the corresponding value is 1 rem/yr.

For mills of recent design where most of the tailings are expected to be under water for the operating life time of the plant, radon release rates from the wet tailings are expected to be about 4% of those from dry tailings. However, when a plant ceases operations, currently the pile is allowed to dry. A 20-foot covering of earth would then be required to reduce radon emissions by 90% compared to an uncovered dry pile.

The highest radiation dose from the model mills at current control levels is to the lung (450 mrem/yr) of individuals that might live within 1 km of the plant. Additional filtration of the air streams can reduce this value to less than 10 mrem per year.

Table 2-17

Health effects resulting from the 100-year dose commitment from radon-222 emanation from a uranium mill tailings pile (250 acres)

Pile produces fuel for 159 GW(e)-yrs.

Exposed population	Health effects committed during plant operation	Health effects committed following plant operation
Eastern United States	1	120
Northern Hemisphere	0.2	80

Health effects from 100-year dose commitment - 200 effects/facility-30 yr

~113 Ci Radon / GW(e)-yr / yr

The model mill currently discharges 4 curies of activity (mostly thorium-230) to the water pathway. This discharge can be eliminated by a catch basin and pumps.

Immediate health effects committed under current control levels are predicted to be 0.7 and 0.01 health effects/facility-30 yr for the mill and the mill tailings pile, respectively, excluding radon. As many as 200 health effects may result from the 100-year dose committment due to radon emanations from large uranium mill tailings piles.

3.0 Conversion Facilities

3.1 General Description of the Uranium Conversion Process(1,2)

Uranium concentrate milled from ore must be converted to the volatile compound uranium hexafluoride (UF_6) in order to be enriched by the gaseous diffusion process. Two different industrial processes are used for uranium hexafluoride production. The "hydrofluor process" consists of reduction, hydrofluorination and fluorination of the ore concentrates to produce crude uranium hexafluoride followed by fractional distillation to obtain a pure product. The wet solvent extraction process employs a wet chemical solvent extraction step at the head end of the process to prepare high purity uranium feed prior to reduction, hydrofluorination, and fluorination steps. Each method is used to produce roughly equal quantities of uranium hexafluoride feed for the enrichment plants. Illustrative flow sheets are given in figures 3-1 and 3-2.

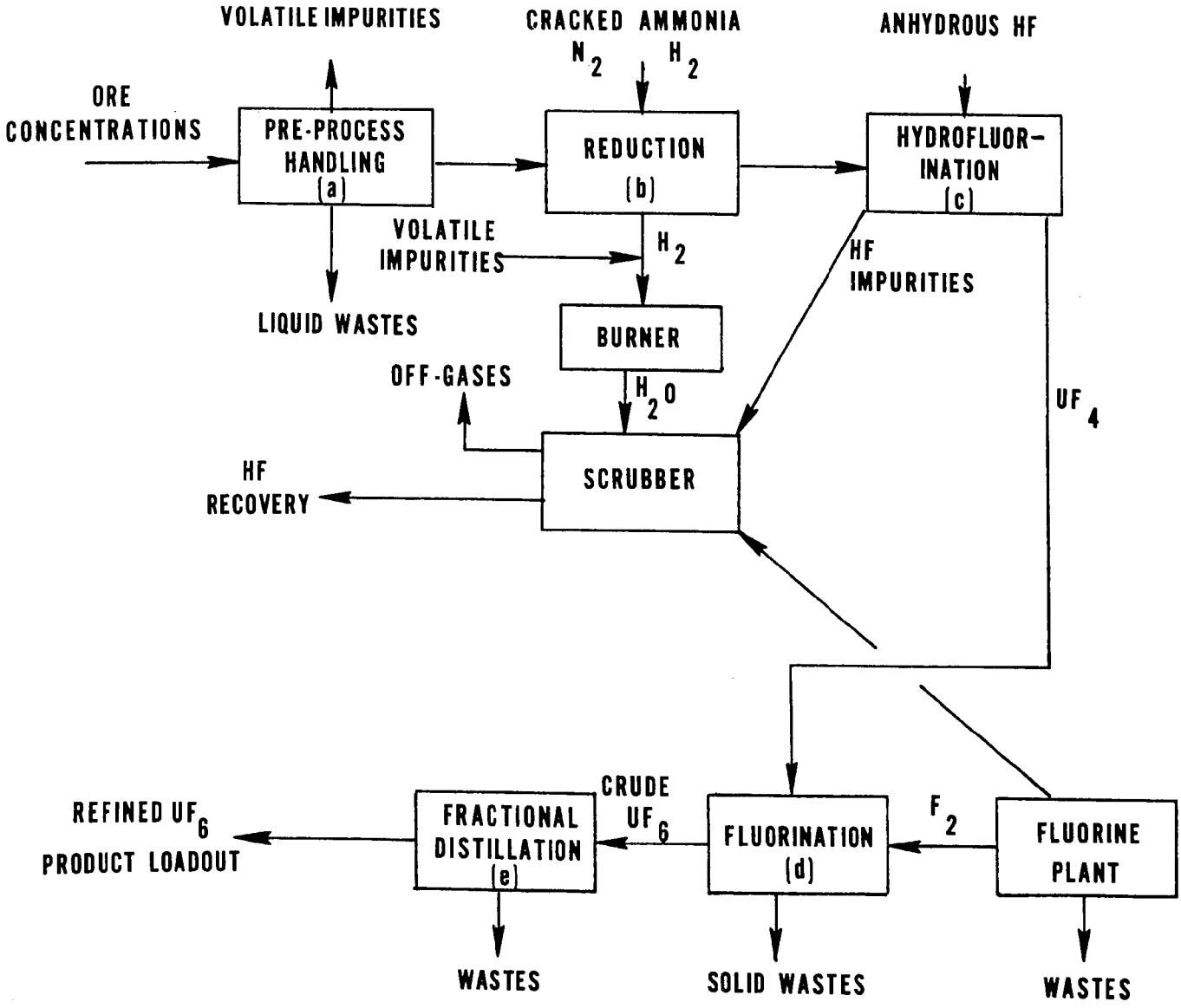

Figure 3-1. UF_6 production-hydrofluor process block diagram

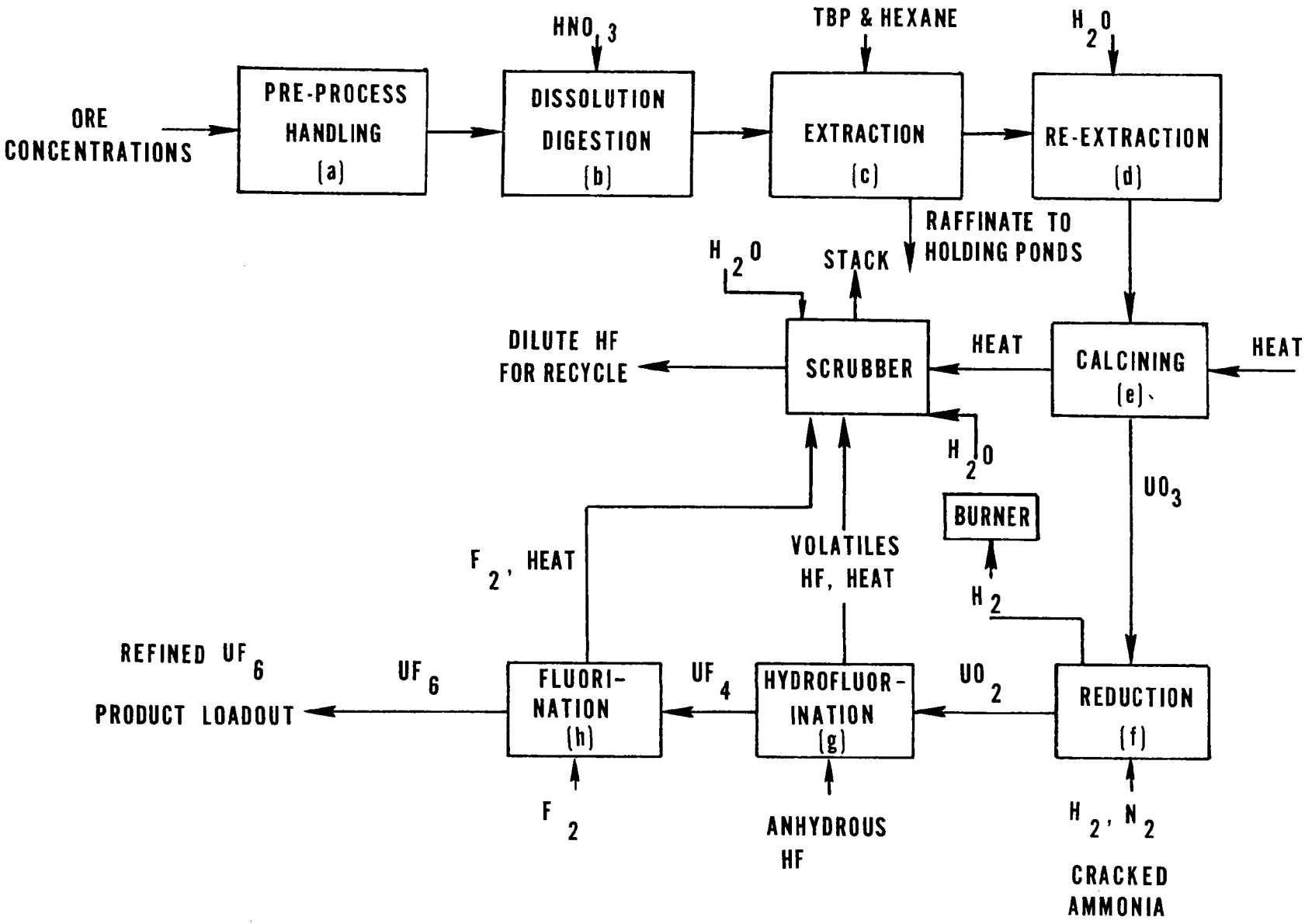

Figure 3-2. UF_6 production-wet solvent extraction-fluorination block diagram

The two commercial plants (3-5) currently in operation process approximately 10,000 metric tons of uranium into uranium hexafluoride per year; 180 metric tons of uranium converted to 270 metric tons of uranium hexafluoride are required to support a GW(e)-yr of electricity generated by light water reactors.

The uranium concentrate feed to a conversion plant contains the equivalent of about 75% to 85% uranium oxide. The conversion process removes essentially all of the remaining impurities and produces a highly purified uranium hexafluoride product. The dry hydrofluor process separates impurities either as volatile compounds or as solid constituents of ash. The wet solvent extraction method separates impurities by extracting the uranium into organic solvent leaving the impurities behind dissolved in an aqueous solution. Therefore, the nature of the radioactive effluents from the two processes differ substantially; the hydrofluor method releases radioactivity primarily in the gaseous and solid state, while the solvent extraction method releases more of its radioactive wastes dissolved in liquid effluents.

Both plant designs stipulate virtually complete recovery of uranium, total utilization of fluorine, and high utilization of the other main reactants, (hydrogen, hydrogen fluoride, ammonia, and nitric acid). The plants are located in relatively sparsely populated areas. The range of population density in the vicinity of the two existing production facilities is 10-15 people per square kilometer (25-40 people per square mile); the region sur-

rounding the plant using the dry hydrofluor process is the more densely populated.

3.2 The Model Conversion Facilities

A system of model plants has been assumed for each segment of the nuclear fuel cycle in order to achieve a common base for comparison of radiation doses, committed health effects, and also radioactive effluent control technology.

The model plant is defined in terms of a contribution to the nuclear fuel cycle that is consistent with current and projected commercial nuclear industry practice. However, because the uranium hexafluoride used by the commercial reactor industry is produced in approximately equal quantities by two different methods, each with its own characteristic amounts of radioactive effluents, two types of model conversion facilities will be assumed. One is based on the wet solvent extraction procedure, the other on the hydrofluor process.

Each type of model plant is assumed to process 5,000 MTU per year, enough to support 28 GW(e)-years of electricity generated by light water reactors. Additional resources required by the model plant are listed in table 3-1. In this table, the requirements of each plant type are appropriately averaged and considered as a single model plant of 5,000 MTU/yr capacity.

Radiation dose rates and health effects that might result from the discharge of radioactive effluents from these model facilities were calculated using standard $(\bar{\chi}/Q)$ values, dose

Table 3-1

Model uranium conversion plant[a]

Production (annual)	5,000 MT uranium as UF_6
LWR/EQA[b]	28
Land-industries	570 hectares
Permanent commitment--land	4 hectares
Water use-LWR/EQA	170 million liters
Air discharge	14 million liters
Surface water discharge	160 million liters
Electrical energy consumption (LWR/EQA)	2.1 million kilowatt hours
Natural gas for LWR/EQA	0.9 million cubic meters

[a]Assumes equal production by each of two current processes.

[b]LWR/EQA = average annual requirements for model 1 GW(e) light water reactor

conversion factors, model pathways, and health effect conversion factors that are common to all facilities considered in this examination of the fuel supply cycle. These assumptions are discussed in appendix A.

3.3 Release of Radioactive Effluents from Conversion Facilities

Because no irradiated material is handled by conversion facilities, all radionuclides present also occur in nature. They are radium, thorium, uranium, and their respective decay products. Some of the decay products are delivered to the facility as impurities in the mill concentrate and others reoccur there by the continuing radioactive decay of the uranium. Uranium is present in the majority of the plant processes, appears in liquid effluents and is essentially the only source of radioactivity in the gaseous effluents. The radium, thorium, and decay products are separated from the uranium in the conversion process and thus appear in the liquid effluents or solid waste associated with specific purifying procedures.

Uranium may appear in the gaseous effluents in several chemical forms. Possible chemical species are U_3O_8, UO_2, UF_4, UF_6, $(NH_4)_2U_2O_7$, and UO_2F_2. In the conversion process employing solvent extraction, uranium is present as uranyl nitrate which may also appear in gaseous effluents. Thus, the uranium may be released as both soluble and insoluble aerosols. Measurements at one facility (5) indicate that about two thirds of the airborne uranium is in an insoluble form, and about one third is in a

soluble form. Because uranium has a low specific activity (0.7 Ci/ MTU), the insoluble uranium is amenable to filtration. The discharge to the environment is through low stacks and vents.

Liquid effluents are associated with the various solvent extraction and scrubber systems so that the radionuclides in these effluents are considered to be in solution. About 1.7 metric tons of uranium, 0.03% of the material processed, appears in the liquid effluent streams.

Conversion of 10,000 MTU/yr to uranium hexafluoride produces an estimated 1,000 metric tons of solid waste (about 450 m^3) that must be shipped to a commercial waste disposal burial site (2). Materials in the solid wastes are filter fines, sediments, pond muds, bed materials, and miscellaneous materials. The wastes are shipped in 208 liter (55 gallon) drums. The activity disposed of per year in the solid waste is estimated as (5,8):

Natural uranium	0.5 Ci
Natural thorium	4.0 Ci
Uranium-234	0.5 Ci
Thorium-234	0.6 Ci
Protactinium-234	0.6 Ci
Radium-226	0.5 Ci
	6.7 Ci

3.3.1 Radioactive Effluents from Model Conversion Facilities

Because regulations have not required the reporting by conversion facilities of the total amounts of each radionuclide discharged per year, the source terms chosen for the two types of model facilities are based on information assembled from a variety of sources that, in many cases, cannot be adequately documented,

although much reliance has been placed on reference 2 and supporting material referred to therein. Source terms listed in tables 3-2 and 3-3 are believed to be reasonably accurate estimates of the quantities of radioactive materials discharged to the air and water pathways by operating facilities but are normalized to a production rate of 5,000 MTU per year.

3.4 Radiological Impact of Conversion Facilities

Estimates of the radiation doses received by individuals in the vicinity of the two types of conversion facilities from their routine effluents are presented in tables 3-4 and 3-5, for doses through the air pathway and the water pathway, respectively. The estimated aggregate doses to the population in the vicinity of conversion facilities are given in table 3-6. The models for the dispersion and radiation dose calculations are discussed in appendix A.

3.5 Health Effects Impact of a Model Conversion Facility

The expected cost in health effects to members of the general population in the vicinity of a model conversion facility are presented in tables 3-7 and 3-8 for facilities using the solvent-extraction process and the hydrofluor process respectively. The models used for the calculation of health effects are given in appendix A.

3.6 Control Technology

3.6.1 Airborne Effluent Control Technology

The major airborne waste control systems for conversion of

Table 3-2

Discharges of radionuclides to the environment from a model conversion facility[a] using the wet solvent extraction process

Radionuclide	Pathway	Possible chemical states	Source term (Ci/yr)
Uranium	Air	U_3O_8, UO_2	0.02 (insoluble)
		UF_6, UO_2F_2	0.008 (soluble)
Uranium	Water	UO_2^{++}	2 (soluble)
Radium-226	Water	Ra^{++}	0.006 (soluble)
Thorium-230	Water	Th^{++}	0.0006 (soluble)

[a]Each facility supports twenty-eight 1 GW(e) power plants

Table 3-3

Discharges of radionuclides to the environment from a model conversion facility[a] using the hydrofluor process

Radionuclide	Pathway	Possible chemical states	Source term (Ci/yr)
Uranium	Air	U_3O_8, UO_2	0.04 (insoluble)
		UF_6, UO_2F_2	0.02 (soluble)
Uranium	Water	UO_2^{++}	0.8 (soluble)
Radium-226	Water	Ra^{++}	[b]
Thorium-230	Water	Th^{++}	[b]

[a] Each facility supports twenty-eight 1 GW(e) power plants

[b] Information not available

Table 3-4

Radiation doses to individuals in the general population in the vicinity of a model conversion plant, through inhalation

Source term (Ci/yr)	Critical organ	Maximum dose to critical organ	
		Individual at plant boundary (mrem per yr/facility-yr)[a,b]	Individual within 80 km (mrem per yr/facility-yr)[a,b]
Wet solvent extraction process			
0.03 (uranium)	Lung	29	7×10^{-3}
	Bone	0.2	5×10^{-5}
Hydrofluor process			
0.06 (uranium)	Lung	72	2×10^{-2}
	Bone	0.5	1×10^{-4}

[a]Each facility supports twenty-eight 1 GW(e) power plants

[b]Listed mrem per yr radiation dose will result from each year of facility operation

Table 3-5

Radiation doses to individuals in the general population in the vicinity of a model conversion plant, through drinking water

Source term (Ci/yr)	Critical organ	Maximum dose to critical organ	
		Individual at plant boundary (mrem per yr/facility-yr)[a,b]	Individual within 300 km (mrem per yr/facility-yr)[a,b]
Wet solvent extraction process			
2 (uranium)	Bone	2	0.2
	Soft tissue	0.2	2×10^{-2}
6×10^{-3} (radium-226)	Bone	9×10^{-3}	9×10^{-4}
	Soft tissue	3×10^{-4}	3×10^{-5}
6×10^{-4} (thorium-230)	Bone	8×10^{-5}	8×10^{-6}
	Soft tissue	-	-
Hydrofluor process			
0.8 (uranium)	Bone	1	0.1
	Soft tissue	0.1	0.01

[a]Each facility supports twenty-eight 1 GW(e) power plants

[b]Listed mrem per yr radiation dose will result from each year of facility operations

Table 3-6

Aggregate dose to the general population in the vicinity of a model conversion facility

Source term (Ci/yr)	Pathway	Critical organ	Aggregate dose to population (dose to critical organs) (rem per yr/facility-yr)[a,b]
Wet solvent extraction process			
0.02 (uranium)	Air	Lung	10
		Bone	0.08
2 (uranium)	Water	Bone	140
		Soft tissue	14
0.006 (radium-226)	Water	Bone	0.6
		Soft tissue	0.02
0.0006 (thorium-230)	Water	Bone	0.005
Hydrofluor process			
0.06 (uranium)	Air	Lung	25
		Bone	0.2
0.8 (uranium)	Water	Bone	55
		Soft tissue	6

[a]Each facility supports twenty-eight 1 GW(e) power plants.

[b]Listed organ rem will result from each year of facility operations.

Table 3-7

Health effects to members of the general population in the vicinity of a model conversion facility using the wet solvent extraction process

Pathway	Critical organ	Health effects per facility-year[a]		
		Mortality	Nonfatal effects	Genetic effects
Air	Lung	0.0005	0	0
	Bone	a	a	a
Water	Bone	0.004	0.002	0
	Soft tissue	0.002	0.002	0.002
Total		0.006	0.004	0.002

Total health effects for 30 years of plant operations are 0.4.

[a]Listed health effects will result from each year of facility operations.

Table 3-8

Health effects to members of the general population in the vicinity of a model conversion facility using the hydrofluor process

Pathway	Critical organ	Health effects per facility-year[a]		
		Mortality	Nonfatal cancers	Genetic effects
Airborne	Lung	0.001	0	0
	Bone	b	b	b
Waterborne	Bone	0.002	0.0009	0
	Soft tissue	0.0008	0.0008	0.0008
Total		0.004	0.002	0.0008

Total health effects for 30 years of plant operations are 0.2.

[a]Listed health effects will result from each year of facility operations.

[b]Not significant compared to bone dose from water pathway.

uranium ore concentrates to uranium hexafluoride combine product recovery and waste control. Table 3-9 lists the systems used within the different product handling areas of the hydrofluor facility (5). Costs are estimated for the bag filter systems using the air flow rates as listed. However, the number of bags per unit differs and thus costs based on air flow rates alone may contain errors.

Several major problems are encountered when adequacy of present waste treatment systems is addressed. Uranium conversion facilities use waste treatment systems which are not discussed in detail in the open literature, and there is lack of discussion on the effectiveness of the waste treatment systems compared to the corresponding source terms for these facilities. For airborne waste treatment, the systems are not specifically aimed at control of radiological hazards, but rather are combined with control of chemically toxic effluents (3). The gaseous wastes contain fluorides, nitrates, and other chemicals which must be removed before being exhausted to the atmosphere.

3.6.2 Waterborne Effluent Control Technology

In conversion facilities the control technology for removal of radioactive materials from liquid effluents is combined with control technology for chemical wastes (5). Of the two existing conversion facilities, one recovers uranium from wastes by a wet chemistry recovery process and the other combines the uranium-bearing wastes with fluoride liquid wastes to be impounded in

Table 3-9

Airborne waste treatment effectiveness and costs for the hydrofluor process

Process	Air cleaning device[a]	Air flow treated (cfm)[c]	Capital cost[b]	Annual operating cost (@ 0.05/cfm)[c]
UO_3	2 bag filters in parallel (50 bags)	1,400	$ 3,000	$ 168
	1 bag filter in series with above (30 bags)	1,400	3,000	84
Fluorinator off-gas	1 bag filter (30 bags) 1 bag filter (20 bags) in series with above	2,300	6,000	138
General dust collection	2 bag filters in series (128 + 128 bags)	6,000	20,000	720
	2 bag filters in series (128 + 96 bags)	6,000	20,000	720
	2 bag filters in series	3,000	14,000	360
	2 bag filters in series (25 + 25 bags)	2,100	10,000	252
	Total		$82,000	$2,580

[a]Based on Allied Chemical Corporation system.

[b]Capital cost based on automatic cleaning bag filters (6).

[c]1.0 cfm equals 1.7 cubic meters per hour.

limestone-lined ponds. Descriptions of those ponds are available (4), but estimates of costs for the waste lagoons at the one plant could be quite inaccurate when applied at other locations because of varying costs of construction. The combining of radiological waste treatment with chemical waste treatment prevents accurate estimation of the fraction of the costs incurred for radioactive waste treatment.

Improvement in control of radioactive materials can be effected by the application of more stringent liquid waste treatment. The improvements in liquid waste treatment which may be brought about by application of the 1972 amendment to the Federal Water Pollution Control Act to control chemical wastes should reduce the impact of radioactive liquid effluents (7). The costs and effectiveness of the technology which would be applied to conversion facilities are presently unknown.

3.6.3 Solid Wastes

Solid wastes from conversion facilities are not expected to result in health effects commitments from wastes onsite. Shipments to commercial burial sites are discussed in section 6.

3.7 Environmental Controls

The effect of environmental control systems on total curies discharged, maximum radiation dose to an individual, total health effects for 30 years of plant operations, capital and annual operating costs of environmental control systems are given in tables 3-10 and 3-11 for both types of model plants. Details of the

Table 3-10

Radiological impact of airborne effluents vs controls for uranium conversion facilities

Controls	Source term (Ci/yr)	Max. dose to the individual (mrem/yr, lung)	Total health effects 30 yr plant operations (H.E./facility-30 yr)	Capital cost (1970 $/fac.)	Annual operating cost (1970 $/fac.)	Present worth (1970 $/fac.)
Wet solvent extraction process						
None	>2	>3,000	>15	0	0	0
Bag filters[a]	0.02	30	0.015	3,000	100	11,000
Additional bag filter in series[b]	0.01	3	0.001	3,000	100	11,000
Hydrofluor process						
None	>6	>7,000	>4	0	0	0
Bag filters[a]	0.06	70	0.04	82,000	2,600	190.000
Additional bag filter in series[b]	0.02	7	0.004	82,000	2,600	190,000

[a]Current level of control

[b]Add-on controls remove insoluble aerosols only

H.E. - health effect

Table 3-11

Radiological impact of waterborne effluents vs controls for uranium conversion facilities

Controls	Source term (Ci/yr)	Max. dose to the individual (mrem/yr, bone)	Total health effects 30 yr plant operations (H.E./facility-30 yr)	Capital cost (1970 $/fac.)	Annual operating cost (1970 $/fac.)	Present worth (1970 $/fac.)
Wet solvent extraction process						
None	>20	>25	>4	0	0	0
Settling ponds[a]	2	2	0.4	-	20,000[b]	240,000
Additional treatment[c]	0.2	0.2	0.04	1,000,000	1,000,000	14,000,000
Hydrofluor process						
None	>8	>10	>1	0	0	0
Settling ponds[a]	0.8	1	0.15	-	20,000[c]	240,000
Additional treatment[b]	0.08	0.1	0.01	1,000,000	1,000,000	14,000,000

[a]Current levels of control

[b]Estimated as one man-year of effort for radioactive materials control

[c]Decontamination factor of 10

H.E. - health effect

economic models are given in appendix B.

3.7.1 Environmental Controls on Airborne Waste Streams

It was assumed that removal of all controls would increase the amounts of radioactive effluent discharged by a factor of >100. The add-on control was assumed to be bag filters that would reduce the amount of insoluble aerosols discharged by a factor of 10. The numbers of health effects as a function of controls were adjusted to conform to these factors.

3.7.2 Environmental Controls on Water Waste Streams

It was not known what the effect of removal of the various waste treatment systems would be, but it was assumed that the quantities discharged would increase by a factor of >10. The add-on control was assumed to be flocculation followed by a settling pond that would reduce the amount of uranium discharged by a factor of 10. The numbers of health effects as a function of controls were adjusted to conform to these factors.

3.8 Summary

The highest radiation doses from these facilities are to the lung (70 mrem/yr; 30 mrem/yr) of individuals living within 1 km of the plants and are caused by insoluble uranium aerosols. It is believed that additional filtration of air streams can reduce this dose rate by at least a factor of 10.

The largest amount of radioactive material discharged is that of 2 curies/yr to the water pathways from a solvent extraction process plant. It is believed that this amount of discharge can

be reduced by at least a factor of 10 by addition of waste treatment systems.

An average of 0.3 health effects are to be expected from 30 years of plant operations under current levels of effluent controls.

4.0 Uranium Enrichment Facilities

4.1 Description of the Uranium Enrichment Industry(1)

Natural uranium contains about 0.7% of fissionable uranium-235. Light-water nuclear power reactors, however, utilize uranium that is enriched in uranium-235 to the range of 2-4%. Gaseous diffusion is the technology that has been developed in this country for performing the enrichment operation. Uranium is enriched by pumping the volatile uranium hexafluoride through a system of numerous porous barriers. These barriers discriminate against the passage of the heavier isotope of uranium by a theoretical maximum enrichment factor of 1.0043. Existing plants would require about 1,700 barrier stages to produce a material of 4% uranium-235. The uranium hexafluoride gas is driven through the barriers by compressors driven by electric motors. It is the compression of the gas that generates process heat which in turn requires cooling water that is ultimately discharged into the environment as thermal effluent. The electric motors require very large quantities of electricity, the generation of which causes additional effluents to be discharged into the environment from the electric power generating plants. The

gaseous diffusion plants also produce uranium hexafluoride depleted in uranium-235 (0.25%) which is stored as a solid in cylinders at the plants.

There are three government-owned gaseous diffusion plants in the United States. They were built between 1943 and 1955 and are located at Oak Ridge, Tennessee; Paducah, Kentucky; and Portsmouth, Ohio, on sites chosen for their remote location and low surrounding population densities. Distances from the gaseous diffusion plants to population centers are given in table 4-1. The plants average about 800 meters to their nearest site boundaries.

Figure 4-1 gives the mode of operation for the existing gaseous diffusion plants. The current complex of plants has a production capacity of about 10,000 metric tons of separative work units (SWU)[2] per year, enough to support 90 GW(e)-years of electricity generated by light water reactors.

It is planned to increase the capacity of the existing three-plant complex by a factor of 2.5 by 1980. This will be accomplished by improving and upgrading the present units and will be

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[2]A separative work unit (SWU) is a measure of the effort expended to separate a quantity of uranium of a given assay into two components, one having a higher percentage of uranium-235. Separative work is expressed in kg units to give it the same dimensions as material quantities.

Table 4-1

Distances to gaseous diffusion plants

from nearby population centers and UF_6 production plants (2)

Gaseous Diffusion Plant	Nearby population centers			UF_6 Production Plants (miles)	
	City	Miles	Population	Allied Chemical Metropolis, Ill.	Kerr-McGee Gore, Okla.
Oak Ridge	Oak Ridge, Tenn.	13	28,000	200	600
	Knoxville, Tenn.	30	170,000		
Paducah	Paducah, Ky	16	31,000	20	500
Portsmouth	Waverly, Ohio	12	5,000	400	900
	Portsmouth, Ohio	20	28,000		

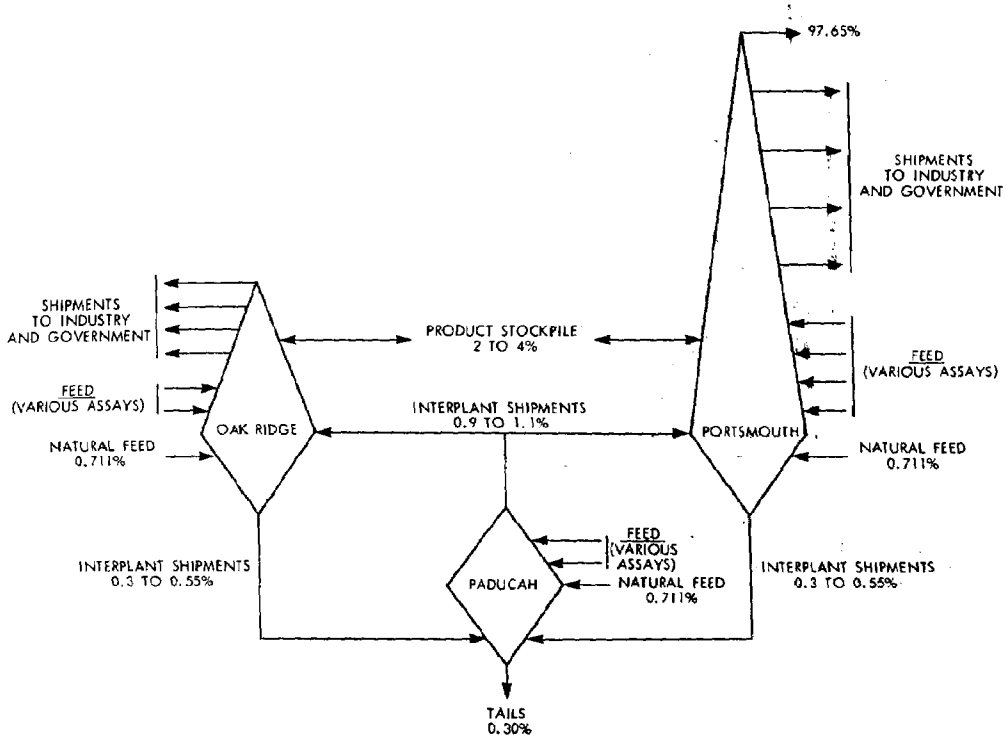


Figure 4-1

MODE OF OPERATION FOR GASEOUS DIFFUSION PLANTS

(% Values Are Weight % U-235)

enough to meet the projected 1980 industry demands. At the present time the existing production capability of the plants is only partially used for commercial production of LWR fuel. Capacity for this purpose can be increased to 10,000 MT SWU only by diverting capacity now used for other government needs.

4.2 The Model Facility

A system of model plants has been assumed for each segment of the nuclear fuel cycle to achieve a common base for comparison of radiation doses, committed health effects and also of radioactive effluent control technology. The model plant is defined in terms of a contribution to the nuclear fuel cycle that is consistent with current and projected commercial industry practice.

The characteristics of the model enrichment plant shown in figure 4-2 are assumed to be those of the current enrichment plant complex described in section 4.1 and are identical to the model plant described in reference (1). The production capacity is 10,000 MT SWU per year and will support the requirements of ninety 1 GW(e) light water reactors.

Radiation dose rates and health effects that might result from the model facility were calculated using standard procedures that are common to all facilities considered in this examination of the fuel supply. These assumptions are discussed in appendix A.

4.3 Release of Radioactive Effluents from Enrichment Facilities

Gaseous diffusion plants are large complexes, processing

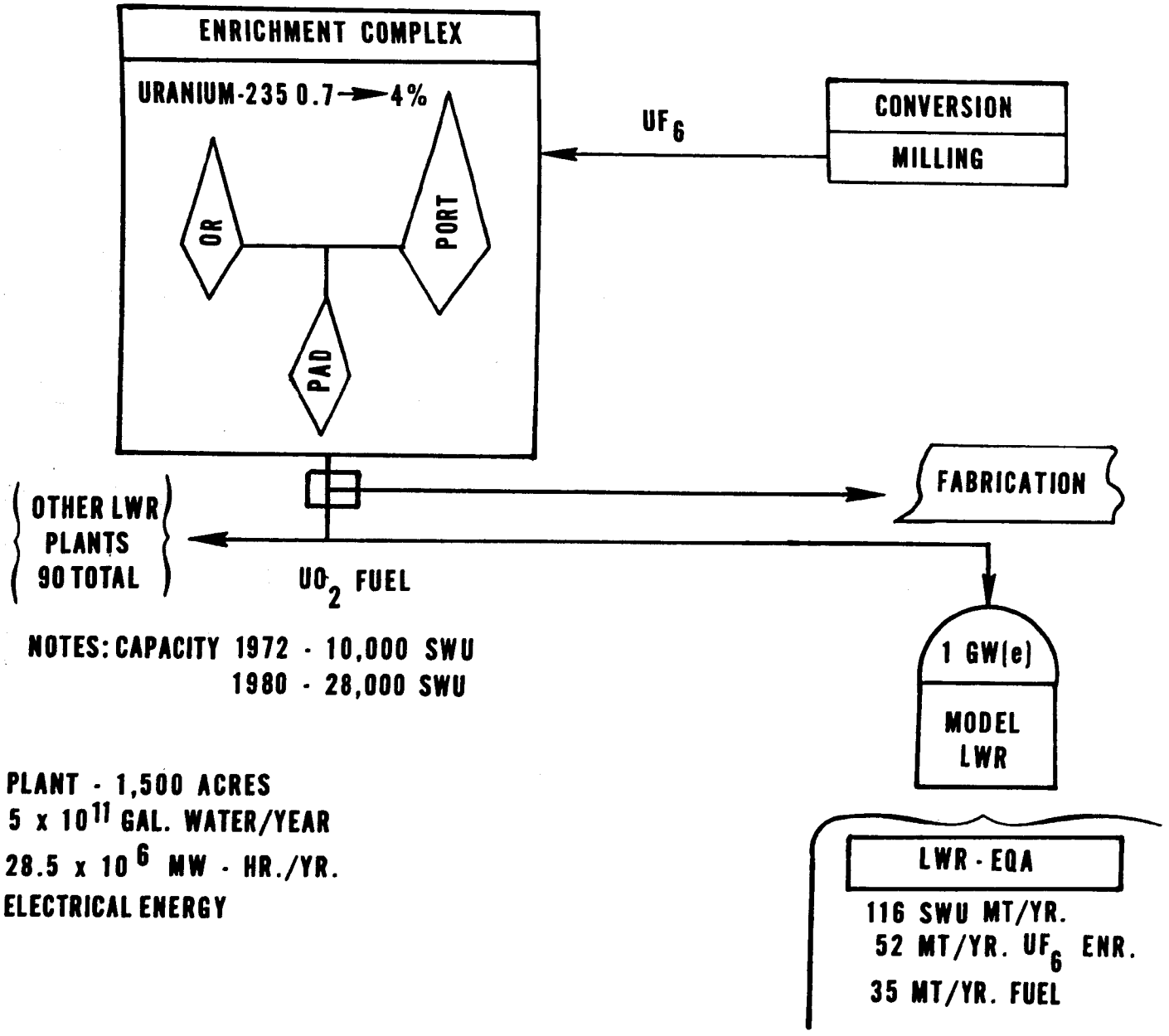


Figure 4-2. Model plant characteristics

large quantities of materials, and having many effluent streams. The effluent streams bearing radioactive materials are limited to a few types, resulting in releases of uranium to the air and river water, most probably as UO_2F_2. Reported release quantities are very small compared to permissible discharge limits and consist solely of uranium. However, the large quantities of uranium necessarily produce radioactive daughter products during storage and processing which must be handled in some manner. Those which decay to uranium-234, a comparatively worthless isotope, can eventually, via the uranium recovery facilities, be shipped out as uranium-234 in the product uranium. Other radionuclides, daughters of uranium-234 and minor contaminants of recycle fuel, must be handled through some waste disposal system. Since they are not reported as being present in effluents, it is assumed for the present that 100 percent of them go into solid waste. The effluent uranium is reported as natural uranium, which is selected as a representative isotopic mix because in producing low-enrichment LWR fuel by gaseous diffusion, the portion of plant capacity which is processing depleted uranium is comparable to the portion which is processing enriched uranium.

Effluent data for uranium enrichment plants are minimal. The Atomic Energy Commission (1) has provided source data on quantities of uranium discharged. These data are given as kilograms of uranium in gaseous and liquid effluents and are "based on releases which occurred in 1971, the annual releases attri-

butable to the support of a 1000 MW(e) LWR...," (2). Because the plants furnish enrichment services for several other purposes, e.g., naval reactor fuel, these data are uncertain; and, the task of determining what portion of the measured effluents that can be attributed to supporting the fuel requirements for LWR's is complicated. While the releases indicated represent roughly 0.007 percent of the material processed, they are remarkably small in view of the amount of processing performed.

Solid wastes consist of sludge from onsite holding ponds. The sludge is collected and buried onsite. The AEC estimates the sludge amounts to less than 1 metric ton of uranium per annual LWR fuel requirement requiring less than 0.01 acre per year (2).

No data other than that provided by the AEC in the "Environmental Survey of the Nuclear Fuel Cycle" has been made available. However, the treatment used by the AEC of converting kg quantities to curies of uranium enriched to 4 percent uranium-235 was considered unsuitable because this is about the upper limit of enrichment for LWR fuel. Because similar amounts of enrichment plant capacity are processing enriched uranium as well as depleted uranium, the uranium-235 enrichment value of natural uranium was selected as the representative enrichment for this calculation. The kilogram releases were accordingly converted to curies of natural uranium (0.7% uranium-235).

Table 4-2 gives the amounts of radioactive material assumed

Table 4-2

Discharges of radionuclides to the environment from a model enrichment facility[a]

Radionuclide	Pathway	Possible chemical states	Source term (Ci/yr)
Uranium	Air	UO_2F_2	0.05
	Water	UO_2^{++}	0.6

[a]Each facility supports ninety 1 GW(e) power plants.

to be discharged from a model enrichment facility.

4.4 Radiological Impact of Enrichment Facilities

Small quantities of uranium are released under controlled conditions to the environment from enrichment facilities. This material is transported through the environment by atmospheric and liquid exposure pathways and results in a dose to man.

4.4.1 Atmospheric Pathways

The atmospheric pathway considered most significant for this analysis was inhalation of and the subsequent dose from airborne uranium. Airborne pathway dose computations are summarized in appendix A and the results are listed in table 4-3.

4.4.2 Liquid Pathways

The most significant liquid exposure pathway was considered to be ingestion of uranium bearing drinking water. Dose computations via this pathway are also discussed in appendix A and the results are listed in table 4-4.

The aggregate dose to individuals listed in table 4-5 were computed by multiplying the respective per capita organ dose by the number of persons receiving the organ dose.

4.5 Health Effects Impact of a Model Enrichment Facility

The health effects impact of a model enrichment facility are discussed in appendix A together with the health effects impact of other fuel supply components. The projected health effects from operation of a model enrichment facility are listed in table 4-6.

Table 4-3

Radiation doses to individuals in the general population in the vicinity of model enrichment plant, through inhalation[a]

Source term (Ci/yr)	Critical organ	Maximum dose to critical organ[b]	
		Individuals at plant boundary (mrem/yr per facility-yr)	Individuals within 80 km (mrem/yr per facility-yr)
0.05 (uranium)	Bone	1	3×10^{-4}

[a]Each facility supports ninety 1 GW(e) power plants.

[b]Listed mrem per yr radiation dose will result from each year of facility operations.

Table 4-4

Radiation doses to individuals in the general population

in the vicinity of a model enrichment plant, through drinking water

Source term (Ci/yr)	Critical organ	Maximum dose to critical organs	
		Individuals at plant boundary (mrem/yr per facility-yr)	Individuals within 300 km (mrem/yr per facility-yr)
0.6 (uranium)	Bone	0.7	0.07
	Soft tissue	0.07	0.007

Table 4-5

Aggregate dose to the general population in the vicinity of a model enrichment facility

Source term (Ci/yr)	Pathway	Critical organ	Aggregate dose to individuals (rem/yr per facility-yr)[a,b]
0.05 (uranium)	Airborne	Bone	0.4
0.6 (uranium)	Waterborne	Bone	40
		Soft tissue	4

[a] Each facility supports ninety 1 GW(e) power plants

[b] Listed aggregate dose will result from each year of facility operations

Table 4-6

Health effects to members of the general population in the vicinity of a model enrichment facility

Pathway	Critical organ	Health effects/facility-yr[a]		
		Mortality	Nonfatal effects	Genetic effects
Air	Bone (bone cancer & leukemia)	1×10^{-5}	7×10^{-6}	0
Water	Bone (bone cancer)	7×10^{-4}	7×10^{-4}	0
	Bone (leukemia)	4×10^{-4}	0	0
	Soft tissue	6×10^{-4}	6×10^{-4}	6×10^{-4}
	Total, water------	2×10^{-3}	13×10^{-4}	6×10^{-4}

Total health effects for 30 years of plant operations--0.1

[a]Listed health effects will result from each year of facility operations.

4.6 Control Technology

4.6.1 Airborne

Present information indicates that the treatment of gaseous wastes in uranium enrichment facilities is part of the product recovery system (3). The costs and efficiencies of this system for uranium removal are unknown (3).

Because uranium enrichment technology is classified for national security reasons, the possible changes in waste management systems for control of uranium isotopes released to the environment are unknown.

4.6.2 Liquid Effluents

The liquid effluents from uranium enrichment facilities pass through process cleanup for recovery of uranium and chemicals such as fluorides and nitrates. The treated water is released to settling ponds which discharge into a nearby stream (1). The costs and effectiveness of existing waste control technology are not presently available.

Possible improvements in liquid effluent control may result from application of the Federal Water Pollution Control Act Amendments of 1972 (5). The costs and effectiveness of the possible improvement are unknown.

4.6.3 Solid Wastes

Miscellaneous solid radioactive wastes are incinerated and the ashes are processed through the uranium recovery plant (1). The tails from the enrichment process are collected and stored

(1). There are no anticipated health effects from solid wastes produced by the enrichment plant.

4.7 Environmental Controls - Enrichment Facilities

Specific information on liquid and gaseous waste control systems is not available. Therefore, a detailed evaluation of the reduction in the radiological impact on the environment from the use of additional controls cannot be made. A first order approximation of the value, in terms of environmental effects, of the present control technology versus no controls is shown in tables 4-7 and 4-8. In these tables, it is assumed that radioactivity releases with no waste control systems in effect would be approximately two and one order of magnitude higher for the air and water waste control system, respectively.

4.8 Summary

The highest radiation dose from the model enrichment facility is expected to be less than 2 mrem per year (bone) to the maximum exposed individual delivered in about equal amounts through inhalation and drinking water.

Approximately 0.1 health effects are to be expected from 30 years of plant operations under current levels of effluent controls.

Less than 1 curie per year of uranium is discharged.

5.0 Fuel Fabrication and Scrap Recovery

5.1 Description of the Fuel Fabrication Process

Fuel for the light water power reactor is fabricated from

Table 4-7

Radiological impact of airborne effluents vs controls for uranium enrichment facilities

Controls	Source term (Ci/yr)	Max. dose to the individual (mrem/yr)	Total health effects 30 yr plant operation (H.E./facility-30 yr)	Capital cost (1970 $/fac.)	Annual operating cost (1970 $/fac.)	Present worth (1970 $/fac.)
None	>5	>0.01 (Bone)	$>3 \times 10^{-2}$	0	0	0
Cold traps[a] and aluminium traps	0.05	1 (Bone)	3×10^{-4}	Unknown	Unknown	Unknown

[a]Current levels of controls.
H.E. - Health effects

Table 4-8

Radiological impact of waterborne effluents vs controls for uranium enrichment facilities

Controls	Source term (Ci/yr)	Max. dose to the individual (mrem/yr)	Total health effects 30 yr plant operation (H.E./facility-30 yr)	Capital cost (1970 $/fac.)	Annual operating cost (1970 $/fac.)	Present worth (1970 $/fac.)
None	> 6	>8	>1	0	0	0
Chemical[a] treatment and settling ponds	0.6	0.7	0.13	-	20,000[b]	240,000

[a]Current levels of controls

[b]Estimated as one man year effort for radioactive materials control
H.E. - Health effects

uranium hexafluoride enriched to 2-4% in the uranium-235 isotope. The slightly enriched uranium hexafluoride is shipped from the uranium enrichment facility to a fuel fabrication facility (in sealed 2,300 kg cylinders) where it is hydrolyzed to uranyl fluoride, converted to ammonium diuranate, and calcined to the dioxide. The dioxide powder is pelletized, sintered, and loaded into stainless steel or Zircaloy tubing which is then capped and welded. A process flowsheet is presented in figure 5-1 (1). The fuel rods, each about 3.7 meters (12 feet) long and slightly less than 13 mm (1/2 inch) in diameter, are assembled in arrays to be handled as fuel assemblies.

A list of fuel fabrication plants, their products, and site data are given in tables 5-1 and 5-2 (1). Scrap material from fuel fabrication is dissolved in nitric acid, purified by solvent extraction, calcined and reduced to the dioxide which may be cycled back to the fabrication process.

The most significant chemical effluents are fluorine, fluorine compounds and nitrogen compounds. The bulk of the fluorine released from the UF_6 appears ultimately as solid CaF_2 resulting from lime neutralization.

5.2 The Model Facility

The model fuel fabrication plant is assumed to be identical to the one described in reference 1. The model site for the plant that is assumed for the purposes of dose calculations has the general characteristics discussed in appendix A of this report.

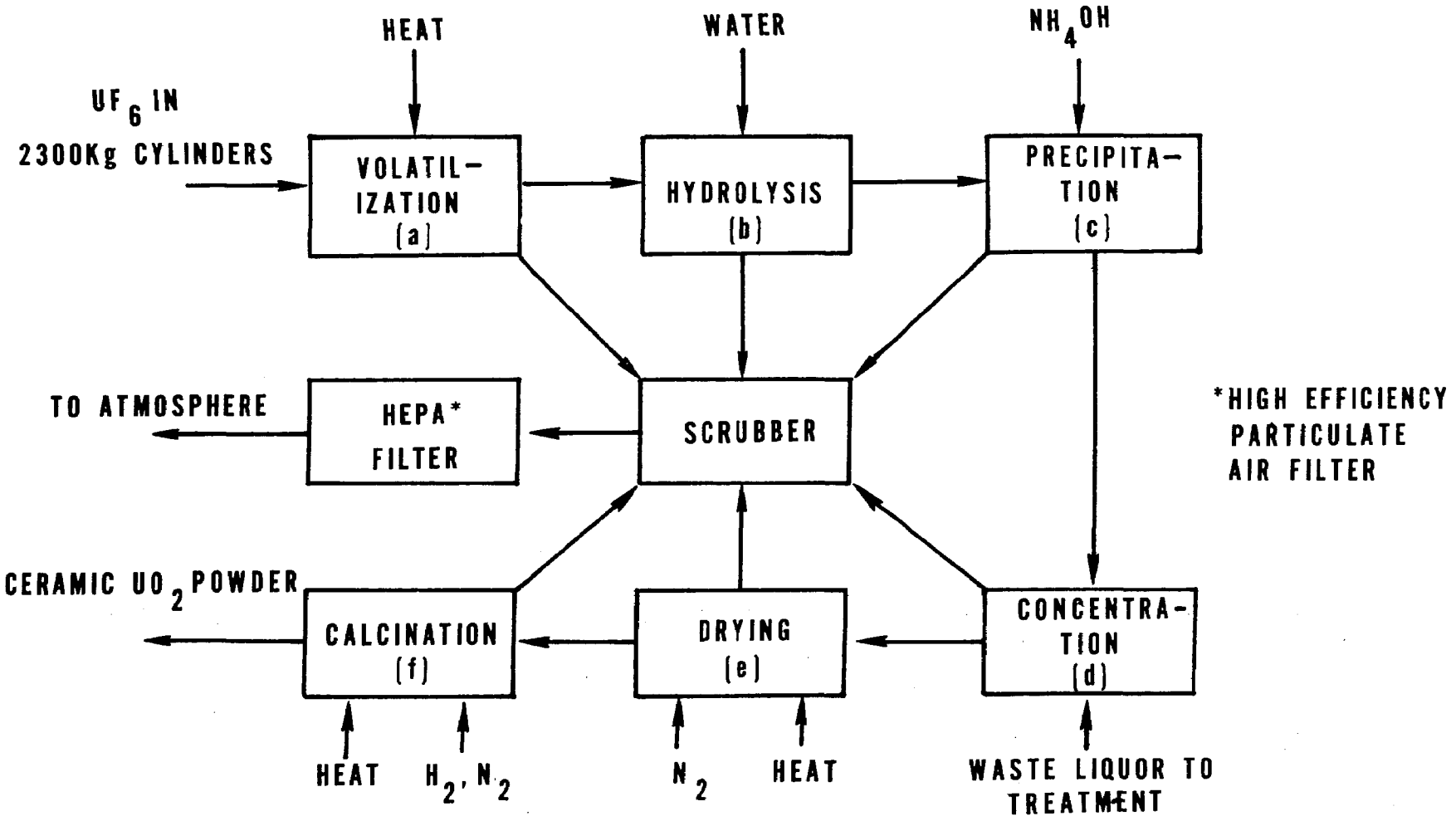


Figure 5-1 Fuel fabrication-chemical processing (ADU) block diagram

Table 5-1

LWR fuel fabrication plants (3)

Licensee	Plant location	Plant feed material	Plant product
Babcock & Wilcox	Lynchburg, Va.	UO_2 pellets	Fuel assemblies
Combustion Engineering	Windsor, Conn.	UO_2 powder	Fuel assemblies
General Electric	Wilmington, N.C.	UF_6	Fuel assemblies
Gulf United Nuclear	Hematite, Mo.	UF_6	UO_2 powder or pellets
Gulf United Nuclear	New Haven, Conn.	UO_2 pellets	Fuel assemblies
EXXON	Richland, Wash.	UF_6	Fuel assemblies
Kerr-McGee[a]	Crescent, Okla.	UF_6	UO_2 powder or pellets
Nuclear Fuel Services[a]	Erwin, Tenn.	UF_6	UO_2 powder or pellets
NUMEC	Apollo, Pa.	UF_6	UO_2 powder or pellets
Westinghouse	Columbia, S.C.	UF_6	Fuel assemblies

[a]Kerr-McGee and Nuclear Fuel Services data are from USAEC Regulatory files.

Table 5-2

Fuel fabrication plants - site size and demography (4)

Plant location	Site size (hectares)	Population density (people/km^2)	Nearby population centers		
			City	Population	Distance (km)
Babcock & Wilcox Lynchburg, Va.	205	16	Lynchburg	54,000	6
Combustion Eng. Windsor, Conn.	215	240	East Granby Windsor	35,000 22,500	5 8
General Electric Wilmington, N.C.	668	20	Castle Hayne Wilmington	700 46,000	3 13
Gulf United Nuclear Hematite, Mo.	61	120	Hematite St. Louis	< 2,500 622,000	1 53
Gulf United Nuclear New Haven, Conn.	31 a	240	Hartford New Haven	158,000 138,000	15 0
EXXON Richland, Wash.	65	8	Richland	26,000	5
Kerr-McGee Crescent, Okla.	405	40	Crescent Oklahoma City	1,600 363,000	8 48
Nuclear Fuel Services Erwin, Tenn.	24	40	Erwin Johnson City	4,700 33,800	2 21
NUMEC Apollo, Pa.	2	160	Apollo Pittsburgh	< 2,500 520,000	0 40
Westinghouse Columbia, S.C.	461	50	Columbia	113,500	13

[a]Shared by manufacturing and research divisions of Olin Corporation and naval reactor fuel operations of United Nuclear Corp.

In addition, the plant has particular characteristics related to its function in the fuel cycle. These are indicated in figure 5-2.

Other requirements for the model plant include:

1. 40.5 hectare (100 acre) site with buffer zone of at least 500 meters to the nearest site boundary.

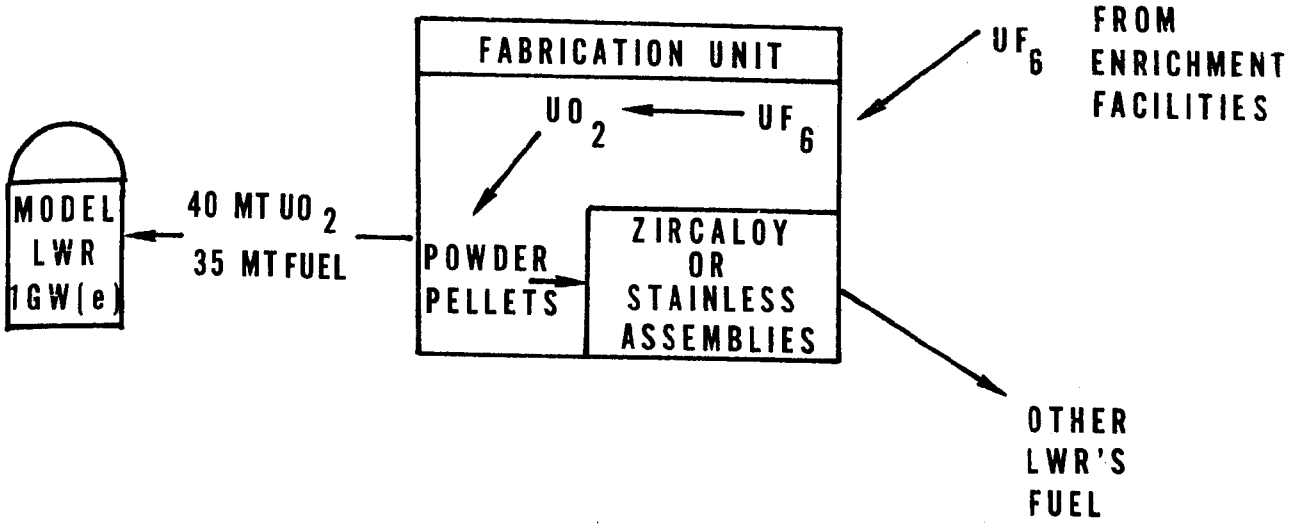
2. 1.7 million liters (450,000 gallons) of water per day or 20 million liters (5.2 million gallons) per LWR annual fuel requirement.

3. 100,000 cubic meters (3.6×10^6 scf) natural gas for process heat in fabrication of annual LWR fuel requirements.

For the purposes of this analysis, a model fuel fabrication plant has a capacity of 3 MTU per day and operates 300 days per year. Assuming a lifetime for the plant of 30 years, 780 fuel requirements for the model LWR are available.

5.3 Radionuclide Effluents from Fuel Fabrication Facilities

The majority of the present fuel fabrication plants perform all the post enrichment operations necessary to produce finished fuel assemblies, including converting UF_6 to UO_2, making the UO_2 pellets, putting the pellets into cladding tubes, and putting the tubes together to form fuel assemblies. Due to leakage, spillage and breakage, some of the enriched uranium is released to the waste streams of the plant, and small quantities escape, most probably as UO_2F_2 or UO_2.



AREA - 40.5 HECTARES
WATER - 1.7 MILLION LITERS/DAY
POWER - 6 MW(e)
-- 1,700 MW-HR. (ANNUAL)
OR 620 MT COAL
OUTPUT - 3 MTU/DAY FOR 300 DAYS/YR
LIFE - 30 YEARS

Figure 5-2 Model fuel fabrication plant

Estimated radioactive effluent quantities are given in table 5-3. The airborne radionuclides are conservatively assumed to be in the form of insoluble aerosols; the radionuclides in liquid effluents are probably in solution. Upon consideration of the available data and the intent to base the estimates upon good current practice, the values used by the AEC in their "Environmental Survey of the Nuclear Fuel Cycle" (3) were used for these estimates. The decay of the uranium isotopes -234, -235, and -238 in natural uranium provides 0.7 Ci per MTU compared to uranium enriched to 3.2 percent uranium-235 which provides about 1.8 Ci per MTU.

The economic incentive to minimize losses of uranium in effluents is only weakly related to the radioactivity of the effluents. Releases at a constant rate in kilograms per year become more hazardous as the radioactivity per kilogram increases. This may be expected to occur. Most of the radioactivity in the uranium released is due to uranium-234 which is relatively worthless. Uranium coming from enrichment facilities is expected to contain higher percentages of uranium-234 in the future as more recycled uranium fuel is used. The recycled fuel will also bring radioactive impurities with it, although at extremely small concentrations. Because some of this material will find its way into effluents, careful watch over the amount and nature of radioactivity in effluents is necessary.

Uranium scrap recovery operations are performed both in the

Table 5-3

Discharge of radionuclides to the environment from a model uranium oxide fuel fabrication facility[a]

Radionuclide	Pathway	Probable chemical state	Source term (Ci/yr)
Uranium[b]	Air	UO_2F_2;UO_2	0.005
Uranium[b]	Water	UO_2^{++}	0.5
Thorium-234	Water	Th^{++}	1

[a]Each facility supports twenty-six 1 GW(e) power plants

[b]Enriched to 3.2 wt % uranium-235.

fuel fabrication plant and in separate facilities. The scrap recovery operations tend to release a larger fraction of the material processed to effluent streams than do the fuel fabrication lines. This happens in part because the scrap recovery operations are relatively small and it is difficult to find economic justification for the expense of high efficiency waste treatment systems for them. Available data from one separate facility indicate that quantities on the order of one percent of the material processed are released in effluents for that operation. Its releases of radioactivity to air and water are comparable to those of fuel fabrication facilities. However, when the scrap recovery is contained within the fabrication plant, scrap recovery effluents become a contribution to the total plant effluents and indistinguishable from them in available data. Effluents from uranium scrap recovery operations are expected to be a minor fraction of those from all LWR fuel fabrication operations.

Some reported and estimated radioactive effluents for specific fuel fabrication plants are given in table 5-4. The data serve to indicate order-of-magnitude values, but many reports do not relate the effluent data to the quantities processed.

Estimates of solid wastes from fuel fabrication facilities involve both onsite burial and shipment offsite to commercial burial sites. Onsite burial of calcium fluoride removed from liquid wastes streams is estimated as 680 MT per year for a 900

Table 5-4

Reported and estimated effluent uranium quantities

Plant	Gaseous	Liquid
Exxon (2)	0.00015 Ci/yr	0.08 Ci/yr
Kerr-McGee (3)	0.04 Ci/yr	0.24 Ci (7.3 mo.)
NUMEC (4)		1.75 Ci (6 mo.)
General Electric (5)[b]	0.0028 Ci/yr[a]	1.36 Ci/yr[a]
	0.002 Ci/yr	
Westinghouse[c]	0.2 Ci/yr[a]	0.3 Ci/yr[a]
GUNF (6)	0.03 Ci/yr	0.1 Ci/yr

[a]Estimated on 900 MT/year basis.

[b]General Electric Company, private communication with AEC (December 1, 1972)

[c]Westinghouse Electric Company, private communication with AEC (December 1, 1972)

MTU/yr model plant (1). The volume of waste buried per year is about 220 cubic meters (290 cubic yards or 7,800 cubic feet). The quantity buried per annual LWR fuel requirement is estimated as 8.4 cubic meters (296 cubic feet) containing 0.06 Ci of uranium.

The volume of solidified waste shipped to commercial waste burial firms is not estimated by the AEC. The radioactivity is estimated to be 0.0025 Ci per annual LWR fuel requirement or 0.07 Ci per year for the 900 MTU model fuel fabrication plant.

5.4 Radiological Impact of a Model Fuel Fabrication Facility

Estimates of the radiation doses received by individuals in the vicinity of a model fuel fabrication facility from their routine effluents are presented in tables 5-5 and 5-6, for doses through the air pathway and water pathway, respectively. The estimated aggregate doses to the population in the vicinity of conversion facilities are given in table 5-7. The models for the dispersion and dose calculations are discussed in appendix A.

5.5 Health Effects Impact of a Model Fuel Fabrication Facility

The expected cost in health effects to members of the general population in the vicinity of a model fuel fabrication facility are presented in table 5-8. The models used for the calculation of health effects are given in appendix A.

5.6 Control Technology

5.6.1 Airborne

Table 5-5

Radiation dose to members of the general population from a model uranium oxide fuel fabrication facility through inhalation

Source term (Ci/yr)	Critical organ	Maximum dose to critical organ	
		Individual at plant boundary (mrem/yr per facility-yr)[a,b]	Individual within 80 km (mrem/yr per facility-yr)
0.005 (uranium)	Lung	10	0.002

[a]Each facility supports twenty-six 1 GW(e) power plants.

[b]Listed mrem/yr radiation dose will result from each year of facility operations.

Table 5-6

Radiation doses to individual in the general population in the vicinity of a model uranium fuel fabrication plant through drinking water

Source term (Ci/yr)	Critical organ	Maximum dose to critical organs	
		Individuals at plant boundary (mrem per yr/facility-yr)[a,b]	Average individual dose within 300 km (mrem per yr/facility-yr)
0.5 (uranium)	Bone	0.6	0.06
	Soft tissue	0.06	0.006

[a]Each facility supports twenty-six 1 GW(e) power plants.

[b]Listed mrem/y radiation dose will result from each year of facility operations.

Table 5-7

Aggregate dose to the general population in the vicinity of a model uranium fuel fabrication plant

Source term (Ci/yr)	Pathway	Critical organ	Aggregate dose to population (rem per yr/facility-yr)[a,b]
0.005 (uranium)	Air	Lung	3
0.5 (uranium)	Water	Bone	34
	Water	Soft tissue	3

[a]Each facility supports twenty-six 1 GW(e) power plants

[b]Listed aggregate dose will result from each year of facility operations

Table 5-8

Health effects to members of the general population in the vicinity of a model fuel fabrication plant

Pathway	Critical organ	Predicted health effects/facility-yr[a,b]		
		Mortality	Nonfatal effects	Genetic effects
Air	Lung	0.0002	0	0
Water	Bone (bone cancer)	0.0006	0.0006	0
	Bone (leukemia)	0.0004	0	0
Water	Soft tissue	0.0005	0.0005	0.0005
	Total	0.0016	0.0011	0.0005

[a]Listed health effects will result from each year of facility operations.
[b]Total health effects for 30 years of plant operations are 0.1.

The model uranium fabrication plant has three major gaseous waste treatment systems. The system for conversion of UF_6 to UO_2 is equipped with a scrubber-demister and one high-efficiency particulate air (HEPA) filter (1). The processes handling UO_2 powder, pellets, and fuel tube loading are exhausted through the HEPA filters (1). Scrap recovery chemical systems exhaust through a scrubber-demister and through one HEPA filter. The solid wastes incinerator also exhausts through one HEPA filter (1). For the purposes of this discussion, the conversion (UF_6 to UO_2) and scrap recovery are assumed to use a common scrubber-demister. Each system is equipped with a HEPA filter. The process systems handling UO_2 are assumed to use two HEPA filters in tandem independent of those on the conversion and scrap recovery systems. The major fraction of airborne particulates is assumed to be from the process systems.

Table 5-9 lists the associated cost parameters for the gaseous waste systems. The AEC-model 900 MTU capacity fuel fabrication plant was used for describing the air cleaning system. Costs of the systems are reported as a range of costs. For the purposes of a later summary table, the midpoint of the range was used. The capital and operating costs listed in table 5-9 were reported as annualized costs in the reference quoted. Selection of the midpoint of the range of costs and inflation may have lead to somewhat of an underestimate of costs for application to specific facilities in 1973. However, the ratio of operating

Table 5-9

Uranium fuel fabrication and scrap recovery gaseous waste treatment effectiveness and costs

Type of equipment	Fraction of uranium removal	Estimated[b] cfm treated	Approximate[c] capital cost ($/cfm)	Annual operating cost ($/cfm)	Capital cost	Annual operating cost	Capital cost	Annual operating cost
Scrubber and demister	0.9(7)	10^5	1.50 - 3.00(2)	0.37 - 0.75(2)	150,000 - 300,000 (1964 $)	37,000 - 75,000 (1964 $)	190,000 - 380,000 (1970 $)[d]	46,000 - 94,000 (1970 $)[d]
Prefilter	-	10^5	0.02 - 0.06	0.015 - 0.025(2)	2,000 - 6,000 (1960 $)	15,000 - 25,000 (1960 $)	2,500 - 7,500 (1970 $)[d]	19,000 - 57,000 (1970 $)[d]
1st HEPA	0.999[a]	10^5	0.42 - 0.60(2)	0.10 - 0.45(2)	42,000 - 60,000 (1960 $)	10,000 - 45,000 (1960 $)	52,000 - 75,000 (1970 $)[d]	12,500 - 56,000 (1970 $)[d]
2nd HEPA	0.99[a]	10^5						
3rd HEPA	0.94(8)	10^5						

[a] First and second banks together reported to remove 0.99999(3) fraction removal apportioned between the filters as indicated.

[b] 1.0 cfm = 1.7 cubic meters per hour.

[c] Costs converted to capital cost from annualized costs using a annual fixed charge rate of 16.6%.

[d] Costs converted to 1970 $ using the Marshall and Stevens Equipment Cost Index.(9)

costs to total annualized costs is 75% which is in agreement with an estimated range of 70-85% in reference 12.

Several major problems are encountered when the question of the adequacy of present waste treatment systems is addressed. Uranium fuel fabrication facilities use waste treatment systems which are not described in detail in the open literature. An additional problem is the minimal amount of discussion of the effectiveness of the waste treatment systems and the corresponding source terms for these facilities. The gaseous wastes also contain fluorides, nitrates, and other chemicals which must be removed before exhausting to the atmosphere.

Added HEPA filters are a possible way of adding waste treatment. A current practice being introduced in uranium fuel fabrication plants is the use of glove boxes. The primary purpose is product containment within the plant, but the quantities released to the environment may also be reduced. Costs of glove boxes or hoods are not included in the estimates of costs of control.

5.6.2 Liquid Effluents

The liquid wastes from scrubber-demisters, drains, and laboratories are collected in settling tanks or ponds. Treatment consists of adding flocculating agents and chemicals for removal of fluorides and nitrates which are discharged from the systems for conversion of UF_6 to UO_2 and scrap recovery (1).

The review of available data on uranium fuel fabrication plants conducted by Battelle was used for estimating the costs of liquid waste treatment (10). For 1970, the total industry had a production of 950 MTU. Table 5-10 lists the data for costs of control. Annualized costs were calculated using the factors and assumptions in appendix B.

Possible improvements in control of radioactive materials could result from the application of more stringent liquid waste treatment. The improvements in liquid waste treatment which may be brought about by application of the 1972 amendment to the Federal Water Pollution Control Act (11) to chemical wastes could reduce the impact of radioactive materials by also reducing their discharge quantities. The costs and effectiveness of the technology which would be applied to fuel fabrication facilties is not known.

5.6.3 Solid Wastes

Solid wastes from fuel fabrication facilities are not expected to result in health effects commitments from wastes onsite. Shipments to commercial burial sites are discussed in another section.

5.7 Environmental Controls - Fuel Fabrication

The waste systems, estimated costs and estimated health effects for airborne releases from fuel fabrication facilties are summarized in table 5-11. The changes in costs and health effects for the addition of another HEPA filter in series are also listed. Table 5-12 lists the present control technology, costs and health

Table 5-10

Uranium fuel fabrication and scrap recovery
liquid waste treatment effectiveness and cost (10)

Type of equipment	Fraction of uranium removal	Capital cost[a]		Annual operating cost[a]	Annualized capital cost[b]	Total annualized cost
		Equipment	Facility			
Settling tanks	0.90(6)	$280,000(6)	$56,000(6)	$28,000(6)	$56,000	$84,000

[a]For 950 MTU throughout 1970.

[b]Annual fixed charge rate on depreciable capital - 16.6%.

Table 5-11

Radiological impact of airborne effluents vs controls for a model uranium fuel fabrication plant

Controls	Source term (Ci/yr)	Max. dose to the individual (mrem/yr, lung)	Total health effects 30 yr plant operations (H.E./facility-30 yr)	Capital cost (1970 $/fac.)	Annual operating cost (1970 $/fac.)	Present worth (1970 $/fac.)
Scrubber and prefilter base	> 500	> 1,000,000	> 500	0	0	0
Plus 1 HEPA	0.5	1,000	0.5	64,000	34,000	530,000
2nd HEPA in series[a]	0.005	10	0.005	64,000	34,000	530,000
3rd HEPA in series	0.0003	0.5	0.0002	64,000	34,000	530,000

[a]Current level of control

H.E. - health effect

Table 5-12

Radiological impact of waterborne effluents vs controls for a model uranium fuel fabrication plant

Controls	Source term (Ci/yr)	Max. dose to the individual (mrem/yr, bone)	Total health effects 30 yr plant operations (H.E./facility-30 yr)	Capital cost (1970 $/fac.)	Annual operating cost (1970 $/fac.)	Present worth (1970 $/fac.)
None	> 5	> 6	> 0.9	0	0	0
Settling tanks[a]	0.5	0.6	0.09	340,000	28,000	1,000,000
Precipitation and flocculation	0.02	0.01	0.002	1,000,000	200,000	4,300,000

[a]Current level of control

H.E. - health effect

effects estimates for radioactive materials released to the water pathway. Costs for liquid waste treatment were calculated using the factors and assumptions in appendix B. For additional liquid waste treatment, it was assumed a two-stage precipitation-flocculation system reported in reference 13 could attain an additional decontamination factor of 50.

5.8 Summary

The highest radiation dose from the model fabrication facility is 10 mrem per year (lung) to the maximum exposed individual living within 1 km of the plant.

Approximately 0.1 health effects are to be expected from 30 years of plant operations under current levels of effluent controls.

Less than 1 curie per year of uranium is discharged.

6.0 Transportation

6.1 Description and Growth Patterns

An overview of the transportation requirements for the LWR nuclear power industry is depicted in figure 6-1. Radioactive materials are currently shipped by rail and truck. However, barge transportation is projected in the near future. A summary of the parameters for each transportation pathway is presented in table 6-1. Predictions of growth patterns are summarized in table 6-2 for a 900 unit LWR program at 1,000 MW(e) each which can reasonably be expected to occur by the year 2000. Plutonium

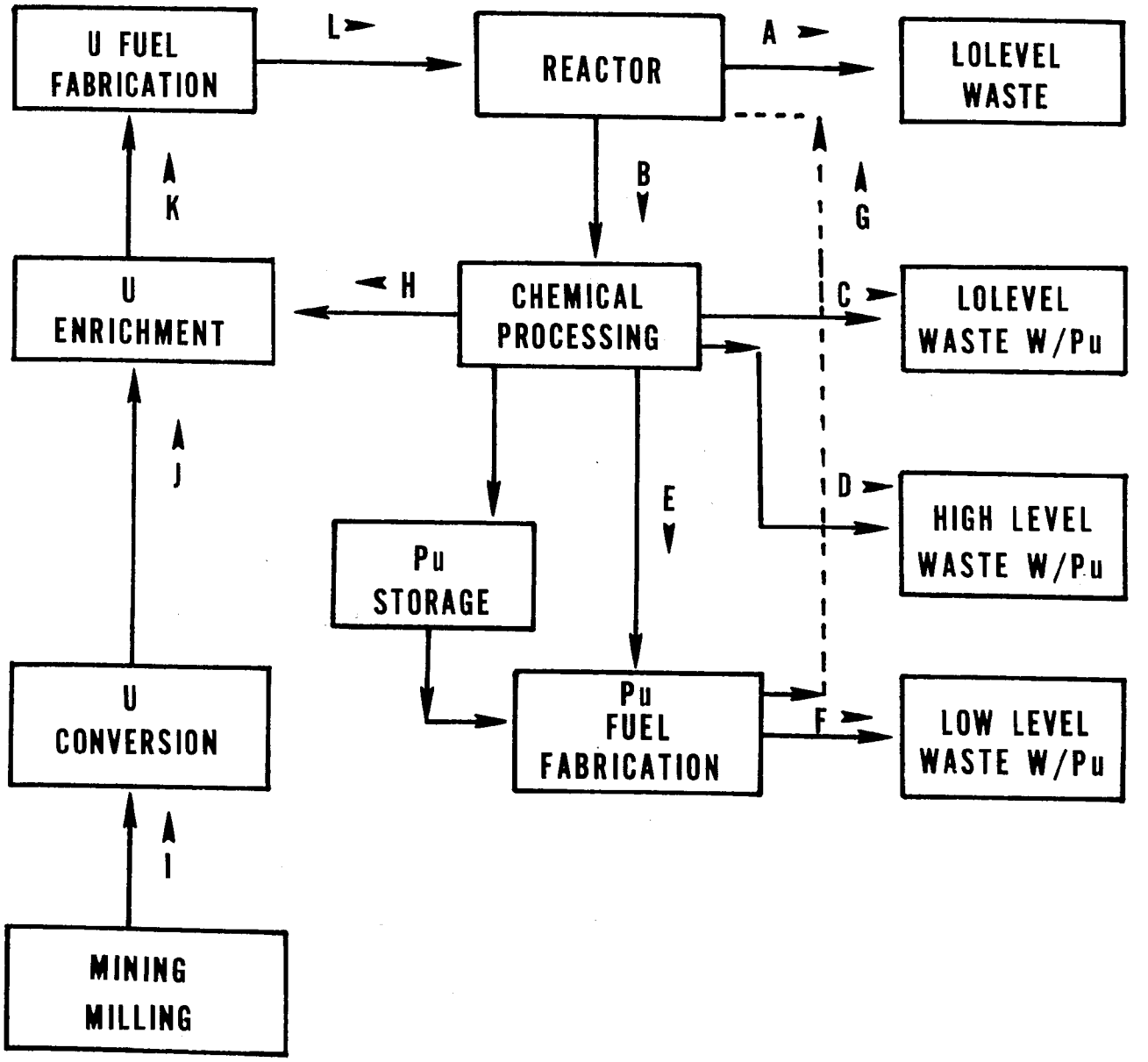


Figure 6-1 Simplified schematic of transportation requirements for the LWR nuclear power industry

Table 6-1

Summary of transportation parameters for the LWR nuclear power industry

Path (See figure 6-1)	Material	Form	Mode	Annual quantity shipped to facility	Quantity per shipment
A	Miscellaneous	Packaged solids	Truck Rail	3,000 drums per reactor	50 drums 150 drums
B	Spent fuel	Spent fuel assemblies	Truck Rail (Barge)	1,500 MT	0.5 MT 3.0 MT
C	Miscellaneous w/transuranics	Packaged solids and cladding	(a)	--	--
D	High level wastes w/transuranics	Solid	Rail	500 casks (141.5 m^3)	10 casks (2.88 m^3)
E	Plutonium oxide	Non-dispersible solid	Truck	15,000 kg[b] per chemical plant	30 kg[b]
F	Miscellaneous w/transuranics	Packaged solids	Truck	(c)	--
G	Recycle fuel $PuO_2 + UO_2$	Fuel assemblies	Truck	(d)	--
H	Uranyl nitrate	Liquid	Truck	500 MT	5 MT
I	Uranium oxide	Powder (yellow cake)	Truck Rail	15,000 MT	15 MT 38 MT
J	UF_6	Powder (natural U)	Truck	10,000 MT	11 MT (11 MT/cask)
K	UF_6	Powder (enriched U)	Truck	750 MT	11 MT (2.2 MT/cask)
L	Fresh fuel	Fuel assemblies	Truck	30 MT	3 MT

[a]Miscellaneous low and intermediate level wastes are currently stored onsite at the reprocessing plant. Appendix F of 10 CFR Part 50 requires decontamination of reprocessing sites upon decommissioning which will probably result in shipment of these wastes to a repository.

[b]Very limited quantities of plutonium have been shipped to the mixed oxide fabrication plants. In addition, the quantity of plutonium permitted in shipping containers varies widely. Thus, it is not possible to estimate the quantities in this path with any accuracy at this time. Shipment of plutonium would range from 900 kg/yr for a 30 MT mixed oxide plant to 4,500 kg/yr for a 150 MT plant assuming a plutonium content of 3% in the fabricated assemblies.

[c]Insufficient information exists at this time to estimate these quantities. However, it is suspected volumes will be large with plant startups and then decrease gradually.

[d]This path will probably be similar to path L.

Table 6-2

Transportation requirements for a 900 LWR nuclear power program

(low enriched uranium oxide fuel)

Path (See figure 6-1)	Number of facilities shipped to/from	Total annual shipments	Average kilometers per shipment	Annual population dose (person-rem/yr)
A	from 900 reactors	54,000 (T) 18,000 (R)	805	150[a]
B	from 900 reactors	54,000 (T) 9,000 (R)	1,610	176[a]
D	from 18 chemical plants	900 (R)	4,025	40
E	from 18 chemical plants	9,000 (T)	<800[b]	Not applicable
H	to 9 enrichment plants	5,400 (T)	1,208	0.24[c]
I	to 6 conversion plants	6,000 (T)	1,610	Negligible
J	to 9 enrichment plants	7,500 (T)	805	Negligible
K	to 36 fabrication plants	2,600 (T)	1,208	0.12[c]
L	to 900 reactors	9,000 (T)	1,610	0.54[c]

[a]90% rail and 10% truck weighted values.

[b]In many cases it is suspected the mixed oxide (PuO_2 - UO_2) fabrication plant will be located close to or adjacent to the chemical reprocessing plant. However, it is difficult to estimate the average shipping distances at this time.

[c]A dose rate of 0.1 mrem per hour at 3 meters from the apparent centerline of the shipping route was used for these shipping paths. Thus the population dose per km is 0.01 that listed in table 6-3.

recycle is shown in the figure and is summarized in table 6-1, since current trends indicate that such recycle will occur in the future.

6.2 Shipping Containers

A variety of containers are required to ship the diverse radioactive materials listed in table 6-1. Rather than describe the individual containers, many of which are under development and/or in design, a list of the major design requirements appears more appropriate:

a. radiation exposure limits external to the container

b. criticality control

c. heat dissipation

d. survival and maintenance of integrity under accident conditions, and

e. weight limitations.

6.3 Exposure Levels

Exposures from the normal transportation of radioactive materials for the nuclear power industry are estimated by assuming that direct radiation is the only source and using the dose rate limit imposed by Department of Transportation (DOT) regulations (10 mrem per hour at 6 feet from the surface of the shipping vehicle)(1). DOT regulations also impose requirements on the dose and temperature at the surface of the vehicle or container. However, these are of little consequence in estimating population exposure. In a recent analysis (2), the AEC used a value

of 10 mrem per hour at 10 feet from the apparent centerline of the shipping route instead of the DOT requirement. This AEC value was used here (10 mrem/hour at 3 meters) for purposes of consistency and comparison.

6.4 Radiological Impact

The dose to the population can be calculated by assuming a uniform population density along a 2 kilometer corridor of the road or track (1 kilometer on either side of the road or track centerline). Since the exposure at any distance from the road is dependent on the given distance and on the speed of the vehicle, it is necessary to integrate the dose to the individual at the given point as the shipment passes. An equation was set up and the solution is:

D = individual dose at given distance (mrem)

$$D = \frac{K}{(ab)^{1/2}} \tan^{-1} \left[t \left(\frac{b}{a} \right)^{1/2} \right]$$

where:

$K = 0.025$ mrem-m^2/s

a = constant = (distance in meters from centerline of path)^2

b = f(t) = (velocity of vehicle in m/s)^2

$$t = \frac{2{,}000}{\text{velocity of vehicle}} \text{ in seconds}$$

The results using this solution and vehicle speeds of 320 km/day (representative of rail transport) and of 966 km/day (representative of truck transport) are listed in table 6-3.

The dose to an individual standing 100 meters from the centerline of the path is 1.04×10^{-4} mrem for the passage of one shipment at 320 km/day.

The AEC presented a similar analysis in a recent report (2) wherein corrections were made for a buildup factor and air attenuation. With these corrections numerical integration was required to solve the equation. An analysis was performed using the simplified method presented here for the same distance parameters used by the AEC. This analysis indicated that the integrated dose for both methods agreed to within 10% of each other. However, there were significant differences in the distribution of the dose with distance from the centerline. The method used here produced lower results close to the shipping path and higher results at the further distances (0.5 to 1.0 km). It is concluded that both methods are acceptable since a uniform population density is considered.

The population dose per kilometer (table 6-3) was obtained by lumping the population in groups at 25 meter intervals from 25 to 150 meters from the road or track centerline and at 100 meter intervals from 200 to 1,000 meters from the centerline, and then multiplying by the individual dose at the respective distance on both sides of the path. These results were then summed for the particular vehicle speed. The population density used was 127.4 persons per square kilometer. These calculations were performed for both rail (320 km/day) and truck (966 km/day)

Table 6-3

Individual dose at a given distance from the apparent centerline of the shipping route for the passage of one shipment[a]

Distance (meters)	Dose at 320 km/day[b] (mrem)	Dose at 966 km/day[b] (mrem)
25	4.24×10^{-4}	1.4×10^{-4}
50	2.11×10^{-4}	6.97×10^{-5}
75	1.49×10^{-4}	4.59×10^{-5}
100	1.04×10^{-4}	3.41×10^{-5}
125	8.22×10^{-5}	2.7×10^{-5}
150	6.81×10^{-5}	2.25×10^{-5}
200	4.99×10^{-5}	1.65×10^{-5}
300	3.22×10^{-5}	1.06×10^{-5}
400	2.34×10^{-5}	7.76×10^{-6}
500	1.82×10^{-5}	5.99×10^{-6}
600	1.45×10^{-5}	4.79×10^{-6}
700	1.2×10^{-5}	3.97×10^{-6}
800	1.02×10^{-5}	3.36×10^{-6}
900	8.6×10^{-6}	2.88×10^{-6}
1,000	7.49×10^{-6}	2.5×10^{-6}
Population dose (person-rem per km)	1.1×10^{-5}	3.7×10^{-6}

[a]Based on 10 mrem per hour at 10 feet (~3 m) from the apparent centerline of the shipping route.

[b]Accurate to one significant figure.

transport.

The population doses were then multiplied by the total kilometers traveled for each path, as presented in table 6-2. Summation of the population doses for each pathway results in the total dose of 367 person-rem/yr for the 900 unit reactor program. This yields a value of 0.41 person-rem per reactor per year. If a load factor as low as 0.7 is used, this planned impact is not more than 0.6 person-rem per GW(e)-yr.

An additional planned exposure must be considered for radioactive material shipments to account for the stops made enroute. Trains stop at rail stations and trucks at truck stops and, in most cases, it is reasonable to expect that the population density will be much greater at the stop than the 127.4 persons per square kilometer used above. However, for the most part, the stops will be of short duration. Therefore, this planned exposure should lead to an impact of about the same magnitude as that from the moving vehicle and it is recommended that the planned impacts listed above be doubled to account for this additional exposure (2). The totals are:

$$\text{Planned impact from normal transportation} = \frac{0.82 \text{ person-rem}}{\text{reactor-year}} = \frac{1.2 \text{ person-rem}}{\text{GW(e)-yr}}$$

These values do not include occupational exposure. It is expected that transport workers will be considered radiation workers with the exception of rail workers (brakemen, conductors, engineers, etc.).

The maximum dose to an individual is expected to occur to railroad brakemen. In the AEC analysis it was estimated that brakemen would spend from 1 to 10 minutes in the vicinity of the cask car during a trip for an average exposure of about 0.5 mrem per shipment. Assuming a highly unusual set of circumstances in which one brakeman receives such an exposure from all incoming shipments of spent fuel to a chemical plant (450 annual shipments), the maximum individual exposure rate would be 225 mrem per year to this particular brakeman. However, it appears difficult to envision such circumstances and, if encountered in practice, steps could be taken to monitor and/or reduce this exposure.

6.5 Health Effect Impact

The conversion factor used to obtain health effects for the radiological impact as estimated in section 6.4 are:

Total body irradiation:

200 deaths per year per 10^6 annual person-rem;

200 nonlethal cancers per year per 10^6 annual person-rem.

Gonadal irradiation:

300 serious effects per year per 10^6 annual person-rem.

Health effects are defined as the sum of lethal, nonlethal and genetic effects.

For the transportation impact in the postulated 900 unit LWR nuclear power program the health effects due to radiation exposure are listed in table 6-4.

6.6 Cost Effectiveness of Reducing Transportation Exposure

Table 6-4

Health effects to members of the general population
from transportation of radioactive materials associated with nuclear fuel cycle

Pathway	Critical organ	Health effects/reactor-yr[a]		
		Mortality	Nonfatal effects	Genetic effects
Direct gamma dose	Whole body	0.0002	0.0002	0.0003

Total health effects for radioactive transportation associated with 30 years of operation of one reactor facility = 0.02

[a]Listed health effects will result from transportation associated with one year of operation of one reactor.

The transportation of radioactive materials in a reactor program must be viewed in perspective when attempting to perform a cost-effectiveness analysis for reducing the radiological impact. The quantity of radioactive material which must be shipped in a reactor program (or per year per reactor) is fixed. The impact resulting from the shipment of this material is dependent on the dose rate at a given distance from the shipping container (or cask). If the dose-rate limit is reduced by a factor of 2, implying the use of additional shielding, the impact will be reduced by a factor of 2. However, from a practical standpoint, it is much more likely that this lower dose rate could best be met by reducing the quantity of material in each shipment since the truck casks are currently at legal weight limits for most states and rail cask weight must be held to some limit to prevent special routing requirements. Crane capacities must also be considered. This situation would lead to a lower impact per shipment but a larger number of shipments, resulting in about the same total impact. Therefore, it is likely that additional restraints (i.e., other than a dose rate limit external to the cask) must be considered to insure that the total transportation impact is reduced if the analysis indicates this is a cost-effective requirement.

The fabrication costs for the General Electric IF 300 rail cask for spent fuel are approximately one million dollars per cask. Current estimates for the larger G.E. IF 400 cask range

between 1.5×10^6 and 1.75×10^6 dollars for fabrication. The capacities of these casks are about 3 MT and about 6 MT, respectively. Design and licensing costs must be written off for some number of casks. For purposes of this analysis, the costs will be increased by about 20% to account for these factors. Using these figures, a straight linear relationship can be developed on a per-shipment basis for cost vs dose reductions for spent fuel shipments. However, the usefulness of this relationship is doubtful as discussed above. The same type of analysis applies to the shipment of high-level solidified waste from the chemical plant to the disposal site.

The shipment of low- and intermediate-level radioactive wastes from reactors to waste disposal sites contributes about 40% of the total impact from transportation (see table 6-2). It appears this exposure can be reduced through the addition of shielding to the transportation vehicle or by segregation of this intermediate-level waste from the low-level waste and the addition of shielding to the intermediate-level shipping container and/or vehicle. About 80% of the radioactivity in these wastes is in about 3% of the volume (2). Thus, segregation with selective shielding appears particularly attractive.

The cost of providing shielding for a truck trailer to reduce the exposure rate by a factor of two is estimated as follows: About 1.26 cm of lead would be placed on the sides, ends, and bottom of the trailer. The added weight of the lead would

be about 18,000 kg, and the cost would be about $7,000. The total cost, including installation and protective steel sheet, would probably be about $20,000. However, this amount of lead would reduce the payload by a factor of at least two and thus the reduction in exposure achieved through use of shielding and the smaller quantity of material per shipment would be offset by more numerous shipments. If segregation of the wastes was required and one section of the truck shielded, it is estimated that about 5,500 kg of lead would be required at a cost of about $2,000, with a total cost of about $7,000. Such shielding would also reduce the payload, but would provide a dose reduction of about 2 for 80% of the waste for a total dose reduction factor of about 1.7. However, there would be additional costs involved in segregating the wastes. It is estimated that this cost would not exceed one man-year per reactor per year resulting in an annual cost of about $20,000. The results of these estimates are shown in figure 6-2.

Route control appears more attractive for the reduction of exposure than the addition of shielding. Figure 6-3 depicts the dose reduction which can be achieved through selection of routes with lower population densities. The population dose due to stops along the route would probably be lower for a lower population density. However, since this factor is not well known in any case, the same value was used as in section 6.4 as a lower limit of exposure.

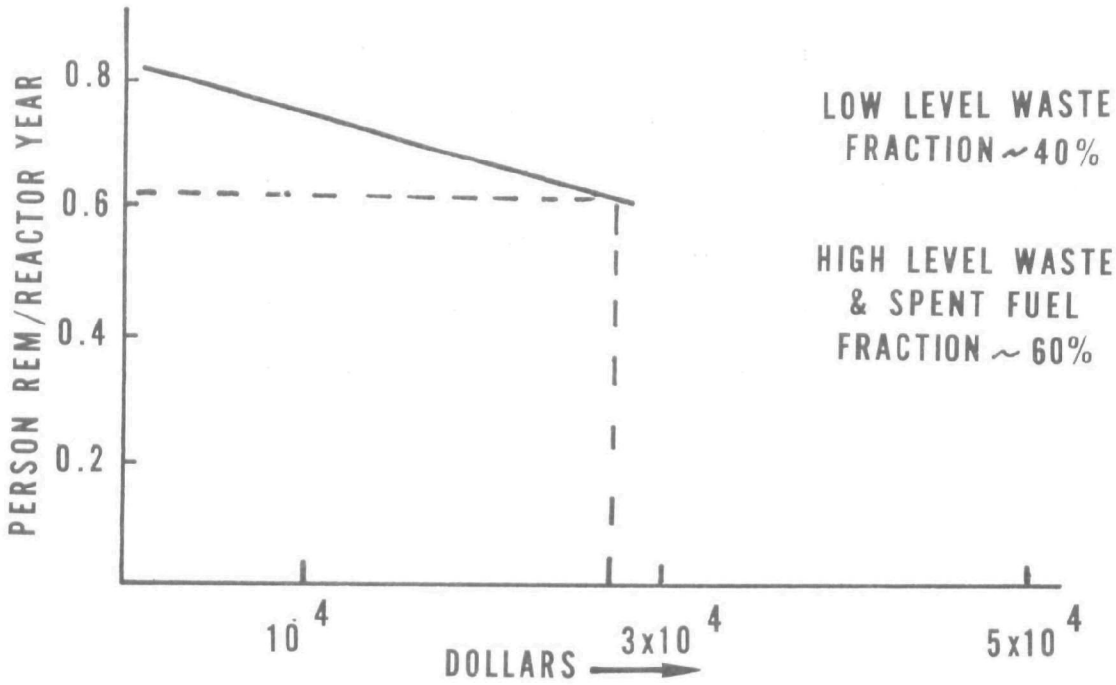


Figure 6-2 Cost effectiveness for segregation of low level wastes with an additional shielding compartment on the truck (1 truck per reactor year)

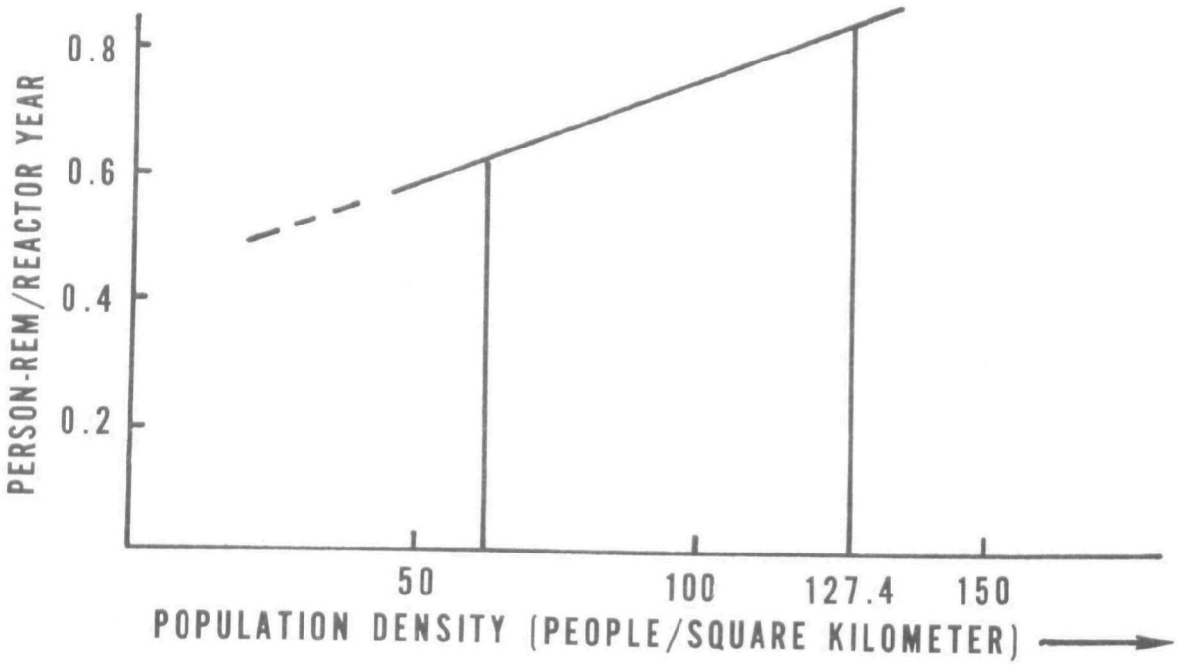


Figure 6-3 Impact reduction available with route control through various areas of population density (impact from stops remains constant)

The costs of route control would be highly variable because of its administrative nature. However, it is estimated that one-half man-year per reactor per year would suffice in accounting for this cost. Additional costs would probably involve the total distance traveled by the vehicles and would be highly site dependent. For purposes of this review a cost of $20,000 per reactor per year would appear adequate for route control requirements.

Appendix A

Exposure Pathway, Radiation Dose and Health Effect Parameters

1.0 Introduction

This appendix explains the assumptions and parameters used in (1) calculating how radioactive effluents discharged from fuel supply operations are transported through the environment, (2) calculating the doses resulting from the interaction of radiation with man, and (3) converting radiation doses to health effects.

The radioactive effluents include uranium and associated daughter products both as aerosols and soluble liquid effluents. Source terms were derived from operations considered reasonable for a "model facility." Generalized "model exposure pathways" were used to relate radioactive airborne and waterborne effluent concentrations to radiation doses to humans. These pathways do not apply to any specific site but were chosen to be representative of what may take place around a "model facility." Dose conversion factors were derived primarily from "ICRP Publication 2" (1) and "Ionizing Radiation: Levels and Effects" (a report of the United Nations' Scientific Committee on the Effects of Atomic Radiation)(2). Conversion factors to express radiation doses as resultant health effects were derived from the BEIR Report (3).

The term "uranium" as used in the Fuel Supply Technical Report is defined as the sum of uranium-234, uranium-235, and uranium-238. It does not include any other radionuclides nor does

it define any particular isotopic ratio. The terms "uranium-238," "uranium-235," "uranium-234," and "thorium-230" are defined as referring to the named isotope only. Daughter products and the effects of daughter products are not included. In contrast, the term "radium-226" may include varying amounts of short-lived daughters depending upon the exact situation.

2.0 Model Exposure Pathways

2.1 Air Pathway

The following assumptions apply to the respective terms used in the atmospheric pathway model:

$(\bar{\chi}/Q)_{max}$ is the meteorological dispersion factor that relates the plant source term to the dispersed radionuclide concentrations in air at a given downwind distance. $\bar{\chi}$ is the yearly average concentration of activity to which a human at the specified downwind distance is exposed (pCi/m^3). Q is the rate at which an airborne radionuclide leaves the site (pCi/s).

It was assumed that the plant boundary lies between 0.5 and 1.5 kilometers from the point of release. $\bar{\chi}/Q$ was then estimated (4) assuming a ten meter stack for three types of sites at 0.5, 1.0, 1.5 kilometers (table A-1). $(\bar{\chi}/Q)_{max}$ was taken to be the average of these values multiplied by 2 to obtain the maximum sector concentration. The standard deviation was roughly estimated as 1/6 the range of these values. This value for $(\bar{\chi}/Q)_{max}$ of $6 \pm 4 \times 10^{-6}$ was used for all fuel supply facilities.

C(a) is the ratio of the average individual radiation dose

Table A-1

Average $(\bar{\chi}/Q)$ values vs distance for h = 10 meters[a] (4)

Site	Distance (km)			
	0.5	1.0	1.5	3.0
River	9×10^{-6}	3×10^{-6}	2×10^{-6}	7×10^{-7}
Lakeshore	4×10^{-6}	1×10^{-6}	5×10^{-7}	2×10^{-7}
Seashore	6×10^{-6}	2×10^{-6}	1×10^{-6}	5×10^{-7}
Average	6×10^{-6}	2×10^{-6}	1×10^{-6}	5×10^{-7}

For the maximum sector concentration, multiply these average values by 2.

[a]h = height of release of gaseous effluent in meters.

within 80 kilometers of the plant to the maximum individual dose at the plant boundary. Calculating C(a) for 50 reactor sites gives a value of 2.3×10^{-4}. This value was accepted for fuel supply facilities.

The radial distance of 80 kilometers was chosen to define the members of the general population exposed to a specific source term because the distance is large enough to include nearby large population centers yet small enough so that the area effected can be considered a local area. Beyond this distance, the diffusion equations that characterize the source term plume are not considered reliable. No correction for particulate depletion from the plume was made.

2.2 Water Pathway

The standard river model does not represent a specific site but was constructed from parameters that are believed to be reasonable and credible. Because most fuel supply facilities are located on rivers, no attempt was made to construct sea coast or lake water exposure pathways. The river model assumes:

1. A river flowing past the outfall at a rate of 280 cubic meters per second.

2. The maximum exposed individuals are those assumed to drink water continuously from the river following complete dilution of effluents by the river water.

3. The average exposed individuals are represented by X people per kilometer of river for 300 kilometers downstream from

the outfall that drink water from the river.

$(\bar{\chi}/Q)_{river}$ is the dispersion factor that relates the plant source term to the dispersed radionuclide concentration in water. $\bar{\chi}$ is the yearly average concentration of activity (pCi/l) to which a human is exposed by drinking water from the river following complete dilution of effluents by the river water. Q is the rate that a waterborne radionuclide leaves the site (pCi/s).

C(w), an additional dilution factor of 0.1, is assumed for members of the general population exposed to the effluents to accommodate additional river dilutions from tributaries, loss of activity to surfaces of the river, multiple uses of water, etc.

The dispersion factors discussed above are summarized in table A-2.

3.0 The "Model" Population at Risk

Table A-3 summarizes the various "model" population groups used for computing somatic and genetic dose effects.

3.1 Through the Air Pathway

The total population within 80 km of 50 reactor sites, taken primarily from environmental impact statements, indicates that there is an average population of 1.5×10^6 people as projected to 1980. The doubling time for this population is assumed to be 40 years based upon an annual increase in population of about 2.5%. Fuel supply facilities are assumed to be similar except for mills which are located in sparsely populated western states. For mills it is assumed that 5.5×10^4 persons are

Table A-2

Dispersion factors for airborne and waterborne pathways

Pathway	$(\bar{x}/Q)_{maximum\ sector}$	$(\bar{x}/Q)_{river}$	C
Airborne	$6 \pm 4 \times 10^{-6}\ \frac{pCi/m^3}{pCi/s}$		$C(a) = 2.3 \times 10^{-4}$[a]
Waterborne		$4 \times 10^{-6}\ \frac{pCi/m^3}{pCi/s}$[b]	$C(w) = 0.1$[c]

[a]Radius of 80 km

[b]River flow - 280 m^3/s; for mills in western states where rivers are smaller, assume a river flow of 14 m^3/s and multiply this factor by 20.

[c]Apparent length of river - 300 km

Table A-3

Population models for air and water pathways

Pathway	Area of Concern	Population	Biological effects
Air	Within 80 km	1.5×10^6 persons	Somatic
		0.8×10^6	Genetic
	Within 80 km of mills	5.5×10^4 persons	Somatic
		1.3×10^4	Genetic
Water	Within 300 km	0.6×10^6 persons	Somatic
		0.3×10^6 persons	Genetic
	For mills, within 300 km of outfall		
		4.4×10^4 persons	Somatic
		2.2×10^4 persons	Genetic

why 1/4 for genetic?

within 80 km of the site. Only one-half of these populations are used for genetically significant dose calculations.

3.2 Through the Water Pathway

It is estimated that 2,000 persons/km of river for 300 km downstream from the outfall drink water taken from the river. This is a total of 0.6×10^6 persons (1980). The number was obtained by dividing the number of people in the various watersheds by the length of rivers over 600 miles long. The population-at-risk for mills was reduced by a number proportional to the population density ratios, 0.037 and multiplied by 2 because it may be assumed that in arid territory, the population is more concentrated around water. The resulting population at risk is 44,000. Doubling time of the population is assumed to be 40 years. Only half of the population is used for genetically significant dose calculations.

4.0 Radiation Dose Conversion Factors

4.1 Airborne Radionuclides

The dose conversion factors for insoluble (class Y) alpha-emitting aerosols given in table A-4 were estimated from the dose conversion factor for an insoluble aerosol of plutonium-239 oxide by a consideration of the ratio of the alpha energies of the radionuclides to the alpha energy of plutonium-239 (table A-5). The continuous inhalation of 1 pCi/m^3 of insoluble plutonium oxide is taken to deliver 12 rem/yr to the lung. This value is based on the ICRP Report of Committee II of 1959 plus a

Table A-4

Dose conversion factors for the airborne pathway

Radionuclide	Aerosol Class	Organ at Risk	Dose conversion factor mrem/yr per pCi/m^3
Plutonium-239	Y	Lung	12,000
Uranium-238	Y	Lung	10,000
	D	Bone	150
Uranium-234	Y	Lung	10,000
	D	Bone	150
Uranium-230	Y	Lung	11,000
Radium-226	Y	Lung[a]	11,000
	D	Bone	300
Radon-222	-	Lung (T.B.)[b]	4

[a]Assume radon-222 and therefore all radium-226 daughters escape from the aerosol particle and only radium-226 (i.e., Σ EF(RBE) n = 49) contributes to the dose.

[b]T.B. means tracheobronchial region.

Table A-5

Dose conversion factors for airborne insoluble particulates

Radionuclide	Σ E (RBE)n[a] (MeV)	Dose conversion factor (mrem/yr per pCi/m^3)
Plutonium-239	53	12,000
Uranium-238	46	10,000
Uranium-235	46	10,000
Uranium-234	46	10,000
Thorium-230	48	11,000
Radium-226	49[b]	11,000

[a] ICRP report of Committee II - assume uranium-238 and uranium-234 equal in energy to uranium-235

[b] Assume all radon escapes from the aerosol particle

correction factor of 8 (5) to convert from the 1959 lung model to the newer lung model recommended by the ICRP Task Force report, "Deposition and Retention Models for Internal Dosimetry of the Human Respiratory Tract" (6).

Dose conversion factors for soluble aerosols were calculated from ICRP Publication 2 values using bone as the critical organ (table A-6).

It was recognized that doses calculated for the lymph nodes in the tracheobronchial region of the lung are much higher (by a factor of 35) than the lung dose if the ICRP "New Lung Model" is used. However, the ICRP has recommended that lymphatic tissue not be considered the critical organ in inhalation exposure to plutonium (7). This apparently is because at present there are no reports of primary tumors of the tracheobronchial lymph nodes and because it is believed that protection of the more radiosensitive lung tissue will provide more than adequate protection to the lymph nodes. Radiation dose to the lymph nodes of the tracheobronchial region will not be used as a criterion for setting environmental standards at this time; and therefore, this dose has not been included in tables A-10 and A-11.

4.2 Waterborne Radionuclides

The uranium dose conversion factors (table A-7) for the waterborne pathway were taken from dose calculations endorsed by the UNSCEAR report (2) for radiation dose rates that result to soft tissue (gonads) and bone surfaces, the tissue at risk for cancer

Table A-6

Dose conversion factors for airborne soluble particulates

Radionuclide	168 h week $(MPC)_a$[a] ($\mu Ci/cm^3$)	Critical organ	Dose conversion factor (mrem/yr per pCi/m^3)
Uranium-238	2×10^{-10}	Bone	150
Uranium-235	2×10^{-10}	Bone	150
Uranium-234	2×10^{-10}	Bone	150
Thorium-230	8×10^{-13}	Bone	38,000
Radium-226	1×10^{-10}	Bone	300
Plutonium	6×10^{-13}	Bone	50,000

[a]ICRP Report of Committee No. II - $(MPC)_a$ is equivalent to 30 rem/yr to bone.

Table A-7

Dose conversion factors for the water pathway

Radionuclide	Organ at risk	Dose conversion factor (mrem/yr per pCi/liter)[a,b]
Uranium-238	Bone	9
	Soft tissue	0.9
	Marrow	2
Uranium-234	Bone	9
	Soft tissue	0.9
	Marrow	2
Thorium-230	Bone	1
	Soft tissue	-
	Marrow	0.2
Radium-226	Bone	12
	Soft tissue	0.4
	Marrow	2.4

[a]It is assumed that adults consume 2 liters/day of water and that the listed dose rates will result if activity in the water is 1 pCi/liter.

[b]Listed are equilibrium dose rates that result from equilibrium body burdens. For example, radon-226 dose rate is not assumed to be reached until 16 years following start of exposure.

induction, following the ingestion of known amounts of uranium naturally present in the human diet (table A-8). While it is recognized that uptake following ingestion in food may not necessarily lead to the same rate of uptake as the ingestion of uranium in drinking water, it is believed that this is the best model currently available for long-term chronic ingestion of uranium. Note that these dose conversion factors refer only to the uranium radionuclide listed and do not include daughter product radiations. These must be listed and calculated separately. However, because the factors are the same for uranium-234, uranium-235, and uranium-238 and the source terms are listed as total uranium (i.e., the sum of uranium-234, uranium-235, and uranium-238), a single calculation suffices.

The thorium-230 dose conversion factor was calculated from ICRP Publication 2 and corrected for surface dose; the radium-226 dose conversion factor was calculated in the same fashion as uranium (tables A-9 and A-10).

The dose to the bone marrow was calculated separately (2). For this site, cancer risks are calculated using the health effect factor for leukemia.

5.0 Health Effect Conversion Factors

Health effect conversion factors were abstracted from the BEIR report and rescaled for the dose to bone surfaces. They are listed in table A-11.

The total integrated radiation dose is multiplied by these

Table A-8

Dose conversion factors for uranium for the water pathway (2)

Intake — 1 µg/day natural uranium
0.68 pCi/day total uranium (^{238}U, ^{235}U, ^{234}U)

Equilibrium body burdens — 0.1 to 0.9 ng/g soft tissue
20 to 30 ng/g bone ash

Dose delivered (calculated by the method of Spiers)

Total uranium at 9.2 pCi/kg bone yields:

0.3 mrad/yr to bone (tabecular bone, surfaces)
0.06 mrad/yr to bone marrow
0.03 mrad/yr to soft tissue (gonads)

Assuming a quality factor of 10, 0.68 pCi total uranium intake per day yields:

3 mrem/yr to bone (tabecular bone, surfaces)
0.6 mrem/yr to bone marrow
0.3 mrem/yr to soft tissue (gonads)

For an intake of 2 liters of liquids per day containing 1 pCi/l total uranium, yields:

8.8 mrem/yr to bone (tabecular bone, surfaces)
1.8 mrem/yr to bone marrow
0.9 mrem/yr to soft tissue (gonads)

Table A-9

Dose conversion factors for radium-226 for the water pathway (2)

Intake 1 pCi/day radium-226

Body burden 7.6 pCi/kg bone
40 pCi/skeleton

Dose delivered (calculated by the method of Spiers)

0.6 mrad/yr to bone (tabecular bone, surfaces)
0.1 mrad/yr to bone marrow
0.02 mrad/yr to soft tissue (gonads)

Assuming a quality factor of 10, 1 pCi ^{226}Ra/day intake yields:

6 mrem/yr to bone (tabecular bone, surfaces)
1 mrem/yr to bone marrow
0.2 mrem/yr to soft tissue (gonads)

An intake of 2 liters of liquids per day containing 1 pCi ^{226}Ra/l yields:

12 mrem/yr to bone (tabecular bone, surfaces)
2.4 mrem/yr to bone marrow
0.4 mrem/yr to soft tissue (gonads)

Table A-10

Dose conversion factors for the water pathway for soluble radionuclides calculated from the recommendations of the ICRP

Radionuclide	168 h week $(MPC)_w$ ($\mu Ci/cm^3$)	Critical organ	Dose conversion factor (mrem/yr per pCi/l)
Thorium-230	2×10^{-5}	Bone	2.5
		Bone surfaces	1.0[a]
		Marrow	0.2[a]

[a]Ratio of bone dose to bone surface dose and bone marrow dose assumed to be the same as for uranium.

Table A-11

Health effects conversion factors

Critical organ	Health effect	Health effects conversion factor (Events per 10^6 rem aggregate dose)		
		Mortality	Nonfatal cancers	Genetic effects
Lung[a]	Cancer of lung	50	0	0
Bone[b]	Cancer of skeleton	16	16	0
Bone marrow[b]	Leukemia	54[a]	0	0
Total soft tissue organs other than bone	Cancers and genetic effects	150	150	300

[a]Insoluble aerosol exposure pathway

[b]Ingestion exposure pathway

factors to give the total health effects that are expected (committed) as the result of the radiation exposure. These effects occur over a period of years following exposure.

For uranium ingested through the water pathway, the majority of this uranium body burden is located in the bone. This uranium is considered to irradiate the bone with the corresponding health effect of cancer of the skeleton. In addition, 20% of the dose is assumed to irradiate the bone marrow with the resulting health effect being leukemia. The mortality risk from leukemia is 54 cases per 10^6 rems to the bone marrow. The risk of mortality from leukemia from bone irradiation by alpha-emitting radionuclides in the bone is therefore 0.2 x 54 cases per 10^6 rems.

For soft tissue organs, including the gonads, uranium burdens are considered to provide an average organ dose with the corresponding health effects assumed to be the sum of all effects on each individual soft tissue organ. In addition, this dose is delivered to the gonads causing genetic health effects.

6.0 Radiological Effects Calculations

Given the source term and the factors discussed above, the various radiation doses to individuals in the general population, and the expected committed health effects resulting from these doses can be calculated. These calculations are presented in tables A-12, A-13, and A-14. A simplified flow sheet for radiation dose and health effects calculations is given in table

Table A-12

Airborne pathway dose calculations for model facilities - current best technology[a]

Model facility	Radio-nuclide	Source term (Ci/yr)	$(\bar{x}/Q)_{max}$ (s/m^3)	Critical Organ	Dose conversion factor ($\frac{mrem/yr}{pCi/m^3}$)	Maximum exposure at boundary (mrem/yr)	C(a)	Average exposure dose within 80 km (mrem/yr)	Persons Exposed	Aggregate Somatic Organ Dose (rem-per yr/facility-yr)
Mill	U total	0.1	6×10^{-6}	Lung	1.0×10^{4}	1.9×10^{2}	2.3×10^{-4}	4.4×10^{-2}	5.5×10^{4}	2.4×10^{0}
	^{226}Ra	0.06	(0.1904)	Lung	1.1×10^{4}	1.3×10^{2}		2.9×10^{-2}		1.6×10^{0}
	^{230}Th	0.06	?	Lung	1.1×10^{4}	1.3×10^{2}		2.9×10^{-2}		1.6×10^{0}
					Total	4.5×10^{2}	Total	10.2×10^{-2}	Total	5.6×10^{0}
Conversion (wet solvent extraction)	U total insoluble	0.015		Lung	1.0×10^{4}	2.9×10^{1}		6.6×10^{-3}	1.5×10^{6}	9.9×10^{0}
	U total soluble	0.008		Bone	1.5×10^{2}	2.3×10^{-1}		5.3×10^{-5}		7.9×10^{-2}
Conversion (hydrofluor)	U total insoluble	0.038		Lung	1.0×10^{4}	7.2×10^{1}		1.7×10^{-2}	1.5×10^{6}	2.5×10^{1}
	U total soluble	0.019		Bone	1.5×10^{2}	5.4×10^{-1}		1.2×10^{-4}		1.9×10^{-1}
Enrichment	U total soluble	0.045		Bone	1.5×10^{2}	1.3×10^{0}		3.0×10^{-4}	1.5×10^{6}	4.4×10^{-1}
Fabrication	U total insoluble	0.005		Lung	1.0×10^{4}	9.5×10^{0}		2.2×10^{-3}	1.5×10^{6}	3.3×10^{0}

a $\frac{Ci}{yr} \times \frac{1\ yr}{3.15 \times 10^{7}} \times \frac{10^{12}\ pCi}{Ci} \times \frac{pCi/m^3}{pCi/s} \times \frac{mrem/yr}{pCi/m^3}$ = mrem/yr;

mrem/yr $\times \frac{10^{-3}\ rem}{mrem} \times$ persons $\times$ facility-yr = rem/yr aggregate dose per facility-yr

Table A-13

Waterborne pathway dose calculations for model facilities - current best technology

Model facility	Radio-nuclide	Source term (Ci/yr)	$(\bar{\chi}/Q)R$ $(\frac{pCi/l}{pCi/s})$ (0.127)	Critical organ	Dose conversion factor $(\frac{mrem/yr}{pCi/l})$	Maximum exposure dose (mrem/yr)	C(w)	Average exposure dose (mrem/yr)	Persons exposed	Aggregate somatic organ dose (rem per yr/facility-yr)
Mill	U total	0.1	1×10^{-6} x 20[a]	Bone	9	2.2×10^{0}	0.10	2.2×10^{-1}	4.4×10^{4}	1.0×10^{1}
				Soft tissue	0.9	2.2×10^{-1}		2.2×10^{-2}		1.0×10^{0}
	^{226}Ra	0.06		Bone	12	1.8×10^{0}		1.8×10^{-1}		8.1×10^{0}
				Soft tissue	0.4	6.2×10^{-2}		6.1×10^{-3}		2.7×10^{-1}
	^{230}Th	3.5		Bone	1	8.9×10^{0}		8.9×10^{-1}		3.9×10^{1}
				Soft tissue	-	-		-		-
Conversion (wet solvent)	U total	2	4×10^{-6}	Bone	9	2.3×10^{0}		2.3×10^{-1}	6.9×10^{5}	1.4×10^{2}
				Soft tissue	0.9	2.3×10^{-1}		2.3×10^{-2}		1.4×10^{1}
	^{226}Ra	0.006		Bone	12	9.1×10^{-3}		9.1×10^{-4}		5.5×10^{-1}
				Soft tissue	0.4	3.1×10^{-4}		3.0×10^{-5}		1.8×10^{-2}
	^{230}Th	0.0006		Bone	1	7.6×10^{-5}		7.6×10^{-6}		4.6×10^{-3}
Conversion (hydrofluor)	U total	0.8	4×10^{-6}	Bone	9	9.1×10^{-1}		9.1×10^{-2}	6.0×10^{5}	5.5×10^{1}
				Soft tissue	0.9	9.1×10^{-2}		9.1×10^{-3}		5.5×10^{0}
Enrichment	U total	0.6	4×10^{-6}	Bone	9	6.9×10^{-1}		6.9×10^{-2}	6.0×10^{5}	4.1×10^{1}
				Soft tissue	0.9	6.9×10^{-2}		6.9×10^{-3}		4.1×10^{0}
Fabrication	U total	0.5	4×10^{-6}	Bone	9	5.7×10^{-1}		5.7×10^{-2}	6.9×10^{5}	3.4×10^{1}
				Soft tissue	0.9	5.7×10^{-2}		5.7×10^{-3}		3.4×10^{0}

[a]Correction factor for smaller size of the western rivers - mill only

Table A-14

Health effect calculations for model facilities - current best technology

Model facility	Critical organ	Aggregate somatic dose (rem/facility-yr)	H.E.F. x 10^{-6}	Genetic correction	Mortalities	Nonfatal cancers	Genetic events	Pathway	Total effects per exposure (effects/facility-yr)	Total effects per 30 yr exposure (effects/facility-30 yr)
Mill	Lung	5.6×10^{0}	50/0/0		2.8×10^{-4}	0	0	Air	2.8×10^{-4}	8.4×10^{-3}
	Bone	5.7×10^{1}	16/16/0		9.1×10^{-4}	9.1×10^{-4}	0	Water		
	Bone[a]		11/0/0[a]		6.3×10^{-4}	0	0	Water	3.1×10^{-3}	9.2×10^{-2}
	Soft tissue	1.3×10^{0}	150/150/300	-/-/0.5	2.0×10^{-4}	2.0×10^{-4}	2.0×10^{-4}	Water		
				Totals	20.2×10^{-4}	11.1×10^{-4}	2.0×10^{-4}		3.4×10^{-3}	10.0×10^{-2}
Conversion (wet solvents)	Lung	9.9×10^{0}	50/0/0		5.0×10^{-4}	0	0	Air	5.0×10^{-4}	1.5×10^{-2}
	Bone	1.4×10^{2}	16/16/0		22.4×10^{-4}	22.4×10^{-4}	0	Water		
	Bone[a]		11/0/0[a]		15.4×10^{-4}	0	0	Water	12.3×10^{-3}	3.7×10^{-1}
	Soft tissue	1.4×10^{1}	150/150/300	-/-/0.5	21.0×10^{-4}	21.0×10^{-4}	21.0×10^{-4}	Water		
				Totals	63.8×10^{-4}	43.4×10^{-4}	21.0×10^{-4}		12.8×10^{-3}	3.9×10^{-1}
Conversion (hydrofluor)	Lung	2.5×10^{1}	50/0/0		12.5×10^{-4}	0	0	Air	1.3×10^{-3}	3.8×10^{-2}
	Bone	5.5×10^{1}	16/16/0		8.8×10^{-4}	8.8×10^{-4}	0	Water		
	Bone[a]		11/0/0[a]		6.1×10^{-4}	0	0	Water	4.9×10^{-3}	1.5×10^{-1}
	Soft tissue	5.5×10^{0}	150/150/300	-/-/0.5	8.3×10^{-4}	8.3×10^{-4}	8.3×10^{-4}	Water		
				Totals	35.7×10^{-4}	17.1×10^{-4}	8.3×10^{-4}		6.1×10^{-3}	1.9×10^{-1}
Enrichment	Bone	4.4×10^{-1}	27/16/0		0.1×10^{-4}	7.0×10^{-6}	0	Air	0.1×10^{-4}	3.0×10^{-4}
	Bone	4.1×10^{1}	16/16/0		6.6×10^{-4}	6.6×10^{-4}	0	Water		
	Bone[a]		11/0/0[a]		4.5×10^{-4}	0	0	Water	3.6×10^{-3}	1.1×10^{-1}
	Soft tissue	4.1×10^{0}	150/150/300	-/-/0.5	6.2×10^{-4}	6.2×10^{-4}	6.2×10^{-4}	Water		
				Totals	17.4×10^{-4}	12.8×10^{-4}	6.2×10^{-4}		3.6×10^{-3}	1.1×10^{-1}
Fabrication	Lung	3.3×10^{0}	50/0/0		1.7×10^{-4}	0	0	Air	1.7×10^{-4}	5.0×10^{-3}
	Bone	3.4×10^{1}	16/16/0		5.5×10^{-4}	5.5×10^{-4}	0	Water		
	Bone[a]		11/0/0[a]		3.7×10^{-4}	0	0	Water	3.0×10^{-3}	9.0×10^{-2}
	Soft tissue	3.4×10^{0}	150/150/300	-/-/0.5	5.1×10^{-4}	5.1×10^{-4}	5.1×10^{-4}	Water		
				Totals	16.0×10^{-4}	10.6×10^{-4}	5.1×10^{-4}		3.2×10^{-3}	9.5×10^{-2}

[a]Bone marrow dose is 20% of bone dose; the H.E.F. for leukemia is multiplied by 0.2 to give a H.E.F. for use with bone dose.
H.E.F. - health effects factor

A-15.

7.0 The 100-Year Radiation Dose Commitment

7.1 Introduction

A year's release of radioactive material from a fuel supply facility causes an immediate commitment of radiation dose to members of the general population in the vicinity of the facility. This dose is delivered when the radioactive effluents, moving quickly through air and water pathways, are taken up by individuals. This takes place during the year in which the effluent is released; and these radiation doses and the resulting health effects have been calculated in previous sections.

The same radioactive material upon release to the biosphere may, over longer periods of time, find its way back to man through slow, secondary pathways such as resuspension-inhalation and food chains. This causes additional radiation dose and health effects. Although these dose rates are usually quite small, the number of people exposed may be very large so that because of the linear, nonthreshold health effects model, the number of predicted health effects may become significant.

The 100-year dose commitment is an attempt to calculate the radiation dose and health effects that will result when a single year's release of radioactive effluent interacts with man during the following 100 years. To estimate the effect of 30 years of facility operations, the 100-year dose commitment for each year of operations is added together. The health effects that result

Table A-15

Radiation dose and health effect calculations - flowsheet

Source term (pCi/s)	x	$(\frac{\chi}{Q})$	=	Maximum exposure concentration (pCi/l or m^3)
	x	Dose conversion factor	=	Dose to maximum exposed individual (organ mrem per yr/facility-yr)
	x	C(a or w)	=	Dose to average exposed individual (organ mrem per yr/facility-yr)
	x	Number of people exposed	=	Total integrated dose (organ rems per yr/facility-yr)
	x	Health effects factor	=	Total number of health effects (effects/facility-yr)

from the 100-year dose commitment are in addition to the effects that occur from the exposure to or immediate uptake of radioactive effluents by people during the year of release.

7.2 Source Terms

Source terms are identical to those used for each model facility in the proceeding sections.

7.3 Exposure Pathway

Natural uranium is released from model facilities of the uranium fuel supply in liquids in a soluble form and to air in both soluble and insoluble forms. The uranium released in solution form is washed to the ocean or is deposited in silt along the river bed. There are presently no known or postulated exposure pathways from uranium buildup in the ocean or silt. The assumption is made that they represent an infinite sink; thus, liquid releases do not contribute to the 100-year dose commitment.

Air releases of natural uranium, whether soluble gases or insoluble particulates, settle out at some point after release rather than remain airborne. The exact pattern of settling depends on the meteorology at the time of release, the geology around the plant, stack height, and other related parameters. It is assumed that 20% of the release will be uniformly distributed within 80 km of the model plant, the remaining 80% will be uniformly distributed across the eastern half of the United States, and that the deposited uranium will be uniformly distributed throughout the top 15 cm of soil.

Uranium occurs naturally in the soil and has an average concentration of 2.8 micrograms per gram of soil, (μg/g) (2). The average daily intake is estimated to be 1 μg; less than 1% of this intake is from inhalation of resuspended particles (2). The major pathway is ingestion through the food chains.

It may be assumed that the increase in concentration of uranium in the soil due to releases from the uranium fuel supply facilities causes a proportional increase in the average daily intake of uranium. Since the radiation dose from background levels of uranium has been calculated (2), the additional dose due to the released uranium can also be calculated.

7.4 Exposure Term

One curie of natural uranium is equivalent to 1.48×10^6 grams. The land area within 80 km of a plant is 2.01×10^{14} cm^2, and of the eastern half of the United States, 3.88×10^{16} cm^2. Assuming uniform mixing within the top 15 cm of soil (2) and a soil density of 1.5 g/cm^3, a one curie release would result in soil concentrations above background of 6.54×10^{-5} and 1.35×10^{-6} μg/g for the local and eastern U.S. land areas, respectively.

Table A-16 gives the background dose to critical organs from naturally occurring uranium and the additional calculated doses due to deposition of a one curie release from a model plant. The additional doses were determined by multiplying the doses from natural uranium by the ratios $(6.54 \times 10^{-5}$ μg/g)/(2.8 μg/g)

Table A-16

Doses to critical organs from naturally occurring uranium and an air release of one curie of uranium from a model plant

Critical organ	Background dose (mrad/yr)	Additional dose from a 1 Ci release	
		Local (mrad/yr)	Eastern United States (mrad/yr)
Soft tissue	0.03	7.0×10^{-7}	1.4×10^{-8}
Gonads	0.03	7.0×10^{-7}	1.4×10^{-8}
Endosteal bone	3	7.0×10^{-6}	1.4×10^{-7}
Bone marrow	0.06	1.4×10^{-6}	2.9×10^{-8}

and $(1.35 \times 10^{-6}$ μg/g)/(2.8 μg/g). To convert each critical organ dose to a dose equivalent (mrads to mrem) it is necessary to multiply the doses by a quality factor. For alpha emitters, the quality factor is defined as 10.

Estimates of the number of health effects expected for a given population dose are based on data presented in the BEIR report (3). Health effects will be defined as the summation of the number of lethal cancers, nonlethal cancers, and, if applicable, genetic effects. The health effects conversion factors for the critical organs are:

Soft tissues	300 health effects per 10^6 person-rem
Gonads	150 health effects per 10^6 person-rem
Endosteal bone	32 health effects per 10^6 person-rem
Bone marrow	54 health effects per 10^6 person-rem

The probability that an average individual in the population will incur a health effect due to a release of uranium is calculated by multiplying the health effects conversion factors by the respective critical organ dose equivalents and summing. This summation results in a single conversion factor relating the anticipated probability of a health effect occurring following the release of one curie of uranium from a model facility. The conversion factors for local and eastern U.S. populations are 5.5×10^{-12} and 1.2×10^{-13} health effects per person per curie released, respectively.

7.5 Population Term

The population of the United States is projected to grow

linearly from 205 million in 1970 to 300 million in 2020. Beyond 2020 it is assumed to remain constant at 300 million persons. The population within 80 km of a model facility is assumed to be 0.7% of the total U.S. population and is assumed to grow as the U.S. population grows. An exception is the population around a model mill. It is assumed to remain constant at 54,000 people. The population of the eastern half of the U.S. is assumed to be 80% of the total U.S. population. It will also grow as the U.S. population grows.

7.6 Evaluation of the 100-Year Dose Commitment

Table A-17 gives the health effects expected to result from the 100-year dose commitment for 30 years of facility operations. For comparison, the health effects expected to result from the immediate dose commitment for 30 years of facility operations is included.

For all types of fuel supply facilities, the 100-year dose commitment causes less than 10% additional health effects when compared to the immediate dose commitment. These calculations refer to uranium discharges only.

Table A-17

Health effects resulting from the 100-year dose commitment from uranium

- 30 years of facility operations -

Model facility	Airborne pathway source term (Ci/yr per facility)	Health effects from 100-yr dose commitment (effects per facility-30 yr)	Health effects from immediate dose commitment (effects per facility-30 yr)
Mill	0.1	0.006 (10%)[a]	0.05[a]
Conversion (wet solvent)	0.023	0.002 (0.5%)[a]	0.4
(Hydrofluor)	0.057	0.005 (3%)[a]	0.2
Enrichment	0.045	0.004 (4%)[a]	0.1
Fabrication	0.005	0.0005 (0.5%)[a]	0.1

[a]Percent of total health effects.

[b]Health effects from uranium immediate dose commitment.

Appendix B

Costs of Control Technology

Capital and operating costs of control technology were obtained from literature sources where available. If not available, an estimate was made based on the judgment of the factors involved in operation of the control systems for a different purpose. An example would be waste settling ponds. Estimates of the annualized costs (combined yearly capital and operating costs for accounting purposes) were made by multiplying the capital cost of a control system by an annual fixed charge rate. The fixed charge rate includes depreciation, interest, taxes, and insurance for a sinking method of depreciation based on a 30-year plant life (1). The fixed charge rate used was 16.6% per year (2). The product of the annual fixed charge rate and the capital cost yields the annualized capital cost which is then added to the annual operating costs. This sum is an estimate of the total annualized costs for the control system. The present worth of a treatment system was calculated from the annualized cost (equivalent to the present worth of the annualized costs using a 7.5% interest rate). Present worth is any future payment or series of payments that will repay a present sum with interest at a given rate. The present worth factor (pwf) for 7.5% for 30 years is tabulated as 11.81 (1). The present worth cost of the system is the equivalent to 11.81 times the total annualized costs.

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