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Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors

Health, Safety and Environment

U.S. Nuclear Regulatory Commission

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Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors

HEALTH, SAFETY & ENVIRONMENT

Chapter IV - Environmental Impact Due to the Implementation of Plutonium Recycle

August 1976

Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission

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CHAPTER IV

ENVIRONMENTAL IMPACT DUE TO THE IMPLEMENTATION OF PLUTONIUM RECYCLE

This Chapter provides an assessment of the health, safety and environmental effects of the entire light water reactor fuel cycle, considering the comparative effects of three major alternatives: no recycle, recycle of uranium only, and recycle of both uranium and plutonium. The assessment covers the period from 1975 through the year 2000 and includes the cumulative effects for the entire period as well as projections for specific years.

Chapter IV is divided into individual sections as follows:

Section A - Summary
Section B - Introduction
Section C - The Light Water Reactor With Plutonium Recycle
Section D - Mixed Oxide Fuel Fabrication
Section E - Reprocessing Plant Operations
Section F - Supporting Uranium Fuel Cycle
Section G - Transportation of Radioactive Materials
Section H - Radioactive Waste Management
Section I - Storage of Plutonium
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Section K - Extended Spent Fuel Storage
Section L - Blending of Plutonium and Uranium at Reprocessing Plants

CHAPTER IV

ENVIRONMENTAL IMPACT DUE TO THE IMPLEMENTATION OF PLUTONIUM RECYCLE

SECTION A

SUMMARY

CHAPTER IV

Section A

SUMMARY

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Section A

SUMMARY

1.0 INTRODUCTION

The total LWR fuel cycle is analyzed with and without recycle to determine the environmental impact of plutonium recycling. The impact attributable to recycling plutonium is determined by estimating the differences between recycling plutonium (and uranium), recycling only uranium from spent LWR fuel, and recycling no fuel at all.

To facilitate the environmental assessment of each step in the fuel cycle, model plants have been developed for this analysis. The models are chosen to represent suitably plants that reflect industry practice. See Table IV A-1 for a summary of fuel cycle model plants. In some steps more than one model is developed to represent technological alternatives that are viable in those steps of the fuel cycle. Each of the steps in the nuclear fuel cycle is evaluated in terms of the impact potential of model plants.

2.0 EFFECT OF RECYCLING ON THE FUEL CYCLE INDUSTRY

Recycling plutonium introduces two additional fuel cycle steps, diminishes the magnitude of material processed in some steps, and alters the isotopic composition of material handled in some steps.

2.1 New Steps Involved In Recycling

2.1.1 Reprocessing

If salvable fissile materials are to be recycled, irradiated LWR fuel is subjected to a reprocessing step that chemically separates the plutonium or uranium or both from the fission products and transuranic wastes. It is estimated that there would be five model reprocessing plants operating by the year 2000 if uranium only or both uranium and plutonium were to be recycled.

Based on the reprocessing plant model, the bulk of the radioactivity in the irradiated fuel is estimated to be retained in the wastes with the exception of some of the gaseous fission products. Gaseous krypton, tritium, carbon-14, and a small amount of the iodine are assumed to be released to the atmosphere. These releases represent the dominant radiological effluents from the reprocessing plants and from the fuel cycle that includes recycling. The releases are from waste treatment systems that are the best proven technology applicable to commercial scale facilities. It should be noted that although the spectrum of radionuclides is somewhat different in

Table IV A-1

FUEL CYCLE MODEL PLANT SUMMARY

Fuel Cycle Step	Model Plant Characteristic*	 Annual Capacity
Mining	Open Pit	200 ST U_3O_8
	Underground	20 ST U_3O_8
Milling	Carbonate/Acid Leach	1,050 ST U_3O_8
UF_6 Conversion	Wet	15,000 MTU
	Dry	15,000 MTU
Enrichment	Gaseous Diffusion	8.75 MTSWU
	Centrifuge	8.75 MTSWU
UO_2 Fuel Fabrication	ADU	1,500 MTU
LWR	PWR	0.8 GWy
	BWR	0.8 GWy
Storage	Spent Fuel Pool	3,500 MTHM
Reprocessing	Purex	2,000 MTHM
MOX Fuel Fabrication	Mechanical Blending	360 MTHM
Transportation	Vechicle Miles	Proportioned to respective plant feeds
Waste Management	Geologic Storage Spent Fuel HLW	15,000 elements 360 m^3
	Surface Burial (other)	28,300 m^3

---

*The chemical-physical process used or type of facility.

irradiated MOX fuel from that in irradiated UO_2 fuel, the result is that the environmental dose commitment from the reprocessing step when plutonium and uranium are recycled is about the same (97%) as that for recycling only uranium.

The U.S. population environmental total body dose commitment from the reprocessing step projected over the 1975-2000 period for recycling uranium only is 1.08×10^6 person-rem or about 0.17% of the natural background radiation of the U.S. population. No recycle would avoid those environmental impacts associated with reprocessing facilities.

2.1.2 Mixed Oxide Fuel Fabrication

An added step in the fuel cycle that is required for the implementation of plutonium recycle is the fabrication of mixed oxide fuel from uranium and recovered plutonium. It is estimated that eight model plants employing 2,400 persons would be needed to accomplish this by the year 2000.

Although the principal radionuclides released to the environment from this fuel cycle step are plutonium isotopes, relatively little radioactivity is released. Consequently the U.S. population environmental dose commitment from the MOX fabrication step is several orders of magnitude less than that of the reprocessing step. For the period 1975 to 2000, the integrated dose commitment (total body) from MOX fabrication for the U.S. population would be about 300 person-rem. Data in Tables IV J-3 through IV J-9, Section J of CHAPTER IV show that the dose commitments from MOX fabrication for other organs are similarly very small in comparison to reprocessing.

2.2 Reduction Of Uranium Needs With Recycling

The recovery of fissile values from spent fuel will reduce the uranium needs of the LWR fuel cycle; accordingly this effect will result in a reduction in the number of model plants needed to supply uranium.

2.2.1 Plants Not Needed With Recycling

Based on the nuclear power growth projected during the period 1975 through 2000, the number of model plants needed in the uranium supply steps of the fuel cycle by the year 2000 have been estimated. Table IV A-2 indicates the number of model plants based on no recycling of fissile values in LWR spent fuel and shows the reduction in model plants achievable if uranium is recycled or if both uranium and plutonium are recycled.

2.2.2 Environmental Effects of Reduced Uranium Needs

The reduction in uranium supply facilities afforded by recycling spent LWR fuel releases more than eight times as many workers for other productive pursuits as would be taken out of the labor market to operate the reprocessing and MOX fabrication facilities to implement recycle. By year 2000, 19,600 fewer persons (predominantly miners) would be needed in the uranium supply steps, if uranium only were recycled;

Table IV A-2

NUMBER OF MODEL PLANTS IN THE URANIUM SUPPLY STEPS OF THE LWR FUEL CYCLE BY 2000

Uranium Supply Step	Total If No Recycle	Reduction If U Only Recycled	Reduction If U and Pu Recycled
Open Pit Mines	240	33	71
Underground Mines	5,600	745	1,645
Uranium Mills	109	14	32
UF_6 Conversion Plants	7	1	2
Enrichment Facilities	6	0	1
UO_2 Fuel Fabrication	9	0	1

47,400 fewer if both Pu and U were recycled. The occupational total body dose commitment would decrease 2.3 x 10^5 with most of the reduction due to decreased uranium mining.

Reduction in the U.S. population dose commitment over the 1975-2000 period from decreased uranium supply steps would only partially compensate for that added by the fuel reprocessing step. As the result of effluents from fewer plants in the uranium supply steps of the fuel cycle, there would be a reduction of 3.9 x 10^5 person-rem (total body) based on uranium only recycling and a reduction of 8.1 x 10^5 person-rem (total body) based on recycling both U and Pu, whereas reprocessing introduces 1.1 x 10^6 person-rem (total body) for both options.

From these steps, 50,000 acres less land would be disturbed with uranium only recycling and 100,000 acres less with recycling of both Pu and U. The reduction in acre-years of unavailability of land for other uses would be 3 x 10^6 and 6 x 10^6, respectively, during the period 1975 to 2000.

The changes in these and other environmental factors are tabulated in Appendix A of Section F, CHAPTER IV.

3.0 RADIOLOGICAL CONSEQUENCES OF PERMITTING PLUTONIUM RECYCLING

Radiological effects have been principally assessed with respect to humans, on the presumption that other biota will not be injured if human exposure is maintained below promulgated standards. Exposures to radionuclides by four principal pathways: submersion, inhalation (including resuspension of deposited particulates), dietary intake, and irradiation from deposited material in the environs, have been taken into account. Appendix A of Section J, CHAPTER IV, explains the methodology used in estimating population doses to various organs from the amounts of radioactivity discharged to the environs by the respective model plants.

The quantities of radioactivity estimated to be discharged from the model plants are within the limitations of present standards; however, they are generally estimated on the high side and should be greater than the average expected for the industry.

Assessments of the following three principal types of exposure have been performed: (1) The population dose commitment averaged over a U.S. population of 250,000,000; (2) the dose commitment of a hypothetical nearest resident to each model plant; and (3) the dose commitment of workers in the respective fuel cycle steps.

3.1 Radiological Impact On Population

Population dose commitments resulting from effluents from each model plant have been computed for the whole U.S. population, not just residents within a limited radius. Some of these are summarized in Table IV A-3.

Occupational dose commitment for the whole fuel cycle is of the same order of magnitude as the population dose commitment. The total dose commitment (population plus occupational) and risk to the U.S. population are summarized for the fuel recycle options in Table IV A-3.

Table IV A-3

TOTAL DOSE COMMITMENT AND CANCER MORTALITY RISK OF THE U.S. POPULATION RESULTING FROM LWR RECYCLE OPTIONS FOR THE PERIOD 1975 THROUGH 2000

	Total If No Recycle	Change	
		If U Only Recycled	If U and Pu Recycled
Population (Million Person-Rem)			
Total Body	3.9	+ 0.7	+ 0.3
Bone	13.0	+ 1.0	0
Lung	1.4	+ 1.0	+ 0.9
Occupational (Million Person-Rem)			
Total Body	4.1	- 0.1	- 0.2
Bone	6.4	- 0.3	- 0.7
Lung	16.0	- 2.0	- 3.0
Total for U.S. Population			
Total Body (Million Person-Rem)	8.0	+ 0.60	0
Total Cancer Mortality Risk	1,080.0	+ 80.0	+ 20.0

For perspective, these values should be compared to the nominal background radiation dose commitment of 6.5×10^8 person-rem that would accrue to the U.S. population in 26 years and to the 3×10^7 cancer mortality projected in this population during its lifetime.

The calculated cancer mortality risk changes of 170 and 100 for the world population, with U recycle only and U and Pu recycle, respectively, are so much smaller than the uncertainty in the estimated cancers from natural radiation that detection of a change in the normal cancer incidence of a population of 4,000,000,000 could not be observed epidemiologically. See Table IV A-4.

Table IV A-4

COMPARISON OF CANCER MORTALITY EXPECTANCY
1975 THROUGH 2000

Expected Total Cancer Incidence in U.S. Population	10,000,000
Estimated Cancers From Natural Background	90,000
Estimated Uncertainty in Estimate of Cancers From Natural Background	± 9,000
Estimated Cancers in U.S. Population From No Recycle*	1,080
Estimated Cancers in U.S. Population From Uranium and Plutonium Recycle*	1,100

---

*Consideration of the world population would increase these estimates to 1,110 and 1,210 respectively.

The linear response theory is generally considered to provide an upper limit to the biological consequences of radiation exposure. Given this fact and the small magnitude of the doses involved here, the increment in cancer risk for fuel cycle alternatives is small--but not detectable--when compared to the normal risks of cancer.

3.2 Radiological Impact On Nearest Neighbors To Fuel Cycle Plants

Annual dose commitments have been computed for hypothetical individuals hypothetically residing near the respective fuel cycle plants. They are adults who live continuously in the vicinity of such plants and eat normal diets derived from food produced at the residence.

These persons would be subject to the maximum differences that could result from changes in the proportions of effluent releases of radionuclides in fuel cycle materials based on the respective fuel cycle options.

For the enrichment and transportation steps in the fuel cycle, which contribute insignificant annual doses of about 0.001 and 0.00005 rem, respectively, the increase in the nearest neighbor total body dose is approximately 40% more than the no recycling option for recycling either uranium or U and Pu. Increases in dose commitments from recycling to neighbors of UO_2 fuel fabrication plants, LWR's and irradiated fuel storage facilities are less than 4%. For these operations, the theoretical nearest neighbor doses are an order of magnitude or more below the unrestricted area limit of 10 CFR Part 20.

For uranium milling, the dose to the nearest neighbor is not projected to change with the implementation of plutonium recycle, but the number of neighbors in this range of exposure will be decreased because recycling decreases the required number of mills. The number of households adjacent to these plants is likely to be low based on the sparsely populated nature of the geographical locations where milling is expected to take place. Thus in this study no benefit is claimed for the reduction in the number of these nearest neighbors who might be exposed if plutonium is recycled.

Further, the number of neighbors living adjacent to the eight MOX fuel fabrication plants for 70 years is unlikely to exceed a few hundred to a few thousand. The risks to nearest neighbors for the several fuel recycling options are too small to be detectable and to provide a clearly defined basis for making a selection of a fuel recycle option purely on the basis of radiological exposure of neighbors living adjacent to fuel cycle plants.

3.3 Radiological Impact On Fuel Cycle Facility Workers

Except for transportation, MOX fuel fabrication, and irradiated fuel reprocessing, occupational exposures diminish with the implementation of recycling. The projected aggregate of occupational exposures for the LWR and its fuel cycle industry at the growth projected from 1975 through 2000 is summarized in Table IV A-3.

Occupational dose commitments are included in Table IV A-3 and the risk to the overall U.S. population including workers discussed in paragraph 3.1. It should be noted here that the decrease in occupational exposure summarized in Table IV A-3 for recycling does not appear to fully offset the increase in environmental dose commitment noted, but the resultant increase shown in Table IV A-3 is from one to two orders of magnitude less than the dose commitment from the no recycling case. Thus neither recycling option is judged to have dose commitments significantly different from those of the no recycling option.

3.4 Radiological Impact Of Accidents

The radiological consequences of accidents of credible magnitudes have been estimated for the respective model plants in the fuel cycle. With the exception of the loss of coolant accident for the GESMO model reactor, the nearest neighbor dose commitments for any accident are predicted to be less than the 10 CFR Part 20 limit for a year's exposure to an individual in an "unrestricted area." Because the frequency of serious radiological accidents in the industry is expected to be far less than one per year for the whole industry, it is considered that the conservative estimates (over-assessment of releases and effects) used to account for normal releases from the model fuel cycle plants have sufficient margin to include the effects of accidental releases over the period of the study.

In the uranium supply steps of the fuel cycle, the per plant impact of an accident would not be significantly different with the implementation of fuel recycling, but there would be fewer model plants, as indicated in Table IV A-2. Therefore, the potential for accidents would be correspondingly decreased.

The additional steps required by recycling, reprocessing of irradiated fuel and the fabrication of MOX fuel, have comparably low radiological impact per accident, as indicated in Table IV A-5, and also low accident occurrence expectancy.

Table IV A-5

ESTIMATED RADIOLOGICAL DOSE COMMITMENTS FROM MODEL PLANT ACCIDENTS

Accident Characterization	Reprocessing UO_2 Fuel	Reprocessing MOX Fuel	Fabrication MOX Fuel
	Rem to Nearest Neighbor		
Criticality	0.056	0.056	0.360
Fire	0.002	0.014	0.027
Explosion	0.011	0.019	0.027
	Person-Rem To Public		
Criticality	629	629	4.2
Fire	18	152	0.8
Explosion	123	213	0.8

In the case of transportation, there would be more transportation mileage to circulate fuel through the recycle steps, and this might increase accident frequency proportionately. However, the radiological consequences of transportation accidents are estimated to be inconsequential.

With plutonium recycling, LWR's could have up to three or four times as much Pu inventory in the core than with UO_2 fueling. However, LWR accident evaluations indicate that the Pu remains nonvolatile. In determining the accident dose resulting from radionuclides escaping the facility, volatiles always predominate. With MOX recycling, the accident radioiodine dose would increase slightly and the dose from other volatiles would decrease somewhat. Thus it is concluded that recycling plutonium in LWR fuel does not significantly affect the consequences of reactor accidents.

4.0 NONRADIOLOGICAL EFFECTS OF PERMITTING PLUTONIUM RECYCLE

The nonradiological impacts of model plants on the environment are considered in terms of pollutants in effluents and in terms of resources consumed.

4.1 Chemical Pollutants

Recycling plutonium would add eight model MOX fuel fabrication plants to the fuel cycle by 2000. Like all commercial plants in all the fuel cycle steps, these new plants would be required to meet individual licensing requirements and MOX fuel fabrication plants would have to be designed and required to operate so that effluent pollutants would not exceed standards for safe concentrations in the immediate environment.

4.1.1 MOX Fuel Fabrication Plant Effluents

Hydrogen, helium, nitrogen, argon, oxides of nitrogen, hydrogen fluoride, and ammonia will be released to the atmosphere by the model MOX fuel fabrication plants.

Treatment systems would reduce the noxious components to the concentrations indicated in Table IV A-6 at a radius of 500 meters (assumed distance to the exclusion area boundary from the plant stack).

Table IV A-6
ANNUAL AVERAGE POLLUTANT CONCENTRATIONS AT
500 METERS FROM MODEL MOX FABRICATION
PLANT, COMPARED TO AIR QUALITY STANDARDS

Compound	Average Concentration	Limiting Standard
Hydrogen Fluoride	6. x 10^{-5} µg/cu m	5 x 10^{-1} µg/cu m
Oxides of Nitrogen	7. x 10^{-2} µg/cu m	1 x 10^{+2} µg/cu m
Ammonia	1. x 10^{-2} µg/cu m	*

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*No primary or secondary environmental air quality standard has been established. The occupational threshold limit value for ammonia, published by ACGIH, is 18,000 µg/cu m. This is approximately 5 times the olefactory threshold.

From the comparison in this table it is judged that no health impact on the environs would result from these effluents.

The MOX fabrication plant would also have wastes of less than 0.5 lb/day phosphate and less than 10 lb/day nitrate in the 28,000 gal/day liquid effluent. This dilutes the discharges to levels that approximate drinking water concentrations before the further dilution is accomplished by the receiving stream.

4.1.2 Effluents From Other Fuel Cycle Facilities

Since the throughputs of most other individual model plants would be the same, whether plutonium recycling were implemented or not, there would be no difference in the concentrations in the local environs of chemicals released with effluents during normal operation. These would, as a matter of course, have to conform with regulations imposed to protect the environments and would be engineered on a case-by-case basis to be in accordance with specific site circumstances.

Irradiated fuel reprocessing would have essentially the same chemical effluents, whether plutonium were recycled or just uranium, but the model reprocessing plants would not exist if there were no recycling.

4.2 Resources Used

Some resource uses are irreversible; others, like water, are transient and provide impact in the form of competition with other local uses. Table IV A-7 indicates the magnitude of major resource uses for the operation of LWR's and their supporting fuel cycle projected for the period 1975-2000, under the three basic fuel management modes. The total requirements are listed for the no recycling option, and parallel columns

Table IV A-7

RESOURCE USE TOTAL, FOR LWR FUEL CYCLE IN OPERATION 1975 THROUGH 2000

Resource	Total If No Recycling	Change From No Recycle Total	
		If U Only Recycled	If Pu and U Recycled
Acre-years Occupied	2.8×10^7	-2.4×10^6	-5.3×10^6
Disturbed Acres	4.1×10^5	-4.2×10^4	-1.0×10^5
Permanently Committed Acres	4.5×10^4	-5.0×10^3	-1.1×10^4
Cubic Meters for Mill Tailings	7.8×10^8	-9.0×10^7	-1.9×10^8
Cubic Meters for Other Radioactive Waste	4.0×10^6	-1.0×10^5	-1.0×10^5
Cubic meters for Chemical Waste	3.2×10^5	$+1.0 \times 10^5$	-2.0×10^4
GWy Electricity	380	+0.6	-2.1
Gallons Fuel Oil (Transportation)	3.5×10^7	$+4.0 \times 10^6$	$+8.0 \times 10^6$
Therms Gas	1.3×10^{10}	-1.4×10^9	-3.0×10^9
Tons Coal	3.7×10^6	$+4.0 \times 10^4$	-5.4×10^5
Tons Natural Uranium	1.2×10^6	-1.3×10^5	-2.9×10^5
Gallons Water	1.3×10^{14}	-1.0×10^{11}	-1.0×10^{13}

indicate the change, + or -, for uranium only recycling and the recycling both plutonium and uranium.

Transient land use is expressed in acre-year units, which is the use of one acre for one year. Permanent land use is expressed in acres. Mining and milling dominate the transient land use. However, the total transient land use of 28 million acre-years represents use of about 0.05% of the land area of the United States during the 26-year period of this study. LWR operation, which will not vary with the fuel management mode used, dominates the volume of radioactive waste generated, the use of electricity (for operating auxiliaries), and the use of water. The second largest volume of radioactive waste results from fuel reprocessing (4% as much volume as from LWR's). The second largest consumption of electrical energy is for enrichment (54% as much as used for operating reactor auxiliaries). The second largest use of water is for cooling at enrichment facilities (17% as much as used by LWR's). These percentages pertain to recycling plutonium and uranium. Transportation in all cases dominates the use of fuel oil, but as noted in CHAPTER IV, Section G, is a very small component of the overall trucking industry.

The impact and value of these uses of resources are discussed in CHAPTERS VI, IX, X, and XI.

5.0 CONCLUSIONS

Recycling plutonium as fuel in LWR's would shift some of the investment of capital and labor from the supplying of raw material for nuclear fuel to the salvaging and reworking of used fuel fissile values. This would probably result in a somewhat reduced need for additional workers in the future. Such a change would result in some reduction in occupational dose to the total work force.

For other segments of the population, the risk of radiation-caused cancer mortality and birth defects from the whole fuel cycle integrated through 2000 is too small to have any epidemiologically detectable impact. Thus the very small theoretical increase in radiological dose commitment to the population if plutonium and uranium are recycled and the somewhat larger increase with the recycle of uranium only over the no recycling option are not judged to provide a real basis for determining that there is a difference in radiological impact resulting from any of the fuel management modes.

Plutonium recycling would conserve most resources involved, although small increases in the volumes of radioactive waste to be stored and the use of fuel oil for transportation would be incurred. These impacts would be offset to some degree by reductions in mill tailings volumes and reduced transportation needs in the uranium supply chain, which for the purpose of this study were not analyzed in detail due to their tenuous link with recycling operations and as a measure of conservatism in expressing recycling benefits.

GESMO

CHAPTER IV

ENVIRONMENTAL IMPACT DUE TO THE IMPLEMENTATION OF PLUTONIUM RECYCLE

SECTION B

INTRODUCTION

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INTRODUCTION

1.0 ENVIRONMENTAL IMPACT OF THE LIGHT WATER REACTOR FUEL CYCLE

CHAPTER IV assesses the prospective environmental impact produced by each segment of the light water reactor (LWR) nuclear fuel cycle, including reactors, and by the industry as a whole, for the following three fuel management options:

- No recycle of uranium or plutonium
- Recycle of uranium only
- Recycle of both uranium and plutonium

This is a generic assessment for each segment of the LWR fuel cycle industry. It is not a substitute for the detailed environmental impact statement for any LWR nuclear facility for which such a statement is required pursuant to 10 CFR Part 51, nor does it replace independent evaluation by NRC of the environmental impact of specific facilities where such an evaluation is required. Rather, this assessment considers the prospective impact on the environment of the recycle of uranium and plutonium as MOX fuel in LWR's, which are currently largely fueled by UO_2 fuels that do not contain recycled uranium or plutonium. CHAPTER IV evaluates the environmental impacts that may result from changes in fuel composition and from the reduction in the amounts of natural resources required if either uranium only or uranium and plutonium are recycled, as compared to no recycle of uranium or plutonium. Environmental impact assessments developed in CHAPTER IV are part of the bases for the assessment of alternatives in CHAPTER VIII and for the cost-benefit analyses in CHAPTER XI.

The growth of the LWR industry was estimated by NRC after considering several different projections of U.S. total nuclear electricity-generating capacity of government and private organizations by modifying two of these projections. The modifications consisted of decreasing the estimate of nuclear generating capacity to reflect the recent withdrawal of commercial high temperature gas cooled reactors from the market and to reflect the assumption that there will be no fast breeder reactors (FBR's). The assumption of no FBR's does not reflect an NRC forecast but rather indicates the view that NRC can make the GESMO decision without the necessity of assuming the existence of the breeder. With no FBR's, virtually all plutonium recovered from LWR spent fuel would be recycled in LWR's. From the list of projections considered, NRC selected two cases, the ERDA moderate high growth with breeder case and the ERDA low growth without breeder case as representing a reasonable range for the estimated growth of U.S. total nuclear electricity-generating capacity for the remainder of the century.

The ERDA moderate high growth with breeder case and the low growth without breeder case project an installed nuclear capacity of 197 and 156 GWe, respectively, in the year 1985; and 893 and 507 GWe, respectively, in the year 2000.

To assess environmental impacts, NRC has chosen to base industry size on the low growth without breeder case because some reactors have been canceled and others have been deferred since the draft GESMO was issued in 1974. It is assumed that all plutonium generated in LWR's will be recycled in LWR's through the year 2000. This removes the perturbation due to FBR's from the analyses, since the FBR's would share, and thus reduce, the environmental impacts related to Pu recycle in LWR's.

CHAPTER III discusses the prospective growth of the LWR fuel cycle industry with respect to the three fuel cycle options under consideration. The three fuel cycle options would result in different environmental impacts.

If spent fuel is disposed of as high level radioactive waste, without recycle of uranium or plutonium, the environmental impacts of the supporting uranium cycle would be at a maximum. Offsetting this would be eliminating the transportation of spent fuel to reprocessing plants, reprocessing, converting recovered uranium to uranium hexafluoride (UF_6), converting plutonium nitrate [$Pu(NO_3)_4$] to plutonium dioxide (PuO_2), fabricating MOX fuel, storing and shipping plutonium products, storing high and intermediate level radioactive waste solutions and subsequently converting them to solid forms for transfer to a Federal repository. Spent fuel elements, after an interim storage period to allow the short-lived fission products to decay, would be shipped to a Federal repository for long term confinement.

If spent fuel is reprocessed to recover and recycle uranium only, spent fuels will be transported to a fuel reprocessing plant for recovery of the usable uranium. Recovered uranium will be converted to UF_6 for reenrichment in uranium enrichment plants, then converted to UO_2 and fabricated into UO_2 fuel elements in the same manner as natural uranium, thereby conserving uranium resources and reducing environmental impacts of mining, milling, and conversion of natural uranium to UF_6. In comparison with the no recycle option, environmental impacts associated with the reprocessing of spent fuel, storage of high and intermediate level radioactive wastes (including plutonium), and subsequent conversion of radioactive liquid wastes to solid forms for transport to a Federal repository are added. There will be no requirement for the fabrication of MOX fuels, nor for the storage and shipment of purified plutonium products. The plutonium will be a transuranic waste, requiring long term confinement in a Federal repository.

If spent fuel is reprocessed to recover and recycle both uranium and plutonium, the fuel cycle operations will be similar to those for uranium recycle, but will also include the purification of plutonium, conversion to PuO_2, fabrication of MOX fuel, and transport of plutonium products for recycle of plutonium as fuel in LWR's. This option conserves additional uranium resources and further reduces the environmental impacts of mining, milling, conversion of natural uranium to UF_6, enrichment, and UO_2

fuel fabrication. However, in addition to the effects attributed to reprocessing, it adds environmental impacts associated with the fabrication of MOX fuel.

2.0 ASSESSMENT OF ENVIRONMENTAL IMPACTS RELATED TO URANIUM AND PLUTONIUM RECYCLE

The forecast growth of the LWR fuel cycle and the definition of the MOX fuel loading in the GESMO model LWR are the primary assumptions in assessing the environmental impact attributable to plutonium recycle. The amount of plutonium available for recycle as fuel in LWR's will increase with the growth of the LWR power industry and the availability of fuel reprocessing capacity when needed.

2.1 The Effect of Plutonium Recycle With Respect to Accidents in Fuel Cycle Facilities

The Nuclear Regulatory Commission requires that the primary assurance of safety be attained through a high degree of reliability and predictability in the design, construction, and operation of a nuclear facility, and through extensive quality assurance actions. In addition, engineered safety systems must be provided to prevent or to mitigate the consequences of accidents.

Each applicant for a construction permit or operating license for a nuclear power plant or a fuel reprocessing plant must provide an analysis and evaluation of the design and performance of the structures, systems, and components of the facility so that the risk to public health and safety resulting from operation of the facility can be assessed. These analyses include a determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and mitigation of the consequences of accidents.

The history of the nuclear fuel cycle industry to date indicates that accidents in fuel cycle facilities have not produced effects on the offsite environment having any significant consequence. In the nuclear fuel cycle industry, safety is the controlling factor. Each applicant, in the license application, is required to analyze potential conditions that could result in the release of radioactivity beyond the plant confines. In addition, the applicant must design, construct and operate the facility in a manner that will minimize such releases. As a further assurance, the NRC staff performs a detailed independent safety evaluation of each fuel cycle facility.

The review includes contamination control considerations, ventilation system performance, exposure assessment for workers, and control of such exposure. Effluents and effluent treatment and monitoring systems for both liquid and gaseous streams leaving the plant are evaluated. Accident analyses that include consideration of criticality, radiation safety, and nonradiation safety problems are performed.

In fuel cycle plants handling special nuclear material, both liquid and solid systems are evaluated for criticality safety. Relative to a uranium criticality accident, the slightly different fission product yield and the presence of plutonium do not significantly increase the effects of a criticality accident. With respect to other types of potential accidents, such as fires or explosions, plutonium recycle introduces plutonium into some plant streams and increases the concentrations of plutonium in certain other plant streams and changes the isotopic composition of the plutonium. This results in a slight increase in the calculated dose commitment to persons who may be exposed to radioactive effluents as a result of such accidents.

WASH-1250, "The Safety of Nuclear Power Reactors (Light Water Cooled) and Related Facilities," dated July 1973, is a comprehensive statement regarding the safety and environmental aspects of nuclear power reactors and their support facilities. Reference is made to this document in CHAPTER IV, Section C, with regard to the introduction of recycle plutonium in LWR's.

A series of potential accidents that could produce environmental effects ranging from trivial to serious are postulated and evaluated in the safety analysis before the individual facility license is issued. Records to date indicate that few, if any, of the accidents involving special nuclear material have had any measurable effects on the environment. Thus the review of postulated accidents for this assessment has concentrated on the more serious accidents of the type that either have occurred or can be realistically postulated.

2.2 Environmental Impact Assessment

The LWR fuel cycle defined for this generic environmental assessment is based on a forecasted future industry since some of the facilities making up the various segments of the prospective industry are not yet designed. Accordingly, the environmental impact assessment is based on model facilities representing typical facilities for each segment of the projected fuel cycle. These model facilities reflect current technology, and in general the environmental impact assessment is based on conservative estimates of effluent releases and other factors.

In this chapter, an assessment of the environmental impact of each segment of the fuel cycle industry has been projected to the year 2000 and integrated from the year 1975 through 2000. The total impact of the three fuel cycle options is compared: no recycle of uranium or plutonium, recycle of uranium only, and recycle of uranium and plutonium as fuel in LWR's. These assessments are based on model facilities representing the elements of the fuel cycle industry; they do not necessarily represent a specific facility, but rather an aggregate of typical facilities in each segment. In this generic assessment, a model facility is described briefly and assessed as a whole, rather than by the individual operations that make up the whole as in environmental statements for specific facilities. The environmental impacts attributed to a model facility may be somewhat overstated with respect to newer plants and future plants. This was done to assure that the assessment encompassed an aggregate of typical facilities for each segment of the fuel cycle.

The assessment does not include anticipated improvements in technology or procedures that may be reasonably expected to reduce environmental impacts as the industry grows from the year 1975 through 2000.

3.0 THE FUEL CYCLE MODEL

The environmental impact from the LWR industry for the three fuel cycle options has been derived from assessments of model facilities representing each segment of the fuel cycle, and transportation and projected waste management operations in the aggregate. This section presents a brief description of the fuel cycle models used.

3.1 Light Water Reactor With Plutonium Recycle

The model reactor is a 1,000 MWe LWR, loaded solely with low enriched UO_2 fuel or with both UO_2 fuel and MOX fuel that contain 1.8% plutonium (1.15 SGR) in the charged heavy metals. The reactor is one of a pair located at a single station. The environmental impact from the normal level of radioactivity released from two reactors, both operating on either UO_2 or the GESMO model MOX loading, is based on three types of reactors: BWR, PWR with U-tube steam generators, and PWR with once-through steam generators.

The concept of expressing the reactor plutonium loading as a multiple of the self-generation reactor (SGR) level is used here. A reactor is an equilibrium SGR when the plutonium recovered from the spent fuel assemblies, which are removed from the core after producing power for a specified period, is approximately equal to the amounts of plutonium in the fuel assemblies when initially placed in the core before irradiation. Although there are fuel management concepts for the use of 100% MOX fuel, and others that use an amount of MOX fuel that is less than the SGR level, the GESMO reactor uses a MOX loading that corresponds to 115% of an SGR loading. At this level, as many as 40% of the fuel rods may contain MOX; the remaining fuel rods contain slightly enriched UO_2. Quantitatively, the 1.15 SGR plutonium loading can be expressed as 1.8% of the weight of the total heavy metal charge. The MOX loading is significant for evaluating the environmental impact of the reactor--it does not affect other parts of the fuel cycle.

3.2 Mixed Oxide Fuel Fabrication

The model MOX fuel fabrication plant produces a nominal 360 MT of MOX fuel per year--enough MOX fuel to supply about 30 model LWR's each operating at a 1.15 SGR level. The plant is located on its own 1,000-acre site. It receives plutonium dioxide and uranium dioxide and manufactures MOX fuel rods, mechanically sizes and blends PuO_2 and UO_2, presses the blended oxides into MOX pellets, sinters the pellets and loads them into tubes, welds end-caps on the tubes to form fuel rods, and ships the MOX fuel rods to a UO_2 fuel fabrication plant for assembly into LWR fuel elements.

Since the MOX fuel fabrication operation does not occur in the LWR industry without Pu recycle, all environmental impacts from this operation are ascribable to Pu recycle.

3.3 Fuel Reprocessing

The model fuel reprocessing plant is a 2,000 MT/yr plant using the Purex solvent extraction process to separate fission products and other actinides from uranium and plutonium. Associated with fuel reprocessing are a separation facility, a UF_6 conversion facility, a PuO_2 conversion facility, and waste solidification and packaging facilities. The model assumes the total release of ^{14}C and the fission products ^3H and ^{85}Kr to the air. This assumption maximizes the impacts attributable to reprocessing.

If neither uranium nor plutonium is to be recycled as fuel in LWR's, the spent fuel becomes the high level radioactive waste of the nuclear industry--no spent fuel will be reprocessed. On the other hand, if either uranium or both uranium and plutonium are to be recycled as fuel in LWR's to conserve fuel resources, then the spent fuels must be reprocessed to recover the uranium or both uranium and plutonium. Most effects attributable to reprocessing would be incurred regardless of whether only uranium or both uranium and plutonium were recycled.

Spent mixed oxide fuels differ from spent UO_2 fuels with respect to the plutonium isotopes, fission products, and transuranium isotopes they contain. The change in environmental impact resulting from reprocessing the industry average mix (11% MOX and 89% UO_2) for the years 1975 through 2000, relative to that from reprocessing UO_2 fuel only, has been assessed and is presented in CHAPTER IV, Section E.

The environmental factors of the five recycle alternatives are tabulated in CHAPTER VIII, Appendix A. A comparative evaluation of all alternatives is given in CHAPTER VIII.

3.4 Supporting Uranium Fuel Cycle Facilities

The supporting uranium fuel cycle operations do not involve the processing of materials containing plutonium and will be carried out following essentially the same procedures as in the existing uranium cycle for LWR's, which includes:

- Uranium mining
- Uranium milling
- Conversion of the mill-produced yellowcake to uranium hexafluoride
- Enrichment of uranium in the fissionable isotope uranium-235
- Conversion of enriched uranium hexafluoride to uranium dioxide
- Fabrication of uranium fuel assemblies

The model for supporting uranium fuel cycle facilities assumes that, during the years 1975 through 2000, approximately 60% of future ore will come from underground mines and 40% from open pit mines. It is assumed that the ore, containing on the average about 0.1% U_3O_8, will be milled with a recovery of 90.5% in plants capable of processing about 3,500 MT of ore per day. The yellowcake will be converted to UF_6 in two existing facilities plus future facilities, each of which will add a capacity of about 15,000 MTU/yr. Enrichment will require expanded capacity of existing gaseous diffusion plants and additional new plants, either gas centrifuge or gaseous diffusion plants. The conversion of slightly enriched UF_6 to UO_2 and fabrication of low enrichment UO_2 fuel assemblies is assumed to take place in expanded existing plants and in new 1,500 MTU/yr fuel fabrication plants.

The recycle of uranium and plutonium would result in a reduction in supporting uranium fuel cycle requirements. Uranium requirements would be reduced by about 22%, relative to the requirements without recycle, if plutonium were recycled as fuel in LWR's. Environmental impacts of the supporting uranium fuel cycle segment also would be reduced if uranium and plutonium were recycled.

3.5 Radiological Assessment

Each section contains an assessment of the radiological effects, if any, for a segment of the nuclear fuel cycle industry. CHAPTER IV, Section J, contains summary tables for comparing the radiological effects from the three fuel cycle options for the total fuel cycle. Population and occupational dose commitment estimates for the total LWR fuel cycle industry over the years 1975 through 2000 are presented there. These are distributed pro rata, per gigawatt-year of electrical energy produced, for a 1,000 MWe LWR to permit ready comparison of the three fuel cycle options on a per reactor basis.

Section J also contains a description of dose commitment models used for this assessment and an evaluation of prospective health effects related to estimated incremental increases in population dose commitments from the LWR fuel cycle industry.

4.0 ENVIRONMENTAL EFFECTS OF A 1,000 MWe LWR FUEL CYCLE

NRC regulations require that environmental reports and statements for individual reactors shall include, in the cost-benefit analysis, the environmental effects of the fuel cycle and transportation of radioactive materials. The Summary and Conclusions volume presents a tabulation of environmental effects normalized to that portion ascribable to an annual reload of a 1,000 MWe nuclear power plant operating at 80% of design capacity. These effects have been calculated from LWR impacts presented in CHAPTER VIII, Appendix A.

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CHAPTER IV

ENVIRONMENTAL IMPACT DUE TO THE IMPLEMENTATION OF PLUTONIUM RECYCLE

Section C
THE LIGHT WATER REACTOR (LWR) WITH PLUTONIUM RECYCLE

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CHAPTER IV

Section C

THE LIGHT WATER REACTOR (LWR) WITH PLUTONIUM RECYCLE

SUMMARY

General

Much of the energy released from light water reactors (LWR's) results from fissioning in situ of plutonium formed in the uranium used as fuel during power operation of the reactors. At the end of its useful life in the reactor the spent uranium fuel contains plutonium that can be recovered and recycled into LWR's as mixed oxide (MOX) fuel, displacing an equivalent amount of enriched uranium fuel. Based on experience and analysis it is concluded that such recycle of plutonium in LWR's can be accomplished without reducing the reactor power rating, or degrading safety and the environment. Current LWR technology is adequate for the design, fabrication, and use of fuel rods containing recycled plutonium as MOX fuel. When the amount of plutonium recovered from the spent fuel removed from the core is approximately equal to the amount of plutonium in the reload fuel when initially placed in the core, the reactor is described as an equilibrium self-generation reactor, hereafter identified as SGR.

To assess the environmental impact of recycling plutonium in LWR's within the limits of demonstrated reactor technology, the quantity of recycled plutonium charged into an LWR has been assumed for this study to be 115% of the SGR value. This type of reactor is called a 1.15 SGR. A model reactor has been developed for this analysis that characterizes a 1.15 SGR in terms of a quantity of plutonium charged as MOX fuel. For the purposes of this statement, an LWR is judged to be within the limits of a 1.15 SGR when the weight (percent) of total plutonium to total heavy metal content (Pu and U) in the as-charged fuel is less than 1.8%. Further, the environmental considerations of the supporting fuel cycle have been based on the annual requirements of this postulated model reactor plant. This concentration should not be considered as a limitation on the amount of plutonium that could be safely and economically used in LWR's.

There are other plutonium recycle fuel management concepts that use more or less than the 1.15 SGR quantities of plutonium, but the plutonium fuel support cycle is not significantly affected. It makes little difference either in spent fuel reprocessing or MOX fuel fabrication whether available plutonium is recycled through all LWR's or only some of the LWR's in operation at any given time. Thus the industry projections for fuel cycle demands, etc., as developed through the use of the NUFUEL computer program are not affected by the mode of reactor recycle, but are dependent on the growth rate of the industry.

Sufficient experience from a technical and economic viewpoint has been gained with experimental and demonstration irradiations to warrant the beginning of large scale use of plutonium in commercial LWR's. In addition to a large number of experimental mixed oxide fuel irradiations, two large scale experimental programs, designed specifically to study utilization of recycled plutonium in LWR's, have been carried out in the United States. Subsequently, more than 2,500 MOX rods were irradiated in commercial U.S. reactor cores.

Environmental Impact

The materials properties and performance of MOX fuels in an LWR are in many cases indistinguishable from the corresponding UO_2 fuels, and in all instances the differences are small. The performance of a MOX model core will be similar to a UO_2 core under normal steady state and load following conditions. Changes in the nuclear and physical properties of mixed oxide fuels relative to UO_2 fuels will somewhat alter fuel behavior during operating transients and accidents, but without significant changes in the consequences. As with UO_2 fuels, essentially all of the fission products normally will be retained within the sealed MOX fuel rods.

Operational Releases to the Atmosphere

Based on experience with UO_2 fuel rods, it can be expected that some fuel cladding defects will occur during normal reactor operation and some of the fission products will be released from the fuel matrix into the primary coolant. The nonvolatile fission products are controlled by the radioactive waste system. The fission gases released to the coolant are removed, treated as necessary, and released to the atmosphere under controlled conditions within the plant operating Technical Specification limits. Similar specifications would be applied, when MOX fuel is used, to achieve the same environmental objective.

Public Health and Safety Considerations

The potential hazards to the public remain relatively unchanged by the substitution of MOX fuel assemblies for UO_2-only fuel assemblies and are noted quantitatively, for normal and accident conditions, in Section C-5.0 of this chapter. The assumed design bases for the 1.15 SGR are the same as those described for UO_2 LWR's in WASH-1250, "The Safety of Nuclear Power Reactors." Nevertheless, the NRC Office of Nuclear Reactor Regulation, in accordance with normal practice, will evaluate each utility application to use MOX fuel assemblies on a case-by-case basis. These evaluations will provide specific assurances that the risks to the health and safety of the public in the vicinity of the nuclear facility will not be affected by a change to MOX fuel. Each new type of UO_2 fuel has been routinely evaluated in the past in the same manner.

1.0 LIGHT WATER REACTORS (LWR's)

General Description[1]--Plants known as light water reactor (LWR) plants use light (i.e., ordinary) water to transfer the heat generated in the nuclear fuel to the steam-generating equipment. There are two types[1] of LWR plants--those using pressurized water reactors (PWR) and those using boiling water reactors (BWR). BWR's generate steam by bulk boiling of pressurized water in the nuclear core. In PWR's, the pressurized water circulating through the nuclear core is not allowed to go into bulk boiling, but rather is used to generate steam in equipment external to the reactor. In both types of plants, pressurized steam is produced to spin a turbine generator and produce electricity. Fossil fueled electric generating plants operate in essentially the same manner.

The heat energy produced during operation of LWR's comes primarily from the fissioning of the ^{235}U atoms in the fuel, with a small contribution (about 5%) from the fissioning of ^{238}U atoms. The fission of ^{238}U occurs only with very energetic neutrons. As the reactor operates, however, another easily fissioned atom, ^{239}Pu, is produced from ^{238}U atoms.* For each gram of ^{235}U consumed in LWR fuel, as much as 0.9 gram of ^{239}Pu is formed within the fuel. Generally, more than half of the plutonium formed undergoes fission in place, thus contributing significantly to the energy produced by the reactor. Plutonium that escapes fission (about 0.25 gram of ^{239}Pu per gram of ^{235}U consumed) and the nonfissile plutonium that is formed can be recovered from the spent LWR fuel at the spent fuel reprocessing plant.

1.1 BWR Description[1,2]

The nuclear steam supply system of a BWR consists primarily of the reactor vessel and equipment inside the vessel. See Figures IV C-1 and IV C-2. The nuclear fuel assemblies are arranged inside a core shroud in the reactor vessel. Water boils in the core, and a mixture of steam and water flows out the top of the core and through steam separators at the top of the core shroud. Steam passes through dryers to remove all but traces of entrained water and then leaves the reactor vessel through pipes to the turbine generator. Water from the steam separators mixed with water returned from the turbine condenser flows downward through the annulus between the core shroud and the reactor vessel, and returns to the bottom of the core. Because the energy supplied to the reactor coolant (water) from the hot fuel is transported directly (as steam) to the turbine, the BWR system is called a "direct cycle" system. Pressure in a typical BWR is maintained at about 1,000 pounds per square inch (psi); at this pressure water boils and forms steam at about 545°F.

The arrangement of the nuclear steam supply system is shown schematically in Figure IV C-1. A diagram of the reactor vessel and internals for a typical BWR is shown in Figure IV C-2. Steam flows from the reactor vessel to the turbine-generator in multiple main steam lines. Both the head of the vessel and the steam separators and dryers are removable for refueling. Neutron-absorbing control and safety

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*On the average, fissioning atoms in LWR fuel each eject somewhat more than two neutrons, one of which is needed to sustain the fission chain reaction. Those neutrons not entering into fission reactions either leak from the core or are captured in the fuel or by surrounding materials. When ^{238}U captures a neutron not sufficiently energetic to cause fission, it transforms spontaneously to neptunium-239, which in turn transforms to plutonium-239 over a relatively short time span.

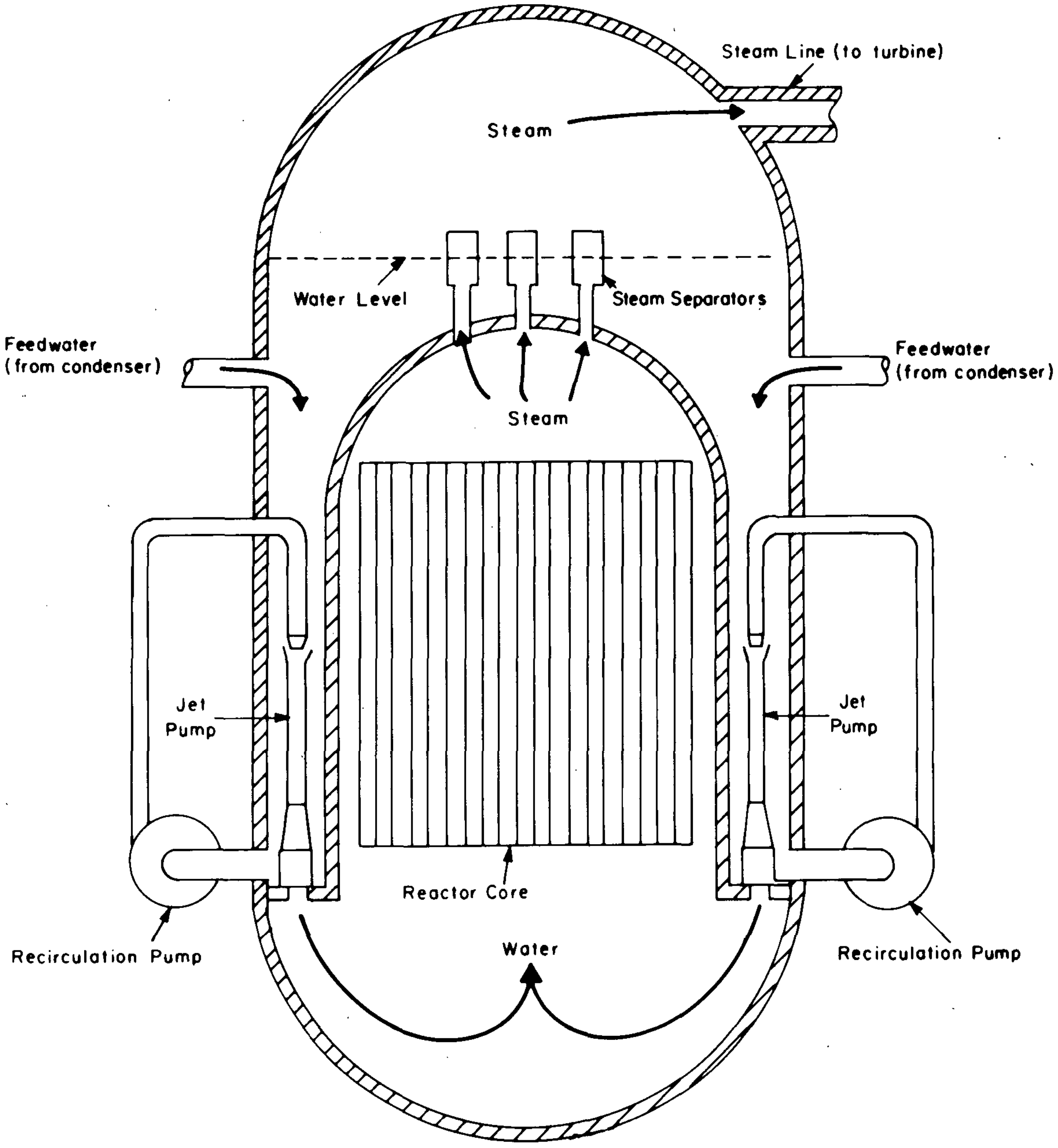

Figure IV C-1 Schematic Arrangement of BWR NUCLEAR STEAM SUPPLY SYSTEM

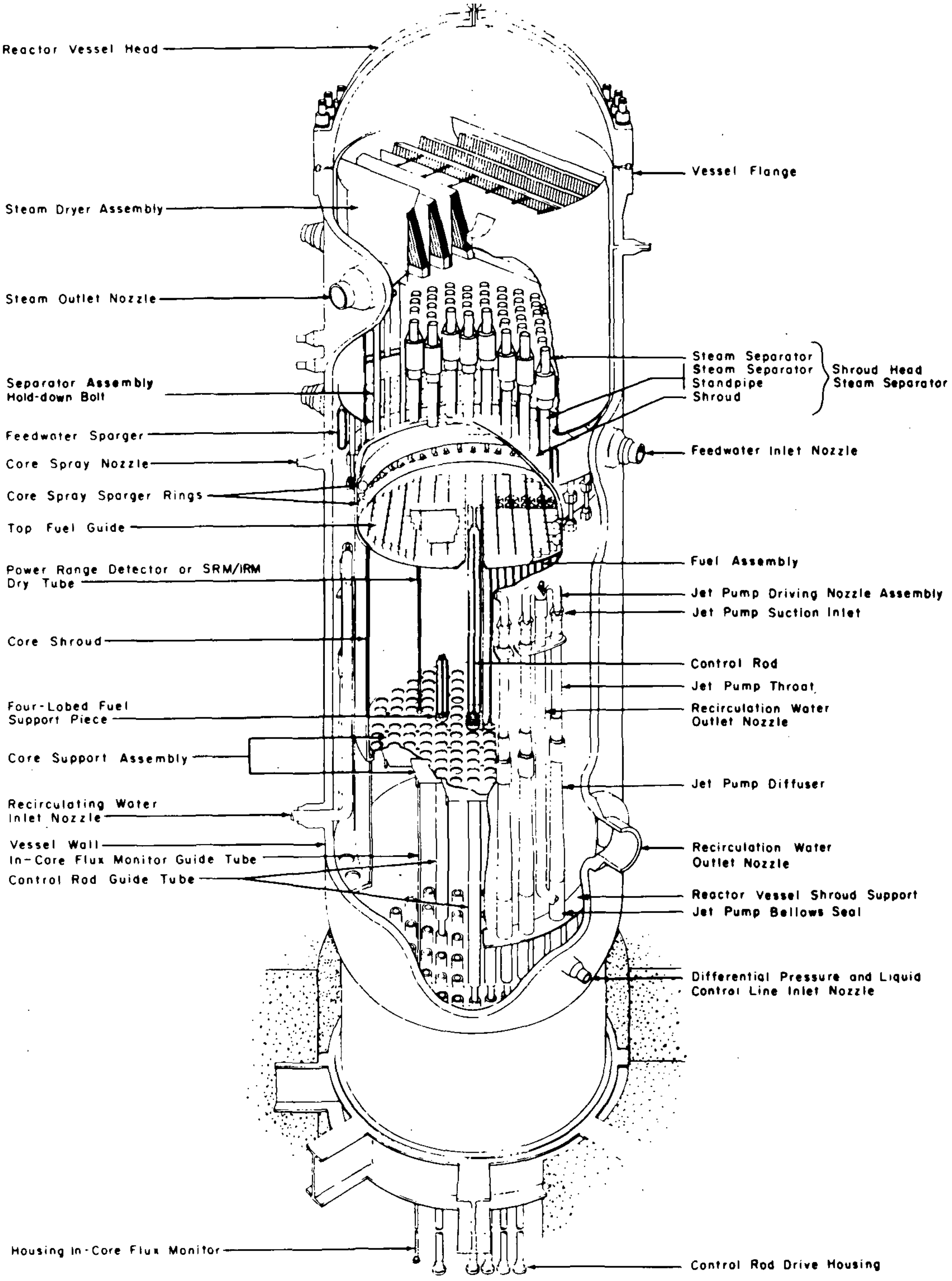

Figure IV C-2 Cutaway View of Internals of Typical BWR Vessel elements in the reactor core are connected to rods that pass through fittings in the bottom head of the vessel and are operated by hydraulic drives mounted below the vessel. Because the reactor heat output is sensitive to the rate of flow of coolant through the core, partial control of the power is effected by varying the jet flow to the recirculation pumps that provide water circulation for core cooling.

In a typical modern BWR, seven by seven or eight by eight square arrays of Zircaloy tubes containing fuel pellets stacked to about 12 feet in height are assembled within metal channels of square cross section. Figures IV C-3 and IV C-4 show a fuel assembly that contains 49 rods. The channel is open at the top and bottom to permit coolant flow upward through the assembly; the closed sides prevent lateral flow between adjacent assemblies in the reactor core. The core of a nominal 1,000 MWe BWR may contain as many as 688 fuel assemblies, each about 5-1/2 inches on a side and about 14 feet in overall length (at 63 fuel rods per 8x8 assembly this is about 43,000 fuel rods per reactor) with a total weight of uranium dioxide of more than 320,000 pounds or 146 metric tons (MT) (approximately 284,000 pounds or 129 MT of heavy metal). Four different ^{235}U enrichments are used in typical eight by eight rod arrays, with the most highly enriched fuel rods placed in the interior, to reduce the local power peaking factor and thereby make fuel rod power production more uniform.

The amount of heat that can be extracted from a reactor core of a given size depends on the rate of recirculation of water through the core. In most recent designs, jet pumps are provided in the annulus outside the core shroud to greatly increase the circulation rate over the natural circulation induced by the boiling in the core.

Boiling water reactors have multiple provisions for cooling the core fuel in the event of an unplanned depressurization or loss of coolant from the reactor. These provisions may differ from plant to plant, but all plants have several independent systems to achieve flooding or spraying, or both, of the reactor core with coolant upon receiving a signal of either high dry well* pressure or low reactor vessel water level. Typical emergency core cooling systems involve either a low pressure core spray system (early BWR's) or a high pressure core spray system (latest BWR's) to assure adequate core cooling in the event of a leak that results in depressurization of the reactor system.

Containment systems of current-design BWR's generally provide both primary and secondary containment. The former is a steel pressure vessel, surrounded by reinforced concrete and designed to withstand peak transient pressures that might occur under the most severe of the postulated, though unlikely, loss of coolant accidents. Under accident conditions, valves would automatically close in the main steam lines from the reactor to the turbine generators, and any steam escaping from the reactor system would be released entirely within the dry well. The resulting increase in dry well pressure would force the air-steam mixture in the dry well down into and through

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*Dry well - The primary containment enclosing the entire reactor vessel and its recirculation pumps and piping, which is connected through large ducts to a lower level pressure suppression chamber, i.e., a large pool of water as shown in Figure IV C-5.

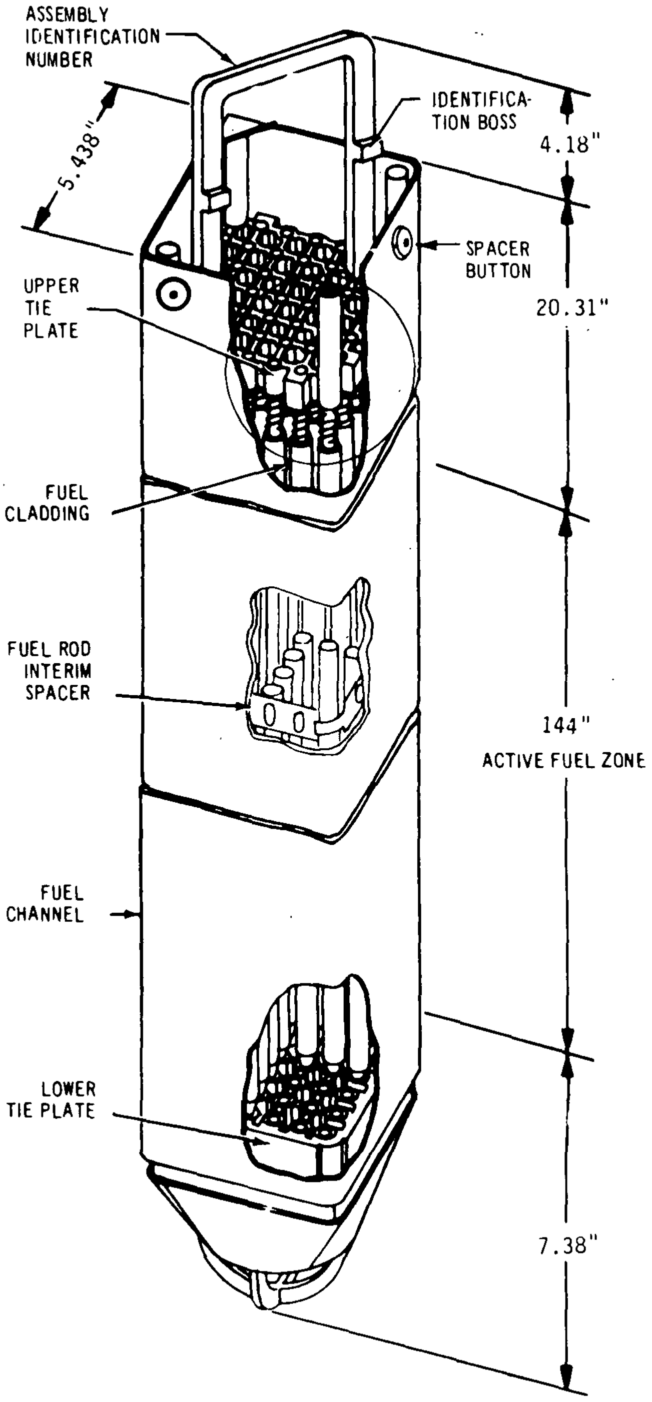
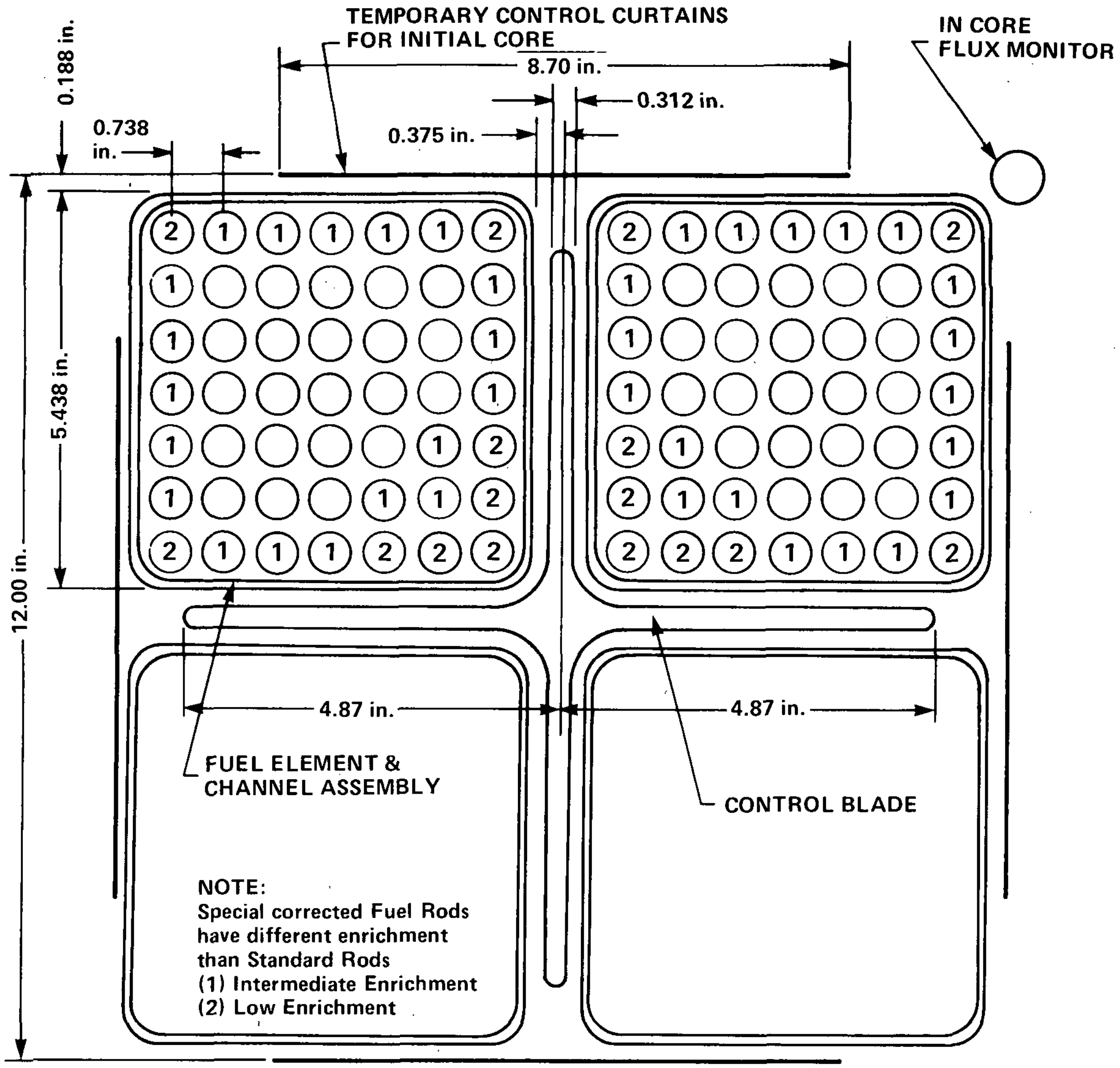


Figure IV C-3 BWR Fuel Assembly



NOTE:
Special corrected Fuel Rods have different enrichment than Standard Rods
(1) Intermediate Enrichment
(2) Low Enrichment

Figure IV C-4 Typical Core Lattice Unit

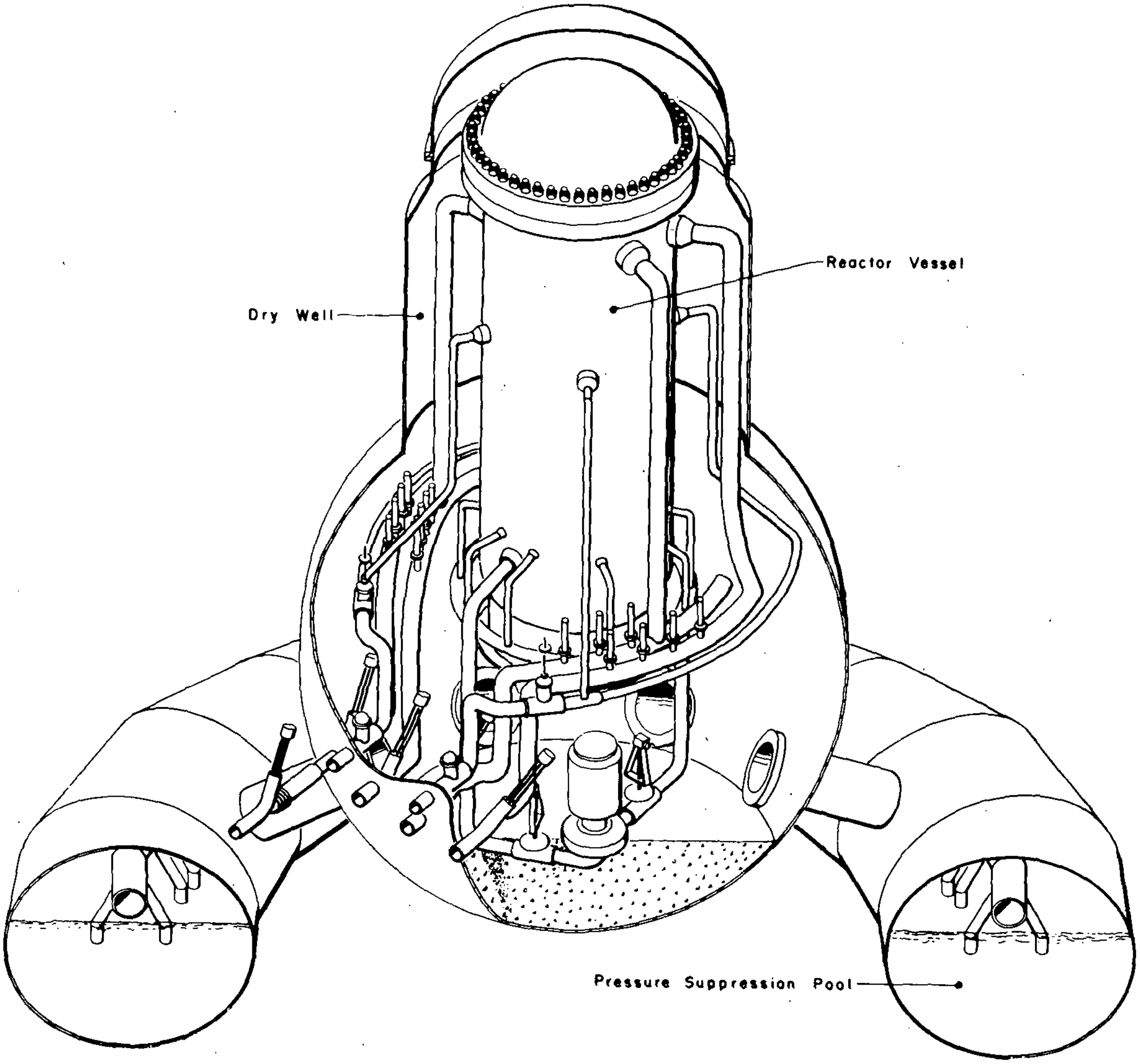


Figure IV C-5 Schematic Arrangement of BWR Primary Containment System the water in the pressure-suppression chamber, where the steam would be completely condensed. Steam released through the pressure-relief valves of the automatic depressurization system would be condensed in the pressure-suppression pool. This pool serves as one source of water for the emergency core cooling system. Systems are provided for the control of combustible gases from metal-water reactions and radiolytic decomposition of the water, to assure that flammable concentrations are not reached in the containment.

The secondary containment system is the reactor building, which houses the reactor and its primary containment system. A typical system is shown in Figure IV C-6. The buildings, substructures, and exterior walls up to a level above the top of the dry well are of poured-in-place reinforced concrete. The secondary containment of operating BWR plants is designed for low leakage and has sealed joints and interlocked double-door entries. Under postulated accident conditions, the normal building ventilation system would automatically shut down and the building would be exhaust ventilated, so as to maintain a slight negative pressure, by two parallel standby systems discharging through the plant stack or roof exhaust system (elevated releases) to minimize any possible ground level release through important seals or other leakage paths. These gas treatment systems include high efficiency particulate air (HEPA) filters and solid adsorbents for trapping radioactive halogens, particularly iodine, that might have leaked from the primary containment.

The most advanced BWR plants use a separate free-standing leak-tight containment shell inside of a sealed building (see Figure IV C-7), providing a further barrier to the escape of gaseous effluents, as well as shielding to further reduce radiation levels from sources inside the containment during normal and accident conditions.

1.2 PWR Description[1,3]

PWR's employ primary and secondary coolant systems for transferring energy from the reactor to the turbine and are called indirect cycle systems. The high pressure water heat transport circuit, comprising the reactor vessel, piping, the necessary pumps, and the inner tube side of the steam generators, is termed the primary system; the lower pressure steam generator to turbine circuit is called the secondary system. A schematic arrangement of a 1,000-MWe PWR system, with four steam generators and one pump for each steam generator, is shown in Figure IV C-8.

The primary system pressure maintained in a typical large PWR system, about 2,250 (psi), permits water to be heated to about 650°F without boiling. The high pressure water, heated to an average temperature of around 600°F, is piped out of the reactor vessel into two or more "steam generators." Heat from the high pressure reactor coolant water is transferred through heat exchanger tubes into a secondary water system, causing this water to boil and produce 1,000-psi, 545°F steam at the turbine inlet.

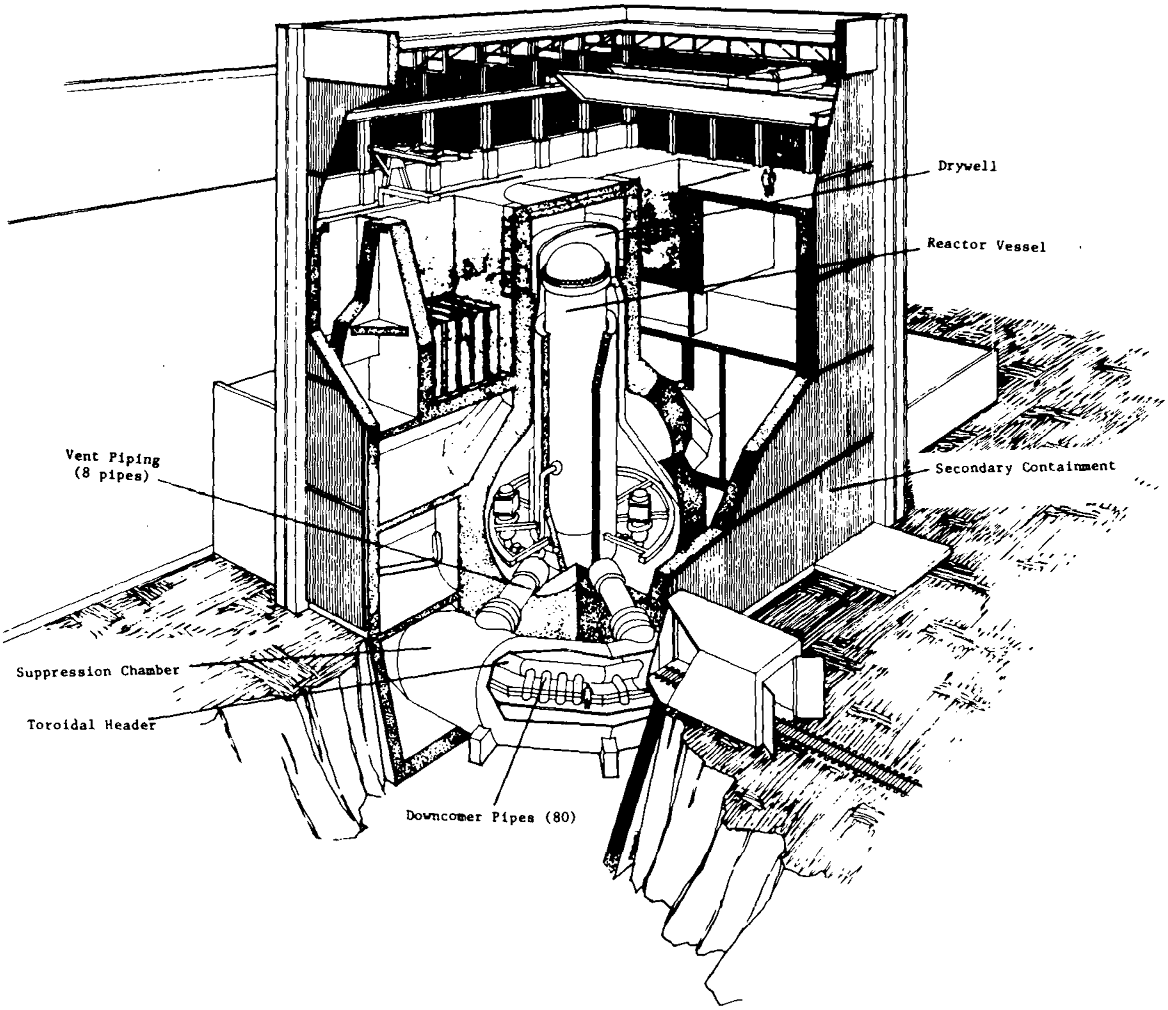


Figure IV C-6 BWR Secondary Containment Building Showing Primary Containment System Enclosed

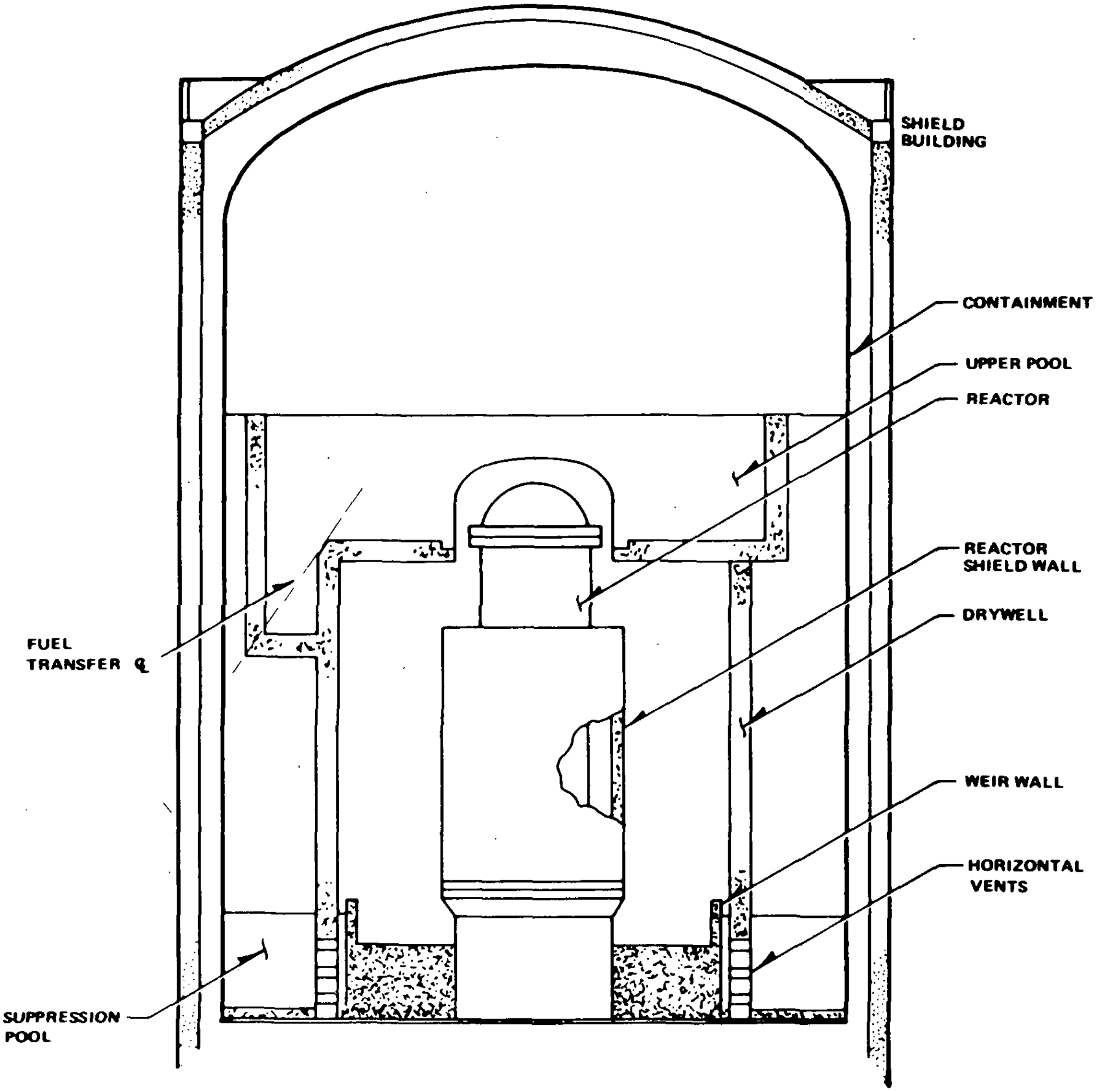


Figure IV C-7 Multiple Containment System for Recent Large BWR Plants

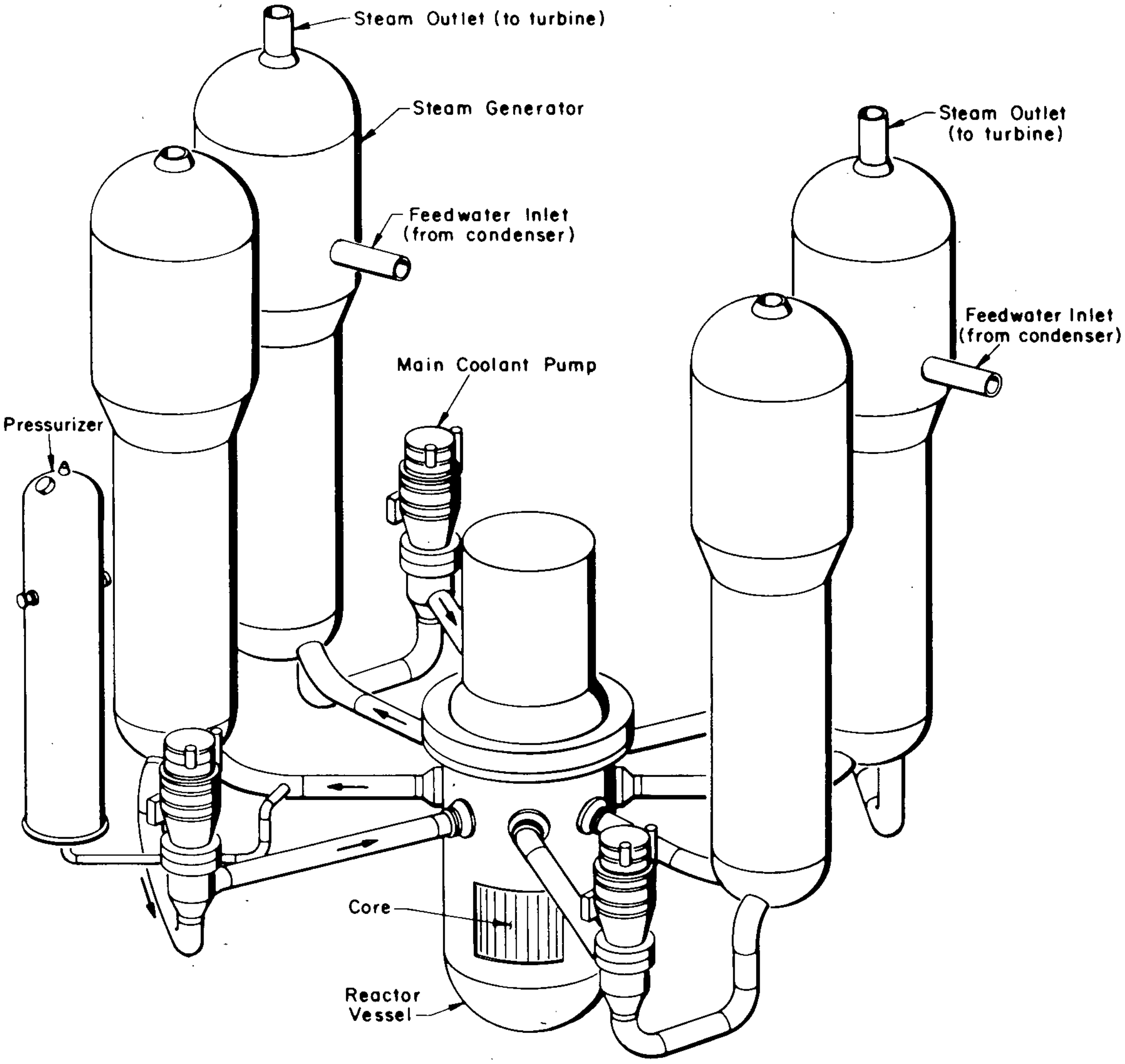


Figure IV C-8 Schematic Arrangement of PWR NUCLEAR STEAM SUPPLY SYSTEM

A cutaway view of a typical PWR reactor vessel and its internals is shown in Figure IV C-9. The vessels have removable top heads (for refueling) and are provided with fittings to accommodate mechanisms for driving neutron-absorbing control rods into and out of the core to control the nuclear chain reaction. Additional control of the chain reaction is provided through the use of variable-concentration neutron-absorbing chemicals, such as boric acid, dissolved in the primary system coolant.

One type of large (1,000 MWe) PWR core contains approximately 50,000 fuel rods in about 190 assemblies totaling about 92 metric tons of slightly enriched uranium dioxide. For the large PWR described herein, 264 fuel rods with fuel pellets stacked to about a 12-foot height are assembled into a 17x17 bundle of square cross section, which normally is about 8-1/2 inches on a side. PWR fuel assemblies are not surrounded by a channel as are the BWR assemblies. The fuel rods are arranged in relatively open arrays. See Figures IV C-10 and IV C-11.

The PWR plant circulates the primary coolant through large steam generators. High performance primary-coolant pumps are designed to operate at 650°F at pressures up to 2,500 pounds psi and are manufactured to stringent specifications. Steam supply systems of PWR's are equipped with pressurizers (see Figure IV C-8) to maintain required primary coolant pressure during steady state operation, to limit pressure changes caused by coolant thermal expansion and contraction as plant loads change, and to prevent coolant pressure from exceeding the design pressure of the entire primary system. The reactor vessel, steam generators, pumps and all other parts of the primary system including the pressurizer are located within the containment.

The major function of the emergency core cooling system of a PWR is to supply sufficient water to cool the core in the event of a break that permits water to leak from the primary system. This break may be very small, or it may be a rupture of the largest coolant pipe in the system. Emergency core cooling systems of PWR's consist of several independent subsystems, each characterized by redundancy of equipment and flow path, to assure reliability of operation and continued core cooling even in the event of failure of any single component. Although the arrangements and designs of PWR emergency core cooling systems vary from plant to plant, depending on the vendor of the steam supply system, all modern PWR plants employ both accumulator injection systems and pump injection systems, with redundancy of equipment to assure operation.

Most present day PWR containments are constructed of reinforced concrete with a steel liner. See Figure IV C-12. All are sized and designed to withstand the maximum temperature and pressure that would be expected from the steam produced if all the water in the primary system were expelled into the containment. Two additional measures are taken in PWR plants to minimize the potential for escape to the environment of any accidental release of radioactive materials. In some plants, cold water sprays are provided to condense the steam resulting from a major escape of primary system coolant; in others stored ice is used for this purpose. By condensing the steam and thus lowering the containment pressure, the driving force for outward leakage is reduced.

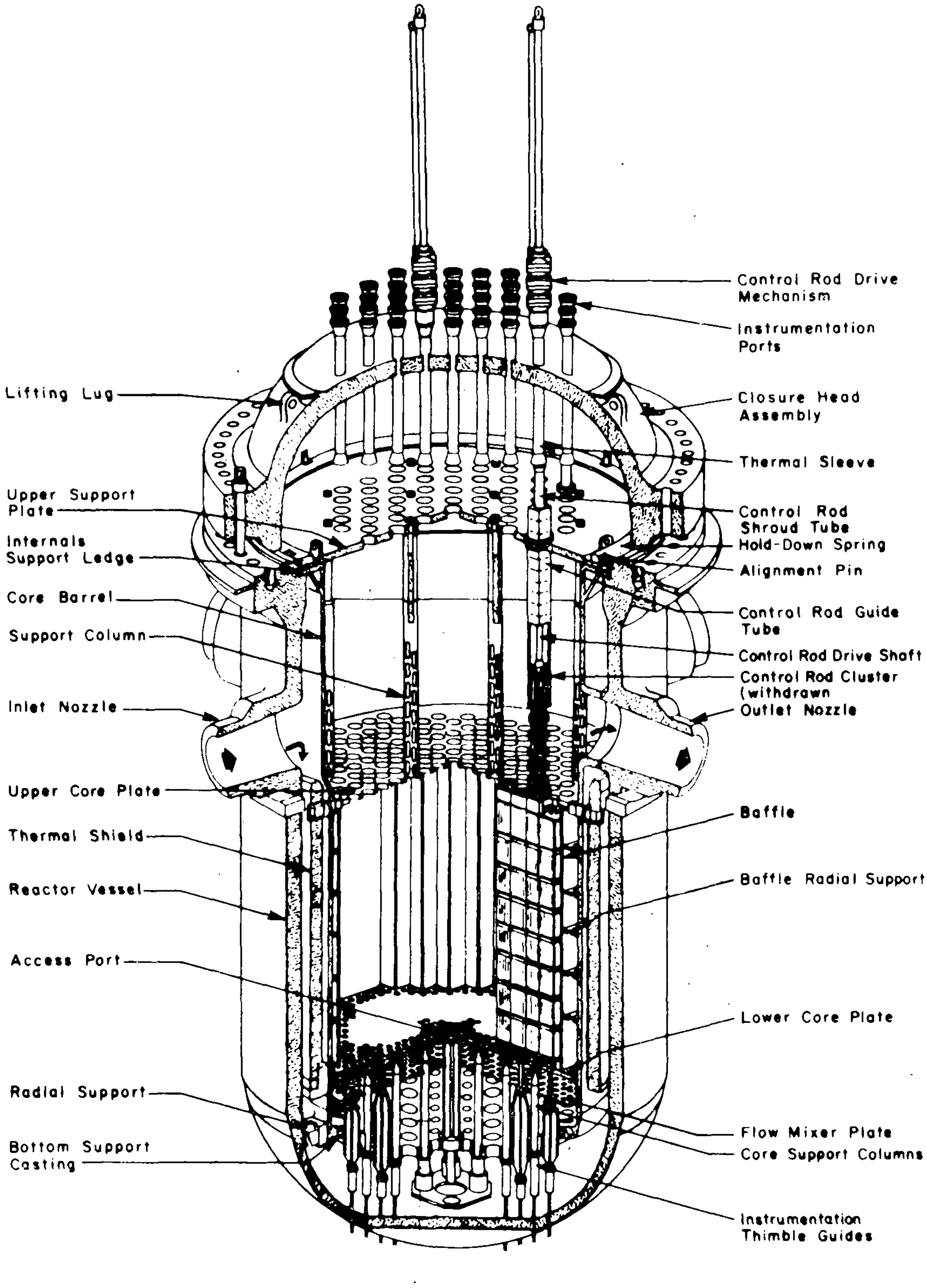


Figure IV C-9 Cutaway View of Internals of Typical PWR Vessel

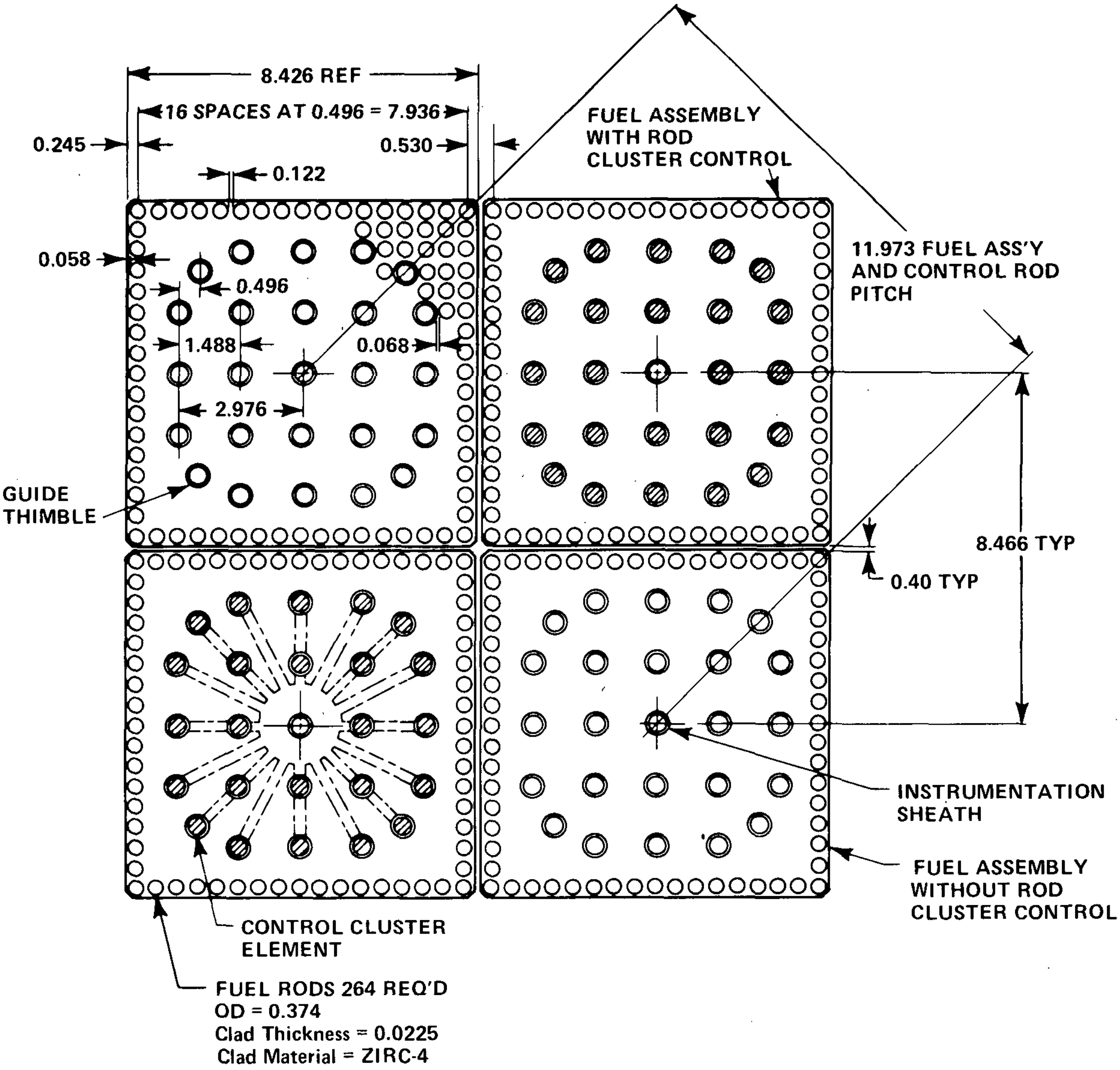


Figure IV C-10 Fuel Assembly Cross Section 17 x 17

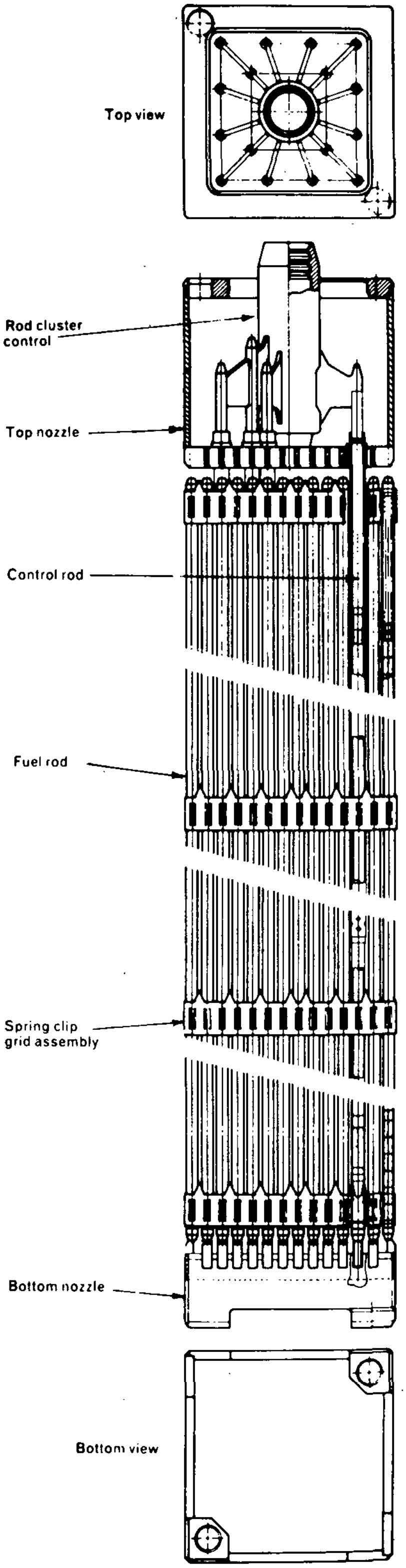


Figure IV C-11 PWR Fuel Assembly

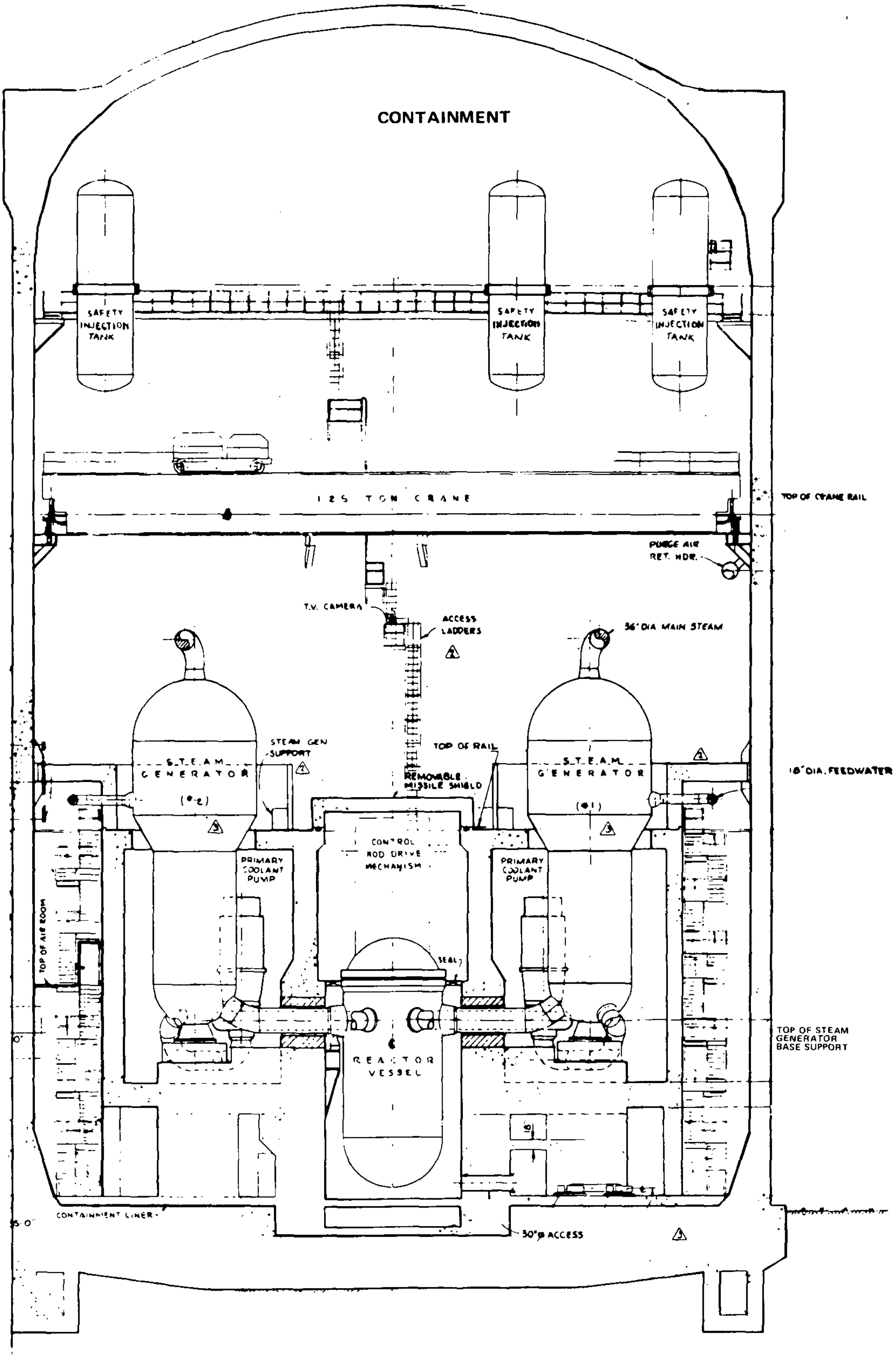


Figure IV C-12 PWR Containment

Another safety measure provides blowers to recirculate containment atmosphere through filters and absorption beds to remove airborne radioactive materials. When sprays are used in the containment, chemicals are usually added to the spray solution to increase the retention of airborne radioactive materials that dissolve in or become entrained by the spray. Systems are also provided for control of hydrogen from both metal-water reactions and radiolytic decomposition of the water, to assure that the amounts of H_2 and O_2 in the containment are below flammable concentrations.

More detailed discussions of design considerations for specific safety system practices for assuring safety and analyses of hypothetical accident sequences for the PWR and BWR are presented in References 1, 2 and 3.

REFERENCES

1. U.S. Atomic Energy Commission (AEC), "The Safety of Nuclear Power Reactors (Light Water Cooled) and Related Facilities," WASH-1250, July 1973.

2. General Electric Co. (GE), Standard Safety Analysis Report, BWR/6, April 30, 1973.

3. Westinghouse Nuclear Energy Systems, RESAR (Rev. 3) Reference Safety Anaysis Report, June 1972.

2.0 RECYCLE PLUTONIUM AS FUEL IN LWR's

Starting in the year 1957, the Atomic Energy Commission (AEC) carried out a plutonium recycle research and development program at a total cost of about $112 million. Of this, about $84.5 million was expended for research and development, about $2.5 million for the Saxton Plutonium Experiment and $25 million for facilities.* In addition, $10 million is estimated to have been invested in research and development (R&D) by industry. AEC's direct support of the plutonium utilization program terminated at the end of fiscal year 1972, though indirect support continued through FY 1973 through cooperative efforts with industry under which AEC supplied plutonium for recycle demonstrations at a reduced charge.[1]

Mixed oxide fuel assemblies[2] that are interchangeable with UO_2 fuel assemblies have been designed and irradiated in both BWR's and PWR's. In PWR's, as shown in Figure IV C-13, control rods (RCC's) are located within the fuel assembly lattice. When plutonium is substituted for ^{235}U in the nearby fuel rods, the higher neutron absorption cross section results in a decrease in effective control rod worth. Because more than half of the fuel assemblies in this type of PWR do not contain rod cluster controls (RCC's), it is possible to use plutonium in these fuel assemblies without significantly affecting overall reactor control worth. However, fuel management requirements for fuel assembly interchangeability during the fuel irradiation lifetime significantly reduce the number of core positions that can be occupied by MOX fuel assemblies. Because of the intimate arrangement of RCCs and fuel rods, PWR's containing both mixed oxide (PuO_2 and UO_2) and uranium oxide (UO_2) fuel elements probably will operate with the mixed oxide fuel in separate and distinct subassemblies. The MOX fuel assemblies would not be placed in core positions containing RCC,s.

Control rods in BWR's (and some PWR's) are external to the fuel assembly, as shown in Figure IV C-14; but every bundle in a BWR is adjacent to a control blade. Loss of control rod worth in BWR's is precluded by locating MOX fuel rods in locations away from the periphery of a bundle. It is possible to use MOX fuel in more than one-third of the rods in a BWR core[2,5] without affecting control worth by utilizing the bundle island concept where only rods of greater than average ^{235}U enrichment are replaced with mixed oxide rods.

The isotopic composition of plutonium varies with fuel exposure as typified by Figure IV C-15, and with repeated reycle of recovered plutonium in LWR's.[4] In one generic BWR design[5] for a plutonium recycle reactor, approximately 25% of the fuel rods in reload fuel assemblies contained plutonium, but the fissile plutonium inventory in these rods was calculated to be 42% of the total fissile inventory in the fuel assembly. With successive recycles, the buildup of ^{236}Pu, ^{238}Pu, ^{240}Pu and ^{242}Pu increases and a greater amount of fissile plutonium, ^{239}Pu and ^{241}Pu, is required to

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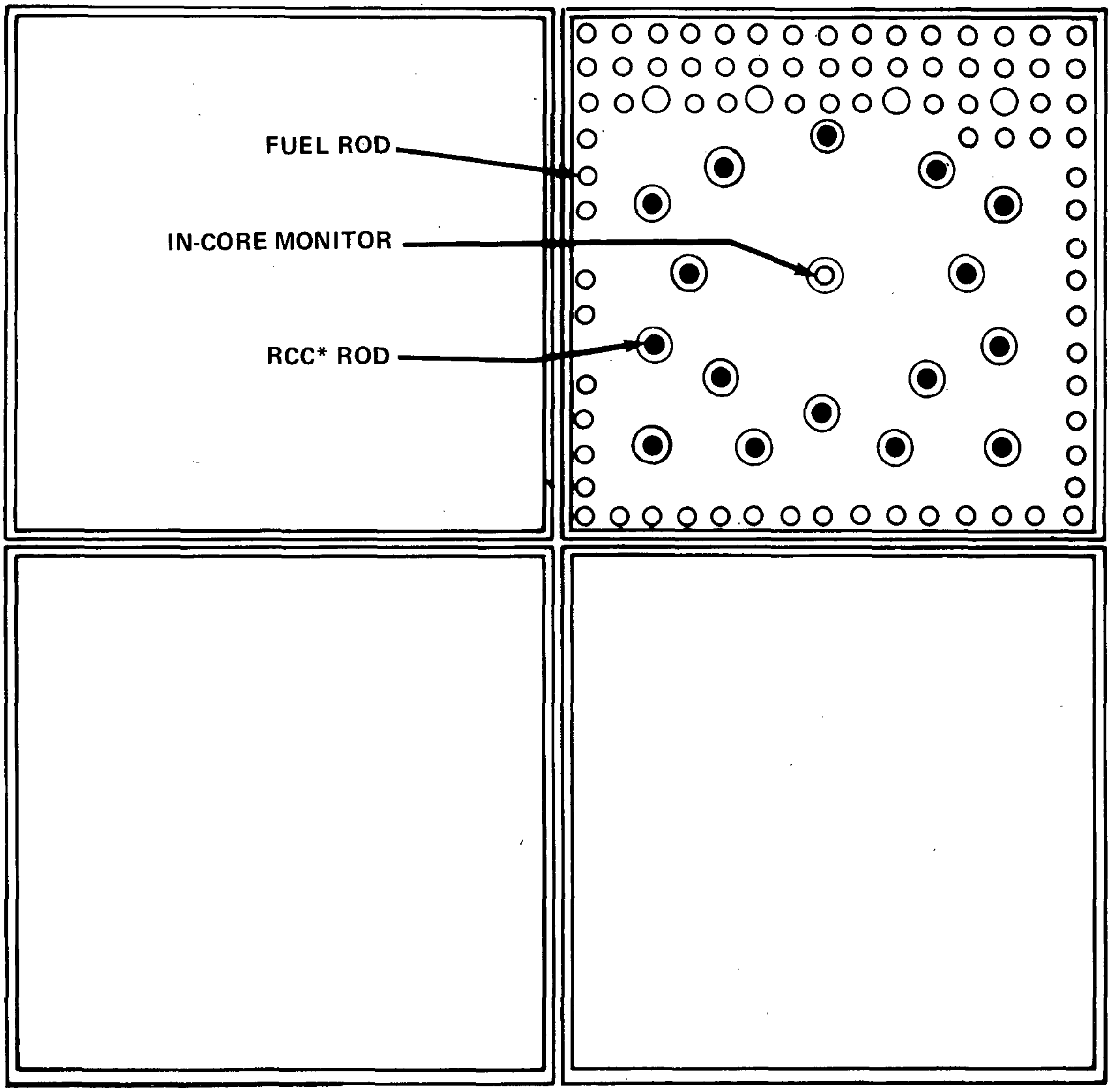
*See CHAPTER II and CHAPTER IV, Section C-3.0 for more information on Pu background.



*RCC – Rod Cluster-Control

Figure IV C-13 Typical PWR Assemblies

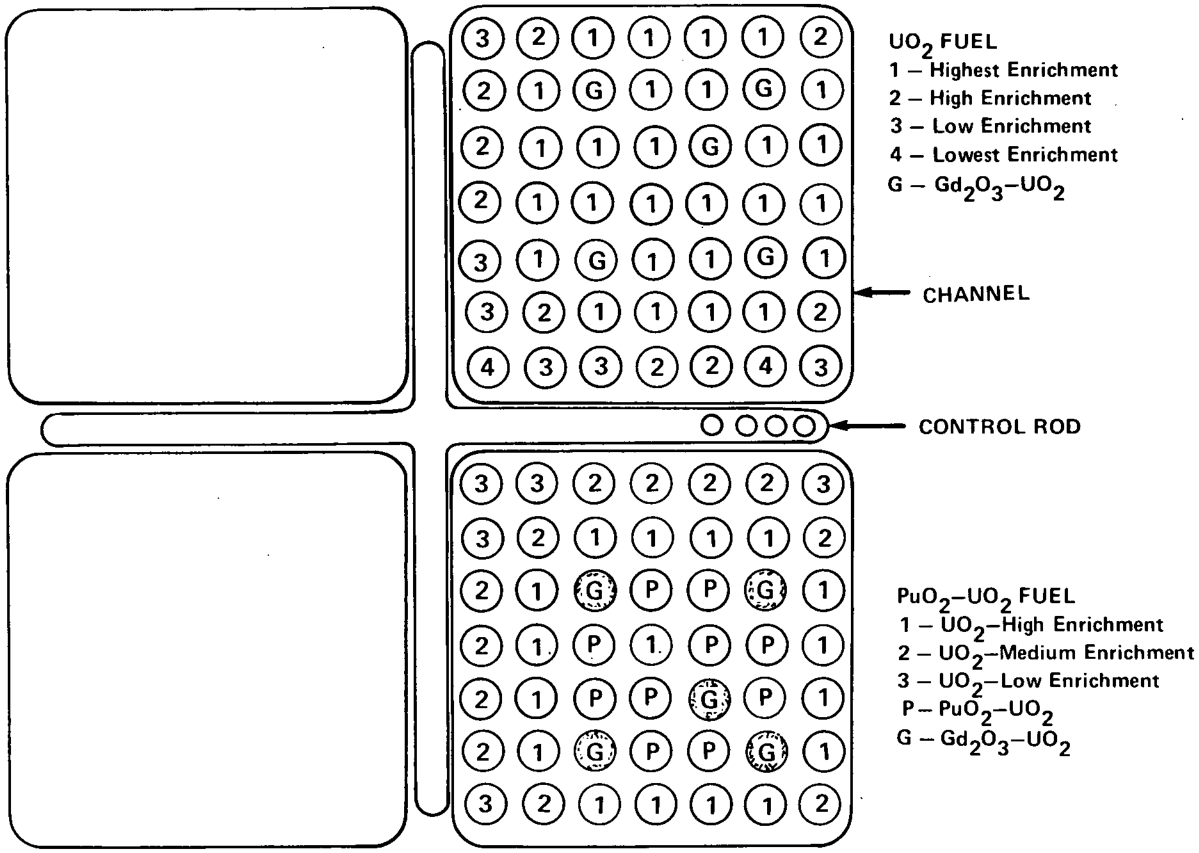


Figure IV C-14 Typical BWR Fuel Module With Control Rod

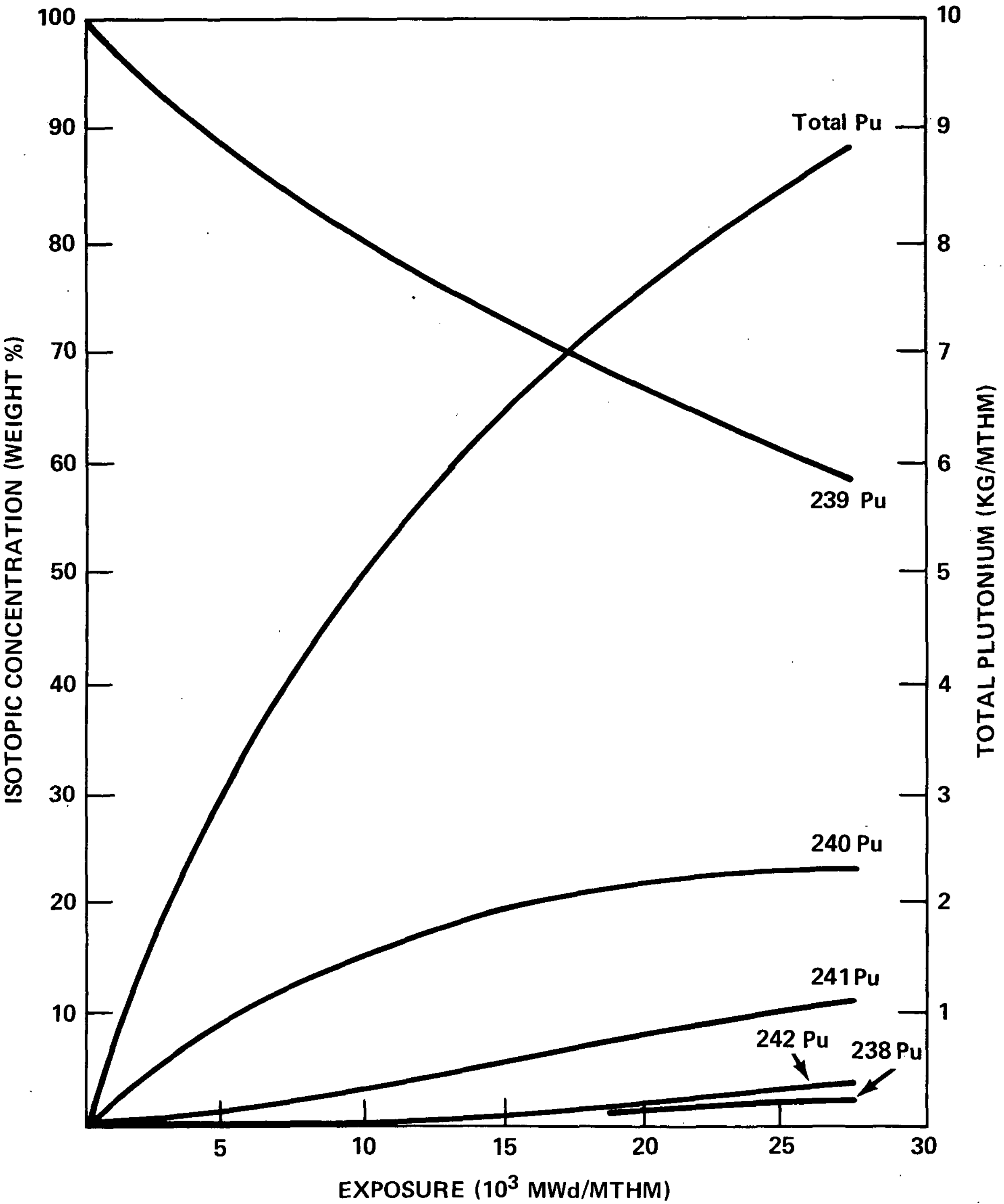


Figure IV C-15. Plutonium Composition vs. Fuel Exposure (Model BWR) compensate for parasitic capture.[1] This has been illustrated by Deonigi in "The Value of Pu Recycle in Thermal Reactors."[6] For calculational purposes, Deonigi assumed that all available plutonium was recycled beginning in the year 1974. For these assumed conditions he noted that during the next decade the nuclear industry would pass through a transition from 1975 where most of the fuel would be discharged at low exposure, to a situation in the year 1985 where the discharged fuel would be predominantly at equilibrium exposures. For these assumed conditions the average composition of plutonium available for recycle is shown in Table IV C-1.

Table IV C-1

AVERAGE COMPOSITION OF PLUTONIUM[6]
AVAILABLE FOR RECYCLE--PERCENT

Year	^{236}Pu	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
1975	0.006	1.0	64	22	10	3
1980	0.007	1.5	58	24	11	5
1985	0.007	1.7	54	25	12	7

The table shows the relative decrease of fissile ^{239}Pu with time, and the significant increase of nonfissile ^{242}Pu with irradiation time.

If plutonium recycle is approved and because plutonium will probably be utilized in reactors designed for enriched uranium fuels and mixed oxide fuel assemblies will probably be designed to be mechanically, neutronically, and operationally interchangeable with uranium elements, the MOX fuel assemblies must perform at least as well as their uranium counterparts. The results of mixed oxide fuel testing programs to date have shown this is the case.[3]

Operation of LWR's with recycle plutonium must be in accordance with the regulations and technical specifications imposed by the Nuclear Regulatory Commission to insure safe plant operation, including requirements for engineered safety features and limits on radioactive releases to the environment.

2.1 Assessment of Nuclear Power Plant Safety

The Nuclear Regulatory Commission requires that the primary assurance of safety be attained through a high degree of reliability and predictability obtained by application of rigorous standards in the design, construction, and operation of the nuclear facility, and through extensive quality assurance controls. In addition, in accordance with the defense in depth concept, safety features and engineered safeguards systems are provided to prevent or to mitigate the consequences of accidents postulated to occur in spite of these measures.

Thus, the Commission's regulations, as stated in Section 50.34 of 10 CFR Part 50, require that each applicant requesting a construction permit or operating license for a nuclear power plant provide an analysis and evaluation of the design and performance of the structures, systems, and components of the facility, so that potential risk to public health and safety resulting from operation of the facility can be assessed. The analyses must include a determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

The conditions to be analyzed range from relatively trivial events that result in essentially no risk to the public (release within the criteria for routine operation), which might occur with moderate frequency, to accident situations with a theoretical potential for large consequences, which are very unlikely to occur. The range of accidents considered can be categorized into three classes:

- Events of moderate frequency (anticipated operational occurrences) leading to no abnormal radioactive releases from the facility

- Events of small probability with the potential for small radioactive release from the facility

- Potentially severe accidents of extremely low probability, postulated to establish the performance requirements of engineered safety features and used in evaluating the acceptability of the facility site

It is highly desirable for both safety and economic reasons that the first class, moderate frequency events, such as partial loss of forced reactor coolant flow, should result in reactor shutdown with no radioactive release from the fuel and with the plant capable of readily returning to power after corrective action is taken. Analysis and evaluation of these moderate frequency conditions, examples of which are listed in Part A of Table IV C-2, offer the opportunity to detect and correct faults that might otherwise lead to more serious failures in a particular plant design. Safety is enhanced if all those events that can be identified as having a reasonable chance of occurring are shown to be covered by design considerations for protection against such occurrence or against their ability to produce significant damage to the plant if they do occur.

The second class of events, such as complete loss of forced reactor coolant flow or partial loss of reactor coolant from small breaks or cracks in pipes, must be shown to present minimal radiological consequences. The actual occurrence of such accidents may, however, prevent resumption of plant operation for a considerable time because of the potential for failure of the cladding of some fuel rods and the consequent requirement for replacement and cleanup. Evaluation of these accidents (examples given in Part B of Table IV C-2) must show that under accident conditions the engineered safety features and containment barriers function effectively to eliminate or reduce to an insignificant level the potential for radioactive release to the environment. These studies also show the effectiveness of safety features designed into the facility to

Table IV C-2

POSTULATED REACTOR FACILITY ACCIDENTS

A. Moderate Frequency Events (no abnormal radioactive release from the facility)

- Withdrawal of control rod at maximum speed because of malfunction or error
- Failure of one safety rod to scram when required
- Partial loss of normal forced reactor coolant flow
- Unintentional startup of an inactive reactor coolant loop
- Loss of external electrical load, turbine trip, or both
- Loss of offsite electrical power
- Excessive load increase
- Loss of normal feedwater flow
- Inadvertent depressurization of the primary coolant system

B. Infrequent Accidents of Small Probability (abnormal radioactive release possible, but not expected)

- Small leaks and breaks in pipes (or minor leaks in large primary or secondary system pipes)
- Inadvertent loading of a fuel assembly into an improper position
- Complete loss of normal forced reactor coolant flow
- Complete loss of all alternating current (AC) power (station blackout)
- Major leakage in radioactive waste decay tank

C. Highly Unlikely Accidents (postulated for evaluating site acceptability)

- Major rupture of pipes containing reactor coolant up to and including double-ended rupture of largest pipe in the primary coolant system (loss of coolant accident)
- Major secondary or steam system pipe rupture up to and including double-ended rupture of a main steam pipe
- Control rod ejection
- Severe fuel handling accident
- Tornadoes, floods, and earthquakes

* * *

cope with unlikely accidents, and show the margins of safety that exist in the reactor facility design by indicating the type of failures that can be accommodated without raising safety concerns.

To provide additional defense in depth, extremely unlikely accidents of the third class, Part C of Table IV C-2, are postulated to occur in spite of the low probability of occurrence and the prevention steps taken. These hypothetical events are evaluated under highly conservative assumptions, as for example a major rupture of a pipe in the reactor coolant system, a loss of coolant accident (LOCA), along with degraded performance of safety systems. Consideration of such postulated accidents, and the provision of engineered safety features to assure that potential radiological consequences would result in a low risk to the public health and safety, are required by the NRC reactor site criteria (10 CFR Part 100).

Postulated occurrences involving sequences of successive failures more serious than those required to be considered in the design bases of protection and engineered safety features could result in severe consequences. Defense in depth (multiple physical barriers) quality assurance, control of design, manufacture, construction, and operations, continued surveillance testing, and conservatism in design are all applied to provide and maintain a high degree of assurance that potential accidents in this class are, and will remain, sufficiently small in probability that the environmental risk is very low.

2.2 Assessment of Site Acceptability

NRC Regulation 10 CFR Part 100, "Reactor Site Criteria," requires that for the purposes of radiological dose calculations, two boundaries be considered in connection with a proposed nuclear power plant site: the "exclusion area" boundary and the "low population zone" boundary. The exclusion area is defined as that area surrounding the reactor in which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property from the area. The low population zone is the area immediately surrounding the exclusion area, which may or may not contain residents, and its boundary must be no more than three quarters of the distance from the reactor to the nearest boundary of a population center containing about 25,000 residents.

In addition, the site criteria state that the radioactivity release assumed for use in evaluation of site acceptability should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, but that would result in potential hazards not exceeded by those from any accident considered credible. A number of hypothetical accidents, called design basis accidents (DBA), are considered in making this evaluation.

On the basis of the DBA analyses and other pertinent information, the NRC staff determines whether or not the specific reactor design, including the engineered safety features, located at the specific site proposed by the applicant provides acceptable protection to the public. As part of this determination, the staff takes into consideration the population density within the low population zone as well as other factors, such as number and size of highways, schools, factories, hospitals, etc., the scope and extent of advance emergency planning, and the actual distribution of residents within the zone.

2.3 Design Basis Accidents - Nuclear Power Plants

The hypothetical accidents evaluated during the safety review of power reactors include

- Loss of the reactor coolant resulting from postulated major ruptures in the primary coolant system piping (loss of coolant accidents)

- Damage to reactor fuel during reactor operation as a result of rapid power increase beyond design limits, caused by uncontrolled withdrawal of control rods at maximum speed because of postulated control system failures

- Damage to reactor fuel during refueling operations such as dropping an irradiated fuel assembly

- In PWR's, system transients resulting from major ruptures in the reactor secondary or steam system piping.

For each of these types of accidents, the potential exists for breaching of the fuel rod cladding and release of radioactive material from the reactor fuel, transport of a portion of this radioactive material through leakage paths in the containment barriers and finally, for leakage of some portion of it to the environment. The analysis of each type of accident is performed to assure that adequate safety features have been engineered into the plant, in the form of passive barriers or active systems, to limit the consequences of a release of fission products from the reactor fuel, and to show that the maximum radiological doses would not exceed the values specified in 10 CFR Part 100, even with highly pessimistic assumptions.

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1. J. Haley, "End of AEC Buyback; Beginning of Commerical Plutonium Recycle," Third Edition, American Power Conference, Chicago, Ill., April 1971.
2. S. Strauch, AEC Paper, "Status of Pu Recycle in LWR's," August 1973.
3. M. D. Freshley, Battelle Pacific Northwest Laboratories, "UO_2-PuO_2 - A Demonstrated Fuel for Plutonium Utilization in Thermal Reactors," Nuclear Technology, May 1973.
4. C. A. Sege, AEC Paper, "Pu Recycle Possibilities in LWR's," September 1972.
5. R. H. Stone, "Plutonium Utilization Study," GE letter to AEC, July 10, 1973.
6. D. E. Deonigi, Battelle Pacific Northwest Laboratories, "The Value of Plutonium Recycle in Thermal Reactors," Nuclear Technology, May 1973.

3.0 EFFECTS OF Pu UTILIZATION ON REACTOR PERFORMANCE

3.1 Summary

There is now sufficient reactor performance experience with experimental and demonstration irradiations to warrant large scale use of recycled plutonium fuels in commercial LWR cores. In the event that plutonium recycle is approved, fuel loadings at the 1.15 SGR level could result in cores with as many as 40% of the fuel rods as mixed oxides. This amount of recycle plutonium is not related to any identified safety limitation.

The nuclear properties of mixed oxide fuels differ to some extent from UO_2 nuclear properties, notably the increased neutron cross section of the plutonium isotopes and the corresponding decrease in control rod worth. The altered nuclear properties can be accommodated in most cases by using various rod placement and enrichment schemes that make it feasible to design fuel assemblies that are interchangeable with the spent UO_2 assemblies they replace.

The materials properties and performance of mixed oxide fuels are in many cases indistinguishable from the corresponding UO_2 fuels, and in all cases the differences are small. Inhomogeneity of fissile material in physically blended mixed oxide fuel pellets could potentially cause a change in fuel performance, but the degree of homogeneity can be controlled during fabrication. Evaluation of any substantial differences between MOX and UO_2 fuel will be required.

The performance of a mixed oxide core will be similar to that of a UO_2 core under the normal steady state and load following conditions. However, changes in the nuclear and physical properties of mixed oxide cores will alter their behavior somewhat during transients and accidents. For example, the steam-line break accident consequences within a PWR mixed oxide core may require more reactivity control. A loss of coolant accident (LOCA), on the other hand, is generally less severe when compared with UO_2 core LOCA consequences. In any case none of the consequences of the postulated accidents will change enough to increase the public risk significantly.

Because no alterations of the core mechanical arrangement are required in order to accommodate the mixed oxide fuel rods, the thermal-hydraulic considerations are unaffected. Hence, this important aspect of core design does not require a generic review and accordingly is not included in the following generic review of plutonium utilization in LWR's.

3.2 LWR Plutonium Irradiation Experience

Plutonium fissioning in oxide fuels is not unique to the recycled plutonium fuels of current interest. A typical UO_2 core near the end of an equilibrium cycle, for example, at a core-averaged exposure of 20,000 MWd/MTU, will derive approximately 50% of its power from the fissioning of bred-in plutonium isotopes. Thus, in one sense, the use of plutonium as a fuel in LWR's does not represent a new situation.

Large numbers of experimental mixed oxide fuel rods have been irradiated in test facilities, and related experience with mixed oxide fuels has come from the liquid metal, fast breeder reactor (LMFBR) program. Two large scale experimental AEC sponsored programs designed specifically to study the utilization of recycled plutonium in LWR's have been carried out in the United States. The first of these, the Plutonium Utilization Program (PUP),[1] resulted in irradiation between 1961-1968 of more than 4,000 mixed oxide fuel rods in the plutonium recycle test reactor (PRTR),* constructed for this purpose. A companion facility, the Plutonium Recycle Critical Facility (PRCF), still in operation, has produced additional nuclear design information on mixed oxide fuels for LWR's. The second large experimental program was the Plutonium Recycle Experiment,[2] conducted in the Experimental Boiling Water Reactor (EBWR) in the years 1966 and 1967. In those years, nearly 1,300 mixed oxide fuel rods were irradiated.

The experimental mixed oxide work is important, but it is also important to summarize here the prototypical irradiations of mixed oxide fuels in nuclear power plants. See Chapter II. A large number of reports have been written on the mixed oxide fuel programs, hence the references in this section should be considered as key references to a much larger body of literature. Table IV C-3 shows recent demonstration plutonium irradiation experience in the United States.

Westinghouse conducted the Saxton Plutonium Project[3] for the AEC in the Westinghouse-built Saxton reactor. More than 600 mixed oxide fuel rods were irradiated in Core II and Core III of this reactor, with some of the rods reaching peak burnups in excess of 50,000 MWd/MTU. The Edison Electric Institute (EEI), with some AEC support, sponsored a Westinghouse Plutonium Recycle Demonstration Program[4] in the San Onofre (Southern California Edison) reactor. Four 14 by 14 demonstration assemblies containing Zircaloy-clad mixed oxide fuels were irradiated to peak burnups of about 25,000 MWd/MTU.

General Electric, under EEI sponsorship, is evaluating the utilization of plutonium in BWR's. The irradiations in this program[5] are being made in the Big Rock Point (Consumers Power) reactor. A total of 32 MOX fuel rods of various fuel pellet designs, in addition to three nine by nine bundles, each containing 68 annular pellet mixed oxide rods, have been irradiated, with peak burnups in excess of 30,000 MWd/MTU. In addition, General Electric has irradiated a few MOX rods in Dresden 1 (Commonwealth Edison) and currently has five MOX bundles in the Quad Cities 1 reactor.

General Atomic Company (formerly United Nuclear) and Commonwealth Edison joined in sponsorship of the Dresden Plutonium Recycle Demonstration Program.[6] Eleven prototype six by six mixed oxide assemblies were loaded in Dresden 1 in 1969. These fuel assemblies were of the island design, in which MOX rods are located in the central region of the bundle. Irradiation of some of these assemblies is continuing with peak burnups around 20,000 MWd/MTU.

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*See CHAPTER II.

Table IV C-3

RECENT LWR MIXED OXIDE FUEL EXPERIENCE IN THE UNITED STATES

FUEL VENDOR	REACTOR	TYPE	SIZE MWe	NUMBER OF PLUTONIUM RODS	PEAK POWER kW/ft	PEAK BURNUP MWd/MTU	PLUTONIUM IRRADIATION DATES
Westinghouse	Saxton	PWR	3	638	21	51,000	1966-72
"	San Onofre	PWR	430	720	7	25,300	1970-73
General Electric	Dresden 1	BWR	200	4	10	20,300*	1967-Present**
"	Big Rock Point	BWR	70	236	15	31,800*	1968-Present**
"	Quad Cities 1	BWR	800	48	15	8,000*	1974-Present
Gulf Nuclear	Dresden 1	BWR	200	99	15	19,800*	1969-Present**
Exxon	Big Rock Point	BWR	70	528	11	17,300*	1972-Present
Nuclear Fuel Services	Big Rock Point	BWR	70	292	10	16,400*	1973-Present

*Approximate burnup as of December 1975.

**Some of the mixed oxide fuels listed here have been discharged

Exxon Nuclear Company has supplied a number of reload assemblies containing mixed oxide fuel for the Big Rock Point reactor, and these assemblies currently have peak burnups in excess of 17,000 MWd/MTU. Consumer's Power has an additional 292 mixed oxide rods, supplied by Nuclear Fuel Services, in the Big Rock Point Reactor. Combustion Engineering, through its affiliate ALKEM, has available recent experience with MOX fuels from two German reactors; while Babcock & Wilcox's subsidiary, NUMEC, has fabricated mixed oxide fuels in large quantities for both the Saxton and LMFBR programs.

The irradiations, summarized in Table IV C-3, have confirmed the general similarity between UO_2 and mixed oxide fuels and have demonstrated that mixed fuel assemblies can be fabricated to be interchangeable with UO_2 fuel assemblies.

A further discussion of reactor experience with plutonium is given in CHAPTER II. Pointed out in the following paragraphs are the differences, where distinguishable, between MOX and UO_2-only fuels and any significance that can be attached to these differences.

3.3 Nuclear Design of Mixed Oxide Cores

The nuclear characteristics that are important to safety and performance of an LWR fueled with mixed oxide are discussed in the following paragraphs. These characteristics include power peaking effects, control rod worths, control requirements, reduction in the delayed neutron fraction, and the importance of ^{135}Xe on stability. The various reactivity coefficients are also discussed, and some comments on calculational methods and the adequacy of available data are provided. Portions of the discussion apply to partial MOX fuel loadings up to approximately half of the fuel rods in the core. Larger loadings may require additional fuel and control management, which could represent a special application and hence are not considered in this discussion.

3.3.1 Effects of ^{135}Xe on Reactor Spatial Stability and Control

A typical PWR, with fuel rods 12 feet long or longer, becomes less stable in the axial direction after 1/2 to 2/3 of the first cycle burnup. This is caused by the spatial buildup and burnout of ^{135}Xe, which has very large thermal neutron cross sections, and the time lag between the formation of ^{135}I and its decay into ^{135}Xe. Although ^{239}Pu has a slightly higher fission product yield of ^{135}Xe than ^{235}U (6.6% vs. 6.4%), the fact that the thermal neutron flux in a core containing mixed oxide fuel assemblies is smaller than the flux in an all UO_2 core means that less of the ^{135}Xe is destroyed by neutron absorption and more of it is destroyed by radioactive decay. Hence, shifts in power density have a much smaller local reactivity effect in mixed oxide cores than they do in UO_2 cores. The degree to which stability is improved by MOX fuel is dependent on the fraction of the core that contains mixed oxides. With a large fraction of mixed oxides present, a 12-foot long reactor may be considerably more stable than a UO_2-only core. Benefits from improved stability with MOX may make load following less difficult; the need for part-length rods in PWR's and the need for careful control rod operation to maintain a stable power distribution during load changes may be reduced from that of the UO_2-only core.

3.3.2 Reactivity Coefficients

The effects of reactivity coefficients on the operation and safety of the reactor are discussed in paragraph 3.3.5. This section discusses the physical phenomena that give rise to the differences between mixed oxide and UO_2 coefficients.

Moderator Temperature Coefficients

The major factors that affect the moderator temperature coefficient of LWR's are the decrease in water density as temperature increases and the shift in the thermal neutron spectrum with temperature. As the moderator temperature rises, the moderator density decreases. A decrease in moderator density primarily increases the absorption of epithermal neutrons, more neutrons are captured in the absorption resonances, and tends to shift the thermal neutron spectrum to higher energies. As neutrons slow down in the LWR lattice, a substantial fraction is captured in the neutron absorption resonances of various isotopes. In a UO_2 lattice, the major resonance absorber is ^{238}U. Because ^{238}U does not fission in the epithermal range, the effect on reactivity is negative. If nonfissionable isotopes such as ^{240}Pu and ^{242}Pu are present, the effect is to increase resonance absorption and make the moderator coefficient more negative.

The reactivity effects due to changes in the thermal neutron spectrum are complex. As illustrated in Figure IV C-16, the various plutonium isotopes have large overlapping resonances in the 0.1 eV* to 5 eV range. Although the peak of the thermal neutron spectrum in a typical LWR is in the range 0.035 eV to 0.04 eV, the distribution of thermal neutrons extends into the range 0.1 eV to 1.0 eV. As the temperature increases the neutron spectrum shifts so that more neutrons are at higher energies. This causes more neutrons to be absorbed in nonfissionable ^{240}Pu. Also, more neutrons are absorbed in the 0.3 eV resonance of ^{239}Pu. As shown in Figure IV C-17, absorption in this resonance produces significantly fewer neutrons per absorption than at lower energies. Both of these effects make the moderator temperature coefficient more negative. The moderator coefficient in a reactor at equilibrium with self-generated plutonium is approximately 5.0×10^{-5} per °F more negative than in an equilibrium UO_2 reactor.

If soluble poison is used in the reactor, the poison concentration will decrease as the temperature is increased, because the moderator density is decreasing. This produces a positive component to the moderator coefficient. There would be little difference between UO_2 and mixed oxide cores for a given excess reactivity being controlled by soluble poison.

Void Reactivity Coefficients

Because PWR's have very low void fractions, the use of some mixed oxide fuel assemblies would make little difference in steady state operation. For BWR's, where the void percent V is large, the effect on the void coefficient is important. Some

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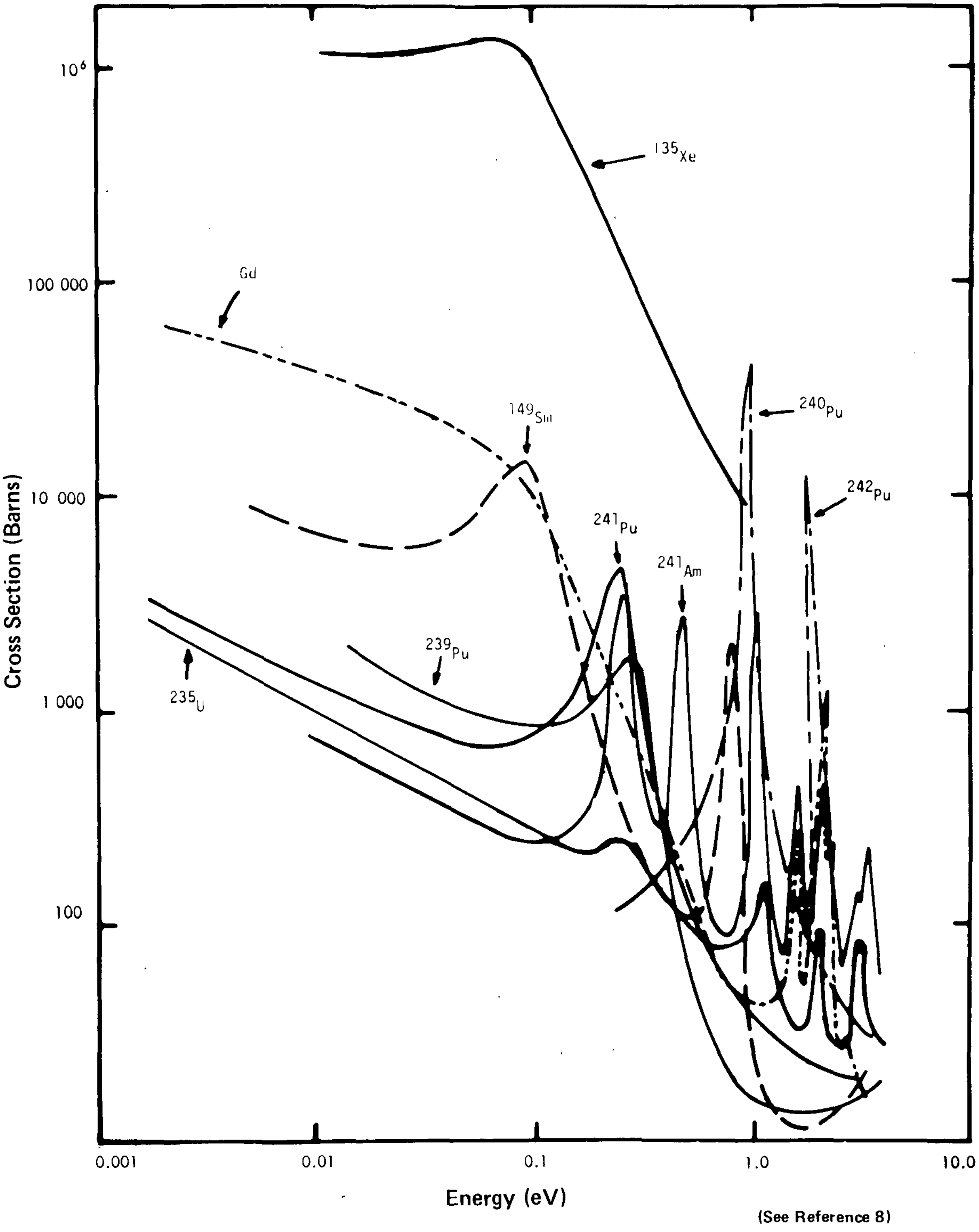
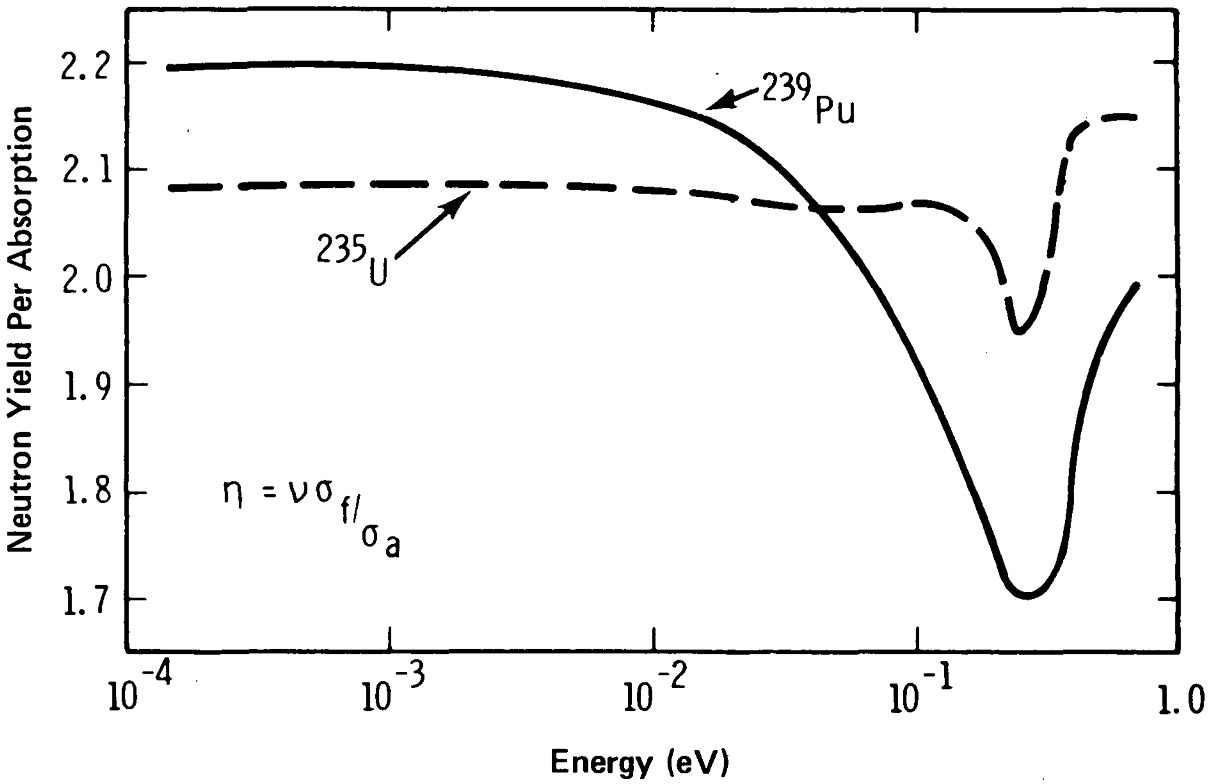
*eV = electron volt



(See Reference 8)

Figure IV C-16 Cross Section Structure Below 3 eV



(See Reference 8)

Figure IV C-17 Neutron Yield Per Absorption for ^{235}U and ^{239}Pu typical values of $(\Delta k_{eff}/k_{eff})/\Delta V$ for a BWR are -9×10^{-4} per percent for UO_2-only fuel and -10×10^{-4} per percent V for a self-generated mixed oxide fuel. Changes of this magnitude are significant.

Doppler Coefficient

As the fuel temperature is increased, the vibrational energy of the fuel atoms is increased. This effectively broadens the resonances leading to greater neutron absorptions. The result is a negative fuel temperature coefficient called the Doppler coefficient. The primary Doppler effect occurs in the fertile isotopes that have even numbers of both neutrons and protons in their nuclei. The effect is small in fissile isotopes, because resonances have low cross sections. In a UO_2-only fueled core, the effect is primarily due to ^{238}U, but as other isotopes are generated, ^{236}U, ^{240}Pu, and ^{242}Pu become important. In general, for LWR's containing mixed oxide fuel, the Doppler coefficient is about 10% more negative. For example, typical values for a BWR are -1.40×10^{-5} $(\Delta k_{eff}/k_{eff})/°F$ for a UO_2-only design and -1.55×10^{-5} $(\Delta k_{eff}/k_{eff})/°F$ for a design using self-generated mixed oxide fuel.

3.3.3 Local Power Peaking

The isotopes ^{239}Pu and ^{241}Pu have fission cross sections that are approximately twice that for ^{235}U in a typical LWR thermal neutron spectrum. Because approximately the same number of fissile atoms for a given mass of fuel is required to attain the same fuel burnup with either ^{235}U or a mixed oxide, severe power peaking might be expected if MOX fuel were isolated in a region of all UO_2 fuel. However, the mean free path of thermal neutrons in LWR lattices is short. Because most of the neutrons that reach thermal energies are absorbed in fissile material, the amount of moderator in the vicinity of a fuel rod determines the number of neutrons that are thermalized near, and absorbed by, a fuel rod. In tight BWR lattices this factor can dominate more than the difference in uranium and plutonium fission cross sections. BWR fuel vendors, in general, propose an island design concept wherein UO_2 fuel rods are adjacent to the large control rod water slots to minimize power peaking in the internal MOX fuel rods, the island. The large number of neutrons slowing down in the water slots would cause undesirable power peaking in the MOX rods if those rods were placed on the periphery of the fuel bundle. In PWR's, which have relatively uniform lattices, some vendors' calculations indicate that an acceptable local power peaking distribution can be obtained with two plutonium concentration zones in the mixed oxide fuel assemblies, which are interspersed with all UO_2 fuel assemblies. Other vendors may use as many as four plutonium concentrations and a region of all UO_2 fuel on the outside of the fuel assembly.

Measurements of local power peaking in reactors containing mixed oxide fuels may involve greater uncertainties than in cores that contain only UO_2 fuels. Calculational uncertainties, introduced in extrapolating from a flux-thimble measurement in a UO_2 assembly to the hot rod in a mixed oxide fuel assembly nearby, may introduce several percent more error than is currently present. This difficulty in normalizing relative powers in UO_2 and mixed oxide rod gamma scan measurements has been encountered in the analysis of some critical experiments.[7] In the techniques used for measuring relative power densities by utilizing gross gamma scan data, the difference in the gamma spectrum and the ratio of gamma decay heat to beta particle decay heat must be known. Uncertainties in these data, as well as other experimental uncertainties, could lead to experimental errors a few percent larger for MOX lattices than measurements in UO_2-only critical experiments. Each set of measurements, therefore, must be evaluated carefully when comparisons are made to calculations.

3.3.4 Control Rod Worth

It is feasible to design the core of a reactor utilizing recycled plutonium fuel so that its safety and performance characteristics will approach the characteristics of an all UO_2 core. To accomplish this, it may be necessary to limit the number of fuel rods in the core that have plutonium as the major fissile material.

Because the fission cross sections of ^{239}Pu and ^{241}Pu in a typical LWR are approximately twice that of ^{235}U, the thermal neutron flux in a mixed oxide fuel assembly would be approximately half that in an all UO_2 fuel assembly, if both contained the same fissile content and operated at the same power level. When a control rod is inserted in a mixed oxide fuel assembly, the lower neutron flux results in fewer neutrons being absorbed in the control rod relative to the fuel, and thus the reactivity worth of the control rod is reduced. The relative reduction in worth is dependent on the moderator temperature, because the ratio of ^{239}Pu to ^{235}U fission cross section varies strongly with moderator temperature (1.28 at 68°F vs. 2.62 at 590°F). At normal operating temperatures, this loss can be 25-30% of the total control rod worth.

Limiting the quantity of mixed oxide rods in the core is feasible and can be accommodated by the way that different types of fuel assemblies are located or reshuffled in the fuel management scheme. For example, it is possible to avoid use of control rods in mixed oxide fuel assemblies in PWR's and to locate them away from islands of mixed oxide rods in BWR's, so that control rod worths are not substantially reduced. (If too large a fraction of the core contains mixed oxides, flexibility is lost and substantially more control rods may be required to provide the necessary control and safety margins.) The model reactor described in Section IV C-4.0 contains a small enough amount of plutonium that control requirements and fuel management flexibility are assured for most licensed LWR's.

3.3.5 Control Requirements

Reactivity control requirements are of three types:

- Those required for normal daily operation
- Those required for long term effects such as fuel burnup
- Those required to limit the effects of anticipated operational transients and accidents

The effect of mixed oxide fuel on control requirements is strongly dependent on the fuel management scheme used. With the assumption that the fraction of mixed oxide fuel rods in the core is small enough to allow flexibility in fuel management, as is the case with the model reactor, the effects on control requirements for normal daily operation in contrast to uranium-only cores are summarized here.

- The Doppler effect, because of changes in fuel temperature, is increased.

- The moderator temperature coefficient is more negative. This increases control requirements for power changes and for going from hot standby to cold shutdown status in BWR's. PWR's use soluble poisons to provide the hot standby to cold shutdown requirements, but control rod requirements for reactor trip (scram) are increased.

- Reactivity changes due to moderator voids (bubbles) result in an increased negative reactivity. This is a small effect in PWR's, but is a significant effect in BWR's.

- PWR control rod insertion requirements to follow load changes may be somewhat increased because of the more negative power coefficients, especially at end of cycle where the capacity of the soluble boron system to compensate for core reactivity changes is limited. The increased stability due to ^{135}Xe effects will reduce the requirements on the control system. The control rod insertion restrictions to limit the consequences of a control rod ejection accident will not be changed appreciably.

- The reactivity effect due to ^{135}Xe is reduced.

- The worst stuck-rod control requirement is affected by fuel loading patterns; it may be unchanged or even reduced.

Long term reactivity changes occur, because of depletion of fissile material and buildup of neutron absorbing fission products in the fuel. In general, the same target burnup can be achieved by adjusting the amount of fissile material in the different types of fuel rods, and control requirements will not be appreciably affected. The rate of change of long term reactivity with burnup for mixed oxide fuel is much smaller than for UO_2 fuel. This is because ^{240}Pu is an excellent fertile material and upon neutron capture produces fissile ^{241}Pu.

The soluble boron worth is less in PWR cores containing mixed oxide fuel assemblies than in UO_2 cores. Because the reactivity requirements for burnup are somewhat less in a mixed oxide core, the soluble boron content compared to burnup is not much different. In BWR's, the flatter burnup characteristics of mixed oxide fuels are a definite advantage, but this is offset to a large extent by the increased control rod requirements due to more negative moderator temperature and void reactivity coefficients.

Control requirements to limit anticipated operational occurrences and accidents are primarily related to the reduced soluble boron worth in a PWR core that contains mixed oxide fuel assemblies. Both PWR's and BWR's depend on soluble boron injection systems as the second independent control system, which is capable of bringing the reactor subcritical in the cold condition. The slower time response of this type of system with a mixed oxide core may be compensated for by increasing the initial charging rates of the boron injection system. In the PWR steam break accident the larger moderator temperature and void reactivity coefficients with MOX fuel may require more control rod worth or faster acting boron injection systems to insure the consequences of this accident are acceptable.

3.3.6 Delayed Neutron Fraction and Prompt Neutron Lifetime

The delayed neutron fraction and prompt neutron lifetime are important to the analysis of rapid transients. As shown in Table IV C-4, the delayed neutron fractions (β) for the plutonium isotopes, especially for ^{239}Pu, are less than that of ^{235}U.

Comparing practical designs of all-UO_2 cores with cores containing mixed oxides reveals that β for the UO_2 design is 0.0074 at the beginning of life compared with 0.0065 for the GESMO model reactor. At high exposure, the difference diminishes to 0.0054 for β for the UO_2 core compared with 0.0051 for the model core. Thus near the end of a LWR fuel cycle the differences between MOX cores and UO_2-only cores diminish noticeably.

Table IV C-4

DELAYED NEUTRON FRACTIONS

ISOTOPE	β
^{235}U	0.0067
^{238}U	0.0164
^{239}Pu	0.0022
^{240}Pu	0.0029
^{241}Pu	0.0054
^{242}Pu	0.0051

The higher absorption cross sections of plutonium decreases the prompt neutron lifetime in plutonium recycle cores. Typical values of this lifetime in PWR's are 20 to 25 μsec for UO_2-only cores compared with 16 μsec for the model reactor. In BWR's typical values are 33 μsec for UO_2 and 30 μsec for the model reactor. Even for large reactivity accidents, changes of this magnitude are acceptable.

3.3.7 Calculational Methods and Adequacy of Data

A comprehensive review of the status of experimental work on plutonium, both in operating reactors and in critical experiments, has been given in a recent paper by Uotinen, et al.[8] The paper, with its 130 references, also discusses problem areas in calculational techniques.

Experimental data for mixed oxide cores are not as extensive as for UO_2 cores. Critical experiments have been small in size, and the larger neutron leakage effects introduce additional uncertainty in the data. Also, the limited experimental data on localized quantities, such as fuel cell neutron reaction rates for the various isotopes, make it more difficult to determine whether calculations are in agreement or not. In the energy region below 3 eV, the complicated cross section structure shown in Figure IV C-16 makes it difficult to insure that compensating effects are not obscuring errors in analyzing the experiment. For these reasons, there must be more conservatism in MOX core design calculations than UO_2 core design calculations.

In order to calculate quantities such as the moderator temperature coefficient of reactivity accurately, there is need for a neutron thermalization computer technique that adequately treats the complicated resonance cross section region below 3 eV. It is desirable to have the thermal cutoff--between the fast and thermal calculations--well above the 1.05 eV resonance of ^{240}Pu. Commonly used codes such as THERMOS have a weakness in that the number of groups available ($\sim$ 35) does not give sufficient resolution to treat resonances properly. In principle, it is possible to generate libraries for use in integral transport theory codes with any number of thermal groups. This is not normally done in the industry for the heterogeneous lattices. Another technique is to perform Monte Carlo calculations in the range 0-3 eV and then to compute correlation factors for use with codes that have a thermal cutoff of 0.625 eV.

In determining effective fast group cross sections, a calculational method that explicitly determines the self-shielding and Doppler broadening in the ^{240}Pu and ^{242}Pu resonances is needed. A typical method is that of Nordheim,[10,11] which has been incorporated in several fast-neutron-spectrum codes such as GAM-II.[12]

Particle self-shielding effects in mixed oxide fuels are probably unimportant from a nuclear standpoint because most vendors are considering fuels in which all but a few volume percent of the PuO_2 particles are smaller than 20μ to 50μ in diameter.

Uncertainties in the calculation of safety related quantities such as reactivity coefficients, control rod worths, and power distributions can be accommodated in the design. Some increase in design margins may be necessary to allow for a possible increase in the uncertainty of core parameters in a MOX core and may involve economic penalties. Therefore, continued improvement in the data base and calculational techniques is well justified.

3.4 Characteristics of Mixed Oxide Fuels

In the following sections the physical properties and performance characteristics of a nominal 5% PuO_2 - 95% UO_2 mixed oxide fuel (subsequently referred to as 5% Pu fuel) are compared with those of UO_2 fuel. Because the average MOX fuel rod will contain about 5% PuO_2, this value has been chosen as an example. Most of the properties and characteristics discussed exhibit small deviations from the UO_2-only cases. Several of these performance characteristics, however, represent essential differences between UO_2 and mixed oxide fuels; and, in particular, microscopic plutonium inhomogeneity is uniquely associated with recycled fuels. The subject of plutonium inhomogeneity is, therefore, reviewed in greater detail than the other items. A brief description of the fabrication of mixed oxide fuel pellets is contained in CHAPTER IV, Section D.

3.4.1 Physical and Mechanical Properties of $(U,Pu)O_2$

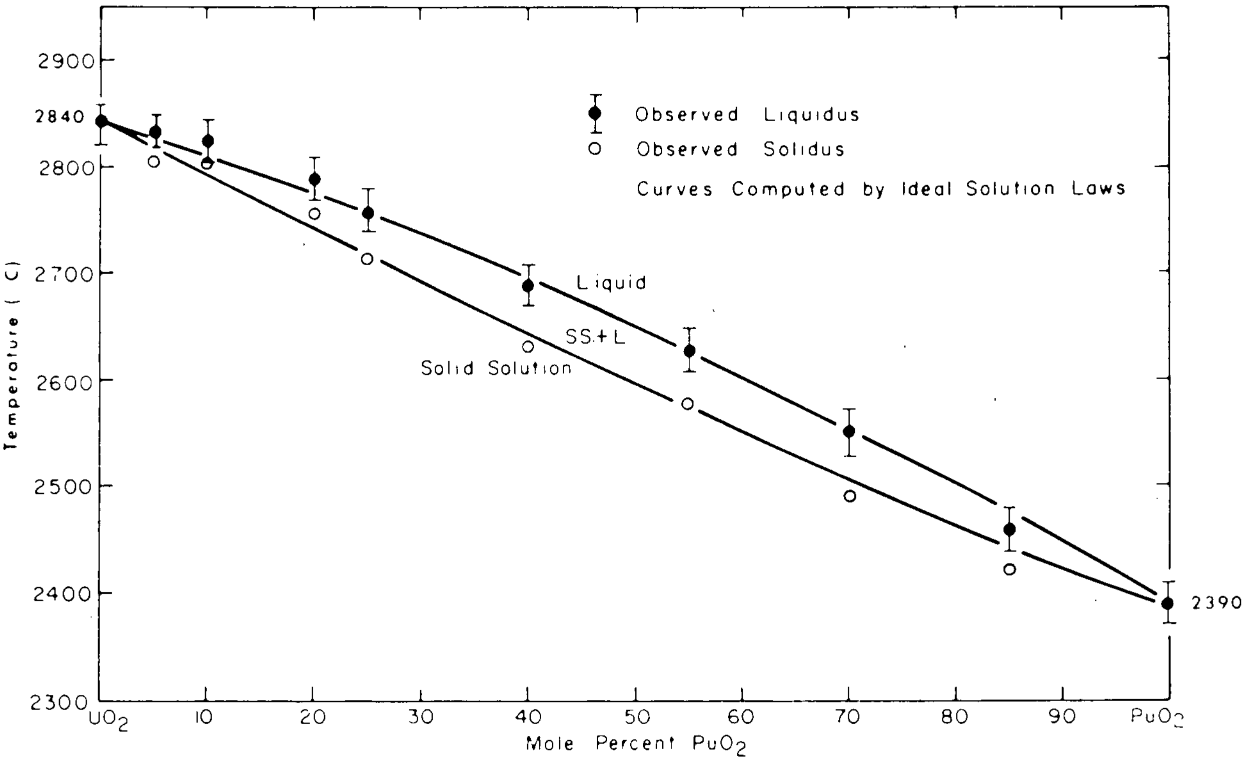
PuO_2 like UO_2 has a face-centered cubic fluorite lattice structure,[13,14] with a room temperature lattice constant of 5.3960 Å for PuO_2 compared with 5.4704 Å for UO_2. Because of the smaller lattice parameter and slightly greater mass, PuO_2 has a theoretical density of 11.46 g/cu.cm, compared to the UO_2 density of 10.96 g/cu.cm. For a 5% Pu mixed oxide fuel, the theoretical density will be only 0.2% greater than the density of UO_2, and this change is generally less than the uncertainties in any measured value.

Because of their ionic and crystalline similarities, UO_2 and PuO_2 form a complete solid solution ranging from pure UO_2 to pure PuO_2. The melting point of stoichiometric PuO_2 is approximately 2,390°C,[14,15] which is significantly lower than the melting point of stoichiometric UO_2 at approximately 2,840°C.[13,15]

The phase diagram of Lyon and Baily[16] is shown in Figure IV C-18. This diagram indicates a reduction in the melting point of about 25°C for a 5% Pu fuel from that of a UO_2 fuel. The lower melting point of mixed oxide fuels must be accounted for in fuel rod thermal design.

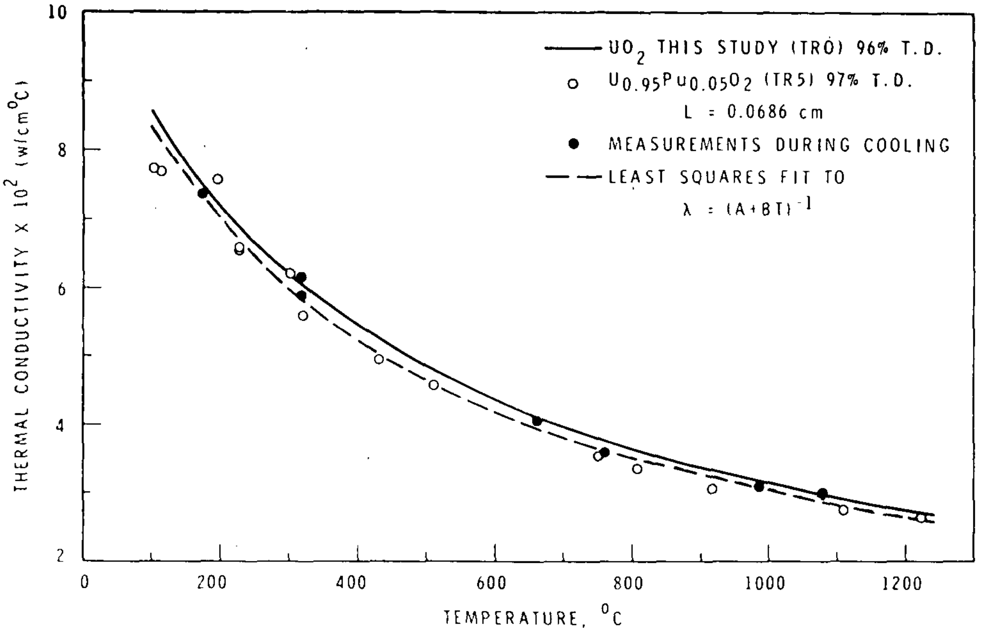
The addition of PuO_2 to UO_2 has been found to lower the thermal conductivity of the oxide fuel. At 500°C, the measurements of Gibby[17] indicate a reduction in thermal conductivity of about 5% for 5% PuO_2 fuel compared with UO_2 fuel, as shown in Figure IV C-19. Calza-Bini et al.[18] have found that the integral conductivity $\int^{melt} k dT$, which is a measure of the power at which fuel centerline melting takes place, is reduced about 5% for a 4% mixed oxide fuel. Therefore, for a 5% PuO_2 fuel, it can be expected that thermal conductivity will be reduced about 5% compared with UO_2, independent of the temperature.

The linear thermal expansion coefficient of stoichiometric PuO_2 has been measured by Tokar et al.[19] and found to be very close to measured values for UO_2. Figure IV C-20 shows the UO_2 data of Brett and Russell[20] and Conway et al.[21] along with the PuO_2 data. No systematic differences between the UO_2 and PuO_2 data are apparent. Therefore, for a 5% Pu mixed oxide, it is expected the thermal expansivity will be the same as for UO_2.



(See Reference 16)

Figure IV C-18 Solid-Liquid Phase Diagram for the UO$_2$-PuO$_2$ System



(See Reference 17)

Figure IV C-19 The Thermal Conductivity of $U_{0.95}Pu_{0.05}O_2$ as a Function of Temperature

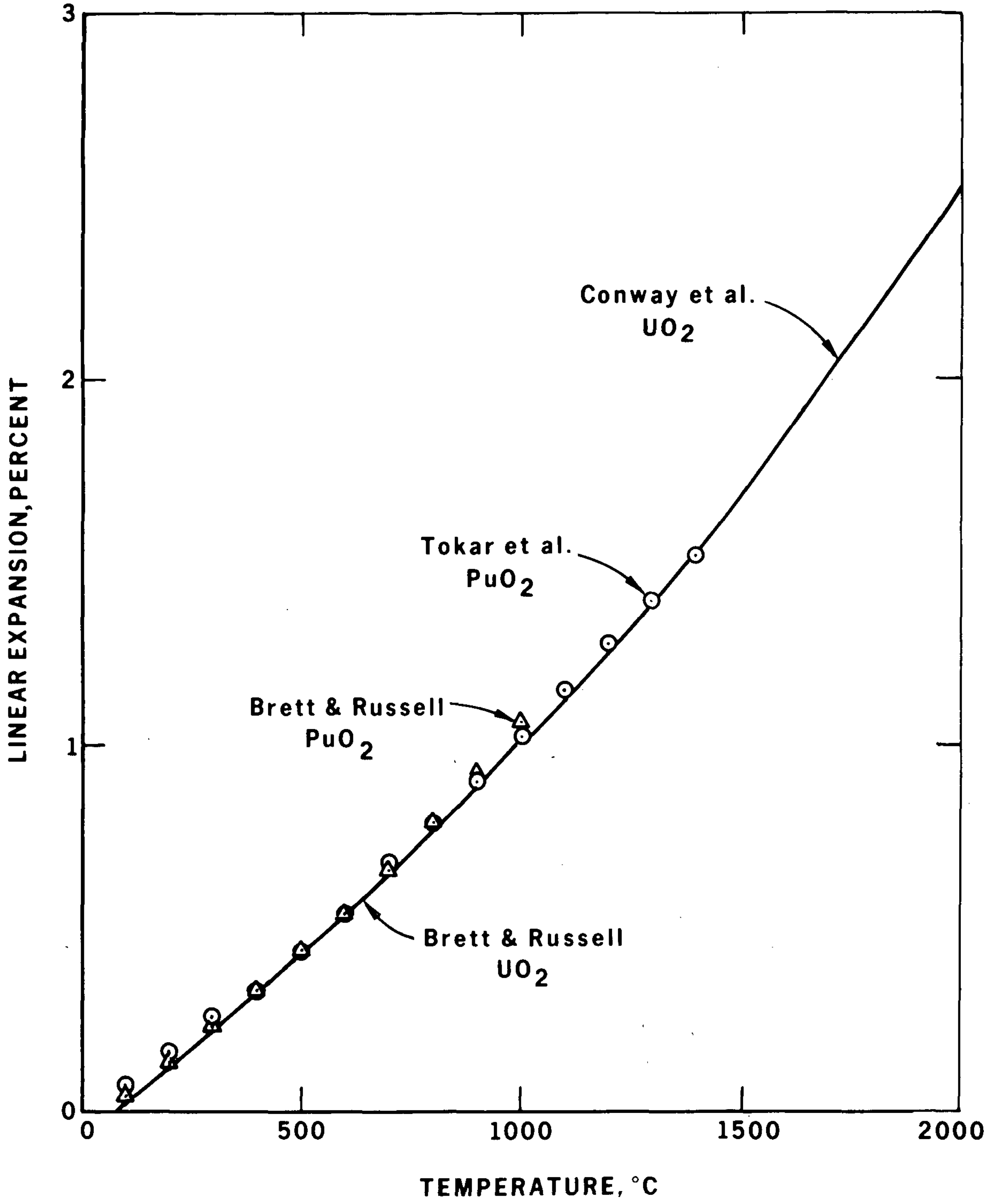


Figure IV C-20 Linear Thermal Expansion of Stoichiometric UO_2 and PuO_2

Recent correlations of enthalpy (stored energy) data have been reported by Chasanov et al.[22] for UO_2 and by Gibby et al.[23] for 20-25% Pu mixed oxides (breeder fuels). The high temperature enthalpy of a fuel is important in accident analysis, and at high temperatures these two correlations agree very closely. For example, at the 2,390°C melting point of PuO_2, both correlations give enthalpies of 54.42 kcal/mol. The temperature derivative of an enthalpy function gives the specific heat, and at 2,390°C the correlations give specific heat values of 34.7 cal/mol°K for UO_2 and 33.2 cal/mol°K for high plutonium content MOX (20-25%). It is clear, therefore, that the enthalpy and specific heat of a 5% Pu MOX fuel are very similar to the corresponding UO_2 quantities.

The brittle fracture strength of UO_2 and of 20% Pu mixed oxide (breeder fuels) has been measured in four point bending by Roberts and Wrona.[24] They found the fracture strength to be lower for the mixed oxide, but conclude that this relative weakness was caused by fabrication flaws and impurities in the mixed oxide specimens. Recycled plutonium fuels may contain more impurities than fresh UO_2 fuels, and the altered fabrication process for plutonium fuels may introduce additional fabrication imperfections, as evidenced by Roberts and Wrona. It is conceivable that mixed oxide fuels will have a lower fracture strength than UO_2 fuels. Although none of the current methods used to analyze fuel rod thermal performance utilizes a fracture stress parameter, it is thought that any reduction in strength that would lead to fuel cracking would improve the thermal performance of the fuel by improving the fuel-cladding-gap thermal conductance.

The rate of plastic flow under stress for mixed oxide fuels has been studied as a function of plutonium content by Evans et al.[25] As the plutonium content is increased, a definite softening of the material is reported. From the data it can be seen that a 5% increase in plutonium content will in general increase the creep rate about 20%. However, data from another source shows a smaller change. As in the case for fracture strength, creep rate is not used in current thermal performance analyses and so far no problems have been identified that would result from using a fuel with greater plasticity. Increased plasticity may, in reality, be beneficial by reducing pellet cladding interaction.

3.4.2 Performance Characteristics of MOX Fuel and UO_2 Fuel

3.4.2.1 Homogeneity

In UO_2 fuels the fissile material ^{235}U is distributed homogeneously, on an atomic scale, among the nonfissile isotopes. Most mixed oxide fuels, on the other hand, will be made from a blending of UO_2 and PuO_2 powders that leads to a less homogeneous fuel because the PuO_2 powder particles contain nearly all of the fissile material and the UO_2 powder particles contain nearly all of the nonfissile material. The physical mixing, or blending, of plutonium recycle fuels is likely to be a common fabrication practice because of the proposed restrictions[26] on shipment of plutonium nitrate solutions and economic considerations that make it desirable to handle plutonium separately from uranium as much as possible in processing.

Although individual PuO_2 powder particles, generally < 50-μm diameter, will result in a certain inhomogeneity in the fuel, there is concern that larger particulates can form during powder mixing by the agglomeration of many individual PuO_2 powder particles. Agglomerates larger than 500 μm in diameter have been reported.[27] The existence of such gross fissile-atom agglomerates might affect fuel rod performance, and, therefore, plutonium homogeneity should be controlled and the effects of a manufacturer's degree of fissile-atom inhomogeneity must be assessed.

There are several phenomena related to plutonium particle size that may affect the performance of mixed oxide fuels, including a change of fuel reactivity, degradation of the Doppler coefficient, the creation of local power spikes, and the possibility of localized cladding failure during a large transient (accident). These effects are discussed in the following paragraphs.

3.4.2.2 Self-Shielding Effects on Fuel Reactivity

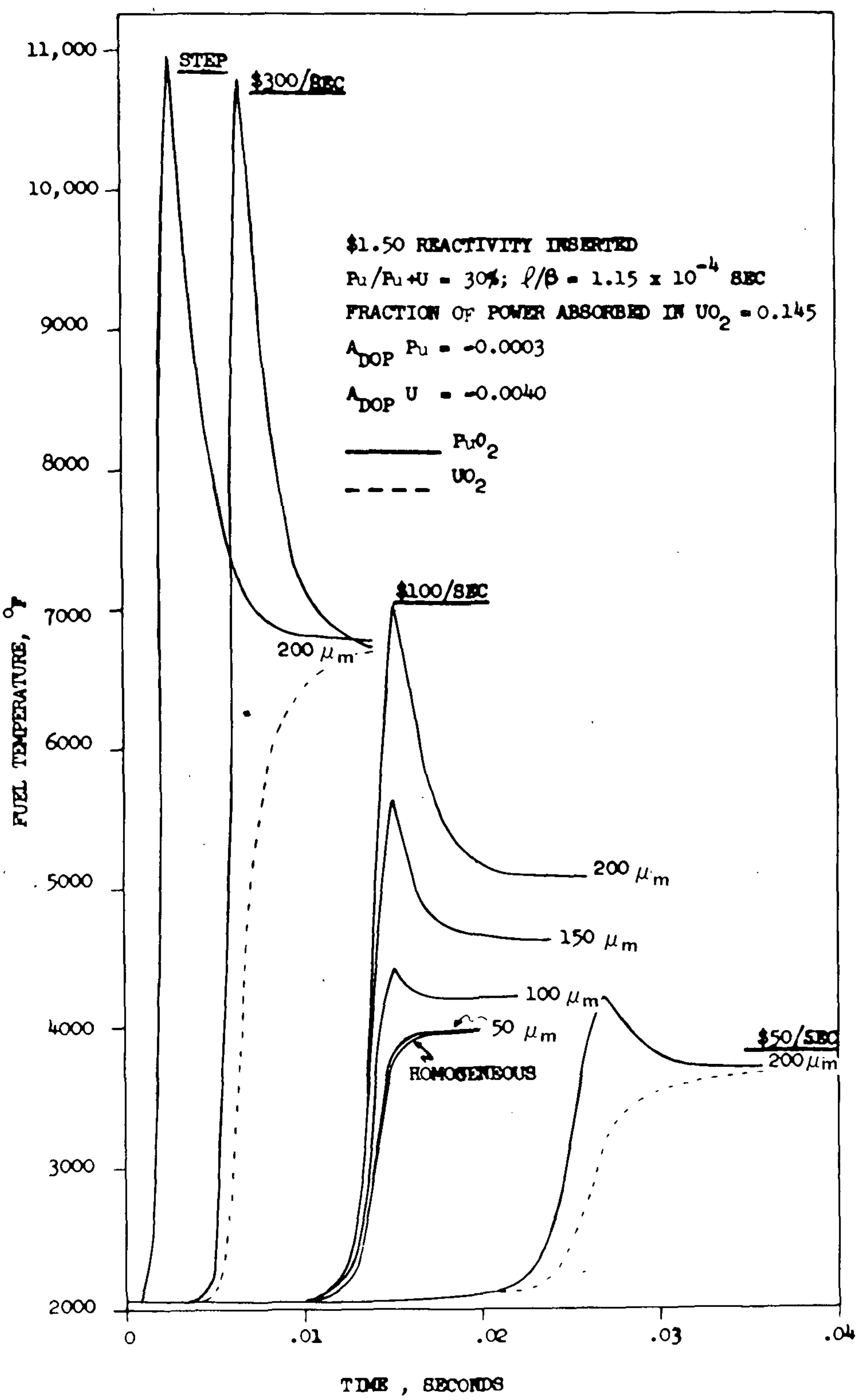
When the fissile material is distributed in discrete high concentration particles, rather than homogeneously, the self-shielding of fissile plutonium will result in a loss of reactivity. This effect will diminish as burnup progresses. On the other hand, self-shielding of the 0.3-eV ^{239}Pu resonance, which has a low neutron yield per absorption, will increase reactivity. A positive increase can also occur from increased self-shielding of the 1.05-eV ^{240}Pu resonance. The combination of these effects can cause either a positive or negative reactivity effect, depending on the ^{240}Pu concentration. Calculations by the fuel vendors indicate that fuel with 20-μm to 50-μm particles, which contain 20% to 25% of their plutonium inventory as ^{240}Pu, will exhibit a reactivity that is approximately 0.1% lower than homogeneous fuel. This is an insignificant difference compared with the excess reactivity (on the order of 10%) built into a typical core.

3.4.2.3 Doppler Coefficient

During a power transient, the inhomogeneous distribution of fissile material will cause a time delay in the ^{238}U temperature change and hence slow down Doppler feedback. Bailey et al.[28] have investigated this effect analytically for fast reactors, and their results are shown in Figure IV C-21, where the average temperatures of the PuO_2 matrix are shown as functions of time. Their calculations show that, for a given reactivity insertion rate, the fuel temperature response is not affected by particle size up to approximately 75 μm. However, for substantially larger particles, fuel temperatures increase with particle size. Although reactivity insertion rates in an LWR are lower than in an LMFBR, the ratio of UO_2 volume to PuO_2 volume is increased, so that large PuO_2 particles may have some effect on Doppler feedback in LWR's. Evaluation of particle size effects on Doppler feedback will therefore be required in the analysis of potential accidents.

3.4.2.4 Local Power Spikes and Hot Spots

Large agglomerates of fissile material located near the fuel cladding will cause local power spikes, which may create local hot spots on the cladding. The effect of a



(See Reference 28)

Figure IV C-21 Fuel Temperature Response to Reactivity Insertions for Various PuO_2 Particle Sizes

400-μm plutonium particle located on the surface of a pellet adjacent to the cladding has been estimated by Westinghouse[29] using two and three-dimensional analytical thermal models. Results from the TAP three-dimensional code show that the power spike never exceeds 20%. As would be expected, the two-dimensional codes give larger spikes, although the spikes are very localized, and decrease to below 20% a short distance (80 mils) from the particle. The thermal conductivities of the fuel, gap, and cladding, however, will dissipate this heat spike and reduce its effect on the local thermal-hydraulic performance of the rod.

To measure the effects of local heat-flux spikes, Hill et al.[30] performed departure from nucleate boiling (DNB) water tests on a 14-foot nonuniformly heated four by four rod bundle. A 20% heat-flux spike was generated in three adjacent rods over a 6-inch length at the axial location where DNB was most likely to occur. Additional tests were run under the same conditions without the power spiked rods. Figure IV C-22 shows the predicted versus measured DNB heat flux for the spiked bundle and the unspiked bundle. The results indicate that the measured spike effect is so small that it lies within the repeatability of the DNB measurement. Because the heat-flux spike for a 400-μm PuO_2 particle would be much less severe than the one used in the rod bundle tests, it can be concluded that large particles should not have an adverse effect on the DNB heat flux during steady state operation. Nevertheless, large particles could have an effect on cladding integrity during a severe power transient, reactivity initiated accident. Two fuel rod failure thresholds are used in analyzing these design basis accidents.[31] The first, and less severe, is based on incipient failure wherein the gaseous fission products are released through a breach in the cladding. This threshold is used to calculate the number of failed rods during the postulated accident so that radiological releases can be evaluated. The second threshold concerns the expulsion of high temperature fuel into the coolant, sometimes referred to as prompt fuel dispersal. This higher threshold is a design limit above which a coolable fuel geometry cannot be assured.

Tests of the effects of reactivity initiated transients have been conducted in the Special Power Excursion Reactor Tests (SPERT) and Transient Reactor Tests (TREAT) programs to establish threshold energies for incipient failures and for prompt-dispersal failures of UO_2 fuel pins. Limited SPERT tests[32] have been performed on unirradiated rods containing mixed oxide fuel. The average plutonium particle size in these prototypical fuels was 40-50 μm; the maximum was 200-300 μm. Both annular and solid pellet geometries were investigated. The incipient-failure energies for mixed oxide pins containing annular or solid pellets were consistent with values for homogeneous UO_2 fuel pins.[33] The values for mixed oxide fuels were in the energy range of 225 to 275 cal/g. A conservative value of 170 cal/g is currently used by General Electric for this threshold.

Transient tests in SPERT have also been conducted on unirradiated UO_2 pellets containing very large (550-μm) discrete PuO_2 particles located at different spatial positions in each pellet.[34] Results show that large particles adjacent to the cladding reduce the incipient cladding-failure threshold energy to about 213 cal/g, slightly

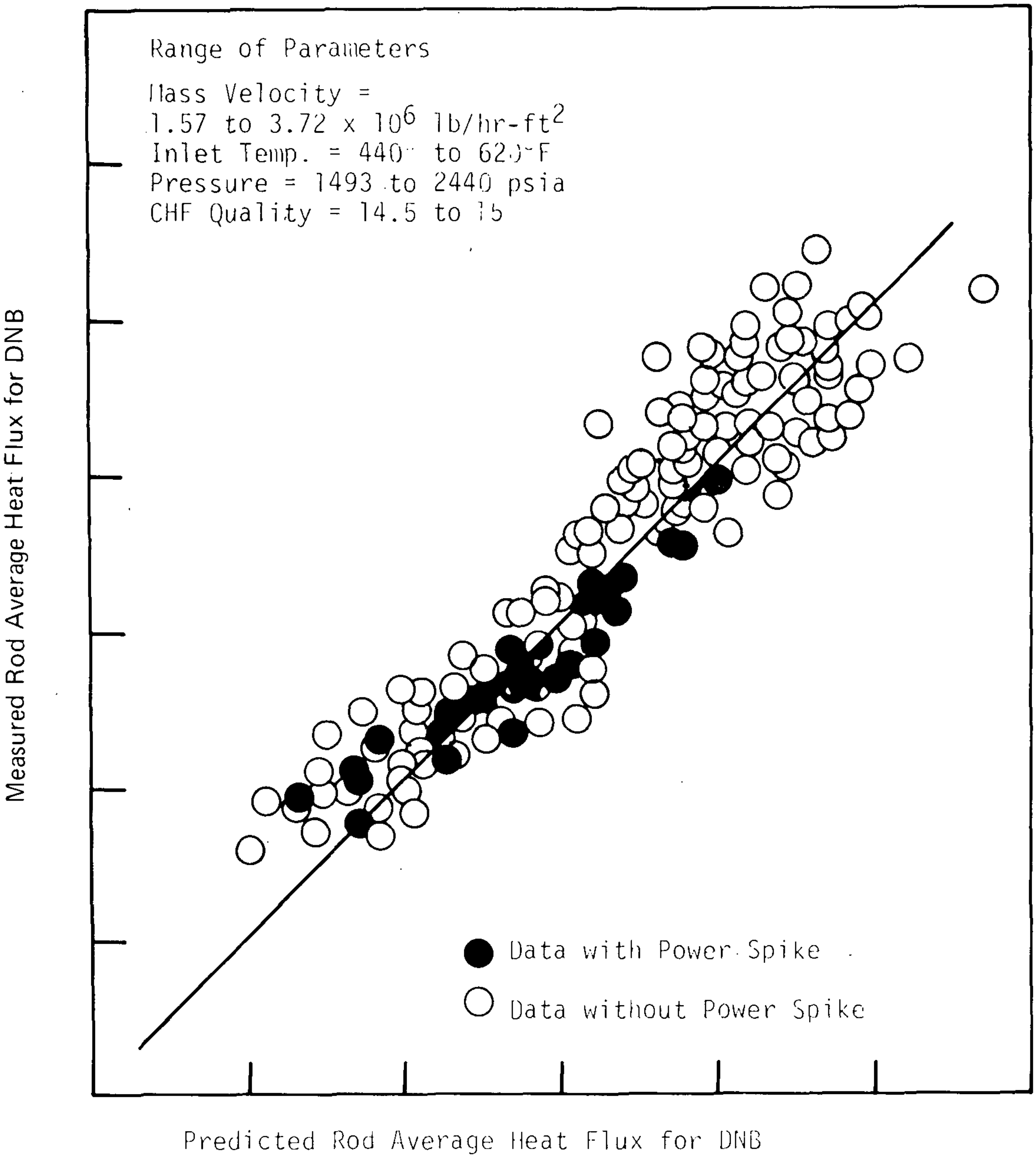


Figure IV C-22 Comparison of DNB Data With Power Spike and Without Power Spike (Predictions for the Power Spike Tests Were Made with the No-Spike Model) below the values obtained for homogeneous fuel pins. Failures in these grossly inhomogeneous fuels were caused by the localized melting and perforation of the cladding and resulted in expulsion of the PuO_2 particle, which was located close to the surface. Transient tests at energy levels high enough to cause prompt fuel dispersal were not performed with these inhomogeneous fuels.

General Electric[35] is studying the failure probability for fuels containing large PuO_2 particles, and transient tests on such fuels have been performed in a TRIGA reactor to correlate the failure to the peak neutron flux and the particle diameter. General Electric has concluded that the perforation threshold for fuels with large particles is dependent upon the peak neutron flux and not the total energy deposited.

It is evident that under some circumstances the incipient cladding failure threshold can be lowered by the presence of large PuO_2 particles in the fuel. An evaluation of this effect will therefore be required in the individual reactor safety analysis for the particular fuel homogeneity characteristics under consideration.

The prompt-dispersal energy threshold, for annular-pellet prototypical mixed oxide rods, was also investigated[32] and found to be between 329 and 414 cal/g, which is slightly above the threshold for homogeneous UO_2 fuel pins. High energy tests to produce fuel dispersal with solid pellets were not performed, but tests with these fuels up to 277 cal/g resulted in no loss of fuel material. A value of 280 cal/g is currently taken by the Commission as a conservative value for the prompt-dispersal threshold for UO_2 fuels. The slightly higher threshold of annular pellets may be attributed in part to their ability to readily accommodate molten-fuel-volume increases. In the milder incipient-failure mode, the severity of the failure at 275 cal/g for the rod containing annular pellets was much less than for the rod containing solid pellets at approximately the same energy.

It is believed that fuel dispersal results when a substantial fraction of the fuel is molten at the time of cladding failure. For an energy (enthalpy) insertion of 280 cal/g, some, but not all, of the UO_2 fuel is molten. The enthalpy required to melt UO_2 is 332 cal/g. Because the enthalpies of UO_2 and MOX fuels are very nearly the same at high temperatures (see the previous discussion of enthalpy and specific heat), no change is expected in the energy required to produce massive melting of MOX fuel. Because reactivity initiated fuel failure tests are difficult to perform and large numbers of tests have not been conducted additional tests with both UO_2 and MOX are scheduled in the Commission's safety research program at the Power Burst Facility. In the meantime the 280 cal/g value is considered a conservative limit for both UO_2 and MOX fuels.

3.4.2.5 Segregation of Plutonium

Measurable segregation of plutonium and uranium atoms has been observed in LMFBR mixed oxide fuels. This type of fissile-atom inhomogeneity is another way that recycled plutonium fuels can differ from UO_2 fuels. Meyer[36] reports that segregation typically results in a local concentration increase of 35 to 50% at the central-void edge, and that no segregation is observed when a central void is not formed. Figure IV C-23 shows an example for an LMFBR fuel in which segregation has taken place. If the maximum allowable power level is determined by centerline melting or some related design temperature, then segregation of fissile material toward the fuel centerline will lower the allowable power.

For LWR fuels, there are a number of circumstances that reduce the importance of this effect:

- LWR fuels are operated at power levels that preclude central-void formation, and therefore under steady-state conditions, plutonium segregation is not expected.

- The design limit centerline temperature in LWR fuels is based on solid pellet geometry. If overpower operation produces a central void, and concomitantly plutonium segregation, the resulting geometry change to an annular shape would lower the centerline temperature significantly, perhaps several hundreds of degrees. This temperature reduction, due to geometric effects, should overshadow any temperature increases due to an increased fissile-atom concentration near the center of the fuel rods.

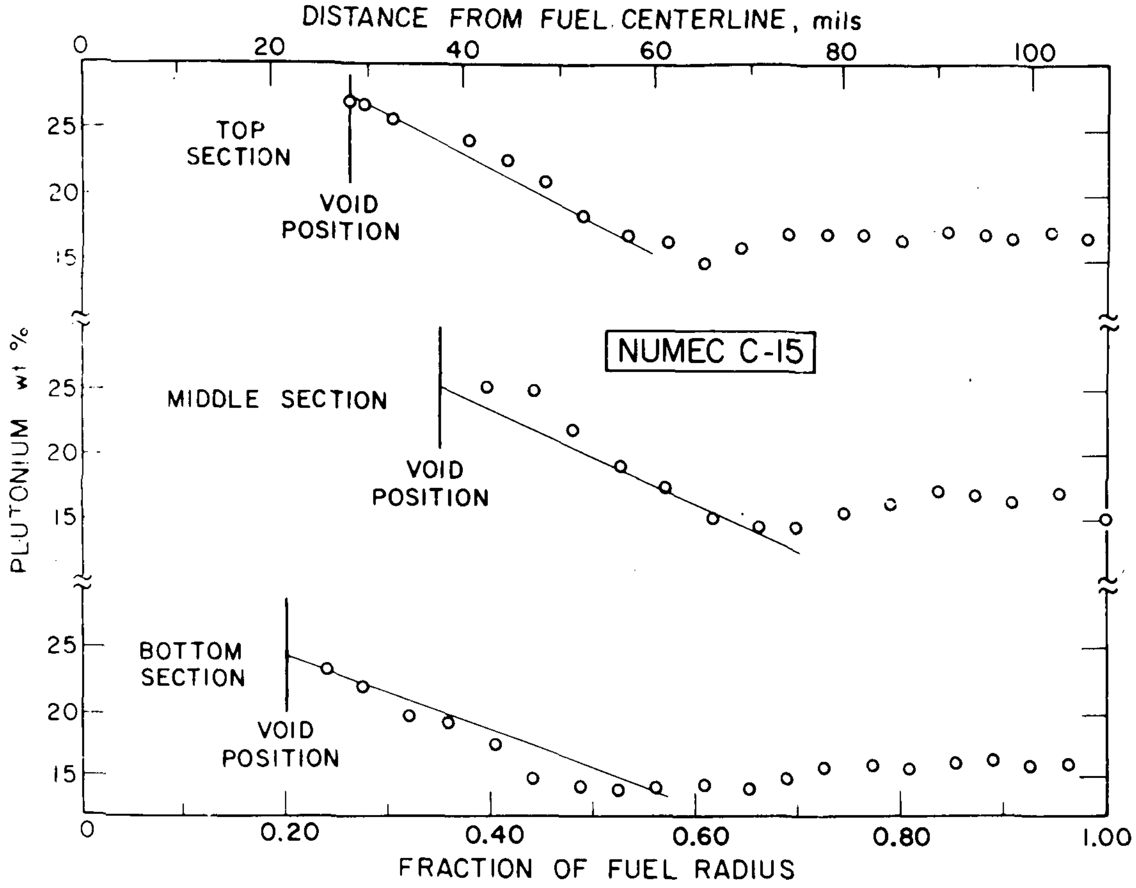
- More than half of the effect in LMFBR fuel is due to lowering of the melting point as local plutonium concentrations near the fuel centerline are increased from 20% to about 30%. A corresponding 50% increase in plutonium concentration from 5% to 7.5% in LWR MOX fuels would have a much smaller effect, changing the melting point by only 10 to 15°C.

- Flux depression in LWR fuels lowers the reactivity, compared with LMFBR fuels, of fissile concentrations at the fuel rod centerline. Because flux depression is even more pronounced in plutonium fuels, the reactivity of plutonium near the fuel centerline is even further reduced.

Although segregation of fissile atoms in recycled mixed oxide fuels is conceivable, the phenomenon seems largely mitigated in LWR's in a thermal flux using current design methods.[37]

3.4.2.6 Radial Temperature Profiles

As a result of higher neutron cross sections, plutonium atoms in the periphery of a fuel rod will capture relatively more of the thermal neutrons than uranium atoms would capture, consequently the thermal flux in the center of the fuel rod will be depressed. Thus, compared with a UO_2 fuel, a mixed oxide fuel will develop less of its power in the central portion of the rod and the resulting temperature profile will be significantly altered.



(See Reference 36)

Figure IV C-23 Typical Measured Plutonium Distributions in a Mixed Oxide Fuel Irradiated in a Fast Flux

Figure IV C-24 shows the radial temperature profiles[38] for a 3% enriched UO_2 fuel and a 4% plutonium mixed oxide fuel, both operated at 13 kW/ft. These parameters are typical of enrichments and peak-power levels that are expected in current LWR's. The centerline temperature of the mixed oxide fuel is substantially lower (approximately by 75°C) than the centerline temperature of the UO_2 fuel. There is a corresponding reduction of the volumetric-average temperature, a measure of the stored energy of the fuel, of about 25°C for mixed oxide fuel compared with UO_2 fuel. Figure IV C-24 includes only the effects of the different nuclear properties of the two fuel types. Changes in thermal conductivity and melting point, discussed in previous sections, will tend to increase centerline temperatures and lower the allowable power rating of a mixed oxide fuel. These trends toward higher temperatures tend to offset the effects illustrated in Figure IV C-24. Recent calculations[37] that consider plutonium concentrations up to 5% indicate that the offsetting tendencies of flux depression and thermal conductivity almost exactly compensate each other. Thus it is unlikely that any change in allowable power rating related to fuel temperatures will be required. These effects, however, must be included in mixed oxide fuel analyses for individual license applications.

3.4.2.7 Fuel Densification

In-reactor densification of LWR fuels produces a small reduction in length and diameter of the fuel pellets, and an analysis of the effects of fuel densification is required by regulation.[39] Extensive sintering during fabrication of UO_2 or MOX pellets can produce densities of 95 to 97% of their theoretical density (TD). Pellets that are not completely sintered may experience additional densification in-reactor, but the last few volume percent of porosity, as typical in most ceramics, is not removed. Thus the potential maximum density (100% TD) and a somewhat lower practical maximum density (∿97% TD) limit the amount of densification that can take place. Consequently, the potential for gross differences in the densification behavior of UO_2 and MOX fuels does not exist.

The mechanisms of fuel densification are not completely understood although work is being done in this area.[40-42] The kinetics of densification in MOX might be different if in-reactor diffusion rates are different, but no comparative measurements of radiation-induced diffusion for uranium and plutonium have been made. It has been suggested[42-43] that the extent of densification in commercial MOX fuels might be less than in UO_2 fuels because of the small particulate inhomogeneities that exist, but this effect has not been verified. It has been demonstrated[44-47] that fuel microstructures (pore size and grain size) play a very important role in determining the densification behavior of a fuel, and it is likely that microstructure, controlled during fabrication, and not plutonium concentration, will determine the densification behavior of MOX fuels.

In-reactor densification data on MOX fuels are becoming available and are being reviewed by NRC. Westinghouse[43] has measured the densification of MOX and comparable UO_2 fuels in both the Saxton and San Onofre reactors. Although there is variability in both the UO_2 and MOX data, there is no indication from the data of a significant difference in the behavior of UO_2 and MOX fuels. General Electric[48] has reported that

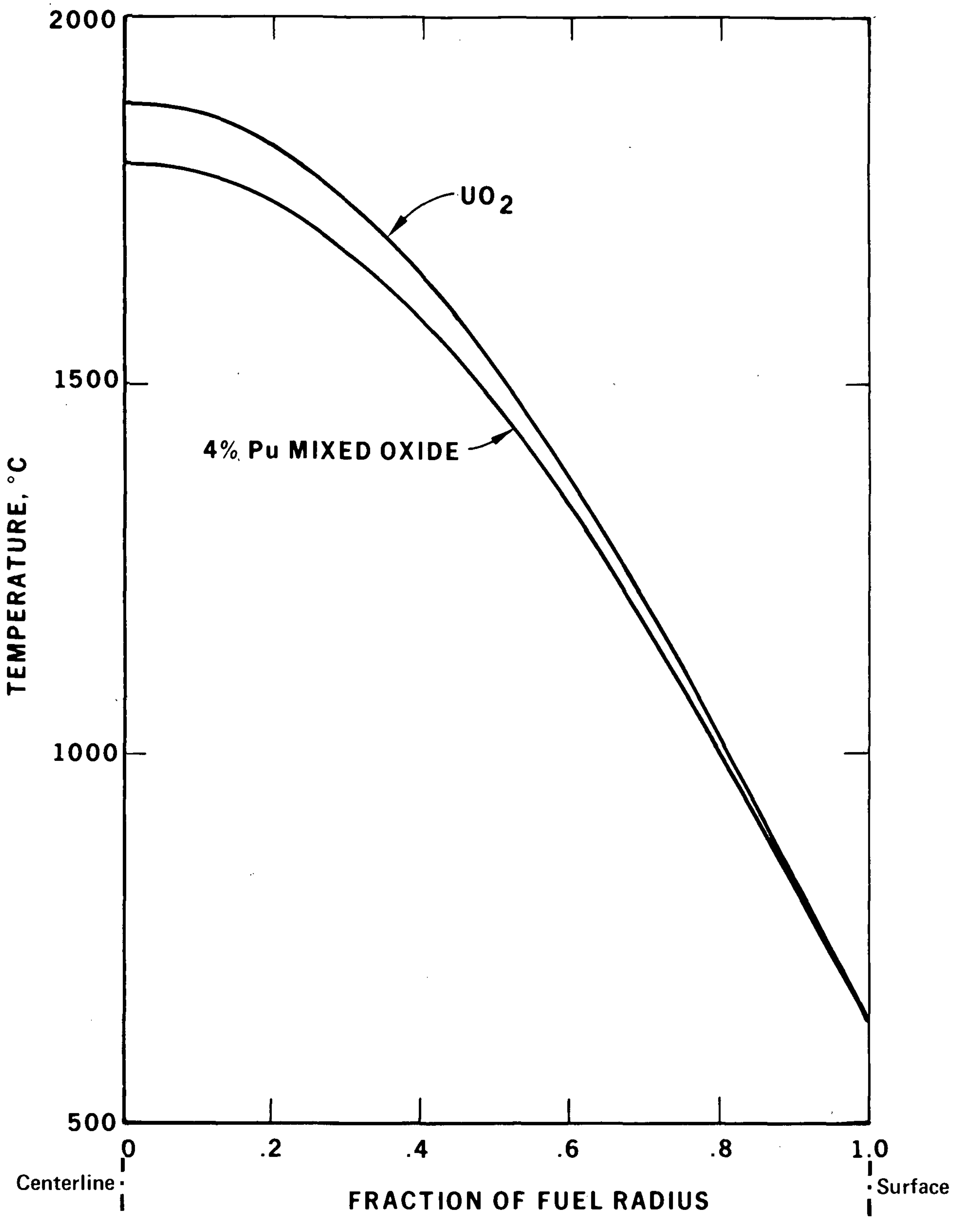


Figure IV C-24 Radial Temperature Profiles of 4% Enriched Mixed Oxide and 3% Enriched UO_2 Fuels, Both Operating at 13 kW/ft.

preliminary densification data for MOX fuels show no significant differences between UO_2 and MOX fuels. Exxon,[49] on the other hand, found significantly less densification in MOX fuels than in UO_2 fuels in the Big Rock Point reactor. Most U.S. fuel vendors are currently making densification measurements on MOX fuels with the expectation that their current densification models can be applied to both UO_2 and MOX fuels. Verification will be required for individual license application to use recycle plutonium in MOX fuel.

3.4.2.8 Plenum Gases

When fission gases and sorbed impurity gases are released from oxide fuel, the plenum gas, helium, is diluted and the thermal conductivity of the gas in the fuel cladding gap is reduced. The possibility that there may be an increase in fission gas release rates for mixed oxide fuels has been considered. Carrol and Sisman[50] report a significant increase in gas release for ∿20% Pu mixed oxide test pellets; however, Stoddart[51] has recently criticized those results and discussed other data that show no difference between MOX and UO_2 fuels.

Beyer and Hann[52] point out that sweep-gas data taken at low heat ratings, such as the Carrol and Sisman data, are often more than an order of magnitude lower than comparable sealed-capsule or fuel rod data. Consequently, Beyer and Hann reject such data from their analysis. Recent MOX gas release data[53] taken on irradiated fuel rods from Saxton (Core III) are in good agreement with the UO_2 and MOX data cited by Stoddart. It is concluded, therefore, that changes in gas release due to the addition of small amounts of plutonium to the fuel should not be important.

The principal volatile impurity released from UO_2 is nitrogen, which is found to exist in the fuel as a second phase in the form of uranium nitrides.[54] Because plutonium also forms a nitride,[14] it is likely that a 5% Pu mixed oxide fuel will absorb nitrogen in a similar manner. The quantity of sorbed gas to be used in a sorbed-gas-release model is measured in a quality control procedure.

3.4.2.9 Swelling Rates

Swelling in oxide fuels is a consequence of the creation of approximately two fission product atoms for each uranium or plutonium atom destroyed in the fission event. Some of the fission products are gaseous and some are solid. Anselin[55] has studied the behavior of solid fission products in both UO_2 and mixed oxide fuels. He concludes that there is no difference in the solid contribution to swelling rates for ^{235}U and ^{239}Pu fuels. Although the production of gaseous fission products is slightly different for ^{235}U and ^{239}Pu fuels, the total quantity of gas produced is almost unchanged,[55] so that a change in swelling rate due to fission gases is not expected. Current fuel-swelling models are empirical and use mixed oxide data in some cases. It is unlikely that fuel swelling models will require modification to describe mixed oxide fuels.

3.4.2.10 Fuel Pellet Cracking

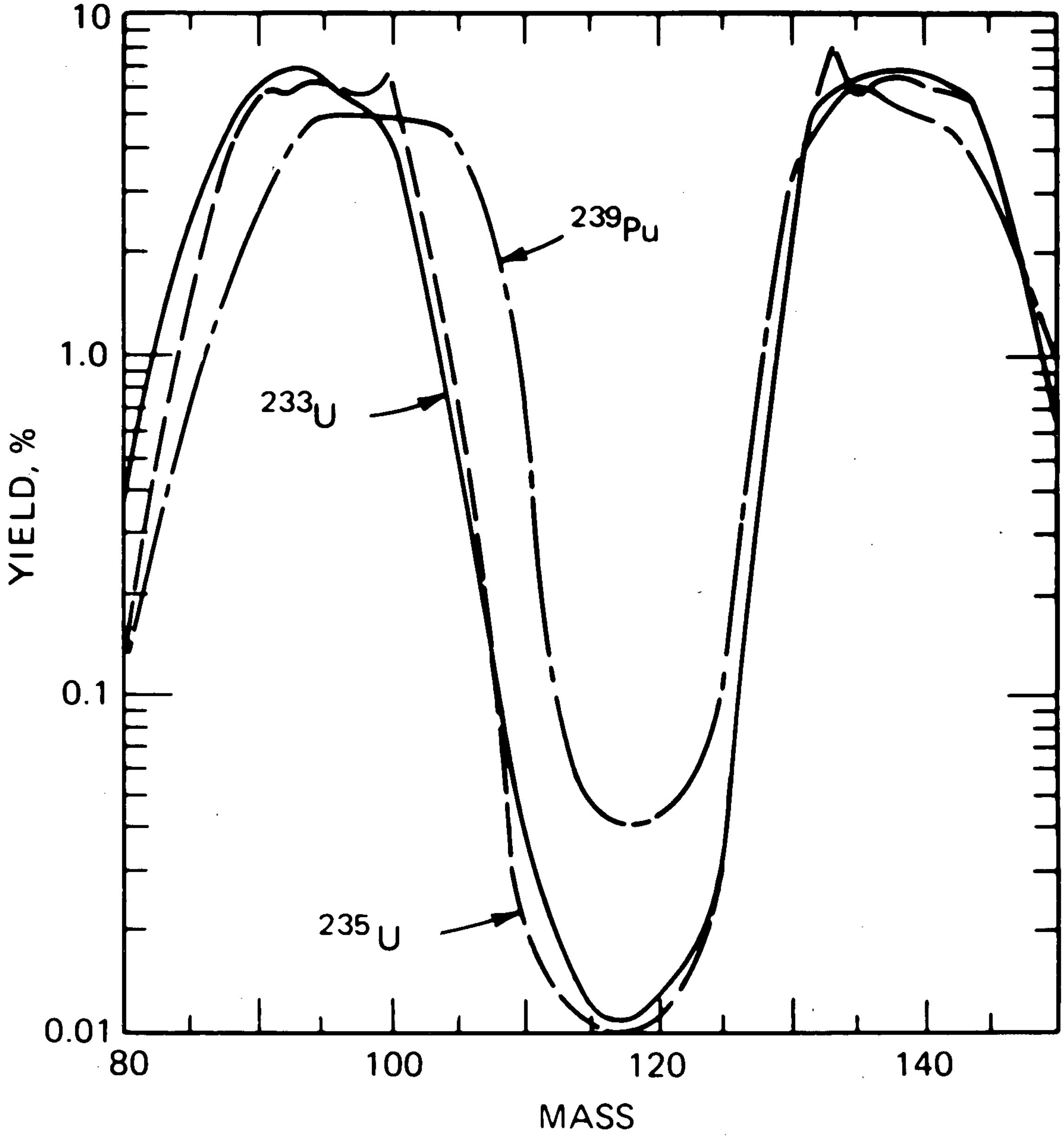
Fuel cracking and relocation of the fractured pieces affects the thermal performance of a fuel rod by generally improving the gap conductance. The mechanisms are not fully understood theoretically; however, the mechanical properties, fracture stress and creep rate may play a role in this phenomenon. It is not possible to state with confidence how these altered properties would affect cracking and relocation, but it should be pointed out that a weaker fuel, a fuel with a lower fracture stress, would not necessarily be detrimental, because an increased tendency for cracking could be beneficial. Because of a lack of detailed understanding of fuel cracking and relocation, all of the current models are empirical. Mixed oxide data should be included in the data base for an adequate model.

3.4.2.11 Fission Fragment Nuclides

The fragments from fissioning of either uranium or plutonium cluster in abundance around mass numbers of approximately 100 to 140, as shown in Figure IV C-25. See Johnson et al.[56] Although the distributions of fission products are similar for the three isotopes shown, it can be seen in the figure that plutonium fissioning produces a slight shift to the right in the lower peak of the yield profile. As a consequence, there is a substantial decrease in the yield of zirconium (A = 91) and a substantial increase in the yields of ruthenium (A = 101) and palladium (A = 106). These changes are significant for two reasons: these metals are among the most abundant fission products; and zirconium, which is reduced in quantity, is an oxygen getter, while ruthenium and palladium, which are increased in quantity, are noble metals. Consequently, for plutonium fuels, an increase in the quantity of available oxygen is expected with increasing burnup. This is an important consideration for LMFBR fuels where oxygen participates in corrosion of the stainless steel cladding. However, no problems have been identified with increases in oxygen activity in zircaloy clad LWR fuels, and the fission product and oxygen/metal differences between UO_2 and mixed oxide fuels are probably inconsequential.

3.4.2.12 Annular Fuel Pellets

Because of the greater neutron absorption of plutonium, there will be a tendency to design mixed oxide fuels for higher water-to-fuel ratios. For BWR's it is likely that annular pellets will be used to increase the water-to-fuel ratio. While there are other advantages to using annular pellets, such as lower central temperatures, there is the possibility of relocation of fissile material through the central hole, and there are some changes in the mechanical behavior compared with solid fuels. Prototypical irradiations[5] of annular mixed oxide pellets have shown that small quantities of fuel particles collect in the bottom of the fuel rod, but that the quantities are so small as to have no adverse effect on fuel performance. Other tests[57] have shown that, under transient overpower conditions, annular pellet rods exhibit different behavior than solid pellet rods, although the cladding failure threshold energies are similar. There is a substantial amount of irradiation experience with annular pellet fuels, but the geometric effects must be assessed in comparison with the more common solid pellet fuel in the safety analysis of a particular reactor design.



(See Reference 56)

Figure IV C-25 Mass-Yield Curves for Thermal-Neutron Fission of ^{233}U, ^{235}U, and ^{239}Pu.

3.4.2.13 Fuel Pellet Density

Another way to increase the water-to-fuel ratio for both BWR's and PWR's is to lower the fabricated density of the fuel pellets. Although no intention to do this has been expressed by any of the U.S. fuel vendors, the incentive is present. Lowering the fabricated density tends to increase fuel densification; no difficulties are foreseen from such a procedure. Techniques have been developed by the fuel vendors to produce relatively stable fuel microstructures, and all of the NRC-approved densification models account for variations in the initial fuel density.

3.4.2.14 Burnable Poisons

The reactivity worth of absorbers or poisons is reduced in mixed oxide cores, as described in paragraph 3.3. When burnable poisons like gadolinium are fabricated into the fuel pellets, the increased demand for poison can be accommodated by either increasing the number of rods containing poison or increasing the concentration of poison in a rod. In the latter case, concentrations might be increased beyond the current practice, and the effect of larger concentrations of poison material would have to be assessed. Current design trends, however, are toward lower burnable poison concentrations, so it is unlikely that a problem will be encountered with rods containing burnable poison.

3.5 Behavior of MOX LWR's

In the previous sections, differences between UO_2 and mixed oxide fuels have been examined. Many of the core design features important in the safety analyses are identical for the two fuel types. Mechanical properties of the cladding and structural members, fatigue life, cladding collapse, fretting corrosion, crud deposit, fuel assembly geometry, coolant flow, flow induced vibrations, seismic, and other considerations are unchanged. These similarities, along with the relative insignificance of the changes indicated in paragraphs 3.3 and 3.4 for mixed oxide fuels, insure the similarity in behavior of mixed oxide and UO_2 fuels, which has been demonstrated by irradiation experience.

As a consequence of the similarities of cores with UO_2-only and those containing mixed oxide fuels, no changes are judged to be necessary in the methodology or calculational techniques that are used in required safety analyses. The calculations for steady state, transient, and accident behavior will be performed for each plant using accepted methods that are currently employed for UO_2 cores, except that the affected parameters will be modified to reflect the use of plutonium fuels. For illustrative purposes, a description of the qualitative changes in behavior that can be expected for MOX fueled LWR's will be presented.

3.5.1 Normal Operation

Steady state behavior of plutonium recycle cores can be expected to be similar to UO_2 cores. The more negative Doppler, void, and moderator temperature coefficients, as well as the decreased ^{135}Xe effect, make a MOX fueled LWR more stable when operating at steady state. The decreased delayed neutron fraction and prompt neutron lifetime are not significant during steady state operation or when reactivity changes are made slowly.

Microscopic inhomogeneity of fissile material will have a negligible effect on heat flow characteristics. Macroscopic plutonium segregation is not expected, and fuel centerline temperatures will be very nearly the same for a given power level. The densification of mixed oxide fuel pellets is not expected to differ from UO_2 densification. The thermal expansion and mechanical properties of the mixed oxides are so close to the corresponding properties of UO_2 that differences are not measurable in well controlled laboratory tests. Thermal and mechanical properties of the cladding and structural members are not changed, and the thermal hydraulics are unaltered.

As discussed in paragraph 3.3, the more negative reactivity coefficients in a mixed oxide core require more control to change the reactor power level since the power coefficient is more negative. In BWR's, the reactivity change from cold-shutdown to hot-operating conditions is somewhat increased, but this increase is largely offset by the reduction in control requirements for long term burnup of mixed oxides. Scram reactivity rates and power shaping requirements must also be considered. In PWR's, load changes and ^{135}Xe reactivity transients are usually controlled with the soluble poison system during most of the cycle. Later in the cycle, when the required concentrations of poison in the coolant becomes excessive, control rods are used to make load changes. Soluble-poison control for load following is little different with UO_2 or mixed oxides. Although greater insertion of the control banks for a given power change will sometimes be required and hence, make load follow more restrictive, the lessened effect of ^{135}Xe redistributions will have a beneficial effect in reducing the need for control rod motion to maintain acceptable axial power distribution.

3.5.2 Transients and Accidents

Events of moderate frequency that produce anticipated operational transients can be categorized as three general types:

- Those that cause an increase in power

- Those that cause an increase in coolant temperature

- Those that cause a decrease in the coolant temperature

The more negative Doppler, moderator temperature, and void reactivity coefficients in a mixed oxide fueled reactor will make the first type of transient, such as uncontrolled rod-bank withdrawal, less severe. The smaller delayed neutron fraction and shorter prompt neutron lifetime potentially make the first type of transients more severe for the MOX reactors, but the more negative coefficients are controlling. The boron dilution transient would be less severe with mixed oxide fuels because the soluble poison worth is less. Plutonium segregation could occur during sustained overpower operation that caused centerline temperatures to rise, but the consequences of such segregation are judged to be unimportant.

An example of the second type of transient is the loss of turbine load. In this case, the more negative moderator temperature coefficient of a mixed oxide core would make the temperature and pressure transients less severe in a PWR, but potentially more severe in a BWR because of the reactivity increase due to rapid void collapse.

In the third type of transient, such as startup of an inactive reactor coolant loop, the more negative moderator and void coefficients of a mixed oxide core would tend to be somewhat detrimental. Because sufficient shutdown margin is always maintained, however, the consequences of this transient are not serious.

The more serious design basis accidents that are postulated for LWR's have been analyzed by several fuel vendors by comparing plutonium recycle cores to UO_2 cores. The more negative Doppler, moderator temperature, and void reactivity coefficients compensate for the lower delayed neutron fraction (β) and the shorter prompt neutron lifetime (ℓ^*), and the consequences of the accidents are comparable for mixed oxide and UO_2 cores except for those accidents to be mentioned.

3.5.2.1 PWR Steam Line Break

The PWR steam line break accident results in a rapid cooling of the core and a potential return to criticality because of the negative moderator coefficient. To prevent this, more control rods or a higher boron injection rate may be required. More restrictive fuel management will be required to minimize such changes.

3.5.2.2 Rod Ejection

The postulated rod-ejection accident for a MOX fueled reactor may be more or less severe, depending on the core design. More negative reactivity coefficients and lower ejected-rod worths are advantageous, while the smaller β, ℓ^*, and delayed Doppler feedback, when large PuO_2 agglomerates are present, are detrimental.

3.5.2.3 Loss of Coolant

The consequences of a loss of coolant accident are not appreciably different between UO_2 and mixed oxide cores. Several factors tend to make the accident less severe with mixed oxides. Additional flux depression will compensate for the lower thermal conductivity of mixed oxide fuel so that the stored energy will be somewhat reduced. Especially where annular pellets are used, the stored energy of a mixed oxide fuel might be significantly reduced. The smaller β and shorter ℓ^* make the decay of neutron fissioning after the accident more rapid, resulting in less residual fission power. After 100 seconds, the fission product decay heat[58-60] is several percent less for ^{239}Pu than for ^{235}U fissions because of the different fission product yields. The net energy per fission[61] for ^{239}Pu is 2% to 3% higher than ^{235}U fissions, thus requiring fewer fissions for the same energy output. The result is a somewhat lower short term decay heat for mixed oxide rods for a given power rating. At the end of a cycle when over 50% of the fissions in a UO_2 core are from plutonium, the difference between mixed oxide fuel and UO_2 fuel decay heat would be reduced.

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4.0 MODEL PLUTONIUM RECYCLE REACTOR FOR GESMO

Plans to recycle plutonium in the reload fuel assemblies for licensed operating light water reactors (LWR's) or fuel assemblies in cores of new LWR's will be reviewed by the NRC Office of Nuclear Reactor Regulation on case-by-case terms. This review will provide individual assurances that the risk and hazards to the health and safety of the public will remain acceptably low as provided by UO_2-only cores.

A model plutonium recycle LWR has been developed for GESMO purposes to relate the environmental consideration of the supporting fuel cycle to the annual operation of a representative power plant. Plutonium recycle in LWR's, within the values identified for the model reactor, can be accomplished with existing (demonstrated) technology without imposing significantly new restrictions on the operation of LWR's. The plant performance characteristics for the GESMO model reactor are not changed from the UO_2-only plant performance characteristics; this will be verified for any LWR that is authorized to recycle plutonium.

Plant design, site cooling water requirements, and, as reported later in Section 5.0, environmental considerations in the vicinity of the nuclear plant are unaffected by the presence of additional amounts of plutonium contained within the sealed clad fuel rods in the core of the GESMO model LWR. Potential environmental effects at an LWR are dependent on the differences in fuel composition. The basis for and the derivation of the GESMO model LWR, identified as the 1.15 SGR, are reviewed in the following sections.

The isotopic composition of the plutonium recycled in the GESMO model LWR is representative of the plutonium that may be commercially available, no earlier than the year 2000, containing a large amount of aged plutonium in contrast to the plutonium from UO_2-only cores that will be available initially if plutonium recycle in LWR's is authorized. The environmental impact analysis therefore is more conservative in this respect because of increased isotopic plutonium toxicity.

The model plutonium recycle reactor contains uranium enriched fuel rods (UO_2 only) and blended or mixed PuO_2 and UO_2 fuel rods (MOX rods). As many as 40% of the rods in the model LWR may be MOX rods, depending on the enrichment of the UO_2 rods that are replaced by equivalent MOX rods and the quality (isotopic composition) of the plutonium. The configuration of the model reactor fuel rod assemblies with MOX rods will be unchanged from the UO_2-only fuel rod assemblies and the amount of plutonium in the MOX rods will be controlled so that fuel assemblies containing MOX fuel rods are interchangeable with UO_2-only fuel rod assemblies, except for some PWR core positions that are also used for reactor control. On the basis that fuel rod integrity is not affected by using plutonium instead of enriched uranium in some of the new fuel rods (IV C-3) and that power histories for the UO_2 and MOX LWR's are identical, the potential environmental effects of the model plutonium recycle LWR's are dependent exclusively on the differences in core nuclide inventory caused by the change in new fuel composition. These differences, based on the comparison of UO_2-only and GESMO model core nuclide inventories in the following sections, are relatively insignificant with respect to the controlled release of radionuclides in the gaseous and liquid effluents during normal reactor operation or the escape of gaseous radioactivity during accidents that include the low probability design basis accidents.

4.1 Basis for the GESMO Model Reactor

Plutonium is formed by the interaction of neutrons with ^{238}U in the uranium fuel of all LWR's during normal operation for the production of energy. Much of the plutonium fissions in place as it forms, to the extent that approximately 35% of the total energy produced by uranium fueled reactors is due to the fissioning of plutonium formed within the enriched uranium fuel rods. Most of the plutonium produced in uranium reactors can fission and generate heat, but nonfissile plutonium isotopes are also produced. Not all of the fissile plutonium produced actually fissions in place. Some of the nonfissile isotopes, notably ^{238}Pu and ^{240}Pu are fertile in that neutron capture by them produces the fissile isotope ^{239}Pu or ^{241}Pu. After the plutonium has been recycled many times, the ratio of fissile to nonfissile plutonium atoms is reduced. The spent (depleted) fuel removed from a uranium only fueled reactor after a normal residence time, typically 3 or 4 years, contains the residual fissile plutonium produced during the irradiation period, the nonfissile plutonium isotopes, and *other transuranium nuclides, together with the residual* ^{235}U, *other transuranium* nuclides, together with the residual ^{235}U, ^{238}U, and fission products. Spent fuel assemblies are normally stored at the reactor site for at least 120 days in a water pool provided for that purpose to allow sufficient decay of short lived radioactive elements for the safe offsite shipment in shielded and cooled containers to long term storage areas. If plutonium recycle in LWR's is approved, the spent fuel will be sent to fuel reprocessing plants where uranium and plutonium can be recovered for fabrication into MOX fuel rod assemblies and reinsertion into LWR's displacing an equivalent number of UO_2-only fuel assemblies. If plutonium recycle is not approved the spent fuel will eventually be shipped to a long term storage disposal site, (assuming uranium recycle only is not practical).

If the plutonium recovered from LWR spent fuel rods, after each of the annual core refueling operations, is recombined with uranium (natural uranium, for example), fabricated into MOX fuel rods and reinserted into the LWR core along with sufficient low enriched uranium fuel rods to satisfy core reactivity requirements, and if this procedure is repeated for about 30 successive refuelings, equilibrium will be achieved. Equilibrium is achieved when the amount of plutonium recovered from the spent MOX and UO_2-only fuel rods will be equal to the plutonium in the new MOX fuel rods charged into the reactor 3 or 4 years earlier. This is described as an equilibrium self-generation reactor (SGR). At equilibrium the SGR recycles all of the plutonium that it produces.

This concept of plutonium recycle is not the only possible fuel management mode: others include utilization of more or less plutonium than is recovered from the spent fuel of one SGR. For example, it is possible that in the more distant future all of the fuel rods in a limited number of LWR's could be mixed oxide, thereby eliminating any dependence of such reactors on uranium enrichment facilities. Instead, there would be a new need for plutonium from other operating uranium reactors to supply such "plutonium burners."

Operation in this mode contrasts sharply with the plutonium recycle conditions represented by the SGR, in that if the number of operating LWR's remained constant (no growth), all of the plutonium produced in LWR's could be recycled in about one-third of the operating LWR's whereas the SGR offers the potential utilization of nearly all operating LWR's to recycle the plutonium that is produced. It should be noted, with respect to the generic radiological safety evaluation for supporting the fuel cycle, that plutonium recovery from spent fuel and plutonium transportation between the chemical separation plants and fuel fabrication plants is not noticeably sensitive to the number of LWR's used to recycle all of the plutonium that is produced.

4.1.1 Model Pu Recycle Reactor - 1.15 SGR

The probable mix of plutonium recycle modes for the period between 1975 and 1985 would be as follows based on a survey by ORNL of the opinion of vendors, utilities, and associated experts:

- 10% of LWR's would not recycle any plutonium.
- 40% of LWR's would recycle at SGR or will operate between 2/3 SGR and SGR.
- 45% of LWR's would recycle above SGR. Most will be just above SGR, but some will be routinely paired with a uranium reactor and recycle the plutonium generated in both.
- 5% would recycle in excess of 2 SGR.

More than self-generation quantities of plutonium can be recycled safely under current technology, but a precise upper limit has not been determined. The SGR has been selected as the reference basis for the model plutonium recycle reactor because the existing UO_2 reactor technology is generally applicable and new uncertainties, associated with plutonium recycle of larger quantities in LWR's, are negligibly low; also

- Most plutonium will be recycled, if plutonium recycle in LWR's is approved by the Nuclear Regulatory Commission, in LWR's near SGR levels.
- Plutonium recycle at SGR levels allows utilization in LWR's of all the plutonium recovered from LWR's.
- Plutonium can be recycled in existing LWR's at approximately SGR levels without significant reactor modifications or other emergency core cooling systems (ECCS) improvements to maintain LWR safety margins, but each proposal to recycle plutonium in LWR's will be reviewed and approved by the NRC to verify this and provide individual assurances that reactor safety margins and operating characteristics are acceptable.

To provide design flexibility and to reflect the potential for greater than SGR quantities of recycle plutonium and at the same time provide an arbitrary limit that can be used for analytical purposes related to the environmental impact (including a typical reactor site) evaluation, the GESMO model reactor has been set at 115% of the equilibrium SGR. The 1.15 SGR level corresponds to an average plutonium content of 1.8 weight percent of the heavy metal (Pu and U) in the as charged fuel; 100 times the weight of plutonium, 590 kg,* divided by the total heavy metal 32,200 kg, equals 1.8. For an equivalent PWR, the corresponding weight of recycle plutonium is 485 kg. Conversion of a UO_2-only LWR to a 1.15 equilibrium SGR by recycling the plutonium recovered plus 15% more from some other source would require about half of the time required for SGR only to reach equilibrium, 16 vs. 30 years. The use of 1.15 SGR as a model reactor should not be interpreted to mean that there is an inherent safety or environmental limit at 1.15 SGR (or close to this value) on the use of recycle plutonium in reactors. This is not the case. On the other hand, it should not be concluded that there is not some limit beyond 1.15 SGR at which the safety or environmental consequences of the use of recycle plutonium in reactors are not comparable to that of UO_2. To identify this limit precisely was not considered to be justified in light of results of the survey of the industry plans for the use of recycled plutonium in currently designed LWR's.

4.1.2 Basis for Selection of the GESMO Model Reactor

From the beginning of the year 1970 to the end of 1975 (6 years) a total of 45 nuclear power plants began commercial production of electricity in the United States. The average net power capacity of these 45 nuclear plants[2] is slightly less than 800 MWe. Over the 5-year period beginning in 1976, a total of 45 additional plants, now under construction, with average net power capacity slightly in excess of 1,000 MWe are scheduled to begin commercial operation.

At the end of 1975, there were eight operating nuclear power plants with rated net power capacity 5 to 12% greater than 1,000 MWe. The current limit on the size (capacity) of nuclear power plants imposed by NRC is 3,800 MWt. The most recent plans call for construction to begin on two plants of this size in time to begin commercial operation in 1983 and 1984. The economy of plant size is illustrated by one study,[3] which concludes that half ownership in a 1,150 MWe nuclear plant would cost $220 million less than construction of a 575 MWe coal fired plant.

The model reactor, 1,000 MWe net power capacity, is representative of the largest nuclear power plants now operative and the average of those that are scheduled to start in the next 5 years. Plutonium recycle in accordance with the GESMO Model LWR should not affect the power rating of the reactor.

Thermal Efficiency - 32.6%

Thermal efficiency is a measure of the efficiency with which the plant converts thermal to electrical energy. Most of the currently licensed LWR's operate at a rated power level with thermal efficiencies between 32 and 33%; some operate at efficiencies as low as 31.5% and others as high as 35%. LWR's under construction are expected to perform at slightly higher thermal efficiency. The value selected for the model is

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*Total plutonium - Refer to Table IV C-9 nearly representative of existing LWR's and perhaps a little low for those now under construction. Actually, the 32.6% value chosen for the GESMO model reactor results in core power rating and core nuclide inventories that are slightly larger than reality; thus, calculated radiation dose levels are therefore conservatively on the high side. LWR thermal efficiency is unaffected by plutonium recycle.

Plant Factor - 0.80

Plant factor (sometimes called capacity factor) is defined as the ratio of the average power load of an electric power plant to its rated capacity. A 0.80 power factor is equivalent to plant operation at 100% power level for 80% of the time or 80% power level for 100% of the time. The plant cannot operate at 100% power level continuously because of the need to shut the plant down periodically for refueling, maintenance, inspection and testing, and demand for power may be satisfied at less than 100%. GESMO calculations based on the 0.80 power factor for the GESMO model LWR will result in an average power level that is higher than normally attainable. Therefore, the power level dependent inventory of short lived radionuclides will be conservatively high with respect to environmental effects. By contrast, the inventory of the long lived radionuclides is proportional to burnup and is not sensitive to power level at any given exposure. The plant factor for LWR's is not expected to change because of plutonium recycle.

4.1.3 Reactor Types - BWR and PWR

For comparative power ratings, the boiling water reactor core is larger than the pressurized water reactor core.[4] One-fourth of the fuel assemblies in a typical BWR core is replaced annually by an equal number of 2.6% average enriched uranium fuel assemblies. One-third of a typical PWR core is replaced annually with 3.2% average enriched uranium fuel. The total annual requirement for ^{235}U is approximately the same for each of the two LWR type reactors, but because the typical BWR core is about 33 percent larger than a PWR core, the inventory of long half-life nuclides including plutonium accumulated in a BWR core is noticeably larger. Because it is representative of the highest inventory of long lived nuclides in LWR's, the BWR has been selected as the GESMO model LWR. The environmental impacts of plutonium recycle in both PWR's and BWR's are presented later (CHAPTER IV, Section C-5.0).

The 1,000 MWe BWR core contains about 129 metric tons[4] heavy metal (MTHM) as charged, and a typical reload for a 1,000 MWe BWR unit contains 32.25 MTHM. Core size does not change if plutonium is recycled; therefore, the GESMO model BWR core contains 129 MTHM, and the reload fuel assemblies contain 32.25 MTHM. By contrast a representative 1,000 MWe PWR core with or without plutonium recycle contains about 81 metric tons of heavy metal as charged, and a typical reload weighs about 27 MTHM.

4.1.4 Industry Mix of BWR's and PWR's

There were about 56 operating LWR's[2] in the United States at the end of the year 1975. Forty-one percent of these plants are BWR's; 59% are PWR's. At the same time, 48 of the 67 LWR's under construction were PWR's (72%). Of the 84 LWR's planned for construction, 56 (67%) are PWR's. The GESMO LWR growth projections assume that one-third of the LWR's are BWR's, and two-thirds of the LWR's are PWR's. The overall economics of plutonium utilization in LWR's may favor one type of reactor over the other, but it is unlikely that this factor would influence the mix of BWR's and PWR's.

4.1.5 Recycle Plutonium Composition

Repeated recycling of plutonium changes its isotopic composition, as indicated by the ORNL[5] calculated values shown in Table IV C-5.

Table IV C-5

CALCULATED PLUTONIUM COMPOSITION - PERCENT

	1 Pu Recovered From Spent U Fuel	2 Pu After One 4-year Recycle	3 Pu After Two 4-year Recycles	4 Pu Recycle Model BWR
^{238}Pu	1.9	3.46	4.87	3.4
^{239}Pu	57.9	38.2	29.4	41.7
^{240}Pu	24.7	29.4	33.5	29.2
^{241}Pu	11.0	17.2	17.4	15.2
^{242}Pu	4.4	11.7	14.9	10.4
Pu_f*	68.9	55.4	46.8	57.0

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*$Pu_f = {}^{239}Pu + {}^{241}Pu$

If the plutonium recovered from spent uranium rods is blended with natural uranium, as assumed for this evaluation (uranium recovered from spent fuel or uranium tailings may also be used in place of natural uranium) and recycled in an LWR for 4 additional years of irradiation, the composition of the recovered plutonium changes as shown in Column 2. ^{239}Pu decreases from 58% to 38%, and for the example chosen, ^{241}Pu increases from 11% to 17%. After one recycle, the fraction of fissile plutonium (Pu_f) decreases from 69% to 55%. At the end of two normal 4-year plutonium recycle periods, the calculated Pu_f fraction decreases to 47% as obtained from Column 3 by combining ^{239}Pu and ^{241}Pu. As can be seen from Table IV C-5, the fraction of Pu_f decreases with each additional recycle. If the total plutonium in reload fuel assemblies remains the same while the Pu_f fraction in recovered plutonium decreases, the plutonium will be distributed in fewer mixed oxide fuel rods to compensate for the neutron absorption in nonfissile plutonium. More ^{235}U enriched fuel rods will be required to satisfy core reactivity requirements.

It is expected that the identity of recycled plutonium will not be retained from cycle to cycle in LWR's as shown in Table IV C-5. Instead, the recycle plutonium for each reload is expected to be a blend of plutonium recovered from the spent fuel of all LWR's. It will therefore be a higher quality than assumed for the 1.15 SGR analysis: each fuel reload will have more fissile and less parasitic plutonium. Refer to MOX Fuel Fabrication CHAPTER IV, Section D.

The model reactor core nuclide inventory allows for recycle plutonium of the composition shown in the fourth column of Table IV C-5. This is the plutonium composition calculated for the 1.15 equilibrium SGR, which is described in this section, paragraph 4.2. Recycle plutonium of higher quality than calculated for the GESMO model reactor plutonium, such as plutonium that will be available initially, if plutonium recycle is authorized, will result in a core nuclide inventory with less plutonium. The environmental effects during normal operation or following accidents including DBA's will not change significantly from the GESMO model values. New (high quality) plutonium reduces the requirement for ^{235}U enrichment. More of the plutonium fissions to produce power; thus, core inventory of plutonium is less than the corresponding value for the GESMO model reactor. The differences in radioactive fission gas releases due to plutonium composition variations will be negligibly small. The model LWR therefore adequately represents the environmental impact of plutonium recycle in LWR's for the plutonium composition variations that can be expected through the year 2000, assuming the low LWR growth rate presented in CHAPTER I.

4.1.6 Chemical Waste Released to the Environment

To obtain estimates of the amount of chemical wastes released to the environment from nuclear reactors, environmental statements from modern PWR and BWR reactors were used, with the realization that future design changes could change the amounts discharged. In addition, assigning such estimates to the total industry obscures site specific and individual plant operating characteristics that ultimately control the quantities of chemical wastes released.

Calculation of the amount of chemical materials released to the atmosphere was based on the data reported in the final environmental statements related to construction of the BWR Hartsville Nuclear Plants[6] and the draft environmental statement related to construction of the PWR Marble Hill Generating Station.[7] Both stations utilize oil fired auxiliary steam generators as a steam supply for unit startup, building heating, and other minor plant uses. The steam generators will use No. 2 fuel oil. The gaseous effluents from burning the No. 2 fuel oil were estimated on the basis of U.S. Environmental Protection Agency (EPA) air pollution emission factors. To normalize the amount of effluents to a 1,000 MWe plant, a linear relationship between unit capacity and chemicals released to the atmosphere was assumed.

The quantities of chemical materials discharged in metric tons per year are listed in Table IV C-6 for a composite LWR. These values are not changed by plutonium recycle from the UO_2-only LWR values. The model for an oil fired auxiliary steam generator assumes that the average power level is 50% and that rated power is 20 MWt.

Table IV C-6

CHEMICAL MATERIALS DISCHARGED

Chemicals	Composite Model LWR MT/YR
SO_x	150
NO_x	180
CO	4
Hydrocarbons	4
Particulates	13

The estimates of chemical discharges for release to water bodies are based on those used in the draft GESMO for once-through cooling systems where 2,790 MT of sulfates and 232 MT of chlorides are discharged annually to water bodies. For plants using cooling towers[6,7] to dissipate the condenser discharge heat instead of once-through cooling water the amount of sulfuric acid and sodium hypochlorite in the blowdown to the main water body is about 1,400 and 92 MT/year respectively. These values vary widely from site to site. Plutonium recycle in LWR's does not affect the chemical discharges to water bodies and the LWR values are therefore equally applicable to the GESMO model LWR. If cooling towers are assumed for all model plants, the normal river or lake water quantities of chemicals and solids would be concentrated and discharged in cooling tower blowdown. The concentrations of the chemicals from cooling towers for the Hartsville 1,285 MWe per unit are listed in Table IV C-7 as an example.

Cooling tower drift would reduce the amount of water discharged to water bodies and increase the amount of water and chemicals discharged to air. The suspended solids from the cooling tower for a 1,000 MWe plant are estimated to be about 25 MT per year.

Other impacts and energy uses of the 1,000 MWe composite PWR and BWR are presented in Table IV C-8.

4.2 Model Plutonium Recycle Reactor

It has been assumed in the calculations for the model plutonium recycle reactor that 2 years is the elapsed time between removal of spent fuel from the reactor and the recycling of the recovered plutonium in new fuel assemblies inserted during core refueling. This period includes storage of spent fuel elements at the reactor site, transportation to chemical recovery plants, further storage, fuel reprocessing, temporary storage of recovered plutonium, transportation of plutonium to a fuel fabrication facility, production of MOX fuel rods, transportation of MOX reload fuel assemblies to a reactor site, and, finally, temporary storage of the fuel assemblies at the reactor site until the plant is shut down for fuel reloading. It would be desirable to recycle the plutonium as soon as possible because of the 15-year half-life decay period for fissile ^{241}Pu. A delay in recovery and reinsertion of recycle plutonium into a reactor core represents a loss in energy potential. Excessive delay in inserting plutonium recycle reload assemblies after completion of fuel assembly fabrication

Table IV C-7

EXPECTED CONCENTRATION OF EFFLUENTS FROM COOLING TOWER BLOWDOWN (Hartsville Nuclear Plant)

Parameter	Cumberland River, mg/l	CF = 2[*]		CF = 6.6[**]	
		Effluent Concentration, mg/l	Concentration[***] in River After Mixing, mg/l	Effluent Concentration, mg/l	Concentration[***] in River After Mixing, mg/l
Dissolved Solids[†]	110	228	121.8	734	172.4
Suspended Solids[†]	39	78	42.9	257	60.8
Ammonia[†]	0.11	0.22	0.12	0.73	0.17
Fluoride[††]	0.06	0.12	0.07	0.40	0.09
Chloride[††]	3	6	3.3	20	4.7
Sulfate[†]	19	42.8	21.4	130.2	30.1
Total Phosphate[†]	0.27	0.54	0.30	1.78	0.42
Silica[††]	4.7	9.4	5.2	31	7.3
Total Iron[††]	1.0	2.0	1.1	6.6	1.56
Total Manganese[†]	0.20	0.4	0.22	1.32	0.31
Total Copper[†]	0.20	0.4	0.22	1.32	0.31
Total Zinc[†]	0.27	0.54	0.30	1.78	0.42
Total Chromium[†]	0.01	0.02	0.011	0.07	0.016
Total Aluminum[††]	1.6	3.2	1.76	10.56	2.5
Total Nickel[†]	8.50	1.0	0.55	3.3	0.78
Total Lead[†]	0.12	0.24	0.13	0.79	0.19
Total Cadmium[†]	0.016	0.03	0.017	0.11	0.025
Total Boron[†]	0.20	0.4	0.22	1.32	0.31
Sodium[†]	2.7	7.5	3.2	20	4.4
Potassium[††]	1.9	3.8	2.1	12.54	2.96
Mercury[††]	0.0006	0.0012	0.0007	0.004	0.0009

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[*] CF (concentration factor) = 2: factor by which concentration of parameters in raw river water are multiplied in heat dissipation system during normal plant operation.

[**] CF (concentration factor) = 6.6: factor by which concentration of parameters in raw river water are multiplied in heat dissipation system during 30-day holdup of blowdown.

[***] Concentration of parameters at edge of mixing zone after initial 9:1 dilution with ambient river water.

[†] Maximum concentration in samples at Cumberland River Mile (CRM) 285 during March 1973-June 1974 (ER Supp. 1).

[††] Maximum concentration in samples taken at CRM 313.5 in May, July, and September 1973 (from ER Table 3.6-1).

therefore results in a loss of fuel bundle reactivity. It can be projected that recovery of plutonium from spent fuel will be accomplished in less than 2 years at some future time.

Table IV C-8

IMPACTS AND ENERGY USES (Composite PWR BWR)[6,7]

Annual Use or Discharge	Composite Model LWR
Electrical Energy for Plant Auxiliaries, MW Yr	40
Fuel Oil,* gal	3.4×10^6
Acres of Land Per Reactor Site	500
Acres Committed to Use	50
Acres Permanently Committed	2
Gallons of Water to Atmosphere, by Cooling towers	6.5×10^9
Gallons of Water to Water Bodies, by Cooling towers	6.5×10^9
Gallons of Water to Water Bodies, by Direct cooling	3×10^{11}
Thermal Dissipation, Btu	5.0×10^{13}
Waste Solids, cu. m	260

*50% Rated Power (Table IV C-6)

Nearly all of the plutonium in spent fuel assemblies removed from LWR's can be recovered, fabricated into mixed oxide fuel rods and recycled in LWR's. However, small amounts of plutonium remain in the radioactive waste, ^{241}Pu decays noticeably during the assumed 2-year interval between removal of spent fuel assemblies and reinsertion of the recovered plutonium in the form of MOX fuel rods, and there is waste in the fabrication of MOX fuel pellets and rods. To allow for this incomplete plutonium recovery and decay, the 1.15 SGR model calculations assume that 276 kg of plutonium are recovered from the spent fuel assemblies of uranium cores that contain the 281 kg of plutonium as presented in Table IV C-9. This value when increased by 15% (1.15 SGR) and reduced to allow for ^{241}Pu decay and fabrication waste becomes 311.4 kg for Type A fresh reload fuel in Table IV C-9. Uranium-235 in the reload fuel assemblies is correspondingly reduced by an amount that is dependent on the plutonium substitution value.

The fresh reload plutonium composition is based on core calculations[8] for the Browns Ferry Nuclear Power Plant operated by TVA. These calculations consider variations in physics parameters, such as neutron flux, that are not possible in the ORIGEN[9] computer program as it currently exists. Because the calculations are based on core design methods that have been experimentally verified, uncertainty relative to the isotopic composition of the plutonium in each of the reload types is reduced.[5] The resultant small differences between the composition of plutonium recovered and recycled two years later are evident by comparing, for example, the composition of the plutonium in spent UO_2 fuel (Table IV C-9) to the composition of plutonium in fresh reload Type A fuel.

After four annual refuelings with recycle plutonium in accordance with Table IV C-10, the amount of mixed oxide fuel substituted for enriched fuel is 28% as shown in Figure IV C-26. At the end of the fourth year the spent fuel contains residual plutonium in the MOX fuel rods of Type A reload, plus the net plutonium generated in the

Table IV C-9

COMPOSITION OF 1.15 SGR MODEL RELOADS, BWR*

172 RELOAD FUEL ASSEMBLIES

ISOTOPIC WEIGHT OF FRESH FUEL WHEN CHARGED INTO CORE

(kg)

Reload Type	U	A	B	C	D	E
$^{234}\mathrm{U}$	10.1	7.7	7.6	7.0	7.0	6.8
$^{235}\mathrm{U}$	838.5	665.5	653.2	610.7	610.8	592.7
$^{236}\mathrm{U}$	90.3	65.0	63.2	57.1	57.1	54.5
$^{238}\mathrm{U}$	31,315	31,200	31,220	31,085	31,095	31,000
Total U	32,250	31,940	31,940	31,760	31,770	31,660
$^{238}\mathrm{Pu}$	-	6.0	6.0	13.2	13.3	20.3
$^{239}\mathrm{Pu}$	-	178.1	180.2	227.0	227.4	246.7
$^{240}\mathrm{Pu}$	-	78.4	76.9	133.6	131.7	172.0
$^{241}\mathrm{Pu}$	-	34.4	34.4	65.3	69.7	89.8
$^{242}\mathrm{Pu}$	-	14.6	13.8	41.2	40.6	61.2
Total Pu	-	311.4	311.2	480.3	482.8	589.3

ISOTOPIC WEIGHT OF SPENT FUEL WHEN DISCHARGED FROM THE CORE

(kg)

$^{234}\mathrm{U}$	5.00	3.97	3.89	3.76	3.78	3.81
$^{235}\mathrm{U}$	233	193	190	182	183	180
$^{236}\mathrm{U}$	186	142	139	127	127	122
$^{237}\mathrm{U}$	0.27	0.21	0.21	0.19	0.19	0.18
$^{238}\mathrm{U}$	30,600	30,500	30,500	30,400	30,400	30,300
$^{237}\mathrm{Np}$	20.9	16.2	15.9	14.7	14.6	14.1
$^{239}\mathrm{Np}$	1.63	1.66	1.67	1.63	1.62	1.60
$^{238}\mathrm{Pu}$	7.46	10.9	10.8	16.0	16.3	21.4
$^{239}\mathrm{Pu}$	168	222	225.1	241	243	250
$^{240}\mathrm{Pu}$	66.5	111	111.1	136	137	155
$^{241}\mathrm{Pu}$	29.9	60.3	60.0	79.8	80.7	94.9
$^{242}\mathrm{Pu}$	9.23	27.6	27.2	46.9	47.4	62.3
Total Pu	281	431.8	434.2	519.7	524.4	583.6

*Refer to Table IV C-10 and Figure IV C-26.

Table IV C-9

ISOTOPIC WEIGHT IN SPENT FUEL, (Cont.)

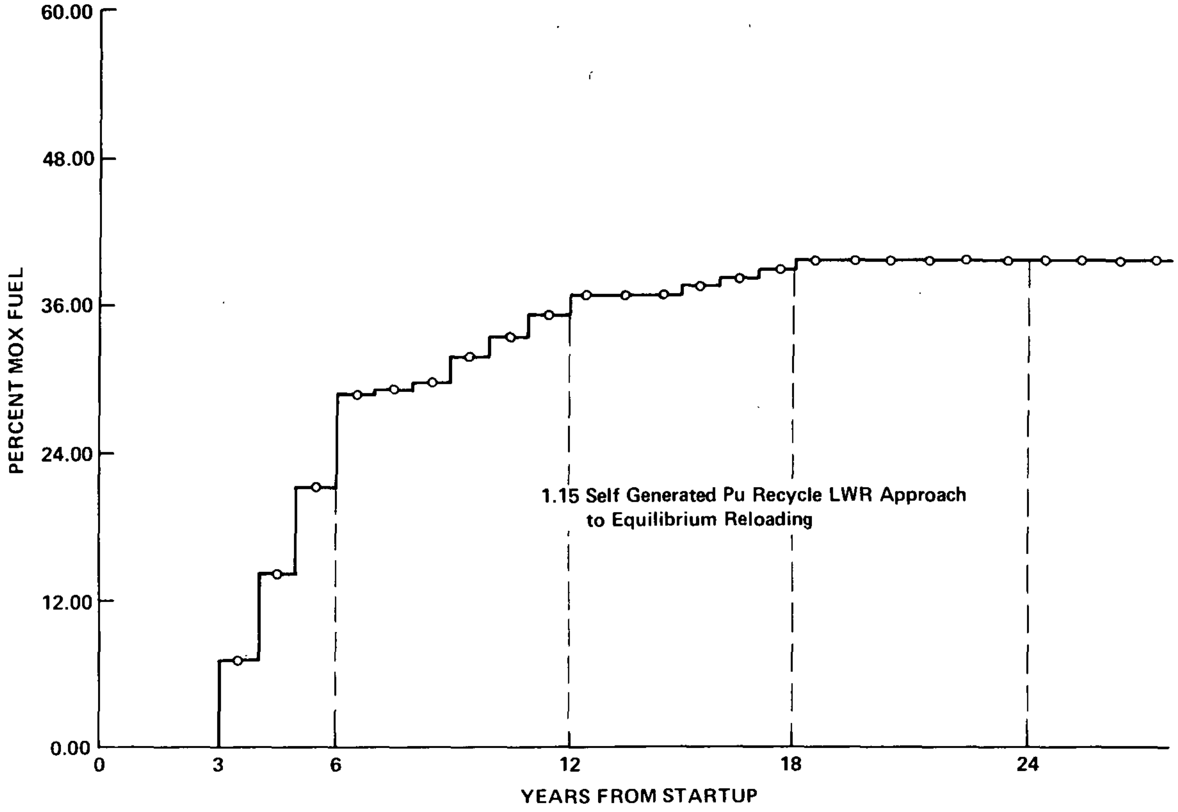
(kg)

	U	A	B	C	D	E
^{241}Am	1.06	3.20	3.17	5.02	5.26	6.38
^{242}Am	0.002	0.19	0.19	0.31	0.32	0.40
^{243}Am	2.32	13.3	13.1	25.7	25.3	36.4
^{242}Cm	0.298	1.00	1.00	1.47	1.49	1.76
^{244}Cm	0.63	7.67	7.53	14.9	14.1	20.5
^{245}Cm	0.038	0.88	0.87	1.69	1.55	2.23
^{246}Cm	0.0038	0.0	0.0	0.17	0.15	0.21

Table IV C-10 Chronology of Reactor Reloads[5]

YEAR	RELOAD	RELOAD TYPE	BWR REACTOR CONFIGURATION	RELOAD TO REPROCESSING	TYPES TO FUEL FABRICATION
	STARTUP	UUUU			
1			UUUU		
	1	U		U	–
2			UUUU		
	2	U		U	A
3			UUUU		
	3	A		U	A
4			AUUU		
	4	A		U	A
5			AAUU		
	5	A		U	B
6			AAAU		
	6	B		U	B
7			BAAA		
	7	B		A	B
8			BBAA		
	8	B		A	C
9			BBBA		
	9	C		A	C
10			CBBB		
	10	C		B	C
11			CCBB		
	11	C		B	D
12			CCCB		
	12	D		B	D
13			DCCC		
	13	D		C	D
14			DDCC		
	14	D		C	E
15			DDDC		
	15	E		C	E
16			EDDD		
	16	E		D	E
17			EEDD		
	17	E		D	E
18			EEED		
	18	E		D	E
19			EEEE		
	19	E		E	E
20			EEEE		

REFER TO TABLE IV C-9
FOR A, B, C, D, E & U



(See Reference 5)

Figure IV C-26 Percent MOX vs. Years From Startup of Uranium Fueled Core

UO_2-only and MOX rods during normal reactor operation. A comparison of spent fuel Types U and A in Table IV C-9 shows the substantial increase in the amount of plutonium present in the spent fuel and attributed to plutonium recycle. When the plutonium from Type A spent fuel is recovered, fabricated with 15% additonal plutonium into mixed oxide fuel rods and recycled in the same LWR two years later (Figure IV C-26) the mixed oxide fuel in the core increases noticeably as shown by the increase in plutonium for reload Type C fuel (Table IV C-9) and the core after the sixth plutonium reload (Figure IV C-26). Continued in accordance with Table IV C-10 and the plutonium compositions presented in Table IV C-9 equilibrium conditions are approximated for the 1.15 SGR in 16 years as shown in Figure IV C-26. Thereafter, the amount of additional plutonium added to the recycle plutonium is just sufficient to maintain the annual total plutonium charged to the 1.15 SGR model reactor at a constant value 15% higher than the equilibrium SGR value. This addition amounts to about 4.3% of the plutonium contained in the model 1.15 SGR spent fuel. It should be noted that the amounts of additional plutonium added to the plutonium recovered from one LWR have been chosen so that over the entire reactor lifetime the fresh MOX fuel of the 1.15 SGR reactor contains, year by year, at least 15% more MOX fuel than an identical reactor that utilizes only its own self-generated plutonium as it becomes available.

Table IV C-10 shows the chronology of the assumed 1.15 SGR reload types from loading through reprocessing and fuel fabrication. Each of the reload types A through D is used for 3 years successively to replace spent fuel assemblies until equilibrium conditions are approached with reload Type E. The calculational model results in a greater concentration of aged plutonium than will occur, if as anticipated, the plutonium from all LWR's is blended together, in effect diluting the plutonium recovered from older plants with the plutonium from the newer plants that are started up each year. The model 1.15 SGR, as described, yields the most adverse composition of plutonium relative to toxicity that can result by recycling plutonium recovered from LWR's at the GESMO model reactor rates.

The calculational results show that 16 years are required to approximate the equilibrium inventory of radionuclides in the core and spent fuel when the plutonium is recycled as described. However, an operating uranium LWR may be converted to recycle plutonium and achieve equivalent GESMO model LWR equilibrium conditions in 3 or 4 years, or new LWR's may begin operating using the model reactor plutonium concentrations of 1.8% in the as charged heavy metal fuel in the initial core without exceeding the environmental values calculated for the GESMO model LWR.

4.3 Difference Between Nuclide Inventories of the GESMO Model and UO_2-Only BWR's

Typically,the nuclear core is refueled annually. The most depleted fuel assemblies (spent fuel), normally about 1/4 of the fuel assemblies in a BWR core, are removed and replaced with reload fuel assemblies. About 172 spent fuel assemblies in a 1,000 MWe BWR are replaced annually with 172 new reload fuel assemblies.

4.3.1 Reload Fuel Assemblies (New Fuel) - Plutonium and Uranium Content

It can be seen in Table IV C-11 that a typical reload of 172 bundles contains 32.2 metric tons (MT) of uranium, of which 0.84 MT is ^{235}U. The average enrichment of equilibrium reload BWR assemblies is, therefore, 2.6%. If the GESMO model BWR fuel assemblies are used instead of standard BWR fuel assemblies, the isotopic inventories of uranium and plutonium in the new fuel assemblies would be as shown in Table IV C-11, Column 2 and 172 reload fuel assemblies would contain 0.593 MT of ^{235}U, 0.246 MT of ^{239}Pu, and 0.09 MT of ^{241}Pu. The total fissile fuel, 0.929 MT $^{235}U + Pu_f$, is noticeably greater than 0.840 MT of ^{235}U in BWR reload fuel to satisfy equivalent reactivity and fuel lifetime requirements. About 36% of the total fissile fuel is recycle plutonium. It can also be shown, because the recycle plutonium is mixed with natural uranium containing 0.7% ^{235}U, that about 40% of the fuel rods are mixed oxide rods and that the average fissile fuel in the MOX rods ($PU_f + {}^{235}U$) is 3.3%, in contrast to 2.6% for the enriched fuel rods. If the average uranium enrichment in the model reactor UO_2 rods is less than 2.6%, because the UO_2 rods with the highest enrichment in the fuel assembly interior are preferentially replaced with Pu_f, there will be fewer MOX rods and the Pu_f concentrations will be greater than the value that has been calculated here for the equilibrium 1.15 SGR. Column 2 also shows that the total weight of plutonium, 589 kilograms, is 1.8% of the total heavy metal, 32,200 kilograms, as noted earlier.

4.3.2 Spent Fuel Assemblies - Nuclide Inventory

Comparisons of spent fuel from an equilibrium BWR using enriched uranium only with the fuel discharged from the GESMO model LWR in Table IV C-12* shows that the spent GESMO model fuel contains about twice the amount of plutonium in the spent BWR fuel assemblies, about seven times as much ^{242}Pu, significant increases in the actinides such as americium, a slight increase in iodine, and a slight decrease in krypton. It is also evident that the various radioisotopes of americium and curium decay very slowly. The noble gases and iodine, except for ^{129}I and ^{85}Kr, decay to insignificance during the period that fuel assemblies must be retained at the reactor site in the spent fuel storage pool.

4.3.3 Core Nuclide Inventory

The comparisons in Tables IV C-11 and IV C-12* are important in assessing potential fuel handling and transportation impacts. The comparisons in Table IV C-13* are related to the environmental impacts of reactor operation with recycle plutonium.

Table IV C-13 includes the 54 radionuclides[10] that are most important in the calculation of radiation doses to the public following hypothetical accidental releases of radioactivity to the atmosphere. The radionuclide inventories are based on average burnups of one-fourth, one-half, three-fourths and full burnups (27,500 MWd/MTHM) just prior to reactor refueling, when the oldest and most irradiated one-fourth of the fuel in the core is replaced. The average power density is assumed to

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*One of the tables at the end of paragraph 4.3.3.

Table IV C-11

172 RELOAD FUEL ASSEMBLIES - BWR[5]

Heavy Metal Fuel Composition

Grams

Nuclides	(1) Uranium Only*	(2) GESMO Model**
^{234}U	1.01×10^4	6.77×10^3
^{235}U	8.39×10^5	5.93×10^5
^{236}U	9.03×10^4	5.45×10^4
^{238}U	3.13×10^7	3.10×10^7
^{238}Pu	-	2.03×10^4
^{239}Pu	-	2.46×10^5
^{240}Pu	-	1.72×10^5
^{241}Pu	-	8.98×10^4
^{242}Pu	-	6.12×10^4
Total	3.22×10^7	3.22×10^7

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*^{234}U and ^{236}U content of 0.0312 wt% and 0.28 wt% respectively, taken from Figures 3 and 4 of Minor Isotopes Committee Report Draft of October 28, 1974, for BWR reload cores.

**Assumes blending old plutonium that has been recycled one, two, and three times with new plutonium formed in uranium.

Table IV C-12

NUCLIDE INVENTORY[5]

172 SPENT FUEL ASSEMBLIES, BWR

Grams

Nuclide	Uranium Only		GESMO MODEL	
	At Discharge	120-day Decay	At Discharge	120-day Decay
^{230}Th	5.03×10^{-2}	5.49×10^{-2}	3.62×10^{-2}	3.97×10^{-2}
^{232}Th	1.67×10^{-2}	1.84×10^{-2}	1.07×10^{-2}	1.18×10^{-2}
^{231}Pa	2.44×10^{-2}	NC	1.71×10^{-2}	NC
^{232}U	1.47×10^{-2}	1.68×10^{-2}	1.07×10^{-2}	1.23×10^{-2}
^{233}U	1.70×10^{-1}	NC	1.31×10^{-1}	NC
^{234}U	5.00×10^{3}	NC	3.81×10^{3}	NC
^{235}U	2.33×10^{5}	NC	1.80×10^{5}	NC
^{236}U	1.86×10^{5}	NC	1.22×10^{5}	NC
^{237}U	2.68×10^{2}	2.08×10^{-3}	1.83×10^{2}	3.61×10^{-3}
^{238}U	3.06×10^{7}	NC	3.03×10^{7}	NC
^{237}Np	2.09×10^{4}	2.11×10^{4}	1.41×10^{4}	1.43×10^{4}
^{239}Np	1.63×10^{3}	1.85×10^{-3}	1.60×10^{3}	2.83×10^{-2}
^{236}Pu	2.79×10^{-2}	2.58×10^{-2}	2.08×10^{-2}	1.93×10^{-2}
^{238}Pu	7.46×10^{3}	7.60×10^{3}	2.12×10^{4}	2.21×10^{4}
^{239}Pu	1.68×10^{5}	NC	2.50×10^{5}	NC
^{240}Pu	6.65×10^{4}	NC	1.55×10^{5}	NC
^{241}Pu	2.99×10^{4}	2.95×10^{4}	9.50×10^{4}	9.35×10^{4}
^{242}Pu	9.23×10^{3}	NC	6.24×10^{4}	NC
^{241}Am	1.06×10^{3}	1.52×10^{3}	6.72×10^{3}	8.18×10^{3}
^{242m}Am	3.85×10^{1}	NC	4.19×10^{2}	NC
^{242}Am	2.08	4.6×10^{-4}	9.74	5.02×10^{-3}
^{243}Am	2.32×10^{3}	NC	3.42×10^{4}	NC
^{242}Cm	2.98×10^{2}	1.80×10^{2}	1.80×10^{3}	1.09×10^{3}
^{243}Cm	2.23×10^{0}	NC	1.09×10	NC
^{244}Cm	6.30×10^{2}	6.22×10^{2}	1.85×10	NC

Table IV C-12

NUCLIDE INVENTORY[5] (Cont.)

Grams

Nuclide	Uranium Only		GESMO MODEL	
	At Discharge	120-day Decay	At Discharge	120-day Decay*
^{245}Cm	3.77×10	NC	1.99×10^{3}	NC
^{246}Cm	3.82×10^{0}	NC	1.97×10^{2}	NC
^{247}Cm	4.46×10^{-2}	NC	3.09×10^{0}	NC
^{248}Cm	2.68×10^{-3}	NC	2.39×10^{-2}	NC
^{249}Bk	2.58×10^{-5}	1.98×10^{-5}	2.98×10^{-3}	2.29×10^{-3}
^{249}Cf	4.88×10^{-6}	1.09×10^{-5}	7.96×10^{-4}	1.49×10^{-3}

Curies

Nuclide	Uranium Only		GESMO MODEL	
	At Discharge	120-day Decay	At Discharge	120-day Decay*
^{3}H	1.84×10^{4}	1.80×10^{4}	2.13×10^{4}	2.09×10^{4}
^{14}C	1.35×10^{-7}	NC	1.07×10^{-7}	NC
^{133}Xe	3.13×10^{7}	5.38×10^{0}	3.23×10^{7}	5.54×10^{0}
^{129}I	9.87×10^{-1}	9.93×10^{-1}	1.13×10^{0}	1.14×10^{0}
^{131}I	1.69×10^{7}	5.66×10^{2}	1.78×10^{7}	5.96×10^{2}
^{85}Kr	2.77×10^{5}	2.71×10^{5}	2.46×10^{5}	2.40×10^{5}
^{90}Sr	1.93×10^{6}	1.92×10^{6}	1.69×10^{6}	1.67×10^{6}
^{134}Cs	5.25×10^{6}	4.70×10^{6}	5.35×10^{6}	4.79×10^{6}
^{137}Cs	2.78×10^{6}	2.75×10^{6}	2.92×10^{6}	2.90×10^{6}
Total	2.70×10^{9}	1.07×10^{8}	2.79×10^{9}	1.12×10^{8}

ELEMENTS

Curies

Nuclide	Uranium Only		GESMO MODEL	
	At Discharge	120-day Decay	At Discharge	120-day Decay*
Kr	6.90×10^{7}	2.71×10^{5}	6.48×10^{7}	2.40×10^{5}
Xe	1.60×10^{8}	3.73×10^{2}	1.67×10^{8}	3.93×10^{2}
I	2.27×10^{8}	5.67×10^{2}	2.34×10^{8}	5.97×10^{2}

NC no change

*For comparative decay. Absolute values are sensitive to the number of MOX rods per bundle and amount of plutonium per rod.

Table IV C-13

NUCLIDE INVENTORY PRIOR TO REFUELING[5]

BWR CORE - 688 FUEL ASSEMBLIES

Includes 54 Biologically Significant Nuclides*

Radionuclide	Half Life (days)	BWR Curies $\times 10^{-8}$		BWR (grams)	
		U Only	GESMO MODEL	U Only	GESMO MODEL
^{3}H	4.48×10^{3}	0.000355	0.000387	3.66	4.00
^{4}He	stable	0.0	0.0	14.8	2.28×10^{2}
^{14}C	2.09×10^{6}	3.38×10^{-7}	2.68×10^{-7}	7.58	6.01
^{58}Co	71.0	0.0089	0.0097	28.2	30.7
^{60}Co	1.92×10^{3}	0.0048	0.0041	4.27×10^{2}	3.62×10^{2}
^{85}Kr	3.95×10^{3}	0.0078	0.0067	2.01×10^{3}	1.71×10^{3}
^{85m}Kr	0.183	0.22	0.19	2.61	2.29
^{87}Kr	0.0528	0.42	0.37	1.50	1.30
^{88}Kr	0.117	0.62	0.54	4.87	4.25
^{86}Rb	18.7	0.00025	0.00018	0.305	0.220
^{89}Sr	52.1	0.72	0.62	2.57×10^{3}	2.19×10^{3}
^{90}Sr	1.00×10^{4}	0.054	0.045	3.79×10^{4}	3.17×10^{4}
^{91}Sr	0.403	1.04	0.92	29.2	25.7
^{90}Y	2.67	0.055	0.046	10.1	8.45
^{91}Y	59.0	0.92	.80	3.75×10^{3}	3.27×10^{3}
^{95}Zr	65.2	1.2	1.2	5.76×10^{3}	5.56×10^{3}
^{97}Zr	0.71	1.4	1.4	75.6	75.0
^{95}Nb	35.0	1.2	1.2	2.95×10^{3}	2.84×10^{3}
^{99}Mo	2.8	1.6	1.6	3.30×10^{2}	3.31×10^{2}
^{99m}Tc	0.25	1.4	1.4	25.7	25.8
^{103}Ru	39.5	1.0	1.1	-	-
^{105}Ru	0.185	0.81	0.98	12.1	14.6
^{106}Ru	3.66×10^{2}	0.34	0.47	1.02×10^{4}	1.39×10^{4}
^{105}Rh	1.50	0.61	0.76	73.5	91.2
^{127}Te	0.391	0.062	0.072	2.36	2.72
^{127m}Te	1.09×10^{2}	0.011	0.013	1.17×10^{2}	1.38×10^{2}
^{129}Te	0.048	0.32	0.34	1.50	1.62

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*Based on hypothetical accidents involving core melting of uranium-only fueled LWR[10]

Table IV C-13 (Cont.)

NUCLIDE INVENTORY PRIOR TO REFUELING[5]

Radionuclide	Half Life (days)	BWR Curies x 10^{-8}		BWR (grams)	
		U Only	GESMO MODEL	U Only	GESMO MODEL
^{129m}Te	34.0	0.048	0.052	1.62×10^2	1.75×10^2
^{131m}Te	1.25	0.13	0.13	16.0	16.7
^{132}Te	3.25	1.2	1.2	3.90×10^2	4.02×10^2
^{127}Sb	3.88	0.065	0.075	24.6	28.3
^{129}Sb	0.179	0.34	0.37	6.02	6.49
^{131}I	8.05	0.84	0.87	6.76×10^2	7.03×10^2
^{132}I	0.0958	1.2	1.3	11.9	12.3
^{133}I	0.875	1.7	1.7	1.48×10^2	1.48×10^2
^{134}I	0.0366	1.9	1.9	7.13	7.06
^{135}I	0.280	1.5	1.5	42.6	42.3
^{133}Xe	5.28	1.6	1.6	8.50×10^2	8.51×10^2
^{135}Xe	0.384	0.44	0.60	17.5	23.6
^{134}Cs	7.50×10^2	0.114	0.108	8.76×10^3	8.28×10^3
^{136}Cs	13.0	0.042	0.051	56.1	68.4
^{137}Cs	1.10×10^4	0.072	0.074	8.30×10^4	8.47×10^4
^{140}Ba	12.8	1.5	1.4	2.00×10^3	1.96×10^3
^{140}La	1.67	1.5	1.5	2.72×10^2	2.67×10^2
^{141}Ce	32.3	1.3	1.3	4.56×10^3	4.4×10^3
^{143}Ce	1.38	1.3	1.3	1.95×10^2	1.89×10^2
^{144}Ce	2.84×10^2	0.84	0.79	2.63×10^4	2.47×10^4
^{143}Pr	13.7	1.2	1.2	1.79×10^3	1.73×10^3
^{147}Nd	11.1	0.6	0.6	7.18×10^2	7.02×10^2
^{230}Th	2.92×10^7	3.01×10^{-11}	2.16×10^{-11}	0.155	0.111
^{232}Th	5.14×10^{12}	4.25×10^{-17}	2.68×10^{-17}	0.0389	0.0245
^{234}Th	24.1	4.12×10^{-7}	4.03×10^{-7}	0.00178	0.00176
^{231}Pa	1.19×10^7	2.59×10^{-11}	1.78×10^{-11}	0.0543	0.0373
^{233}Pa	27.4	3.68×10^{-7}	2.40×10^{-7}	0.00180	0.00117
^{232}U	2.63×10^4	5.78×10^{-9}	4.05×10^{-9}	0.0270	0.0189
^{233}U	5.90×10^7	5.58×10^{-11}	4.10×10^{-11}	0.589	0.433
^{234}U	9.02×10^7	1.63×10^{-6}	1.19×10^{-6}	2.63×10^4	1.92×10^4
^{235}U	2.60×10^{11}	3.42×10^{-8}	2.50×10^{-8}	1.60×10^6	1.21×10^6

Table IV C-13 (Cont.)

NUCLIDE INVENTORY PRIOR TO REFUELING[5]

Radionuclide	Half Life (days)	BWR Curies x 10^{-8}		BWR (grams)	
		U Only	GESMO MODEL	U Only	GESMO MODEL
^{236}U	8.73×10^{9}	4.16×10^{-7}	3.66×10^{-7}	6.56×10^{5}	4.29×10^{5}
^{237}U	6.74	0.939	0.604	1.15×10^{3}	7.40×10^{2}
^{238}U	1.65×10^{12}	4.11×10^{-7}	4.08×10^{-7}	1.23×10^{8}	1.22×10^{8}
^{237}Np	7.79×10^{8}	3.75×10^{-7}	2.43×10^{-7}	4.31×10^{4}	3.45×10^{4}
^{239}Np	2.35	18.5	17.3	7.97×10^{3}	7.42×10^{3}
^{236}Pu	10.5×10^{2}	3.5×10^{-7}	2.18×10^{-7}	5.72×10^{-2}	4.08×10^{-2}
^{238}Pu	3.25×10^{4}	0.0025	0.013	1.49×10^{4}	7.71×10^{4}
^{239}Pu	8.9×10^{6}	0.00036	0.00064	5.80×10^{5}	1.05×10^{6}
^{240}Pu	2.4×10^{6}	0.00038	0.0014	1.72×10^{5}	6.19×10^{5}
^{241}Pu	5.35×10^{3}	0.066	0.27	6.50×10^{4}	3.64×10^{5}
^{242}Pu	1.48×10^{8}	6.38×10^{-7}	1.15×10^{-5}	1.64×10^{4}	2.61×10^{5}
^{241}Am	1.5×10^{5}	0.000065	0.00072	1.89×10^{3}	2.10×10^{4}
^{242m}Am	5.53×10^{4}	6.48×10^{-6}	0.000108	66.6	1.11×10^{3}
^{242}Am	0.669	0.0285	0.284	4.76	35.1
^{243}Am	2.80×10^{6}	6.82×10^{-6}	0.000193	3.55×10^{3}	1.01×10^{5}
^{242}Cm	1.63×10^{2}	0.017	0.154	5.18×10^{2}	4.66×10^{3}
^{244}Cm	6.63×10^{3}	0.00076	0.032	9.40×10^{2}	3.97×10^{4}
^{246}Cm	1.72×10^{6}	1.69×10^{-8}	9.06×10^{-7}	5.46	2.94×10^{2}
^{247}Cm	5.99×10^{9}	5.64×10^{-14}	3.94×10^{-12}	0.0639	4.46
^{248}Cm	1.29×10^{8}	1.60×10^{-13}	1.32×10^{-11}	0.00390	0.322
^{249}Bk	3.15×10^{2}	6.48×10^{-10}	6.54×10^{-8}	3.88×10^{-5}	0.00391
^{249}Cf	1.29×10^{5}	2.66×10^{-13}	3.90×10^{-11}	6.52×10^{-6}	9.55×10^{-4}

be 18.21 MW/MTHM except for the last 60 days of reactor operation. For the first half of this period, power level is reduced to 60%. For the last half of this period reactor power level is 100%. These assumptions are necessary because the inventory of the short lived radionuclides that reach equilibrium during irradiation is directly proportional to the power density. The inventory of the long lived radionuclides is proportional to burnup and is not sensitive to power density at any given exposure.[10] The values presented in Table IV C-13, therefore, are representative of the highest inventory of radionuclides that can occur during normal operation of LWR's fueled with UO_2-fuel only or with the GESMO model mixed oxide fuel. The differences between the UO_2 and GESMO model are readily apparent by comparison of the inventories for each type of core. For example, the inventory of ^{131}I is somewhat larger for the 1.15 SGR core than the UO_2 core but the ^{85}Kr inventory is smaller. The total amount of plutonium (fissile and nonfissile) in the GESMO model reactor is only three times greater than the total amount of plutonium in the uranium-only core. The nonfissile plutonium isotopes in the model core are five times greater than the nonfissile plutonium isotopes in the uranium-only core.

The changes in radioactive releases to the environment during normal and accident conditions due to changes in the core inventory of radionuclides such as those that have been identified for a BWR are considered in Section 5.0, which follows. The release of very small quantities of radioactivity into the primary coolant through the cladding in which the fuel is sealed is evaluated, as well as large radioactive releases that accompany fuel clad failures resulting from postulated design basis accidents.

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5.0 ENVIRONMENTAL IMPACT

Final environmental statements prepared as part of the requirement for a full term operating license consider various aspects of locating a nuclear power plant at a specific site. The final environmental statement assesses the environmental impact and adverse effects of

- Reassignment of agricultural and natural land for the plant facilities
- Fogging effects if cooling towers are used
- Chemical wastes released to bodies of water
- Installation of transmission lines
- Alteration of the benthic population in bodies of water adjacent to the thermal effluent discharge as a result of water temperature elevation
- Creation of a potential source for cold shock mortality of fish attracted to the discharge in the event that the reactor is shut down during cold weather
- Possible mortality of some of the passing drift organisms through entrainment into the cooling system followed by chemical, mechanical, or thermal shock
- Water losses due to cooling tower evaporation
- Discharges of small quantities of radioactive gases and liquids to the environment
- Creation of a very low probability risk of accidental radiation exposure to nearby residents

Changing the fuel composition to include recycle plutonium causes relatively small changes to the fission products formed during reactor power operation. Except for fuel rods with defects in the fuel cladding, all of the fission products are retained within the sealed fuel rod. However, from past reactor experience during normal reactor operation, some defects are expected to occur, and some of the fission product gases will be released from the fuel matrix into the primary reactor coolant. Some of the very low probability accidents such as the design basis loss of coolant accident can cause fuel damage resulting in the escape of a larger amount of the fission product gases into the containment. All of the other aspects of reactor operation with recycle plutonium mixed oxide fuel rods in place of enriched fuel rods remain unchanged, including the probability of reactor accidents. Only the last two items of the preceding list of environmental impacts remain for further analysis.

5.1 Radioactivity Source Terms

The following sections consider the changes in fission product and actinide inventories due to the use of recycle plutonium in place of some of the enriched uranium and the associated changes in radioactive releases to the environment during normal and accident conditions. The consequences of such releases are expressed for reactor comparative evaluation in radiation equivalent units at various locations in the vicinity of the plant. With consideration of both the fission products and actinides, calculations show that the use of recycle plutonium in LWR's does not change significantly the hazards to the public that have been previously determined to be within acceptable limits, and on this basis operation of nuclear power plants with recycle plutonium may be authorized. Risks to the public related to transportation of reload fuel assemblies and spent fuel assemblies containing recycle plutonium are evaluated in CHAPTER IV, Section G, Transportation of Radioactive Materials.

5.1.1 Comparison of Expected Releases of Radioactive Materials From LWR's Using Mixed Oxide Fuels and UO_2 Fuels

The expected releases of radioactive materials in effluents from LWR's using mixed oxide fuels have been compared with those from LWR's using UO_2 fuels. The model used to calculate the releases considers the releases to be proportional to the concentrations of radioactive materials in the reactor coolant and specifies concentrations of radioactive materials in the coolant of reactors fueled with UO_2 fuels. This model is presented in Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors" dated April 1976, NUREG 0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors," and NUREG 0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," both dated April 1976.

Ratios were calculated for the concentrations of fission products and transuranium isotopes in the coolant of reactors fueled with mixed oxide relative to those with UO_2 fuels using the model contained in the Final Environmental Statement Concerning Proposed Rule Making Actions for Operation to Meet the Criterion "As Low As Practicable," WASH-1258, Volume 2. This model considers the leak rate of each isotope into the coolant from fuel having cladding defects to be equal to the product of the cycle-average isotopic inventory in the core, the fraction of fuel having cladding defects, and the escape rate coefficient. The fraction of fuel having cladding defects and the escape rate coefficient are assumed to be the same for the mixed oxide fuels as for the low enrichment uranium oxide fuels. The difference in the concentrations of radioactive materials in the reactor coolant will result only from the different isotopic inventories of radionuclides calculated to be present in the mixed oxide fuels versus the low enrichment uranium oxide fuels. These, in turn, will depend on the relative fission rates of fissile plutonium and ^{235}U in the two cores and the fission product yields of the individual fissile isotopes. The fissioning of the ^{239}Pu that is produced in UO_2 fuels by neutron capture in ^{238}U is accounted for in the calculational model, which includes fission product yields for the thermal neutron fission of ^{239}Pu. The isotopic composition of the fresh and spent fuel and the model reactor core for the

GESMO model reactor are as presented in CHAPTER IV, Section C-4.0, Tables C-11, C-12, and C-13 respectively.

The annual releases of radioactive materials from reactors using UO_2 fuels were calculated using the GALE computer code and the models and parameters given in NUREG 0016 and 0017 previously mentioned. The release rates for mixed oxide fueled reactors were calculated by multiplying the release rates used for UO_2 fuels by the ratio of the reactor coolant concentration for the MOX fueled reactor to that for the UO_2 fueled reactor.

For the UO_2 fueled LWR, the tritium source term is calculated using parameters given in NUREG 0016 and 0017. Table IV C-13 shows that the tritium inventory in the core of the GESMO model reactor is approximately 10% higher than in a UO_2-only fueled reactor. This increase is the result of the higher fission yield for tritium from the fission of ^{239}Pu compared to ^{235}U. For a BWR, it is assumed that all of the tritium released is produced by fission, and the tritium release is therefore increased by 10% for the GESMO model reactor in contrast to the uranium-only BWR.

For a PWR, the neutron activation of boron-10 accounts for production of approximately 33% of the tritium produced and the remainder is due to ternary fission. A calculation similar to that done for the BWR would result in an increase in the production of tritium in a PWR mixed oxide core of 7%.

Source terms for gaseous ^{14}C, ^{41}Ar, and radioactive particulates (principally ^{60}Co) released in gaseous effluents have been included in the evaluations. These source terms are based on a review of current data from operating reactors using UO_2 cores. Because ^{14}C, ^{41}Ar, and ^{60}Co are formed by activation of water, air, and structural materials and are not a direct function of the fissioning isotope, the release rates are considered to be approximately equal to those determined from current plant operating experience.

The parameters used in the source term calculations for operating variables, such as leakage from the primary system, partition factors, radwaste system variables and equipment decontamination factors, are given in Tables IV C-14 and IV C-15. The bases for these parameters are given in NUREG 0016 and 0017.

The models chosen for preparation of the source terms are a BWR, a PWR with once-through steam generators, and a PWR with recirculating U-tube type steam generators. The operating variables used are those given in the American Nuclear Society Standard N237, "Source Term Specification" (May 1976) for BWR's and PWR's. The standard reference LWR's are based on a 3,400-MWt rated power level. Its range of applicability includes the 3,067-MWt rated power level used to comparatively evaluate the GESMO model LWR's and uranium-only LWR's. The differences in power level result in insignificant differences in annual releases. The radwaste systems for each type plant contain equipment and features typical of current operating plants; however, the models are typical cases and the source terms calculated for these models are not directly applicable to a particular operating reactor.

Table IV C-14

SOURCE TERM PARAMETERS, BWR

1.	Reactor Power Level	3400 MWt
2.	Plant Capacity Factor	0.80
3.	Releases of Radioactivity to the Primary Coolant	
	a. Noble Gases	60,000 μCi/sec
	b. Iodine-131	5.0×10^{-3} μCi/g
4.	Tritium Released in Liquid Waste and Gaseous Waste Combined	0.025 Ci/yr/MWt
5.	Weight of Water in Reactor Vessel	3.8×10^{5} lb
6.	Reactor Cleanup Flow Rate	1.3×10^{5} lb/hr
7.	Steam Flow Rate	1.5×10^{7} lb/hr
8.	Fraction of Feedwater Flowing Through Condensate Demineralizers	1.0
9.	Turbine Gland Seal Leakage Rate (Assuming Clean Steam Supplied to Gland Seal)	Iodine Source Term Is Negligible
10.	Partition Factor for Radioiodine in the Reactor Vessel	0.02
11.	Radioiodine Input Rate to Main Condenser Off-Gas System	5 Ci/yr
12.	Startup of Main Condenser (Mechanical) Vacuum Pump, Gaseous Source Term	
	a. ^{133}Xe	2,300 Ci/yr
	b. ^{135}Xe	350 Ci/yr
	c. ^{131}I	0.03 Ci/yr
13.	Holdup Time for Charcoal Delay System	43 Days Xe 2 Days Kr
14.	Iodine Release Fraction Through Charcoal Delay System	negligible
15.	Air In-leakage Rate to the Main Condenser	30 scfm
16.	Decontamination Factors for Evaporators, Miscellaneous Radwaste Evaporator	
	a. All Nuclides Except Iodine	10^{4}
	b. Iodine	10^{3}

(Note: Assumes 24-ton 0°F temperature charcoal delay system, catalytic recombiner, 10-minute holdup time prior to entry into charcoal delay system.)

Table IV C-14 (Cont.)

17. Decontamination Factors (DF) for Demineralizers

	Anion	Cs, Rb	Other Nuclides
Reactor Coolant Cleanup	10	2	10
Condensate	10	2	10
High Purity Waste*	10^2(10)	10(10)	10^2(10)
Low Purity Waste*	10^2(10)	2(10)	10^2(10)

18. Liquid Waste Flows

Source	Gallons Per Day	Fraction of Primary Coolant Concentration
Containment, Auxiliary Building and Fuel Pool Equipment Drains	3,720	0.01
Dry Well Equipment Drain	3,400	1.0
Radwaste Building Equipment Drain	1,060	0.01
Turbine Building Equipment Drain	2,960	0.01
Containment, Auxiliary Building, and Fuel Pool Floor Drains	2,000	0.01
Dry Well Floor Drains	700	1.0
Radwaste Building Floor Drains	1,000	0.01
Turbine Building Floor Drains	2,000	0.01
Laboratory Drains	500	0.02
Chemical Lab Waste	100	0.02
Condensate Demineralizer Regeneration	3,400**	†
Resin Rinse	5,000	0.002
Detergent Waste (Laundry)	450	††

19. Summary of Variable Parameters for Gaseous Radwaste Treatment System

a. Main Condenser/Air Ejector

(1)	Volume:	30 cfm/reactor, assuming 30 cfm in-leakage for a 3-shell condenser
(2)	Treatment:	Catalytic recombiner, 10-minute holdup, 0°F charcoal delay bed (24-ton) and HEPA filter.
(3)	Holdup Time:	Xe 43 days Kr 2 days

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*For an evaporator polishing demineralizer or for the second demineralizer in series, the DF is given in parentheses.

**Based on one demineralizer being regenerated every 3.5 days.

†Activity levels based on radionuclide inventory on resins using DF for condensate demineralizer and a constant regeneration efficiency.

††Activity levels listed in GALE code, see Table 2-32 of NUREG-0016.

Table IV C-14 (Cont.)

b. Turbine Gland Seal System

(1) Volume: 600 scfm/reactor
(2) Treatment: Clean steam
(3) Holdup Time: Not applicable
(4) DF: Not applicable

c. Containment Building Ventilation

(1) Volume: 100,000 scfm/reactor
(2) Treatment: Charcoal Adsorbers
(3) DF, Iodine: 10

d. Turbine Building Ventilation

(1) Volume: 500,000 scfm/reactor
(2) Treatment: Clean steam on valves 24 inches and larger
(3) DF: Not applicable

e. Auxiliary Building Ventilation

(1) Volume: 20,000 scfm/reactor
(2) Treatment: Charcoal Adsorbers
(3) DF, Iodine: 10

f. Radwaste Building Ventilation

(1) Volume: 20,000 scfm/reactor
(2) Treatment: No charcoal adsorbers
(3) DF: Not applicable

20. Core Input Parameters

(1) Mixed Oxide Fueled Core

Weight of uranium in reactor: 145 metric tons
Weight % ^{235}U: 1.84
Weight % fissile Pu: 1.04

(2) Uranium Oxide Fueled Core

Weight of uranium in reactor: 145 metric tons
Weight % ^{235}U 2.7
Weight % fissile Pu: 0

21. Liquid Waste Treatment Variables and Parameters

a. High Purity Wastes

(1) Volume: 16,100 gpd/reactor
(2) Treatment: Filter, two mixed-bed demineralizers in series, 99% recycled
(3) Fraction of Primary Coolant Activity: 0.22
(4) Collection Time (days): 0.6
(5) Holdup Time (days): 1
(6) DF:

I	10^3
Cs	10^2
Others	10^3

Table IV C-14 (Cont.)

b. Low Purity Wastes

(1) Volume: 5,700 gpd/reactor
(2) Treatment: Filter, evaporator, 90% recycled
(3) Fraction of Primary Coolant Activity: 0.13
(4) Collection Time (days): 2
(5) Holdup Time (days): 10
(6) DF: I 10^3
Cs 10^4
Others 10^4

c. Chemical Wastes

(1) Volume: 4000 gpd/reactor
(2) Treatment: Filter, evaporator, polishing demineralizer, 100% discharge of treated effluent
(3) Radioactivity Concentration: This parameter is determined by the GALE code. Includes 3400 gpd/reactor of condensate demineralizer regenerant solutions and 600 gpd of laboratory drains.
(4) Collection Time (days): 2.8
(5) Holdup Time (days): 10
(6) DF: I 10^4
Cs 10^5
Others 10^5

d. Detergent Wastes

(1) Volume: 450 gpd/reactor
(2) Treatment: Filter, 100% discharge
(3) Radioactivity Concentration: 1×10^{-4} μCi/cc
(4) Collection Time (days): 1
(5) Holdup Time (days): 1
(6) DF: I 1
Cs 1
Others 1

Table IV C-15

SOURCE TERM PARAMETERS, PWR

1.	Reactor Power Level	3400 MWt
2.	Plant Capacity Factor	0.80
3.	Fraction of Fuel Releasing Radioactivity to the Primary Coolant (Zircaloy Clad Fuel)	0.12%
4.	Tritium Released in Liquid Waste and Gaseous Waste Combined	0.4 Ci/yr/MWt
5.	Weight of Water in Primary System	5.5×10^{5} lb
6.	Weight of Water in Secondary System	3.7×10^{6} lb
7.	Number of Steam Generators	
	a. Recirculating U-tube Steam Generator	4
	b. Once-Through Steam Generator	2
8.	Weight of Liquid in Each Steam Generator	
	a. Recirculating U-tube Steam Generator	1.1×10^{5} lb
	b. Once-Through Steam Generator	5×10^{4} lb
9.	Steam Flow Rate	1.5×10^{7} lb/hr
10.	Fraction of Feedwater Flowing Through Condensate Demineralizers	0.65
11.	Letdown Flow Rate	75 gpm
12.	Letdown Cation Demineralizer Flow Rate	7.5 gpm
13.	Shim Bleed Flow Rate (Assumes base load operation)	1 gpm
14.	Primary-to-Secondary Leakage	100 lb/day
15.	Leakage into the Containment Building	1%/day of primary coolant noble gas inventory, 0.001%/day of primary coolant iodine inventory
16.	Turbine Building Leakage Rate	1700 lb/hr
17.	Auxiliary Building Leakage Rate	160 lb/day
18.	Steam Generator Blowdown Rate	
	a. Recirculating U-tube Steam Generator	75000 lb/hr
	b. Once-Through Steam Generator	Not Applicable
19.	Primary System Volumes Degassed for Cold Shutdown	2/yr
20.	Holdup time for Primary System Gases	60 days
21.	Containment Building Volume	2×10^{6} cu. ft
22.	Frequency of Containment Purge	
	a. High Volume Purge	4/yr
	b. Low Volume Purge	Continuous 1000-cfm purge

Table IV C-15 (Cont.)

23. Partition Factors for Radioiodine

Steam Generator, Internal Partition	
a. Recirculating U-tube	0.01
b. Once-Through	1
Steam Generator Blowdown Tank Vent	
a. Recirculating U-tube (tank vented to condenser	0
b. Once-Through	Not Applicable
Auxiliary Building Leakage	0.0075
Turbine Building Steam Leakage	1
Main Condenser/Air Ejector	
a. Volatile Iodine Species	0.15
b. Nonvolatile Iodine Species	0

24. Primary Coolant Volatile Iodine Species Fraction — 0.05

25. Decontamination Factors for Evaporators

	All Nuclides Except Iodine	Iodine
Miscellaneous Radwaste Evaporator	10^4	10^3
Boric Acid Evaporator	10^3	10^2

26. Decontamination Factors for Demineralizers

	Anion	Cs, Rb	Other Nuclides
Mixed Bed (Li_3BO_3)	10	2	10
Mixed Bed (H^+OH^-)			
Condensate (Deep Bed)	10	2	10
*Radwaste	10^2(10)	2(10)	10^2(10)
*Cation Bed (H+)	1(1)	10(10)	10(10)

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*For two demineralizers in series, the DF for the second demineralizer is shown in parenthesis. For polishing demineralizers after an evaporator, use value for the second demineralizer in series.

Table IV C-15 (Cont.)

27. Liquid Waste Flow Rates

Source	Gallons Per Day	Fraction of Primary Coolant Activity
Containment Building Sump	40	1
Auxiliary Building Floor Drains	200	0.1
Laboratory Drains	400	0.002
Sample Drains	35	1
Turbine Building Floor Drains	7200	**
Miscellaneous Sources	700	0.01
Shim Bleed	1440	†
Equipment Leaks to Reactor Coolant Drains	240	1
Condensate Demineralizer Regeneration	3400*	*
Detergent Waste (Laundry, Showers, Decontamination)	450	††

28. Summary of Variable for Gaseous Radwaste Treatment System

a. Primary System Gases

(1) Number of Decay Tanks	4
(2) Holdup Time in Decay Tanks	60 days
(3) Fill Time of Decay Tanks	30 days

b. Secondary System Gases

(1) Condenser Air Ejector

(a) Volume: 20 scfm/reactor
(b) Treatment: Charcoal Adsorbers
(c) DF, Iodine: 10

(2) Steam Generator Blowdown Vent

(a) Volume: 1,200 cfm/reactor
(b) Treatment: Vented through Condenser
(c) Partition Factor (PF), Iodine: 0

c. Containment Purge

(1) Low Volume Purge Vent

(a) Volume: 1,000 scfm
(b) Treatment: Charcoal Adsorbers
(c) DF, Iodine: 10

(2) High Volume Purge Vent

(a) Volume: 50,000 scfm
(b) Treatment: Charcoal Adsorbers
(c) DF, Iodine: 10

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*Based on one demineralizer being regenerated every 3.5 days. Activity levels based on radionuclide inventory on resins using DF for condensate demineralizers and constant regeneration efficiency.

**Main Steam activity

†Letdown System effluent activity

††Activity levels listed in GALE Code, see Table 2-20 NUREG 0017.

Table IV C-15 (Cont.)

d. Building Ventilation

(1) Auxiliary Building

(a) Volume: 80,000 scfm/reactor
(b) Treatment: Charcoal Adsorbers
(c) DF, Iodine: 10

(2) Turbine Building

(a) Volume: 500,000 scfm/reactor
(b) Treatment: No Charcoal Adsorbers
(c) DF: Not Applicable

29. Core Input Parameters

(1) Mixed Oxide Fueled Core

Weight of Uranium in Reactor	95 Metric Tons
Weight % ^{235}U	2.03
Weight % Fissile Pu	1.12

(2) Uranium Oxide Fueled Core

Total Weight of Uranium in Reactor	95 Metric Tons
Weight % ^{235}U	3.00
Weight % Fissile Pu	0

30. Liquid Waste Treatment Variables and Parameters

a. Clean Wastes

(1) Volume: 1,700 gpd/reactor. Waste Consists of 1,440 gpd Shim Bleed That Has Been Processed Through Letdown Demineralizers and 240 gpd of Equipment Leaks at PCA
(2) Treatment: Filter, Mixed-Bed Demineralizer, Evaporator, Mixed-Bed Polishing Demineralizer, 90% Recycled
(3) Fraction of Primary Coolant Activity Calculated by GALE Code Based on DF's of Letdown Demineralizer
(4) Collection Time: 20 days
(5) Holdup Time: 20 days
(6) DF I: 10^5
Cs: 2×10^4
Others: 10^6

b. Dirty Wastes

(1) Volume: 1,000 gpd/reactor
(2) Treatment: Filter, Evaporator, Polishing Demineralizer, 90% recycled
(3) Fraction of Primary Coolant Activity 0.1
(4) Collection Time: 2 days
(5) Holdup Time: 4 days
(6) DF I: 10^4
Cs: 10^5
Others: 10^5

c. Steam Generator Blowdown (Recirculating U-tube steam generators only)

(1) Volume: 216,000 gpd/reactor
(2) Treatment: Filter, two Demineralizers in Series, Route to Main Condenser Hotwell
(3) Fraction of Primary Coolant Activity: At Secondary Coolant Activity
(4) Collection Time: 0
(5) Holdup Time: 0
(6) DF I: 10^3
Cs: 10^2
Others: 10^3

Table IV C-15 (Cont.)

d. Condensate Demineralizer Regenerant Wastes

(1) Volume: 3400 gpd/reactor
(2) Treatment: Evaporator, Mixed Bed Polishing Demineralizer, 90% Recycled
(3) % Primary Coolant Activity: This Parameter Determined by GALE Code. Assumes Regeneration of One Demineralizer Every 3.5 days With 100% Regeneration Efficiency.
(4) Collection Time: 2.5 days
(5) Holdup Time: 5 days
(6) DF I: 10^4
Cs: 10^5
Others: 10^5

e. Turbine Building Floor Drains

(1) Treatment: 100% Discharged Without Treatment
(2) Volume: 7,200 gpd/reactor
(3) Fraction of Primary Coolant Activity - At Main Steam Activity

f. Detergent Waste

(1) Treatment: Filter, 100% Discharged
(2) Volume: 450 gpd/reactor
(3) Radioactivity Concentration: 1×10^{-4} μCi/gm
(4) Collection Time: 1 day
(5) Holdup Time: 1 day
(6) DF I: 1
Cs: 1
Others: 1

5.1.2 Tables - Summary of Source Terms

Tables IV C-16 through IV C-21 summarize the liquid and gaseous source terms. Each table shows the calculated releases for a MOX fueled GESMO model LWR, with the corresponding calculated releases for a UO_2 fueled LWR with parallel operating characteristics and parameters.

5.2 Radiological Effects of Normal Operations

The introduction of mixed oxide fuels into light water cooled power reactors requires the examination of the radiological effects that might result as a consequence of normal operations. In this section an examination is made of the effects of mixed oxide fuel utilization in three different reactor systems, and a comparison of the results obtained is made with those that would exist were the reactors fueled with uranium oxide.

Estimates of doses from radionuclides released to the atmosphere and to the hydrosphere were made for a site with two 3,400 MWt power reactors. Refer to CHAPTER IV, Section J, Appendix A, for the definition of dose as used in this analysis. For the boiling water reactor site, a site boundary distance of 1,300 meters was assumed with an atmospheric dispersion factor of 1.0×10^{-6} sec/m^3. For the pressurized water reactor site, a site boundary distance of 1,000 meters was assumed with an atmospheric dispersion factor of 1.5×10^{-6} sec/m^3. Other parameters describing the region surrounding the sites were assumed to be the same for both types of reactors. Dispersion of radionuclides released as liquid effluent was estimated by assuming that the reactors are located on a river at the start of a large watershed (e.g., the Mississippi River watershed). Use of river water that would result in radiation doses to persons was assumed to exist at selected intervals so that individual and integrated population doses could be estimated. Some site and reactor characteristics are listed in Table IV C-22. Details concerning the overall dose modeling are given in CHAPTER IV, Section J, Appendix A.

5.2.1 Radiation Doses to Species Other Than Man

The pathways by which biota other than man may receive radiation doses in the vicinity of a nuclear power station are shown in Figure IV C-27*. Two comprehensive reports[1,2] concerned with radioactivity in the environment and these pathways contain a more detailed explanation of the subjects that are to be discussed. Although no guidelines have been established for desirable limits for radiation exposure to species other than man, it is generally agreed that the limits established for humans are also conservative for these species.[3]

5.2.1.1 Radioactivity in the Environment

Aquatic organisms living in water containing radionuclides discharged from the two uranium oxide or mixed oxide fueled reactors described here could be subjected to radiation in their environment. The maximum doses to aquatic organisms could be delivered to

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*See page IV C-110

Table IV C-16

CALCULATED ANNUAL RELEASE OF RADIOACTIVE MATERIALS IN LIQUID EFFLUENTS FOR 1,000 MWe BWR, Ci/Yr*

Nuclide	GESMO Model 1.15 SGR Mixed Oxide Fuel	Uranium Only Fuel
	Corrosion & Activation Products	
^{24}Na	0.00015	0.00023
^{32}P	0.00014	0.00014
^{51}Cr	0.0043	0.0054
^{54}Mn	0.0015	0.0011
^{55}Fe	0.00095	0.0018
^{59}Fe	0.00003	0.00004
^{58}Co	0.006	0.0043
^{60}Co	0.0077	0.0094
^{64}Cu	0.00041	0.00062
^{65}Zn	0.00027	0.00034
^{69m}Zn	0.00004	0.00005
^{69}Zn	0.00004	0.00005
^{187}W	0.00001	0.00001
^{239}Np	0.00062	0.00057
	Fission Products	
^{89}Sr	0.00011	0.00014
^{90}Sr	0.00001	0.00001
^{90}Y	0.00001	0.00001
^{91}Sr	0.00004	0.00005
^{91m}Y	0.00002	0.00003
^{91}Y	0.00008	0.0001
^{92}Y	0.00001	0.00001
^{93}Y	0.00004	0.00005
^{95}Zr	0.00001	0.00001
^{95}Nb	0.00001	0.00001
^{99}Mo	0.00019	0.00019

Table IV C-16 (Cont.)

^{99m}Tc	0.00024	0.00024
^{103}Ru	0.0002	0.00017
^{103m}Rh	0.00002	0.00002
^{105}Rh	0.00003	0.00002
^{129m}Te	0.00006	0.00005
^{129}Te	0.00003	0.00003
^{131}I	0.28	0.26
^{133}I	0.0008	0.0008
^{134}Cs	0.014	0.013
^{135}I	0.00009	0.00009
^{136}Cs	0.00005	0.00003
^{137}Cs	0.025	0.024
^{137m}Ba	0.00013	0.00013
^{140}Ba	0.00024	0.00025
^{140}La	0.00025	0.00026
^{141}Ce	0.00004	0.00004
^{143}Pr	0.00003	0.00003
Total Except Tritium	∿ 0.3	∿ 0.3
Tritium	47	43

---

*Annual release of a nuclide not shown is less than 10^{-5} curie.

Table IV C-17

CALCULATED ANNUAL RELEASE OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS FOR 1,000 MWe BWR, Ci/Yr*

Nuclide	GESMO Model 1.15 SGR Mixed Oxide Fuel	Uranium Only Fuel
Noble Gases		
^{41}Ar	25	25
^{83m}Kr	*	*
^{85m}Kr	120	150
^{85}Kr	210	290
^{87}Kr	150	200
^{88}Kr	190	240
^{89}Kr	a	*
^{131m}Xe	19	18
^{133m}Xe	*	*
^{133}Xe	3200	3200
^{135m}Xe	750	740
^{135}Xe	1100	1100
^{137}Xe	*	*
^{138}Xe	1400	1400
Total Noble Gases	7100	7300
Others		
^{131}I	0.33	0.3
^{133}I	1.1	1.1
^{14}C	9.5	9.5
^{3}H	47	43
Particulates	0.25	0.25

---

*Annual release less than 1 curie.

Table IV C-18

CALCULATED RELEASE OF RADIOACTIVE MATERIALS IN LIQUID EFFLUENTS FOR 1,000 MWe PWR WITH U-TUBE STEAM GENERATORS, Ci/Yr*

Nuclide	GESMO Model 1.15 SGR Mixed Oxide Fuel	Uranium Only Fuel
	Corrosion & Activation Products	
^{51}Cr	0.00005	0.00008
^{54}Mn	0.00079	0.001
^{55}Fe	0.00004	0.00007
^{59}Fe	0.00004	0.00005
^{58}Co	0.0069	0.0047
^{60}Co	0.0072	0.0088
^{239}Np	0.00003	0.00004
	Fission Products	
^{83}Br	0.00005	0.00007
^{86}Rb	0.00001	0.00002
^{89}Sr	0.00002	0.00002
^{91}Y	0.00006	0.00008
^{99}Mo	0.0057	0.0057
^{99m}Tc	0.0052	0.0052
^{127m}Te	0.00001	0.00001
^{127}Te	0.00002	0.00002
^{129m}Te	0.00006	0.00005
^{129}Te	0.00004	0.00004
^{130}I	0.00018	0.00019
^{131m}Te	0.00006	0.00006
^{131}Te	0.00001	0.00001
^{131}I	0.054	0.052
^{132}Te	0.00075	0.00073
^{132}I	0.0024	0.0023
^{133}I	0.042	0.043
^{134}I	0.00002	0.00002
^{134}Cs	0.026	0.025
^{135}I	0.011	0.011
^{136}Cs	0.0026	0.0018
^{137}Cs	0.034	0.033
^{137m}Ba	0.0085	0.0084
Total Except Tritium	∿ 0.2	∿ 0.2
Tritium	260.	240.

---

*Annual release of a nuclide not shown is less than 10-5 curie.

Table IV C-19

CALCULATED RELEASE OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS FOR 1,000 MWe PWR WITH U-TUBE STEAM GENERATORS, Ci/Yr*

Nuclide	GESMO Model 1.15 SGR Mixed Oxide Fuel	Uranium Only Fuel
Noble Gases		
^{41}Ar	25	25
^{83m}Kr	1	1
^{85m}Kr	13	16
^{85}Kr	340	470
^{87}Kr	3	3
^{88}Kr	17	23
^{89}Kr	*	*
^{131m}Xe	87	82
^{133m}Xe	120	120
^{133}Xe	12000	12000
^{135m}Xe	*	*
^{135}Xe	90	86
^{137}Xe	*	*
^{138}Xe	*	*
Total	13000	13000
Others		
^{131}I	0.027	0.025
^{133}I	0.023	0.023
^{14}C	8	8
^{3}H	1200	1100
Particulates	0.06	0.06

---

*Annual release less than 1 Curie.

Table IV C-20

CALCULATED ANNUAL RELEASE OF RADIOACTIVE MATERIALS IN LIQUID EFFLUENTS FOR 1,000 MWe PWR WITH ONCE-THROUGH STEAM GENERATORS, Ci/Yr

Nuclide	GESMO Model 1.15 SGR Mixed Oxide Fuel	Uranium Only Fuel
	Corrosion & Activation Products	
^{51}Cr	0.0001	0.00017
^{54}Mn	0.00082	0.001
^{55}Fe	0.00008	0.00015
^{59}Fe	0.00007	0.00009
^{58}Co	0.008	0.0055
^{60}Co	0.0073	0.0089
^{239}Np	0.00008	0.0001
	Fission Products	
^{83}Br	0.00005	0.00007
^{86}Rb	0.00001	0.00002
^{89}Sr	0.00003	0.00004
^{91}Sr	0.00003	0.00004
^{91m}Y	0.00002	0.00002
^{91}Y	0.00014	0.00017
^{99}Mo	0.034	0.035
^{99m}Tc	0.034	0.034
^{127m}Te	0.00003	0.00002
^{127}Te	0.00006	0.00005
^{129m}Te	0.00014	0.00013
^{129}Te	0.0001	0.00009
^{130}I	0.00012	0.00013
^{131m}Te	0.00008	0.00008
^{131}Te	0.00001	0.00001
^{131}I	0.029	0.028
^{132}Te	0.0018	0.0018
^{132}I	0.003	0.0029
^{133}I	0.027	0.027
^{134}I	0.00004	0.00004
^{134}Cs	0.02	0.02
^{135}I	0.0086	0.0088
^{136}Cs	0.0032	0.0022
^{137}Cs	0.03	0.029
^{137m}Ba	0.0047	0.0046
Total Except Tritium	∿ 0.2	∿ 0.2
Tritium	260	240

Table IV C-21

CALCULATED ANNUAL RELEASE OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS FOR 1,000 MWe PWR WITH ONCE-THROUGH STEAM GENERATORS, Ci/Yr*

Nuclide	GESMO Model 1.15 SGR Mixed Oxide Fuel	Uranium Only Fuel
^{41}Ar	25	25
^{83m}Kr	1	1
^{85m}Kr	13	16
^{85}Kr	340	470
^{87}Kr	3	3
^{88}Kr	17	23
^{89}Kr	*	*
^{131m}Xe	87	82
^{133m}Xe	120	120
^{133}Xe	12000	12000
^{135m}Xe	*	*
^{135}Xe	90	86
^{137}Xe	*	*
^{138}Xe	*	*
Total	13000	13000
Others		
^{131}I	0.027	0.026
^{133}I	0.024	0.024
^{14}C	8	8
^{3}H	1200	1100
Particulates	0.06	0.06

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*Annual release less than 1 Curie

Table IV C-22

SITE AND REACTOR CHARACTERISTICS OF MIXED OXIDE FUELED REACTOR COMPLEX

Thermal Power Level	3,400 MWt/unit
Electrical Power Level	1,180 MWe/unit
Power Factor	0.8
Coolant Dilution Rate	100 scfs/unit
Distance to Boundary -PWR	1,000 meters
Distance to Boundary -BWR	1,300 meters

waterfowl, fish, or waterplants with the particular maximum receptor being dependent upon the type of reactors presumed installed at the station. The doses received by these organisms would be principally a consequence of physiological phenomena exhibited by those organisms that concentrate, within their body structures, a number of elements that can be present in their aqueous environment. Estimates have been made of the quantities of elements present in the aqueous habitat and the individual species. Values of these biological accumulation factors for a number of chemical elements by freshwater fish and water plants are used in calculating doses to these species. A detailed discussion of the models used for biota dose estimation is given in CHAPTER IV, Section J, Appendix A.

The quantities and species of radionuclides expected to be discharged annually from each of the proposed two reactor systems have been estimated and are provided in CHAPTER IV, Section C-5.0, Tables IV C-14 through IV C-21.

The liquid radioactive wastes given in the reference tables are diluted by the total annual water discharged from the site. The two-unit reactor complex is assumed to have a total operational blow down rate of 200 cubic feet per second. This flow rate provides dilution such that the maximum concentrations of radioactivity at the outfall, neglecting additional dilution and tritium, would be 3.4×10^{-9} microcuries per milliliter of water. The maximum tritium concentration would be 3.2×10^{-6} microcuries per milliliter. Following discharge into the receiving water, further dilution occurs as the effluent mixes with the river as it moves downstream from the point of discharge.

In addition to doses delivered from radionuclides discharged into the environment, exposures can occur due to radiation from turbine units of boiling water reactors. This radiation is virtually independent of fuel type (i.e., uranium oxide, mixed oxide) and arises primarily from the decay of short lived ^{16}N produced in the reactor coolant and carried over to the turbine with the steam. Doses from direct radiation will decrease rapidly with distance from the turbine building.

5.2.1.2 Dose Rate Estimates

The ingestion and direct radiation doses to fish, plants, and waterfowl that live permanently in the vicinity of the discharge outflow have been estimated for each of the three reactor systems considered and for both uranium oxide and mixed oxide fueled conditions. These dose estimates are provided in Table IV C-23.

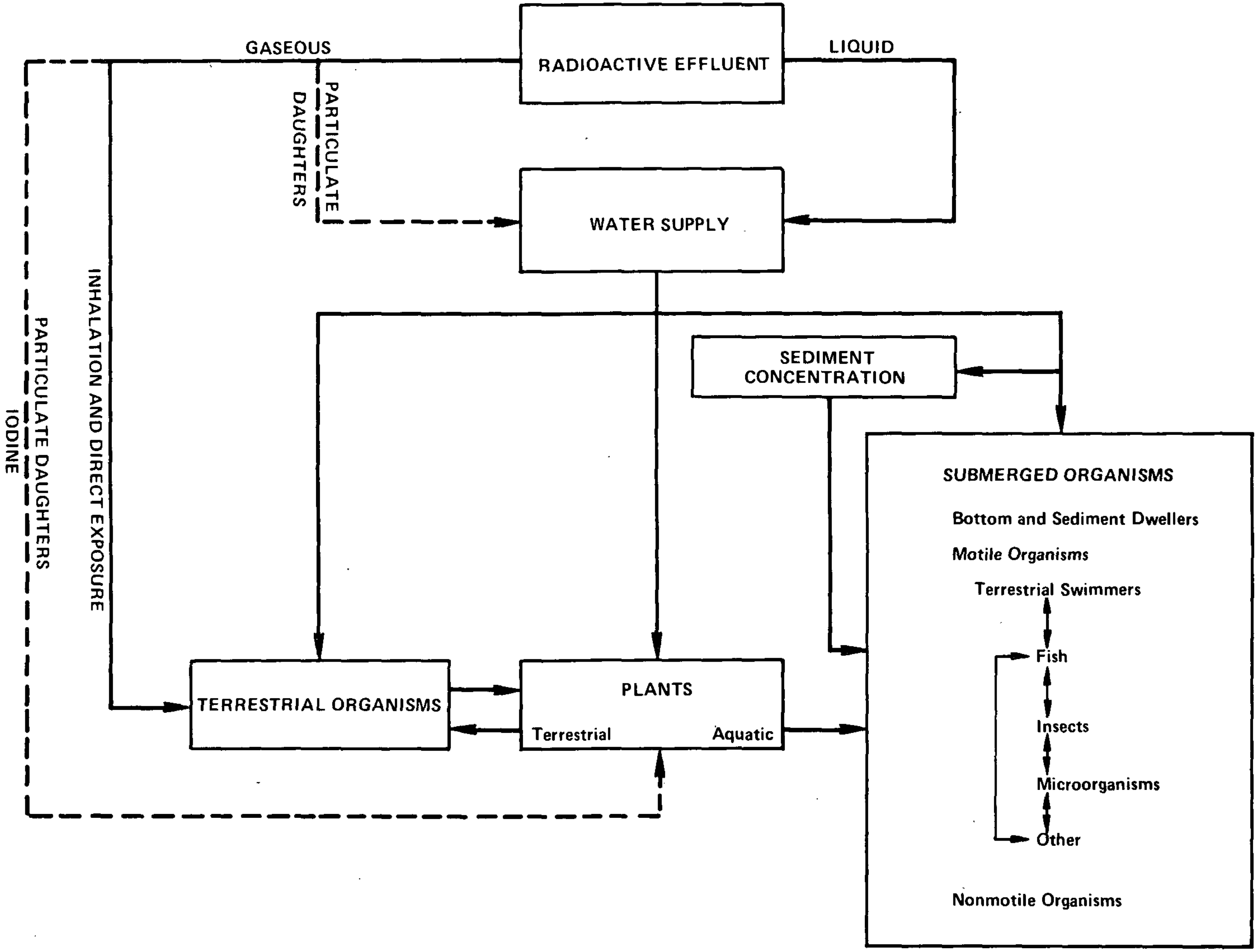

Figure IV C-27 Generalized Exposure Pathways for Organisms Other Than Man

However, it is unlikely that any of the mobile life forms would spend their life spans at these activity concentrations. Also, additional dilution and radioactive decay of the discharge radionuclides prior to their uptake by biota would contribute to reducing the estimated doses to aquatic biota. Therefore, the doses quoted for these organisms are considered to be upper values.

Table IV C-23

ANNUAL DOSES TO BIOTA VIA AQUEOUS EFFLUENT FROM NORMAL REACTOR OPERATIONS*

(Two Units)

Reactor System	Annual Dose (mrad) Fish	 Plants	 Waterfowl
BWR-1**	13	12	44
BWR-2***	13	12	44
PWR-1†	15	2.8	42
PWR-2††	16	3.0	44
PWR-3†††	14	5.0	36
PWR-4§	14	5.0	38

---

*Doses are calculated for radioactivity concentrations corresponding to effluent river dilution.

**BWR-1 denotes two 1,000 MWe boiling water reactors fueled with uranium oxide fuel.

***BWR-2 denotes two 1,000 MWe boiling reactors fueled with mixed oxide fuel.

†PWR-1 denotes two 1,000 MWe pressurized water reactors, U-tube steam generators, fueled with uranium oxide fuel.

††PWR-2 denotes two 1,000 MWe pressurized water reactors, U-tube steam generators, fueled with mixed oxide fuel.

†††PWR-3 denotes two 1,000 MWe pressurized water reactors, once-through steam generators, fueled with uranium oxide fuel.

§PWR-4 denotes two 1,000 MWe pressurized water reactors, once-through steam generators, fueled with mixed oxide fuel.

The difference between doses for a given reactor system as a consequence of uranium oxide and mixed oxide fuel utilization is not large. Because of minor differences in the quantities of radioactive species released from different reactor types and fuels and differing bioaccumulation factors for the various aquatic life forms, one fuel type cannot be said to deliver greater doses to aquatic species as a whole than another. Differences between reactor types, as may be seen in Table IV C-23, are of more significance to dose than the differences in fuel types.

Terrestrial organisms that live in the plant environs could receive radiation doses originating from the turbine complex. Because of specific design details, only boiling water reactor systems would produce observable dose rates ($\gtrsim$ 5 mrem/year) in the plant vicinity. It has been estimated that the typical boiling water reactor system used in this analysis could produce dose rates of the order of 10 mrem/year at a distance of 2,000 meters from the plant. The type of fuel used by the reactors would have virtually no effect on the dose rates produced by this mechanism.

5.2.2 Radiological Impact on Man

During normal operation of reactors small quantities of fission products and induced activities are released to the environment. The designs of the plants are such that radiation doses received by persons beyond the site boundaries as a consequence of plant releases are within the design objectives imposed by Appendix I to 10 CFR Part 50. Estimates have been made of the quantities of radionuclides expected to be released annually from the station under normal operating conditions. Those estimates of effluent activities, provided in Tables IV C-16 through IV C-21, are based upon operating experience gained from past plant performance and estimates of future plant performance obtained by assessing planned modifications in design and operating procedures.

5.2.2.1 Exposure Pathways

Radiation doses that might be delivered to individuals at locations on and beyond the boundaries of the site have been calculated for the six possible combinations of reactor types and fuels. The calculated doses are based upon the estimated activities present in liquid and gaseous releases from two identical reactors operating on the site. These doses result from the most significant pathways, shown on Figure IV C-28, by which man may receive radiation doses from the effluents released from the site. The assumptions used in the calculations attempt to realistically characterize the dilutions of the radionuclides in the atmosphere and coolant water, the biological accumulation of radionuclides in food chains leading to man, and in the assumed use by man of the plant surroundings and the products of the surroundings.

Liquid Effluents--The liquid waste systems that result in discharges to the river described in Tables IV C-16, IV C-18, and IV C-20 provide estimates of annual liquid radionuclide releases per reactor unit and fuel type. Dilution factors for these sources were calculated using the methods described in CHAPTER IV, Section J, Appendix A.

Gaseous Effluents--The gaseous waste system has been described; Tables IV C-17, IV C-19, and IV C-21 provide estimates of annual gaseous radionuclide releases per reactor unit and fuel type. Individual and population doses due to gaseous radionuclides from the various possible combinations of reactor units and fuels have been calculated.

Direct Radiation--Exposure to direct and scattered radiation arises primarily from the decay of short lived ^{16}N produced in the reactor coolant. In a boiling water reactor, the ^{16}N is carried from the reactor to the turbine with the steam. Based on measurements made at operating boiling water reactors, it has been estimated that the reactor system used in this analysis would produce dose rates of 10 mrem/yr at a distance of 2,000 meters from the plant. This dose rate could be received only by an individual who is continuously occupying this particular location. By accounting for occupancy times, this dose would be reduced.

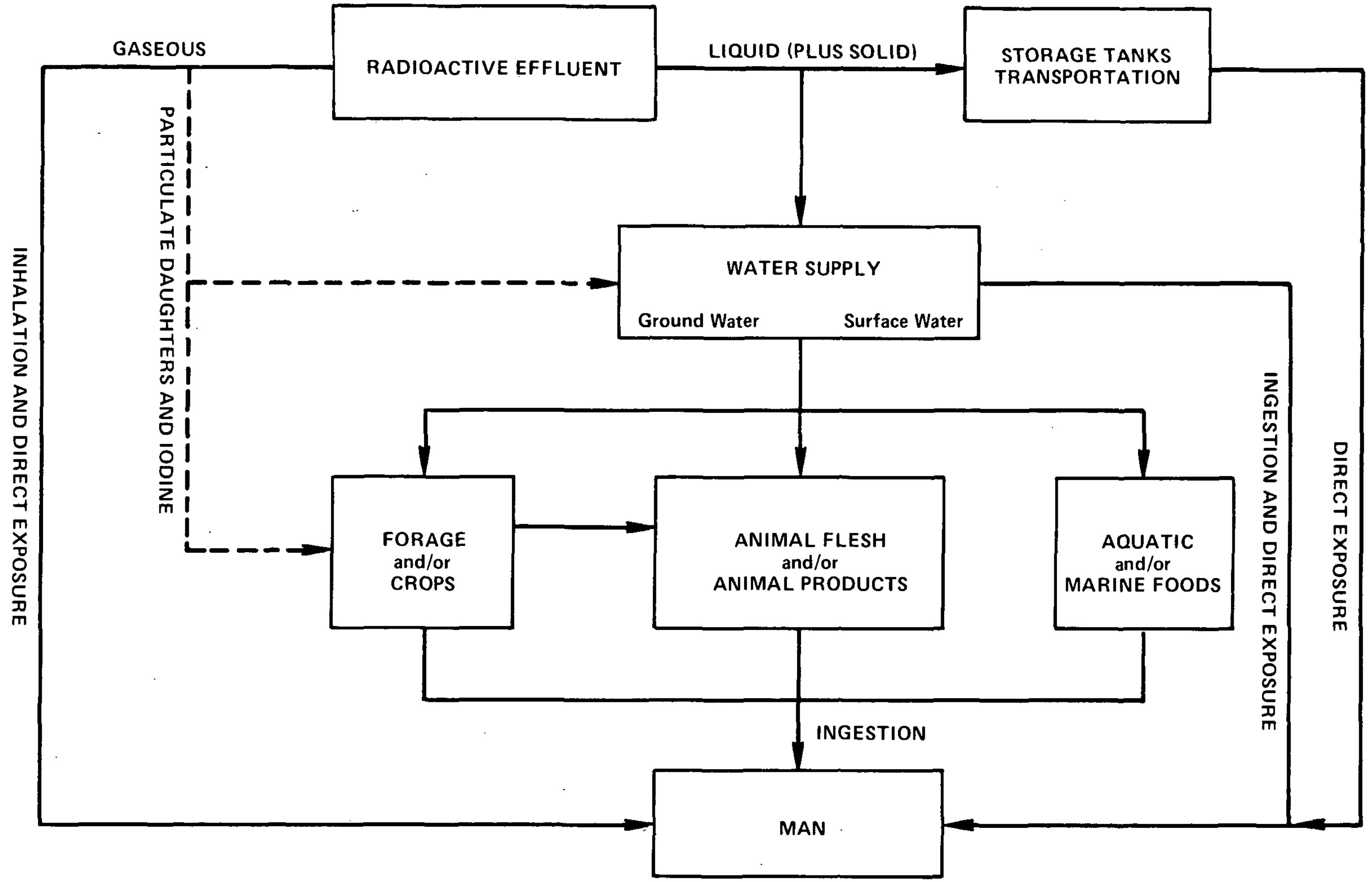

Figure IV C-28 Generalized Exposure Pathways To Man

Direct radiation doses from pressurized water reactors are negligible ($\leq$ 5 mrem/year).

In the case of both BWR's and PWR's, the type of fuel would have virtually no effect on direct radiation dose rates.

5.2.2.2 Maximum Individual Dose Rate Estimates

Individual doses were estimated at assumed site boundary locations for the six combinations of reactor units and fuels. For radionuclides released to the atmosphere, a hypothetical individual was assumed to exist at the site boundary and to be exposed via the pathways described in detail in CHAPTER IV, Section J, Appendix A. These pathways consist of external exposure from submersion in the airborne noble gases, external exposure from standing on particulate radionuclides deposited on the ground, internal exposure from inhaling airborne radionuclides, and internal exposure from ingesting foodstuffs contaminated by radionuclides released to the atmosphere.

For radionuclides released to the receiving water (assumed here to be a river), a hypothetical individual is assumed to be obtaining drinking water and fish and to be using the river bank for recreational purposes near the discharge region and to be consuming foodstuffs produced in fields irrigated by the contaminated river water.

Calculated annual individual doses are listed in Tables IV C-24 through IV C-29 and are typical of doses calculated for actual reactor sites. Comparison of comparable reactor systems using different fuels shows that in no case is dose significantly altered by changes in fuel types. It was concluded that the calculated dose to individuals from normal operations is perturbed very slightly by the introduction of mixed oxide fuel into reactor systems originally fueled with uranium oxide.

5.2.2.3 Population Doses From All Sources Resulting From Normal Operations

A fraction of the radioactive effluent released to air and water from the plant site will become available to man before it is immobilized or decays. The population in the reactor surroundings will also be exposed to radiation from irradiated fuel and other radioactive wastes shipped from the site to approved disposal sites. Thus, the introduction of the plant will result in an incremental increase in radiation exposure over that due to natural background.

Integrated population person-rem doses that may result from radionuclides released to the atmosphere were estimated using the same pathways considered for individual dose estimations. As discussed in CHAPTER IV, Section J, Appendix A, average population density and land productivities are used to estimate the population exposure.

Integrated population doses that might result from the use of river water by the population downstream of the plant have also been estimated. The population exposures

Table IV C-24

CALCULATED ANNUAL INDIVIDUAL DOSES FROM ATMOSPHERIC RELEASES
(in mrem)

Nuclide Group	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
GESMO MODEL - BWR (MOX)								
Tritium	7.81×10^{-3}	7.81×10^{-3}	7.81×10^{-3}	7.81×10^{-3}	7.81×10^{-3}	7.81×10^{-3}	7.81×10^{-3}	7.81×10^{-3}
Radiocarbon	4.23×10^{-1}	4.23×10^{-1}	2.11	4.23×10^{-1}	4.23×10^{-1}	4.23×10^{-1}	4.23×10^{-1}	4.23×10^{-1}
Radioiodine	3.00×10^{-2}	3.77×10^{-2}	1.25×10^{-2}	1.25×10^{-2}	1.25×10^{-2}	8.68	1.70×10^{-2}	1.25×10^{-2}
Nobles Gases--								
Krypton-85	1.60×10^{-4}	1.60×10^{-4}	1.60×10^{-4}	1.60×10^{-4}	1.60×10^{-4}	1.60×10^{-4}	3.47×10^{-4}	1.36×10^{-2}
Others	7.96×10^{-1}	7.96×10^{-1}	7.96×10^{-1}	7.96×10^{-1}	7.96×10^{-1}	7.96×10^{-1}	8.26×10^{-1}	1.80
Other Nuclides	2.59	2.73	4.51	2.65	2.53	2.47	2.53	2.47
TOTALS	3.85	3.99	7.44	3.89	3.77	1.24	3.81	4.72
BWR - (UO_2 Only)								
Tritium	7.15×10^{-3}	7.15×10^{-3}	7.15×10^{-3}	7.15×10^{-3}	7.15×10^{-3}	7.15×10^{-3}	7.15×10^{-3}	7.15×10^{-3}
Radiocarbon	4.23×10^{-1}	4.23×10^{-1}	2.11	4.23×10^{-1}	4.23×10^{-1}	4.23×10^{-1}	4.23×10^{-1}	4.23×10^{-1}
Radioiodine	2.78×10^{-2}	3.60×10^{-2}	1.81×10^{-2}	1.81×10^{-2}	1.81×10^{-2}	7.93	1.61×10^{-2}	1.18×10^{-2}
Noble Gases--								
Krypton-85	2.21×10^{-4}	2.21×10^{-4}	2.21×10^{-4}	2.21×10^{-4}	2.21×10^{-4}	2.21×10^{-4}	4.97×10^{-4}	1.87×10^{-2}
Others	8.44×10^{-1}	8.44×10^{-1}	8.44×10^{-1}	8.44×10^{-1}	8.44×10^{-1}	8.44×10^{-1}	8.75×10^{-1}	2.02
Other Nuclides	2.59	2.73	4.51	2.65	2.53	2.47	2.53	2.47
TOTALS	3.89	4.04	7.49	3.93	3.81	1.17	3.85	4.95

$X/Q = 1.0 \times 10^{-6}$ sec/cu. m

Table IV C-25

CALCULATED ANNUAL INDIVIDUAL DOSES FROM LIQUID RELEASES
(mrem)

Nuclides	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung
GESMO MODEL - BWR (MOX)							
Totals	2.97×10^{-1}	4.86×10^{-2}	3.22×10^{-1}	4.02×10^{-1}	1.55×10^{-1}	1.39	6.69×10^{-2}
BWR - (UO_2 Only)							
Totals	3.15×10^{-1}	5.20×10^{-2}	3.34×10^{-1}	4.26×10^{-1}	1.66×10^{-1}	1.50	7.22×10^{-2}

Table IV C-26

CALCULATED ANNUAL INDIVIDUAL DOSES FROM ATMOSPHERIC RELEASES
(in mrem)

Nuclide Group	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
GESMO MODEL - U-TUBE PWR (MOX)								
Tritium	1.99×10^{-1}	1.99×10^{-1}	1.99×10^{-1}	1.99×10^{-1}	1.99×10^{-1}	1.99×10^{-1}	1.99×10^{-1}	1.99×10^{-1}
Radiocarbon	3.76×10^{-1}	3.76×10^{-1}	1.88	3.76×10^{-1}	3.76×10^{-1}	3.76×10^{-1}	3.76×10^{-1}	3.76×10^{-1}
Radioiodine	2.11×10^{-3}	1.95×10^{-3}	7.27×10^{-4}	7.27×10^{-4}	7.27×10^{-4}	6.79×10^{-1}	9.76×10^{-4}	7.27×10^{-4}
Noble Gases--								
Krypton-85	2.59×10^{-4}	2.59×10^{-4}	2.59×10^{-4}	2.59×10^{-4}	2.59×10^{-4}	2.59×10^{-4}	5.61×10^{-4}	2.20×10^{-2}
Others	2.00×10^{-1}	2.00×10^{-1}	2.00×10^{-1}	2.00×10^{-1}	2.00×10^{-1}	2.00×10^{-1}	2.13×10^{-1}	4.62×10^{-1}
Other Nuclides	1.12	1.11	1.51	1.21	1.10	1.05	1.08	1.05
TOTALS	1.90	1.89	3.79	1.99	1.88	2.50	1.87	2.11
U-TUBE PWR (UO_2 ONLY)								
Tritium	1.83×10^{-1}	1.83×10^{-1}	1.83×10^{-1}	1.83×10^{-1}	1.83×10^{-1}	1.83×10^{-1}	1.83×10^{-1}	1.83×10^{-1}
Radiocarbon	3.76×10^{-1}	3.76×10^{-1}	1.88	3.76×10^{-1}	3.76×10^{-1}	3.76×10^{-1}	3.76×10^{-1}	3.76×10^{-1}
Radioiodine	1.96×10^{-3}	1.83×10^{-3}	6.81×10^{-4}	6.81×10^{-4}	6.81×10^{-4}	6.30×10^{-1}	9.15×10^{-4}	6.81×10^{-4}
Noble Gases--								
Krypton-85	3.59×10^{-4}	3.59×10^{-4}	3.59×10^{-4}	3.59×10^{-4}	3.59×10^{-4}	3.59×10^{-4}	7.76×10^{-4}	3.04×10^{-2}
Others	2.04×10^{-1}	2.04×10^{-1}	2.04×10^{-1}	2.04×10^{-1}	2.04×10^{-1}	2.04×10^{-1}	2.17×10^{-1}	4.67×10^{-1}
Other Nuclides	1.12	1.11	1.51	1.21	1.10	1.05	1.08	1.05
TOTALS	1.88	1.88	3.78	1.98	1.86	2.46	1.85	2.10

$X/Q = 1.5 \times 10^{-6}$ sec/cu. m

Table IV C-27

CALCULATED ANNUAL INDIVIDUAL DOSES FROM ATMOSPHERIC RELEASES
(in mrem)

Nuclide Group	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
GESMO MODEL ONCE-THROUGH PWR MOX								
Tritium	1.99×10^{-1}	1.99×10^{-1}	1.99×10^{-1}	1.99×10^{-1}	1.99×10^{-1}	1.99×10^{-1}	1.99×10^{-1}	1.99×10^{-1}
Radiocarbon	3.76×10^{-1}	3.76×10^{-1}	1.88	3.76×10^{-1}	3.76×10^{-1}	3.76×10^{-1}	3.76×10^{-1}	3.76×10^{-1}
Radionuclides	2.11×10^{-3}	1.96×10^{-3}	7.32×10^{-4}	7.32×10^{-4}	7.32×10^{-4}	6.80×10^{-1}	9.83×10^{-4}	7.32×10^{-4}
Noble Gases--								
Krypton-85	2.59×10^{-4}	2.59×10^{-4}	2.59×10^{-4}	2.59×10^{-4}	2.59×10^{-4}	2.59×10^{-4}	5.61×10^{-4}	2.20×10^{-2}
Others	2.00×10^{-1}	2.00×10^{-1}	2.00×10^{-1}	2.00×10^{-1}	2.00×10^{-1}	2.00×10^{-1}	2.13×10^{-1}	4.62×10^{-1}
Other Nuclides	1.12	1.11	1.51	1.21	1.10	1.05	1.08	1.05
TOTALS	1.90	1.89	3.79	1.99	1.88	2.50	1.87	2.11
ONCE THROUGH - PWR - (UO_2 ONLY)								
Tritium	1.83×10^{-1}	1.83×10^{-1}	1.83×10^{-1}	1.83×10^{-1}	1.83×10^{-1}	1.83×10^{-1}	1.83×10^{-1}	1.83×10^{-1}
Radiocarbon	3.76×10^{-1}	3.76×10^{-1}	1.88	3.76×10^{-1}	3.76×10^{-1}	3.76×10^{-1}	3.76×10^{-1}	3.76×10^{-1}
Radioiodine	2.04×10^{-3}	1.90×10^{-3}	7.09×10^{-4}	7.09×10^{-4}	7.09×10^{-4}	6.55×10^{-1}	9.52×10^{-4}	7.09×10^{-4}
Noble Gases--								
Krypton-85	3.59×10^{-4}	3.59×10^{-4}	3.59×10^{-4}	3.59×10^{-4}	3.59×10^{-4}	3.59×10^{-4}	7.76×10^{-4}	3.04×10^{-2}
Others	2.04×10^{-1}	2.04×10^{-1}	2.04×10^{-1}	2.04×10^{-1}	2.04×10^{-1}	2.04×10^{-1}	2.17×10^{-1}	4.67×10^{-1}
Other Nuclides	1.12	1.11	1.51	1.21	1.10	1.05	1.08	1.05
TOTALS	1.88	1.88	3.78	1.98	1.86	2.46	1.85	2.10

$X/Q = 1.5 \times 10^{-6}$ sec/cu. m

Table IV C-28

CALCULATED ANNUAL INDIVIDUAL DOSES FROM LIQUID RELEASES
(mrem)

Nuclides	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung
GESMO MODEL U-TUBE PWR (MOX)							
TOTAL	5.93×10^{-1}	1.57×10^{-1}	3.50×10^{-1}	7.54×10^{-1}	3.48×10^{-1}	4.60×10^{-1}	2.09×10^{-1}
U-TUBE PWR - (UO_2 ONLY)							
TOTAL	4.95×10^{-1}	1.46×10^{-1}	2.88×10^{-1}	6.27×10^{-1}	2.99×10^{-1}	3.00×10^{-1}	1.86×10^{-1}

Table IV C-29

CALCULATED ANNUAL INDIVIDUAL DOSES FROM LIQUID RELEASES
(mrem)

Nuclides	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung
GESMO ONCE-THROUGH - PWR (MOX)							
TOTAL	5.13×10^{-1}	1.57×10^{-1}	2.95×10^{-1}	6.49×10^{-1}	3.14×10^{-1}	3.16×10^{-1}	1.98×10^{-1}
ONCE-THROUGH PWR (UO_2 ONLY)							
TOTALS	5.66×10^{-1}	1.46×10^{-1}	3.39×10^{-1}	7.21×10^{-1}	3.30×10^{-1}	4.39×10^{-1}	1.96×10^{-1}

that might occur as a consequence of drinking water consumption, ingestion of fish, consumption of irrigated crops, and recreational activities in the downstream region of the river have been calculated.

The calculated annual population doses from the liquid and gaseous effluents are listed in Tables IV C-30 through IV C-34. As may be seen, the population dose values are only slightly dependent upon fuel type. In no case is the population dose significant when it is recalled that each member of the population receives an individual total body dose of about 100 mrem/year from natural background.

The transportation of cold (new) fuel to a reactor, or irradiated fuel from the reactors to a fuel reprocessing plant, and of solid radioactive wastes from the reactor to disposal sites is described in CHAPTER IV, Section G.

Dose Assessment--Using conservative estimates, the total population dose from all effluent pathways received by the population would be no more than about 500 person-rem per year. Operation of these two-unit plants would be a minor contributor to the radiation dose that persons living in the area normally receive from natural background radiation.

It may be concluded that, under normal operations, the effects of fuel type (of the fuels considered) is not significant in terms of the radiological impact of the example reactor systems.

5.3 Accident Analysis Source Terms for Plutonium Recycle Reactors

Plutonium recycle reactors potentially have substantial increases in plutonium content over uranium oxide cores; e.g., a 1.15 SGR core may contain three to four times the amount of plutonium found in a uranium oxide core of similar exposure. However, plutonium is not a factor in the comparison of source terms because engineered safety features would limit temperatures to well below the point at which plutonium vaporization would occur in both uranium and mixed oxide charges. Radioactive halogen and noble gas isotopes constitute the principal source terms for evaluating the radiological consequences of design basis accidents.[2,3] The implications of the higher plutonium content were evaluated with respect to the halogen and noble gas fission products, in order to assess changes, if any, in the accident analysis source terms.

Table IV C-35 indicates the difference in selected fission products between the model plutonium recycle reactor and a typical uranium-only reactor. The comparison is in terms of element dose ratios for the krypton, xenon, and iodine radioisotopes, which are used in accident analyses. It can be seen that where increases occur, they are relatively small.

Table IV C-30

CALCULATED ANNUAL POPULATION DOSE FROM ATMOSPHERIC RELEASES

(person-rem)

Nuclide Group	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
GESMO MODEL BWR (MOX)								
Tritium	5.17×10^{-1}	5.17×10^{-1}	5.17×10^{-1}	5.17×10^{-1}	5.17×10^{-1}	5.17×10^{-1}	5.17×10^{-1}	5.17×10^{-1}
Radiocarbon	8.64×10^{1}	8.64×10^{1}	4.32×10^{2}	8.64×10^{1}	8.64×10^{1}	8.64×10^{1}	8.64×10^{1}	8.46×10^{1}
Radioiodine	1.41×10^{-1}	1.35×10^{-1}	3.93×10^{-2}	3.93×10^{-2}	3.93×10^{-2}	5.03×10^{1}	5.31×10^{-2}	5.31×10^{-2}
Noble Gases--								
Krypton-85	5.87×10^{-2}	5.87×10^{-2}	5.87×10^{-2}	5.87×10^{-2}	5.87×10^{-2}	5.87×10^{-2}	1.27×10^{-1}	6.97
Others	1.31	1.31	1.31	1.31	1.31	1.31	1.39	3.18
Other Nuclides	1.12×10^{1}	1.19×10^{1}	2.34×10^{1}	1.14×10^{1}	1.06×10^{1}	1.02×10^{1}	1.05×10^{1}	1.02×10^{1}
TOTALS	9.94×10^{1}	1.00×10^{2}	4.57×10^{2}	9.97×10^{1}	9.89×10^{1}	1.49×10^{2}	9.90×10^{1}	1.05×10^{2}
BWR - (UO_2 ONLY)								
Tritium	4.73×10^{-1}	4.73×10^{-1}	4.73×10^{-1}	4.73×10^{-1}	4.73×10^{-1}	4.73×10^{-1}	4.73×10^{-1}	4.73×10^{-1}
Radiocarbon	8.64×10^{1}	8.64×10^{1}	4.32×10^{2}	8.64×10^{1}	8.64×10^{1}	8.64×10^{1}	8.64×10^{1}	8.64×10^{1}
Radioiodine	1.30×10^{-1}	1.27×10^{-1}	3.67×10^{-2}	3.67×10^{-2}	3.67×10^{-2}	4.59×10^{1}	4.96×10^{-2}	3.67×10^{-2}
Noble Gases--								
Krypton-85	8.11×10^{-2}	8.11×10^{-2}	8.11×10^{-2}	8.11×10^{-2}	8.11×10^{-2}	8.11×10^{-2}	1.75×10^{-1}	6.87×10^{1}
Others	1.34	1.34	1.34	1.34	1.34	1.34	1.42	3.26
Other Nuclides	1.10×10^{1}	1.19×10^{1}	2.34×10^{1}	1.14×10^{1}	1.06×10^{1}	1.02×10^{1}	1.05×10^{1}	1.02×10^{1}
TOTALS	9.94×10^{1}	1.00×10^{2}	4.57×10^{2}	9.97×10^{1}	9.89×10^{1}	1.44×10^{2}	9.90×10^{1}	1.07×10^{2}

NOTE: U.S. person-rem from ^{13}H is 97.975% of the worldwide value
U.S. person-rem from ^{14}C is 48.351% of the worldwide value
U.S. person-rem from ^{85}Kr is 18.085% of the worldwide value

Table IV C-31

CALCULATED ANNUAL POPULATION DOSES FROM LIQUID RELEASES
(Person-rem)

Nuclides	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung
GESMO MODEL BWR (MOX)							
TOTAL	1.40	1.00	6.71×10^{-1}	1.82	1.22	4.15×10^{1}	1.00
BWR - (UO_2 ONLY)							
TOTAL	1.54	1.11	7.04×10^{-1}	1.99	1.35	4.47×10^{1}	1.12

Table IV C-32

CALCULATED ANNUAL POPULATION DOSES FROM ATMOSPHERIC RELEASES

(Person-rem)

Nuclide Group	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
GESMO MODEL U-TUBE PWR (MOX)								
Tritium	1.32×10^{1}	1.32×10^{1}	1.32×10^{1}	1.32×10^{1}	1.32×10^{1}	1.32×10^{1}	1.32×10^{1}	1.32×10^{1}
Radiocarbon	7.68×10^{1}	7.68×10^{1}	3.84×10^{2}	7.68×10^{1}	7.68×10^{1}	7.68×10^{1}	7.68×10^{1}	7.68×10^{1}
Radioiodine	1.08×10^{-2}	8.93×10^{-3}	2.58×10^{-3}	2.58×10^{-3}	2.58×10^{-3}	4.02	3.45×10^{-3}	2.58×10^{-3}
Noble Gases--								
Krypton-85	9.51×10^{-2}	9.51×10^{-2}	9.51×10^{-2}	9.51×10^{-2}	9.51×10^{-2}	9.51×10^{-2}	2.06×10^{-1}	8.05
Others	3.88	3.88	3.88	3.88	3.88	3.88	4.01	9.25
Other Nuclides	4.81	4.76	7.32	5.43	4.68	4.32	4.49	4.32
TOTALS	9.88×10^{1}	9.87×10^{1}	4.08×10^{2}	9.94×10^{1}	9.86×10^{1}	1.02×10^{1}	9.87×10^{1}	1.12×10^{2}
U-TUBE PWR (UO_2 ONLY)								
Tritium	1.21×10^{1}	1.21×10^{1}	1.21×10^{1}	1.21×10^{1}	1.21×10^{1}	1.21×10^{1}	1.21×10^{1}	1.21×10^{1}
Radiocarbon	7.68×10^{1}	7.68×10^{1}	3.84×10^{2}	7.68×10^{1}	7.68×10^{1}	7.68×10^{1}	7.68×10^{1}	7.68×10^{1}
Radioiodine	9.98×10^{-3}	7.84×10^{-3}	2.40×10^{-3}	2.40×10^{-3}	2.40×10^{-3}	3.73	3.22×10^{-3}	2.40×10^{-3}
Noble Gases--								
Krypton-85	1.31×10^{-1}	1.31×10^{-1}	1.31×10^{-1}	1.31×10^{-1}	1.31×10^{-1}	1.31×10^{-1}	2.84×10^{-1}	1.11×10^{1}
Others	3.88	3.88	3.88	3.88	3.88	3.88	4.01	9.25
Other Nuclides	4.81	4.76	7.32	5.43	4.68	4.32	4.49	4.32
TOTALS	9.77×10^{1}	9.77×10^{1}	4.07×10^{2}	9.83×10^{1}	9.76×10^{1}	1.01×10^{2}	9.77×10^{1}	1.14×10^{2}

NOTE: U.S. person-rem from ^{3}H is 97.975% of the worldwide value.
U.S. person-rem from ^{14}C is 48.351% of the worldwide value.
U.S. person-rem from ^{85}Kr is 18.085% of the worldwide value.

Table IV C-33

CALCULATED ANNUAL POPULATION DOSES FROM ATMOSPHERIC RELEASES
(Person-rem)

Nuclide Group	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
GESMO MODEL - ONCE THROUGH PWR (MOX)								
Tritium	1.32×10^1	1.32×10^1	1.32×10^1	1.32×10^1	1.32×10^1	1.32×10^1	1.32×10^1	1.32×10^1
Radiocarbon	7.68×10^1	7.68×10^1	3.84×10^2	7.68×10^1	7.68×10^1	7.68×10^1	7.68×10^1	7.68×10^1
Radioiodine	1.08×10^{-2}	8.43×10^{-3}	2.59×10^{-3}	2.59×10^{-3}	2.59×10^{-3}	4.02	3.46×10^{-3}	2.59×10^{-3}
Noble Gases--								
Krypton-85	9.51×10^{-2}	9.51×10^{-2}	9.51×10^{-2}	9.51×10^{-2}	9.51×10^{-2}	9.51×10^{-2}	2.06×10^{-1}	8.05
Others	3.88	3.88	3.88	3.88	3.88	3.88	4.01	9.25
Other Nuclides	4.81	4.76	7.32	5.43	4.68	4.32	4.49	4.32
TOTALS	9.88×10^1	9.87×10^1	4.08×10^2	9.94×10^1	9.86×10^1	1.02×10^2	9.87×10^1	1.12×10^2
ONCE THROUGH - PWR (UO_2 ONLY)								
Tritium	1.21×10^1	1.21×10^1	1.21×10^1	1.21×10^1	1.21×10^1	1.21×10^1	1.21×10^1	1.21×10^1
Radiocarbon	7.68×10^1	7.68×10^1	3.84×10^2	7.68×10^1	7.68×10^1	7.68×10^1	7.68×10^1	7.68×10^1
Radioiodine	1.04×10^{-2}	8.16×10^{-3}	2.50×10^{-3}	2.50×10^{-3}	2.50×10^{-3}	3.88	3.35×10^{-3}	2.50×10^{-3}
Noble Gases--								
Krypton-85	1.31×10^{-1}	1.31×10^{-1}	1.31×10^{-1}	1.31×10^{-1}	1.31×10^{-1}	1.31×10^{-1}	2.84×10^{-1}	1.11×10^1
Others	3.88	3.88	3.88	3.88	3.88	3.88	4.01	9.25
Other Nuclides	4.81	4.76	7.32	5.43	4.68	4.32	4.49	4.32
TOTALS	9.77×10^1	9.77×10^1	4.07×10^2	9.83×10^1	9.76×10^1	1.01×10^2	9.77×10^1	1.14×10^2

NOTE: U.S. person-rem from ^3H is 97.975% of the worldwide value.
U.S. person-rem from ^{14}C is 48.351% of the worldwide value.
U.S. person-rem from ^{85}Kr is 18.085% of the worldwide value.

Table IV C-34

CALCULATED ANNUAL POPULATION DOSES FROM LIQUID RELEASES
(Person-rem)

Nuclides	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung
GESMO MODEL U-TUBE PWR - (MOX)							
TOTAL	5.67	5.05	8.80×10^{-1}	6.30	5.07	1.35×10^{1}	5.12
U-TUBE PWR (UO_2 ONLY)							
TOTAL	5.15	4.68	7.22×10^{-1}	5.67	4.64	9.04	4.71
GESMO MODEL ONCE-THROUGH PWR (MOX)							
TOTAL	5.55	5.05	7.38×10^{-1}	6.07	5.00	9.58	5.10
ONCE-THROUGH PWR (UO_2 ONLY)							
TOTAL	5.27	4.67	8.52×10^{-1}	5.87	4.70	1.28×10^{1}	4.73

Table IV C-35

COMPARISON OF THE GESMO MODEL MOX AND URANIUM ONLY FUEL ACCIDENT ANALYSIS DOSE SOURCE TERMS

Element	MOX/Uranium-Only Dose Ratio Pu 1-2 (1/4 fuel cycle) (7,586 MWd/MT)	 Pu 2-3 (Spent Fuel) (26,604 MWd/MT)
Kr	0.55*	0.67*
Xe	1.01*	0.91*
I	1.05*	0.99*
	1.14**	1.03**

---

*Whole Body Dose Ratio

**Thyroid Dose Ratio

Pu-1, Pu-2	Fresh plutonium recovered from spent uranium fuel from a reactor using UO_2 only fuel and a reactor that contains some MOX rods, respectively
Pu 1-2	Plutonium recovered from spent MOX rods fabricated from Pu-1 fuel
Pu 2-3	Plutonium recovered from spent MOX rods fabricated from Pu-2 fuel

At worst, some SGR fuels exhibit as much as 14% increase in the iodine thyroid dose source term. More typically, the dose source terms decrease somewhat except for the iodine thyroid dose source term which shows about a 3% increase.

Thus, it is concluded that the accident analysis source terms for plutonium recycle reactors as typified by the SGR model do not present any significant increases over those used for uranium oxide reactors.

5.3.1 Source Term Evaluation for Plutonium Recycle Reactors

Section 4.0 describes the model used in calculating fission product inventories for the plutonium recycle reactor based on 1.15 SGR. A typical uranium-only reactor was included in the analysis for the purpose of comparison to the GESMO model LWR fission product inventories.

The calculated plutonium recycle fuel fission product inventories for the selected radionuclides exhibit some departures from those identified with uranium oxide fueled LWR's. For example, Table IV C-36 presents 1.15 SGR to BWR inventory ratios for a limited variety of plutonium isotopic mixtures and irradiation exposures. Only ^{135}Xe shows a significant increase. However, when the isotopic xenon inventories are weighted by dose conversion factors, the departure from uranium oxide fueled LWR's is less pronounced as illustrated in the last two columns in Table IV C-36 where the entries represent 1.15 SGR to dose ratios by element for plutonium recycle fuels at burnups shown.

On the basis of the elemental SGR and BWR dose ratios listed in Table IV C-36 it can be concluded that the xenon and iodine source terms are essentially the same for

Table IV C-36

COMPARISON OF MOX AND UO_2 SPECIFIC INVENTORIES RATIOS AND DOSE RATIOS FOR SELECTED FISSION PRODUCTS-BWR†

	MOX/UO_2†† Specific Inventory Ratio*				MOX/UO_2†† Element Dose Ratio	
	Pu-2	Pu-2-3	Pu-1-2-3	Pu-1-2	Pu-2-3	Pu-1-2
			(1/4 fuel cycle)	Spent Fuel	(1/4 fuel cycle)	
Burnup	26,604 MWd/MT	26,604 MWd/MT	30,403 MWd/MT	7,596 MWd/MT	26,604 MWd/MT	7,596 MWd/MT
	(1)	(2)	(3)	(4)	(5)	(6)
^{83m}Kr	0.583	0.548	0.531	0.356		
^{85}Kr	0.657	0.639	0.630	0.531		
^{85m}Kr	0.762	0.746	0.727	0.582	0.67**	0.55**
^{87}Kr	0.742	0.719	0.704	0.546		
^{88}Kr	0.754	0.735	0.722	0.568		
^{89}Kr	0.735	0.711	0.697	0.535		
^{131m}Xe	1.04	1.02	1.02	1.15		
^{133m}Xe	0.982	0.963	0.954	0.981		
^{133}Xe	0.982	0.972	0.963	0.989		
^{135m}Xe	0.972	0.954	0.945	0.954	0.91**	1.01**
^{135}Xe	1.47	1.63	1.65	1.75		
^{137}Xe	1.00	0.991	0.980	1.03		
^{138}Xe	1.01	0.991	0.980	1.05		
^{131}I	1.04	1.03	1.02	1.15		
^{132}I	1.03	1.01	1.01	1.10		
^{133}I	1.05	0.963	0.954	0.989	0.99**	1.05**
^{134}I	0.963	0.954	0.936	0.945	1.03***	1.14***
^{135}I	0.963	0.954	0.945	0.954		

---

*Pu-1 (Ref.)	"Fresh" plutonium recovered from uranium fuel that was exposed in an all-uranium reactor.
Pu-2	"Fresh" plutonium recovered from uranium fuel that was exposed in a reactor containing some fraction of mixed oxide fuel.
Pu-1-2	Plutonium recovered from spent fuel rods fabricated from Pu-1 fuel.
Pu-2-3	Plutonium recovered from spent fuel rods fabricated from Pu-2 fuel.
Pu-1-2-3	Plutonium recovered from spent fuel rods fabricated from Pu that has been recycled two or more times.

**Whole Body Dose

***Thyroid Dose

†Based on ORNL version of ORIGEN Computer Results

††The 1.15 SGR (GESMO MODEL) uses recycle plutonium assumed to contain 33% Pu-2, 32% Pu-2-3, and 35% Pu 1-2-3.

GESMO model and uranium only LWR's. The krypton source term is lower by a factor of about 0.7 for plutonium fuels at the end of a full irradiation cycle. For 1/4 fuel cycle exposure (i.e., approximately one year) the differences between MOX and uranium-only fuel rods are more noticeable. For example, the ratio of doses due to Kr, Xe, and I for Pu 1-2 MOX at 7596 MWd/MT to uranium-only fuel of the same exposure compared to end of cycle fuel ratios are noticeably different as shown in Table IV C-36, columns 5 and 6.

A comparison of the uranium fueled BWR and PWR fission product inventories where fuel irradiation exposure is assumed to be the same for both types of LWR's is shown in Table IV C-37. The agreement is acceptable.

5.4 Environmental Impact of Postulated Accidents in LWR's Utilizing Mixed Oxide Fuel

This section discusses the environmental impact of postulated accidents for light water reactors utilizing mixed oxide fuel. It is similar to the plant accident section prepared for each environmental statement issued for light water reactors currently being reviewed under NEPA requirements. The assessment of the environmental impact of postulated accidents for mixed oxide cores is that it is not significantly different from the environmental impact from currently proposed uranium oxide cores. Although the fission yields of some isotopes vary slightly for the mixed oxide core, the conclusions of this evaluation would not be any different for a light water reactor using uranium oxide fuel.

A high degree of protection against the occurrence of postulated accidents listed in Table IV C-38 in light water reactors is provided through correct design, manufacture, operation, and the quality assurance program used to establish the necessary high integrity of the reactor system. These points are considered in the Commission's safety evaluations on individual licensing cases. Deviations that may occur are handled by protective systems to place and hold the plant in a safe condition. Notwithstanding this, the conservative postulate is made that serious accidents might occur, even though they may be extremely unlikely, and engineered safety features are installed to mitigate the consequences of those postulated events that are judged credible. The use of mixed oxide fuel has no impact on the capability of engineered safety features to perform their intended function.

The probability of occurrence of accidents and the spectrum of their consequences to be considered from an environmental effects standpoint have been analyzed using best estimates of probabilities and realistic fission product release and transport assumptions. For site evaluation in the Commission's safety reviews, extremely conservative assumptions are used for the purpose of comparing calculated doses resulting from a hypothetical release of fission products from the fuel against the 10 CFR Part 100 siting guidelines. Realistically computed doses that would be received by a population or environment from the accidents that are postulated would be significantly less than those usually presented in Safety Evaluations and within the Maximum Permissible Concentration (MPC) of 10 CFR Part 20.

Table IV C-37

COMPARISON OF ORNL CALCULATED BWR URANIUM INVENTORIES WITH PWR URANIUM INVENTORIES, Ci/MWt

	BWR*	PWR**	f_{whb}***	f_{th}†
^{83m}Kr	1.515×10^3	1.505×10^3	.000691	
^{85}Kr	3.586×10^2	3.520×10^2	.00104	
^{85m}Kr	5.532×10^3	5.425×10^3	.0242	
^{87}Kr	9.716×10^3	1.023×10^4	.285	
^{88}Kr	1.473×10^4	1.510×10^4	.320	
^{89}Kr	1.843×10^4	1.853×10^4	.369	
^{131m}Xe	2.385×10^2	2.368×10^2	.000166	
^{133m}Xe	1.568×10^3	1.270×10^3	.00114	
^{133}Xe	5.252×10^4	5.200×10^4	.0461	
^{135m}Xe	1.421×10^4	1.398×10^4	.0344	
^{135}Xe	8.664×10^3	8.550×10^3	.0480	
^{137}Xe	6.330×10^4	5.124×10^4	.343	
^{138}Xe	6.127×10^4	4.950×10^4	.528	
^{131}I	2.840×10^4	2.875×10^4	.378	.713
^{132}I	4.022×10^4	4.075×10^4	.260	.0256
^{133}I	5.317×10^4	5.225×10^4	.133	.254
^{134}I	6.011×10^4	5.875×10^4	.346	.0122
^{135}I	4.607×10^4	4.575×10^4	.223	.0593

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*BWR inventory corrected for burnup and enrichment to correspond to 44,000 MWd/MT and 3.3% enrichment.

**Reference PWR inventory calculated with ORIGEN at 44,000 MWd/MT and 3.3% enrichment.

***Whole body dose (whb) weighting factor.

†Thyroid Dose weighting factor.

Table IV C-38

CLASSIFICATION OF POSTULATED ACCIDENTS AND OCCURRENCES*

Class	NRC Description	Examples of Accidents
1	Trivial incidents	(None identified.)
2	Small releases outside containment	Releases through steam line relief valves; small spills and leaks of radioactive materials outside containment.
3	Radioactive waste system failure	Equipment leakage or malfunction; release of waste gas storage tank contents; release of liquid waste storage tank contents.
4	Fission products to primary system (BWR)	Fuel cladding defects; off-design transients that induce fuel failures above those expected.
5	Fission products to primary and secondary systems (PWR)	Fuel cladding defects and steam generator leaks; off-design transients that induce fuel failure above those expected and steam generator leak; steam generator tube rupture.
6	Refueling accident	Fuel bundle drop; heavy object drop onto fuel in core.
7	Spent fuel handling accident	Fuel assembly drop in fuel storage pool; heavy object drop onto fuel rack; fuel cask drop.
8	Accident initiation events considered in design-basis evaluation in plant Safety Analysis Report	Loss-of-coolant accidents; break in instrument line from primary system that penetrates the containment; rod ejection accident (PWR); rod drop accident (BWR); steam line breaks.
9	Hypothetical sequence of failures more severe than Class 8	Not considered.

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*From proposed Annex to Appendix D, 10 CFR Part 50.

The Commission issued guidance to applicants on September 1, 1971, requiring the consideration of a spectrum of accidents with assumptions as realistic as the state of knowledge permits. Standard accident assumptions and guidance were issued by the Commission as a proposed amendment to Appendix D of 10 CFR Part 50 on December 1, 1971. Nine classes of postulated accidents and occurrences ranging in severity from trivial to very serious were identified by the Commission. In general, accidents in the high potential consequence end of the spectrum have a low occurrence rate and those on the low potential consequence end have a higher occurrence rate.

The dose which might be received by an assumed individual standing at the site boundary in the downwind direction was estimated using the assumptions in the proposed Annex to Appendix D previously mentioned. Accident consequence estimates for a PWR and a BWR are presented in Tables IV C-39 and IV C-40 for a postulated river site. Estimates of the integrated exposure that might be delivered to the population within 50 miles of the site for the year 2000 are also presented in these tables. The site chosen for analysis was the river site described in the final environmental statement[41] on the as low as practicable rule.

To rigorously establish a realistic annual risk, the calculated doses in Tables IV C-39 and IV C-40 would have to be multiplied by estimated probabilities. The events in Classes 1 and 2 represent occurrences which are anticipated during plant operations; and their consequences, which are very small, are considered within the framework of routine effluents from the plant. Except for a limited amount of fuel failures and some steam generator leakage (PWR case), the events in Classes 3 through 5 are not anticipated during normal plant operation; but events of this type could occur sometime during the 40-year plant lifetime. Accidents in Classes 6 and 7 and small accidents in Class 8 are of similar or lower probability than accidents in Classes 3 through 5 but are still possible. The probability of occurrence of large Class 8 accidents is very small. Therefore, when the consequences indicated in Tables IV C-39 and IV C-40 are weighted by probabilities, the environmental risk is very low.

The postulated occurrences in Class 9 involve sequences of successive failures more severe than those required to be considered in the design bases of protection systems and engineered safety features. Their consequences could be severe. However, the probability of their occurrence is judged so small that their environmental risk is extremely low. Defense in depth (multiple physical barriers), quality assurance for design, manufacture and operation, continued surveillance and testing, and conservative design are all applied to provide and maintain a high degree of assurance that potential accidents in this class are, and will remain, sufficiently small in probability that the environmental risk is extremely low.

The NRC has performed a study to assess more quantitatively the risks from accidents in uranium fueled LWR's. The initial results of these efforts were made available for comment in draft form on August 20, 1974 and released in final form on October 30, 1975 as WASH-1400. This study, called the Reactor Safety Study, is an

Table IV C-39

SUMMARY OF RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENTS* PWR (WITH OR WITHOUT RECYCLE PLUTONIUM)

Class	Event	Estimated Fraction of 10 CFR Part 20 limit at site boundary**	Estimated Dose to Population 50 mile radius, person-rem
1.0	Trivial incidents	***	***
2.0	Small releases outside containment	***	***
3.0	Radwaste system failures		
3.1	Equipment leakage or malfunction	0.099	8.2
3.2	Release of waste gas storage tank contents	0.39	32
3.3	Release of liquid waste storage contents	0.011	0.9
4.0	Fission products to primary system (BWR)	NA	NA
5.0	Fission products to primary and secondary systems (PWR)		
5.1	Fuel cladding defects and steam generator leaks	***	***
5.2	Off-design transients that induce fuel failure above those expected and steam generator leak	0.002	0.19
5.3	Steam generator tube rupture	0.13	11
6.0	Refueling accidents		
6.1	Fuel bundle drop	0.021	1.7
6.2	Heavy object drop onto fuel in core	0.36	30
7.0	Spent fuel handling accident		
7.1	Fuel assembly drop in fuel storage pool	0.013	1.1
7.2	Heavy object drop onto fuel rack	0.052	4.3
7.3	Fuel cask drop	0.32	26
8.0	Accident initiation events considered in design basis evaluation in the SAR		
8.1	Loss-of-coolant accidents		
	Small break	0.22	34
	Large break	1.8	880
8.1(a)	Break in instrument line from primary system that penetrates the containment	NA	NA
8.2(a)	Rod ejection accident (PWR)	0.18	88
8.2(b)	Rod drop accident (BWR)	NA	NA
8.3(a)	Steamline breaks (PWR's) outside containment		
	Small Break	<0.001	<0.1
	Large break	0.001	0.11
8.3(b)	Steamline break (BWR)	NA	NA

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*The doses calculated as consequences of the postulated accidents are based on airborne transport of radioactive materials resulting in both a direct and an inhalation dose. Evaluation of the accident doses assumes that the environmental monitoring program and appropriate additional monitoring (which could be initiated subsequent to a liquid release incident determined by in-plant monitoring) would detect the presence of radioactivity in the environment in a timely manner such that remedial action could be taken if necessary to limit exposure from other potential pathways to man.

**Represents the calculated fraction of a whole body dose of 500 mrem, or the equivalent dose to an organ.

***These radionuclide releases are considered in developing the gaseous and liquid source term, and doses included in the Routine Release sections.

Table IV C-40

SUMMARY OF RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENTS* BWR (WITH OR WITHOUT RECYCLE PLUTONIUM)

Class	Event	Estimated Fraction of 10 CFR Part 20 limit at site boundary**	Estimated Dose to Population 50 mile radius, person-rem
1.0	Trivial incidents	***	***
2.0	Small releases outside containment	***	***
3.0	Radwaste system failures		
3.1	Equipment leakage or malfunction	0.17	14
3.2	Release of waste gas storage tank contents	0.68	56
3.3	Release of liquid waste storage contents	<0.001	<0.1
4.0	Fission products to primary system (BWR)		
4.1	Fuel cladding defects	***	***
4.2	Off-design transients that induce fuel failures above those expected	0.007	1.4
5.0	Fission products to primary and secondary systems (PWR)	NA	NA
6.0	Refueling accidents		
6.1	Fuel bundle drop	0.004	0.3
6.2	Heavy object drop onto fuel in core	0.03	2.4
7.0	Spent fuel handling accident		
7.1	Fuel assembly drop in fuel storage pool	0.007	0.53
7.2	Heavy object drop onto fuel rack	0.012	0.99
7.3	Fuel cask drop	0.25	21
8.0	Accident initiation events considered in design basis evaluation in the SAR		
8.1	Loss-of-coolant accidents		
	Small break	<0.001	<0.1
	Large break	0.058	39
8.1(a)	Break in instrument line from primary system that penetrates the containment	<0.001	<0.1
8.2(a)	Rod ejection accident (PWR)	NA	NA
8.2(b)	Rod drop accident (BWR)	0.008	1.6
8.3(a)	Steamline breaks (PWR's) outside containment	NA	NA
8.3(b)	Steamline break (BWR)		
	Small Break	0.006	0.49
	Large break	0.03	2.5

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*The doses calculated as consequences of the postulated accidents are based on airborne transport of radioactive materials resulting in both a direct and an inhalation dose. Evaluation of the accident doses assumes that the environmental monitoring program and appropriate additional monitoring (which could be initiated subsequent to a liquid release incident determined by in-plant monitoring) would detect the presence of radioactivity in the environment in a timely manner such that remedial action could be taken if necessary to limit exposure from other potential pathways to man.

**Represents the calculated fraction of a whole body dose of 500 mrem, or the equivalent dose to an organ.

***These radionuclide releases are considered in developing the gaseous and liquid source terms and doses included in the Routine Release sections.

effort to develop realistic data on the probabilities and consequences of accidents in water-cooled power reactors, in order to improve the quantification of available knowledge related to nuclear reactor accident probabilities. The Commission organized a special group of about 50 specialists under the direction of Professor Norman Rasmussen of MIT to conduct the study. As with all new information developed which might have an effect on the health and safety of the public, the results of these studies will be assessed on a timely basis within the regulatory process on generic or specific bases as may be warranted. Although there are differences in the inventories of various radionuclides in the 1.15 SGR as compared to the uranium fueled reactor as illustrated in Table IV C-13, these differences are not large enough to affect the conclusion that the consequences of large accidents in a 1.15 SGR are comparable to those from a uranium fueled reactor. Because no mechanisms have been identified which would increase the probabilities of large releases in the 1.15 SGR, the overall risks from a 1.15 SGR (considering both probabilities and consequences) can be concluded to be comparable to those of uranium fueled reactors.

Tables IV C-39 and IV C-40 indicate that the realistically estimated radiological consequences of the postulated accidents would result in exposures of an assumed individual at the site boundary which are less than or comparable to those which would result from a year's exposure to the Maximum Permissible Concentrations (MPC) of 10 CFR Part 20. The use of mixed oxide fuel does not significantly affect computed doses. The principal increase in dose source term for the GESMO model reactor over a uranium only reactor is about 4% and due to a corresponding increase in ^{131}I production. The calculated doses due to Xe and Kr are lower. Plutonium inventories are about a factor of three or four greater in the mixed oxide core, but engineered safety features would limit temperatures to well below the point at which plutonium vaporization would occur in both uranium and mixed oxide charges. The tables also show the estimated integrated exposure of the population within 50 miles of the plant from each postulated accident. Any of these integrated exposures would be less than that from naturally occurring radioactivity. When considered with the probability of occurrence, the annual potential radiation exposure of the population from all the postulated accidents is an even smaller fraction of the exposure from natural background radiation and in fact, is well within naturally occurring variations in the natural background. It is concluded from the results of the realistic analysis that the environmental risks from postulated radiological accidents are exceedingly small and need not be considered further.

REFERENCES

1. U.S. Atomic Energy Commission, Directorate of Regulatory Standards, "Environmental Survey of Transportation of Radioactive Materials to and From Nuclear Power Plants," WASH-1238, December 1972.

2. "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Regulatory Guide 1.3, June 1974.

3. "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Regulatory Guide 1.4, June 1974.

4. Final Environmental Statement Concerning Proposed Rule Making Action, WASH-1258, July 1973.

GESMO

CHAPTER IV

ENVIRONMENTAL IMPACT DUE TO THE IMPLEMENTATION OF PLUTONIUM RECYCLE

SECTION D

MIXED OXIDE FUEL FABRICATION

CHAPTER IV

Section D

MIXED OXIDE FUEL FABRICATION

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MIXED OXIDE FUEL FABRICATION

SUMMARY

General

Recycle of plutonium into fuel for light water cooled reactors (LWR's) requires production of a mixed uranium dioxide-plutonium dioxide (MOX) fuel, a step that is not a part of today's uranium fuel cycle. According to "low growth" industry estimates, plutonium recycle begins at the earliest in about the year 1981. Using these estimates, it has been projected that approximately 25,000 MT of mixed oxide fuel will be fabricated for LWR use by the year 2000. This would amount to about 13% of the total LWR fuel requirement for the 26-year study period of this generic statement.

The impacts of mixed oxide fuel fabrication will be encountered only if plutonium is recycled.

Mixed oxide fuel must be fabricated in specially designed fuel fabrication plants. A model commercial scale MOX fuel fabrication plant was developed for purposes of preparing this statement to represent the industry norm. The model plant is assumed to have a production capacity of 360 MT/yr of mixed oxide fuel--enough to maintain about thirty 1,000 MWe power plants operating at a 1.15 equilibrium SGR loading.

Although several processes, with essentially the same environmental impact, could be used for the manufacture of mixed oxide fuel, the process assumed in this evaluation is the one indicated as most likely for commercial use: blending ceramic grade PuO_2 powder from the reprocessing plant with ceramic grade natural UO_2 powder (depleted or recycled UO_2 could be used with no significant change in environmental impact). The powders are granulated and cold-pressed into pellets; the pellets are sintered, ground, washed, and loaded into Zircaloy tubes, which are then hermetically sealed by welding caps into the ends. The sealed rods (tubes) are shipped to enriched uranium oxide fuel fabrication plants for assembly into fuel bundles, elements, or assemblies.

The model MOX plant design is also assumed to include the capability to reprocess and recycle scrap and to process and package radioactive waste generated on site.

The total environmental effects and radiological impacts of the mixed oxide fuel fabrication industry shown in Tables IV D-1 and IV D-2, as projected for the 20-year period 1981-2000, were derived by multiplying the annual model plant effects and

Table IV D-1

ENVIRONMENTAL EFFECTS OF THE MOX FUEL FABRICATION INDUSTRY (1981-2000)

Environmental Effects	Industry Total Years 1981-2000
Radioactivity in Effluents	mCi(α)*
Mixed Oxide, (including U and ^{241}Am), Airborne	640
Mixed Oxide (including U and ^{241}Am), Waterborne	130

Chemical Effluents	Industry Total Years 1981-2000
Gaseous	kg
HF	26
NO_x	3.0×10^4
NH_3	4100
Liquid	
PO_4	5900
NO_3	1.2×10^5
Use of Natural Resources	
Land Use**	7.8×10^4 acre-years
Water Use	1.9×10^9 gallons
Energy Use	4.3×10^6 MWh
Thermal Effluent	9.3×10^{12} Btu

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*The term mCi(α) is used because it is a common analytical practice to use the alpha activity as a measure of the biological impact of a plutonium nuclide mixture, even though there may be more beta-emitting ^{241}Pu activity present. The computations of person-rem and organ dose include the contribution of beta dose from ^{241}Pu.

**Land use is based on 78 plant years to account for land use needed for construction prior to operation. All other impacts are based on 71 plant years, e.g., the number of operating plant years required to produce about 25,000 MT of MOX from the year 1981 through the year 2000.

Table IV D-2

RADIOLOGICAL IMPACTS: DOSE COMMITMENT* FROM THE MOX FUEL FABRICATION INDUSTRY

	Annual	Total (1981-2000)		
	Dose Commitment for an Individual**	Dose Commitment for the General Population	Dose Commitment for Occupational Exposures	
Normal Operation	Closest Theoretical Resident, (mrem/yr)	Total Population, (person-rem)	Plant Employees-Total, (person-rem)	
Organ			Internal	External†
Whole Body	3.7	300		25,000
Bone	171.0	14,000		
G.I. Tract	0.15	16		
Lung	3.9	300	4,000	
Liver	17.7	1,400		
Kidney	15.2	1,250		
Skin	0.07	5.6		
Accidents***				
Bone	26.5	0.5		
Thyroid	360	4.2		

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*See CHAPTER IV, Section J, Appendix A for explanation of exposure modes and duration incorporated into the dose commitment determination.

**Individual dose commitments are based upon the 40th and last assumed year of operation of the model plant. All other dose commitments are based on the total industry from 1981 to 2000. Since the individual dose commitment for the water pathway is insignificant compared to that from the air pathway, the individual dose commitment has been based on the air pathway.

***The population dose commitment for accidents is based on the occurrence of one accident of each type (fire, criticality, and explosion) during the 1981-2000 period.

†External whole body occupational exposure includes penetrating exposure to the whole body from neutron and gamma radiation.

impacts by the number of model plant years required to produce the projected total of 25,000 MT of mixed oxide fuel for the period: 25,000 MT ÷ 360 MT/yr ≈ 71 model plant years.

The impacts shown to occur during the 1981-2000 period covered by this section are the total environmental impacts resulting from the commercial mixed oxide fuel fabrication industry from the year 1975 to the year 2000.

Environmental Effects

Radioactivity in Effluents - It is estimated that from 1981-2000, the mixed oxide fuel fabrication industry would release a total of about 640 mCi(α) to the atmosphere and about 130 mCi(α) to water bodies.

Chemical Effluents - The airborne chemical effluents include small quantities of fluoride, nitrogen oxides and ammonia. Liquid effluents consist of treated sanitary waste, cooling tower blowdown, and plant maintenance solutions. Estimates indicate that concentrations of all chemicals in these effluents will be well below established Federal and State standards and will have an insignificant impact on the environment. The quantities of chemicals estimated to be released by the mixed oxide fuel fabrication industry are about 1% to 5% of the estimated quantities that would be released if an equivalent amount of fuel were provided by the UO_2 fuel fabrication industry. For detailed data on the UO_2 fuel fabrication industry, refer to CHAPTER IV, Section F-4.0.

Use of Natural Resources - The land commitment for a model plant on a separate site is assumed to be 1,000 acres. The total land commitment of the mixed oxide fuel fabrication industry for the period 1981-2000 has been conservatively projected to be 7.8×10^4 acre-years, assuming that all plants are located at sites separate from other fuel cycle facilities. The land commitment would be reduced substantially if mixed oxide fuel fabrication plants were collocated with fuel reprocessing plants or with low enriched UO_2 fuel fabrication plants. Calculations show that deposition of plutonium on the plant site land will be so slight that the land may be returned to unrestricted use after plant decommissioning. If plutonium is not recycled, it is estimated that an additional temporary land commitment of about 5.5×10^6 acre-years and an additional permanent commitment of about 12,000 acres will be incurred by the uranium mining and milling industry to produce an equivalent amount of UO_2 fuel.

The total 20-year industry water use of 1.9×10^9 gallons, energy use of 4.3×10^6 MWh and thermal effluent of 9.3×10^{12} Btu are comparable to the water and energy use and thermal effluents of about two or three large commercial laundries or 250 to 300 private homes during the same period. Approximately twice this amount of energy and water would be consumed if an equivalent amount of fuel were to be produced at enriched UO_2 fuel fabrication plants.

Radiological Impacts

Closest Theoretical Resident - Radioactive effluents released to water bodies by a model mixed oxide fuel fabrication plant result in insignificant dose commitments to individuals. Under conservative assumptions tending to overstate dose commitments, the annual dose commitment to the bone of the closest theoretical resident, resulting from the water pathway, is 0.004% of the average organ dose each individual in the U.S. population receives from natural sources. Annual dose commitments to other body organs and the whole body of the closest theoretical resident are about an order of magnitude less than the bone dose commitment.

The airborne pathway is clearly the dominant pathway for exposure to the closest theoretical resident. This individual is considered as continuously residing and deriving his entire livelihood at a point 500 meters in a predominantly downwind direction from a plant. Calculations tending to overstate dose commitments show that the potential average annual dose commitment to this closest theoretical resident is about 171 mrem to the bone, 18 mrem to the liver, 15 mrem to the kidney, 4 mrem to the lung, 4 mrem to the whole body, and less than one mrem to the G.I. tract and skin. The estimated bone dose commitment to the closest theoretical resident is about twice that which an average individual receives annually from natural background irradiation* and about 35% of the limit recommended by the NCRP and the ICRP for individuals in the general population. The liver, kidney, lung and whole body dose commitments are respectively about 3.5%, 3.0%, 1.0%, and 0.5% of these recommendations. Realistically, doses from the actual plant operation are expected to be much lower. Further, plant effluents will be kept as low as reasonably achievable.

General Population - Airborne radioactive effluents released by the mixed oxide fuel fabrication industry during the 1981-2000 period will contribute an estimated dose commitment to the population of the United States to the bone (the maximally affected organ) of about 14,000 person-rem. Population liver and kidney dose commitments are estimated to be about one-tenth, and the lung dose commitment about one-fortieth, of the population bone dose commitment.

The population bone dose commitment, resulting from waterborne effluents, has been conservatively estimated at 16.5 person-rem. Population dose commitments to other organs of the body and the whole body, calculated for the water pathway over the 20-year period, are about an order of magnitude less than the population bone dose commitment.

The total population bone dose commitment from both the waterborne and airborne pathways from the mixed oxide fuel fabrication industry, from the year 1981-2000, is estimated to total about 14,000 person-rem to the population of the United States. This estimate was made by assuming the form of plutonium to be insoluble for lung dose calculations and soluble for other organ dose calculations which maximizes the estimated doses to all organs. Population dose commitments to other organs of the body and the whole body are at least an order of magnitude less.**

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*Average natural background irradiation is assumed to be about 100 mrem/yr.

**If plutonium is assumed to be insoluble for dose calculations of organs other than the lung, these doses would be reduced by about an order of magnitude.

The dose commitment to the population of the United States from MOX fuel fabrication plant operation from 1975-2000 is about 1 x 10^{-3} of the total dose commitment to the United States population from the recycle LWR industry during this period.

Occupational Exposure - Conservative estimates of external and internal dose commitments to employees of a model mixed oxide fuel fabrication plant indicate each may receive an average annual whole body dose commitment of approximately 1.2 rem from external radiation and 0.2 rem from internal radiation. These doses correspond to about 23% and 1%, respectively, of the maximum annual dose limit for individual occupational exposure specified in 10 CFR Part 20.

Total occupational dose commitments for the mixed oxide fuel fabrication industry during the time period under consideration are about 25,000 person-rem to the whole body and about 4,000 person-rem to the lung.

Accidents - The analysis indicates that the probability of major accidents occurring at the model MOX fuel fabrication plants is quite low. Radiological impacts resulting from upper limit accidents have been determined. The results show that the effects of a criticality accident occurring at the model MOX fuel fabrication plant are similar to the effects of a criticality accident occurring at a model enriched uranium fuel fabrication plant or in the plutonium conversion section of a reprocessing plant. The dose commitments resulting from a criticality accident at a model MOX fuel fabrication plant have been estimated to be 360 mrem (thyroid) to the closest theoretical resident and 4.2 person-rem (thyroid) to the U.S. population. The impact from a fire in a model MOX plant is considered the same as the impact from an explosion in the plant. The dose commitment for either of these accidents is estimated to be 21.5 mrem (bone) to the closest theoretical resident and 0.7 person-rem to the bone of the U.S. population. The bone dose commitments are about 12% of the closest theoretical resident and about 0.1% of the U.S. population dose commitments resulting from 1 year of normal model MOX fuel fabrication plant operations.

1.0 PRODUCTION OF MIXED OXIDE FUELS FOR LWR's

Recycle of plutonium into light water cooled reactor fuel would require production of mixed uranium dioxide-plutonium dioxide fuel pellets--a step that is not part of the uranium fuel cycle. Mixed oxide fuel rods may be used to form separate fuel assemblies of mixed oxide fuel (PWR concept), or islands of mixed oxide fuel surrounded by enriched UO_2 rods in a single assembly (BWR concept). With either concept, other assemblies in the core may consist entirely of enriched UO_2 rods.

This environmental statement is based on a study period from the year 1975-2000 and a "low growth" estimate in which plutonium recycle in the LWR industry begins no earlier than the year 1981. Use of this growth model indicates that about 25,000 MT of mixed oxide fuel will be manufactured for LWR's from 1981-2000, assuming no breeder use of plutonium. This is about 13% of the total LWR fuel requirement for the 1975 to 2000 period. Introduction of fast breeder reactors would result in a slight decline in the use of plutonium in light water reactors, because some of the LWR-generated plutonium would be used to fuel breeders.

1.1 Status of Mixed Oxide Fuel Fabrication Industry

Five licensees have possession limits high enough to enable them to produce mixed oxide fuel for light water reactors on a pilot plant scale in the near future. Table IV D-3 lists these organizations. The total annual production capacity of the existing U.S. fuel fabrication industry for mixed oxide fuel for LWR's is estimated to be 50 to 75 MT per year. Table IV D-4 gives geographic and demographic data for the five existing MOX fuel manufacturing plants.

At present, there is little plutonium available in the United States from commercial sources. But as more LWR power plants become operational, more plutonium will become available as reprocessing capability is provided. If all the available plutonium were to be recycled promptly, additional MOX fuel fabrication facilities would be required in the early 1980's. Westinghouse has submitted a license application for a commercial scale mixed oxide fuel fabrication plant that could start up approximately 2 to 3 years after construction begins.[3] Many of the siting and design characteristics of this proposed plant are used as the basis for the model MOX plant in this study. CHAPTER III further discusses the mixed oxide fuel fabrication industry through the year 2000.

Existing pilot scale mixed oxide fabrication plants, with the exception of Exxon, can use either plutonium nitrate solution or solid plutonium dioxide, or both, as the plutonium feed material. NRC regulations prohibit shipment of plutonium compounds in liquid form,[4] except in small amounts, after the year 1978. Hence it is assumed that the commercial scale mixed oxide fuel fabrication plants will use some solid form as the starting material. Allied-General Nuclear Services (AGNS) has submitted an application to the Commission to build a plutonium nitrate-plutonium dioxide conversion facility as a part of its Barnwell Nuclear Fuel Plant complex.[5] Exxon has applied for a license for a fuel reprocessing facility that will include plutonium conversion capability.

Table IV D-3

EXISTING MOX FUEL FABRICATION PLANTS[1]

Licensee	Plant Location	Feed Material	Plant Product	Pu Possession Limit (kg)	Est. Production Capacity (MT/yr)
Babcock & Wilcox	Parks Township, Pa.	Nitrate solution or PuO_2, U nitrate or (U and Pu) O_2	$(U,Pu)O_2$ fuel rods	2000	20
Exxon Nuclear	Richland, Wash.	$UO_2 + PuO_2$	$(U,Pu)O_2$ fuel assemblies	10 unencapsulated; 100 total	15
General Electric	Pleasanton, Calif.	Nitrate solution (U and Pu)	$(U,Pu)O_2$ fuel rods	150	3
Kerr-McGee	Crescent, Okla.	Nitrate solution (U and Pu)	$(U,Pu)O_2$ fuel rods	360	5-10
Westinghouse	Cheswick, Pa.	Pu nitrate solution PuO_2, U nitrate, or UO_2	$(U,Pu)O_2$ fuel rods	120	10-15

Table IV D-4

GEOGRAPHIC AND DEMOGRAPHIC DATA FOR EXISTING MOX FUEL FABRICATION PLANTS[1,2]

Licensee and Plant Location	Site Size (Acres)	Nearby Population Centers		
		City	Population	Distance (miles)
Exxon Nuclear Richland, Washington	160	Richland Pasco Kennewick	35,000 25,000 20,000	3 12 20
General Electric Pleasanton, California	1,600	San Jose San Francisco	450,000 720,000	20 35
Kerr-McGee Crescent, Oklahoma	1,000	Oklahoma City	363,000	30
Babcock & Wilcox Parks Township, Pennsylvania	40	Pittsburgh	520,000	40
Westinghouse Cheswick, Pennsylvania	125	Pittsburgh	520,000	18

1.2 Plutonium Isotopic Composition

Several plutonium isotopes are generated in LWR uranium fuel during irradiation. If this plutonium is then recycled as fuel to LWR's, the isotopic composition of the plutonium in irradiated mixed oxide fuel will be different from that in the previously irradiated uranium only fuel. The isotopic composition of plutonium that is to be used at the fabrication plant will be a function of the fuel irradiation history, the amount of plutonium product blending at the fuel reprocessing plant, and the elapsed time between the recovery of plutonium at the reprocessing plant and its use at the fabrication plant.

The isotopic compositions of four mixtures of recycle plutonium are shown in Table IV D-5. The first recycle feed represents the plutonium composition without recycle, estimated for uranium fuel irradiated to 33,000 MWd/MT in LWR's. The second, third, and fourth recycle plutonium is the calculated isotopic composition of recycle plutonium, aged 1 year between reprocessing and fabrication, resulting from successive recycle fuel reloadings containing plutonium at the 1.15 SGR level.[6] The specific activities of these mixtures and a calculated 20-year industry average specific activity are shown in Table IV D-6.

The calculated isotopic composition of the industry average recycle plutonium was used in this report for estimating source terms and radiological exposures.

Table IV D-5

ISOTOPIC CONTENT OF PLUTONIUM

AGED 1 YEAR AFTER REPROCESSING[6]

Isotope	1st Recycle (weight %)	2nd Recycle (weight %)	3rd Recycle (weight %)	4th Recycle (weight %)
^{238}Pu	2.5	3.2	4.2	5.
^{239}Pu	57.	40.	34.	31.
^{240}Pu	23.	30.	30.	27.
^{241}Pu	11.	15.	16.	16.
^{242}Pu	5.2	10.	15.	20.
*^{241}Am	0.63	0.81	0.86	0.86

*Because of decay of ^{241}Pu during the first year after reprocessing.

Table IV D-6

SPECIFIC ACTIVITY OF AVERAGE RECYCLE PLUTONIUM

Recycle	Specific Activity Ci/g α	Specific Activity Ci/g β	MT Fabricated 1981-2000
1st	.54	13	780
2nd	.68	17	250
3rd	.84	18	49
4th	1.0	18	5

Average 20-year specific activity-Ci/g.

$\alpha = 0.59$

$\beta = 14$

1.3 Mixed Oxide Fuel Fabrication Technology

Production of fuel rods containing MOX sintered ceramic fuel pellets is a combination of chemical and mechanical operations.* A chemical process is used in some existing MOX fuel fabrication plants to convert purified $Pu(NO_3)_4$ to PuO_2 powder, which is then mechanically blended with ceramic grade UO_2 to produce mixed oxide for fabrication into pellets. Alternatively, MOX powder can be produced by the coprecipitation of ammonium diuranate (ADU) and plutonium hydroxide mixed crystal, by the addition of ammonia to a mixture of the nitrate solutions. Other methods have also been used to prepare mixed oxide powders for fuel manufacture.[7] Scrap recovery operations can

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*This environmental impact statement assumes that liquid to solid plutonium conversion operations will be performed at the reprocessing plant. Since some of the liquid to solid operations are now used in some existing MOX pilot fuel fabrication plants, they are included here to provide a complete description of proven conversion processes.

involve nitric acid dissolution and recovery of plutonium and uranium by conventional wet chemical processes. An alternative scrap recovery method involves oxidation and subsequent reduction of dry PuO_2 and UO_2. Descriptions of the dominant processes for conversion of plutonium from nitric acid solutions to PuO_2 or mixed oxide are given below, together with a description of the mechanical processing for the manufacture of mixed oxide fuels.

1.3.1 Oxalate Process

The principal steps involved in producing PuO_2 from $Pu(NO_3)_4$ by the oxalate process are outlined in Figure IV D-1 and are described briefly below:

- Adjustment of feed solution acidity by addition of water or nitric acid and valence adjustment by addition of H_2O_2 or other reducing agents
- Precipitation of plutonium by the addition of oxalic acid
- Filtration to obtain (1) a plutonium oxalate cake and (2) a filtrate which is sent to the liquid radwaste system or recycled to the scrap recovery system
- Drying of the moist filter cake
- Calcination of Pu oxalate to PuO_2
- Milling and screening to the desired screen size to produce the PuO_2 powder

1.3.2 Coprecipitation Process

The principal steps in this process, shown diagrammatically in Figure IV D-2, are

- Mixing of the two nitrate solutions
- Precipitation of mixed plutonium hydroxide and ammonium diuranate by the addition of ammonia
- Filtration to yield a moist cake product and to collect filtrate that is sent to the liquid radwaste system or recycled to the scrap recovery system
- Drying of the moist cake to produce a $Pu(OH)_4$ and ammonium diuranate (ADU) mixed solid
- Calcining to PuO_2-U_3O_8 mixed powder
- Reduction of PuO_2-U_3O_8 to $(U, Pu)O_2$
- Milling and screening to the desired screen size to produce the MOX powder.

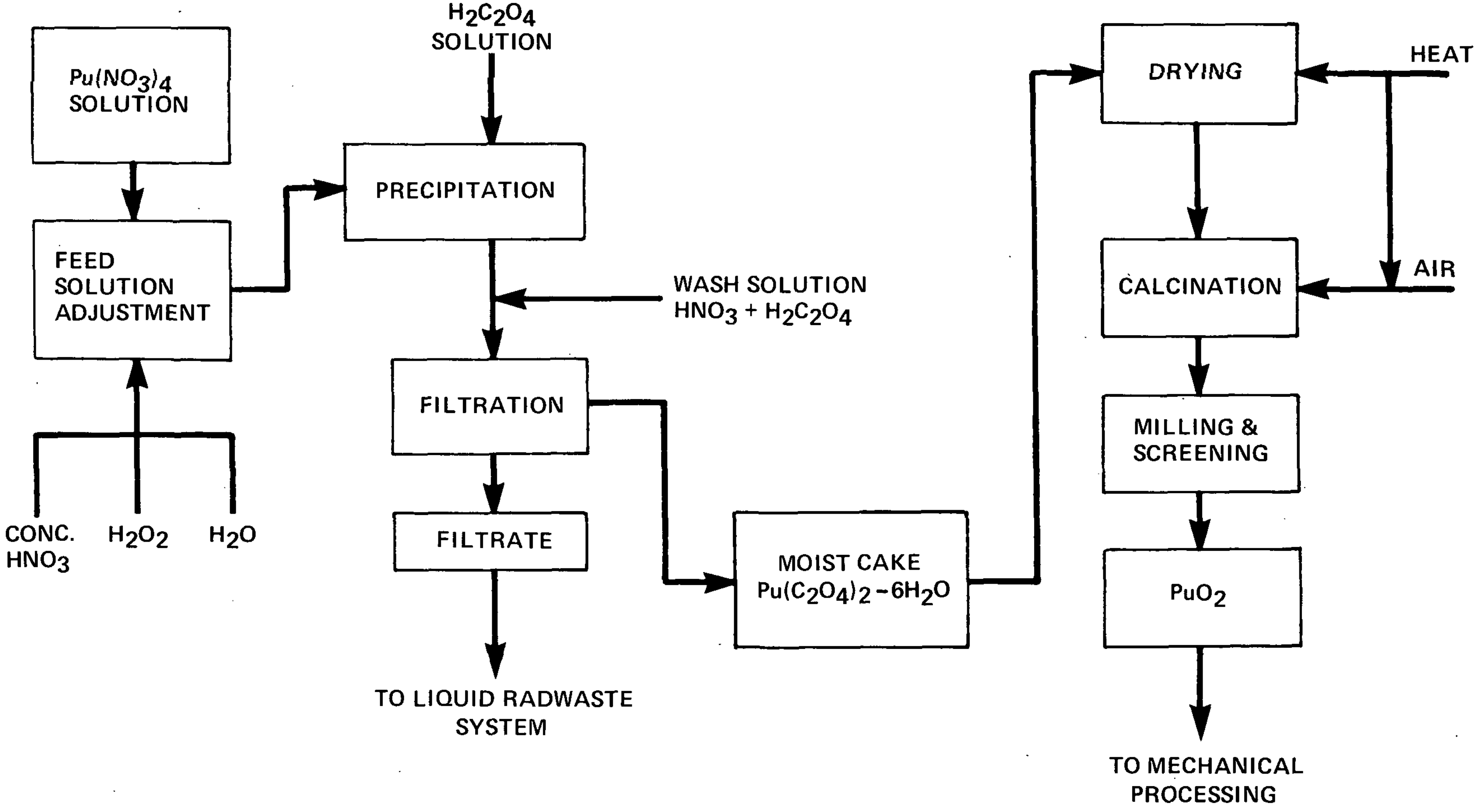
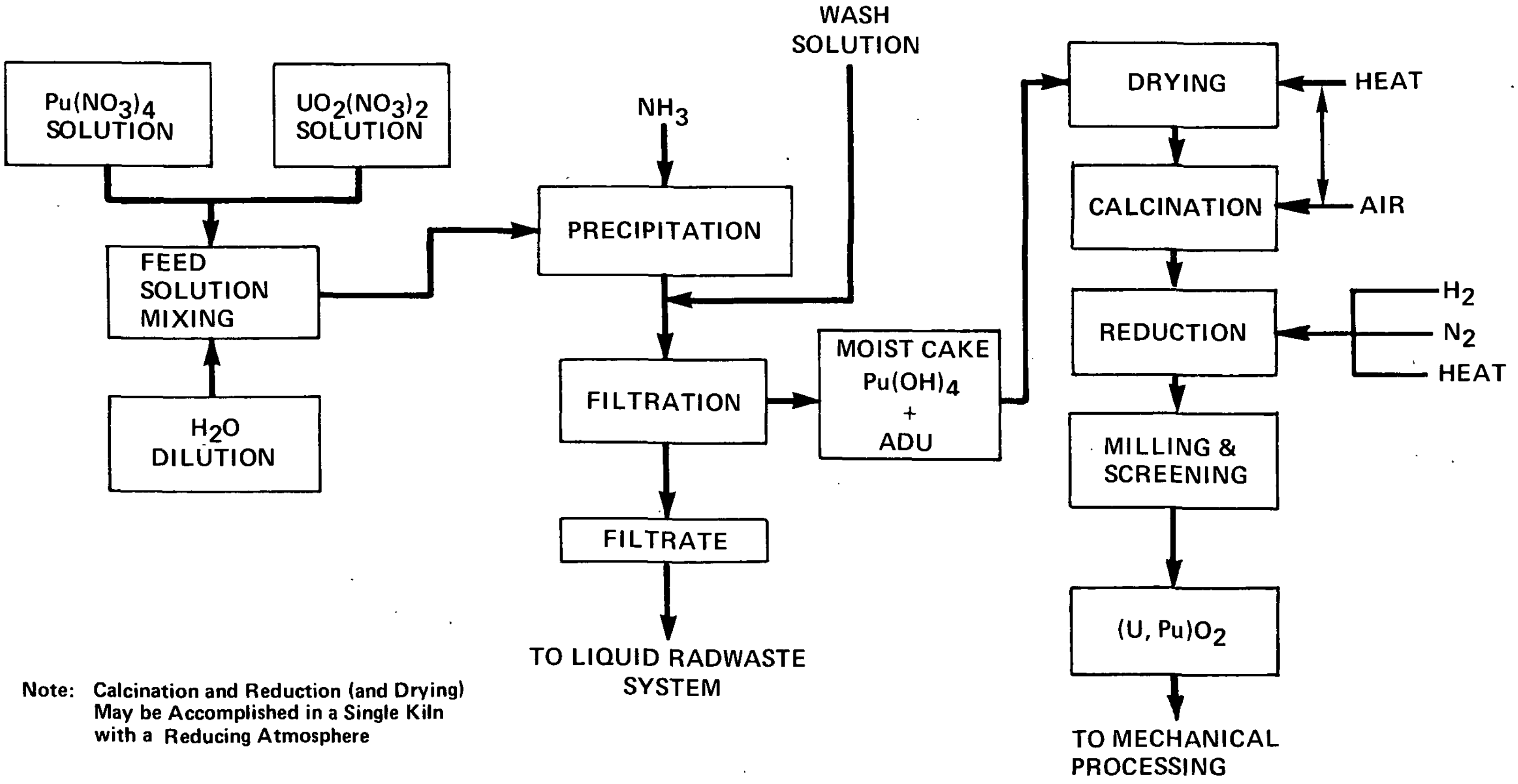


Figure IV D-1 Oxalate Conversion Flowsheet for PuO_2



Note: Calcination and Reduction (and Drying) May be Accomplished in a Single Kiln with a Reducing Atmosphere

Figure IV D-2 Coprecipitation Conversion Flowsheet for Mixed $(U, Pu)O_2$

1.3.3 Mechanical Processing of Feed Powders to Mixed Oxide Fuel Rods[1]

Mixed oxide fuel is produced from feed powders by mechanical processing (Figure IV D-3). Feed powders are converted to pellets and loaded into fuel rods. The principal steps are

- Blending of feed powders for the process (these may include PuO_2, UO_2, MOX or recycled scrap)
- Pretreatment of mixed UO_2 and PuO_2 powders by comminution, compaction, and granulation to the desired consistency
- Pelletizing
- Sintering of the pellets*
- Grinding pellets to finished dimensions
- Cleaning and drying the pellets
- Loading the pellets into fuel rods, decontaminating the rod ends, and welding the end caps
- Decontamination of the entire rod, if necessary
- Inspection of completed rod for integrity
- Packaging in DOT approved containers for shipment

Sintering furnace hearth sizes are limited by design and construction considerations, so that mixed oxide sintering furnaces will probably be limited to about 50 MT/yr each. To reach the model plant design capacity, it is expected that multiple lines of equipment would be used and that each line would contain more than one sintering furnace.

1.4 Scrap Recovery

In manufacture of mixed oxide fuel, a fraction of the mixed oxide pellets are imperfect and are rejected during testing and inspection procedures associated with the plant product quality assurance program. In addition, excess mixed oxide powder may be blended and mixed oxide pellets manufactured to ensure that adequate finished product is produced to meet contractual commitments. Such material and most of the grinder fines, designated as clean scrap that does not require a chemical purification, may be processed through sequential crushing, air oxidation, and N_2-H_2 reduction cycles. The

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*Sintering may include a binder removal step at lower temperatures.

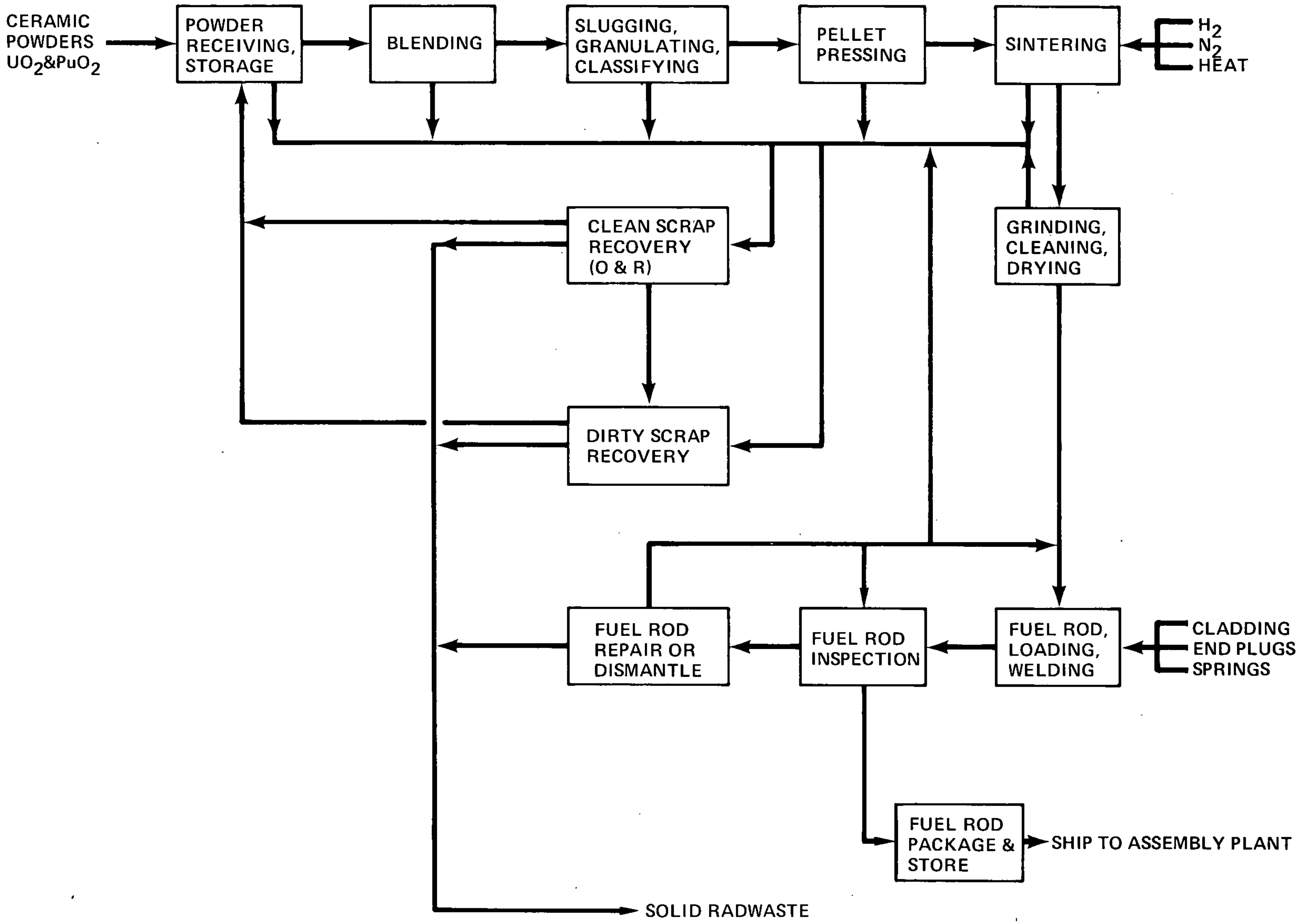

Figure IV D-3 Fuel Fabrication – Mechanical Processing Block Diagram resulting powder, after milling and size classification, may be recycled into the production process at the powder blending step, as shown on the upper portion of Figure IV D-4.

In the fuel fabrication process, small fractions of the mixed oxide powder and pellets are contaminated with impurities from the process equipment--through corrosion, chipping, etc.--and are designated as dirty scrap that requires chemical purification. Other process residues classified as dirty scrap may include such contaminants as analytical residues, metal scrap, plastic, rubber, cellulosic cleaning materials, and organics. Various head-end treatments include washing or leaching of solids, incineration of combustibles, or dissolution of this scrap with nitric acid containing a small amount of fluoride. The resulting impure nitrate solutions are then processed through solvent extraction for purification of plutonium and uranium. In commercial scale plants, the uranium and plutonium may be recovered as a mixture or separately. The process operations of scrap dissolution and solvent extraction are similar to operations used for dissolution and extraction in fuel reprocessing plants. An alternative method of purification is the use of ion exchange, which results in a high purity plutonium product, but loses all uranium to the waste stream and necessitates a proportionately higher uranium makeup at the head end of the process. Recovered plutonium and uranium nitrate solutions from scrap purification are converted to solid forms suitable for recycle to the main process stream by either oxalate or ammonia precipitation.

1.5 Radwaste Systems

During normal operation of the plant, processing of liquid and solid radioactive material will produce contaminated effluents. These effluents are treated by the radioactive waste treatment system, so that releases of radioactivity are maintained at levels that are as low as reasonably achievable (ALARA).

The liquid radwaste system, consisting of evaporator-fractionator systems, is used to reduce the volume of contaminated liquids and to recover process chemicals and water for recycle to the process. Because mixed oxide fuel fabrication is a dry process, there are only a few support operations yielding liquids that may be plutonium contaminated. These operations include

- Wet processing of contaminated scrap
- Process off-gas scrubbing
- Wet decontamination operations
- Miscellaneous liquid waste generating activities such as laundry and personnel showers

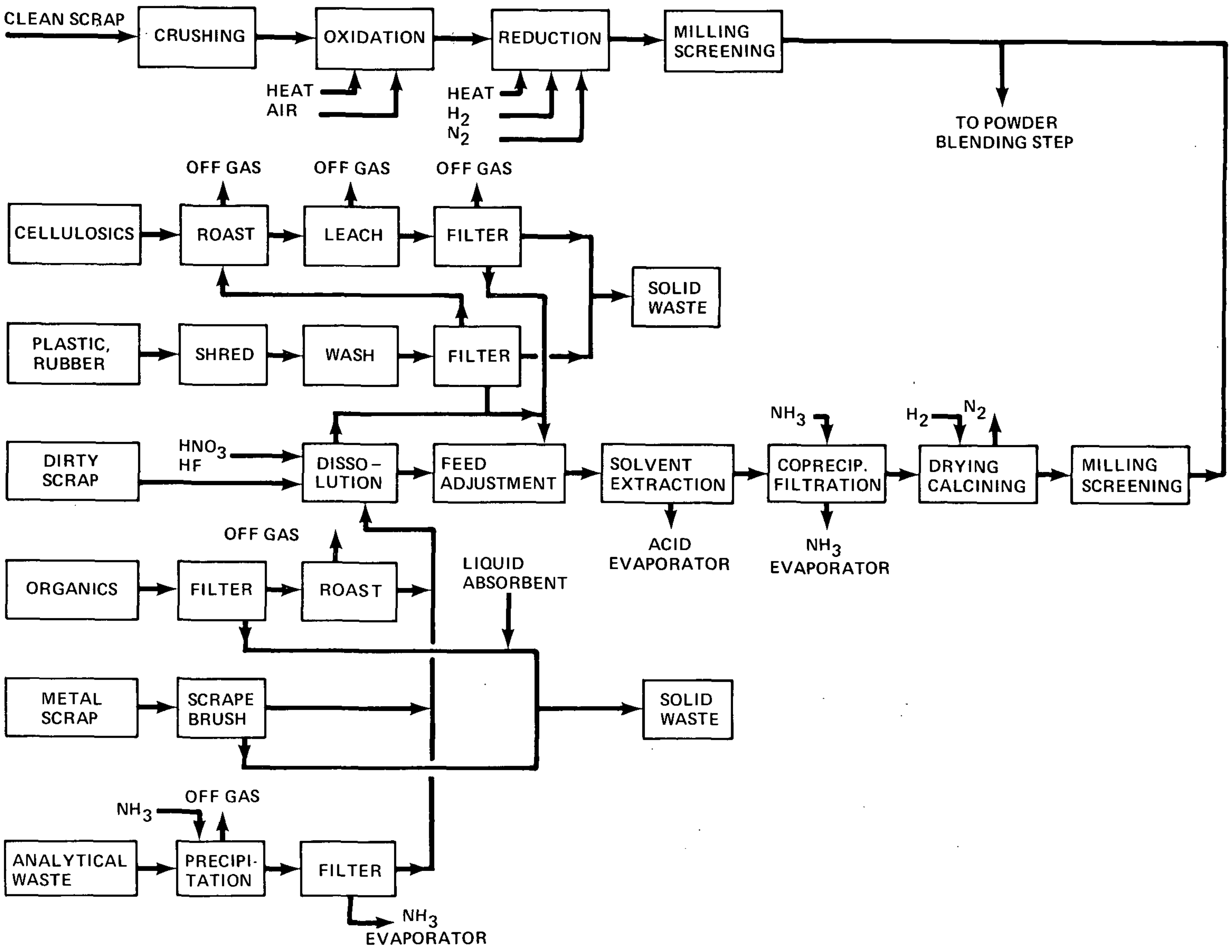

Figure IV D-4 Scrap Recovery Processes

Most of the distillate or excess water will be sampled and released from the plant if the level of radioactive material is below the limits set in 10 CFR Part 20.

The solid radwaste system is designed to solidify residual liquids such as excess process water and evaporator bottoms and to package residual solids like room trash, incinerator ash and contaminated equipment for disposal in accordance with NRC regulations.

Present technology for processing liquids containing plutonium that cannot be recovered economically but is at too high a concentration to release to the environment is to incorporate the liquids into a solid matrix, either with cement or an equivalent solidification agent.

Although future technology cannot be predicted, a reasonable assumption is that plutonium-contaminated liquids will be reduced to the minimum practicable volume by evaporation, perhaps to dryness, before being processed and packaged for shipment to a repository. Such solids as paper, cans, and filters are compacted and packaged in drums at the fabrication plant for disposal, compatible with further treatment at a Federal waste repository.

In consideration of airborne effluents, a series of high efficiency particulate air (HEPA) filters in the plant ventilation system will remove airborne radioactive material. The concentration of radioactive material released to the environment through the HEPA system will be a small fraction of the limits presented in 10 CFR Part 20.

Plants may use one or all of the radwaste operations, depending on the plant processing and throughput. Flowsheets for the model mixed oxide fabrication plant radwaste system are presented later in this section.

2.0 COMMERCIAL SCALE MIXED OXIDE FUEL FABRICATION PLANT*

2.1 Model Plant

2.1.1 Plant Description

The model commercial scale mixed oxide fuel fabrication plant is patterned after the proposed Westinghouse Recycle Fuels Plant and is assumed to be typical of mixed oxide fuel fabrication plants during the 1981 to 2000 period.[8] It has been assumed to have an actual production rate of 360 MT/yr of mixed oxide fuel, enough plutonium reload fuel for about 30 reactors operating at the 1.15 SGR loading. Major plant design characteristics are given in Table IV D-7 and are described below.

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*Much of the information on model plant siting and process design is based on information contained in Reference 8.

Table IV D-7

COMMERCIAL SCALE MODEL MIXED OXIDE FUEL FABRICATION[1,8] PLANT CHARACTERISTICS

Production Rates:	
Operating Capacity	360 MT (U, Pu)O_2/year
Scrap Recycle Rate	36 MT (U, Pu)O_2/year
	30 MT clean scrap
	6 MT dirty scrap
Process Procedures	
Product:	Dry blend of PuO_2 and UO_2 (to pelletize and encapsulate in LWR fuel rods)
Scrap:	Clean: oxidation and reduction
	Dirty: nitric acid dissolution, solvent extraction to recover nitrate solution, coprecipitation and calcination
Liquid Waste Treatment:	Process chemicals are recovered and recycled; all contaminated liquids are either, (1) solidified in concrete, or (2) evaporated, with water being recycled or solidified with evaporator bottoms, or, (3) sampled and released as liquid effluent
Site Size:*	1,000 acres
Ventilation Rate:	45,000 cfm
Electrical Requirements:	7 MWe
Water Requirements:	70,000 gal./day
Cooling Water Blowdown	12,000 gal./day
Sanitary Water	15,000 gal./day
Potentially Contaminated Liquids	5,600 gal./day
Cooling Tower Capacity	15,000,000 Btu/hr

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*Although mixed oxide fuel fabrication plants may be collocated with reprocessing or uranium fuel fabrication plants, the model MOX plant has been assumed to be located on a separate site.

The model mixed oxide fuel fabrication plant is assumed to be located on a site in a rural location, separate from any other plant involved in the LWR fuel cycle. The site size is 1,000 acres, near the upper range of anticipated sizes for this type of facility. The distance from the plant to the nearest boundary is 500 meters. The plant complex includes a fuel manufacturing building designed to withstand the effects of natural phenomena (including tornadoes, earthquakes, and floods), a separate warehouse, administration facilities, field storage tanks (for chemicals such as ammonia, nitrogen, and hydrogen), a guardhouse, a site and plant security building, and other miscellaneous facilities. The structures are air conditioned and electrically heated for personnel comfort.

2.1.2 Design Criteria

2.1.2.1 Quality Standards

Structures, systems, and components of the manufacturing facilities important to safety are designed, fabricated, erected, and tested in accordance with quality assurance criteria in Appendix B of 10 CFR Part 50. Fuel rods will be produced under rigid quality control and quality assurance programs.

2.1.2.2 Systems Important to Safety

Systems important to safety in the model MOX plant are designed to withstand the effects of the most severe natural phenomena expected to occur at the site--including earthquakes, tornadoes, and flooding--without loss of capability to perform their safety functions. These systems also will be appropriately protected against dynamic effects, including the effects of missile impacts that result from equipment failure or external events.

2.1.2.3 Fire Protection

Structures, systems, and components are designed and located to reduce the effects of fires and explosions. Each process enclosure has appropriate automatic fire detection and protection systems with alarms and sufficient redundancy to permit reliable fire protection. Final filters are protected by sprinklers, mist eliminators, and roughing filters to reduce the effect of heat, smoke, water, and large loadings of particulate matter.

2.1.2.4 Physical Security and Accountability

Safeguards considerations are covered in a separate portion of this statement devoted to the environmental impact of safeguards.

2.1.2.5 Reliability and Testability--Redundancy and Independence

All systems necessary for the safe operation of the model MOX plant are designed for functional reliability and in-service testability. These systems have inherent redundancy and independence sufficient to prevent loss of their safety function during abnormal conditions and would assure the capability for safe shutdown of plant operations in the event of an emergency.

2.1.2.6 Confinement Zones, Ventilation Systems, and Cooling Systems

Multiple barriers are used to limit release of plutonium from the model MOX plant manufacturing building. These include both a series of structural barriers to form zones or areas and zoned ventilation systems. Primary confinement is provided in Restricted Access Areas (RAA) by process enclosures such as shielded gloveboxes or hot cells within which the plutonium handling equipment is located. Outside the RAA there may be an area used for operating and maintenance, designated as Limited Access Area (LAA), which serves to contain any leakage of contamination from the RAA. The limited access barrier forms a fire and shielding wall. The final confinement is provided by the building walls, which enclose the Normal Access Areas (NAA).

Pressure differentials are maintained between areas so that air flows from non-contaminated areas into areas of potentially higher contamination levels ($p\ RAA < p\ LAA < p\ NAA <$ atmospheric pressure). Differentials are maintained by automatically controlled zone ventilation systems that are equipped with redundant, independent emergency power supplies.

The model MOX plant exhausts process air through a minimum of three high efficiency particulate air (HEPA) filters, with the first HEPA filter usually located on the glovebox. The two final stages have in-place testability. Figure IV-D-5 shows the schematic air flow diagram. The total building ventilation rate is approximately 45,000 cfm.

A double confinement concept is also used in the process cooling water system design. Process cooling is accomplished by a closed primary loop circulating system that is cooled by a secondary cooling water system. Heat is removed from the secondary system via a cooling tower.

2.1.2.7 Instrumentation and Controls

Instrumentation in the model MOX plant monitors system functions for normal operating ranges, anticipated operational occurrences, and accident conditions. Appropriate controls maintain these systems within specified operating ranges.

2.1.2.8 Electrical Power Systems; Emergency Power

Onsite emergency power systems and an offsite electrical power system will permit certain process systems and systems important to safety to function when needed. These emergency power sources are designed to be independent, reliable, and testable.

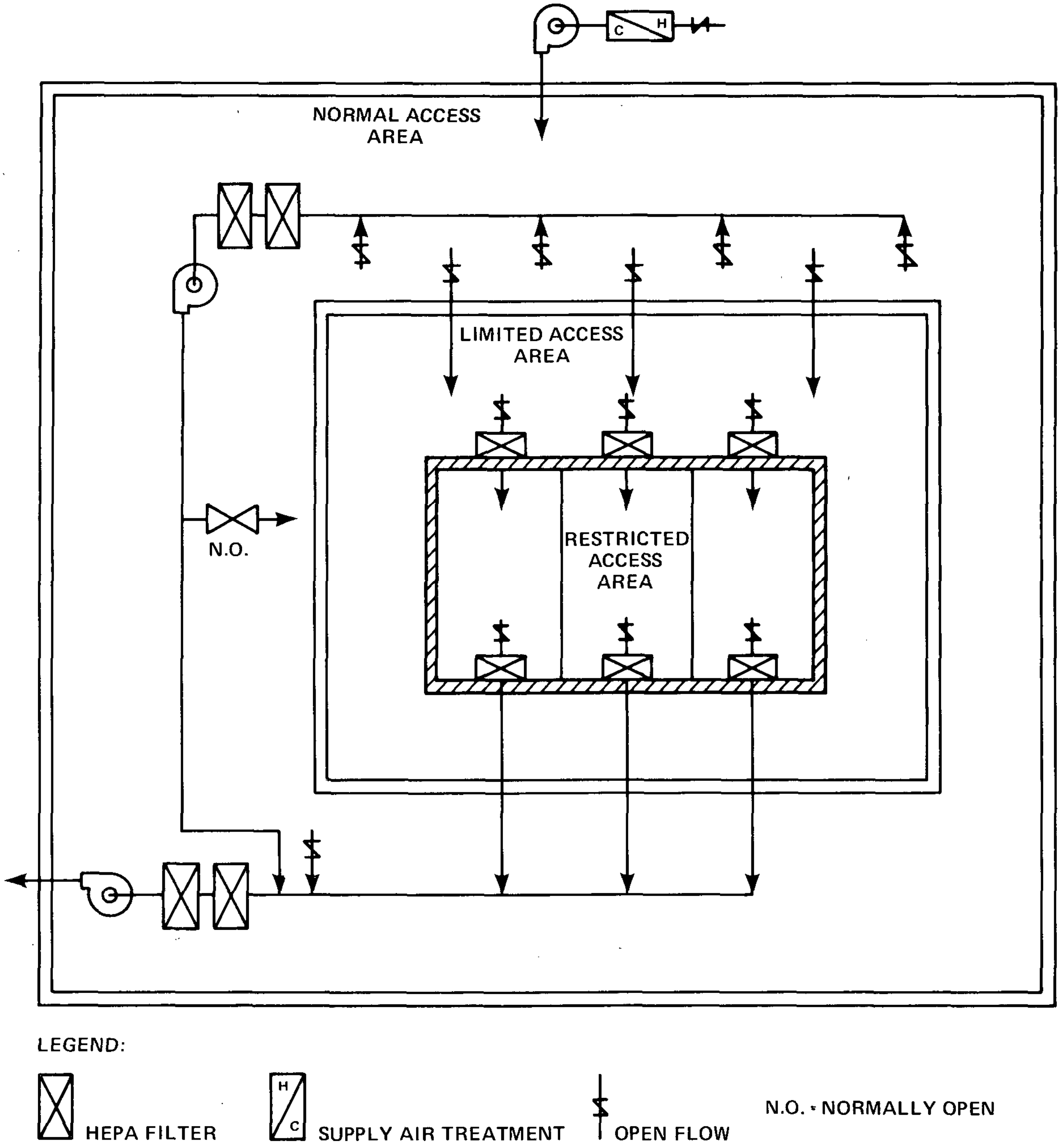

Figure IV D-5 Simplified Air Filtration Flow Diagram

2.1.2.9 Criticality Prevention

The model MOX plant is designed and operated to reduce the probability of criticality accidents. Equipment is designed to prevent criticality by utilizing favorable geometry or fixed neutron poison controls where practicable; standard operating procedures limit masses and concentrations of fissile material in process equipment and, in combination with administrative control, reduce the probability of criticality accidents. All spilled process water and water that may be used in fire control is drained to geometrically favorable or fixed neutron poison controlled tanks.

2.2 Model MOX Plant Processes

2.2.1 Product Manufacture

It is assumed that the model MOX plant receives its feed material in the form of recycled plutonium dioxide and natural uranium dioxide. Enriched, recycled, or depleted UO_2 could be used with no significant change in environmental impact. The mixed oxide fuel is manufactured in the model MOX plant by mechanical blending, pressing, and sintering operations, all within gloveboxes or cell enclosures, to produce fuel pellets for onsite assembly into fuel rods. Figure IV D-3 shows the process flowsheet.

The feed material, PuO_2 powder and UO_2 powder, is received and stored on site. As the process requires, this material is fed into a blender in specified quantities, blended, sampled and stored as a mixed oxide (MOX powder). The MOX powder is then compacted into wafers and fed into a granulator. The resulting granules are classified according to size. Acceptable granules are fed into a pellet press to produce green pellets, which are then inspected and loaded into sintering boats. Rejected pellets, along with undersized particles from the granulator, are cycled to clean scrap.

An automatic system transfers the loaded boats from green pellet storage to the sintering furnace, and as boats of sintered pellets leave the furnace, another system moves each boat to an indexed storage slot. Specified samples are taken from storage for process control checks such as density, homogeneity, and metal-to-oxygen ratio. Underfired pellets are resintered; overfired or otherwise unacceptable pellets are cycled to clean scrap recovery. Acceptable pellets are stored until needed, then ground to specification for loading into fuel rods. (Grinding sludge is also cycled to clean scrap recovery.)

The cladding tubes, end plugs, and springs received from offsite sources are cleaned and inspected. After the first end plug is welded into the tube and inspected, the tubes are stored until needed. After loading the sintered pellets into the tube (rod) to a specified stack height, the rod surface is decontaminated and a second plug is welded into place. The rods are then monitored and decontaminated as required, inspected, stored, and packaged for shipment. Facilities are available to repair or dismantle rods if necessary. Contaminated scrap metal and impure pellets are sent to dirty scrap for recovery or for waste packaging. Undamaged pellets are recycled through the rod loading area. Damaged pellets are recycled through clean scrap recovery.

2.2.2 Scrap Recovery

The total amount of oxide recycled as either clean or dirty scrap is estimated to be about 10% of the plant throughput. It is assumed that dirty scrap comprises one-sixth of the scrap and that clean scrap comprises the remainder. Selection of the type of process for recycling clean scrap is a choice of the individual fuel manufacturer (Figure IV D-4). In the model MOX plant, dirty scrap is assumed to be reclaimed by a wet process, and clean scrap by an oxidation-reduction process.

2.2.2.1 Dirty Scrap Recovery

Dirty scrap is composed of impure MOX from the process as well as MOX removed from metal scrap, liquid analytical wastes, cellulosics, plastics and rubber, and some organic wastes. Dirty scrap is processed through a series of roast, scrub, wash-leach, and filter systems. The procedure followed to reduce contamination levels in each of these scrap items is determined by the nature of the material itself. See the lower portion of Figure IV D-4 for the flowsheet of the dirty scrap recovery process.

Dirty MOX scrap is dissolved in a nitric-acid/fluoride solution. This, along with filtrate from the wash-leach processing, is adjusted for acidity and valence and sent through a Purex solvent-extraction process to recover mixed uranyl nitrate-plutonium nitrate solutions. This solution is then coprecipitated with ammonia and filtered. The resulting precipitates are dried, calcined, and reduced to $(U,Pu)O_2$, which, after a milling and screening process, is suitable for recycle.

Solvent extraction raffinate containing nitric acid is sent to the acid evaporator-fractionator system. Water from this fractionator is collected in the water recovery tank; the nitric acid is recycled to process operations. Ammonia vapors and filtrate are piped to the ammonia evaporator-fractionator system. Water is also recovered from this system. See Figure IV D-6 for the evaporator system flowsheet and paragraph 2.2.3.1 for a description of the liquid waste treatment system.

2.2.2.2 Clean Scrap Recovery

This process receives all mixed oxide, including grinder sludge, that is chemically uncontaminated, yet inadequate to meet specifications of size, shape, density, moisture content, etc. The material is sent through an oxidation-reduction process consisting of alternate exposure of the clean scrap to air, and then to hydrogen diluted with nitrogen to below the lower explosive limit. The clean scrap flowsheet is shown on the upper portion of Figure IV D-4.

2.2.3 Radwaste System

The radwaste system consists of liquid and solid waste treatment systems, with appropriate packaging and disposal methods as described below and in Figure IV D-6.

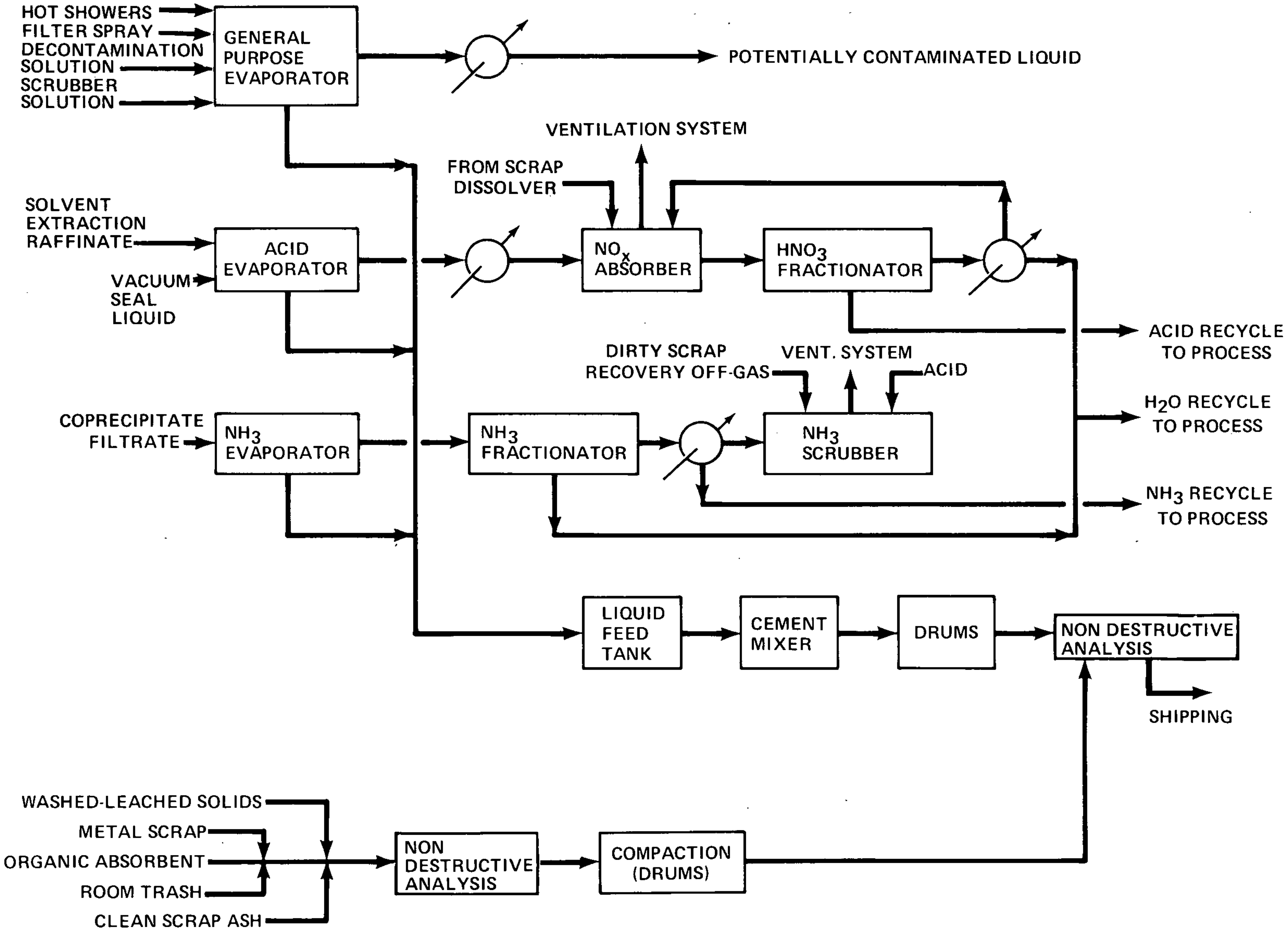

Figure IV D-6 Radwaste System

2.2.3.1 Liquid Waste Treatment

The liquid waste treatment system for the model plant accepts all discard streams that could be potentially contaminated from the main process, scrap recovery, and support areas, and provides for

- Evaporation of trace contaminated liquids to reduce volume
- Processing of liquid waste streams from wet scrap processes to recover and recycle chemical reagents and water
- Processing of off-gas from the wet scrap recycle processes to scrub out chemical pollutants
- Collection, sampling, and release of potentially contaminated liquids that meet 10 CFR 20 levels into the sanitary waste water system.

The design of the liquid waste treatment system is directed toward reducing the volume of waste sent to waste packaging and reducing the volatile chemical pollutants released from the plant.

The flow sheet for the liquid waste treatment system is shown in Figure IV D-6.

2.2.3.2 Solid Waste Treatment

The solid waste treatment system accepts discarded solid residue from the main process and scrap recovery areas and liquid and salt residues from the liquid waste treatment system.

Solid wastes are compacted to minimize volume.

The liquid waste residues are treated by a cementing operation, which results in immobilizing liquids in a concrete matrix. Technology used for preparing a "calcine" product from high level waste might also be used to process liquid waste to a solid.

2.2.3.3 Waste Packaging

The solid waste and solidified liquid waste are packaged in accordance with transportation regulations--normally in 55-gallon drums, and are assayed by a nondestructive method for accountability purposes and to ensure control of transuranium elements released to offsite shipment.

The process for the radwaste packaging system is shown in Figure IV D-6.

2.3 Plant Airborne Effluents

Small quantities of various airborne effluents are released from the mixed oxide manufacturing processes. The radioactive airborne effluents are plutonium and uranium isotopes, and their progeny.

2.3.1 Radioactive Airborne Effluents from the Model MOX Plant

Industrial experience indicates that a small fraction--approximately 10^{-9}--of the material that is processed through a plant is dispersed to the atmosphere via the filtered exhaust system. It is to be expected that future plants will release smaller fractions than this because of larger plant sizes and more stringent regulations under which they will be designed, fabricated, and operated.[9]

It is judged to be conservative to apply the 10^{-9} release fraction actually observed in the smaller plants to predict losses from the larger and newer plants considered typical in this generic assessment of the Pu recycle industry. On the basis of a 360-MT/yr plant with a release factor, the gaseous effluents are calculated to contain 360 mg of mixed oxide with 9 mCi of alpha activity per year. Applying an atmospheric diffusion factor of 5.4×10^{-6} sec/m^3 that has been estimated for this plant (see CHAPTER IV, Section J, Appendix A) results in a maximum alpha concentration of about 2.0×10^{-15} μCi/ml in the air at the plant boundary, which is 0.2% of the stated 10 CFR Part 20 limit for insoluble Pu in uncontrolled areas.

2.3.2 Nonradiological Process Effluents

Mixed oxide fuel fabrication operations may give rise to the release of non-radioactive gaseous effluents such as nitrogen, hydrogen, argon, helium, oxides of nitrogen (NO_x), hydrogen fluoride, and ammonia. Very small amounts of solvent vapors will result from hardware cleaning work outside the manufacturing building.

Nitrogen-hydrogen gas mixtures, below the explosive limit, are used to provide a reducing atmosphere both in pellet sintering furnaces and in reactors that are used for treating clean recycle scrap powder and pellets. Argon and helium are used in welding and in fuel rod leak-testing procedures. No atmospheric pollution problem will result from the use of these gases.

Chemicals used in scrap recycle and waste treatment operations, however, may be released in the vapor phase and are potential atmospheric pollutants. These are hydrogen fluoride (HF), oxides of nitrogen (NO_x), and ammonia (NH_3).

Hydrogen Fluoride

Recent studies show that mixed oxide powders or pellets that have been sintered at 1,650°C for an extended period dissolve readily in nitric acid with little selective dissolution of the uranium before plutonium. Since most of the scrap recycle requiring dissolution and purification will be generated in the unit operations after sintering, it is judged that the primary scrap dissolving work can be done with 12 molar nitric acid. An estimated 20%, or 5 kg of the 24 kg/day mixed oxide dissolved, requires either nitric acid-hydrofluoric acid or nitric acid-calcium fluoride treatment to accelerate dissolution of plutonium. Recent studies have shown that calcium fluoride (CaF_2) is an efficient source of the fluoride ion to accelerate the dissolution of plutonium, because its limited solubility provides a slow, continuous release of the fluoride ion.[10]

At the same time, CaF_2 may reduce the vaporization of HF compared to the direct use of hydrofluoric acid. Thus it is calculated that less than 1 g/day of HF vapor will be released from the fabrication plant on an annual average basis.

Nitrogen Oxides

Oxides of nitrogen will be generated in the scrap dissolution operation and in the ammonium diuranate-plutonium hydroxide calcination operation after coprecipitation. The amount of NO_x released has been calculated to be 1,730 g/day based on the following assumptions:

- Treatment of 24 kg/day recycle mixed oxide with nitric acid, followed by a second contact with nitric acid-calcium fluoride on a 5-kg/day residue
- The amount of NO_x released from nitric acid reacting with impurities such as iron, nickel, and chromium will be equal to that released from mixed oxide dissolution
- Off-gas scrubbers will remove 90% of the NO_x and all the HNO_3 vapor

Based on the scrap recovery system operating 250 days per year, the 1,730 g/day would result in an annual average release of 432 kg of NO_x. With an airflow of 45,000 cfm (Table IV D-6) and the atmospheric diffusion factor of 5.4×10^{-6}, NO_x concentration at the 500-meter site boundary would be <0.01 μg/cu meter compared to the EPA air quality standard of 100 μg/cu meter.[11]

Ammonia

Based on a dirty scrap recycle rate of 2%, the uranium-plutonium solution resulting from solvent extraction of the impure nitrate solutions described above will amount to about 200 liters/day. The purified solutions of uranium and plutonium will be coprecipitated with ammonia or ammonium hydroxide and filtered; the cake will be calcined in a reducing atmosphere to UO_2 and PuO_2 (mixed oxide) suitable for recycle to the powder blending operation. It has been calculated that the NH_3 releases from the fabrication facility will be 233 g/day for each of the 250 working days, or 160 g/day on an annual average basis. The model plant has an ammonia recovery and recycle system that is designed for 99% recovery, based on extensive industrial experience. Based on the 45,000-cfm facility ventilation exhaust, the NH_3 concentration discharged to the atmosphere would be about 0.1 part per million, far below the OSHA guideline of 50 parts per million.

2.3.3 Nonradiological Utility Effluents

The assumption has been made that heating, ventilating, and air-conditioning systems for the model MOX plant will be electrically operated. The amount of NO_x

released from the operation of a diesel powered emergency generator is considered to be insignificant, because it will be operated only a few hours a year.

2.4 Plant Liquid Effluents

Plant liquid effluents include both uncontaminated and potentially contaminated liquids. The latter includes laundry water, laboratory sink drainage, floor scrub liquids, regular shower water, fire water, and distillate from the general purpose evaporator. The uncontaminated liquid effluents are primarily sanitary waste water and cooling tower blowdown. Table IV D-8 summarizes the volume of liquid effluents released from the model MOX plant each day.

Table IV D-8

LIQUID EFFLUENTS FROM THE MODEL MOX FUEL FABRICATION PLANT

Stream	Amount (gal/day)
Potentially Contaminated Liquids	5,600
Sanitary	16,000
Cooling Tower Blowdown	12,000

2.4.1 Radioactive Liquid Effluents

Potentially contaminated liquids are kept in a holdup tank until analysis determines whether they must be sent to the radwaste system or disposed of as uncontaminated liquid waste. The decision level for transfer to the radwaste is chosen on a plant-by-plant basis, considering which route will result in the lower overall environmental impact. For this statement, the potentially contaminated liquid will be assumed to be disposed of as uncontaminated waste if its plutonium concentration is below the limits given in 10 CFR Part 20. Even though most of the time the potentially contaminated liquid will contain very much less than the limits set in 10 CFR Part 20.106(a), for this study it has been assumed that the annual average concentration is 7% of the limit of 5×10^{-6} μCi/ml given in 10 CFR Part 20, Appendix B, Table II, Column 2. It should be noted that the potentially contaminated liquid effluent stream is diluted by more than a factor of 5 by sanitary waste and cooling tower blowdown. Also, dilution by surface water will further reduce the liquid waste concentration.

The quantity of plutonium released to the environment is estimated to be 1.8 mCi(α)/yr, or about 3.1 mg Pu/yr. This is based on a release to the environment of 5,600 gal/day, 250 days/yr, of potentially contaminated liquid at 7% of the maximum permissible concentration.

2.4.2 Nonradiological Liquid Effluents

Table IV D-7 lists the chemical and sanitary liquid effluent flows for the model plant. Approximately 20 gal/m of sanitary and cooling tower blowdown are discharged together with 1/3 gal/m of such streams as uncontaminated laboratory waste and scrub water. Small amounts of phosphate (<0.5 lb/day) and nitrate (<10 lb/day) are present in the discharge stream.

2.5 Plant Thermal Wastes

Mixed oxide fuel fabrication is not an energy intensive process, and the radioactive heat generated by the plutonium feed material is not large (about .02 w/g). Essentially, all electrical energy used in the model MOX plant site is discharged as heat via the cooling tower(s), either as evaporative loss or in the blowdown. In addition, showers, sinks, and laundry wastes can contain heated water. It is estimated that the model MOX plant will release about 15,000,000 Btu/hr to the biosphere. This heat release is equivalent to the heat loss of about 75 to 100 private homes during winter months.

2.6 Plant Solid Radioactive Wastes

Solid waste generation has been estimated as approximately 10,000 cu ft per year containing 22 kg of PuO_2, or ∿ 0.12% by weight of the PuO_2 throughput. The waste consists of 710 cu ft from used HEPA filters, 1,900 cu ft from solidified liquid wastes, 5,000 cu ft from general process waste, and 2,500 cu ft from discarded major process components. Solid waste will be packaged in approved shipping containers for transfer to a Federal waste repository. Table IV D-9 summarizes the volumes and contents of solid wastes.

Table IV D-9

SUMMARY OF SOLID WASTE VOLUME AND PuO_2 CONTENT

Waste Stream	Approximate Volume Before Packaging (cu ft)	Number of Container Drums 55 gallon	Number of Container Drums 80 gallon	PuO_2 Content (kg)
HEPA Filters	710	30	162	13
Solidified Liquid Wastes	1,900	250	--	0.7
General Process Waste	5,000	685	--	7.8
Major Process Components	2,500	-	238	0.5
TOTALS	∿ 10,000	965	400	22.0

3.0 ENVIRONMENTAL IMPACT OF SITE PREPARATION AND PLANT CONSTRUCTION

The model MOX plant is assumed to be built on a large tract of land previously used for farming, ranching, or timbering. Construction requires onsite roads and erection of the plant and auxiliaries. Construction practices must meet local ordinances.

Local dewatering of the site may be required. Retention basins, if required for the dewatering activities, would be constructed and operated according to the guidelines of the United States Environmental Protection Agency and the USDA Soil Conservation Division.

The major effect of construction will be the removal of the plant site from any previous productive use, regrading and tree cutting on a relatively small part (about 5%) of the total site, possible disturbance of local wildlife, and possible local and temporary silting of streams. No part of the construction activity differs from that required to construct a comparably sized building for nonnuclear manufacturing. The large exclusion area and measures taken during construction to limit noise, dust, and surface water contamination tend to minimize the effect of construction on the offsite environment.

4.0 ENVIRONMENTAL IMPACT OF PLANT OPERATION

4.1 Land Use

As a condition of the plant license, access to the model MOX plant site will be barred except with permission and authorization of the owner. With a 500-meter distance from the plant to the nearest boundary, plant operations will not have a high impact on the surroundings. Plant operations are not expected to interfere with the use of adjacent land or streams.

4.2 Water Use

The fabrication of mixed oxide is not a water-intensive process. About 50 gpm of water will be provided for operation of the model MOX plant, the major use being cooling tower makeup and sanitary water.

Plant effluents must meet local, State, and Federal requirements, and a permit to discharge water must be obtained from the United States Environmental Protection Agency.

The plant cooling tower is sized to remove about 15,000,000 Btu/hr. Drift from cooling towers of this size will not extend more than 150 meters (500 feet) under normal conditions, and the plume (condensed water vapor) is not expected to extend beyond the site boundary, under most atmospheric conditions.

4.3 Radiological Impact of Plutonium in Effluents on Man in the Offsite Environment

Mixed oxide fuel fabrication operations involve uranium, plutonium, and their daughters. The alpha emitting plutonium isotope, ^{238}Pu and the beta emitting isotope ^{241}Pu are the main contributors to potential human exposures from the operations of the fuel fabrication plant. Although organisms other than man may be exposed to radiation from these effluents, discussion of exposure in this section is limited to that of man. For a detailed discussion of plutonium toxicity and behavior in the environment, see CHAPTER IV, Section J, Appendix C.

As stated in paragraph 2.3.1, a conservative estimate of the annual releases from the model mixed oxide fabrication plant is 360 mg of MOX containing an average of 9.0 mCi(α) of plutonium and uranium isotopes per year as airborne particulates and about 73 milligrams of MOX containing 3.1 milligrams of plutonium, an average of 1.8 mCi(α) to a surface water stream.

4.3.1 Potential Offsite Exposure from Airborne Plutonium

In evaluating potential exposure from radionuclides, two classes of exposure are considered: the potential annual dose commitment of the closest theoretical resident and the potential annual dose commitment, in person-rem, to the general population.*

Table IV D-10 gives the dose commitment from airborne effluents that the closest theoretical resident might receive and the dose commitment that the U.S. population could potentially receive from 1 year of plant operation. It can be seen that the potential bone dose commitment (bone is the critical organ) received from 1 year of exposure might be 171 millirem to the closest theoretical resident, and a total of about 197 person-rem to the bone of the U.S. population. In actual plants it is expected that doses will be maintained at levels below those shown in the table by a combination of minimization of plant effluents, location of effluent release points, location of plant boundaries, and other measures that are available to licensees in design of a specific plant for a specific site.

The tabulated dose commitments include irradiation by airborne radioactive material passing in the atmosphere, inhalation of radioactive material in the atmosphere from plant emissions prior to deposition, irradiation from radioactive material deposited in the habitat, radioactive material ingested as a consequence of obtaining a full diet from the contaminated habitat and inhalation of deposited radioactive material resuspended into the atmosphere.

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*See CHAPTER IV, Section J, for an evaluation of health effects from plutonium exposures of these levels, as well as the calculational methods and parameters used to derive these dose commitments.

Table IV D-10

AVERAGE CLOSEST THEORETICAL RESIDENT AND TOTAL U.S. POPULATION DOSE COMMITMENTS FOR AIRBORNE EFFLUENTS FROM A MODEL MOX PLANT (360 MT/yr)

Conservatively assumed to release
360 mg MOX/yr [9.0 mCi(α)] to the atmosphere

Organ*	Closest Theoretical Resident at 500m (millirem)	Total U.S. Population (person-rem)
Whole Body	3.7	4.2
Bone	171.	197.
G.I. Tract	0.15	0.23
Lung	3.9	4.4
Liver	17.7	19.
Kidney	15.2	18.
Skin	0.07	0.08

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*Plutonium and americium have been assumed to be insoluble in deriving the lung dose and soluble in deriving all other doses. These assumptions maximize doses to all organs.

4.3.2 Potential Offsite Exposure from Waterborne Plutonium

Table IV D-11 lists the dose commitment to the closest theoretical resident and the U.S. population from waterborne plutonium. Comparison of the waterborne dose commitment in Table IV D-11 with the airborne dose commitment in Table IV D-10 shows that waterborne effluents do not contribute significantly to either the closest theoretical resident or U.S. population exposures.

Table IV D-11

AVERAGE ANNUAL DOSE COMMITMENTS* TO THE CLOSEST THEORETICAL RESIDENT AND THE U.S. POPULATION FROM A MODEL MOX PLANT WATERBORNE EFFLUENT

Assumed to release
73 mg MOX, 3.1 mg/Pu/yr [1.8 mCi(α)] to a water course

Organ*	Closest Theoretical Resident (mrem)	U.S. Population (person-rem)
Whole Body	1.1×10^{-4}	0.006
Bone	4.4×10^{-3}	0.23
G.I. Tract	4.4×10^{-4}	0.021
Liver	5.2×10^{-4}	0.029
Kidney	5.2×10^{-4}	0.027

---

*Soluble forms of Pu and Am assumed to calculate dose commitments.

4.3.3 Total Population Exposure

Table IV D-12 presents the estimated total potential U.S. population dose commitment from gaseous and liquid effluents from 1 year of operation of a model mixed oxide fuel fabrication plant. This table makes it clear that the potential exposure from airborne plutonium is the dominant pathway for consideration.

Table IV D-12 shows potential annual dose commitments from the model plant calculated for the U.S. population to be 197 person-rem to the bone, 19.3 person-rem for liver, 17.5 person-rem for kidneys, 4.4 person-rem for lung, and 4.2 person-rem for whole body exposure. This compares with the 25,000,000 person-rem annual natural background irradiation experienced by the population of the United States.

The effects of radiation exposure on health are discussed in CHAPTER IV, Section J.

Table IV D-12

TOTAL ANNUAL DOSE COMMITMENTS* TO THE U.S. POPULATION FROM MODEL MOX PLANT, AIRBORNE AND WATERBORNE EFFLUENTS

Organ	Dose Commitment (person-rem) Airborne	Waterborne	Total
Whole body	4.2	0.006	4.2
Bone	197	0.23	197
Thyroid**	0.075		0.075
G.I. Tract	0.23	0.021	0.25
Lung	4.4		4.4
Liver	19.3	0.029	19.3
Kidney	17.5	0.027	17.5
Skin	0.08		0.08

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*See CHAPTER IV, Section J, Appendix A, for explanation of exposure modes and duration incorporated into the dose commitment determination.

**From criticality accident only.

4.3.4 Ground Contamination

A consideration with respect to land use both during plant operation and after plant shutdown is ground contamination from particulate airborne plutonium deposited on soil and plants. To calculate surface contamination accruing from this deposition, which is a function of the concentration time integral in the atmosphere, Ψ, and the deposition velocity V_g, the following expression is used:[12]

areal concentration, $\mathrm{Ci/m^2} = \Psi V_g$

where Ψ is the concentration time integral*, $\mathrm{Ci\text{-}sec/m^3}$, and

V_g is the deposition velocity, m/sec.

For the model plant and the atmospheric diffusion characteristics associated with it, it has been calculated that the maximum annual deposition of plutonium would occur on the plant property 100 meters from the release point and would be on the order of 6×10^{-9} $\mathrm{Ci/m^2}$ per year.

The maximum deposition 500 meters from the plant would be on the order of 5×10^{-10} $\mathrm{Ci/m^2}$ per year. Assuming the annual release of plutonium to be 15 mg, the maximum offsite accumulation for 20 years of plant operation could exceed the fallout that has already accumulated in the environment within 1,000 meters of the plant by about a factor of 3. Assuming no downward migration of plutonium in the soil, a 20-year surface accumulation on the plant site or in the offsite environment would amount to less than the 5×10^{-7} $\mathrm{Ci/m^2}$ in the top millimeter of soil proposed as the standard for ground contamination.[13] Thus it is expected that plant sites used for mixed oxide fuel production can be returned to unrestricted use after plant shutdown. A comprehensive survey of plutonium contamination and removal of any deposited radioactive material would be required for each specific case prior to facility license termination. At the present time the NRC has no standard for acceptable ground contamination.

4.4 Occupational Exposure

Radioactive materials will be present in certain plant areas and will lead to both internal and external exposure for individuals working in these areas.

Estimates have been made of the occupational exposures that might be expected in the model fuel fabrication facility, based on the design limit radiation levels in plants of similar design and projected model plant radiation zones.[14] The estimated total penetrating (gamma and neutron) radiation dose to all workers employed in a model plant is 350 person-rem per year for a plant with 300 employees. This is an average of 1.20 rem per employee per year, about 23% of the annual individual occupational dose permitted by 10 CFR Part 20 regulations, or about 12 times the average annual background dose received by each individual in the general U.S. population.

Internal exposures add a relatively small amount to the occupational dose. The dose commitment to the lung or any organ that results from 1 year's exposure is estimated to average 0.19 rem per employee for the 300 employees, or a total of 57 person-rem per year. This is about 1% of the current annual individual occupational internal dose limit to organs of 15 rem/yr recommended by the ICRP.

4.5 Environmental Impact from Nonradiological Effluents

The model mixed oxide fuel fabrication plant releases, on an annual average daily basis, are comprised of about 1.0 g/day HF, 1200 g/day NO_x, and 160 g/day NH_3. The estimated atmospheric diffusion factor at the plant boundary (500 meters from the release point) is 5.4×10^6 sec/cu meter. Applying this diffusion factor to the chemical releases results in average plant boundary concentrations of

Effluent	Concentration (μg/cu m)
HF	6.0×10^{-5}
NO_x	7.4×10^{-2}
NH_3	1.0×10^{-2}

The offsite HF concentration is about 0.012% of the most restrictive State standard of 0.5 μg/cu meter;[15] NO_x is 0.074% of the EPA standard of 100 μg/cu meter, and NH_3 is about one-millionth of the occupational limit established by the American Conference of Government Industrial Hygienists.[11]

No significant effects on man or the environment are expected to accrue from these effluents.

5.0 ACCIDENTS

5.1 Systems Important to Safety

Mixed oxide fuel fabrication plants are required by 10 CFR 70.22(f) to be designed, fabricated, constructed, tested, and operated under rigid quality assurance programs. Quality assurance comprises all those planned and systematic actions necessary to provide adequate confidence that structures, systems, components, and operations will perform satisfactorily in service (10 CFR Part 50, Appendix B).

All operations at MOX fuel fabrication plants that involve handling plutonium, except when it is contained in shipping containers or sealed fuel rods, are carried out within shielded process enclosures such as cells or gloveboxes. These enclosures provide confinement of plutonium in the event of equipment failure. The process building, essential equipment, and supporting systems are designed to withstand impacts due to natural phenomena related to tornadoes, earthquakes, and floods.

During the life of a mixed oxide fuel fabrication plant, some equipment (or accessory) failures may occur. Monitors are installed to detect such failure or process upset conditions that can cause safety related damage. Some corrective action is automatically provided. The ventilation system is designed to function during and after accidents, and to pass all plant ventilation air through high efficiency particulate air (HEPA) filters before it is released to the atmosphere. In summary, a model mixed oxide fuel fabrication plant is

- Designed, fabricated, constructed, tested, and operated according to a quality assurance plan

- Designed to cope with accidents

- Designed to minimize the offsite consequences of potential accidents

5.2 Upper Limit Accidents

A wide spectrum of credible accidents for these plants has been analyzed and their potential consequences estimated.

Some incidents, such as punctures or tears of gloves or other glovebox malfunctions, are expected to occur as part of the normal operation. Other more serious accidents--such as glovebox window breakage--will occur far less often, although the offsite consequences from such accidents are judged to be insignificant, they are included in the estimate of airborne effluents resulting from normal operations. Upper limit accidents that may occur include a criticality incident, an explosion, or a fire. These upper limit accidents are analyzed for offsite environmental effect potential.

5.3 Criticality

Nuclear criticality safety is a major consideration in plant design and NRC licensing review. Plant equipment is designed and arranged to preclude accidental criticality, unless several independent failures occur simultaneously.

For operations under administrative control (that is, where adherence to specific operating procedures is necessary to preclude criticality), the NRC requires that two independent levels of review be carried out by technically qualified personnel prior to start of operations. All personnel involved in the operations receive instructions in the specific procedures and periodic training sessions in criticality safety. No changes are permitted in the approved procedures without the two-level review and approval.

There have been no criticality accidents to date in process operations where undermoderation is a primary method of control, and few in aqueous or moderated systems. The number of fissions has been estimated to be 10^{18} in an accidental criticality. In calculating the effects, it is assumed that all of the noble gases krypton and xenon and 25% of the iodine formed by the fissions would escape. In addition, it is assumed that 500 grams of Pu would become airborne in a glovebox by the accidental criticality excursion. The ventilation filters are assumed to remain intact because a criticality is not an explosive process. The decontamination factor of three HEPA filters in series has been assumed to be 2×10^7, lower by a factor of 50 than the decontamination factor assumed for normal operating conditions.[15] Calculations show that the maximum offsite individual dose commitment results from absorption of fission product iodine in the thyroid and amounts to 360 mrem. This is comparable to the dose to the closest theoretical resident from a criticality accident at a UO_2 fabrication plant. The slightly different fission product yield and the presence of small amounts of plutonium particulate do not significantly alter the effects of a PuO_2 criticality accident relative to those of a UO_2 criticality accident.

5.4 Fire

Unlike a criticality excursion or an explosion, a fire usually is not an instantaneous event and very often starts from a small flame source. The design, construction, and operation of fuel fabrication plants considers in detail the possibility of a fire and equipment and procedures for fire prevention. Regulatory Guide 3.16[17] presents methods acceptable to the NRC for a fire protection program that should prevent, detect, extinguish, limit, or control fires and explosions and their hazards and damaging effects. Licensees must operate within these or acceptable equivalent constraints. Under these conditions, the probability of a fire of the magnitude considered in this statement is considered highly unlikely. In general, operators have time to react to and extinguish small fires. The process materials, oxides of uranium and plutonium, are not themselves flammable. The final filters are protected against fire by water spray systems installed in the duct some distance upstream of the final filters. Mist deflectors or collectors are installed between the water spray system and the filters to remove large drops of water. The water from the sprays collects in the bottom of the ducts and flows to a fire-water collection tank. This tank is either a safe-geometry tank or a fixed-poison-controlled tank to preclude the possibility of a criticality accident as the result of a fire.

The final HEPA filters are located some distance from the gloveboxes. This separation distance and the water spray system should be sufficient to protect the filters against the effects of an explosion or fire, but the fire or explosion is assumed to destroy the local filters on the gloveboxes. Plutonium and uranium oxides reach the final filters. Based on an assumed room volume of 1,000 cu m and an air loading of 100 mg/cu m for plutonium oxide particulate,[7,16,17] 100 grams of plutonium would reach the filters. Each of the filters is expected to remove 99.9% of the particulate reaching it, so that a total of 0.1 mg of Pu passes through the filters.

The probability of a major fire in a modern plutonium facility is small. However, in the unlikely event of a fire, the calculated total additional dose commitment from the accident would be approximately 10% of the annual dose commitment estimated to accrue from normal operations.

5.5 Explosion

An explosion might occur in a mixed oxide fuel fabrication plant at locations where an explosive mixture of vapors in air could be present. There is a potential for the existence of combustible gases at the sintering furnace and in the clean scrap reduction operation. In addition, flammable solvents are used in the dirty scrap recycle operation and may be used for cleaning fuel rods and during cleanup and maintenance operations. These operations are essentially the only ones that have a potential for supporting an explosion.

To minimize potential explosive mixtures, the sintering furnaces and the clean scrap reduction operations use hydrogen diluted with an inert gas so that the mixture is below the explosive limit. The hydrogen-inert gas mixture is produced outside the production building proper by means of a gas mixing system designed to control the gas introduced into the production building to a mixture containing less than 6.5 percent hydrogen. Combinations of this mixture of inert gas and air cannot form explosive mixtures. Therefore a hydrogen explosion in this plant is considered highly unlikely and has not been assessed in detail for this study.[19]

The dirty scrap process uses a commercial organic solvent, generally kerosene, that is widely used and is not considered to present a major hazard of explosion. However, the possibility exists that more flammable solvents might be used in cleanup or maintenance operations. The quantity of such solvents will be kept to a minimum and their use controlled. Thus the possibility of a flammable solvent explosion is unlikely but for conservatism it has been considered in this study.

The consequences of an explosion are similar to those of a fire. The amount of plutonium reaching or passing through the filters is estimated to be the same as that estimated for the fire (see above) and would have the same relatively minor offsite consequences.

5.6 Accident Summary

Table IV D-13 summarizes the offsite effects of the upper limit accidents. Although the plutonium dioxide powder, which forms the major part of the inventory of the MOX fuel fabrication plant, is insoluble, the potential dose commitments for the bone are based on soluble plutonium.

Table IV D-13

SUMMARY OF OFFSITE IMPACTS FROM MODEL MOX PLANT ACCIDENTS

Type of Accident	Pu Release [μCi(α)]	Closest Theoretical Resident Dose Commitment (mrem)	U.S. Population Dose Commitment (person-rem)
Criticality	0.29	360 (thyroid)	4.2 (thyroid)
Fire	53	21.5 (bone)	0.7 (bone)
Explosion	53	21.5 (bone)	0.7 (bone)

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*The meteorology used in this assessment results in an atmospheric diffusion factor (Ψ/Q value) of 1.3×10^{-4} at 500 meters (Pasquill Stability Category D, wind speed = 3 m/sec, 10 meter release point). Plutonium has been assumed to be soluble.

It is judged that upper limit accidents will occur very infrequently. For purposes of this impact assessment, it has been assumed that an upper limit accident of each type analyzed will occur annually in the MOX fuel fabrication industry. This is a conservative assumption that significantly overstates the actual impact expected from upper limit accidents over the life of these facilities.

6.0 ENVIRONMENTAL IMPACT OF MOX INDUSTRY FROM 1981-2000

Table IV D-1 summarizes the environmental impact of the mixed oxide fuel fabrication industry from the year 1981 to the year 2000. The total industry impact except for land requirements is based on the number of 360 MT plant years required to produce 25,000 MT of MOX fuel, in 20 years or 71 plant years. The land requirements are based on the number of plant years each plant would be present at a site in order to provide a smooth growth curve without increasing the production of operating plants above 360 MT/yr and that number has been calculated to be 78 plant years.

7.0 CONTRIBUTIONS TO THE GENERAL EXPOSURE

The mixed oxide fuel fabrication industry will release small amounts of plutonium, americium, and uranium. The total industry release of plutonium and americium from 1981 through 2000 is estimated to be about 770 mCi(α). The total 1981-2000 dose commitment to the bone, the limiting organ, of the U.S. population is calculated to be 14,000 person-rem. The dose commitment to the population of the U.S. from MOX fuel fabrication plant operation for 1975-2000 is about 1×10^{-3} of the total dose commitment to the U.S. population from the LWR industry during this period.

Maximum ground deposition of α-emitters released from the normal operation of fuel fabrication plants over a 20-year period would occur within the facility site boundaries and is expected to be about equal to that already accumulated from worldwide weapons testing fallout (1 to 3 mCi/km^2). A deposition of this magnitude would be difficult to detect because of the variation of weapons testing fallout in the soil.

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GESMO
CHAPTER IV
ENVIRONMENTAL IMPACT DUE TO THE
IMPLEMENTATION OF
PLUTONIUM RECYCLE

SECTION E
REPROCESSING PLANT OPERATIONS

CHAPTER IV

Section E

REPROCESSING PLANT OPERATIONS

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Section E
REPROCESSING PLANT OPERATIONS

SUMMARY

General

Irradiated spent fuel can be reprocessed to recover uranium and plutonium that contain usable fissile materials. With no recycle of uranium and plutonium there would be no fuel reprocessing industry. The spent fuel elements themselves would constitute high level radioactive waste. See CHAPTER IV, Section H. In this respect, the fuel reprocessing industry is a contractual service that is predicated on the net benefits related to the conservation of natural resources and, thus, is contingent upon the recycle of uranium (U) and plutonium (Pu) as fuel in light water reactors (LWR's). Environmental effects from reprocessing are partially offset by a reduction of effects associated with decreased mining, milling, and enrichment requirements if U and Pu are recycled as fuel in LWR's. See CHAPTER XI, Section 4.0.

This section assesses the environmental effects which may occur if spent fuels are reprocessed to recover only U for recycle as UO_2 fuels in LWR's or to recover U and Pu for recycle as UO_2 and MOX fuels in LWR's.

The Industry Today

Three commercial nuclear fuel separations facilities have been built in the United States, but none of these plants is currently reprocessing spent fuel. If a decision is made to permit spent fuel to be reprocessed to recover and recycle U and Pu in LWR fuels, Allied-General Nuclear Services could complete the necessary additional facilities and provide commercial spent fuel reprocessing services within about 3 to 4 years after such a decision. General Electric has elected not to operate the Midwest Fuel Recovery Plant, now called the Morris Operation, because of technical difficulties uncovered during preoperational tests. This facility is now being used to store spent fuel. Nuclear Fuel Services, Inc. is moving toward a decision whether to continue or terminate their West Valley operation. This facility is now being used to store spent fuel. They estimate that it would take about 10 years to modify existing facilities and add the facilities necessary to continue commercial reprocessing services at West Valley.

Capital costs for a complex of facilities for reprocessing 1,500 MT/yr of heavy metals in spent fuels, such as may be provided by Allied-General Nuclear Services at the Barnwell Nuclear Fuel Plant (BNFP), are projected to be on the order of $600 million in 1976 dollars, excluding cost escalation.[1] The fuel reprocessing plant complex will consist of the following:

- Fuel receiving and storage facility
- Separations facility
- UF_6 conversion facility
- PuO_2 conversion facility
- Waste solidification and packaging facilities

The Future Industry

The current projection of the amounts of spent fuel expected to be discharged from LWR's and the expected growth of the spent fuel reprocessing industry if these fuels are reprocessed are shown in Table IV E-5. By the year 1980, the backlog of spent fuels available for reprocessing is expected to total more than 4,000 metric tons. The projected spent fuel load would require the equivalent of five model fuel reprocessing plants by the year 2000. It is estimated that the next new reprocessing plant could be designed and constructed within 8 to 10 years; thus might begin operation in about the mid-1980's. According to the current projection, if Pu is recycled, during the period through the year 2000 mixed oxide fuel could make up approximately 11% of the total LWR fuel available for reprocessing; the remaining 89% would be UO_2 fuel. Over this period, the recovery and recycle of U and Pu as fuel in LWR's would reduce virgin uranium requirements by approximately 22% and also would reduce uranium enrichment requirements by about 14%. Most of the projected spent fuel load would be reprocessed in plants not yet designed.

Effects of Reprocessing Mixed Oxide Fuel

Irradiated mixed oxide (MOX) fuel differs from irradiated UO_2 fuel in fission product content, transuranium element content, and decay heat generation. The increased plutonium content in MOX can be accommodated by appropriate flowsheet adjustments which would have only a slight effect on reprocessing plant capacity or operating cost. If the mixed oxide fuel pellets have a high degree of homogeneity and are sintered at about 1,650°C for some 4 hours, there should be little difference between the dissolvability of irradiated UO_2 fuel and irradiated MOX fuel. However, the solidified high level radioactive waste from reprocessing an overall average fuel mix of 11% MOX and 89% UO_2 fuel would have about 25% more decay 10 ten years after fuel irradiation, because of the increased ^{244}Cm content, than an equivalent amount of similarly aged solidified high level radioactive waste from UO_2 fuel. This increase in decay heat could cause a slight increase, about 1%, in the cost for disposal of high level waste at the Federal repository because of additional space requirements.

The Model Fuel Reprocessing Plant

Future plants are likely to have larger capacities than those now in place. Existing reprocessing plants and proposed plants have design capacities which range from 300 to 2,100 MT/yr. The design capacity of later reprocessing plants could exceed 3,000 MT/yr by year 2000. For this evaluation the model reprocessing plant is taken to be similar to current designs, with a design throughput capacity of 2,000 metric tons (MT) of heavy metal, U and Pu, per year. The reprocessing plant complex is defined in Figure IV E-1, in Section E-1.0. It is assumed that the quantities of radioactive effluents will vary directly with throughput and the related annual average feed mix reprocessed. Thus, the prospective spent fuel reprocessing industry is defined in terms of equivalent 2,000 MT/yr model fuel reprocessing plants, and the environmental assessment is based on the spent fuel load and the equivalent number of model fuel reprocessing plants required to reprocess the spent fuel load in any given year. For example, if U only or both U and Pu are recycled by the year 2000, five 2,000 MT/yr model fuel reprocessing plants would be required to reprocess approximately 10,000 MT of heavy metal as it becomes available in spent fuel discharged by LWR's.

Environmental Effects

Radioactive effluents from the prospective spent fuel reprocessing industry are summarized in Table IV E-1. Compared to recycle of U only, Pu recycle adds to the amounts of ^{3}H, ^{106}Ru, ^{129}I, and transuranium elements, and decreases the amounts of ^{14}C, ^{85}Kr, and ^{90}Sr that would remain in the environment at the end of year 2000 from reprocessing. These effluents have been conservatively estimated based on reprocessing a total fuel load composed of UO_2 fuel or 11% MOX + 89% UO_2 fuel, aged 160 days, with the release to the atmosphere of 100% of the ^{3}H, ^{14}C, and ^{85}Kr present in the irradiated fuel. Table IV E-19, in Section E-5.0, shows the cumulative environmental impact of the prospective U.S. spent fuel reprocessing industry, which could occur if the spent fuels are reprocessed, over the period from the year 1975 through year 2000. In general, the nonradiological environmental effects of reprocessing operations do not change with Pu recycle, except for an increase in the decay heat of the inventory of stored high level radioactive waste and a related slight increase in cooling water requirements for reprocessing MOX fuels. The bulk of the chemical effluents are related to the burning of fossil fuels associated with the process heat used in the reprocessing operations. Only about 10% of the land used would be permanently restricted for future use. The thermal discharges and uses of land, water, and electricity by the prospective reprocessing industry are less than 0.5% of the totals for the entire fuel cycle industry.

Radiological Impact

The prospective radiological impact on the population expected if spent fuels are reprocessed is shown in Table IV E-2. Over a 26-year period, the total body dose commitment to the world's population (5 billion people) would be increased by about 1.8 million person-rem, about 0.02% of that received from natural radiation. Spent fuel reprocessing would contribute about 20% of the 9 million person-rem total body dose commitment to the world's population from the total LWR fuel cycle industry in the United States.* Tables IV E-12 and 13 in Section E-3.0 show the dose commitment

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*Health risks from radiation are discussed in CHAPTER IV, Section J, Impact of Radiation on Populations.

Table IV E-1

REPROCESSING INDUSTRY RADIOACTIVE EFFLUENTS

Basis: The projected U.S. fuel reprocessing industry, years 1975 through 2000.

- Total fuel mix is 11% MOX + 89% UO_2 fuel with Pu recycle.*
- Average fuel exposure is 33,000 MWd/MT.
- Radionuclides present in the environment at the end of year 2000.
- 100% of ^{3}H, ^{14}C, and ^{85}Kr is released to the atmosphere.**

Curies Remaining in the Environment at the End of Year 2000 by Source

Radionuclide	UO_2 Fuel	UO_2 + MOX Fuel
^{3}H	42,000,000	42,600,000
^{14}C	78,300	75,000
^{85}Kr	876,000,000	841,000,000
^{129}I	110	114
Other Fission Products	156	163
Pu, Am, & Cm	62	83

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*Without U or Pu recycle, there is no requirement for reprocessing spent fuel.

**No credit taken for prospective retention of these radionuclides.

to the nearest exposed individual living in the vicinity of a 2,000 MT/yr fuel reprocessing plant which reprocesses UO_2 fuel only, or reprocesses 11% MOX + 89% UO_2 fuels. Introduction of Pu recycle may decrease population dose commitments slightly because of decreased amounts of ^{14}C and ^{85}Kr associated with irradiated MOX fuel. However, the dose commitment to the nearest exposed resident may increase about 1% to total body, from 7.5 to 7.55 mrem/yr, because of increased amounts of tritium, and may increase about 11% to bone, from 23.7 to 26.4 mrem/yr, resulting for the most part from increases in the amounts of transuranic radionuclides that may be released to the environment if Pu recycle fuel is reprocessed. With respect to ground contamination, it is estimated that the maximum local accumulation of alpha activity in soil resulting from deposition of transuranic particulates over 40 years would increase because of Pu recycle about 60% more than from reprocessing UO_2 fuels. This maximum local concentration in soil expected to result from reprocessing Pu recycle fuel would be about equal to the concentration that has accumulated to date from fallout. The average annual occupational exposure associated with the overall operation of a 2,000 MT/yr spent fuel reprocessing plant complex is estimated to be approximately 1,250 person-rem. In general, Pu recycle may result in increased exposure (about 100 person-rem) to those operating personnel in the PuO_2 conversion facility whose activities involve physical handling of increased quantities of plutonium.

Table IV E-2

EFFECT OF PLUTONIUM RECYCLE ON POPULATION DOSE COMMITMENT FROM FUEL REPROCESSING PLANTS

Basis: The projected U.S. fuel reprocessing industry, years 1975 through 2000.

- Total fuel mix is 11% MOX + 89% UO_2 fuel with Pu recycle.
- Average fuel exposure is 33,000 MWd/MT.
- Fuel has aged 160 days from reactor discharge.
- 100% of ^{3}H, ^{14}C, and ^{85}Kr is released to the atmosphere.*

U.S. Population Dose Commitment in Thousands of Person-Rem

	^{3}H	^{14}C	^{85}Kr	^{129}I, ^{131}I	Other Fission Products	U, Pu, Am, Cm	Total
U.S. Population							
Total Body							
UO_2 Fuel	638	365	66	3	0	1	1,080
UO_2 + MOX Fuel	645	350	64	3	10	2	1,070
Bone							
UO_2 Fuel	638	1,820	66	2	62	65	2,650
UO_2 + MOX Fuel	645	1,750	64	2	61	92	2,610
Thyroid							
UO_2 Fuel	638	365	66	852	6	--	1,930
UO_2 + MOX Fuel	645	350	64	869	6	--	1,930
Lung							
UO_2 Fuel	638	365	144	2	8	2	1,160
UO_2 + MOX Fuel	645	350	138	2	8	4	1,150
Skin							
UO_2 Fuel,	638	365	5,630	2	5	--	6,640
UO_2 + MOX Fuel	645	350	5,390	2	5	--	6,390
Occupational Exposure							
UO_2 Fuel							72
UO_2 + MOX Fuel							78
Foreign Population (Total Body)							
UO_2 Fuel							703
UO_2 + MOX Fuel							675
Natural Radiation**							
U.S. Population							600,000
Foreign Population							11,400,000

Note: Population dose commitments expected from each segment of the U.S. total LWR fuel cycle industry, and for the three fuel cycle options, are compared in CHAPTER IV, Section J, Appendix E, Tables J(E)-1 through -8.

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*No credit taken for prospective retention of these radionuclides.

**Stated for comparison.

Accidents

Plutonium recycle may slightly affect the offsite consequences in the event of an accident, because of the change in transuranic radionuclide concentrations associated with reprocessing MOX fuel. The maximum potential offsite exposure in the event of an accident exists during reprocessing of a fuel lot made up entirely of PWR MOX fuel elements. Should an upper-level* accident occur in a plant during reprocessing of PWR MOX fuel elements, it is estimated that the maximally exposed offsite individual might receive a dose commitment to thyroid of up to about 56 mrem, or a dose commitment to bone of up to about 19 mrem. This maximum bone dose commitment to an individual from a reprocessing plant accident is comparable to about 19% of the annual dose from natural radiation.

Environmental Impact, Years 1975 through 2000

The cumulative impact related to the prospective growth of a U.S. fuel reprocessing industry, expected if spent fuels are reprocessed, is summarized in Table IV E-19 in Section E-5.0. In general, Pu recycle does not result in any significant change in the environmental impact associated with spent fuel reprocessing.

Population dose commitments expected from each segment of the U.S. total LWR fuel cycle industry, for the three fuel cycle options, are summarized in CHAPTER IV, Section J, Appendix E. Over the years 1975 through 2000, if spent fuel is reprocessed to recover and recycle uranium only, the total body dose commitment to the world's population from the U.S. total LWR fuel cycle industry is expected to increase about 16%, when compared to the no recycle of U or Pu option. On the same basis, if spent fuel is reprocessed to recover and recycle U and Pu, the total body dose commitment to the world's population is expected to increase about 9%. About 20% of the total body dose commitment to the world's population that is expected to result from the total LWR fuel cycle industry in the United States would be attributable to reprocessing, if spent fuels were reprocessed to recover and recycle U only or both U and Pu. For comparison, the LWR's would cause about 30% of the world's population total body dose commitment that would be attributable to the total LWR fuel cycle industry in the United States, if U only or both U and Pu are recycled as fuel in LWR's.

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*The more serious accidents of the types that have occurred or realistically can be postulated.

1.0 THE IRRADIATED FUEL REPROCESSING INDUSTRY

Irradiated fuel can be reprocessed to recover usable fissile material (both U and Pu) from spent reactor fuel elements.* The buildup of fission products and the depletion of fissile material requires that the reactor operator annually replace between one-fourth and one-third of the fuel elements and redistribute partially spent fuel remaining in the reactor core. The fuel elements removed from the reactor still contain about one-fourth of the original ^{235}U plus plutonium produced from ^{238}U during irradiation and, in the case of plutonium recycle, a portion of the original plutonium content. Fuel reprocessors recover the uranium and plutonium products for the account of the reactor operator. Additionally, pursuant to Federal regulations, reprocessors are required to convert the radioactive wastes to forms acceptable for transfer to Federal repositories. The recovered uranium is sent to a plant for reenrichment and eventual recycle to reactors in replacement fuel elements. The recovered plutonium can be converted to plutonium dioxide and then blended with UO_2 for recycle in replacement fuel elements. Capital costs for a complex of facilities for reprocessing nuclear reactor fuels, such as may be provided by Allied-General Nuclear Services at the Barnwell Nuclear Fuel Plant, are projected to be on the order of $600 million in 1976 dollars, excluding cost escalation.[1] For additional information related to cost estimates for reprocessing services, see CHAPTER XI.

Figure IV E-1 shows, as a simplified flow diagram, the major operations performed at the spent fuel reprocessing plant complex. In general, the fission products are separated from the uranium and plutonium, and the uranium and plutonium are recovered as purified nitrate solutions in the separation facility. Radioactive wastes must be converted to forms acceptable for transfer to Federal waste repositories, which will be provided for the long term confinement of various types of radioactive wastes. Plutonium nitrate will be converted in the plutonium conversion facility to a solid form, such as plutonium dioxide (PuO_2), which may be blended with UO_2 for recycle as LWR mixed oxide fuel elements, stored for future use, or shipped as waste to a Federal repository. Uranium nitrate will be converted to uranium hexafluoride (UF_6), in the UF_6 conversion facility, for return to the fuel cycle in the chemical form for feed to the enrichment plants.

Spent fuel assemblies are received at the reprocessing plant via truck or rail in heavily shielded shipping casks. See Transportation of Radioactive Materials, CHAPTER IV, Section G. The fuel assemblies are spaced "arrays" of sealed tubular rods containing UO_2 or mixed oxide fuel pellets. Tubular rods encapsulating the fuel are made of Zircaloy, although stainless steel tubing also has been used. During irradiation, the uranium is partially transformed into fission products (some of which are gases) and transuranium elements.

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*During the years 1975 through 2000, recovery and recycle of U and Pu as fuel in LWR's would reduce virgin uranium requirements by approximately 22% and would reduce uranium enrichment requirements by approximately 14%.

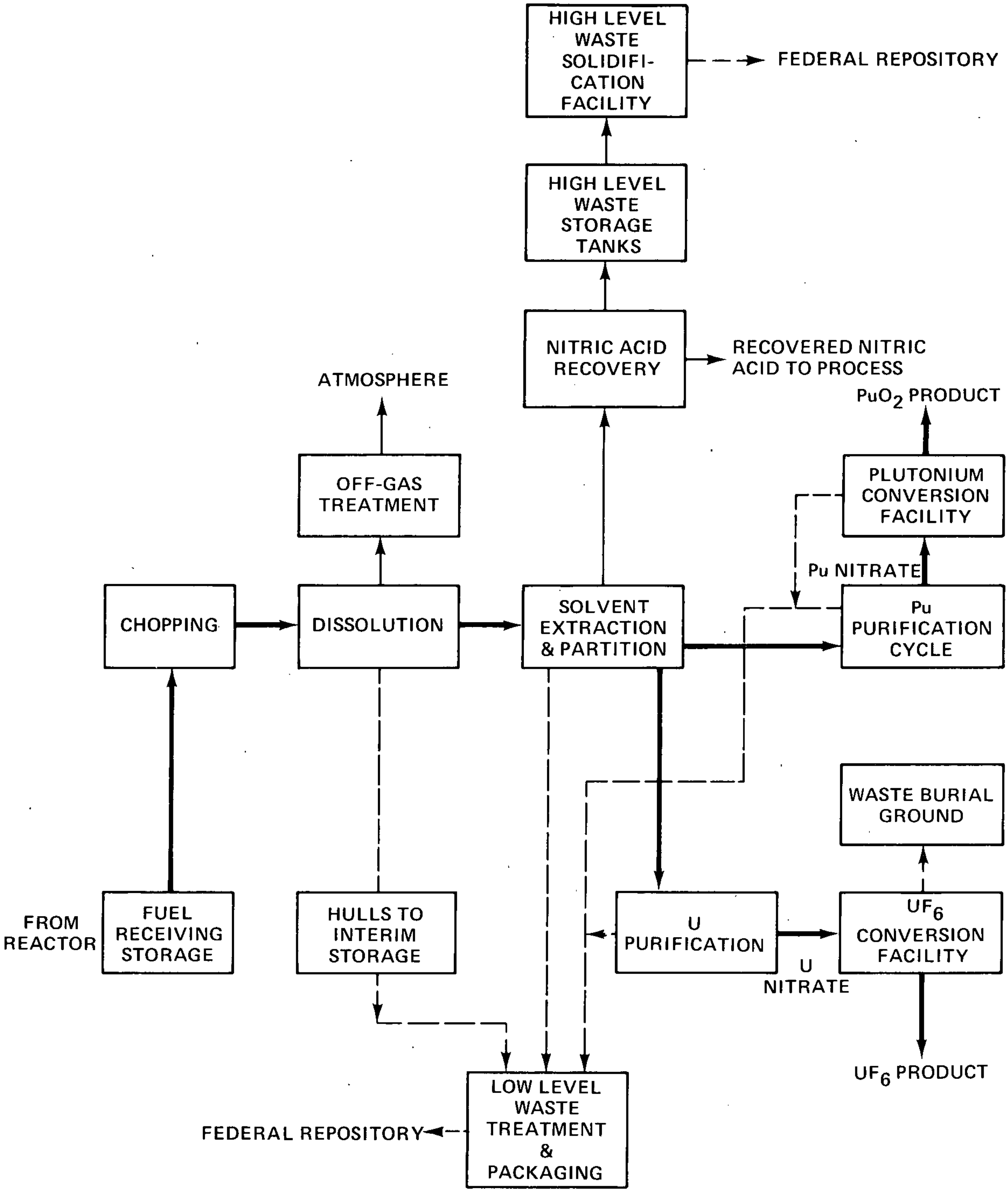

Figure IV E-1 Simplified Flow Diagram of Reprocessing Plant Complex

Fuel elements are stored at least 150 days after removal from the reactor to permit the short half-life radionuclides to decay away before the fuel elements are cut into short lengths to expose fuel material for dissolution. In general, all planned commercial plants utilize mechanical means for breaking up the fuel elements. Chemical systems for recovering special nuclear material are adaptations of the Purex process, for which the technology and risks are well defined.[2] However, specific methods for mechanical disassembly of the fuel elements, dissolution of fuel material, and recovery of uranium and plutonium differ somewhat from plant to plant.

Exposed fuel material is dissolved by hot nitric acid, leaving behind the chopped tubing (hulls). The nitric acid concentration is adjusted and then the solution, which now contains the dissolved fuel material, is processed through a series of solvent extraction or ion exchange systems. Solvent extraction which separates the fission products and transuranium elements from uranium and plutonium and then separates and purifies uranium and plutonium, is carried out in a centrifugal contactor, or a series of pulse columns, or both. Ion exchange, an alternative method for purifying plutonium, is carried out in equipment containing beds of ion exchange resin.[3] To ensure that essentially all of the uranium, transuranium elements, and fission products have been removed, the hulls are again soaked in hot nitric acid, rinsed in water, and monitored.

The purified uranyl nitrate is converted to uranium hexafluoride (UF_6), which is the chemical form of uranium feed for reenrichment in the uranium enrichment plants. The process for converting uranyl nitrate to UF_6 is shown in Figure IV E-2.[4]

It is planned to convert the purified plutonium nitrate product solution ($Pu(NO_3)_4$) to solid plutonium dioxide (PuO_2). Pursuant to 10 CFR Part 71.42, plutonium shipments after June 17, 1978, must be in the form of a doubly confined solid.[5] This also would be the preferred form for storing large inventories of recovered plutonium. Accordingly, reprocessors plan to provide facilities for converting the plutonium nitrate product solution to a solid form, such as plutonium dioxide. A process for converting plutonium nitrate to plutonium dioxide is shown in Figure IV E-3.[6]

The high level radioactive waste contains most of the fission products and transuranium elements that were separated from the uranium and plutonium products. These wastes are confined initially in high integrity tanks within vaults and then are converted to encapsulated immobile solids. Pursuant to Appendix F of 10 CFR Part 50, high level radioactive liquid wastes must be converted to solids within 5 years after generation. Within 10 years after generation these solid wastes must be transferred to a Federal repository for storage. For additional information relevant to the long term confinement of radioactive wastes, see CHAPTER IV, Section H.

In addition to the high level radioactive wastes, a variety of other contaminated wastes results from operations of the separation facility, the UF_6 conversion facility, the PuO_2 conversion facility, and waste handling. Most of these other wastes may be contaminated by long life fission products, or transuranic radionuclides, or both.

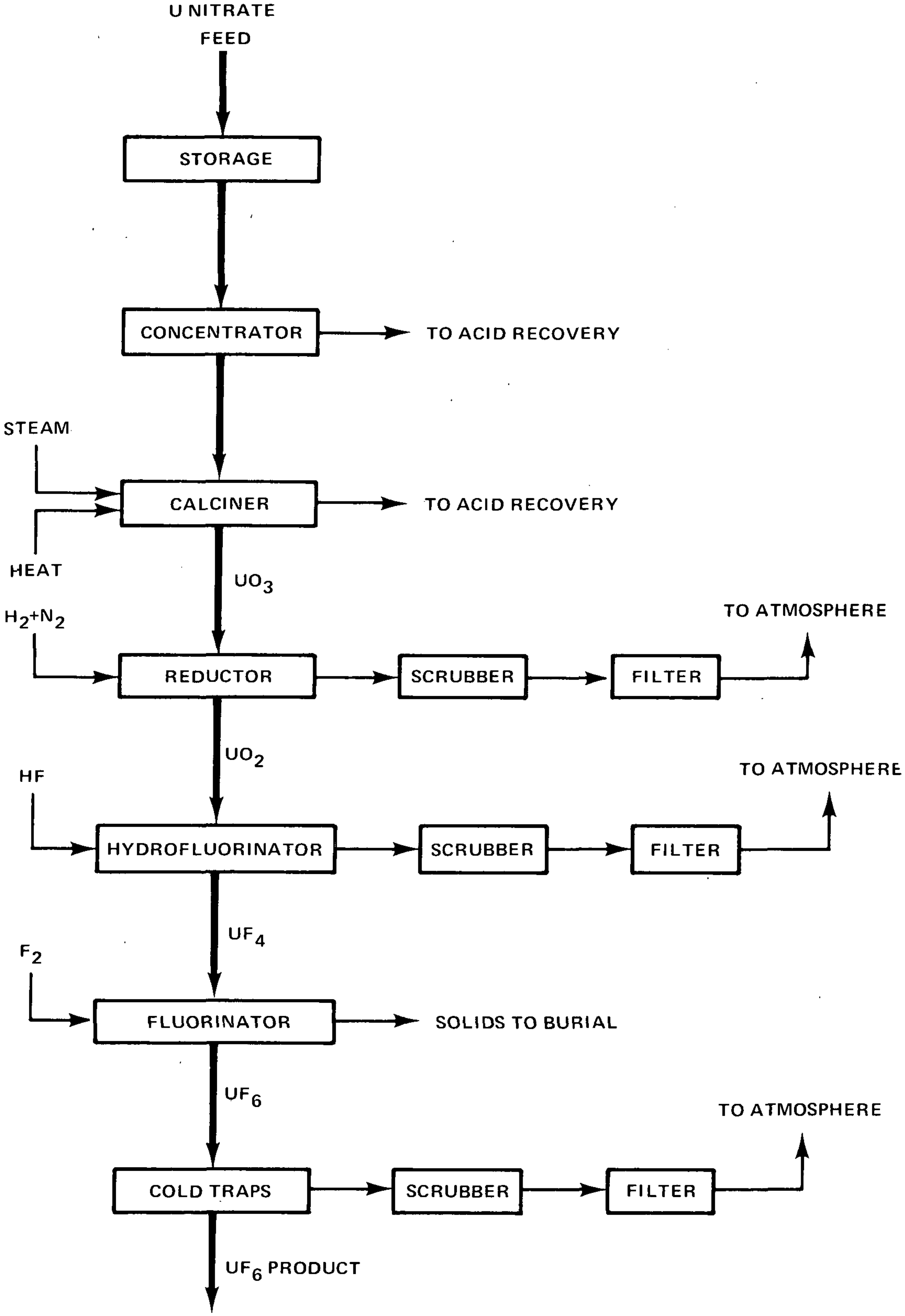

Figure IV E-2 UF_6 Conversion Flow Diagram

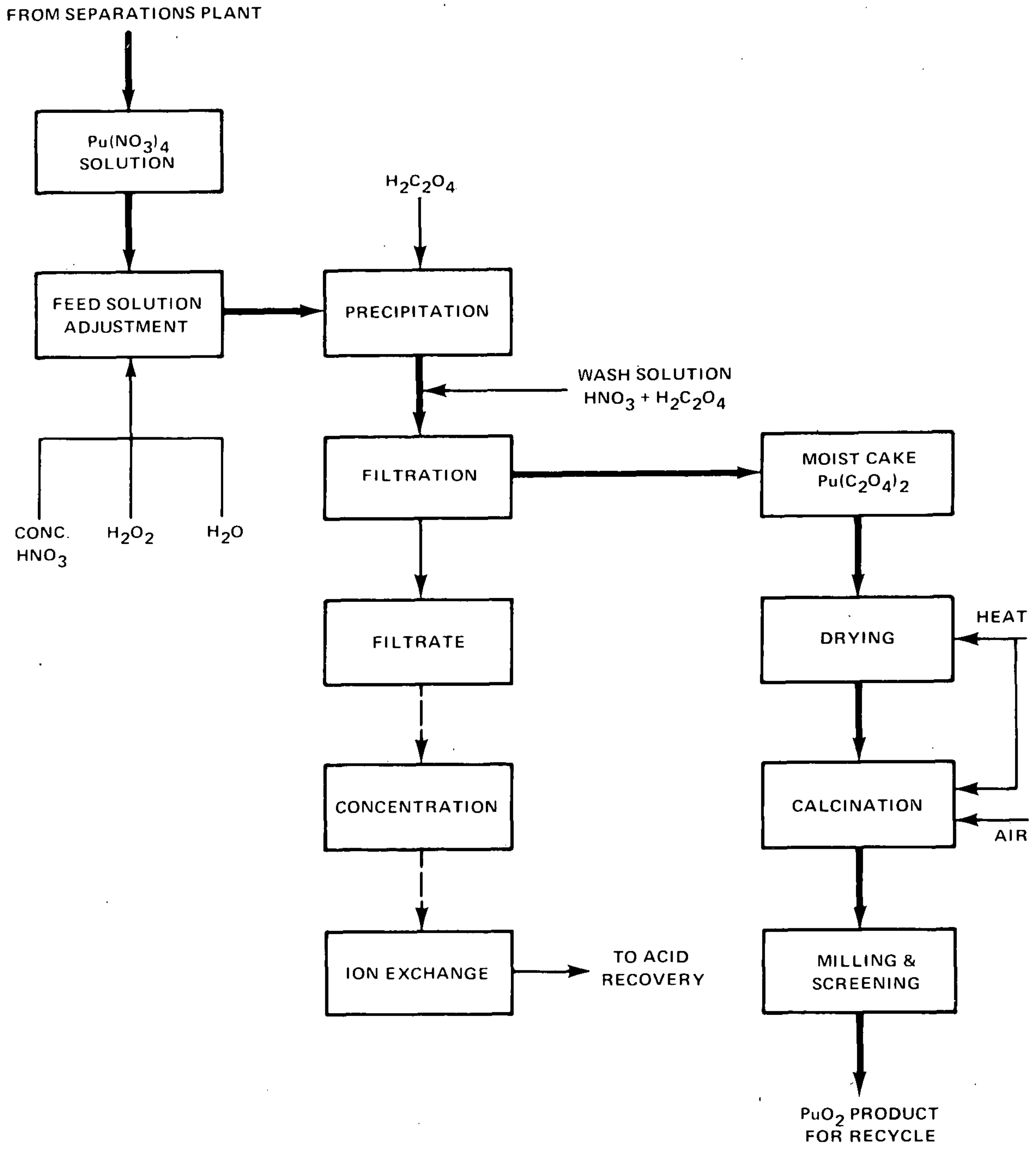

Figure IV E-3 PuO_2 Conversion Flow Diagram

The Commission is considering the amendment of its regulations in 10 CFR Part 20, "Standards for Protection Against Radiation," to prohibit the disposal by burial in soil of transuranium elements.[7] The proposed amendment would require that waste materials containing transuranium elements be transferred to a Federal repository as soon as practicable, but within 5 years after generation. If this amendment is adopted essentially all radioactive waste materials from reprocessing plants, except possibly some solid waste associated with the UF_6 conversion process, would have to be handled so as to enable timely transfer to Federal repositories.

1.1 The Industry Today[8]

Three commercial nuclear fuel reprocessing plants have been constructed in the United States, each located on a relatively large, remote site. None of these plants is now able to reprocess irradiated fuel, but by the early 1980's the United States could have nuclear fuel reprocessing plants in operation. Site location, size, and demographic data for each plant are given in Table IV E-3. The waste treatment systems for each of the three constructed plants are summarized in Table IV E-4.

The first commercial reprocessing plant constructed by Nuclear Fuel Services was a demonstration facility to foster development of a privately owned commercial reprocessing industry. The plant began operation in 1966 and between April of that year and December 1971 successfully recovered purified uranium and plutonium products from approximately 600 metric tons of irradiated fuel. All of the uranyl nitrate product solution returned by NFS to the AEC was acceptable for conversion to UF_6 for use as feed to the gaseous diffusion plants; however, most of it was converted to metal for government programs. Some of the plutonium product was stored as an aqueous solution of $Pu(NO_3)_4$ in a facility provided by the New York State Atomic and Space Development Authority (ASDA). In early 1972, the plant was shut down for modifications and to expand reprocessing capacity to approximately 750 metric tons annually. If this plant is to serve the needs of the nuclear power industry for spent fuel reprocessing services, in addition to proposed major modifications related to the separations facility, NFS would have to add

- A facility to convert uranyl nitrate to UF_6
- Facilities to convert plutonium nitrate to plutonium dioxide and to store plutonium dioxide
- Facilities to convert the high level radioactive liquid wastes to a suitable solid form for transfer to a Federal repository
- Facilities to handle and package other radioactive wastes, such as hulls and intermediate or low level beta, gamma, and transuranic contaminated wastes, for transfer to a Federal repository

Table IV E-3

IRRADIATED FUEL REPROCESSING PLANT SITE DATA AND DEMOGRAPHY

Plant and Location	Plant Capacity MTU/day*	Site Size, Acres	Major Nearby Cities	Population	Distance Miles
Nuclear Fuel Services West Valley, N.Y.	1**	3,500	Buffalo, N.Y.	463,000	28
			Olean, N.Y.	19,000	25
			Springville, N.Y.	4,350	4.5
Morris Operation*** Morris, Ill. (General Electric)	1	890†	Morris, Ill.	8,000	8
			Joliet, Ill.	79,000	15
			Aurora, Ill.	74,000	27
Barnwell Nuclear Fuel Plant Barnwell, S.C. (Allied-General Nuclear Services)	5	1,700††	Barnwell, S.C.	4,500	7.5
			Aiken, S.C.	16,000	26
			Augusta, Ga.	60,000	33

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*Metric tons of uranium per day.

**NFS has applied for a license to operate at 2.5 MTU/day.

***Formerly called the Midwest Fuel Recovery Plant, now used to store spent fuel.

†Adjacent to the Dresden nuclear reactor site of 2,230 acres.

††Adjacent to ERDA's Savannah River Plant exclusion area.

Table IV E-4

FUEL REPROCESSING PLANTS--MAJOR WASTE SYSTEMS

Plant	Gaseous Effluents	Liquid Wastes: High Level	Liquid Wastes: Other than High Level	Solid Wastes
NFS	To atmosphere via iodine scrubber, HEPA filters, 61 meter stack.	Stored onsite as neutralized liquid in tanks in vault.*	Controlled release to creek.	Stored onsite in drums in impervious clay deposits.
MFRP	To atmosphere via iodine scrubber, sand filter, 91 meter stack.	Converted to solid. Stored onsite in containers under water in storage basin.**	Stored onsite in tanks in vault as immobile salt cake.	Stored onsite in underground lined vault.
BNFP	To atmosphere via iodine scrubber, silver zeolite adsorber, HEPA filters, 100 meter stack.	Stored onsite as acidic liquid in tanks in vault.***	Stored onsite in tanks in vault.	Stored onsite in concrete or steel drums.

Note: Information in this table has been compiled from the safety analysis reports for these facilities.

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*Plans to add tanks for storage of high level radioactive liquids in acidic conditions for ultimate conversion to solids and transfer to a Federal storage facility.

**Prior to shipment to Federal storage facility.

***Plans for solidification of high level radioactive liquid wastes for transfer to Federal storage facility.

Nuclear Fuel Services, Inc. is moving toward a decision whether to continue or terminate reprocessing services. They estimate that it may take up to 10 years to modify the separations facility and complete the above additional facilities. Meanwhile, modification of the NFS separations facility and design of additional facilities have been suspended pending, among other things, resolution of issues related to widescale use of mixed oxide fuel in LWR's. NFS is also a party in a contested proceeding before an Atomic Safety and Licensing Board (ASLB) on matters related to the National Environmental Policy Act of 1969 (NEPA).

The second reprocessing plant to be constructed, the Midwest Fuel Recovery Plant, now called the Morris Operation, was a development project to determine whether small reprocessing plants could be economic and could be located in areas relatively near groups of nuclear power reactors. However, as the General Electric Company has publicly disclosed, the results of the preoperational tests revealed technical difficulties related to operating certain systems remotely. Unless corrected, these problems would result in low productivity and high unit operating costs.[9] Accordingly, General Electric has elected not to operate this facility, pending a study to determine whether to modify the facility or abandon it. At present, this facility is being used to store spent fuel.

The third reprocessing plant, the Barnwell Nuclear Fuel Plant (BNFP), is scheduled for completion of construction of the Separations Facility and the UF_6 Facility in late 1976. Additional facilities are needed to achieve the following:

- Convert the plutonium nitrate product to solid plutonium dioxide and store and ship plutonium dioxide

- Handle and package various radioactive wastes containing transuranium elements for transfer to a Federal repository

- Convert the high level and intermediate level radioactive liquid wastes to solid form for transfer to a Federal repository

Design of these additional facilities has been suspended pending, among other things, resolution of issues related to widescale use of mixed oxide fuel in LWR's, and the definition of criteria and charges for the acceptance of various radioactive wastes at Federal repositories. The BNFP is also the subject of a contested proceeding before an Atomic Safety and Licensing Board on matters, related to NEPA, regarding whether the construction permit should be continued, modified, or terminated, and whether the Fuel Receiving and Storage Station and the Separations Facility should be licensed for operation. Future operation of BNFP depends, in part, upon resolution of issues being litigated.

On January 28, 1976, Exxon Nuclear Company, Inc., filed an application (Docket No. 50-564) with NRC for licenses to construct and operate a Nuclear Fuel Recovery and Recycling Center (NFRRC) on a 160-acre site located within a 2,500-acre tract in Roane County, Tennessee. Initially, Exxon Nuclear proposes to construct a facility that would have a capacity to store up to approximately 7,000 metric tons of irradiated nuclear fuel. Exxon Nuclear also proposes to construct a complex of facilities for reprocessing nuclear reactor fuels. Earliest date for completion of construction of the initial chemical processing capacity of the NFRRC (up to 1,500 metric tons of uranium and plutonium per year) is the year 1984; the latest date is 1988. Exxon Nuclear plans to increase the chemical processing capacity for NFRRC (up to 2,100 metric tons of uranium and plutonium per year) by the year 1992. An estimate of total capital costs for the construction of the initial capacity of NFRRC is approximately $600 million in 1975 dollars, excluding interest on construction. Final costs could exceed $1 billion depending on interest on construction and inflation experienced during the years in question.[10]

In 1976, the United States does not have government or commercial capability to reprocess spent power reactor fuel. However, detailed design and construction of the BNFP facilities that are required in addition to the Separations Facility and the UF_6 Facility could be completed in about 40 months. Accordingly, the nation could have the overall capability to reprocess spent power reactor fuel and initiate plutonium recycle about 3 to 4 years after resolving the question of whether widescale use of mixed oxide fuel in LWR's is acceptable.

1.2 The Future Industry

The future of a fuel reprocessing industry is related to the widescale recycle of plutonium as mixed oxide fuel in LWR's. If there is a net benefit, there is a basis for ongoing reprocessing services. Whether spent fuels are reprocessed at this time solely to recover and recycle uranium is an alternative considered in this study (see CHAPTER XI). There would be no fuel reprocessing industry if the spent fuels are not reprocessed; the spent fuels would become the high level radioactive waste. An industrywide comparison of environmental effects related to no recycle, recycle of only uranium, and recycle of both uranium and plutonium is presented in CHAPTER VIII, Appendix A.

The current projection of the amounts of spent fuel to be discharged from LWR's, and the prospective growth of the spent fuel reprocessing industry if these fuels are reprocessed, are shown in Table IV E-5. If these spent fuels are not reprocessed, they may be regarded as radioactive waste. For additional information relevant to the disposal of spent fuel as waste, see CHAPTER IV, Section H. Accordingly, the remainder of this section addresses the environmental effects related to the alternatives of reprocessing spent fuel for recovery of uranium and recovery of plutonium for widescale use of mixed oxide fuel in LWR's.

Within 3 to 4 years after a decision to recycle U, or to recycle U and Pu, Allied-General Nuclear Services could complete necessary additional facilities and provide reprocessing services. By the year 1980, the backlog of spent fuels available for reprocessing may total more than 4,000 metric tons. It is estimated that the next new reprocessing plant could be designed and constructed within 8 to 10 years; thus it might begin operation about 4 years after BNFP. Thereafter, additional reprocessing plants, plus expansion, should provide the additional capacities needed for reprocessing services, as shown in Table IV E-5.

Table IV E-5

PROSPECTIVE FUEL REPROCESSING LOAD
(Metric Tons of Heavy Metals)

Year Ending	Reactor Discharges	Stored Fuel	Fuel Reprocessed	Equivalent Plants*
1975	550	1,700		
1976	800	2,500		
1977	900	3,400		
1978	1,100	4,000	500**	1
1979	1,250	4,250	1,000	
1980	1,350	4,100	1,500	
1981	1,550	4,150	1,500	
1982	1,800	4,200	1,750	
1983	2,250	4,450	2,000	
1984	2,450	4,650	2,250	
1985	2,950	5,350	2,250***	
1986	3,500	6,000	2,850	2
1987	4,050	6,500	3,550	
1988	4,650	6,900	4,250	
1989	4,950	7,600	4,250	
1990	5,450	8,200	4,850	3
1991	6,150	8,800	5,550†	
1992	6,650	9,200	6,250	
1993	7,150	9,500	6,850	4
1994	7,800	9,750	7,550	
1995	8,400	9,900	8,250	
1996	9,000	10,650	8,250	
1997	9,550	10,950	9,250††	5
1998	10,100	10,800	10,250	
1999	10,450	11,000	10,250	
2000	10,950	11,700	10,250	

---

*Number of equivalent 2000 MT/yr model plants.

**Earliest start of reprocessing without conversion of plutonium to solid form.

***Start reprocessing 1st Pu recycle (98.4% UO_2 + 1.6% MOX).

†Start reprocessing 2nd Pu recycle (91% UO_2 + 9% MOX).

††Start reprocessing 3rd Pu recycle (86% UO_2 + 14% MOX).

Current projections of plutonium recycle in LWR's indicate that by the year 1990 mixed oxide fuel will make up approximately 9% of the total LWR fuel reprocessed; by the late 1990's, the maximum amount of mixed oxide fuel reprocessed in any one year would be approximately 18% of the total LWR fuel reprocessed. The remaining 82% would be UO_2 fuel. For the years 1975 through 2000, the total industrywide fuel mix would average approximately 11% MOX fuel and 89% UO_2 fuel.

1.2.1 Effect of Reprocessing Mixed Oxide Fuel

Initially, only UO_2 fuel will be available for reprocessing. If U and Pu are recycled, reprocessing of MOX fuel might begin about the year 1985. Environmental impacts from reprocessing operations may differ for UO_2 fuel and mixed oxide fuel, because irradiated mixed oxide fuel differs from irradiated UO_2 fuel in

- Plutonium content
- Dissolvability
- Fission product and transuranium element content
- Decay heat generation of fuel and wastes.

1.2.1.1 Plutonium Content

Based on total fuel requirements over a 30-year life, a 1.15 SGR plutonium recycle BWR would discharge MOX fuel containing an average of about 19 kg Pu/MT. Uranium oxide fuel irradiated to 33,000 MWd/MT would contain about 9.5 kg Pu/MT. An industrywide mix composed of 89% UO_2 fuel and 11% MOX fuel would contain an average of about 12 kg Pu/MT. Certain MOX fuel pins irradiated to 33,000 MWd/MT in a PWR may contain the equivalent of about 49 kg Pu/MT. Because of the in-process inventory associated with continuous reprocessing operations, a reprocessing plant operator will have limited flexibility in scheduling fuel reprocessing operations to segregate small batches of uranium or plutonium. In general, operations are scheduled by reactor fuel lots. Thus, a 1.15 SGR plutonium recycle fuel lot might contain an average of about twice the amount of plutonium as would a UO_2 reactor fuel lot. Certain PWR MOX elements, however, might contain up to five times the concentration of plutonium compared to a UO_2 fuel element irradiated to 33,000 MWd/MT. Such variances can be accommodated by appropriate flowsheet adjustments. Assuming the total spent fuel load between year 1975 and year 2000 is composed of 89% UO_2 fuel and 11% MOX fuel, the 28% increase in plutonium throughput, compared to reprocessing only UO_2 fuel, would have only a slight effect on reprocessing capacity or operating cost.

1.2.1.2 Dissolvability

Initial experiments as to dissolvability of unirradiated, mechanically blended PuO_2-UO_2 fuel pellets indicated that an appreciable amount of the plutonium might remain undissolved after a series of leaches in boiling 8 molar nitric acid.[11] These experiments, however, also indicated that irradiated fuel pellets are somewhat more soluble than unirradiated pellets. Subsequent experiments determined that the fabrication and irradiation histories of mixed oxide fuels have an important effect on both the dissolution rate and the dissolvability of the fuel:[12,13] if the unirradiated mixed oxide fuel pellets have a high degree of homogeneity and have been sintered at about 1,650°C for about 4 hours, there should be little difference between the dissolvability of irradiated UO_2 pellets and irradiated PuO_2-UO_2 pellets.

Although there may be initial problems with dissolving some MOX elements, such as establishing optimum acid concentrations and time cycles, these problems would be solved shortly after MOX fuel reprocessing began. Those irradiated mixed oxide fuels that had inadequate control of fabrication variables (homogeneity and sintering temperature), however, might contribute to a slight overall increase in the amount of plutonium in the high level radioactive waste. For this reason, it is expected that reactor operators will require that LWR mixed oxide fuels produced by mechanical blending achieve homogeneity and be sintered.

1.2.1.3 Fission Product and Transuranium Element Content

Plutonium fissioning produces a slight shift in the fission product yield spectrum toward the higher mass number isotopes, relevant to the mass-yield profile, as shown in Figure IV C-25. Thus, the composition of fission products in mixed oxide fuel is slightly different than that in UO_2 fuel. Table IV E-6 gives a ratio comparison, in relation to the content in UO_2 fuel, of the content of biologically important fission products in a 1.15 SGR plutonium recycle fuel lot and in the industrywide average feed to a reprocessing plant.[14] These ratios fall within the variations that occur because of differences in the exposure and age of the various reactor fuel batches that will be reprocessed. For the purpose of this evaluation, the effluent releases are assumed to vary directly in relation to the feed composition as shown in Table IV E-6: plutonium recycle would cause a slight increase in the amount of tritium, ruthenium, and iodine, and a slight decrease in the amount of radioactive carbon, krypton, and strontium, in the off-gas effluent from the reprocessing plant.

Table IV E-7 compares weight ratios of the transuranium element content in irradiated UO_2 fuel with the average annual feed to the reprocessing plants and the average 1.15 SGR plutonium recycle fuel lot. As indicated, recycling plutonium results in the assumed 1.15 SGR mixed oxide fuel composition. In comparison to UO_2 fuel, this increases the overall amounts of americium by a factor of 2 to 4, increases the overall amounts of curium by a factor of 2 to 6, changes the isotopic composition of the plutonium, and increases the amount of plutonium that would be processed in the year 2000 by about 50%. These changes affect the composition of normal and accidental releases of radioisotopes in the plant's effluents.

The americium and curium elements follow the fission products, for the most part, from the codecontamination cycle of the Purex process to the high level radioactive waste treatment system. Consequently, the increased americium and curium content in plutonium recycle fuel is expected to have little, if any, effect on shielding requirements or employee exposure in the separations facility. To protect employees from radiation exposure and to maintain exposure levels as low as can be reasonably achievable, the isotopic composition and increased amounts of plutonium associated with mixed oxide fuel are factored into the design of plutonium conversion and fuel fabrication facilities.

Table IV E-6

CONCENTRATION OF PRINCIPAL FISSION PRODUCTS IN IRRADIATED SPENT FUEL

Basis: Ratios relative to UO$_2$ fuel irradiated at 26 MW/MT to 33,000 MWd/MT and aged 160 days, with U and Pu recycle.

Principal Fission Products	Half Life	Cumulative Average Years 1975-2000 (89% UO$_2$ + 11% MOX)	Year 2000 (82% UO$_2$ + 18% MOX)	1.15 SGR Fuel* (60% UO$_2$ + 40% MOX)
^{3}H	12y	1.01	1.02	1.05
^{14}C	5,730y	0.96	0.93	0.84
^{85}Kr	10.4y	0.96	0.93	0.85
^{90}Sr	28y	0.96	0.93	0.83
^{95}Zr	65d	0.99	0.98	0.96
^{106}Ru	1y	1.08	1.13	1.29
^{129}I	1.6×10^{7}y	1.04	1.05	1.14
^{131}I	8.05d	1.0	1.0	1.0
^{134}Cs	2.1y	1.0	1.0	1.0
^{137}Cs	30y	1.0	1.0	1.0
^{144}Ce	284d	0.98	0.98	0.95

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*Average fuel lot for the years 1985-2000, from 1.15 SGR's not yet at equilibrium.

Table IV E-7

TRANSURANIUM ISOTOPES IN IRRADIATED SPENT FUEL

Basis: Ratios relative to UO$_2$ fuel; irradiated at 26 MW/MT to 33,000 MWd/MT and aged 160 days, with U and Pu recycle.

Isotope	Half Life	UO$_2$ Fuel (grams/MT)	Cumulative Average Years 1975-2000 (89% UO$_2$ + 11% MOX) (Weight Ratios)	Year 2000 (82% UO$_2$ + 18% MOX) (Weight Ratios)	1.15 SGR Fuel* (60% UO$_2$ + 40% MOX) (Weight Ratios)
^{238}Pu	86y	235	1.44	1.79	2.65
^{239}Pu	2.4×10^{4}y	5,250	1.16	1.27	1.60
^{240}Pu	6.6×10^{3}y	2,230	1.40	1.68	2.48
^{241}Pu	13y	1,280	1.32	1.70	2.19
^{242}Pu	3.9×10^{5}y	468	1.75	2.38	3.81
Total Pu		9,463	1.27**	1.49**	2.02**
^{241}Am	458y	59	1.64	2.10	3.35
^{243}Am	7.5×10^{3}y	84	2.39	3.82	6.18
^{242}Cm	162d	6	1.96	2.69	4.57
^{244}Cm	18y	19	3.72	5.84	10.95

---

*Average fuel lot for the years 1985-2000, from 1.15 SGR's not yet at equilibrium.
**Curie ratios are 1.32, 1.69, and 2.20.

Secondary effects of increased amounts of transuranium elements in mixed oxide fuel occur in the high level radioactive waste. Americium-243 decays to ^{239}Pu, ^{242}Cm decays to ^{238}Pu, and ^{244}Cm decays to ^{240}Pu (Figure IV E-4).[15] Thus, in addition to an increase in decay heat, the increase in americium and curium isotopes may cause slight increases, over long periods of time, in the plutonium content of the high level radioactive waste. Because the high level radioactive wastes are not released to the environment, this change would have no significant effect.

1.2.2 Heat Content of High Level Radioactive Waste

The decay heat from americium and curium in UO_2 fuel is small relative to the fission product decay heat, only about 4%-5% during 150 days' to 10 years' decay after reactor discharge. However, because of increased amounts of americium and curium, the total high level radioactive waste from reprocessing mixed oxide fuel could have a decay heat content (after 10 years) about 25% greater than the decay heat content of wastes from UO_2 fuel. In fact, after 3 years, the decay heat content of high level radioactive waste from a 1.15 SGR mixed oxide fuel lot could be about 40% greater than from a UO_2 fuel lot. After 10 years, the ^{244}Cm in the high level radioactive waste from mixed oxide fuel lots would contribute about as much decay heat as would the fission products. Thus, a reprocessing plant that solidifies the high level radioactive wastes as generated, without blending the wastes from UO_2 fuel and MOX fuel, could have substantial variations in the heat load, or amount of waste, from canister to canister. Reprocessing plants that store an inventory of liquid high level radioactive waste prior to solidification would have less variations in the power density of high level radioactive waste. The solidified high level radioactive waste from the total (years•1975-2000) reprocessing fuel mix of 11% MOX fuel and 89% UO_2 fuel would have about 25% more decay heat 10 years after irradiation (because of the increased ^{244}Cm content) than an equivalent amount of similarly aged solidified waste from UO_2 fuel. Depending on how these solidified high level radioactive wastes will be transported to and stored at a Federal repository, an increase in decay heat content could cause a slight increase of about 1% in disposal costs related to added space requirements for storing high level solid waste at the Federal repository. See CHAPTER IV, Section G, Transportation of Radioactive Materials, and Section H, Radioactive Waste Management.

2.0 THE MODEL FUEL REPROCESSING PLANT

If plutonium is recycled as fuel in LWR's, over the period from the year 1975 through the year 2000 the amount of mixed oxide fuel reprocessed is expected to total about 11% of the total fuel reprocessing load, most of this to be reprocessed in plants not yet designed. Existing and proposed fuel reprocessing plants have design capacities which range from 300 to 2,100 MT/yr. However, the design capacity of later reprocessing plants could exceed 3,000 MT/yr by year 2000. Although future plants are likely to be larger capacity plants than those currently constructed, for this evaluation, the model fuel reprocessing plant is taken to be similar to current designs, located on a 2,000-acre site in a sparsely populated area, and have a design throughput capacity of 2,000 metric tons of heavy metal (U and Pu) per year. The process system is described by Figures IV E-1, E-2, E-3, and E-5.

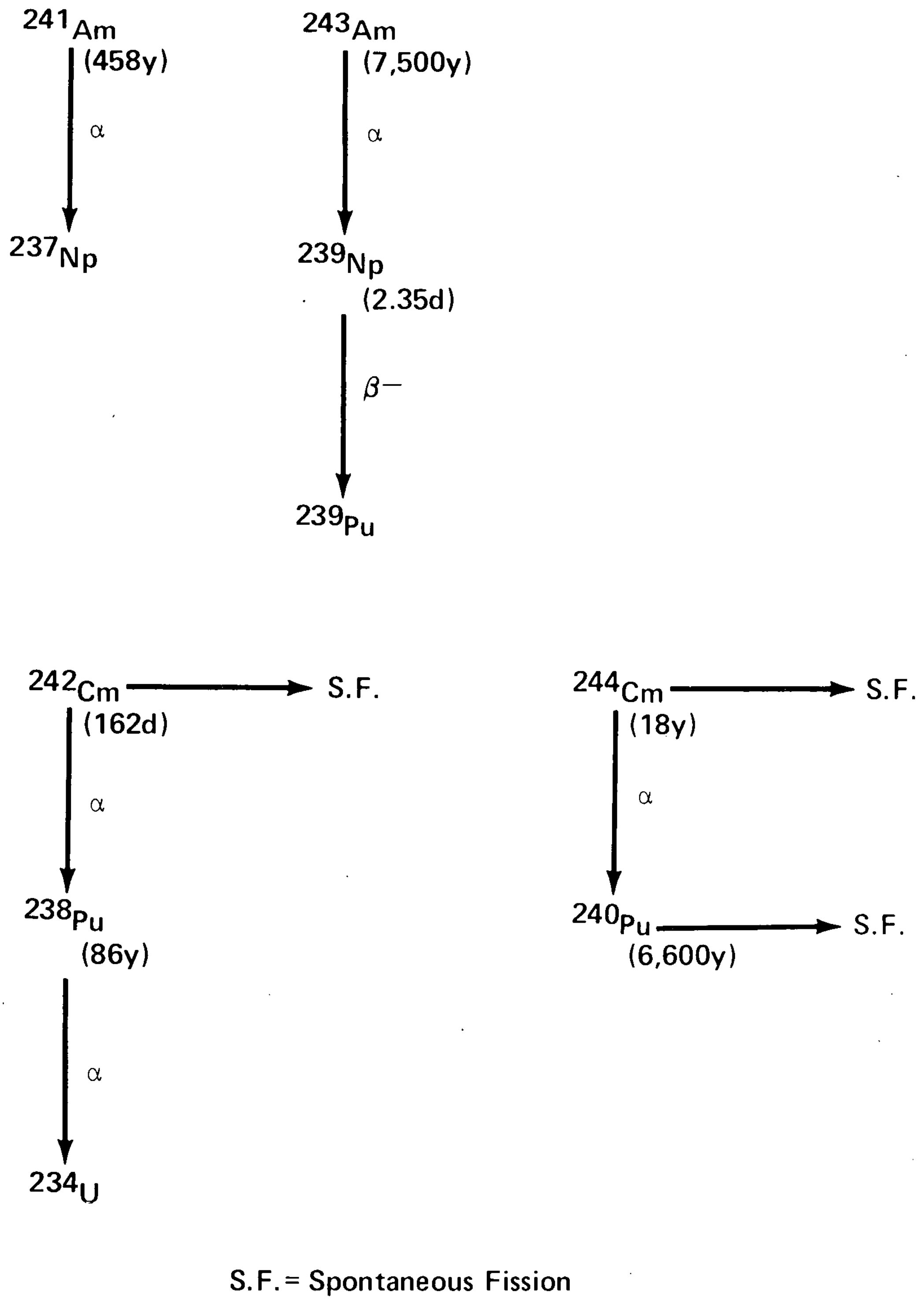

Figure IV E-4 Decay Schemes of Selected Transuranium Isotopes

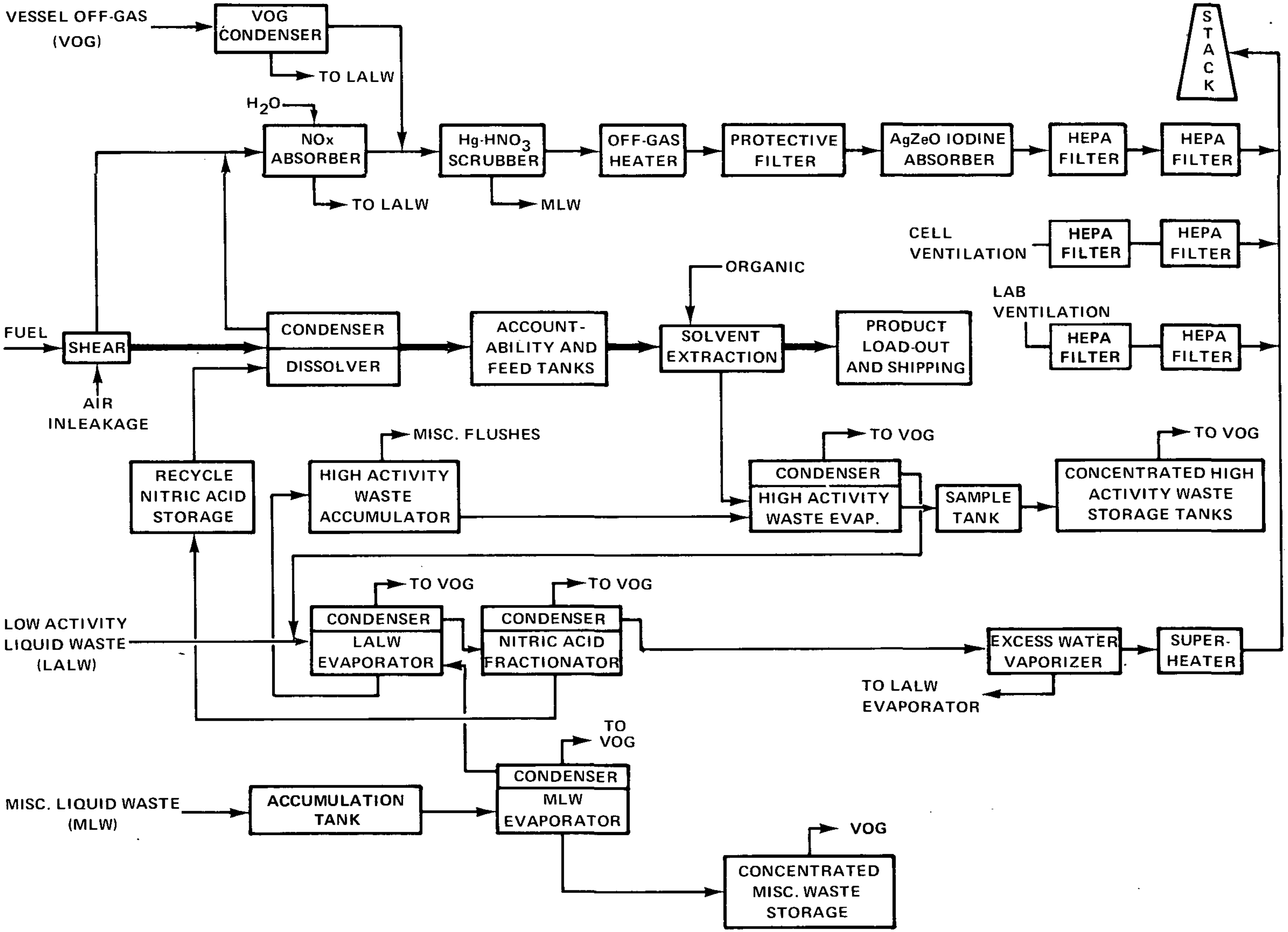

Figure IV E-5 Generic Reprocessing Plant Effluent Systems

Initially the model plants reprocess only UO_2 fuel. When plutonium recycle begins, no significant changes in reprocessing throughput, chemical effluents, or power requirements are expected. However, there may be a slight change in the amount and isotopic composition of the transuranic radionuclides released in the off-gas effluent from the plant. It is assumed here that radioactive effluents will vary directly with throughput and the plant's annual average feed mix.

Structures, systems, and components that confine radioactive materials are designed to withstand the effects of earthquakes and other natural phenomena such as tornadoes, hurricanes and floods. Cooling water, from wells or a surface stream, is returned to a surface stream via cooling towers and cooling ponds. All process and ventilation air which might contain radioactive contaminants is treated or filtered, or both, before discharge to the atmosphere via a 100-meter tall stack.* The plant has standby diesel equipment for emergency power and water supply, an emergency water distribution system, water treatment systems, boilers to produce steam for heating and process needs, air compressors, electrical switchgear, and sanitary waste treatment systems.

Shielding and administrative controls such as special work permits and limited access for maintenance operations, protect personnel from exposure to radiation. Ventilation air flow is controlled so that air flows from personnel-inhabited areas of low (or no) contamination to process areas of progressively higher contamination, and then to filtration systems before discharge to the atmosphere.

The introduction of plutonium recycle fuel is not expected to cause any significant change in chemical effluents from the plant. The average annual concentration of chemical pollutants (NO_x, SO_x, CO, F) at the plant boundary is not expected to exceed Federal or State standards.[16] However, the conversion of $Pu(NO_3)_4$ to PuO_2 using the oxalate conversion process will result in a small increase in CO emissions. Also, the increased heat content of plutonium recycle fuels and its resultant high level radioactive waste could cause a slight increase in cooling water requirements. The estimated environmental impact of a 2,000 MT/yr fuel reprocessing plant that reprocesses an annual average fuel load comprised of 11% MOX fuel and 89% UO_2 fuel (irradiated to 33,000 MWd/MT and cooled 160 days before reprocessing) is tabulated and compared to reprocessing UO_2 fuel only in Table IV E-8.

2.1 Potential for Reducing Radioactive Effluents from Reprocessing Plants

The estimated release of radioactive effluents from the 2,000 MT/yr model reprocessing plant, shown in Table IV E-8, is based on the lower range of the expected performance for each component in the overall effluent treatment system (Figure IV E-5). It has been conservatively assumed that the irradiated fuel is reprocessed about 160 days after being discharged from the reactor and that all of the ^{3}H, ^{14}C, and ^{85}Kr in this fuel is released at the reprocessing plant. Given the amounts of irradiated spent fuel shown to be in storage in Table IV E-5, during the next decade

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*In general, the dispersion of effluents depends upon the height of release, meteorological conditions, and the distance from the release.

Table IV E-8

SUMMARY OF ENVIRONMENTAL CONSIDERATIONS RELATED TO REPROCESSING 2000 MT/Yr Model Plant

	UO_2 Fuel	11% MOX + 89% UO_2 Fuel
Natural Resource Use		
Land (acres)		
Temporarily committed	1,800	1,800
Permanently committed	200	200
Total	2,000	2,000
Water (millions of gallons/yr)		
Discharged to air	375	400
Discharged to water	3,125	3,150
Total	3,500	3,550
Fossil Fuel (per year)		
Electrical Energy (MW-hr x 10^3)	230	230
Equivalent Coal (MT x 10^3)	84	84
Fuel Oil (gal x 10^6)	13	13
Propane Gas (gal x 10^3)	120	120
Effluents		
Chemicals (MT/yr)		
Gases		
SOx	310	310
NOx	1,250	1,250
Hydrocarbons	27	27
CO	24	26
Particulates	33	33
Fluorides	3	3
Liquids		
SO_4	<1	<1
NO_3	-	-
Cl	5	5
Na+K	<1	<1
Fe	-	-
Thermal Btu/yr x 10^9		
Fossil Fuel	2,700	2,700
Decay Heat	1,600	1,700
Total	4,300	4,400
Radiological (Ci/yr)		
Gases (including entrained matter)		
^3H	1,030,000	1,040,000
^{14}C	1,360	1,300
^{85}Kr	22,800,000	21,900,000
^{90}Sr	0.32	0.31
^{106}Ru	8.13	8.77
^{129}I	1.91	1.98
^{131}I	7.5	47.8
Other fission products	10	10
U	2.23×10^{-3}	2.69×10^{-3}
Puα	0.05	0.07
Puβ	1.28	1.76
^{241}Am	8.13×10^{-4}	1.33×10^{-3}
^{243}Am	6.44×10^{-5}	1.54×10^{-4}
^{242}Cm	8.49×10^{-2}	1.66×10^{-1}
^{244}Cm	6.11×10^{-3}	2.27×10^{-2}
Population Dose Commitments (person-rem)		
Occupational	1,250	1,350
U.S. Population	18,800	18,600
Foreign Population	12,200	11,700
Total	32,250	31,650

most of the spent fuel is likely to be aged 2 or more years before being reprocessed. Consequently, the amounts of ^{106}Ru, ^{144}Ce, ^{242}Cm, and the decay energy released from reprocessing spent fuel are likely to be substantially less than indicated in Table IV E-8, because this table is based on a 160-day cooling period. There would be essentially no ^{131}I released from reprocessing the longer cooled spent fuel.

The amount of ^{14}C that may be in irradiated LWR fuels has not been measured, and thus the fraction, if any, that is released from a reprocessing plant is unknown. In this assessment the maximum amount of ^{14}C that would likely be in LWR fuels has been calculated and it is assumed that all of the carbon in the fuel is oxidized and released from the plant during dissolution of the fuel. This hypothesis significantly affects the assessment of radiological effects associated with the release of radioactive effluents from the reprocessing plant: the estimated ^{14}C releases account for approximately 20% of the estimated total body dose commitment and approximately 30% of the estimated bone dose commitment to the hypothetical nearest resident. Also, the estimated ^{14}C releases account for approximately 33% and 55% of the estimated total body dose commitments, and 67% and 86% of the estimated bone dose commitments, to the U.S. population and foreign population, respectively. Because the amount of ^{14}C in MOX fuels would be somewhat less than the amount in UO_2 fuel, because of the lower neutron flux in MOX fuel, the estimated reduction in ^{14}C releases associated with Pu recycle tends to offset increases in dose commitments related to increased amounts of ^{3}H and transuranium elements that may be released during the reprocessing of Pu recycle fuels. The relative effect of the estimated releases of principal radionuclides is shown in Tables IV E-9 through IV E-13.* It should be noted that if appreciable amounts of ^{14}C are actually present and released during the dissolution of irradiated LWR fuel, the ^{14}C could be removed from the off-gas effluent by proven technology which has been commercially demonstrated. One approach for removal of ^{14}C is to adsorb CO_2 on molecular sieves; another is removal by an aqueous caustic scrubber. If necessary, a combination of the two systems could be used.

The state of the art for various effluent treatment methods varies from well established technology used in existing installations to advanced technology currently in the small engineering scale of development. A report prepared by Oak Ridge National Laboratory (ORNL), "Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing 'As Low As Practicable' Guides--Nuclear Fuel Reprocessing," ORNL-TM-4901, dated May 1975, provides, in Section 4.3, a description of various treatment methods. The following is a brief summary of the potential for further reducing releases of particulates, ^{85}Kr, ^{3}H, iodine, and ruthenium from fuel reprocessing plants.

2.2. Particulate Removal

For this assessment, particulate release factors of from 2×10^{-9} to 5×10^{-9} were assumed. These are believed to understate treatment systems' capabilities, and thus are conservative by 2 to 4 orders of magnitude with respect to expected performance of the off-gas effluent treatment systems that will be used in current or future plants. The radioactive effluent estimates in Table IV E-8 include releases from the

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*Tables IV E-9 through IV E-13 follow paragraph 3.1

PuO_2 conversion, the UF_6 conversion, and the high level radioactive waste solidification facilities. They are not significant additions relevant to the radioactive effluents from the separations facility. Because a multiple effluent treatment system, as illustrated in Figure IV E-5, has not yet been operated, there is no actual measured data to confirm the expected overall performance of such systems. Therefore, a conservative release estimate has been made, based on the lower values of the demonstrated performance of the various components making up the overall effluent treatment system.

In general, through the use of appropriate deentrainment mechanisms, the condensate from evaporation of a radioactive solution can be reduced in nonvolatile radioactivity to only 10^{-4} to 10^{-6} of the radioactivity of the feed solution.[17] Plans are to use several stages of evaporation to decontaminate the vaporized process condensate discharged to the atmosphere. Knock-out pots, scrubbers, and adsorption beds are used to remove entrained radioactive particles from the process off-gases. Each of these unit operations is capable of removing 90% or more of the radioactivity. High efficiency particulate air (HEPA) filters have demonstrated removal efficiencies of >99.95%.[8,19] Sand filters have about the same efficiency as a single stage of HEPA filters.[20] For the most part, process off-gases pass through two or more stages of filtration. Thus, it is expected that the proposed multiple effluent treatment system will remove radioactive particulate to as low a level as can be reasonably achievable.

2.3 Krypton-85 Removal

A selective absorption and a cryogenic system are being developed for the possible future removal of noble gases. Required are further engineering development and demonstration in a pilot plant or reprocessing plant on the off-gas from the dissolution of irradiated LWR fuels. Either process may be ready for use by about the year 1983. However, for this assessment, it was assumed that all of the ^{85}Kr is released to the environment, which assesses fuel reprocessing with the maximum radiological impact related to ^{85}Kr.

The principal concern arising from the release of ^{85}Kr from reprocessing plants is the dose commitment (person-rem) delivered to populations. Over the years 1980-2000, the United States might contribute approximately 20% of the total ^{85}Kr dose commitment to the world population. Estimated dose commitments as a result of assumed ^{85}Kr releases from all worldwide facilities through the year 2000 are about 0.03 mrem whole body per year or about 1/3000 that of natural background radiation.[21]

2.4 Tritium Removal

Because most LWR's utilize Zircaloy-clad fuel, essentially all (∿99%) of the tritium resulting from ternary fission will be contained in the fuel elements transported to the fuel reprocessing plants. The tritium content of the UO_2 fuel is assumed to be approximately 515 Ci/MT at a burnup of 33,000 MWd/MT in PWR fuel enriched to 3.3% ^{235}U: the tritium content of the fuel averages approximately 15.6 Ci/1,000 MWd. The fissioning of plutonium increases the yield of tritium in Pu recycle fuel.

Tritium generated in LWR fuel is subjected to high heat fluxes, and hence it migrates to some extent through the fuel. The affinity of the zirconium cladding for hydrogen results in a significant part of the tritium (30-60% of the total) combining with the cladding.[22] This combined tritium is immobile under the conditions of reprocessing operations. In this assessment, it was assumed that 100% of the ^3H present in the irradiated fuel is released to the atmosphere as ^3H or HTO (tritiated water vapor). This assesses fuel reprocessing with the maximum radiological impact related to ^3H.

Because there is no practical way to separate and remove the tritium from water, it will be released, for the most part, along with excess process condensate. The alternatives for disposal of excess condensate are to discharge to the atmosphere as a vapor or discharge to surface streams as a liquid effluent. In certain cases, depending upon the geology of the site, deep well disposal also might be a viable alternative to mitigate the release of ^3H to the hydrosphere. The NRC staff evaluates these alternatives for each facility to determine which alternative would result in the lower radiological impact on the regional population.

A voloxidation process has been demonstrated on a laboratory scale, to remove tritium from the fuel prior to mixing with process solutions. In this process, the sheared fuel is heated in a rotary calciner to about 550°C in air or oxygen to release the volatile fission products via thermal evolution or oxidation before dissolution. The major emphasis is centered on the removal and isolation of tritium to prevent the mixing of the tritium with natural water in the dissolver. However, rotary calciner engineering development has been limited to tests with unirradiated UO_2 and to design studies. Much additional engineering development and a demonstration of such a system in a pilot plant with irradiated LWR fuel will be required to make this system practical.

2.5 Ruthenium and Iodine Removal

For this assessment, it was assumed that the irradiated fuels were reprocessed about 160 days after discharge from the reactors. However, as indicated in Table IV E-5, there is likely to be a substantial amount of irradiated fuel in storage that would be cooled 2 or more years after discharge from reactors. Fuels cooled more than 200 days from reactor discharge would have little residual ^{131}I remaining in the fuel. For each year the irradiated fuel remains in storage, the ^{106}Ru content is reduced by roughly a factor of two. Thus, fuel cooled 2 years would have only one-fourth of the ^{106}Ru initially present in the irradiated fuel. The assumption that the fuels are reprocessed about 160 days after discharge from the reactors assesses the fuel reprocessing plant with the maximum radiological impacts related to releases of ^{106}Ru, ^{131}I, ^{144}Ce, and ^{242}Cm, because the amounts of these shorter lived radionuclides diminish with time due to radioactive decay in the spent fuel.

Although a retention factor of 10^8 was assumed for ruthenium radionuclides, the overall retention factor for ruthenium is expected to be in excess of 10^9. Addition of sugar to waste concentrators, to assure the presence of NO_2, suppresses the volatilization of RuO_4. Under these conditions, it is estimated that less than 10^{-5} of the total Ru will enter the process off-gas treatment system. The off-gas treatment system is estimated to provide a retention factor of 10^4 to 10^5; for a total Ru retention of 10^9 to 10^{10}.

During fuel dissolution, more than 90% of the iodine in the irradiated fuel is expected to enter the off-gas treatment system; the remaining 10% or less will enter the process streams. The liquid waste effluent treatment system is expected to be the limiting factor governing the amount of iodine that might be released from a fuel reprocessing plant. Because there is very little data concerning retention of iodine in the liquid waste treatment and acid recovery systems, it was conservatively assumed that the retention factor for ^{129}I might be as low as 40. This includes a retention factor of about 4 for mercuric nitrate addition to the vaporizer feed to suppress iodine volatility.

Because processes for removal of iodine from liquid effluents are not well developed, limited laboratory-scale studies have been conducted to define promising systems that might be used, if necessary, to improve iodine retention. But, before such systems can be made practical, further engineering-scale development and demonstration will be necessary. These potential iodine removal systems include

- Volatization from water or dilute acid by steam stripping which can remove up to 95% of iodine in some cases. This approach is effective only at iodine concentrations above $\sim 10^{-5}$ moles of iodine per liter. The efficiency of the steam stripping operations can be improved and the system applied to lower concentrations by the addition of stable ^{127}I, followed by treatment with ozone to enhance isotopic exchange.

- Removal by factors of 100 or more from liquid streams by sorption on macroreticular resins or charcoal appears possible. Charcoal is effective for water, and various resins are effective over specific acid ranges up to 6 molar nitric acid.

- Addition of mercuric nitrate to evaporator or vaporizer bottoms to suppress iodine volatility may produce retention factors of approximately ten.

3.0 CHANGES IN RADIOLOGICAL ENVIRONMENTAL EFFECTS FROM REPROCESSING MIXED OXIDE AND URANIUM OXIDE FUELS

As noted, plutonium recycled as fuel in LWR's will not affect the amount of fuel to be reprocessed. The principal changes will come from changes in radioactive effluents. The estimated release of radioactive effluents from the 2,000 MT/yr model reprocessing plant is shown in Table IV E-8. Derivations of the dose commitments are described in CHAPTER IV, Section J.

3.1 Normal Operations--Population Dose Commitment

Table IV E-9 gives the estimated population dose commitment from a model plant reprocessing UO_2 fuel; Table IV E-10 gives the estimated population dose commitment from the same plant reprocessing 89% UO_2 fuel and 11% mixed oxide fuels; Table IV E-11 presents the estimated change in population dose commitment caused by introduction of mixed oxide fuel.

Population dose commitment estimates are affected most by the release of ^{3}H, ^{14}C, and ^{85}Kr to the atmosphere during reprocessing operations, which result in an increased exposure to the world population of 5 billion people. As shown in Table IV E-11, the introduction of Pu recycle fuels may decrease U.S. population total body dose commitments by about 1%, largely because of a decrease in the amount of ^{14}C associated with Pu recycle fuel. During the reprocessing of Pu recycle fuel, increased releases of ^{106}Ru, ^{129}I, and transuranic radionuclides result in increased exposures to the GI tract, thyroid, and bone, respectively. However, these increases are, for the most part, offset by decreases associated with lower releases of ^{14}C. To put these exposure estimates into perspective, the world population receives about 500,000,000 person-rem of exposure each year from natural background radiation. A 2,000 MT/yr model fuel reprocessing plant processing Pu recycle fuel (89% UO_2 fuel + 11% MOX fuel) may cause a 0.006% increase in radiation exposure to the world population. The total body dose commitment added to the U.S. population of approximately 250 million persons would be about 18,600 person-rem. Compared to exposure from natural radiation, this would amount to a 0.07% increase in the total body dose commitment to the U.S. population.

In CHAPTER IV, Section J, Appendix E, Table IV J(E)-9 summarizes for each segment of the total LWR fuel cycle industry the estimated total body dose commitments to the population per gigawatt year of electrical energy produced. Here it is shown that the environmental plus occupational total body dose commitment to the U.S. population from the total LWR fuel cycle industry is expected to increase about 16% with the recycle of uranium only and to increase about 11% with recycle of both U and Pu when compared to the no recycle of U or Pu option. These increases result, for the most part, from the reprocessing of spent fuel. Table IV J(E)-9 also shows that much of the dose commitment attributable to reprocessing would be offset by reductions in the mining and milling industry. If spent fuel is reprocessed, fuel reprocessing would contribute about 19%, and LWR's about 33%, of the total body dose commitment to the U.S. population that is expected to result from the total LWR fuel cycle industry.

3.2 Normal Operations--Nearest Exposed Resident

It is assumed that the nearest resident lives about 1,300 meters (about 0.8 miles) from the plant's stack. Dose calculations represent a 50-year dose commitment to an individual living at this location for one year. See CHAPTER IV, Section J, Appendix A. Tables IV E-12 and IV E-13 show the estimated nearest resident's dose commitment from a 2,000 MT/yr model fuel reprocessing plant which reprocesses UO_2 fuel only or reprocesses 89% UO_2 fuel + 11% MOX fuel. The dose commitment to the total

Table IV E-9

POPULATION DOSE COMMITMENT FROM UO_2 FUEL

Basis: 2,000 MT/yr fuel reprocessing plant
33,000 MWd/MT fuel exposure, aged 160 days
100% release of ^{3}H, ^{14}C, and ^{85}Kr to air

				Person-Rem				
Nuclide	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
^{85}Kr	1,150	1,150	1,150	1,150	1,150	1,150	2,500	97,700
^{3}H	11,100	11,100	11,100	11,100	11,100	11,100	11,100	11,100
^{14}C	6,340	6,340	31,600	6,340	6,340	6,340	6,340	6,340
^{129}I, ^{131}I	57	41	31	31	31	14,800	33	31
Other F.P.	167	9,090	1,090	212	351	100	135	95
Uranium & Transuranics	25	2	1,120	124	104	---	31	1
U.S. Total	18,800	27,700	46,100	18,900	19,000	33,500	20,100	115,000
Foreign Total	12,200	12,200	39,100	12,200	12,200	12,200	12,200	449,000
World Total	31,000	39,900	85,200	31,100	31,200	45,700	38,400	564,000

Table IV E-10

POPULATION DOSE COMMITMENT FROM PU RECYCLE FUEL

Basis: 2,000 MT/yr fuel reprocessing plant
33,000 MWd/MT fuel exposure, aged 160 days
Average feed mix 89% UO_2 fuel + 11% MOX fuel
100% release of ^{3}H, ^{14}C, and ^{85}Kr to air

			Person-Rem					
Nuclide	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
^{85}Kr	1,100	1,100	1,100	1,100	1,100	1,100	2,400	93,700
^{3}H	11,200	11,200	11,200	11,200	11,200	11,200	11,200	11,200
^{14}C	6,070	6,070	30,400	6,070	6,070	6,070	6,070	6,070
^{129}I, ^{131}I	58	42	32	32	32	15,100	34	32
Other F.P.	165	9,760	1,070	213	368	100	137	96
Uranium & Transuranics	38	4	1,590	323	161	1	76	1
U.S. Total	18,600	28,200	45,400	18,900	18,900	33,600	19,900	111,000
Foreign total	11,700	11,700	37,600	11,700	11,700	11,700	17,500	430,000
World Total	30,300	39,900	83,000	30,600	30,600	45,300	37,400	541,000

Table IV E-11

CHANGE IN POPULATION DOSE COMMITMENT

Basis: Compared to UO_2 fuel

	Person-Rem							
Nuclide	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
^{85}Kr	-50	-50	-50	-50	-50	-50	-100	-4,000
^{3}H	-100	+100	+100	+100	+100	+100	+100	+100
^{14}C	-270	-270	-1,200	-270	-270	-270	-270	-270
^{129}I, ^{131}I	+1	+1	+1	+1	+1	+300	+1	-1
Other F.P.	-2	+670	-20	+1	+17	---	+2	+1
Uranium & Transuranics	+13	+2	+470	+200	+57	+1	+45	--
U.S. Total	-200	+500	+700	---	-100	+100	-200	-4,000
Foreign Total	-500	-500	-1,500	-500	-500	-500	-800	-19,000
World Total	-700	---	-2,200	-500	-600	-400	1,000	-23,000

Table IV E-12

DOSE COMMITMENT TO NEAREST RESIDENT, UO_2 FUEL

Basis: 2,000 MT/yr fuel reprocessing plant
33,000 MWd/MT fuel exposure, aged 160 days
100% release of ^{3}H, ^{14}C, and ^{85}Kr to air
$\frac{X}{Q} = 3.7 \times 10^{-8}$ sec-m^{-3}

mrem/year

Nuclide	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
^{85}Kr	0.43	0.43	0.43	0.43	0.43	0.43	0.93	36.3
^{3}H	4.22	4.22	4.22	4.22	4.22	4.22	4.22	4.22
^{14}C	1.58	1.58	7.88	1.58	1.58	1.58	1.58	1.58
^{129}I, ^{131}I	0.29	0.23	0.19	0.19	0.19	57.7	0.20	0.19
Other F.P.	0.83	33.4	4.34	1.00	1.50	0.56	0.78	0.56
Uranium & Transuranics	0.15	-	6.64	0.73	0.61	-	0.19	-
Total	7.50	39.9	23.7	8.15	8.53	64.5	7.89	42.9

Table IV E-13

DOSE COMMITMENT TO NEAREST RESIDENT, MOX FUEL

Basis: 2,000 MT/yr fuel reprocessing plant
33,000 MWd/MT fuel exposure, aged 160 days
Average feed mix 89% UO_2 fuel + 11% MOX fuel
100% release of ^{3}H, ^{14}C, and ^{85}Kr to air
$\frac{X}{Q} = 3.7 \times 10^{-8}$ sec-m^{-3}.

	mrem/year							
Nuclide	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
^{85}Kr	0.41	0.41	0.41	0.41	0.41	0.41	0.89	34.8
^{3}H	4.27	4.27	4.27	4.27	4.27	4.27	4.27	4.27
^{14}C	1.51	1.51	7.57	1.51	1.51	1.51	1.51	1.51
^{129}I, ^{131}I	0.30	0.23	0.19	0.19	0.19	58.8	0.20	0.19
Other F.P.	0.83	35.9	4.27	1.01	1.57	0.56	0.79	0.57
Uranium & Transuranics	0.23	0.02	9.67	1.16	0.95	-	0.30	-
Total	7.55	42.4	26.4	8.55	8.90	65.6	7.96	41.4

body of the nearest resident exposed to the reprocessing plant's effluents would increase about 7.6% compared to exposure received from natural radiation. Reprocessing of Pu recycle fuel would add approximately 11% more to the bone dose commitment that would result from reprocessing only UO_2 fuels, an increase from 23.7 mrem/yr to 26.4 mrem/yr.

3.3 Ground Contamination

The total amount of depositable fission products emitted from the reprocessing plant does not change significantly when mixed oxide fuel is introduced. Table IV E-6 shows that cesium and strontium, two elements that are of interest, either do not change (^{137}Cs) or decrease (^{90}Sr) when mixed oxide fuel is reprocessed. Iodine-129 increases by 4%. Transuranics released from the reprocessing plant might increase from 0.14 αCi/yr without plutonium recycle to 0.26 αCi/yr with Pu recycle. In the case of Pu recycle, the 40 year maximum accumulation of transuranic alpha activity from reprocessing plant effluents may be 2×10^{-9} $\alpha Ci/m^2$, with 10% in the top centimeter of soil. This concentration is of the order of 0.01% of the proposed allowable standard.[23] With Pu recycle, the maximum accumulation of alpha activity will be approximately equal to the amount that has been accumulated to date from fallout.

3.4 Occupational Exposure

Confinement of radioactive material is the primary method for radiological protection of plant personnel, shielding provides the primary means of protecting personnel from radiation. To ensure adequacy, shielding design for existing plants is conservatively based on high burnup Pu recycle fuel and potential abnormal process conditions.

To determine occupational exposure, plant operation and laboratory work force exposures were estimated, based on exposure conditions set by design bases, the personnel needed to carry out various operations, and the types of activity and associated occupancy time in various work zones. In general, facilities were divided into access areas, based on the work to be performed and the associated occupancy requirements: normal access nonradiation zone (0.1 mr/hr), normal access work zone (1.0 mr/hr), and limited access work zone (10 mr/hr). On the assumption that the average exposure of maintenance personnel and other service personnel, except administrative, would equal the average exposure of operating personnel, the estimated average annual occupational exposures for a model reprocessing plant are the values shown in Table IV E-14.

The occupational exposure estimates are based on normal operations and include relatively minor incidents. They do not include those exposures that might result from steps that would be taken to decontaminate an area in the event of a major accident, nor do they include exposures that might be associated with decontamination of facilities and equipment related to repair, modification, or replacement of process components or systems after unexpected equipment failures.

Table IV E-14

AVERAGE ANNUAL OCCUPATIONAL EXPOSURE

Facility	Number of Persons*	Person-Rem
Separations	200	400 4
UF_6 Conversion	100	150
Pu Conversion	225	400-500**
Waste Solidification and Packaging	150	300
Total	675	1250-1350

*Does not include administrative and support personnel who are not engaged in activities related to handling radioactive materials.

**Allowance for exposure related to handling a greater quantity of Pu associated with Pu recycle.

In general, the hand and forearm exposure of personnel working in the PuO_2 conversion facility will be less than 30 and 15 rem per year, respectively.* Because process systems are behind shielding and are operated remotely, the increase in transuranic radionuclides associated with Pu recycle is not expected to have any significant effect on occupational exposure. However, the increased amount of ^{238}Pu and ^{241}Pu, and the total amount of plutonium, associated with Pu recycle may result in increased exposure to those operating personnel in the PuO_2 conversion facility who may physically handle plutonium in glove boxes, etc. It is assumed here that about 75% of the occupational exposure associated with the operation of the PuO_2 conversion facility is directly or indirectly related to the Pu throughput.

3.5 Accidents

The reprocessing plant complex can be broken down into four major processing areas:

- Separation of irradiated fuel to uranyl nitrate, plutonium nitrate, and various waste streams
- Conversion of plutonium nitrate to PuO_2
- Conversion of uranyl nitrate to UF_6
- Waste solidification and handling.

Because about the same amount of uranium is converted to UF_6 with or without Pu recycle, there is no change in the safety considerations related to the operation of the UF_6 plant. Accident evaluations related to a UF_6 plant can be found in the Draft Environmental Statement for the BNFP UF_6 Facility.[24]

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*The limit specified in 10 CFR Part 20.101(a) for hands and forearms is 18-3/4 rem per calendar quarter.

Plutonium recycle may affect the offsite consequences of accidents that might occur in the reprocessing plant because the isotope composition of plutonium changes with plutonium recycle. The effect of Pu recycle on accident effects from either separations or PuO_2 conversion are herein assessed in relation to hypothetical upper level* accidents that might occur in these facilities. The average composition of plutonium will be nearly the same in the separation sections and in the PuO_2 conversion part of the plant. However, because of smaller cells and equipment, accidents postulated to occur in the PuO_2 conversion facility would have less consequences than those postulated for the separations facility.

At separations facilities and PuO_2 conversion facilities, operations in which equipment failures or accidents could disperse significant amounts of radioactive contaminants are performed within process cells or buildings. These structures are designed to maintain adequate confinement capability even in the event of accidents or natural phenomena much more severe than have been experienced for such facilities or in the locality of the plant. It is expected that some equipment failures will occur during the life of these facilities. Accordingly, monitors are provided to detect process or equipment failures and either activate corrective action or signal a need for corrective action. Moreover, these facilities are designed to cope with failures or accidents; cells, buildings and equipment can be decontaminated for repair or replacement of equipment. A ventilation system assures that contaminated air from any inadvertent releases within the facility will be routed through high efficiency filters that remove airborne radioactive particulates before the ventilation air is discharged to the stack. In summary, separation facilities and PuO_2 conversion facilities are designed to assure adequate margins of safety to prevent accidents, cope with potential accidents, and mitigate the consequences of accidents if they should occur as a result of multiple failures of systems or procedures.

A wide spectrum of credible accidents and consequences is analyzed during a safety review. Although major equipment failures or spills of radioactive materials within the facility might disrupt operations and cause shutdown for cleanup and repair, it is not expected that such occurrences would result in the release of significant amounts of radioactivity to contaminate the offsite environs. Radioactive contamination would be confined within the process cell or within the process building, and little, if any, could escape to contaminate the vicinity beyond the plant's exclusion area. Only a few accidents involving radioactive materials have occurred in existing facilities, and none resulted in significant contamination beyond the immediate vicinity of the plant. Experience gained from these accidents has resulted in improved safety procedures and features so that the probability of similar occurrences is very small. Some examples of accidents that might occur in fuel separations or PuO_2 conversion facilities that would have little, if any, discernible impact on the offsite environment are listed, followed by a brief summary of the more significant potential accidents and an assessment of possible consequences irrespective of low probability of occurrence.

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*The review of postulated accidents for this assessment has concentrated upon the more serious accidents of the type that either have occurred or realistically can be postulated.

The following are examples of types of accidents that are postulated with respect to the analysis of irradiated fuel separation and PuO_2 conversion operations. Those listed are not expected to result in discernible contamination offsite, nor discernible exposure to the public, with or without plutonium recycle.

Abnormal Events	Potential Consequences
Separations:	
Fuel cask drop into cask unloading pool	Possible rupture of fuel pins and release of fission gas to atmosphere; contamination of storage pool water
Fuel element hung up in air during transfer to shear	Possible overheating and rupture of fuel pins and release of fission gases to atmosphere
Ignition of zirconium fines	Small fire of short duration--little, if any, damage
Rapid chemical reaction in dissolver	Vessel pressurized, seals blown, fission gases released to atmosphere; cell contaminated
Leak in recovered acid line	Contamination of cell or pipe trench in area of leak
Leak of any vessel or line confining radioactive material	Contamination of cell or pipe trench; transfer material to spare tank space, decontaminate cell and equipment, repair failure or replace equipment
Excessive entrainment of radioactivity in evaporator overheads	Contamination of recovered acid or condensate which must be recycled
Failure of an iodine scrubber	Reduced iodine removal efficiency of system; shut down plant until adequate efficiency restored
Filter failure	Detectable increase of radioactivity in stack effluent; shut down plant and replace filter
Loss of ventilation zone differential-pressure control	Possible migration or radioactivity from controlled area to limited access area; correct deficiency and decontaminate building area

Abnormal Events	Potential Consequences
Separations (Cont.)	
Solvent fire	May plug filter, contaminate cell and ventilation on exhaust ducts; could require extensive cleanup of cell and ducts, and replacement of filter, while plant shut down for repairs
UF_6 Conversion	
HF tank leak	Air concentration of toxic HF could be lethal to unprotected persons in vicinity of tank
PuO_2 Conversion	
Glove Failure	Local contamination of operating area
Window Failure	Local contamination of operating area
Equipment leak or release of plutonium to glovebox	Contamination of glovebox; transfer material to installed spare storage tank; decontaminate glovebox and repair equipment as necessary

Upper level accidents that may occur at separations facilities or PuO_2 conversion facilities include:

- Criticality
- High level radioactive waste concentrator or calciner explosion
- Plutonium product concentrator explosions

For the purpose of assessing the maximum impacts of upper level accidents involving plutonium recycle, it was assumed that an accident might occur during reprocessing of a batch of MOX fuel elements from a PWR, which during the latter part of the period might contain the third recycle of plutonium. (Less than 0.1% of the fuel reprocessed through the year 2000 would be of this type.)

The dose commitments from the hypothetical accidents that might occur in the fuel separations section and plutonium dioxide conversion section of the plant have been calculated for the maximum individual exposed and for the local population. These doses are given in Table IV E-15. It can be seen that Pu recycle may increase the dose commitment related to potential accidents. The maximum individual bone dose, the hypothetical plutonium concentrator explosion accident, is 19 mrem with Pu recycle. Without Pu recycle, the bone dose is 11 mrem. In the event of an accidential explosion, the estimated population bone dose is 213 person-rem with Pu recycle and 123 person-rem without Pu recycle. The maximum bone dose commitment to an offsite individual would be comparable to about 19% of the annual dose from natural background radiation. The amount of plutonium released would be equivalent to about 0.38% of the release limit, as set forth in 10 CFR Part 20.106(a), for an unrestricted area.

3.5.1 Criticality Accident

A criticality accident is unlikely in a separations facility or PuO_2 conversion facility, because equipment and process limitations are designed to prevent such incidents. Safe spacing is assured in storage basins by physically spacing the fuel elements in storage racks in a safe array. Process systems and controls are designed to prevent an unsafe condition. Nevertheless, a criticality accident of 10^{19} fissions is assumed. This yield is approximately an order of magnitude greater than the yield that has been experienced for Pu systems in past accidents.[25] It is further assumed that all noble gases and 50% of the halogens (or halides) are discharged from the plant stack. The dose commitments would be essentially the same for UO_2 fuel or MOX fuel.

The dose commitment to the thyroid (56 mrem to an individual; 629 person-rem to the population) dominates the dose commitment to other organs.

Table IV E-15

ENVIRONMENTAL IMPACT OF ACCIDENTS
FUEL REPROCESSING PLANT

Maximum Offsite Individual Dose Commitment* mrem

Accident	UO_2 Fuel	PWR MOX Fuel	Change
Criticality	56	56	0
Waste Concentrator Explosion	2.6	6.9	+ 4.3
Pu Evaporator Explosion	11	19	+ 8
Fire	1.6	13.5	+12

Local Population Dose Commitment person-rem

Accident	UO_2 Fuel	PWR MOX Fuel	Change
Criticality	629	629	0
Waste Concentrator Explosion	29	78	+ 49
Pu Evaporator Explosion	123	213	+ 90
Fire	18	152	+134

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*Critical organ for all accidents except criticality is bone. Critical organ for criticality accident is thyroid.

3.5.2 Waste Concentrator Explosion Accident

During operation of the separations facility solvent extraction process, solvent degradation products are generated and may be carried over into the waste streams. Under extreme conditions in early pilot plant operations, these nitrated degradation products (red oil) have caused concentrator explosions. However, red oil explosions can be prevented by installing equipment to eliminate the accumulation of organic materials in the waste, and by controlling the process temperature in the concentrator.

Concentrators are installed in highly shielded cells, having a volume of about 100,000 cu ft (3,000 m^3). In the unlikely event of an accident, the explosion is estimated to disperse about 150 gallons (600 liters) of high level radioactive waste solution into the cell in the form of a finely divided mist. A substantial portion of the mist would rain out or plate out on the cell surfaces. Droplets remaining in the air (10 mg/m^3) would be carried through the ventilation ducts to the high efficiency filters. Moisture separators upstream of the filters would knock out most of the mist.

The plant ventilation filters are located some distance from the separations plant process cells. Most of the explosive energy would be expended in destruction of the concentrator. Pressures developed by the explosion would be dampened by expansion into the cell and would be further attenuated in the ductwork. The final filters are not expected to be affected.

It has been estimated that plateout of the droplets on the cell walls and floors and removal by the filtration system will result in a reduction in the fraction of material released to 3.6×10^{-8}. Material leaving the final filter has been estimated to be 30.5 mg of high level radioactive waste solution in the form of an aerosol.

Table IV E-16 identifies those nuclides that would contribute significantly to the offsite dose, and summarizes the offsite bone dose commitment that might result from this hypothetical accident. The maximum offsite dose commitment to an individual is estimated to be about 2.6 mrem (bone) for UO_2 fuel, and about 6.9 mrem (bone) for MOX fuels. This is an increase of only about 3-7% of the exposure received from natural radiation.

3.5.3 Plutonium Concentrator Explosion Accident

The postulated explosion of a plutonium concentrator in the reprocessing plant is typical of upper level accidents by which plutonium could be released to a cell or glovebox area. Typically, the plutonium processing equipment tends to be smaller, and installed in smaller rooms (cells or gloveboxes) than the waste concentrator previously discussed. The release rate is derived by assuming that the room (cell or glovebox) atmosphere contains the same mass of aerosol (10 mg/m^3) as the atmosphere of the waste concentrator cell. For a 1,000 m^3 plutonium concentrator cell volume, the postulated accident would release about 2.2 mg of plutonium.

Table IV E-17 shows the radionuclide releases and the bone dose commitment to the maximally exposed offsite individual. It can be seen that the bone dose commitment from the postulated accident with Pu recycle would be higher than that without Pu recycle. However, the exposure to an individual offsite would be less than 11%-19% of that received from natural radiation.

3.5.4 Other Accidents

A range of other accidents is possible, but the accident evaluations show smaller effects than those tabulated above. Among these are:

- First cycle solvent fire
- Ion exchange resin (Pu cycle) fire
- Second cycle (Pu cycle) solvent fire
- Fuel element drop

For example, estimates have been made of the potential maximum individual dose commitment (offsite) from first and second cycle solvent fires. The estimated bone dose commitments are:

- Without Pu recycle = 1.6 mrem
- With Pu recycle = 13.5 mrem

Table IV E-16

WASTE CONCENTRATOR EXPLOSION EFFECTS

Nuclide	Half Life	Radioactivity Released in Accident mCi		Accident Bone Dose Contribution mrem	
		UO_2 Fuel	MOX Fuel	UO_2 Fuel	MOX Fuel
Pu		0.02	0.15	0.01	0.06
^{241}Am	458y	0.007	0.05	0.02	0.17
^{242}Cm	162d	0.76	12.3	0.04	0.60
^{244}Cm	18y	0.05	3.25	0.07	4.38
^{90}Sr	29y	2.80	1.62	1.92	1.14
^{106}Ru	1y	2000	3400	0.02	0.04
^{144}Ce	284d	27	23	0.04	0.03
Other F.P.		1.5	1.5	0.48	0.48
			Total	2.6	6.9

Table IV E-17

PLUTONIUM PRODUCT EVAPORATOR EXPLOSION EFFECTS

Plutonium Isotope	Half-Life	Radioactivity Released in Accident mCi		Accident Bone Dose Contribution mrem	
		UO_2 Fuel	MOX Fuel	UO_2 Fuel	MOX Fuel
238	86y	1.02	2.11	5.94	12.27
239	2.4×10^4y	0.08	0.04	0.51	0.28
240	6,540y	0.11	0.13	0.76	0.88
241	13y	2.94	4.01	3.98	5.42
			Total	11.2	18.9

4.0 ENVIRONMENTAL EFFECTS IN THE YEAR 2000

Table IV E-5 shows the amount of spent reactor fuel expected to be discharged from LWR's and the prospective growth of the spent fuel reprocessing industry if these fuels are reprocessed. Because most of the spent fuel would be reprocessed in plants that are not yet designed, the prospective growth of the fuel reprocessing industry is shown as the number of equivalent 2,000 MT/yr model plants. Future plants are likely to have larger throughput capacity than those currently constructed. Therefore, the number of actual plants in operation is likely to be less than 5 equivalent model plants indicated in Table IV E-5. The radiological impacts from 5 model plants, about year 2000, are summarized in Table IV E-18.

In general, the growth of Pu recycle may result in a small decrease in the population dose commitment relative to reprocessing only UO_2 fuel, due to small changes in the isotopic content of radioactive effluents released from the reprocessing plants. By about the year 2000, as a result of spent fuel reprocessing, the annual average U.S. population dose commitment to the total body would increase only about 0.34% compared to natural background radiation. It is thus concluded that the reprocessing of spent irradiated fuel containing recycle U and Pu would not significantly increase environmental effects associated with reprocessing operations.

5.0 ENVIRONMENTAL IMPACT OVER 26 YEARS

If the LWR spent fuel discharges projected in Table IV E-5 were reprocessed and the recovered plutonium was recycled as fuel in LWR's, the growth of the U.S. fuel reprocessing industry might result in the overall environmental impact indicated in Table IV E-19, which summarizes the cumulative impact over the period from the year 1975 through 2000. The estimated amounts of various significant radionuclides, corrected for decay, that might accumulate in the environment are given as of the end of the year 2000. Compared to the total dose received from natural radiation, the prospective U.S. spent fuel reprocessing industry might contribute an additional integrated total body dose commitment of about 0.02% to the world population over the period from the year 1975 through 2000. The additional integrated total body dose commitment to the U.S. population over the same period would be about 0.2% of that

Table IV E-18

REPROCESSING INDUSTRY'S RADIOLOGICAL EFFECTS

(About year 2000 - Equivalent to 5 Model Plants)

RADIOACTIVE EFFLUENTS

(Curies)

Radionuclide	UO_2 Fuel	18% MOX + 82% UO_2	Difference
^{3}H	5.3×10^6	5.4×10^6	$+0.1 \times 10^5$
^{14}C	6,970	6,470	-500
^{85}Kr	117×10^6	109×10^6	-8×10^6
^{129}I	9.8	10.3	+0.5
Other F.P.	338	346	+8
Pu, Am, & Cm	7.3	12.9	+5.6

U.S. POPULATION DOSE COMMITMENT IN THOUSANDS OF PERSON-REM

Organ	UO_2 Fuel	18% MOX + 82% UO_2	Difference
Total Body	96	95	-1
GI Tract	142	146	+4
Bone	236	230	-6
Liver	96	96	0
Kidney	96	97	+1
Thyroid	171	172	+1
Lung	103	102	-1
Skin	591	580	-11
Occupational Exposure	6.3	6.8	+0.5

Natural Radiation Exposure*	Person-rem (thousands)
World Population	560,000
U.S. Population	28,000

*Stated for comparison

received from natural radiation. Therefore, it is concluded that the reprocessing of spent irradiated fuel containing recycled U and Pu would not significantly increase environmental effects associated with reprocessing operations.

In general, the thermal discharges and uses of land, water, and electricity by the prospective spent fuel reprocessing industry are less than 0.5% of the totals for the entire LWR fuel cycle industry. See CHAPTER VIII, Appendix A. CHAPTER XI, Section 4.0, compares the effects of alternatives on environmental and economic factors.

Population dose commitments expected from each segment of the U.S. total LWR fuel cycle industry, for the three fuel cycle options, are summarized in CHAPTER IV, Section J, Appendix E. Over the years 1975 through 2000, if spent fuel is reprocessed to recover and recycle uranium only, the total body dose commitment to the world's population from the U.S. total LWR fuel cycle industry is expected to increase about 16%, when compared to the no recycle of U or Pu option. On the same basis, if spent fuel is reprocessed to recover and recycle U and Pu, the total body dose commitment to the world's population is expected to increase about 9%. About 20% of the total body dose commitment to the world's population that is expected to result from the total LWR fuel cycle industry in the United States would be attributable to reprocessing, if spent fuels were reprocessed to recover and recycle U only or both U and Pu. For comparison, the LWR's would cause about 30% of the world's population total body dose commitment that would be attributable to the total LWR fuel cycle industry in the United States, if U only or both U and Pu are recycled as fuel in LWR's.

Table IV E-19

FUEL REPROCESSING CUMULATIVE ENVIRONMENTAL IMPACT

(Years 1975 Through 2000)

Natural Resource Use	UO_2 Fuel	UO_2 + MOX Fuel
Land (acres)	1×10^4	1×10^4
Water (gallons)	2.02×10^{11}	2.04×10^{11}
Fossil Fuel		
Electrical Energy (MW-hr)	1.32×10^7	1.32×10^7
Equivalent Coal (tons)	4.80×10^6	4.80×10^6
Fuel Oil (gallons)	7.5×10^8	7.5×10^8
Propane Gas (gallons)	6.9×10^6	6.9×10^6
Effluents		
Chemicals (MT)		
Gases*		
SO_x	1.78×10^4	1.78×10^4
NO_x	7.20×10^4	7.20×10^4
Hydrocarbons	1.56×10^3	1.56×10^3
CO	1.40×10^3	1.50×10^3
Particulates	1.90×10^3	1.90×10^3
Fluorides	1.56×10^2	1.56×10^2
Liquids		
SO_4	50	50
NO_3	-	-
Cl	290	290
Na + K	50	50
Fe	-	-
Thermal (Btu)		
Fossil Fuel	1.56×10^{14}	1.56×10^{14}
Decay Heat	0.92×10^{14}	0.97×10^{14}
Total	2.48×10^{14}	2.53×10^{14}

Table IV E-19 (Cont.)

FUEL REPROCESSING CUMULATIVE ENVIRONMENTAL IMPACT
(Years 1975 through 2000)

Effluents (Cont.)

Radionuclides** (curies)	UO_2 Fuel	UO_2 + MOX Fuel
^{3}H	4.20×10^{7}	4.26×10^{7}
^{14}C	7.83×10^{4}	7.50×10^{4}
^{85}Kr	8.76×10^{8}	8.41×10^{8}
^{90}Sr	16	15
^{106}Ru	75	83
^{129}I	110	114
^{131}I	8	8
Other F.P.	57	57
U	0.13	0.16
Pu, α	2.7	3.8
Pu, β	58	76
^{241}Am	0.6	0.8
^{243}Am	3.71×10^{-3}	8.88×10^{-3}
^{242}Cm	0.3	0.9
^{244}Cm	0.3	1.1

Population Dose Commitment*** (person-rem)		
Occupational	7.20×10^{4}	7.77×10^{4}
U.S. Population	1.08×10^{6}	1.07×10^{6}
Foreign Population	7.0×10^{5}	6.8×10^{5}
Total	1.86×10^{6}	1.83×10^{6}

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*As gases, excluding those from burning coal for power generation.

**Radionuclides, corrected for decay, remaining in the environs at the end of year 2000.

***Cumulative dose commitment from natural radiation over 26-year period is 1.2×10^{10} person-rem. Dose commitment added to the world population equals approximately 0.02% of exposure to natural radiation. Dose commitments expected from each segment of the U.S. total LWR fuel cycle industry, and for the three fuel cycle options, are summarized and compared in CHAPTER IV, Section J, Appendix E, Tables J(E)-1 through -8.

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CHAPTER IV

ENVIRONMENTAL IMPACT DUE TO THE IMPLEMENTATION OF PLUTONIUM RECYCLE

SECTION F

SUPPORTING URANIUM FUEL CYCLE

VOLUME 3

CHAPTER IV

ENVIRONMENTAL IMPACT DUE TO THE IMPLEMENTATION OF PLUTONIUM RECYCLE

SECTION F

SUPPORTING URANIUM FUEL CYCLE

CHAPTER IV
Section F
SUPPORTING URANIUM FUEL CYCLE

SUMMARY

General

A number of the steps involved in the projected mixed oxide fuel cycle for use with light processing operations. These steps, which will be carried out following essentially the same procedures as in the existing uranium fuel cycle for light water reactors (LWR), include

- Uranium mining
- Uranium milling
- Conversion of the mill produced yellowcake (U_3O_8) to uranium hexafluoride (UF_6)
- Enrichment of uranium in the fissionable isotope uranium-235
- Conversion of enriched uranium hexafluoride to uranium dioxide (UO_2)
- Fabrication of uranium fuel assemblies

Because none of these operations will at any time involve the processing of plutonium containing materials (except in trace quantities in recycled uranium), no adverse differential environmental impacts will occur in these areas from plutonium recycle in LWR's. The recycle of uranium only or both uranium and plutonium would decrease the need for uranium; thus, recycle would decrease the environmental effects from the operations listed.

The natural uranium requirements for an LWR of 1,000 MWe capacity designed to operate on low enriched uranium fuel are about 550 to 625 tons* of U_3O_8 for initial fuel loading and about 200 tons of U_3O_8 for annual refueling. Projections and estimates of fuel management schemes for use of recycled uranium or both recycled uranium and plutonium in LWR's are shown in Table IV F-1. These projections and estimates indicate that over the time period 1975 through 2000, recycling the uranium recovered from spent light water reactor fuel would reduce projected uranium mining and milling requirements

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*Throughout Section F, unless prefaced by "metric" or followed by MT, all tons are short tons. This practice is employed in this section to be consistent with the milling industry, which uses this unit of measure, along with the ERDA NUFUEL Program printouts.

Table IV F-1

EFFECTS OF URANIUM AND PLUTONIUM RECYCLE ON SUPPORTING DOMESTIC URANIUM FUEL CYCLE INDUSTRY

PROJECTED CUMULATIVE PRODUCTION REQUIREMENTS FOR PERIOD 1975 THROUGH 2000*

Fuel Cycle Step	No U or Pu Recycle	With U Recycle	With U and Pu Recycle
Uranium Mining** Millions MT Ore	1,603	1,435	1,245
Uranium Milling Thousands ST U_3O_8	1,600	1,431	1,242
UF_6 Conversion Thousands MTU	1,210	1,082	916
Enrichment Thousands MT SWU***	608	613	523
Fuel Fabrication Thousands MTU	189	189	163†

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*U.S. NRC Projections of Domestic Nuclear Fuel Cycle - Based on U.S. ERDA NUFUEL Computer Program, Forecast of Nuclear Power, April 1976 (Low-Growth Projection).

**Calculated assuming ore assay of 0.10% U_3O_8 and 90.5% mill recovery.

***Enrichment production rates are expressed in terms of separative work units (SWU). Enrichment plant capacity is a function of both the quantities of material fed to and withdrawn from the process and the isotopic assay of each of these streams (not throughput quantities alone as is the case with many other processes). The SWU quantifies an enrichment effort by weighting the importance of plant flows and their assays. Thus, for a uranium enrichment plant, the SWU capacity defines the overall effort expendable by the plant in separating a quantity of feed of a given assay into two components, one having a higher percentage of uranium-235 (the product) and one having a lower percentage (the tails). Separative work is expressed in mass units, usually kilograms, to give it the same dimensions as material quantities.

For example, for a plant to produce a 1 kg mass of 3% uranium-235 enriched uranium from natural uranium feed (0.711% uranium-235) with the waste stream assay at 0.30 uranium-235, 3.4 SWU are needed. To produce a 1 kg mass of 4% uranium-235 under similar conditions, 5.3 SWU are required.

†Does not include about 24,000 MTU used in mixed oxide fuel fabrication.

by about 10.5 percent. Recycling recovered plutonium in addition to the uranium would yield approximately an additional 11.8% decrease in mining and milling requirements compared with the no-recycle option for an overall potential reduction of about 22.3 . Similarly, the estimated reductions in UF_6 conversion requirements from uranium mill concentrates are about 10.6% with recycle of recovered uranium and an additional decrease of about 13.7% (based on the no-recycle option) is projected with recycle of both uranium and plutonium. The overall savings in UF_6 conversion requirements resulting from uranium and plutonium recycle is estimated at 24.3% of the no recycle requirements.

Recycling only recovered uranium, which normally contains a slightly higher concentration of ^{235}U than natural uranium but also contains a small quantity of ^{236}U, increases the separative work required in the enrichment step by about 0.9% of the no recycle requirements. The total effect on separative work demand to meet domestic requirements of recycling both plutonium as well as uranium over the period 1975 through 2000, however, is an overall decrease of 85,000 MT SWU, which represents about 14% of the no recycle requirements. Although no reduction in UO_2 fuel fabrication requirement results from recycling the uranium recovered from spent LWR fuel, recycling both the uranium and plutonium decreases the load on the enriched UO_2 fuel fabrication industry by about 13%.

Environmental Effects

Reduced processing requirements lead to correspondingly reduced impacts on the environment. In those instances where the reduced requirement might result in a decrease in the total number of facilities needed to produce the projected output, additional benefits would be expected to accrue from reduced construction requirements. Assuming that the average ore grade continues to decrease, as is projected, a further benefit that might be derived is the extension of time that higher grade ore is available to the industry. This benefits the environment in that the higher ore grade requires fewer tons to be mined and milled to meet industry requirements thereby reducing the environmental impacts of these operations. A possible added benefit of stretching out the higher grade uranium reserves is the additional time provided for developing improved milling processing techniques. As indicated in CHAPTER IV, Sections A and B, CHAPTER VIII, Appendix A, and CHAPTER XI, Section 4.0, the reduction in environmental impact of the supporting uranium fuel cycle resulting from uranium only or uranium and plutonium recycle will be offset to an extent by the impacts of the recycle related facilities such as reprocessing and MOX plants.

The most significant cumulative industrywide decreases in the environmental impacts of the supporting uranium fuel cycle resulting from recycling uranium or both uranium and plutonium in LWR's will be summarized. A detailed table showing additional estimated environmental effects of the projected uranium fuel cycle covering the period 1975 through the year 2000 with no recycle, with recycle of uranium only, and with recycle of both uranium and plutonium is presented as Appendix A to this section.

Additional effects produced by recycling recovered uranium or both uranium and plutonium to LWR's include the conservation of manpower, materials, and money resulting from the reduced production requirements of the supporting uranium fuel cycle. These impacts are discussed in CHAPTER XI as part of the cost-benefit analysis for this study.

SUMMARY OF SIGNIFICANT EFFECTS OF FUEL RECYCLE ON INTEGRATED ENVIRONMENTAL IMPACTS OF SUPPORTING URANIUM FUEL CYCLE
FOR PERIOD 1975 THROUGH 2000

Uranium Mining Industry

- Reduction of mining facilities by about 745 underground mines and about 33 open pit mines due to uranium recycle and by about 1,645 underground and about 71 open pit mines due to decreased requirements resulting from recycling both uranium and plutonium

- Reduction of ^{222}Rn emissions by about 2,500,000 curies due to uranium recycle and by about 5,300,000 curies due to recycling both uranium and plutonium

- Reduction in ore production needed by about 168 million metric tons with uranium recycle and an additional reduction by 190 million metric tons with uranium and plutonium recycle; a total decrease of 358 million metric tons, more than 50 times the total uranium mining industry production of ore in the year 1974

- Decreased disturbance of land by about 45,000 acres with uranium recycle and by about 97,000 acres (more than 150 square miles) with recycle of both uranium and plutonium

- Reduction in maximum critical organ (kidney) dose commitment to the general population from about 11.3×10^6 person-rem with no recycle to about 10.1×10^6 person-rem with uranium recycle and to about 8.8×10^6 person-rem with both uranium and plutonium recycle

- Decrease in whole body occupational dose commitment from about 1,140,000 person-rem with no recycle to about 1,020,000 person-rem with uranium recycle and to about 900,000 person-rem with both uranium and plutonium recycle

- Reduction in maximum critical organ (lung) occupational dose commitment by about 670,000 person-rem due to uranium recycle and by an additional decrease of about 750,000 person-rem with recycle of both uranium and plutonium, for a total of 1,420,000 person-rem.

Uranium Milling Industry

- Reduction of the total number of uranium mills needed in the year 2000 from about 109 with no recycle to about 95 with uranium recycle and to about 77 with recycle of both uranium and plutonium

- Conservation for future use of about 170,000 tons of U_3O_8 with uranium recycle and of about 360,000 tons of U_3O_8 with both uranium and plutonium recycle; a total potential reduced requirement more than 30 times the total annual production of the uranium milling industry in the year 1974

- Reduction of land committed to permanent use for storage of mill tailings by about 3,900 acres with uranium recycle and an additional reduction of about 5,000 acres with recycle of both uranium and plutonium, for a total of 8,900 acres

- Decrease in the release of ^{222}Rn from active mill tailings areas of about 470,000 curies with uranium recycle and a total reduction of about 990,000 curies with recycle of both uranium and plutonium

- Reduction in the mill tailings generated of about 170 million metric tons containing about 570,000 curies of activity with uranium recycle and an additional decrease of about 190 million metric tons containing about 630,000 curies of activity with uranium and plutonium recycle, for a total of 360 million metric tons containing about 1,200,000 curies

- Reduction in maximum critical organ (kidney) dose commitment to the general population of about 220,000 person-rem with uranium recycle and an additional decrease of about 250,000 person-rem with recycle of both uranium and plutonium, for a total of 470,000 person-rem

- Decrease in whole body occupational dose commitment from about 560,000 person-rem with no recycle to about 500,000 person-rem with uranium recycle and to about 440,000 person-rem with both uranium and plutonium recycle

- Decrease in occupational critical organ (lung) dose commitment by about 500,000 person-rem with uranium recycle and by an additional 500,000 person-rem with uranium and plutonium recycle, for a total of 1,000,000 person-rem.

UF_6 Conversion Industry

- The total environmental impact of the UF_6 conversion industry is quite small compared with several other steps in the supporting uranium fuel cycle. Consequently the potential benefits of recycling uranium or both uranium and plutonium, although producing reductions in environmental impacts of about 10.6 and 24.3%, respectively, from the no recycle option, represent an insignificant percentage of the impact of the total supporting uranium fuel cycle.

Uranium Enrichment Industry

- Reduction in the number of new 8.75 million SWU capacity enrichment facilities needed about the year 2000 from a total of 3 with no recycle or uranium recycle only to 2 with recycle of both uranium and plutonium. Eliminating the need for this one facility would represent a capital investment saving of about $3 billion.

- Increase in the estimated electrical power requirements over the period 1975 through 2000 from about 1,340 million megawatt hours (MWh) with no recycle to about 1,346 million MWh with uranium recycle and a decrease to about 1,223 million MWh with both uranium and plutonium recycle if the first new enrichment plant uses the gaseous diffusion process and subsequent ones use the gas centrifuge process. Corresponding electrical power requirements with all new plants using the gas centrifuge process are 1,167, 1,170, and 1,107 million MWh, respectively. Concomitant with the reduction in electrical power requirements, there is an overall potential reduction in coal consumed from 1975 through 2000 by about 26 million metric tons as a result of recycling both uranium and plutonium with the first new enrichment facility a gaseous diffusion plant and by about 13 million metric tons with all new plants using the gas centrifuge process.

- Reduction of gaseous pollutants associated with the decreased coal consumption. This includes reducing the 26 year period total emission of sulfur oxides by about 340,000 metric tons and the nitrogen oxide emissions by about 270,000 metric tons with recycle of both uranium and plutonium, if the first new enrichment plant uses the gaseous diffusion process. The potential reductions in pollutant emissions are about 50% of the above if all new facilities are gas centrifuge plants.

UO_2 Fuel Fabrication Industry

As indicated in Appendix A, the total environmental impact of the UO_2 fuel fabrication industry is negligible in comparison with the preceding steps in the supporting uranium fuel cycle. Consequently, the decrease in environmental impact resulting from the approximately 13% reduction in UO_2 fuel fabrication requirements obtained with uranium and plutonium recycle is not considered to be significant.

1.0 THE URANIUM MINING INDUSTRY

1.1 Current Status

The domestic uranium mining industry produced the equivalent of about 12,600 tons of U_3O_8 in 1974 with two major methods employed in the ore recovery process. Open pit mining accounted for 58% of the ore produced from 31 facilities ranging in production up to 700,000 metric tons (MT) of ore per year and an additional 40% was produced from 123 underground mines ranging up to 230,000 MT/yr of ore.[1] Open pit mining is normally used when the ore body lies under relatively friable material at depths up to a few hundred feet; underground mining is employed when the ore body is at depths greater than about 400 feet or when it lies under rock strata requiring a great deal of blasting. Other methods of uranium recovery, a small fraction of the nation's total capacity, include heap leaching from low grade ores, recovery from mine water or mill raffinates, and in situ leaching operations. Although only about 2% of the U_3O_8 was produced by all of these methods in the year 1974, there has been a recent increased interest in the utilization of in situ solution mining for producing U_3O_8 from marginal ores, as well as recovery of byproduct uranium from wet process phosphoric acid and copper dump leach liquor.

About 42% of all uranium ore was produced in New Mexico and about 35% in Wyoming, in 1974; most of the remainder came from Colorado, Utah, Texas, and Washington.[2] The assay averaged about 0.18% or 3.6 pounds of U_3O_8 per ton of ore.

1.2 U.S. Uranium Supply

Estimates of ore reserves and potential resources have been made by the AEC since the 1950's. This effort will be continued by the Energy Research and Development Administration (ERDA). Uranium resource figures published by the AEC prior to the year 1975 consisted of estimates of ore reserves for the known uranium districts and estimates of potential resources in general proximity to those districts. The resource base was confined to the western United States, where most known districts are located, chiefly in the Colorado Plateau, Wyoming Basins, and South Texas.

In 1973, the resource assessment program was expanded to encompass the entire United States, and its goals and missions were redefined. The National Uranium Resource Evaluation program (NURE) was established at that time to evaluate uranium resources and to identify potentially favorable areas throughout the United States.[3]

A preliminary evaluation has been completed by ERDA geologists and the results are scheduled to be published in 1976. A more comprehensive NURE report, based on the results of greatly expanded uranium related investigations during the next four years, is targeted for early 1980. The major objective of NURE is a more complete evaluation of the country's potential uranium resources, to help assure timely discovery and utilization.

According to the conservative low growth projections assumed for the current study, the estimated total U.S. uranium requirement to fulfill the needs of the nuclear power industry from the year 1975 through the year 2000, with no recycle of uranium or plutonium, is about 1,600,000 tons of U_3O_8 as yellowcake produced by the mills.

This requirement is reduced to about 1,430,000 tons with uranium recycle and to about 1,240,000 tons with recycle of both uranium and plutonium recovered from spent reactor fuel. As shown in Table IV F-2, the no recycle requirement is comparable to the combined total \$30 forward cost reserves and probable resources, and the U and Pu recycle requirements are roughly equivalent to the combined total of all \$10 forward cost estimated reserves and resources. The indicated estimates of reserves are derived from drill hole and other engineering data provided by the uranium industry and the estimates of potential resources are based on a preliminary assessment resulting from the initial phase of the National Uranium Resource Evaluation (NURE).[3] Material designated as "reserves" is in ore known to exist, while "potential resources" represent the additional uranium that, hopefully, will be developed in the future. As indicated, the estimates of potential resources are placed in three classes (probable, possible, and speculative) to reflect the degree of reliability of the quantities tabulated. Reliability is greatest in the "probable" class, where there has been extensive exploration and where mines have been developed, thus defining ore habits, the nature and extent of the favorable host rocks, etc. Reliability is least in the "speculative" category, where areas of favorability must be inferred solely from literature survey, geological reconnaissance of formation outcrops, and examination of the logs and cuttings of wells drilled for petroleum exploration or other purposes.

Estimated reserves at a forward cost of \$30 per pound U_3O_8, excluding byproduct sources, are about 640,000 tons and the total estimated reserves and potential resources in the United States are estimated at about 3,700,000 tons of U_3O_8. These resources are considered to be in deposits such as those currently being mined, primarily tabular pods in sandstones, conglomerates, and veins sometimes referred to as "conventional" deposits. There are also large quantities of uranium deposits in limestone, shale, lignite and phosphate rock, and disseminated deposits in granitic, metamorphic, and volcanic rock (unconventional ores), but the uranium in these materials occurs in low concentrations (<100 parts per million) as shown in Figure IV F-1. Cost of production from such sources would be much higher than from conventional ores, and very large tonnages would need to be mined and milled to produce any substantial amounts of uranium. As a consequence, the environmental impact of using "unconventional" ores would also be significantly greater, and because the uranium requirements through the end of the century can apparently be met with "conventional" ore, the use of "unconventional" ores is not assessed in detail in this analysis.

The conventional uranium deposits are located in the western part of the United States, principally in the States of Arizona, Colorado, New Mexico, Texas, Utah, Washington, and Wyoming. Figure IV F-2 shows the significant uranium areas of the United States. Most conventional uranium deposits are small, containing less than 100 tons of U_3O_8. A relatively few large deposits contain the bulk of the reserves since about 93 of the total \$30 forward cost reserves are located on about 11% of the total number of uranium properties.[6]

Table IV F-2

ESTIMATED U.S. URANIUM RESOURCES[4]

U_3O_8 Forward* Cost Up to: $/lb	Percent of U_3O_8	Reserves	Potential Resources**			Total
			Probable	Possible	Speculative	
		Cumulative Thousands of Tons U_3O_8				
10	0.17	270	440	420	145	1,275
15	0.13	430	655	675	290	2,050
30	0.08	640	1,060	1,270	590	3,560
Byproduct						
1975-2000***	-	780	-	-	-	3,700

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*Forward costs are those operating and capital costs yet to be incurred at the time an estimate is made. Profit, cost of money, and sunk costs for property acquisition, exploration, mine development, and mill construction are not included. The various forward costs are independent of the market price at which the estimated quantities of U_3O_8 would be sold. Price is determined by the market supply-demand relationship at any given time and would be expected to be significantly higher than the indicated forward costs.

**"Probable" potential resources are those estimated to occur in known uranium districts and are further postulated to be

- in extensions of known deposits
- in new deposits within trends or areas of mineralization that have been identified by exploration.

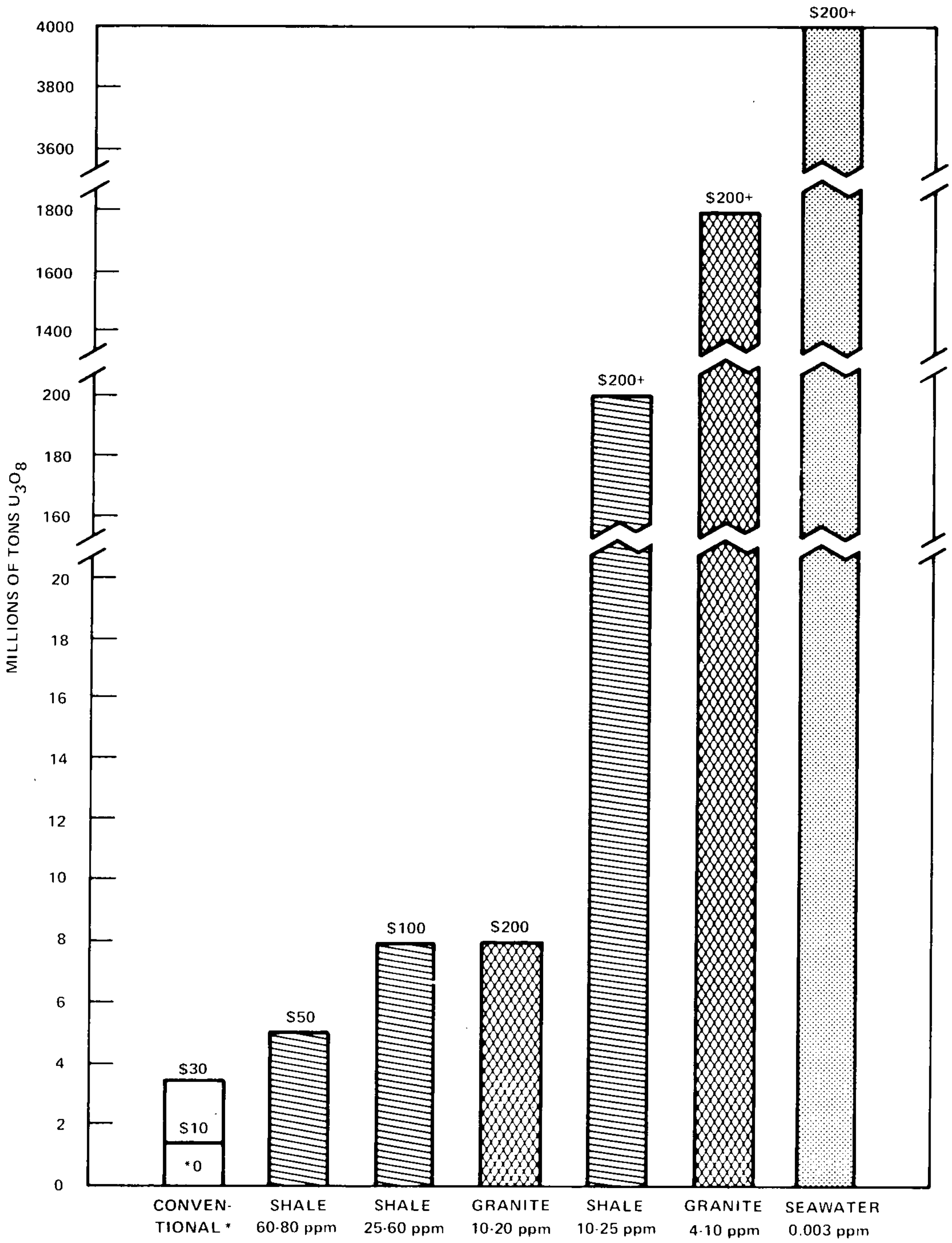
"Possible" potential resources are those estimated to occur in new deposits in formations or geologic settings productive elsewhere

- within the same geologic province or subprovince under similar geologic conditions
- within the same geologic province or subprovince under different geologic conditions.

"Speculative" potential resources are those estimated to occur in new deposits

- in formations or geologic settings not previously productive within a productive geologic province or subprovince,
- within a geologic province or subprovince not previously productive.

***Byproduct of phosphate and copper production.



* 800 - 1900 ppm GJO - 100 (75)

(See Reference 5)

Figure IV F-1 Estimated U. S. Uranium Resources in Various Formations and Concentrations

Figure IV F-2 Significant Uranium Areas of the United States

1.3 Foreign Uranium Supply[7]

Reasonably assured foreign resources at $15 per pound U_3O_8 (excluding the Communist Bloc) are currently estimated at 1,040,000 tons of U_3O_8 with estimated potential additional $15 per pound resources at 650,000 tons of U_3O_8. More than 75% of the former resources are located in Australia, Canada, South Africa, and southwest Africa and the remainder primarily in northern and central Africa and in Europe. Of the estimated additional resources, about 81% are located in Canada and Australia. At $30 per pound U_3O_8, the reasonably assured resources are estimated at 1,770,000 tons of U_3O_8 with about 82% located in Sweden, Australia, south and southwest Africa, Canada and Spain, and about 66% of the estimated additional $30 per pound U_3O_8 reserves of 1,040,000 tons are located in Canada and Spain. Total currently estimated foreign resources are presented in Table IV F-3.

Table IV F-3

ESTIMATED FOREIGN RESOURCES (EXCLUDING COMMUNIST BLOC)7

Thousand Tons U_3O_8

	Reasonably Assured	Estimated Additional
	$15/lb U_3O_8	
Australia	360	107
S & SW Africa	262	10
Canada	187	421
Niger	52	26
France	48	33
Algeria	36	--
Gabon	26	6
Spain	13	11
Argentina	12	18
Other*	42	15
Total (Rounded)	1,040	650
	$30/lb U_3O_8	
Sweden	390	--
Australia	360	100
S & SW Africa	343	44
Canada	216	545
Spain	134	138
France	71	53
Niger	65	39
Algeria	36	--
Argentina	22	48
Other**	134	69
Total (Rounded)	1,770	1,040

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*Includes Central African Republic, Japan, Mexico, Portugal, Sweden, Turkey, Yugoslavia and Zaire.

**Includes, in addition to *; Denmark, Finland, Italy, Korea and the United Kingdom.

Despite a possibly better supply position with respect to low cost ores than is currently evident, it seems unlikely that the United States can count on the availability of large amounts of low cost foreign uranium. As indicated in Figure IV F-3, steep increases in foreign requirements are expected to make heavy demands on the projected foreign production capability. Indeed, without an expansion of the foreign production capability or the utilization of higher cost ores, or both, foreign requirements are expected to exceed foreign production capability beyond 1983.

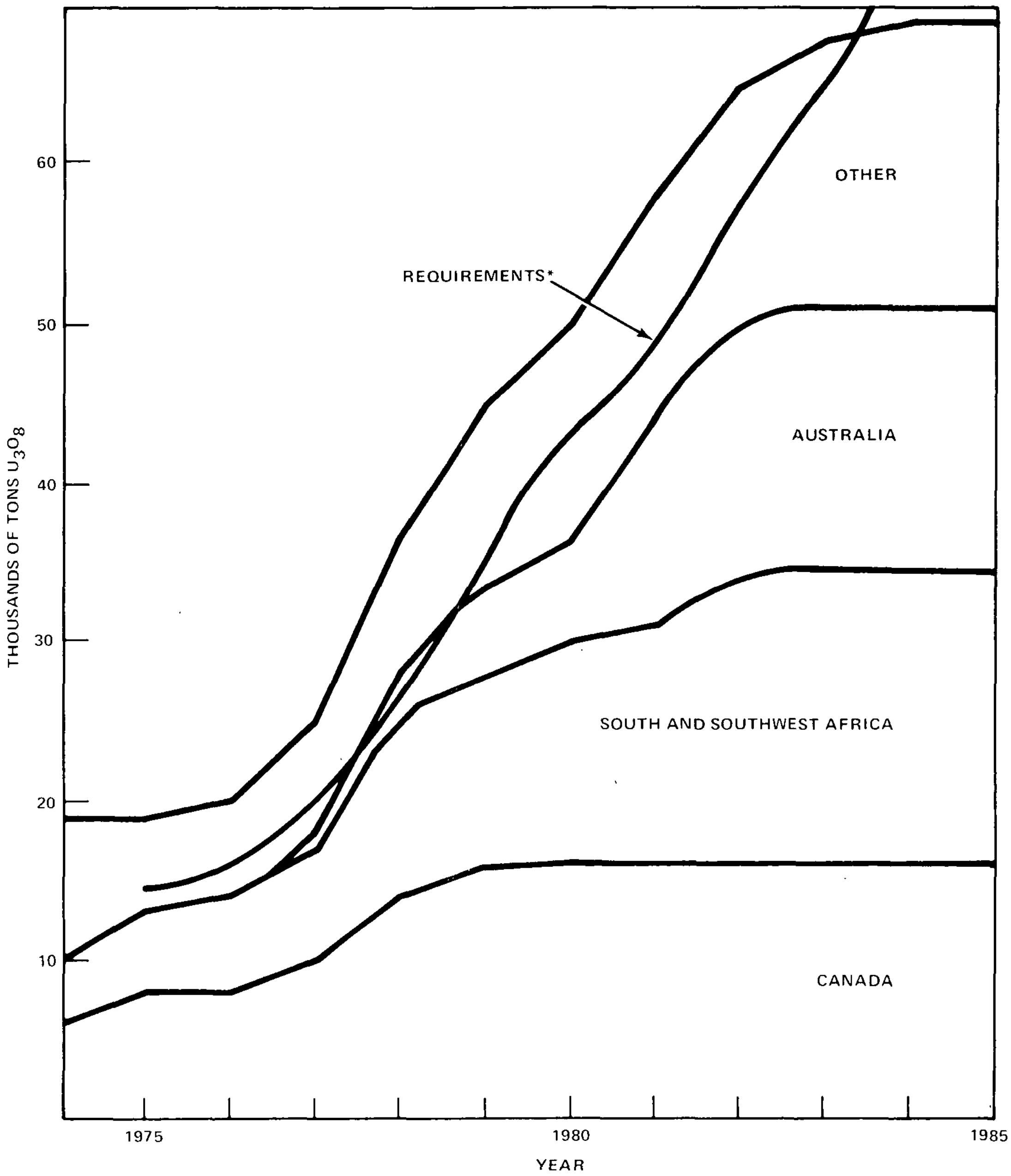
This picture is compounded by limitations on the production rates attainable, due to physical and economic constraints, and uncertainty as to foreign governments' policies regarding uranium production and supply to others. Although it is not possible to forecast accurately the extent to which foreign uranium will be available as a source for United States use over the long term, the present picture and projections lead to the conclusion that there will not be an assured availability of substantial quantities of low cost foreign uranium for United States use through the end of the century.

1.4 Estimated Domestic Availability and Consumption of Uranium

Uranium requirements in the United States are currently about 13,000 tons of U_3O_8 per year and, with no recycle of uranium or plutonium to light water reactors, are expected to increase rapidly to about 47,100 tons per year by 1985, 98,200 tons per year by 1995, and about 113,900 tons per year by the year 2000.

Exploration will be necessary to develop and establish U.S. uranium resource supplies to match this expansion. This will necessitate substantial capital investment several years in advance of needed production and construction of new mining and milling facilities. Between the years 1973 and 1990 this investment has been projected at $10 billion, of which $6 billion would be for exploration. An additional $8 billion will be required in the period 1991-2000.[8] The need for extensive exploration stems from the expectation that existing ore reserves will be exhausted within the next two decades. The potential resources listed in Table IV F-2 have yet to be discovered and additional potential resources must be identified. Although there is some indication that the estimated quantities exist and will be found, it will take significant time and effort to discover and delineate the deposits. Exploration in the United States over the last few years has not expanded reserves significantly and increased efforts will be needed in the future to maintain a satisfactory resource base.

As a practical matter, ore reserves at any time should be at least capable of meeting requirements for the following 8 years according to the estimate that 8 years is the approximate idealized lead time between start of exploration and initial production of U_3O_8.[9] This lead time includes the time for exploratory geology, land acquisition, exploration and development drilling, as well as mine and mill construction. A reserve base is necessary to justify investment in mines and mills, for amortization of capital, and for contracting for sale of products. With no uranium or plutonium recycle, about 1,600,000 tons of U_3O_8 must be produced to satisfy requirements from the start of the year 1975 through the end of 2000. An 8-year reserve at that



* ERDA 1975 TAILS ASSAY
0.20 TO 10/1/77
0.25 TO 10/1/79
0.275 TO 10/1/81
0.30 AFTER

(See Reference 7)

Figure IV F-3 Foreign (Non-Communist) Uranium Production Capability and Requirements time would be about 900,000 tons of U_3O_8. In the United States today there is about a 13-year reserve of U_3O_8 in the \$15 per pound forward cost category. The practically achievable production of U_3O_8 from ore reserves and probable potential \$15 per pound resources, however, is expected to fall behind demand, possibly in the mid 1980's. To meet requirements beyond that time will require discovery of additional \$15 per pound resources, or production from higher cost and lower grade conventional ores or non-conventional ores, or both.

Under none of the recycle options considered in the present study is the quantity of uranium available in the \$15 per pound reserves and probable resources sufficient to satisfy the total uranium requirements through the year 2000. The additional uranium required must then be obtained from lower grade \$30 per pound reserves and probable resources along with some higher grade possible and speculative resources.

It appears possible that the shortage could be made up from \$30 forward cost reserves and probable resources alone. Thus, the ore delivered to the mills for processing during the year 2000 might consist of essentially all \$30 forward cost material (0.08% U_3O_8). Under the assumption that no major finds of highly concentrated ore will take place, the present study is based on the estimated ore grade processed in the year 2000 being about 0.08% U_3O_8. The 1974 mill recovery from 0.18% U_3O_8 grade ore feed was 93.2%. It is assumed that attainable recovery in the year 2000 might be midway between this value and that obtained with a constant tailings concentration of 0.0122% U_3O_8 (84.8% recovery), because some improvement in recovery through improved technology might be justified by the higher value of the yellowcake product. It is also assumed that the grade of ore processed, along with the mill recovery, will decrease at a uniform rate over the period from 1975 through 2000. Thus, the present study is based on projected estimates of an average ore feed grade of 0.1% U_3O_8 and a production weighted average mill recovery of 90.5% for the period from 1975 through 2000.

1.5 Projected Mining Industry

Mining production in the United States in 1974 was 12,600 tons of U_3O_8. From projected power growth data, it is apparent that in order to meet energy requirements, the mining industry will be called upon to expand production drastically as shown in the following tabulation:

PROJECTED URANIUM REQUIREMENTS
(Equivalent Tons U_3O_8 in Yellowcake)

Year	No U or Pu Recycle	With U Recycle	With U and Pu Recycle
1980	25,300	25,300	21,700
1985	47,100	46,700	39,700
1990	73,800	62,700	58,600
1995	98,200	86,200	73,100
2000	113,900	98,800	80,500

The decreased demand for uranium resulting from uranium or both uranium and Pu recycle would be conservatively expected to result in a proportionate reduction in the environmental impact of the mining industry.

The average annual production from underground mines remained close to 18,000 tons of ore per mine for the 7 years prior to 1975 but dropped to a 14,000 ton annual rate for the first half of 1975; the average annual production of open pit mines increased from 80,000 tons of ore per mine in 1969 to about 140,000 tons in 1973 and 1974.[10] Actual production rates and operating lifetimes of both underground and open pit mines are likely to be quite variable. To facilitate estimation of the environmental impacts of the projected mining industry in this analysis, it has been assumed that, on the average, production over the years 1975 through 2000 will be 40% from model open pit mines each producing 200,000 tons of ore per year, and 60 percent from model underground mines each producing 20,000 tons of ore per year.

No significant change in present mining methods is presently foreseen for future operations with the possible exceptions of increased in situ leaching particularly in Wyoming and Texas and the indicated shift to underground mining becoming the predominant method of the future. Assuming that the required resources will be located as conventional deposits, by the year 2000 as many as 5,800 mines may have to be in operation to supply industry demands under the no-recycle operating option.

1.6 Environmental Considerations

Underground mining is characterized by service buildings, a head frame with a truck loading facility, a mine waste pile, and frequently a flow of water pumped to surface drainage channels from underground sumps, after settling to remove suspended solids and treatment with barium salts (and sulfate) for removal of sufficient ^{226}Ra to meet applicable USEPA effluent standards. The area occupied by hoisting and loading facilities, shops, warehouse, changehouse, and office may be only a few acres, but the reach of underground tunnels may be a mile or more. Although the volume of the mine waste pile can be equal to the volume of processed ore, this material can be distributed in such a manner as to have little effect on the appearance of the surroundings. The total area affected permanently by underground mining is considered to be negligible.

Open pit mining is characterized by large open excavations, large piles of earth and rock overburden placed nearby, a network of operating roads and yards, and a flow of mine water pumped into local surface drainage after treatment similar to that for underground mine discharge. Shops, warehouse, office and changehouse structures are usually nearby, and an assortment of heavy earthmoving equipment is present. During much of the operating life of the mine, overburden is used for backfilling the mined out areas, thus minimizing haulage and the period of the physical and topographical impact of the operations. It is not possible to begin restoration efforts until sufficient mined out area is exposed to permit reclamation to proceed; during the later stages of mining and cleanup, the work of filling the final pit is not economically feasible. The final pit areas are sometimes converted to small lakes rather than attempting to achieve restoration to conditions similar to the surrounding region.

The environmental effects of the present uranium mining industry, including accident considerations, have been discussed in detail in WASH-1248, "Environmental Survey of the Uranium Fuel Cycle."[11] The environmental consequences of the projected mining industry are discussed below.

1.6.1 Use of Natural Resources

Land use data for open pit mines indicate that about one-third of the total land involved is disturbed temporarily by the actual mining operation while the remaining two-thirds remains idle.[12] Dusting and blowing associated with the excavated area are minimized by sprinkling the haulageways. Most of the land occupied by the mine itself can be reclaimed and restored before the site is abandoned. Ultimately, the operator grades and contours the piles of overburden and other mine waste remaining on the site to make it compatible with the surrounding countryside.

The effect of underground mining on surface land use varies considerably, depending upon the area covered by the dumped waste from constructing mine shafts, haulageways and access tunnels as well as that produced during normal mining operations. It is of interest that the land surface in the normally semiarid mining regions is rarely suitable for anything else other than possibly the grazing of cattle or sheep.

In both underground and open-pit mining, widely variable and sometimes quite large quantities of water may have to be pumped from the mines. Although this pumping lowers the water table in the immediate vicinity of the wells and sumps, the effect is temporary. When pumping is discontinued, the water table returns to its normal level. During the life of the mine, some of this water may be used as process water for an adjacent uranium mill or, if potable, for watering livestock. In most cases, most of the water recycles through natural seepage and evaporation and eventually returns to the ground water. Since any discharge will be required to meet applicable USEPA effluent standards, the radioactivity reaching surface drainage is not expected to produce any measurable consequence.

Mining operations also consume a relatively small amount of electrical energy, primarily for the operation of ventilation fans and dewatering pumps; in addition, diesel fuel is consumed by the earthmoving equipment.

1.6.2 Effluents

The primary chemical gaseous effluents from uranium mining derive from the burning of fossil fuels for required power and use of diesel oil for driving mining equipment. The drainage water carries some suspended solids, but these solids can be reduced by settling pond treatment and natural seepage. The primary solid waste material is the rock and earth overburden, most of which is ultimately returned as backfill in open pit mining, or distributed in such a manner as to have a minor impact on the appearance of the land in the case of underground mines.

Uranium and its daughters are released to the atmosphere when the ore body is exposed and broken up during either underground or open pit mining operations. The airborne radionuclides discharged from underground mines are rapidly reduced by atmospheric dispersion and reach normal background levels at the site boundaries. U.S. Bureau of Mines measurements of radon concentrations in existing open pit mines revealed no significant alpha concentrations.[13] Therefore, the concentrations of airborne radionuclides in unrestricted areas are expected to be indistinguishable from natural background.

Mine drainage water can contain significant levels of radioactivity from dissolved and suspended uranium and its daughters. The activity is removed from the water and returned to the ground by ion exchange during seepage through the soil. When it is economically feasible, uranium values are recovered from the mine water before it is discharged.[14]

1.6.3 Occupational Exposure

Personnel engaged in the uranium mining industry currently are subjected to internal and external radiological exposures. "External gamma radiation intensities in domestic (underground) uranium mines seldom exceed 2.5 mR per hour, and the average intensities are only a fraction of this. It is accordingly unlikely that uranium miners will be exposed to external whole-body radiation doses as large as the Radiation Protection Guide (RPG) recommended by the Federal Radiation Council (FRC) for occupational radiation exposure, 5 rems per year. However, in mining occasional high grade ore pockets (5 percent or greater U_3O_8) external radiation levels may necessitate limitation of personnel exposure."[15] Because both radon daughter and external exposures arise from the same source, it is reasonable to assume that the distributions of the two exposure situations are similar. It is fortuitous that both of these exposures have the same numerical upper limit, namely 5, W.L.M. and REM. Thus, the average value from a log-normal plot of reported radon daughter exposures[16] may be assumed to be numerically equal to an external whole-body exposure of 1.1 Rem/employee/year.

Film badge surveys in open pit mines indicate that external whole-body radiation doses are generally less than 50 mR per month. It is reasonable to assume that some exposure above background is experienced by open pit mine employees. For purposes of quantifying whole-body exposures, it has been assumed that the average whole-body dose is 10 percent of the RPG or 0.5 rem/employee/year. This amounts to a dose equal to about double the natural background dose for the areas of the country where the mines are located.

Based on these average exposures it is estimated that the total mining work force over the period 1975 through 2000 would receive a dose commitment of about 1.14×10^6 person-rem from external sources, if no uranium or plutonium is recycled to light water reactors. With uranium recycle, the external dose commitment would be reduced to about 1.02×10^6 person-rem, while with recycle of both uranium and plutonium the dose commitment would be further reduced to about 0.88×10^6 person-rem.

Internal exposures occur in underground mines primarily through inhalation of radon and radon daughters with subsequent potential exposure to the bronchial epithelial tissue of the lungs. It is estimated that over the period 1975 through 2000, the critical organ exposure will be approximately 6.4×10^6 person-rem to the lungs of the mine worker population if uranium and plutonium are not recycled. With uranium recycle, this organ dose commitment would be reduced to about 5.7×10^6 person-rem to the mine work force, while with recycle of both uranium and plutonium, the dose commitment would be decreased to about 4.9×10^6 lung-rem. The significance and potential effects of these exposures are discussed in Section J of CHAPTER IV.

1.6.4 Accident Considerations

Flooding or collapse of mine walls could result in halting mining operations. During recovery, the required increased pumping rate might cause some increased concentration of suspended solids in the discharged mine water. However, due to the low uranium content of the ore and corresponding low concentration of radioactivity present, it is not likely that the water reaching unrestricted areas would be much different from that in natural springs in the vicinity.

Resumption of underground mine ventilation after a power failure would result in a transient condition during which higher than normal concentrations of radon would be discharged to the atmosphere. Such an occurrence is not expected to affect the off-site environment measurably particularly since no increase in total radon release would be involved.

Accidental releases of mine drainage water from settling basins could result in the transport of solids containing uranium and its daughter products to nearby streams. Because of the small amounts and dilute nature of radioactive material involved, however, the effects on the environment would be insignificant.

An accidental fire involving petroleum products in the mine is credible, but unlikely because of industrial safety precautions. A fire could release combustion products, but radioactive material releases would not be involved.

1.6.5 Contributions to the General Exposure

Both the BEIR report[17] and the U.S. Environmental Protection Agency report, "Estimates of Ionizing Radiation Doses in the United States 1960-2000,"[18] conclude that although uranium mining activities increase the amount of surface uranium and its decay products, these activities do not cause measurable increases in environmental radioactivity outside the immediate vicinity of the mines.

In order to provide a quantitative estimate of the exposure resulting from uranium mining operations over the years 1975 through 2000, it has been assumed that ^{222}Rn is the only significant radioactive material released from the mining industry, that the maximally exposed individual lives 500 meters (about 0.3 mile) from an operating facility, and that all of his food is produced at this location. It is further assumed that the population density in the area of major particulate deposition is 10 persons per sq mi, and that the dispersal of radon before it decays results in decay products impinging on areas with an average population density of about 100 persons per sq mi. Table IV F-4 shows that the annual dose commitment to the closest theoretical resident, from an open-pit mine producing 200,000 tons of 0.1% U_3O_8 content ore per year, ranges from less than 1 millirem to a number of organs such as the skin, thyroid and GI tract to as much as 8.7 mrem to the lung and 4.7 mrem to the kidney, compared with the natural background whole body dose commitment of about 100 mrem/yr. In the unlikely event of an individual living about 0.3 mile from a 20,000 ton per year underground mine, it is estimated that the dose commitment would be one-tenth of the tabulated values.

Table IV F-4

DOSE COMMITMENT* FROM U.S. URANIUM MINING INDUSTRY AIRBORNE EFFLUENTS

Organ	Closest** Theoretical Resident mrem/yr	Integrated Total Population Dose Commitments from 1975 through 2000, person-rem		
		No U or Pu Recycle	With U Recycle	With U & Pu Recycle
Whole Body	0.29	2,970,000	2,660,000	2,310,000
GI Tract	0.27	116,000	104,000	90,000
Bone	2.1	9,690,000	8,670,000	7,520,000
Liver	1.5	2,400,000	2,140,000	1,860,000
Kidney	4.7	11,300,000	10,100,000	8,760,000
Thyroid	0.0006	7,300	6,500	5,700
Lung	8.7	910,000	810,000	705,000
Skin	0.0006	7,300	6,500	5,700

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*See IV J, Appendix A for explanation of exposure modes and duration incorporated in the dose commitment determination.

**Based on estimated ^{222}Rn effluent from open-pit mine producing 200,000 tons per year of 0.1% U_3O_8 ore for each option. $\chi/Q = 5.4 \times 10^{6}$ sec/cu m at 500 meters from ground level release.

Table IV F-4 also presents the total population dose commitment produced by the entire mining industry over the years 1975 through 2000, with and without uranium or plutonium or both being recycled to the light water reactor industry. These data indicate that for the population dose and the critical organ (kidney), uranium recycle decreases the exposure to the population by about 1,200,000 person-rem, and uranium and plutonium recycle produces an additional decrease of about 1,340,000 person-rem. The 26-year integrated whole body dose commitment of about 3 million person-rem shown in Table IV F-4 for the no recycle option represents less than one half of one percent of the total whole body dose to the U.S. population received from natural background radiation. Section J of CHAPTER IV presents a discussion of the health effects of these radiological exposures.

1.7 General Environmental Impact

Although it is anticipated that further implementation of the ALARA* philosophy will produce significant improvements in controlling the environmental impact of conventional uranium mining operations, these improvements may be offset by the required large expansion of the industry, coupled with the need to process more ore per unit of U_3O_8 as the ore grade declines. Thus it is expected that future industrywide environmental impacts will tend to increase in proportion to the increasing production rates.

In production of power from light water reactors, recycling the uranium recovered from spent reactor fuel is estimated to reduce the requirements for fresh uranium from ore production by about 10.5% (from 1.6×10^6 tons to 1.43×10^6 tons U_3O_8) over the period 1975 through 2000. Recycling the plutonium recovered from the spent reactor fuel is estimated to reduce the fresh uranium requirements by an additional 13.2% (from 1.43×10^6 tons to 1.24×10^6 tons) over the same time period. With the environmental effects of the mining operation assumed to be proportional to the amount of ore produced, the overall effect of uranium and plutonium recycle would then be about a 22.5% (from 1.6×10^4 metric tons to 1.24×10^9 metric tons ore) reduction in the environmental impact of the projected uranium mining industry. This reduction would be roughly equivalent to more than 50 times the total environmental impact of the existing mining industry, which produced about 7 million tons of ore in the year 1974.

It should be pointed out that a small additional benefit could be derived by plutonium and uranium recycle, because it is assumed that the average ore grade continues to decrease from the year 1975 to 2000. In this event, fewer tons of lower grade ore would have to be mined to meet the LWR fuel cycle requirement when using MOX fuel, resulting in an additional reduction in environmental impact. A further benefit of plutonium recycle is associated with stretching out the time during which the higher grade uranium reserves meet the LWR industry demand. This additional time will provide for exploration and for development of improved mining and pollution abatement techniques under conditions involving the use of higher grade ores which will have a lower environmental impact than if low grade ores were being processed.

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*As Low As Reasonably Achievable.

The estimated impacts of the mining industry projected for the years 1975 through 2000, without recycle, with recycle of uranium only, and with both uranium and plutonium recycle, are summarized in Table IV F-5. The indicated net decrease is the direct result of reducing the requirement for uranium ore and with it, reducing the estimated total number of operating facilities.

Table IV F-5

SUMMARY OF INTEGRATED ENVIRONMENTAL IMPACTS OF URANIUM MINING INDUSTRY OVER THE PERIOD 1975 THROUGH 2000*

	No U or Pu Recycle	With U Recycle	With U and Pu Recycle
Industry Production (Millions of MT of Ore)	1,600	1,440	1,250
Number of Facilities	5,840	5,060	4,130
Natural Resource Use			
Land (Acres)			
Area Disturbed	334,000	289,000	237,000
Overburden Moved			
For Open Pit Mines (Millions of MT)	19,200	17,200	14,900
Water (Billions of Gallons)			
Discharged to Ground	2,990	2,670	2,320
Effluents			
Radioactive Gases (Curies)			
^{222}Rn**	23,700,000	21,200,000	18,400,000
Radiological Dose Commitment (Person-rem)			
Occupational (Whole Body)	1,200,000	1,000,000	900,000
Total Population (Whole Body)	2,970,000	2,660,000	2,310,000

See Appendix A for estimate of additional environmental impacts.

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*Based on USERDA NUFUEL Computer Program, Low-Growth Forecast of Nuclear Power, April 1976, and average ore grade of 0.1% U_3O_8.

**Based on ^{222}Rn concentration of 1,500 pCi/l in ventilation air discharged from operating underground mines and assumption that calculated release of about 1.5 Ci ^{222}Rn per metric ton of ore mined is also applicable to open pit mining operations.

The most significant effects of recycling uranium or both uranium and plutonium on the projected 1975 through 2000 uranium mining industry include

- A reduction in required 0.1% U_3O_8 average grade ore production of about 170 million metric tons with uranium recycle and an additional reduction of 190 million metric tons with uranium and plutonium recycle. The total decrease of 360 million metric tons, is equivalent to more than 50 times the total 1974 production of the present mining industry.

- A decrease in the amount of land that must be temporarily removed from other uses of about 45,000 acres with recycle of uranium, and an additional reduction of 52,000 acres with uranium and plutonium recycle. The total decrease, 97,000 acres, is equivalent to more than 150 square miles.

- About 2.0 billion metric tons of overburden will remain undisturbed as a result of the decreased ore production requirements associated with uranium recycle with an additional 2.3 billion metric ton reduction in overburden removal resulting from the recycle of uranium and plutonium. The total decrease in the amount of overburden, about 4.3 billion metric tons, is equivalent to a pile of earth with a height equal to that of a five story building covering an area equal to that of the entire city of Washington, D.C.

- A reduction of about 3,000,000 curies in the amount of radon-222 released in the mining operation as a result of uranium recycle, with an additional reduction of 3 million curies produced by plutonium recycle. This decrease in radon release is equivalent to about 0.37% of the total radon that is emitted annually from the entire land area of the world.

2.0 THE URANIUM MILLING INDUSTRY

2.1 Current Status

At the end of the year 1975 there were 14 conventional mills operating in the United States, with a combined nominal processing capacity of 27,250 tons of ore per day. One additional mill was on standby, three more either in need of or undergoing renovation, two facilities were under construction and two more were planned.[19,20] During the year 1974, the average tonnage processed by the operating mills was 23,000 tons per day (TPD) containing an annual total of 12,442 tons of U_3O_8.[21] The currently operating mills range from 400 to 7,000 tons per day in nominal ore capacity with most of the facilities falling into the range of 1,000 to 2,000 tons per day. All of the active mills are located in the western States with three in New Mexico (13,500 TPD nominal ore capacity), five in Wyoming (8,150 TPD), two in Colorado (1,750 TPD), two in Utah (1,700 TPD) and one each in Texas (1,750 TPD) and Washington (400 TPD).[19]

In addition to these conventional uranium mills, there were six commercial or pilot scale solution mining facilities in operation or planned in Texas, and five research and development solution mining installations in operation and several commercial scale facilities planned in Wyoming. One central facility and one satellite facility for recovering uranium from phosphoric acid were in operation in Florida with additional facilities planned that will be capable of recovering as much as 3,000 tons of U_3O_8 per year by 1980. The potential for recovery of byproduct U_3O_8 from the Florida phosphoric acid industry through the end of the century has been estimated at as much as 70,000 to 100,000 tons. Additional facilities that recover small quantities of uranium from mine water are in operation or are planned in New Mexico, Wyoming and Utah and several heap-leach recovery facilities are planned or are in operation in several of the western States. An additional facility is in the advanced planning stages to investigate recovery of byproduct uranium from copper dump leach liquors in Utah.

There is considerable uncertainty in predicting the quantities of uranium recoverable by the various unconventional processing schemes, but it is estimated that they would contribute a relatively small percentage of the total cumulative requirements. It has been assumed for the present study that the total uranium requirements will be supplied by the conventional mining and milling industries.

Although the uranium extraction process varies from mill to mill depending mainly on the chemical composition of the ore, steps common to all mills processing conventionally mined ores include crushing, grinding, chemical leaching to extract the uranium from the finely divided ore, and recovery of the uranium from the leach solutions. The mill chemical processes employed fall into three general classes:

- Acid leach-solvent extraction

- Acid leach-ion exchange

- Alkaline leach

In acid leach mills, the ore is ground and then leached with sulfuric acid in the presence of an oxidant. After separation from the undissolved solids, uranium is recovered and separated from impurities in the sulfuric acid leach solution either by solvent extraction or by the use of an ion exchange resin. A purified and concentrated uranium solution is then stripped from the organic solvent or eluted from the ion exchange resin. The uranium is then finally precipitated, dried, and packaged for shipment.

In alkaline leach mills, ore is ground much finer than in the acid process and is leached with a sodium carbonate-sodium bicarbonate solution at elevated temperature and pressure, using air or an oxygen-air mixture as an oxidant. After separation of the solid, the liquid is recarbonated with carbon dioxide from the flue gases and recycled to the process. In general, the carbonate leach is more selective for extraction of uranium than the acid leach; thus it may not be necessary to purify the alkaline leach solution before precipitating the uranium with sodium hydroxide. In recent years, however, some alkaline circuit mills have had to purify their yellowcake product by dissolving it in sulfuric acid and reprecipitating with hydrogen peroxide or ammonia in order to meet the allowable sodium content in the product.

2.2 Projected Milling Industry

Although uranium requirements for the next few years will be somewhat below current production capability, by the late 1970's additional milling capacity will be needed. Thereafter, a rapid expansion will be needed to meet the projected demands of the nuclear power industry. The following tabulation shows the estimated future power capacity of light water reactors and the projected uranium milling capacities needed to satisfy these demands with and without recycle of uranium or both uranium and plutonium to LWR's.

PROJECTED INDUSTRY REQUIREMENTS[22]

Year	Projected Nuclear Power Capacity MWe	Corresponding Requirements for Uranium Milling Industry Output, Tons of U_3O_8 No U or Pu Recycle	With U Recycle	With U and Pu Recycle
1980	70,200	25,300	25,300	21,700
1985	156,000	47,100	46,700	39,700
1990	269,000	73,800	62,700	58,600
1995	400,000	98,200	86,200	73,100
2000	507,000	113,900	98,800	80,500

Because the milling industry varies widely, the effects of the industry are difficult to project in detail. Accordingly, a model mill has been developed for this study to characterize the industry effects and to estimate the environmental impacts. It has been assumed that the uranium milling industry over the years from 1975 through 2000 will be made up of the required number of model mills to meet the projected U_3O_8 requirements. The model selected for this characterization is a hypothetical composite facility processing 85% of the ore feed through an acid leach-solvent extraction circuit and the remainder through an alkaline leach circuit. The composite model mill is assumed to have the capacity to process 3,500 metric tons of 0.1% U_3O_8 ore per day, 300 days per year, with a U_3O_8 recovery of 90.5% for an annual output of about 1,050 tons of U_3O_8 as yellowcake. Based on these assumptions, with no uranium or plutonium recycle, 109 model mills would be required in the year 2000. With recycle of uranium to light water reactors the number of model mills required in that year would be decreased to 95; and with recycle of both uranium and plutonium only 77 model mills would be required.

As at present, it is expected that milling facilities will continue to be located close to mining operations to minimize haulage costs and thus will be located in semiarid, isolated regions of the western States where population densities range from 5 to 10 persons per sq mi. Although it is not now feasible to predict the effect of the ALARA philosophy it appears reasonable to assume that significant technological improvements affecting operating efficiency as well as environmental impacts will be attained. However, this analysis takes no credit for such advances.

2.3 Environmental Considerations

The environmental impacts of current milling operations are discussed in the "Environmental Survey of the Uranium Fuel Cycle," WASH-1248.[23] The principal environmental consequences of the milling industry include the following:

2.3.1 Uses of Natural Resources

The largest part of the land disturbed by a conventional uranium mill is devoted to the disposal of mill tailings, because in effect, nearly the entire mass of ore processed by the mill ends up in the tailings. After mills are shut down, the tailings areas are covered with earth and planted to prevent erosion and spread of radioactive material by wind and water. The covered and planted areas are restored to resemble the surrounding terrain. Current practice is to withhold such land from future unrestricted use to minimize potential exposures to the residual uranium and radioactive uranium decay daughters pending completion of studies directed toward development of acceptable procedures for ultimate disposal of mill tailings.

2.3.2 Effluents

In addition to the gaseous effluent release associated with generation of electric power required by the mill, small quantities of sulfuric acid fumes, kerosene, and dust may be released to the atmosphere from uranium mill processes. Airborne concentrations of these contaminants are maintained below EPA standards, and deleterious effects on biota are highly unlikely.

Radioactive airborne effluents consisting of uranium and uranium daughter products are released primarily as particulates from the dust control equipment servicing the ore crushing and yellowcake drying operations. Additional particulates and radon gas are also released from the tailings piles adjoining each of the uranium mills. Estimated airborne radiological emissions from the projected uranium milling industry over the 1975 through 2000 period without recycle, with recycle only of uranium and with both uranium and plutonium recycle are presented in Table IV F-6. As indicated, the airborne radioactive material discharged by the milling industry contains more radon-222 than any other radionuclide, about 3,500,000 to 4,400,000 curies over the 26 year period, depending on whether or not uranium or both uranium and plutonium are recycled. The recycle of U and Pu decreases this emission by about 20% while the recycle of U only effects a decrease of about 9%.

Table IV F-6

ESTIMATED URANIUM MILLING INDUSTRY AIRBORNE EMISSIONS FOR PERIOD 1975 THROUGH 2000 (CURIES)*

Radionuclide	No U or Pu Recycle	With U Recycle	With U & Pu Recycle
^{238}U	260	230	200
^{234}U	260	230	200
^{235}U	12	11	9
^{234}Th	19	17	15
^{230}Th	24	21	19
^{226}Ra	13	12	10
^{222}Rn	4.4×10^6	4.0×10^6	3.5×10^6

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*Based on data from Reference 25.

Liquid and solid chemical and radioactive wastes are discharged to the tailings retention pond. No significant adverse effect is expected to occur in the offsite environment after the mill is decommissioned and the pond area is graded, covered with earth, and revegetated.

Essentially all of the process heat used in the mill operations is discharged to the atmosphere with little effect on the environment other than the potential for some local fogging under certain meteorological conditions.

2.3.3 Occupational Exposure

2.3.3.1 External Exposure

Film badge surveys conducted in uranium mills indicate that external whole body radiation doses to employees are generally less than 50 mrem per month. To provide a quantified estimate of whole body exposures to uranium mill personnel, it is assumed that the average whole body dose commitment is 10% of the 10 CFR 20 permissible level, or 500 mrem/yr per employee. This exposure is equal to about twice the natural background dose commitment for the areas of the country where the majority of uranium mills are located. On this basis, the external dose commitment to the total mill employee population projected over the period 1975 through 2000 is estimated at about 209,000 person-rem if no uranium or plutonium is recycled to light water reactors. With uranium recycle, the total industry population dose commitment would be reduced to about 187,000 person-rem/yr, and with recycle of both uranium and plutonium, the total external exposure to mill personnel would be decreased to about 163,000 person-rem/yr.

2.3.3.2 Internal Exposure

Exposure to Radon Daughters--Limited surveys of radon daughter concentrations in uranium mills indicate that significant concentrations are seldom found. There are areas in some mills, however, where exposure may approach those experienced in underground mines. The number of persons currently exposed to significant concentrations of radon daughters is small, and with the anticipation of improved control methods (primarily improved ventilation), it is assumed that exposure to radon daughters in uranium mills will be negligible.

Exposure to Other Airborne Radioactive Materials--Surveys to determine concentrations of airborne radioactive materials in processing areas of uranium mills indicate that in some operating areas personnel may be exposed to as much as a 2.5×10^{-12} to 5 5×10^{-12} μCi/ml concentration of natural uranium (5 to 10% of 10 CFR 20 MPC values). It is assumed that secular equilibrium, from ^{238}U through ^{226}Ra, generally exists in the airborne material in the ore processing areas of the mills. Generally only a small number of mill employees are exposed to essentially daughter free natural uranium in the final product handling areas of the mills, and a significant number of the mill employees, clerks, warehouse employees, secretaries, and other office personnel, are rarely exposed to concentrations of radioactive materials in excess of natural background. Therefore, this present study is based on the assumption that on the average, all mill employees are exposed to 2.5×10^{-12} μCi/ml "insoluble" airborne natural uranium in secular equilibrium with its daughters. Exposure to air containing this concentration of radionuclides is estimated to produce an annual critical organ (lung) dose commitment to the individual employee of about 11 rem. Exposure of the total milling industry workers over the years 1975 through 2000 (with no uranium or plutonium recycle) to this concentration of radionuclides in the plant air would produce a total occupational lung dose commitment of about 4,800,000 person-rem. With recycle of uranium to LWR's, this lung dose commitment would be decreased to about 4,300,000 person-rem, and with recycle of both uranium and plutonium the lung dose commitment is estimated to be reduced to about 3,800,000 person-rem.

2.3.4 Accident Considerations

A tailings dam failure due to natural causes can be postulated, but is unlikely. Tailings releases which occur from equipment malfunction or human error are not likely to have any appreciable environmental effect based on the few such incidents that have occurred. The materials carried from the tailings pond or pipeline would be deposited through sedimentation over a relatively short distance. Recovery by standard earthmoving equipment would be straightforward, and all waste materials and contaminated soils could be either buried in place or returned to the tailings pond.

A postulated fire in a solvent extraction building is also credible, but the effects would be transitory and largely confined to within a few hundred feet from the building. Recovery would require a radiation survey of the site and some limited removal or burial of contaminated soil. Based on the effects of solvent extraction fires which have occurred in two different mills, it is unlikely that the consequences would be substantially greater than the temporary loss of operating capability.

2.3.5 Contributions to the General Exposure

Population doses attributable to the uranium milling industry are expected to be relatively low because of location in remote and sparsely populated areas, comparatively low specific activity of uranium and its daughter products, and the tendency of uranium and its daughter products (except radon and in some cases radium) to be removed rather quickly from air or water, so that widespread dispersion is unlikely. Except for the possible transport of ^{222}Rn, which is present throughout the world in background radiation, it can be expected that, in general, no significant quantity of radioactive materials would reach heavily populated areas. Although uranium milling activities contribute to the content of radioactive material in the environment, it appears from available measurements that population doses from this source cannot be distinguished from background.[24]

As with mining, in order to derive a quantitative estimate of the individual and population dose commitments resulting from uranium milling operations over the 26-year period 1975 through 2000, conservative assumptions have been used to assess the dose to the maximally exposed individual. These assumptions include that he lives about 500 meters from a model operating mill processing 3,500 tons of ore per day and that his entire food intake is produced at this location. It is further assumed that the population density in the area of major particulate deposition is 10 persons per sq mi and that the dispersal of radon produced in processing the ore as well as that emanating from the tailings piles impinges decay products on areas having an average population density of about 100 persons per sq mi. Estimates of radon discharge from tailings as well as airborne particulate emissions are based on the results of a study conducted by the Oak Ridge National Laboratory to determine the radiation levels and the radioactivity releases that could be considered "as low as practicable" with various technologies employed in the milling of uranium ores.[25]

Estimated whole body dose commitment to the closest theoretical resident, as well as maximum organ dose commitments to this individual produced by airborne effluents, from a uranium mill processing 3,500 tons of ore per day are shown in Table IV F-7. This indicates that in the extremely unlikely situation of an individual residing 500 meters from an operating facility and eating all of his food produced at this location, the individual would receive a whole body dose commitment of 147 mrem along with a dose commitment to the critical organ (bone) of 347 mrem. These dose commitments represent about 29% of the maximum permissible whole body exposure specified in 10 CFR 20 and about 12% of the maximum permissible bone dose recommended by ICRP.

Table IV F-7 also shows the total population dose commitment resulting from operation of the entire uranium milling industry over the period 1975 through 2000 without recycle, with recycle of uranium, and with both uranium and plutonium recycle to the light water reactor industry. The effect of uranium recycle on the total population whole body dose commitment is a reduction of 61,000 person-rem, and uranium and plutonium recycle produces an additional decrease of 68,000 person-rem. The data also indicate that for the critical organ (kidney), uranium recycle decreases the exposure

Table IV F-7

DOSE COMMITMENT* FROM U.S. URANIUM MILLING INDUSTRY AIRBORNE EFFLUENTS

		Integrated Total Population Dose Commitments from 1975 through 2000, Person-Rem		
Organ	Closest Theoretical Resident** mrem/yr	No U or Pu Recycle	With U Recycle	With U & Pu Recycle
Whole Body	147	579,000	518,000	450,000
GI Tract	4.5	22,400	20,000	17,000
Bone	347	1,870,000	1,680,000	1,460,000
Liver	14.3	449,000	402,000	349,000
Kidney	129	2,130,000	1,910,000	1,660,000
Thyroid	1.7	1,500	1,400	1,200
Lung	65.8	175,000	157,000	136,000
Skin	1.7	1,500	1,400	1,200

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*See IV J, Appendix A for explanation of exposure modes and duration incorporated in the dose commitment determination.

**Based on effluents from model mill processing 3,500 metric tons/day of ore for each option. $x/Q = 5.4 \times 10^{-6}$ sec/cu m at 500 meters from ground level release.

to the population by about 220,000 organ-rem, and the recycle of uranium and plutonium reduces the exposure by an additional 250,000 organ-rem. See CHAPTER IV, Section J, for a discussion of the health effects of these radiological exposures.

2.4 General Environmental Impact

It is expected that process modifications in future conventional uranium milling operations will be directed toward improving operating efficiencies, increasing uranium recoveries and decreasing impacts on the environment. It is also expected that a fraction of future uranium requirements may be supplied by solution mining and byproduct recovery operations, which would tend to eliminate the formation of tailings piles and reduce emissions of radon, but could have the potential of some increase in contamination of ground water. Because of the difficulty in assessing the effects of possible process improvements that might decrease the adverse effects per ton of yellowcake produced, in this study no credit is taken for such improvements, and the expansion in uranium requirements is assumed to produce an increase in impact on the environment proportional to the increased demands. As discussed previously, recycling the uranium recovered from light water reactor spent fuel is estimated to reduce the requirement for fresh uranium from the mills by about 10.5% over the period from 1975 through 2000. Recycling uranium and plutonium recovered from the spent reactor fuel is estimated to reduce the fresh uranium requirements by an additional 13% over the same period. If these decreases in U_3O_8 production requirements result in corresponding decreases in environmental impacts, the effect of uranium and plutonium recycle over the 26 year period would be to reduce the environmental effect of the expanded milling industry by an amount equivalent to more than 30 times that of the annual impact of the currently operating mills. A summary of the estimated environmental impacts of the uranium milling industry projected over the period without recycle, with recycle of uranium, and with recycle of both uranium and plutonium recovered from light water reactors is presented in Table IV F-8. Estimates of other environmental effects are presented in Appendix A. The most significant effects of recycle on the projected uranium milling industry from 1975 through 2000 include

- A reduction of about 170,000 tons in the U_3O_8 required for light water reactor fuel if uranium is recycled and an additional decrease of about 190,000 tons if both uranium and plutonium are recycled over the 1975-2000 period. The total reduction in requirements, about 360,000 tons of U_3O_8, is equivalent to about 30 times the total annual production of the uranium milling industry in 1974.

- A decrease of almost 4,000 acres in the land permanently committed for limited use as mill tailings areas, if uranium is recycled, and an additional reduction of 5,000 acres if both uranium and plutonium are recycled. The land permanently committed for limited use as tailings areas represents about 73% of total land requirements of the supporting uranium fuel cycle.

- A decrease of about 470,000 curies in the amount of ^{222}Rn that will be released from active mill tailings areas if uranium is recycled and an additional reduction of about 520,000 curies, if both uranium and plutonium are recycled

- A reduction of 170 million metric tons in the quantity of mill tailings with a corresponding reduction of about 600,000 curies of contained radioactivity in the tailings occurring if uranium is recycled and an additional reduction of 190 million metric tons containing about 800,000 curies of activity, if both uranium and plutonium are recycled. These mill tailings, approximately equal in tonnage to and containing about 85% of the radioactivity initially present in the ore that was processed, represent essentially all of the total output of liquid and solid radioactive wastes generated by the supporting uranium fuel cycle facilities.

3.0 THE UF_6 CONVERSION INDUSTRY

3.1 Current Status

A necessary step in the LWR uranium fuel cycle is conversion of uranium ore concentrates to uranium hexafluoride, a gaseous compound used as feed to a uranium enrichment facility. The total domestic commercial industry for converting natural uranium to uranium hexafluoride currently consists of two plants, with a combined capacity of about 17,300 MT of uranium per year as UF_6.[26] There are two basic processes

Table IV F-8

SUMMARY OF INTEGRATED ENVIRONMENTAL IMPACTS OF URANIUM MILLING INDUSTRY OVER THE PERIOD 1975 THROUGH 2000

	No U or Pu Recycle	With U Recycle	With U & Pu Recycle
Industry Production (ST U_3O_8)	1,600,000	1,430,000	1,240,000
Number Of Model Facilities in Year 2000	109	95	77
Natural Resource Use			
Land Permanently Committed (Limited Use, Acres)*	30,500	26,600	21,600
Water Discharged to Air (Millions of Gallons)	1,150,000	1,030,000	890,000
Fossil Fuel			
Natural Gas (Millions of Therms)	12,000	10,800	9,300
Effluents			
Chemical,			
Solid Wastes, Tailings (Millions of MT)	1,600	1,430	1,240
Radioactive (Curies)			
Gas ^{222}Rn	4,440,000	3,970,000	3,450,000
Solid Wastes, U and Daughters	5,400,000	4,800,000	4,200,000
Radiological Dose Commitment (person-rem)			
Occupational (Whole Body)	561,000	502,000	436,000
Total Population (Whole Body)	579,000	518,000	450,000

Note: See Appendix A for estimate of other environmental impacts.

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*Based on scale-up of data from Reference 25 and each plant uses a different process for production of the UF_6. In the dry process, the uranium concentrate feed is carried through successive reduction, hydrofluorination, and fluorination steps in fluidized bed reactors, after which the crude uranium hexafluoride is purified by fractional distillation. The wet process employs a chemical solvent extraction step at the head end to prepare a high purity uranium oxide feed prior to the reduction, hydrofluorination, and fluorination steps. In this process, a flame reactor rather than a fluidized bed unit is used to carry out the fluorination of the solid uranium tetrafluoride (UF_4) product of the hydrofluorination step. Although both processes deliver the same end product, waste effluents are somewhat different, with the dry process generating a larger quantity of radioactive solid wastes and the wet process, a larger quantity of liquid radioactive wastes.

The dry process removes most of the uranium concentrate impurities either as solid constituents remaining with the calcium fluoride used in the fluid bed fluorination step, or as volatile compounds and still bottoms separated in the UF_6 distillation operation. The majority of the volatile impurities are removed from the off-gases by caustic scrubbers. In the wet process, the bulk of the concentrate impurities is contained in the raffinate stream produced in the solvent extraction purification step. Both manufacturing processes effect essentially complete recovery of uranium and utilization of elemental fluorine as well as high utilization of a number of the other major reactants such as hydrogen, hydrogen fluoride, and ammonia.

The two existing commercial UF_6 production facilities at Metropolis, Illinois, and Sequoyah, Oklahoma, are in areas where the range of population density is 35 to 60 persons per sq mi.[27] The region surrounding the Metropolis plant, which uses the dry process, is the more densely populated.

3.2 Projected UF_6 Conversion Industry

For the current study, it is assumed that essentially all of the U.S. UF_6 conversion demand will be supplied by domestic facilities over the time frame 1975-2000. According to the current plans to increase the capacity of the present domestic wet process plant by the year 1978, the installed U.S. capacity of about 21,900 MTU/yr at that time will be adequate to satisfy projected U.S. conversion needs through the year 1981. However, it is estimated that the conversion capability will have to be expanded to about 87,300 MTU/yr by the year 2000 with no uranium or plutonium recycle, to about 75,500 MTU/yr if only uranium is recycled, or to 59,300 MTU/yr if both uranium and plutonium are recycled. Thus, if the average capacity of conversion plants of the future will be about 15,000 MTU per year, a total of three new domestic conversion facilities will be needed onstream by the year 2000 to meet projected requirements with uranium and plutonium recycle, four new plants will be needed if only uranium is recycled, and five new facilities will be required to meet the demand if neither uranium nor plutonium is recycled.

Selection of the manufacturing process to be used in future conversion facilities will depend largely on the relative unit production costs of the UF_6. An additional factor that might be involved could include a comparison between the cost of producing UO_2 suitable for mixed oxide fuel as an intermediate in a modified wet UF_6 process versus the cost of using one of the conventional UO_2 processes starting with pure UF_6 as the raw material. An important cost to be assessed against the wet process could be that entailed in providing an acceptable method for permanent disposal of the nitrates and radioactive wastes contained in the solvent extraction raffinates. The current practice of storing raffinate solutions is considered to be temporary while better long term solutions to the waste disposal problem are being developed. For the present study, it has been assumed that the future UF_6 conversion industry will consist of the two existing facilities, with the wet process plant upgraded to a capacity of about 9,100 MTU per year plus additional 15,000 MTU per year capacity facilities equally divided between the two processes as required to meet the projected demand with the first new plant to go on stream using the dry process.

3.3 Environmental Considerations

As with mining and milling, the environmental impact of current UF_6 production operations is discussed in the "Environmental Survey of the Uranium Fuel Cycle," in support of light water reactors.[28] The environmental consequences of a UF_6 conversion industry in which both the dry and the wet process are used for the conversion of uranium concentrates to UF_6 are discussed below. Over the period 1975 through 2000, it is estimated that about 59% of the total conversion demand will be provided by dry process and the remainder by wet process facilities.

3.3.1 Use of Natural Resources

To characterize the effects of this process step, model dry process and wet process plants have been developed. In either case, the model UF_6 production facility is assumed to be built on a site occupying about 1,400 acres in a developing industrial area where the previous use of the land was primarily agricultural. Although an estimated 10% of the site is disturbed for roads, fills, and plant structures, it is expected that most of this area can ultimately be restored to its original condition. It is estimated that about 10% of the disturbed land will be permanently committed to building foundations, slabs, and onsite burial of low activity calcium fluoride and other solids. It is expected that the plant sites will be selected with consideration for minimized impact on local recreation, historical factors, and adjacent farming activity. An additional minor consideration may be proximity to uranium enrichment facilities to minimize transportation costs.

The rate of water use for a UF_6 production plant is not unusual for an industrial operation. The wet solvent extraction process uses water for reagent makeup and the aqueous phase of the solvent extraction step. Both methods require water for process cooling systems, and both return cooling water to natural receiving bodies of water.[29,30] In either case, the plants are expected to be located near a water source that would provide a reliable supply and permit final discharge of treated liquid wastes and cooling water. As has been indicated, although the wet process solvent extraction raffinate is currently being retained indefinitely until a permanent disposal procedure is developed, other liquid wastes are treated in holding ponds, sampled, and diluted with other clear aqueous plant effluent streams before being discharged.

The present UF_6 conversion industry uses about 4.8 million gal/day of water for all purposes, of which about 87% is returned to surface waters and the remainder is discharged to the air either as direct cooling tower losses or other evaporation.

Other resources consumed by the UF_6 conversion industry include natural gas used for process and building heat, and electrical power, used for operating plant machinery, generating fluorine, and for plant lighting. Although it is probable that some of the electrical power requirements of all segments of the supporting uranium fuel cycle industries during the years from 1975 through 2000 will be supplied by nuclear generation facilities, in this section the environmental impact produced by power plants has not been considered except for uranium enrichment. which is estimated to require more than 90% of the total electrical power consumed by the entire uranium fuel cycle.

3.3.2 Effluents

Various process off-gases are generated in the production of UF_6 from uranium concentrates. Many of these are combustion products, but some are volatilized solids and gases evolved during calcining and fluorination. Several off-gas treatments are used to minimize the concentrations of airborne effluents released to the environment. Fluorides and oxides of nitrogen are the more significant sources of potential adverse environmental impact. Historically, analyses of airborne concentrations of fluoride as HF in air and concentrations in forage, in the vicinity of a wet solvent extraction plant, indicate fluoride levels below those expected to cause deleterious effects on human health or grazing animals. Long term observation of an area within a 7 mile radius of a dry process plant has not revealed any adverse effects attributable to fluoride releases from the plant.[31]

There are two major aqueous waste streams associated with UF_6 production. Many of the contaminants in this liquid waste are in the raffinate stream from the solvent extraction process, which is not released to the environment, but held in sealed ponds. The second stream is made up mostly of cooling water and dilute scrubber solutions, which represent the bulk of the water use. Some of these aqueous effluents are treated with calcium to precipitate calcium fluoride and diluted with all remaining clear water effluents from the plant before release. The solid calcium fluoride is recovered from settling ponds and ultimately buried on the site.[32] Analyses of water samples taken to measure the amounts of fluoride and nitrate concentrations in the vicinity of a wet process UF_6 production plant showed concentrations of 0.1 ppm and 0.3 ppm, respectively, which are within the 1975 Environmental Protection Agency regulations for drinking water sources.[33]

A small quantity of natural uranium is released from exhaust air as dust. Radioactivity in liquid effluents, originating primarily from natural uranium, is continuously monitored and has averaged less than 2×10^{-6} μCi/ml, less than 10% of the applicable 10 CFR 20 limit for water effluents to the unrestricted area over the history of an existing wet process UF_6 production plant.[34] Estimated total airborne and waterborne radiological emissions from the projected UF_6 industry over the years 1975 through 2000 without recycle, with recycle of uranium, and with both uranium and plutonium recycle are presented in Table IV F-9. As indicated, the total industry airborne emissions are estimated to range from about 7.4 to about 9.5 curies, and total industry waterborne emissions are estimated to range from about 340 to about 450 curies over the 26 year period, depending upon whether or not uranium or both uranium and plutonium are recycled.

The major source of solid radioactive wastes from the dry process is the ash residue from the fluidized bed fluorinators. This residue, containing nonvolatile fluorides and traces of uranium and uranium daughters, is packaged and consigned for burial at a licensed commercial waste burial facility.[35] Much smaller quantities of solid ash wastes are produced in the flame fluorination reactors used in the wet UF_6 process. This material is recycled to the head end of the process where the impurities are converted to liquids that leave in the solvent extraction raffinate. Although a relatively small quantity of solids is produced by neutralizing this raffinate solution with ammonia, most of the radioactive impurities entering with the plant yellowcake feed are converted to solids at this point in the process. These solids also are collected and disposed of at a licensed commercial burial ground. The total quantity of radioactive solids estimated to be generated by the UF_6 industry over the period from 1975 through 2000 ranges from about 97,000 to 127,000 metric tons containing about 21,000 to 28,000 curies, depending upon the industry production requirement that is dictated by the fuel management mode.

Process heat is dissipated from either type of production facility, either directly to the air or by evaporation of cooling water from towers or holding ponds. The temperature of the wet process cooling water is raised about 5°F before being discharged to the receiving stream. The major part of the thermal load from the dry process is discharged to the atmosphere as heated air.

3.3.3 Occupational Exposure

Personnel employed in the UF_6 conversion industry will be exposed to external beta and gamma radiation, as well as to airborne natural uranium and uranium daughters. External exposure to beta radiation for the entire industry over the period 1975 through 2000, without recycle, with recycle of uranium, or with both uranium and plutonium recycle, is estimated at less than 10,100 person-rem and exposure to gamma radiation is estimated to be less than 2,800 person-rem.*

Internal organ doses result from penetrating external gamma radiation plus inhalation of air containing about 1.6×10^{-11} μCi/ml of a mixture of radionuclides assumed to have the airborne emission distribution shown in Table IV F-9. These exposures are estimated to produce an average annual whole body dose commitment to an individual worker of about 0.20 rem with critical organ dose commitments of about 1.5 rem to the lung and about 2.2 rem to the bone. Applying these occupational exposures

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*Calculated from the results of plant measurements supplied by domestic UF_6 plant operators.

Table IV F-9

ESTIMATED UF_6 INDUSTRY INTEGRATED EMISSIONS FOR PERIOD 1975 THROUGH 2000*

(curies)

Radionuclide	No U or Pu Recycle	With U Recycle	With U & Pu Recycle
		Airborne Emissions	
^{238}U	3.5	3.1	2.7
^{234}U	3.5	3.1	2.7
^{235}U	0.15	0.14	0.12
^{234}Th	2.2	2.0	1.8
^{230}Th	0.0095	0.085	0.076
^{226}Ra	0.010	0.0093	0.0083
^{222}Rn	0.036	0.032	0.028
Total	9.49	8.47	7.43
		Waterborne Emissions	
^{238}U	134	120	100
^{234}U	134	120	100
^{235}U	5.8	5.2	4.4
^{234}Th	136	121	105
^{230}Th	40	36	29
^{226}Ra	1.5	1.3	1.2
Total	451	404	340

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*Based on data from References 29 and 30.

to the total industry population over the entire period from 1975 through 2000 indicates that the integrated whole body dose commitment will range from about 3,500 to 4,400 person-rem, and the integrated maximum critical organ dose commitment will range from about 41,000 to 50,000 bone-rem depending upon whether or not uranium or uranium and plutonium are recycled. The overall significance of these exposure levels, which indicate that either recycle option results in reduced dose commitments, is discussed in CHAPTER IV, Section J.

3.3.4 Accident Considerations

Although several credible accidents associated with UF_6 conversion facilities have a potential for offsite effects, there have been no accidents in either of the two operating facilities that have resulted in any adverse effect on the environment. Summary assessments have been reported for a number of postulated accidents including a solvent extraction circuit fire, a ground level release of UF_6 from a failed product cylinder, a raffinate pond dike failure, a uranyl nitrate hexahydrate (UNH) evaporator failure and an HF storage tank rupture.[36]

Although a fire in the solvent extraction circuit of a wet UF_6 plant might result in low specific activity radioactive materials being carried by the generated smoke, most of the uranium and daughters would remain confined by plating out of the smoke particulates on building and equipment surfaces. Traces carried outdoors would deposit close to the building. It is unlikely that there would be measurable radioactive deposition beyond the plant site. Consequently a wet UF_6 plant fire would produce no significant offsite environmental effects, but could result in a localized cleanup situation.

Rupture or failure of a valve on a hot 14 ton UF_6 cylinder could occur outdoors during handling or transfer. In such an accident, assuming that 9,200 pounds of UF_6 are released before the leak is stopped, an individual 1000 meters downwind from the point of release and remaining in the cloud for the entire duration of the release is estimated to receive a 100 mrem radiation dose to the kidney and a 400 mrem dose to the bone compared with the maximum allowable annual occupational dose of 15,000 mrem. The individual would also be exposed to a hydrogen fluoride concentration of about 10 mg/cu m (about 1% of the lethal concentration) but would hardly be expected to remain in the uranium/hydrogen fluoride cloud for the full duration because this HF concentration is readily sensed and represents about 40% of the level causing eye and respiratory discomfort. The HF intake by an individual exposed for the duration of the release would be about 20 mg. This is less than 1% of the lethal single dose intake of fluoride.

A catastrophic failure of the dike of a retention pond containing solvent extraction raffinate could release as much as 8 million gal of contaminated water containing about 1.2×10^{-7} μCi/ml of uranium, 1×10^{-6} μCi/ml of radium-226, and 1×10^{-8} μCi/ml of thorium-230. In the case of the existing wet process facility, the flow in the nearby river would be adequate to reduce these concentrations to levels below the maximum permissible concentrations. However, the chemical content of the raffinate pond could be sufficiently toxic to cause localized fish kills in the plume until river flow has completely diluted the material. This effect would be of short duration and temporary in its effect on the local fish population.

The UNH evaporator at a solvent extraction plant might contain about 2,000 gallons of high specific gravity uranyl nitrate. Adjacent to this could be an equal volume of product stored in a surge tank. Both of these would be installed in a curbed area capable of containing the total volume of both vessels.

A spill might occur as the result of a line break or a valve failure. If the UNH material did leak, it would be caught and retained in the curbed area and only mists and sprays would be released beyond the immediate area with no significant release to the environs offsite. All of the activity involved is associated with freshly separated uranium of low specific activity.

Historically, only one explosion has been associated with evaporator operation, and that resulted from a "red oil" reaction. The "red oil" type of reaction can occur when the temperature exceeds 226°F. This is prevented by continuous venting to the atmosphere and limiting temperatures to a safe level.

When large quantities of acid are stored at any industrial facility there is some likelihood of a spill or leak. For UF_6 production both nitric acid and hydrofluoric acid may be stored in large quantities. Of these, HF poses the greater problem in the event of a leak. Design features, structures and practices are aimed at reducing the incidence and consequences of such accidents. It is common practice to install acid storage tanks in curbed areas with sufficient volume to contain the entire volume of the storage vessel. In addition, the curbed area may contain crushed limestone providing a quantity in excess of that required to neutralize all of the acid available.

In the extremely unlikely event of a catastrophic failure of an acid storage tank there could be some release of vapor and spray in the immediate area of the storage yard. The duration of the release would be relatively short and would not constitute a potential insult to life forms beyond the area in close proximity to the storage facility any more than if the incident occurred in any chemical process industry plant treating non-radioactive materials.

The ultimate consequences of an acid spill would amount to requiring cleanup and repair.

3.3.5 Contributions to the General Exposure

The radioactive materials of interest in the UF_6 conversion industry are the naturally occurring members of the ^{235}U and ^{238}U decay chains. Because of the removal of a number of these materials in the milling process, the relative concentrations of uranium daughters in the uranium concentrate feed is somewhat different from those found in the ore.

The solvent extraction circuit used in the wet UF_6 conversion process is similar in some respects to that used in many uranium mills, and the reduction, hydrofluorination, and fluorination steps may release small amounts of radioactive materials to the atmosphere and to the river serving the plant. The dry process produces solid wastes containing radioactive materials and also releases some radioactivity to the surroundings and to the nearby river. The resultant dose is projected to be below the normal variation in background dose, and represents no measurable radiological impact.

The estimated whole body dose commitment to the closest theoretical resident, as well as the organ dose commitments to an individual resulting from airborne or waterborne effluents discharged by both wet process and dry process 15,000 MTU/yr UF_6 facilities, are presented in Table IV F-10. It should be pointed out that the air and water pathway commitments are not likely to be additive, because the maximum concentrations are unlikely to occur at the same location with intake by the same individual. These data indicate that the annual individual organ exposure is estimated to range from about 0.003 mrem to the thyroid to about 51 mrem to the bone and annual whole body exposures range from about 0.9 to 33 mrem. These values may be compared with the unrestricted area maximum permissible whole body exposure to an individual of 500 mrem/yr indicated by 10 CFR 20 and the maximum permissible organ exposure of 1,500 mrem/yr recommended by ICRP.

Table IV F-11 presents the integrated total population dose commitment produced by the UF_6 conversion industry over the years 1975 through 2000, without recycle, with uranium recycle and with both uranium and plutonium recycle to light water reactors. These data show that for the critical organ (bone), uranium recycle reduces the total population exposure from about 103,000 person-rem to 91,700 person-rem and plutonium recycle further decreases the population dose commitment to 78,800 person-rem. A detailed discussion of the risks associated with radiological exposures is presented in CHAPTER IV, Section J.

3.4 General Environmental Impact

In this study no credit is taken for the advent of technological advances that might be projected to have an effect on the environmental impact of future UF_6 conversion plants. Accordingly, it is assumed that UF_6 production facilities between the present time and the end of the century will resemble in most respects the plants of today, but will be somewhat larger in size. Thus the integrated environmental impact of the domestic UF_6 conversion industry from 1975 through 2000 would be expected to be greater by a factor of about 50 to 70 over the annual impact of the current industry operating at full rated capacity. The estimated reduction in UF_6 requirements over the 26-year period from about 1,210,000 MTU to 916,000 MTU as a result of the potential decrease resulting from uranium and plutonium recycle would be expected to reduce the gross environmental impact by about 24%.

Review of the estimated environmental impacts of the UF_6 conversion industry indicate that with the exception of natural gas consumption, a few liquid chemical effluents such as sodium, sulfate, ammonia and fluoride, and ^{226}Ra in an impounded

Table IV F-10

DOSE COMMITMENT FROM MODEL URANIUM HEXAFLUORIDE CONVERSION PLANT EFFLUENTS TO CLOSEST THEORETICAL RESIDENT*

Annual Dose Commitment, mrem

ORGAN	WET PROCESS PLANT AIR PATHWAY	WET PROCESS PLANT WATER PATHWAY	DRY PROCESS PLANT AIR PATHWAY	DRY PROCESS PLANT WATER PATHWAY
Whole Body	0.91	3.5	5.5	33.1
GI Tract	0.74	1.6	1.3	1.6
Bone	9.17	19.2	38.8	50.8
Liver	0.33	0.47	1.8	0.04
Kidney	16.3	3.9	29.7	1.4
Thyroid	0.31	0.024	0.5	0.003
Lung	6.7	0.024	13.1	0.003
Skin	0.31	-	0.5	-

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*Based on effluents from 15,000 MTU/yr UF_6 plants. $\chi/Q = 5.4 \times 10^{-6}$ sec/cu m at 500 meters from ground level release. Calculated from effluent data reported in References 29 and 30.

Table IV F-11

INTEGRATED DOSE COMMITMENT* FROM U.S. UF_6 CONVERSION INDUSTRY EFFLUENTS OVER PERIOD 1975 THROUGH 2000

Organ	Total Population Dose Commitment, Person-Rem No U or Pu Recycle	With U Recycle	With U & Pu Recycle
Whole Body**	43,300	38,700	34,200
GI Tract**	5,600	5,000	4,200
Bone**	103,000	91,700	78,800
Liver	680	600	490
Kidney**	11,900	10,700	8,900
Thyroid	50	40	40
Lung	980	880	760
Skin	40	30	30

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*See IV J, Appendix A for explanation of exposure modes and duration incorporated in the dose commitment determination. Calculated from effluent data reported in References 29 and 30.

**Majority of person-rem for these organs derived from waterborne effluent exposure.

liquid discharge, the projected impacts represent an insignificant part of the impact of the whole supporting uranium fuel cycle. Moreover, because the total impact of the items listed above is quite small, it has been concluded that the relative impact from the UF_6 conversion industry over the period from 1975 through 2000 without recycle, with uranium recycle only, or with both uranium and plutonium recycle will also be small. The estimated detailed impacts on the environment of the UF_6 conversion industry projected for the period 1975 through 2000, with and without uranium recycle or with both uranium and plutonium recycle are shown in Appendix A.

4.0 THE URANIUM ENRICHMENT INDUSTRY

4.1 General

Naturally occurring uranium contains about 0.7% of the fissionable isotope ^{235}U. Enrichment of this isotope to a concentration in the range of 2% to 4% is necessary to provide fuel for a light water moderated nuclear reactor.[37] Currently, the only process being used in the United States to produce commercial quantities of enriched uranium by separating ^{235}U from ^{238}U is based on gaseous diffusion technology.

The average velocities of gas molecules at a given temperature depend upon their masses. The gaseous diffusion process is based upon the principle that the rate at which a gas escapes through a small hole in a containment vessel is proportional to the speed of the gas molecules. The enrichment of uranium is accomplished by combining this principle with the use of a porous barrier through which the gas molecules diffuse. Using volatile uranium hexafluoride as the gas, the maximum theoretical enhancement in the isotopic content for a single stage is a factor of 1.0043. This degree of enrichment is multiplied by using a large number of stages, or a cascade. The existing plants utilize approximately 1,200 stages to produce material containing 4% ^{235}U.[38] The gas flow through the cascade is provided by compressors driven by electric motors. The compression of gas generates heat that is discharged into the environment. In addition to the enriched UF_6 product, the gaseous diffusion plants produce UF_6 depleted in ^{235}U, called tails. The tails assay at which an enrichment plant is operated is based on a balance of a number of factors that include the availability of uranium feed, the enrichment plant capacity, and electrical power availability. After current improvement programs are completed, the existing ERDA enrichment plants are expected to operate at a tails assay of about 0.3% ^{235}U to sustain the nuclear power plants that have contracted for enrichment services on a long term basis. Therefore, the assay of the tails for this study has been assumed to be 0.3% ^{235}U. This material is stored as solid UF_6 in cylinders at the plants for possible future uses.

Another process now receiving serious consideration as a candidate for meeting the required increased enrichment capability in the 1980's and beyond is the gas centrifuge process, still in the development stage. The apparent advantages of this process include the requirements for only about 10% of the electrical power used in gaseous diffusion, thus producing lower emission of contaminants from fossil fuel plants supplying a portion of the needed electricity for the enrichment plants and lowered heat rejection to the atmosphere. On the other hand, a centrifuge enrichment plant generates a larger solids discard stream than a diffusion plant of like capacity.

Because it appears that gas centrifuge plants can be built on a considerably smaller scale than diffusion plants, at a lower cost, such plants may provide a greater opportunity for competitive participation by private enterprise. Careful evaluation of these factors will be required before a process selection is made for future plants.

A large number of other isotope separation processes based on various mechanisms have been studied, including

- Phase Equilibrium Processes such as fractional distillation, gas-liquid absorption, liquid-liquid extraction, fractional sublimation or fractional crystallization
- Chemical Exchange Processes such as exchange distillation, gas-liquid chemical exchange, liquid-liquid chemical exchange, exchange chromatography and ion exchange
- Diffusion Processes such as sweep and mass diffusion, carrier diffusion, thermal diffusion, diffusion with the application of electric fields, etc.
- Molecular Flow Processes
- Aerodynamic Processes such as the Becker separation nozzle process; the vortex tube and the Fenn-shock separation process
- Photoexcitation Processes such as lasers
- Electromagnetic Processes
- Nuclear Spin Processes

To date, evaluation of these methods indicates that a commercial facility employing one of these processes is not likely to produce significant quantities of separative work in the time scale of interest.[39]

4.2 Current Status

At the present time all of the enrichment services for the U.S. nuclear industry are provided by three government owned and contractor operated gaseous diffusion plants (GDP's). The facilities at Oak Ridge, Tennessee, and at Paducah, Kentucky, are operated by Union Carbide Corporation, Nuclear Division, and the plant at Portsmouth, Ohio, is operated by the Goodyear Atomic Corporation.

The Oak Ridge and Portsmouth plants were built on sites originally chosen for their remote location, low population density, and availability of low cost electrical power. The Paducah plant was built on a government owned site previously occupied by an ordnance works.[40] Population density in the vicinity of the plants ranges from 30 to 40 people per sq mi, the region within a 50 mile radius of the Oak Ridge plant having the highest population density, primarily because of the proximity of Knoxville, Tennessee. Cooling water is obtained from the Clinch River at Oak Ridge, from the Ohio River at Paducah, and from subterranean wells at Portsmouth. The bulk of the process heat load, however, is discharged to the atmosphere via forced draft cooling towers. Water for the three plants was required at the rate of approximately 23 million gal/day at 1972 power levels.[41] Makeup water is required to replace cooling tower evaporation, windage, and blowdown losses. Blowdown is the removal of a portion of the recirculating cooling water containing concentrated dissolved salts to prevent their buildup in the system.

Power for the gaseous diffusion plants is drawn from the grids of three utilities: Tennessee Valley Authority (TVA), Ohio Valley Electrical Corporation, and Electric Energy, Incorporated. These utilities generate power primarily in large coal-fired stations;[42] TVA is rapidly increasing its power availability from nuclear fueled stations.

The existing facilities were built during the years 1943 to 1955 to meet enriched uranium requirements for military use. With the decline of such needs, the output of the three plants was reduced substantially in the 1960's and reached a power demand of 1,900 megawatts (MW) by 1970. At a production level of 10,500,000 separative work units (SWU) per year in 1972 the electrical power load was about 3,250 MW.[41] At full capacity, the three plant complex requires about 6,100 MW of electrical power while operating at a production level of 17,200,000 SWU/yr.

The Federal government is currently engaged in modifying and uprating the three gaseous diffusion plants to increase their capacity to produce enriched uranium for the nuclear power industry. The program is estimated to cost $1.2 billion (1975 dollars) and to result in about a 60% increase in productive capacity to yield an annual capacity of 27,700,000 SWU by the year 1981. This capacity increase will result from two currently authorized programs; the Cascade Improvement Program (CIP)[43] and the Cascade Uprating Program (CUP),[43] and will be accompanied by an increase in electrical power requirements from 6,100 MW, the maximum power demand of the original plants, to 7,380 MW.[44]

4.3 Projected Uranium Enrichment Industry

Despite the CIP expansion and CUP uprating in progress, the projected domestic and foreign demand for enriched uranium, including all existing ERDA commitments, will exceed the production capability of existing enrichment plants in the first half of the 1980's. With domestic nuclear power reactors alone considered, the present GDP's could satisfy domestic separative work demands until sometime between 1990 and 2000. For this study, the entire output of the present GDP's (improved and uprated) is assumed to be available exclusively to satisfy domestic demands, except for government requirements estimated at 1.2×10^6 SWU/yr.

For the projected growth of U.S. nuclear power alone, over the period 1975 through 2000, the U.S. uranium enrichment industry will have to supply domestic enrichment needs of about 608 million SWU if neither plutonium nor recycled uranium is used in LWR fuel, 613 million SWU if uranium is recycled, or 523 million SWU if plutonium and uranium are both recovered from spent LWR fuel and recycled as mixed oxide fuel. Separative work production would of course have to be greater in any of the cases if U.S. enriching facilities are called upon to supply foreign needs.

It might be expected that reenrichment of recycled uranium that has been recovered from spent LWR fuel and contains about 0.8% ^{235}U should require less separative work than natural uranium feed containing 0.7% ^{235}U to attain the desired fuel grade assay. The recycled uranium also contains a small amount of ^{236}U, which was formed by neutron capture in ^{235}U in a previous reactor cycle. Because ^{236}U also parasitically absorbs neutrons in the reactor, additional ^{235}U is required when ^{236}U is initially present in the fuel to compensate for this fuel degradation. As indicated in the projected separative work requirements, the net result of these offsetting effects is a slight increase in separative work needed when recovered uranium is recycled to LWR's. Further discussion of the neutronic penalty for the ^{236}U content of the recycled fuel is presented in CHAPTER VIII, Appendix VIII B.

The additional industry requirements may be met by expanding the existing gaseous diffusion facilities, by constructing additional gaseous diffusion plants on new sites, by installing a number of gas centrifuge facilities, or by any combination of these approaches. Considerations for selecting the separation process to be used in the industry expansion are considered to be outside the scope of this study. A possible schedule of new plant construction through the year 2000, given in Draft ERDA-1543,[45] calls for the first plant to be a gaseous diffusion facility with subsequent enrichment plants using the centrifuge process. However, the centrifuge process may be demonstrated as being commercially practical soon enough to be available for all the new plants needed.* Projections for this study are based on both possibilities. If it is assumed that the new or expanded enrichment facilities will be added in incremental units having a capacity of 8.75 million SWU/yr (one-half of the combined capacities of existing U.S. gaseous diffusion plants),[46] three additional facilities will be needed to meet domestic enrichment requirements through the year 2000 if neither plutonium nor recycled uranium is used in the fuel. The number of new enrichment plants required remains at three if only uranium is recycled, but drops to two if both uranium and plutonium are recycled. These data, with estimated electrical power requirements for the enrichment industry are summarized in Table IV F-12.

It should be pointed out that centrifuge plants with annual capacities of less than 8.75 million SWU may be built at lower cost. Installation of a larger number of small plants is not expected to create an overall greater environmental impact.

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*Another possibility is that in keeping with the assumption above that the present GDP enterprise is exclusively available for domestic needs, the first new U.S. enrichment plant would be necessary for our foreign commitments. The new plant would probably be built in the 1980's and could employ gaseous diffusion. The potential environmental impact of such a plant is not considered to be within the scope of this study.

Table IV F-12

INTEGRATED DOMESTIC REQUIREMENTS OF U.S. URANIUM ENRICHMENT INDUSTRY

1975 THROUGH 2000

	No U or Pu Recycle	U Recycle	U & Pu Recycle
Total Industry Requirements, SWU x 10^6	608.2	613.3	523.4
Supplied by Upgraded Current Facilities SWU x 10^6	487.8	488.4	468.5
Supplied by New Facilities, SWU x 10^6			
a. Assuming the First New Plant Uses Gaseous Diffusion and the Remainder Use the Gas Centrifuge Process:			
Gaseous Diffusion	79.85	81.35	53.9
Gas Centrifuge	40.55	43.55	1.0
b. Assuming All New Plants Use the Gas Centrifuge Process	120.4	124.9	54.9
Number of Additional 8.75 x 10^6 SWU/yr Plants Needed	3	3	2
Electric Power			
a. Assuming the First New Plant Uses Gaseous Diffusion and the Remainder Use the Gas Centrifuge Process:			
MW (Year 2000)	9,727	9,741	9,483
MW-hr (1975-2000)	1,340 x 10^6	1,346 x 10^6	1,223 x 10^6
b. Assuming All New Plants Use the Gas Centrifuge Process:			
MW (Year 2000)	7,567	7,581	7,323
MW-hr (1975-2000)	1,167 x 10^6	1,170 x 10^6	1,107 x 10^6

Basis: 7,060 MW in present plants after currently authorized improvement and upgrading programs (CIP and CUP) are completed producing 26.5 x 10^6 SWU/yr for power plants. Total capacity of present plants is 27.7 x 10^6 SWU/yr (1.2 x 10^6 SWU/yr for government) and total power is 7,380 MW.

2,400 MW per 8.75 x 10^6 SWU/yr in new gaseous diffusion plant.

240 MW per 8.75 x 10^6 SWU/yr in new gaseous centrifuge plant.

4.4 Environmental Considerations

The environmental impacts of uranium enrichment operations are described in detail in the draft "Environmental Statement, Expansion of U.S. Uranium Enrichment Capacity," ERDA-1543,[45] as well as in an earlier document, "Environmental Survey of the Uranium Fuel Cycle," WASH-1248.[47] The more significant environmental effects associated with uranium enrichment are discussed below.

4.4.1 Use of Natural Resources

The requirement of continuous processing through hundreds or thousands of stages makes the present gaseous diffusion plants among the largest industrial facilities in the world with respect to land area under roof. Of the total 1,500 acres devoted to the combined three sites, about 425 acres have been used for buildings, roadways, or storage facilities. Similar commitments of land would be expected for future installations of enrichment facilities on new sites, i.e., about 400 total acres for an 8.75 million SWU/yr gaseous diffusion plant and about 350 acres for a like capacity gas centrifuge plant. Additional land requirements for new plants adjoining the existing enrichment facilities are expected to be considerably less if the required added production capacity is provided in this manner. For this study, however, it has been assumed that the new enrichment plants would be built at separate sites. In all cases, essentially none of the land is committed permanently and could be returned to some active useful purpose after the plant life is completed.

In the year 1972, operation of cooling towers used to dissipate the waste heat from the gaseous diffusion complex resulted in the consumption of about 23 million gal/day of water due to evaporation, windage, and blowdown losses. Although the high rate of evaporation occasionally results in localized misting and fogging under certain meteorological conditions, these phenomena are usually confined to the plant site.

Additionally, with once-through cooling, large quantities of water would be discharged to surface water bodies from fossil fired and from nuclear power plants supplying the large amounts of electrical energy required by the enrichment facilities.

Regardless of the process used in the new plants, the uranium enrichment step required about 92 to 93% of the electrical energy consumed in the entire uranium fuel cycle. As indicated in Table IV F-12, from 1975 through 2000, the total energy required by the U.S. enrichment industry is estimated to be about 1,340 million MW-hr with no uranium or plutonium recycle, about 1,346 million MW-hr with uranium recycle, and about 1,223 million MW-hr with both uranium and plutonium recycled to LWR's if one new gaseous diffusion plant is built before new gas centrifuge plant construction starts. If all new enrichment plants use the centrifuge process, the equivalent total energy requirements are 1,167, 1,170, and 1,107 million MW-hr, respectively. These requirements do not include allowances for any enriched uranium stockpile buildup. Although the power generating system producing the electricity consumed by the present enrichment industry is primarily fossile fueled, it is estimated that through the year

2000 approximately 40%* of the total energy needed by the enrichment industry will be supplied by nuclear reactors. Based on these projections, about 292 million metric tons of coal will be needed from 1975 through 2000 to supply the industry electrical energy requirements with no recycle of spent fuel to the nuclear power industry, about 293 million metric tons will be required with uranium recycle, and about 266 million metric tons with recycle of both uranium and plutonium if the first new enrichment plant uses the diffusion process. If all the new enrichment plants are constructed for the centrifuge process, coal requirements would be reduced over the 26 year period by about 9.5 to 13% of the diffusion process values.

4.4.2 Effluents

The primary source of environmental effects associated with the enrichment industry is the gaseous and particulate effluents from the coal fired power stations that must produce a large part of the electrical energy consumed. An additional environmental impact related to power needs is the heat discharged to the atmosphere and to the condenser cooling water at the electrical generation plant sites. If the first new enrichment plant employs gaseous diffusion, thermal releases from power plants in the year 2000 would be increased by the equivalent of about 25 MW by uranium recycle, but reduced by 428 MW if both uranium and plutonium are recycled. At the enrichment plants, heat is dissipated primarily to the atmosphere, and, although occasional misting and fogging results on the plant site from operation of the cooling towers, the thermal impact is insignificant.

Small quantities of airborne fluorides are generated at the enrichment facilities. Ambient air concentrations are reduced to environmentally acceptable levels (< 1 ppb) through the use of fluoride scrubbers. Particulates and oxides of nitrogen and sulfur are also released by the enrichment plants. Conservative estimates of the offsite concentrations of these contaminants yield levels below EPA standards. In any case, the total quantities of these effluents are insignificant in comparison with the combustion products generated by the supporting electric power plants.

There are a number of chemical species in the liquid effluent streams from the enrichment plants, including nitrates, chromates, phosphates, zinc, chloride, and sulfate ions. Chromate is the limiting species. With effluent treatment to reduce the chromium content to an acceptable concentration, additional dilution by the receiving stream reduces all chemical concentrations to a small fraction of the recommended permissible water quality standards. Chemical discharges to the environment are expected to decrease as pollution control programs are implemented, but the magnitude of improvement cannot be estimated to any degree of accuracy at this time and has not been used in this analysis.

Small quantities of radioactivity are released from the enrichment facilities during normal operations, in both gaseous and liquid effluent streams. The net effect

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*In Appendix VIII A which details the environmental impacts for all segments of the nuclear industry, including enrichment, a figure of 33-1/3' has been used.

of fuel recycle over the period 1975 through 2000 is an estimated increase in the total radioactive releases from about 4 curies under the no-recycle option, to about 540 curies with uranium recycle only, or to about 460 curies with both uranium and plutonium recycle.

4.4.3 Occupational Exposure

Personnel employed in the uranium enrichment industry will be exposed to small amounts of beta and gamma radiation as well as to airborne radiological emmissions. The 1970-1974 annual data for persons involved in Oak Ridge Gaseous Diffusion Plant (ORGDP) operations who were potentially exposed to beta and gamma radiation are given in Table IV F-13. The average number of people on whom measurements were made was 121 per year over a five-year period. The average doses, 0.042 rem/yr from gamma radiation and 0.25 rem/yr from beta radiation, represent about 0.8% and about 1.5%, respectively, of the permissible whole body and skin doses in 10 CFR 20. These doses are below the normal variation in background dose and represent no measurable radiological impact.

Table IV F-13

ORGDP DOSE DATA

Year	Number of People Examined	Dose in mrem/yr Penetrating (Gamma)	Skin (Gamma + Beta)	Beta (by Difference)
1970	115	58	701	643
1971	119	46	312	266
1972	117	50	242	192
1973	124	23	161	138
1974	128	35	67	32
5 Year Averages	121	42	297	254

Individual doses in the year 2000 can be expected to be about the same for each of the three fuel cycle options as is currently experienced at ORGDP. For plants using the same process, the numbers of employees potentially exposed to beta and gamma radiation are assumed to be directly related to production activity. Moreover, the number of potential exposures in a centrifuge enrichment plant is estimated to be double that in a diffusion plant because there are twice as many people employed in areas of exposure potential in a centrifuge facility. On these bases, with the first new enrichment plant using gaseous diffusion, the external whole body occupational dose commitments for the total enrichment industry in 2000 are estimated to be 80 person-rem/yr from gamma radiation and about 500 person-rem/yr from beta radiation under the U and Pu recycle option. Under both the U recycle only and the no recycle options, the gamma dose is estimated at 122 person-rem/yr and the beta dose would be 735 person-rem/yr. Again, these doses are below the normal variation in background dose, and represent no measurable radiological impact.

When all new enrichment plants use the centrifuge enrichment process, exposures are somewhat higher: 100 person-rem/yr from gamma radiation and 600 person-rem/yr from beta radiation under the U and Pu recycle option, and about 140 person-rem/yr from gamma radiation and 850 person-rem/yr from beta radiation under the U-only and the no-U no-Pu recycle options.

Internal occupational exposures to operating and maintenance personnel result from inhalation of plant air containing levels of radionuclides that would not exceed 5.5×10^{-12} μCi/ml. Breathing this air is estimated to produce an annual critical organ (lung) dose commitment to the individual employee of about 0.7 rem under any of the three recycle options. Personnel employed in the enrichment industry in the year 2000 who are potentially exposed to this concentration of radionuclides in plant air would receive a total population dose commitment of about 4,700 to 5,300* lung-rem if neither U or Pu is recycled, 4,700 to 5,300* lung-rem if U only is recycled, or 3,400 to 3,900* lung-rem if both U and Pu are recycled.

4.4.4 Accident Considerations

There are a number of credible accidents associated with uranium enrichment facilities that have a potential for offsite environmental effects. A fire or explosion could release gaseous and liquid chemicals to the environment, but building design will limit the total quantities that could be involved in one accident, and the distance to unrestricted areas enhances dilution of concentrations.

Specific credible process accidents include chemical emissions resulting from leaks in hot and cold UF_6 cylinders and chemical storage facilities such as anhydrous HF tankage. Such accidents would be similar to those discussed in Section 3.3.4 for the UF_6 conversion industry. Hypothetical maximum accidents are discussed in Draft ERDA-1543.[9]

A criticality incident in the low enrichment portions of a diffusion plant is highly improbable. Detailed evaluations of cascade equipment under normal and contingency operations have indicated the inherent nuclear safety of the cascade:

- Where the integrity of the diffusion equipment is not breached, criticality cannot occur in unmoderated uranium hexafluoride in the ^{235}U concentration range involved in light water reactor fuel.

- Criticality is possible in the moderated state; however, accumulation of the necessary quantity of fissile material in a nuclearly favorable configuration is required. Nuclear evaluations have indicated that the abnormal operating contingencies necessary to produce such a situation make the possibility of a criticality condition extremely unlikely.

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*The lower lung-rem value applies to the industry if the first new plant built employs gaseous diffusion and the following use the gas centrifuge. The higher lung-rem value applies to the industry when all new plants use the gas centrifuge.

4.4.5 Contributions to the General Exposure

With no recycle of recovered uranium from LWR spent fuel, the isotopes of interest in enrichment plant effluents are the naturally occurring isotopes ^{234}U, ^{235}U, and ^{238}U. In the event reprocessed uranium is fed to an enrichment plant, reactor produced isotopes ^{232}U, ^{233}U, and ^{236}U, traces of reactor produced plutonium and neptunium and fission product isotopes of technetium, ruthenium, niobium, and zirconium could also be present. Strict acceptance specifications for the feed would limit the radiological and chemical impurities. It is estimated that offsite radiological concentrations in effluent discharges from an 8.75×10^6 SWU/yr enrichment plant with mixed natural and reprocessed uranium feeds will amount to less than 1% of the maximum permissible concentrations listed in Table II, Schedule B of 10 CFR 20.

The population dose commitments resulting from the discharge of airborne and waterborne effluents from the entire projected uranium enrichment industry from 1975 through 2000 are shown in Table IV F-14 for the three recycle options considered in the present study. In contrast to the differences produced by the recycle mode used, only very small differences in dose commitment result from selection of the gas centrifuge process rather than the gaseous diffusion process for the first new enrichment facility.

The estimated annual individual whole body and organ dose commitments to the closest theoretical resident resulting from airborne and waterborne emissions discharged by single 8.75×10^6 SWU/yr gaseous diffusion or gas centrifuge enrichment facility are presented in Table IV F-15. The data indicate the maximum annual individual organ dose commitment to be 31 mrem to the GI tract and the maximum whole body dose commitment is only 1 mrem with both maximum values occurring under the uranium recycle or the uranium and plutonium recycle options. These values represent about 0.2% of the maximum permissible organ dose recommended by ICRP and 0.2% of the unrestricted area maximum permissible whole body dose to an individual specified in 10 CFR Part 20.

In all cases the resultant doses are below the normal radiation background doses and do not represent any measurable radiological impact.

4.5 General Environmental Impact

The estimated impacts of the domestic uranium enrichment industry with no recycle, with uranium recycle only, and with both uranium and plutonium recycle are shown in Table IV F-16 for the following plant construction schemes: (A) when the first new plant uses the gaseous diffusion process and the remaining new plants use the gas centrifuge process; (B) when all new plants use the gas centrifuge process. It should be pointed out that the effluents listed in these tables are based on the discharge data given in Draft ERDA-1543[45] for enrichment activities at gaseous diffusion and gas centrifuge plants. Onsite processes other than enrichment operations, which would produce larger pollutant releases, have not been included in these tabulations. Additional estimated industry impacts are presented in Appendix A.

Table IV F-14

INTEGRATED DOSE COMMITMENT* FROM U.S. URANIUM ENRICHMENT INDUSTRY EFFLUENTS OVER PERIOD 1975 THROUGH 2000

Organ	No U or Pu Recycle (Total person-rem)**	With U Recycle (Total person-rem)**	With U and Pu Recycle (Total person-rem)**
	A. Assuming the First New Plant Uses Gaseous Diffusion and the Remaining New Plants Use the Gas Centrifuge Process		
Whole Body	77	130	110
GI Tract	66	7,070	6,270
Bone	930	950	790
Liver	21	220	190
Kidney	520	2,900	2,550
Thyroid	18	17	15
Lung	490	440	380
Skin	18	110	100
	B. Assuming All New Plants Use the Gas Centrifuge Process		
Whole Body	77	120	110
GI Tract	67	7,350	6,270
Bone	930	950	800
Liver	21	220	190
Kidney	520	3,000	2,550
Thyroid	18	17	15
Lung	490	440	380
Skin	18	120	100

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*See IV J, Appendix A for explanation of exposure modes and duration incorporated in the dose commitment determination.

**Person-rem dose commitment resulting from effluents to the atmosphere and surface water with the major part from airborne emissions.

Table IV F-15

DOSE COMMITMENT FROM MODEL URANIUM ENRICHMENT PLANT EFFLUENTS TO CLOSEST THEORETICAL RESIDENT*

	Annual Dose Commitment, mrem			
	No U or Pu Recycle		With U or U and Pu Recycle	
Organ	Air Pathway	Water Pathway	Air Pathway	Water Pathway
		Centrifuge Enrichment Plant		
Whole Body	0.7	6×10^{-4}	1	0.07
GI Tract	0.6	1×10^{-3}	31	8
Bone	8	9×10^{-3}	8	0.2
Liver	0.2	6×10^{-9}	1	0.2
Kidney	5	2×10^{-3}	14	3
Thyroid	0.2	6×10^{-9}	0.2	0.00007
Lung	6	6×10^{-9}	6	0.02
Skin	0.2	-	1	-
		Gaseous Diffusion Plant		
Whole Body	0.7	3×10^{-4}	1	0.07
GI Tract	0.6	6×10^{-4}	31	8
Bone	8	5×10^{-3}	8	0.2
Liver	0.2	3×10^{-9}	1	0.2
Kidney	5	1×10^{-3}	14	3
Thyroid	0.2	3×10^{-9}	0.2	0.00007
Lung	6	3×10^{-9}	6	0.02
Skin	0.2	-	1	-

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*Based on effluents from 8.75×10^6 SWU/yr centrifuge or gaseous diffusion enrichment plants. $\chi/Q = 5.4 \times 10^{-6}$ sec/cu m at 500 meters from ground level release.

Table IV F-16

SUMMARY OF ESTIMATED ENVIRONMENTAL IMPACTS OF URANIUM ENRICHMENT INDUSTRY OVER PERIOD 1975 THROUGH 2000

(Based on U.S. Requirements Only*)

(A) ASSUMING THE FIRST NEW PLANT USES GASEOUS DIFFUSION AND SUBSEQUENT PLANTS USE THE GAS CENTRIFUGE PROCESS

	No U or Pu Recycle	With U Recycle	With U and Pu Recycle
Industry Production (Millions of SWU)	608.2	613.3	523.4
Number of Facilities	6	6	5
Natural Resources Use			
Land			
(Acres) Temporarily Committed	2,600	2,600	2,250
Acre-Yr Occupancy	46,200	46,550	43,300
Water (Millions of gal)			
Discharged to Air**	423×10^3	425×10^3	389×10^3
Discharged to Water Bodies***	66×10^6	66×10^6	60.4×10^6
Electricity			
Total Requirement (Thousands of MW-hr)	$1{,}340 \times 10^3$	$1{,}350 \times 10^3$	$1{,}220 \times 10^3$
Supplied by Fossil Fuel (Thousands of MW-hr)	804×10^3	808×10^3	734×10^3
Equivalent Coal (Thousands of Metric Tons)	292×10^3	293×10^3	266×10^3
Effluents - Chemical			
Gases (Metric Tons)			
SO_x	3,700,000	3,710,000	3,360,000
NO_x	2,960,000	2,970,000	2,690,000
Solids (cu m)			
Power Plant Sludges	23×10^6	23×10^6	21×10^6
Enrichment Plant Scrap	38,300	40,800	8,600
Thermal (Billions of Btu's)	12.3×10^6	12.3×10^6	11.2×10^6
Radioactive (Curies)			

	Gases			Liquids		
Radionuclide	No U or Pu Recycle	With U Recycle	With U & Pu Recycle	No U or Pu Recycle	With U Recycle	With U & Pu Recycle
^{232}U		2×10^{-3}	2×10^{-3}		3×10^{-4}	3×10^{-4}
^{233}U		1×10^{-5}	1×10^{-5}		2×10^{-6}	1×10^{-6}
^{234}U	3.4	3	2	1×10^{-1}	1×10^{-1}	9×10^{-2}
^{235}U	1×10^{-1}	1×10^{-1}	1×10^{-1}	4×10^{-3}	4×10^{-3}	4×10^{-3}
^{236}U		8×10^{-2}	7×10^{-2}		4×10^{-2}	8×10^{-3}
^{238}U	4×10^{-1}	5×10^{-1}	4×10^{-1}	9×10^{-2}	9×10^{-2}	7×10^{-2}
^{239}Pu		3×10^{-8}	2×10^{-8}		6×10^{-7}	5×10^{-7}
^{237}Np		1×10^{-5}	1×10^{-5}		3×10^{-4}	2×10^{-4}
^{144}Ce		8×10^{-3}	7×10^{-3}		1×10^{-1}	9×10^{-2}
^{137}Cs		8×10^{-3}	7×10^{-3}		1×10^{-1}	9×10^{-2}
^{106}Ru		5×10^{-1}	4×10^{-1}		7	6
^{99}Tc		38	32		491	419
^{95}Zr&^{95}Nb		1×10^{-1}	9×10^{-2}		1	1
Other		8×10^{-3}	7×10^{-3}		7×10^{-3}	1.7×10^{-4}
Total	3.9	42	36	0.2	499	426

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*The environmental impact of the foreign business aspects on the U.S. nuclear industry is not included here. It is covered in the Draft ERDA-1542 "Environmental Statement. U.S. Nuclear Power Export Activities," Vols. I & II, August 1975.

**From enrichment plants only.

***From enrichment plants and power plants in support thereof. Forty percent of the electrical energy used over the period 1975 through 2000 comes from nuclear plants and 60 from coal fired plants.

See Appendix A for estimate of other environmental impacts.

Table IV F-16 (Continued)

SUMMARY OF ESTIMATED ENVIRONMENTAL IMPACTS OF URANIUM ENRICHMENT INDUSTRY OVER PERIOD 1975 THROUGH 2000

(Based on U.S. Requirements Only*)

(B) ASSUMING ALL NEW PLANTS USE THE GAS CENTRIFUGE PROCESS

	No U or Pu Recycle	With U Recycle	With U and Pu Recycle
Industry Production (Millions of SWU)	608.2	613.3	523.4
Number of Facilities	6	6	5
Natural Resources Use			
Land			
(Acres) Temporarily Committed	2,550	2,550	2,200
Acre-Yr Occupancy	45,700	46,000	42,900
Water (Millions of gal)			
Discharged to Air**	370×10^3	370×10^3	353×10^3
Discharged to Water Bodies***	58×10^6	58×10^6	55×10^6
Electricity			
Total Requirement (Thousands of MW-hr)	$1,167 \times 10^3$	$1,170 \times 10^3$	$1,110 \times 10^3$
Supplied by Fossil Fuel (Thousands of MW-hr)	700×10^3	702×10^3	664×10^3
Equivalent Coal (Thousands of Metric Tons)	254×10^3	255×10^3	241×10^3
Effluents - Chemical			
Gases (Metric Tons)			
SO_x	3,220,000	3,230,000	3,050,000
NO_x	2,580,000	2,590,000	2,440,000
Solids (cu m)			
Power Plant Sludges	20×10^6	20×10^6	19×10^6
Enrichment Plant Scrap	97,600	98,400	48,200
Thermal (Billions of Btu's)	10.7×10^6	10.7×10^6	10.1×10^6
Radioactive (Curies)			

	Gases			Liquids		
Radionuclide	No U or Pu Recycle	With U Recycle	With U & Pu Recycle	No U or Pu Recycle	With U Recycle	With U & Pu Recycle
^{232}U		2×10^{-3}	2×10^{-3}		3×10^{-4}	3×10^{-4}
^{233}U		1×10^{-5}	1×10^{-5}		2×10^{-6}	2×10^{-6}
^{234}U	3.4	3	2	1×10^{-1}	1×10^{-1}	9×10^{-2}
^{235}U	1×10^{-1}	1×10^{-1}	9×10^{-2}	5×10^{-3}	5×10^{-3}	4×10^{-3}
^{236}U		8×10^{-2}	7×10^{-2}		1×10^{-2}	9×10^{-3}
^{238}U	4×10^{-1}	4×10^{-1}	4×10^{-1}	1×10^{-1}	1×10^{-1}	8×10^{-2}
^{239}Pu		3×10^{-8}	2×10^{-8}		6×10^{-7}	5×10^{-7}
^{237}Np		1×10^{-5}	1×10^{-5}		3×10^{-4}	2×10^{-4}
^{144}Ce		8×10^{-3}	7×10^{-3}		1×10^{-1}	9×10^{-2}
^{137}Cs		8×10^{-3}	7×10^{-3}		1×10^{-1}	9×10^{-2}
^{106}Ru		5×1^{-1}	4×10^{-1}		7	6
^{99}Tc		38	32		491	419
^{95}Zr&^{95}Nb		1×10^{-1}	9×10^{-2}		1	1
Other		8×10^{-3}	7×10^{-3}		2×10^{-2}	9×10^{-3}
Total	3.9	42	36	0.2	499	426

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*The environmental impact of the foreign business aspects on the U.S. nuclear industry is not included here. It is covered in the Draft ERDA-1542 "Environmental Statement. U.S. Nuclear Power Export Activities," Vols. I & II, August 1975.

**From enrichment plants only.

***From enrichment plants and power plants in support thereof. Forty percent of the electrical energy used over the period 1975 through 2000 comes from nuclear plants and 60% from coal fired plants.

See Appendix A for estimate of other environmental impacts.

The most significant impact differences between the three recycle options are as follows:

- If both uranium and plutonium are recycled, the enrichment requirements in the years 1975-2000 would be reduced by 85 million separative work units in a comparison with requirements under no recycle option. This reduction would result in a decrease in the total number of enrichment plants from six to five. Eliminating the need for this one facility would represent a capital investment saving in excess of $3 billion.

- The reduced separative work demand of the uranium and plutonium recycle option would decrease electrical energy requirements by about 117 million megawatt hours over the period 1975 through 2000, if the first new plant is gaseous diffusion and the remaining plants centrifuge, but by only 60 million megawatt hours if all new plants are centrifuge facilities.

- Assuming 60% of the required electrical energy is generated by coal fired plants, the decreased demand over the period from 1975 through 2000 with the uranium and plutonium recycle option represents coal saving of about 26 million metric tons over the no recycle option if the first new plant is gaseous diffusion, and about 13 million metric tons if all new plants employ the centrifuge process.

- The reduced coal consumption that would result with the uranium and plutonium recycle option would decrease generation of airborne pollutants when compared with the no recycle option. If the first new plant uses gaseous diffusion and the remaining use the gas centrifuge process, over the period 1975 through 2000 sulfur oxides would be reduced by about 340,000 metric tons, and oxides of nitrogen would be reduced by about 270,000 metric tons. The reductions if all plants employed the centrifuge process would be about 170,000 metric tons of SO_x and about 140,000 metric tons of NO_x.

5.0 THE URANIUM FUEL FABRICATION INDUSTRY

5.1 Current Status

The feed material used in fabrication of fuel for light water reactors is uranium hexafluoride that has been enriched to about 2. to 4 in the ^{235}U isotope at the enrichment plants. The enriched UF_6 is converted to UO_2 powder, which is formed into pellets and then loaded and sealed into Zircaloy tubes by capping and welding.

Currently, essentially all of the LWR UO_2 fuel is produced by use of one of the conventional wet ammonium diuranate (ADU) processes, although alternative dry conversion processes have been developed[48] and are being utilized by one facility. Two other facilities are using the dry process for part of their production. The ADU process involves volatilizing the enriched UF_6 received in sealed 2-1/2 ton cylinders, hydrolyzing the UF_6 to form a solution of UO_2F_2, precipitating ammonium diuranate by the addition of ammonium hydroxide, dewatering the ADU by centrifuging or filtering, then drying and reducing the ADU to UO_2 powder in a cracked ammonia atmosphere. The UO_2 powder is then pretreated to obtain the desired consistency, formed into pellets, sintered to the required density, ground to the finished dimensions, and, finally, washed and dried. The completed pellets are loaded into fuel rods, which are then sealed by welding on end caps, and the fuel rods are assembled to form finished fuel assemblies ready for loading into the reactor.

Although most of the off specification material produced during these operations can be recycled with a minimum of rework, some must be sent through a scrap recovery cycle.

The present fuel fabrication industry consists of 9 commercial plants, listed in Table IV F-17, each of which performs part or all of the fuel fabrication operation. Three facilities currently produce complete light water reactor fuel assemblies starting with enriched UF_6 as the feed material; two others start with UO_2 powder or UO_2 pellets to produce fuel assemblies. The four remaining facilities simply produce UO_2 powder or UO_2 powder and pellets from enriched UF_6 as feed for fuel assembly plants. Current capacity of the industry is about 2,700 MTU/yr as fuel assemblies.

Table IV F-17

CURRENT LWR FUEL FABRICATION PLANTS[49]

Licensee	Plant Location	Plant Feed Material	Plant Product
Babcock & Wilcox	Lynchburg, Va.	UO_2 Powder	Fuel Assemblies
Babcock & Wilcox	Apollo, Pa.	UF_6	Fuel Assemblies*
Combustion Engineering	Hematite, Mo.	UF_6	UO_2 Powder or Pellets
Combustion Engineering	Windsor, Conn.	UO_2 Powder	Fuel Assemblies
Exxon Nuclear	Richland, Wash.	UF_6	Fuel Assemblies
General Electric	Wilmington, N.C.	UF_6	Fuel Assemblies
Kerr-McGee	Cimarron, Okla.	UF_6	Fuel Assemblies*
Nuclear Fuel Services	Erwin, Tenn.	UF_6	UO_2 Powder or Pellets
Westinghouse	Columbia, S.C.	UF_6	Fuel Assemblies

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*Not currently producing fuel assemblies being used in LWR's.

5.2 Projected Fuel Fabrication Industry

The present annual industry capacity of 2,700 MTU is adequate to meet LWR needs through the year 1979 whether or not uranium or uranium and plutonium are recycled.

Based upon increasing reactor fuel demands and the fact that contracts for fuel are negotiated several years in advance, plans have been formulated by industry to expand capacity to about 8,200 MTU/yr by about the year 1980. It is estimated that this quantity will be fabricated in 6 plants and that most of the increase will be from the expansion of existing facilities.[50]

Projections indicate that by the end of the 20th century an annual industry capacity of about 13,500 MTU will be required, with or without recycle of uranium, and 10,900 MTU will be required if both uranium and plutonium are recycled.* If there is no recycle or if only uranium is recycled, this would require a capacity increase of 5,300 MTU/yr, probably achieved by building 3 new 1,500 MTU/yr plants and expanding existing facilities. If uranium and plutonium were both recycled, the required increase in capacity would be 2,700 MTU/yr, which would probably require the addition of only 2 new 1,500 MTU/yr plants.

The two or three plants constructed after the year 1980 would probably use the dry process for converting UF_6 to UO_2 powder. This process utilizes fluid bed hydrolysis-reduction techniques and has been under development for some time. Although this process change would result in a substantial reduction of both gaseous and liquid chemical effluents (by eliminating the use of ammonia and recovering fluoride as a salable byproduct) no credit is taken in this study in estimating the quantity of effluents released by the industry.

Thus, the industry at the turn of the century is estimated to consist of 6 pre-1980 facilities using the wet ADU process to convert UF_6 to UO_2 plus two or three 1,500 MTU/yr plants using the dry conversion process. All new facilities would be capable of producing complete fuel assemblies starting with enriched UF_6 as the feed material.

5.3 Environmental Considerations

A detailed description of the environmental impact of current uranium fuel fabrication operations is presented in the Environmental Survey of the Uranium Fuel Cycle.[51] The environmental impact of the uranium fuel fabrication industry is not affected significantly by the recycle of uranium or the recycle of both uranium and plutonium. The impact on the environment from UO_2 fuel fabrication is relatively small compared with that produced by other fuel cycle steps.

5.3.1 Use of Natural Resources

For this study it is assumed that the total land in use by the industry in the year 2000 will be about 8,200 acres with no recycle or with uranium recycle and about 7,200 acres if both uranium and plutonium are recycled. The nine plants currently

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*This statement assumes that natural UO_2 would be used in mixed oxide and that the natural UO_2 would be supplied by the UF_6 plants as an intermediate product.

engaged in various aspects of the uranium fuel fabrication industry are located on sites ranging in size from a few acres to 1,650 acres for a total of about 5,200 acres.[51] The 1,500 MTU/yr plants coming on stream after the year 1980 are assumed to be situated on 1,000 acre sites where about 25 acres are used for plant and ancillary service areas and, to maintain consistency with all other supporting uranium fuel cycle industry sites, there is a buffer zone of at least 500 meters to the closest site boundary. All of this land can be fully reclaimed for other uses after fuel fabrication operations cease.

Most of the water used in the fuel fabrication industry is required for the cooling of process equipment and does not come in contact with uranium or process chemicals during plant operation. The cooling water is used to dilute any contaminated liquid process wastes prior to release offsite and the estimated total industry discharge of about 9 million gal/day should not constitute a significant environmental impact.

Other natural resources consumed by the fuel fabrication industry include electrical power and natural gas to supply process and building heat.

5.3.2 Effluents

The only airborne chemical effluent of any significance from the processing operations is hydrogen fluoride, which is evolved in converting the UF_6 feed to UO_2. The fluoride wastes are currently removed effectively from the gaseous effluent streams by liquid scrubbing and result in site boundary concentrations of fluorides well within permissible limits.

The most significant chemicals contained in liquid effluents are nitrogen compounds generated in the UF_6 to UO_2 conversion step and in scrap recovery operations. The limiting release for nitrogen compounds in liquid effluents is that of ammonia. This nitrogen compound requires a reduction in concentration by a factor of about 1,000 to reach recommended levels for surface water. This reduction can be achieved by natural dilution, if plant effluents are released to large streams. Reductions in concentration of ammonia in liquid effluents could also be achieved by reducing the quantity of ammonia released through installation of an ammonia recovery system, or by using the dry conversion process.

Currently in the ADU process, the scrubber product is combined with other process liquids and treated with lime to precipitate calcium fluoride, which is filtered from the liquid and buried on site. The small quantity of fluoride remaining in solution is released and the concentration is reduced to acceptable levels by dilution in the receiving stream.

The very small quantities of radionuclides now released in the effluent gases and liquids result in concentrations of radioactive material in these media that are about 0.04% and 10%, respectively, of 10 CFR 20 limits at the site boundary. A small amount of uranium is contained in the calcium fluoride solids precipitated from the scrubber liquids. The resulting concentration of less than 5 μCi/lb of solid is not considered to be a health hazard.

The estimated emissions of radioactive materials from the entire uranium fuel fabrication industry over the period 1975 through 2000 are presented in Table IV F-18. Trace quantities of ^{232}U and ^{233}U, along with possible trace amounts of reactor products and fission products, could be present in the enriched UF_6 feed produced from recovered uranium recycled to the enrichment facilities. However, it is estimated that because of the extremely small quantities involved, these radioisotopes, if present, would not contribute measurably to either the occupational exposure or the general population dose commitment attributed to the uranium fuel fabrication industry. The total quantity of radioactive material released as airborne effluents over the 26 year period is estimated at about 1.3 curies. About 310 or 300 curies would be released in liquid discharges from the entire UO_2 fuel fabrication industry depending upon whether or not uranium is recycled. If both uranium and plutonium are recycled, industry emissions are estimated to be reduced by about 13% because total throughput will be reduced by this percentage. The small increases in both airborne and waterborne emissions resulting from uranium recycle, as shown in Table IV F-18, are attributed to the added radioactivity contributed by the ^{236}U content of the uranium recovered from spent fuel.

The thermal load of process cooling water is dissipated to the atmosphere as the water passes through holding ponds and has no adverse environmental effect.

5.3.3 Occupational Exposure

Occupational exposures result from external radiation and from inhalation during normal working hours of air containing about 9.7×10^{-12} μCi/ml of a mixture of radionuclides assumed to have the distribution shown in Table IV F-18. These exposures are estimated to produce an annual whole body dose commitment to an individual of about 0.26 rem along with a critical organ dose commitment of about 9.95 rem to the lung. For comparison, this estimated annual total body dose commitment is about 5% of the 10 CFR 20 limits for annual occupational exposure, or about double that resulting from natural background. The estimated occupational dose commitment to the lung is about two-thirds of the exposure that would be received by an individual breathing air containing the maximum permissible concentration of uranium as specified in 10 CFR 20.

Applying these dose commitments to the total population of the industry for each year over the 26 year, 1975-2000 period and totaling them produces the integrated dose commitments for the entire industry over that period for each of the three recycle options. Personnel employed in the UO_2 fuel fabrication industry are exposed to external beta and gamma radiation as well as inhaled airborne slightly enriched uranium and uranium daughters. Whole body occupational integrated dose commitments for the 26 year period would be 50,500 person-rem for no recycle or recycle of uranium only and 43,800 person-rem for recycle of uranium and plutonium. Critical organ

Table IV F-18

ESTIMATED INTEGRATED UO_2 FUEL FABRICATION INDUSTRY EMISSIONS FROM 1975 THRU 2000* (curies)[5]

Radionuclide	No U or Pu Recycle	With U Recycle	With U & Pu Recycle***
		Airborne Emissions**	
^{234}U	1.00	1.00	0.87
^{235}U	0.028	0.028	0.024
^{236}U	0.00	0.042	0.036
^{238}U	0.12	0.12	0.11
^{234}Th	0.12	0.12	0.11
	1.268	1.310	1.150
		Waterborne Emissions	
^{234}U	236	236	205
^{235}U	6.4	6.4	5.6
^{236}U	0.0	9.8	8.5
^{238}U	29.4	29.4	25.5
^{234}Th	29.4	29.4	25.5
	301.2	311.0	270.1

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*Estimates based upon 1,500 Mt/yr plants using ADU process.

**Based upon ventilation system having scrubbers or roughing filters and one final HEPA filter.

***Emissions are some 13% lower for recycle of both U and Pu because throughputs are lower.

(lung) integrated dose commitments for the industry are estimated at about 1,952,000 person-rem, 1,955,000 person-rem, and 1,693,000 person-rem for no recycle, recycle of uranium only and recycle of both uranium and plutonium, respectively. The estimated dose commitments for recycle of both uranium and plutonium are reduced by 13% because of a like reduction in industry throughout.

The overall significance of these levels of exposure is discussed in CHAPTER IV, Section J.

5.3.4 Accident Considerations

To date there have been no offsite environmental effects from accidents in low enriched UO_2 fuel fabrication plants. There has been a low incidence of accidental occurrences in these plants as a result of strong safety oriented plant design and operations.[51]

Three accidents that have the potential for significant offsite effects were considered:[51]

- Rupture of an overfilled UF_6 cylinder with the resulting UF_6 hydrolysis producing a release of hydrogen fluoride and a soluble uranium compound (UO_2F_2)

- An explosion in a reduction furnace which would release uranium in an insoluble form

- A criticality accident resulting in neutron and gamma radiation and the release of fission products

The UF_6 release would result in a dose commitment to individuals at the site boundary of about 4 mrem to the bone* and about 1.25 mrem to the kidneys.* These doses would be small fractions of the annual average exposure of 130 mrem to the whole body from natural background. The HF concentration at 500 meters is estimated to be about 0.14 mg/cu m, which is less than 10% of the 2 mg/cu m threshold limit for an 8 hour per day occupational exposure.[51]

The hypothetical criticality accident (the U.S. fuel cycle industry has never experienced an accidental criticality with low enriched uranium) could result in dose commitments to the whole body of individuals at the site boundary of about 50 mrem and a thyroid dose commitment of about 260 mrem.* This total body dose commitment is about 40% of the average annual exposure that an individual receives from natural background radiation.

The hypothetical furnace explosion could result in dose commitments to individuals at the site boundary of 1×10^{-4} mrem to the lung,* which is below the normal variation in background dose and represents no measurable radiological impact.

Occupational dose commitments and chemical exposures resulting from accidental releases are also estimated to be quite low because of employee training in accident procedures and use of appropriate protective equipment.

5.3.5 Contributions to the General Exposure

The primary isotopes of interest in the UO_2 fuel fabrication industry are ^{234}U, ^{235}U, ^{236}U, ^{238}U, and ^{234}Th. Non uranium members of the ^{238}U chain beyond ^{234}Th are not considered to be present in sufficient quantities to constitute a significant radiological consideration. The estimated annual whole body dose commitment plus the organ dose commitments to the closest theoretical resident produced by airborne or waterborne effluents discharged from a model UO_2 fuel fabrication plant operating under each recycle option are presented in Table IV F-19. It is highly unlikely that the air and water pathway commitments would be additive for a single individual, because this would require both maximum concentrations to occur at the same point with

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*These exposures were estimated at a site boundary that is 500 meters from the release. Thus, these exposures are 1/25 of those estimated in the referenced document at a site boundary that is 100 meters from the release.

the maximum intake by the same individual. This tabulation indicates that the annual single organ dose commitment ranges from about 4.7 x 10^{-6} mrem/yr to the thyroid or lung up to about 4.1 mrem/yr to the bone from waterborne effluents and from a minimum of about 0.03 mrem/yr to the thyroid or skin to a maximum of 1.3 mrem/yr to the bone from airborne effluents. The data also indicate estimated whole body dose commitments range from about 0.1 mrem/yr via the air pathway to about 0.25 mrem/yr via the water pathway. The minor differences in values indicated between the recycle and no recycle operating options is due to the small quantity of ^{236}U contained in the recycled uranium.

Table IV F-19

DOSE COMMITMENT FROM MODEL UO_2 FUEL FABRICATION PLANT EFFLUENTS TO CLOSEST THEORETICAL RESIDENT*

	DOSE COMMITMENT, mrem/yr			
	NO U OR PU RECYCLE		WITH U OR U AND PU RECYCLE	
ORGAN	AIR PATHWAY	WATER PATHWAY	AIR PATHWAY	WATER PATHWAY
Whole Body	0.11	0.25	0.11	0.26
GI Tract	0.094	0.44	0.10	0.45
Bone	1.3	4.1	1.3	4.3
Liver	0.034	8.2 x 10^{-6}	0.035	8.2 x 10^{-6}
Kidney	0.73	0.97	0.74	1.0
Thyroid	0.032	4.7 x 10^{-6}	0.032	4.7 x 10^{-6}
Lung	0.98	4.7 x 10^{-6}	1.0	4.7 x 10^{-6}
Skin	0.032	---	0.032	---

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*Based on effluents from UO^2 fuel fabrication plants processing 1500 MTU/yr for each option χ/Q = 5.4 x 10^{-6} sec/cu m at 500 meters from ground level release.

Table IV F-20 presents the integrated total population* dose commitment produced by the UO_2 fuel fabrication industry over the period 1975 through 2000 without recycle, with the recycle of uranium, and with recycle of both uranium and plutonium to light water reactors. These data show that for the critical organ (bone), uranium recycle increases the total population dose commitment from 40,600 person-rem to 41,600 person-rem because of the ^{236}U content of the recycled fuel; with recycle of both uranium and plutonium, the dose commitment is decreased to about 36,000 person-rem because of the decrease in total uranium requirements from about 189,000 metric tons to about 163,000 metric tons. A discussion of the risks associated with radiological dose commitments is presented in CHAPTER IV, Section J.

Table IV F-20

INTEGRATED DOSE COMMITMENT* FROM U.S. UO_2 FUEL FABRICATION INDUSTRY

EFFLUENTS OVER PERIOD 1975 THROUGH 2000

Organ	Total Population Dose Commitments, person-rem		
	No U or Pu Recycle	With U Recycle	With U & Pu Recycle
Whole Body**	2,400	2,600	2,200
GI Tract**	2,700	2,800	2,400
Bone**	40,600	41,600	36,000
Liver	5.2	5.2	4.5
Kidney**	6,500	6,800	5.900
Thyroid	4.6	4.7	4.0
Lung	140	150	130
Skin	4.6	4.7	4.0

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*See CHAPTER IV, Section J, Appendix A for explanation of exposure modes and duration incorporated in the dose commitment determination.

**Majority of person-rem for these organs derived from waterborne effluent exposure. Majority of other organ person-rem derived from airborne effluents.

5.4 General Environmental Impact

It is expected that the UO_2 fuel fabrication facilities in existence at about the end of the century will be similar to those of today, except that the plants and the plant sites will be larger. The industry throughput in the year 2000, with no recycle, is projected to be about 13,500 MTU as compared with about 920 MT in the year 1975. The impact on the environment is expected to increase proportionately. Thus, the impact on the general environment would be expected to increase by a factor of about 15. The estimated impacts on the environment of the UO_2 fuel fabrication industry over the period 1975 through 2000 are shown in Appendix A. Except for the minor increases in both airborne and waterborne radioactive emissions due to the ^{236}U content of the recycled uranium, the environmental impact is estimated to remain essentially the same, whether or not uranium is recycled, because the quantity of UO_2 fuel fabricated would be the same in both cases. If both uranium and plutonium are recycled, however, the environmental impact of the UO_2 fuel fabrication industry is estimated to be decreased by about 13.8 because the total demand for low enriched UO_2 would be decreased by this percentage.

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*Based on an estimated U.S. population of 250 million persons.

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CHAPTER IV

Section F

APPENDIX A

ESTIMATED INTEGRATED ENVIRONMENTAL IMPACT OF SUPPORTING URANIUM FUEL CYCLE, 1975 THROUGH 2000

APPENDIX A – ESTIMATED INTEGRATED ENVIRONMENTAL IMPACT OF SUPPORTING URANIUM FUEL CYCLE
1975 THROUGH 2000

	MINING			MILLING			UF_6 CONVERSION			ENRICHMENT CASE A*			ENRICHMENT CASE B**			UO_2 FUEL FABRICATION			TOTAL***		
	No Recycle	U Recycle	U & Pu Recycle	No Recycle	U Recycle	U & Pu Recycle	No Recycle	U Recycle	U & Pu Recycle	No Recycle	U Recycle	U & Pu Recycle	No Recycle	U Recycle	U & Pu Recycle	No Recycle	U Recycle	U & Pu Recycle	No Recycle	U Recycle	U & Pu Recycle
Number of Facilities in Year 2000	5840.	5064.	4125.	109.	95.	77	7.	6.	5	6.	6.	5.	6.	6.	5	9.	9.	7	-		
NATURAL RESOURCE USE																					
(Disturbed Acres)	3.34×10^{5}	2.89×10^{5}	2.37×10^{5}	3.71×10^{4}	3.23×10^{4}	2.62×10^{4}	980.	840.	700.	2.6×10^{3}	2.6×10^{3}	2.3×10^{3}	2.6×10^{3}	2.6×10^{3}	2.2×10^{3}	225.	225.	175	3.7×10^{5}	3.2×10^{5}	2.7×10^{5}
LAND (Acre Years of Occupancy)	2.34×10^{7}	2.09×10^{7}	1.81×10^{7}	1.22×10^{6}	1.90×10^{5}	0.95×10^{6}	1.51×10^{5}	1.40×10^{5}	1.25×10^{5}	4.6×10^{4}	4.7×10^{4}	4.3×10^{4}	4.6×10^{4}	4.6×10^{4}	4.3×10^{4}	1.3×10^{5}	1.3×10^{5}	1.1×10^{5}	2.5×10^{7}	2.2×10^{7}	1.9×10^{7}
(Permanently Committed Acres)	11000.	9500.	7800.	3.05×10^{4}	2.66×10^{4}	2.16×10^{4}	70.	60.	50									-	4.2×10^{4}	3.6×10^{4}	2.9×10^{4}
WATER (Millions of Gallons)																					
Discharged to Air				1.15×10^{6}	1.02×10^{6}	0.89×10^{6}	13.1×10^{3}	11.7×10^{3}	9.6×10^{3}	4.2×10^{5}	4.3×10^{5}	3.9×10^{5}	3.7×10^{5}	3.7×10^{5}	3.5×10^{5}		-		1.6×10^{6}	1.5×10^{6}	1.3×10^{6}
Discharged to Water Bodies							10.9×10^{4}	9.7×10^{4}	8.0×10^{4}	6.6×10^{7}	6.6×10^{7}	6.0×10^{7}	5.8×10^{7}	5.8×10^{7}	5.5×10^{7}	3.1×10^{4}	3.1×10^{4}	2.7×10^{4}	6.6×10^{7}	6.6×10^{7}	6.0×10^{7}
Discharged to Ground	2.99×10^{6}	2.67×10^{6}	2.32×10^{6}													-			3.0×10^{6}	2.7×10^{6}	2.3×10^{6}
ENERGY USE																					
Electricity (GWY)	2.87	2.57	2.23	6.70	6.00	5.21	1.28	1.15	0.96	153.0	153.7	139.6	133.2	133.6	126.4	1.15	1.15	1.00	165.0	164.6	149.0
Gas (Therms)				12.0×10^{9}	10.8×10^{9}	9.3×10^{9}	11.9×10^{8}	10.7×10^{8}	8.7×10^{8}							2.1×10^{8}	2.1×10^{8}	1.8×10^{8}	1.3×10^{10}	1.2×10^{10}	1.0×10^{10}
Fuel Oil (Gallons)	2.01×10^{9}	1.79×10^{9}	1.56×10^{9}							1.3×10^{7}	1.3×10^{7}	1.1×10^{7}	1.3×10^{7}	1.3×10^{7}	1.1×10^{7}			-	2.0×10^{9}	1.8×10^{9}	1.6×10^{9}
Coal (Metric Tons)										2.9×10^{8}	2.9×10^{8}	2.7×10^{8}	2.5×10^{8}	2.6×10^{8}	2.4×10^{8}				2.9×10^{8}	2.9×10^{8}	2.7×10^{8}
EFFLUENTS																					
Thermal Dissipation (Btu × 10^{9})	3.01×10^{5}	2.70×10^{5}	2.34×10^{5}	1.37×10^{6}	1.23×10^{6}	1.06×10^{6}	1.6×10^{5}	1.4×10^{5}	1.1×10^{5}	1.2×10^{7}	1.2×10^{7}	1.1×10^{7}	1.1×10^{7}	1.1×10^{7}	1.0×10^{7}	5.3×10^{4}	5.3×10^{4}	4.6×10^{4}	1.4×10^{7}	1.4×10^{7}	1.2×10^{7}
Chemical Effluents																					
To Atmosphere (MT)																					
SO_x	8.3×10^{4}	7.4×10^{4}	6.4×10^{4}	550	490	430.	3.6×10^{4}	3.2×10^{4}	2.8×10^{4}	3.7×10^{6}	3.7×10^{6}	3.4×10^{6}	3.2×10^{6}	3.2×10^{6}	3.1×10^{6}			-	3.8×10^{6}	3.8×10^{6}	3.5×10^{6}
NO_x	6.8×10^{4}	6.1×10^{4}	5.3×10^{4}	1.09×10^{5}	9.8×10^{4}	8.5×10^{4}	1.7×10^{4}	1.5×10^{4}	1.2×10^{4}	3.0×10^{6}	3.0×10^{6}	2.7×10^{6}	2.6×10^{6}	2.6×10^{6}	2.4×10^{6}				3.2×10^{6}	3.2×10^{6}	2.9×10^{6}
CO										5.9×10^{4}	5.9×10^{4}	5.4×10^{4}	5.2×10^{4}	5.2×10^{4}	4.9×10^{4}				5.9×10^{4}	5.9×10^{4}	5.4×10^{4}
Hydrocarbons	4.9×10^{3}	4.4×10^{3}	3.8×10^{3}	2.2×10^{4}	1.9×10^{4}	1.7×10^{4}	2.4×10^{3}	2.1×10^{3}	1.7×10^{3}	3.0×10^{4}	3.0×10^{4}	2.7×10^{4}	2.7×10^{4}	2.7×10^{4}	2.5×10^{4}			-	5.9×10^{4}	5.6×10^{4}	5.0×10^{4}
NH_3							4.9×10^{2}	4.4×10^{2}	3.9×10^{2}										490.	440.	390.
Fluorides							2.8×10^{2}	2.5×10^{2}	2.0×10^{2}	160.	160.	140.	140.	140	130.	31.	31.	27.	471.	441.	367.
Particulates	9.9×10^{3}	8.8×10^{3}	7.7×10^{3}	5.5×10^{3}	4.9×10^{3}	4.3×10^{3}	5.4×10^{2}	4.8×10^{2}	3.9×10^{2}	1.8×10^{5}	1.8×10^{5}	1.6×10^{5}	1.5×10^{5}	1.6×10^{5}	1.5×10^{5}	3.5×10^{4}	3.5×10^{4}	3.1×10^{4}	2.3×10^{5}	2.3×10^{5}	2.0×10^{5}
To Water Bodies (MT)																					
SO_4							5.7×10^{4}	5.1×10^{4}	4.5×10^{4}	3.3×10^{3}	3.3×10^{3}	2.8×10^{3}	3.3×10^{3}	3.3×10^{3}	2.8×10^{3}				6.0×10^{4}	5.4×10^{4}	4.8×10^{4}
NO_3							9.1×10^{2}	8.2×10^{2}	6.5×10^{2}	2.8×10^{3}	2.8×10^{3}	2.0×10^{3}	3.7×10^{3}	3.8×10^{3}	2.6×10^{3}	6.1×10^{3}	6.1×10^{3}	5.3×10^{3}	9.8×10^{3}	9.7×10^{3}	8.0×10^{3}
Cl							1.5×10^{4}	1.4×10^{4}	1.1×10^{4}	1.5×10^{3}	1.6×10^{3}	1.3×10^{3}	1.5×10^{3}	1.6×10^{3}	1.3×10^{3}				1.7×10^{4}	1.6×10^{4}	1.2×10^{4}
Fluorides							9.5×10^{2}	8.5×10^{2}	7.2×10^{2}	32.	33	26.	37.	29.	29.	173.	173.	150.	1.2×10^{3}	1.1×10^{3}	9.0×10^{2}
Na^{+}							5.2×10^{4}	4.7×10^{4}	4.1×10^{4}										5.2×10^{4}	4.7×10^{4}	4.1×10^{4}
Ca^{++}							3.6×10^{3}	3.3×10^{3}	2.6×10^{3}							0.9	0.9	0.8	3.6×10^{3}	3.3×10^{3}	2.6×10^{3}
NH_3							1.8×10^{4}	1.6×10^{4}	1.4×10^{4}							70	70.	60.	1.8×10^{4}	1.6×10^{4}	1.4×10^{4}
Fe							2.4×10^{2}	2.1×10^{2}	1.9×10^{2}										240.	210.	190.
Solid Wastes Collected † (Cubic Meters)							2.0×10^{5}	1.8×10^{5}	1.6×10^{5}	2.3×10^{7}	2.3×10^{7}	2.1×10^{7}	2.0×10^{7}	2.0×10^{7}	1.9×10^{6}	1.5×10^{5}	1.5×10^{5}	1.3×10^{5}	2.3×10^{7}	2.3×10^{7}	2.1×10^{7}

RADIOACTIVE EFFLUENTS																					
To Atmosphere (Curies)																					
Rn-222	2.4×10^{7}	2.1×10^{7}	1.8×10^{7}	4.4×10^{6}	4.0×10^{6}	3.5×10^{6}	0.036	0.032	0.028	–	–	–	–	–	–	–	–	–	2.8×10^{7}	2.5×10^{7}	2.2×10^{7}
Ra-226	–	–	–	13.	12.	10.	0.010	0.009	0.008	–	–	–	–	–	–	–	–	–	13.	12.	10.
Th-230	–	–	–	24.	21.	19.	0.095	0.085	0.076	–	–	–	–	–	–	–	–	–	24.	21.	19.
Th-234	–	–	–	19.	17.	15.	2.23	1.99	1.78	–	–	–	–	–	–	0.12	0.12	0.11	21.4	19.1	16.9
U-232	–	–	–	–	–	–	–	–	–	–	0.002	0.002	–	0.002	0.002	–	–	–	–	0.002	0.002
U-233	–	–	–	–	–	–	–	–	–	–	0.00001	0.00001	–	0.00001	0.00001	–	–	–	–	0.00001	0.00001
U-234	–	–	–	260.	230.	200.	3.52	3.15	2.70	3.4	3.	2.	3.4	3.	2.	1.	1.	0.87	268.	237.	206.
U-235	–	–	–	12.	11.	9.	0.15	0.14	0.12	0.1	0.1	0.1	0.1	0.1	0.09	0.028	0.028	0.024	12.3	11.3	9.2
U-236	–	–	–	–	–	–	–	–	–	–	0.08	0.07	–	0.08	0.07	0.00	0.042	0.036	–	0.122	0.106
U-238	–	–	–	260.	230.	200	3.52	3.15	2.70	0.4	0.5	0.4	0.4	0.4	0.4	0.12	0.12	0.11	264.	234.	203.
Tc-99	–	–	–	–	–	–	–	–	–	–	38.	32.	–	38.	32.	–	–	–	–	38.	32.
Other Radionuclides****	–	–	–	–	–	–	–	–	–	–	0.62	0.51	–	0.62	0.51	–	–	–	–	0.62	0.51
To Water Course (Curies)																					
Ra-226	–	–	–	–	–	–	1.5	1.3	1.2	–	–	–	–	–	–	–	–	–	1.5	1.3	1.2
Th-230	–	–	–	–	–	–	40.	36.	28.	–	–	–	–	–	–	–	–	–	40.	36.	28.
Th-234	–	–	–	–	–	–	136.	121.	105.	–	–	–	–	–	–	29.4	29.4	25.5	165.	150.	131.
U-232	–	–	–	–	–	–	–	–	–	–	0.0003	0.0003	–	0.0003	0.0003	–	–	–	–	0.0003	0.0003
U-233	–	–	–	–	–	–		–	–	–	2×10^{-6}	1×10^{-6}	–	2×10^{-6}	2×10^{-6}	–	–	–	–	2×10^{-6}	1×10^{-6}
U-234	–	–	–	–	–	–	134.	120.	100.	0.1	0.1	0.09	0.1	0.1	0.09	236.	233.	205.	370.	356.	305.
U-235	–	–	–	–	–	–	5.8	5.2	4.4	0.004	0.004	0.004	0.005	0.005	0.004	6.4	6.4	5.6	12.2	11.6	10.
U-236	–	–	–	–	–	–	–	–	–	–	0.04	0.008	–	0.01	0.009	0.0	9.8	8.5	–	9.8	8.5
U-238	–	–	–	–	–	–	134.	120.	100.	0.09	0.09	0.07	0.1	0.1	0.08	29.4	29.4	25.5	163.	149.	126.
Tc-99	–	–	–	–	–	–	–		–	–	491.	419.	–	491.	419.	–	–	–	–	491.	419.
Other Radionuclides****	–	–	–	–	–	–	–	–	–	–	8.21	7.18	–	8.22	7.19	–	–	–	–	8.21	7.18
Solid Wastes (Cubic Meters)	–	–	–	8.0×10^{8}	7.2×10^{8}	6.2×10^{8}	8.8×10^{4}	7.9×10^{4}	6.6×10^{4}	3.8×10^{4}	4.1×10^{4}	8.6×10^{3}	9.8×10^{4}	9.8×10^{4}	4.8×10^{4}	–	–	–	8.0×10^{8}	7.2×10^{8}	6.2×10^{8}
RADIOLOGICAL DOSE COMMITMENT (Person-Rem)																					
Occupational																					
Whole Body	1.2×10^{6}	1.0×10^{6}	0.9×10^{6}	5.6×10^{5}	5.0×10^{5}	4.4×10^{5}	4.4×10^{3}	4.1×10^{3}	3.5×10^{3}	2.9×10^{3}	2.9×10^{3}	2.3×10^{3}	3.2×10^{3}	3.2×10^{3}	2.5×10^{3}	5.1×10^{4}	5.1×10^{4}	4.4×10^{4}	1.8×10^{6}	1.5×10^{6}	1.4×10^{6}
Bone	1.7×10^{6}	1.6×10^{6}	1.4×10^{6}	2.3×10^{6}	2.0×10^{6}	1.8×10^{6}	5.0×10^{4}	4.6×10^{4}	4.1×10^{4}	2.4×10^{4}	2.5×10^{4}	2.0×10^{4}	2.7×10^{4}	2.7×10^{4}	2.2×10^{4}	6.6×10^{4}	6.6×10^{4}	5.7×10^{4}	3.8×10^{6}	3.7×10^{6}	3.3×10^{6}
Thyroid	1.2×10^{6}	1.0×10^{6}	0.9×10^{6}	2.1×10^{5}	1.9×10^{5}	1.6×10^{5}	4.4×10^{3}	4.1×10^{3}	3.5×10^{3}	2.9×10^{3}	2.9×10^{3}	2.3×10^{3}	3.2×10^{3}	3.2×10^{3}	2.5×10^{3}	5.0×10^{4}	5.0×10^{4}	4.3×10^{4}	1.5×10^{6}	1.3×10^{6}	1.1×10^{6}
Liver	1.3×10^{6}	1.2×10^{6}	1.0×10^{6}	2.1×10^{5}	1.9×10^{5}	1.6×10^{5}	5.1×10^{3}	4.7×10^{3}	4.1×10^{3}	2.9×10^{3}	2.9×10^{3}	2.3×10^{3}	3.2×10^{3}	3.2×10^{3}	2.5×10^{3}	5.0×10^{4}	5.0×10^{4}	4.3×10^{4}	1.6×10^{6}	1.5×10^{6}	1.2×10^{6}
Kidney	1.6×10^{6}	1.5×10^{6}	1.3×10^{6}	2.6×10^{5}	2.3×10^{5}	2.0×10^{5}	1.3×10^{4}	1.2×10^{4}	1.1×10^{4}	6.8×10^{3}	6.8×10^{3}	5.6×10^{3}	7.5×10^{3}	7.5×10^{3}	6.0×10^{3}	5.4×10^{4}	5.4×10^{4}	4.6×10^{4}	1.9×10^{6}	1.8×10^{6}	1.6×10^{6}
GI Tract	1.1×10^{6}	1.0×10^{6}	0.9×10^{6}	2.1×10^{5}	1.9×10^{5}	1.6×10^{5}	2.7×10^{3}	2.5×10^{3}	2.1×10^{3}	1.7×10^{3}	1.6×10^{3}	1.3×10^{3}	1.9×10^{3}	1.8×10^{3}	1.4×10^{3}	5.0×10^{4}	5.0×10^{4}	4.3×10^{4}	1.4×10^{6}	1.3×10^{6}	1.1×10^{6}
Lung	6.4×10^{6}	5.7×10^{6}	4.9×10^{6}	4.8×10^{6}	4.3×10^{6}	3.8×10^{6}	3.4×10^{4}	3.1×10^{4}	2.6×10^{4}	5.9×10^{4}	5.9×10^{4}	4.9×10^{4}	6.5×10^{4}	6.5×10^{4}	5.3×10^{4}	1.95×10^{6}	1.96×10^{6}	1.69×10^{6}	1.3×10^{7}	1.2×10^{7}	1.0×10^{7}
Skin	1.1×10^{6}	1.0×10^{6}	0.9×10^{6}	2.1×10^{5}	1.9×10^{5}	1.6×10^{5}	1.3×10^{4}	1.2×10^{4}	9.5×10^{3}	1.0×10^{4}	1.0×10^{4}	8.1×10^{3}	1.1×10^{4}	1.1×10^{4}	9.0×10^{3}	5.0×10^{4}	5.0×10^{4}	4.3×10^{4}	1.4×10^{6}	1.3×10^{6}	1.1×10^{6}
Total Population																					
Whole Body	3.0×10^{6}	2.7×10^{6}	2.3×10^{6}	5.8×10^{5}	5.2×10^{5}	4.5×10^{5}	4.3×10^{4}	3.9×10^{4}	3.4×10^{4}	77.	130.	110.	77.	120.	110.	2.4×10^{3}	2.6×10^{3}	2.2×10^{3}	3.6×10^{6}	3.3×10^{6}	2.8×10^{6}
Bone	9.7×10^{6}	8.7×10^{6}	7.5×10^{6}	1.9×10^{6}	1.7×10^{6}	1.5×10^{6}	1.0×10^{5}	9.2×10^{4}	7.9×10^{4}	930.	950.	790.	930.	950.	800.	4.1×10^{4}	4.2×10^{4}	3.6×10^{4}	1.2×10^{7}	1.1×10^{7}	9.1×10^{6}
Thyroid	7.3×10^{3}	6.5×10^{3}	5.7×10^{3}	1.5×10^{3}	1.4×10^{3}	1.2×10^{3}	50.	40.	40.	18.	17.	15.	18.	17.	15.	4.6	4.7	4.0	8.9×10^{3}	8.0×10^{3}	7.0×10^{3}
Liver	2.4×10^{6}	2.1×10^{6}	1.9×10^{6}	4.5×10^{5}	4.0×10^{5}	3.5×10^{5}	680.	600.	490.	21.	220.	190.	21.	220.	190.	5.2	5.2	4.5	2.9×10^{6}	2.5×10^{6}	2.3×10^{6}
Kidney	11.3×10^{6}	10.1×10^{6}	8.8×10^{6}	2.1×10^{6}	1.9×10^{6}	1.7×10^{6}	1.2×10^{4}	1.1×10^{4}	8.9×10^{3}	520.	2900.	2550	520.	3000.	2550.	6.5×10^{3}	6.8×10^{3}	5.9×10^{3}	1.3×10^{7}	1.2×10^{7}	1.1×10^{7}
GI Tract	1.2×10^{5}	1.0×10^{5}	9.0×10^{4}	2.2×10^{4}	2.0×10^{4}	1.7×10^{4}	5.6×10^{3}	5.0×10^{3}	4.2×10^{3}	66.	7070.	6270.	67.	7350.	6270.	2.7×10^{3}	2.8×10^{3}	2.4×10^{3}	1.5×10^{5}	1.4×10^{5}	1.2×10^{5}
Lung	9.1×10^{5}	3.1×10^{5}	7.1×10^{5}	1.8×10^{5}	1.6×10^{5}	1.4×10^{5}	980.	880.	760.	400.	440.	380.	490.	440.	380.	140.	150.	130.	1.1×10^{6}	9.7×10^{5}	8.5×10^{5}
Skin	7.3×10^{3}	6.5×10^{3}	5.7×10^{3}	1.5×10^{3}	1.4×10^{3}	1.2×10^{3}	40.	30.	30.	18.	110.	100.	18.	120.	100.	4.6	4.7	4.0	8.9×10^{3}	8.0×10^{3}	7.0×10^{3}

* Case A – Assuming first new enrichment facility is gaseous diffusion plant with subsequent plants using gas centrifuge process.

** Case B – Assuming all new enrichment plants use gas centrifuge process.

*** Based on enrichment Case A.

**** See Tables IV F-15A and IV F-15B for breakdown of radionuclides.

Coal consumption reported is that used in fossil fuel fired power plants supplying 60% of electrical energy consumed. Airborne chemical effluents reported include coal combustion products.

† Includes sludges generated in coal fired power plants supplying electricity for enrichment industry.

†† Minor differences from values reported in CHAPTER VIII APPENDIX A result from differences in rounding off numbers from hand calculations and in procedures employed in integrating impacts over the 26 year period.

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SUMMARY

General

This section presents an analysis of the predicted environmental impact caused by shipments of radioactive material throughout the fuel cycle necessitated by a mature industry involving the use of recycle plutonium, as oxide mixed with uranium oxide, in light water reactors (LWR's). An assessment of the potential environmental impact of transporting fuel materials and solid radioactive wastes for LWR's fueled by enriched uranium has been reported previously.[1,2] Since the impact of radioactive materials transportation in the overall uranium cycle was found to be quite low (3.4×10^{-3} mrem/person/reactor/year),* analysis of the effects of plutonium recycle was performed to determine the differences that would result from those determined in the uranium cycle. The assessment for the uranium cycle was based on the existing transportation industry, operating under the present regulatory requirements of the U.S. Nuclear Regulatory Commission (NRC) and the Department of Transportation (DOT). The environmental impact of recycling plutonium is compared with a) no recycle of either uranium or plutonium and b) uranium recycle only with no plutonium recycle.

Material quantities in this analysis correspond to the predicted annual quantities for the entire nuclear power industry for the 26-year period 1975-2000 (see CHAPTER XI). Estimates of environmental effects are based on representative conditions for such important parameters as radiation level, shipping distance, package content, population density, and accident frequency. The values chosen are judged to be conservative but representative values from the possible value range of parameters. Credit has not been taken for the reduction in transportation steps prior to uranium fuel fabrication due to the recovery of uranium and plutonium from spent LWR fuel assemblies in the mining, milling, UF_6 production, and enrichment steps.

Figure IV G-1 gives a flow diagram for the projected mixed oxide fuel cycle as envisioned for the 26-year period. The flow diagram assumes that mixed oxide (MOX) fuel fabrication plants will be at separate sites from either UO_2 fuel fabrication plants or reprocessing plants. In the event that any collocation of such facilities should take place, the flow diagram would be somewhat conservative for estimating the environmental effects of transportation.

In addition to examining a mature industry in detail, the integrated environmental and radiological factors imposed over the 26-year period 1975-2000 have been

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*This value was obtained by summing contributions from References 1 and 2.

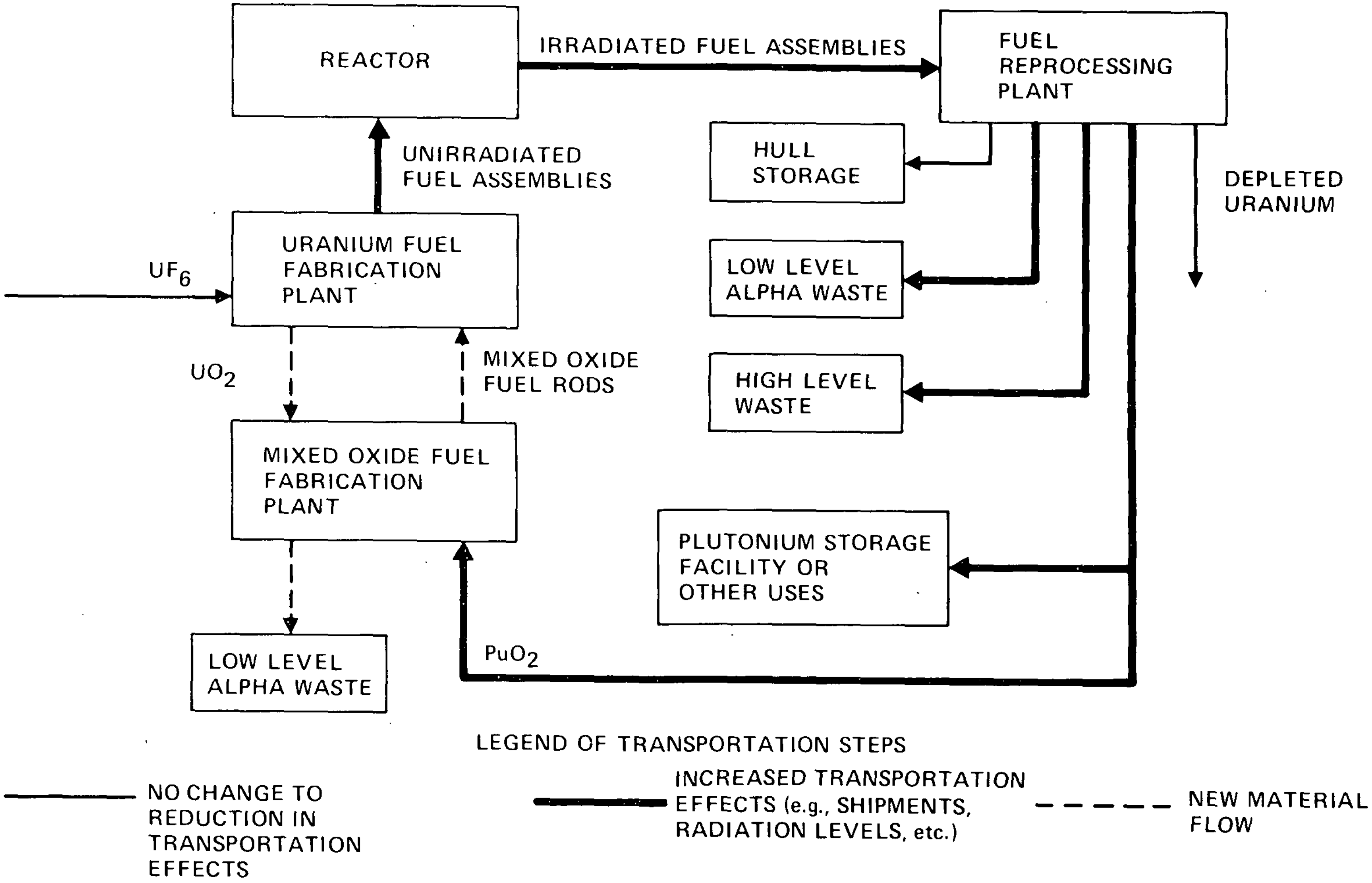

Figure IV G-1 Flow Diagram for Mixed Oxide Fuel Cycle determined. Factors for the 26-year period were calculated by a direct comparison of factors for the year 2000 period with the integrated period based on the throughput. The factors for the year 2000 have been developed in detail in the text.

The Safeguards Supplement to GESMO discusses the safeguards requirements of transporting fissionable material.

Environmental Effects

Tables IV G-1 and IV G-2 present a summary of estimated materials and quantities to be transported for the 26-year period as affected by full scale plutonium recycle compared with no recycle of either uranium or plutonium. See CHAPTER XI.

Implementation of plutonium recycle would result in an approximate 6% overall decrease in vehicle-miles (15 million miles) involved in shipment of fuel materials and wastes (total of Tables IV G-1 and IV G-2) over the no-uranium or plutonium recycle case. This decrease is due primarily to fewer waste shipments to a Federal repository.

There will be no appreciable effect on the environment from heat, weight or traffic density in the shipping of fuel and waste.

The following truck and train mileages obtained from Tables IV G-1 and IV G-2 were used for determining the combustion effluent releases for the integrated 26-year period (1975-2000). The releases are reported in detail in CHAPTER VIII.

	Mileage		
	No U or Pu Recycle	With U Recycle	With U and Pu Recycle
Truck	123,000,000	123,000,000	142,000,000
Train	116,000,000	75,800,000	82,100,000

(Diesel oil consumption was assumed to be 5 miles per gallon for both truck and train shipments.)

Future barge shipments have not been considered in this section because data reported in WASH-1238 show that any such future shipments will reduce the environmental impact.

Radiological Impacts

Tables IV G-3 and IV G-4 present a summary of the estimated 26-year period radiation doses to the transport workers and the general public, if full scale plutonium recycle is instituted. These tables assume the following numbers of different shipping routes in estimating the number of people (general public) exposed during the various types of shipments:

Table IV G-1 *Shipments of Fuel Materials (1975-2000) Under Various Modes of Fuel Management*

TYPE OF SHIPMENT	PROBABLE MODE OF TRANSPORT	QUANTITY SHIPPED (MT)**			NO. OF SHIPMENTS			ESTIMATED AVERAGE SHIPPING DISTANCE (MILES)	TOTAL SHIPPING DISTANCE (MILES)		
		NO U OR Pu RECYCLE	WITH U RECYCLE	WITH U + Pu RECYCLE	NO U OR Pu RECYCLE	WITH U RECYCLE	WITH U + Pu RECYCLE		NO U OR Pu RECYCLE	WITH U RECYCLE	WITH U + Pu RECYCLE
UO_2 TO MOX FUEL FABN. PLANT	TRUCK	—	—	24,100	—	—	1,800	200	—	—	360,000
MOX FUEL RODS TO UO_2 FUEL FABN. PLANT	TRUCK	—	—	25,300	—	—	4,220	200	—	—	1,690,000*
UNIRRADIATED FUEL ASSEMBLIES TO REACTOR	TRUCK	189,000	189,000	189,000	30,800	30,800	30,800	1,000	61,600,000*	61,600,000*	61,600,000*
IRRADIATED FUEL ASSEMBLIES TO FUEL REPROCESSING PLANT OR STORAGE	TRUCK	50,200	50,200	50,200	61,400	61,400	75,500	500	61,400,000*	61,400,000*	75,500,000*
	RAIL	75,200	75,200	75,200	15,300	15,300	16,400	1,000	30,600,000*	30,600,000*	32,700,000*
PuO_2 TO STORAGE OR OTHER USES	TRUCK	—	—	18	—	—	41	300	—	—	24,500*
PuO_2 TO MOX FUEL FABN. PLANT	TRUCK	—	—	1,160	—	—	2,650	300	—	—	1,590,000*
TOTAL					108,000	108,000	131,000		154,000,000	154,000,000	174,000,000

* Includes return trip for empty containers.

** Heavy Metal (uranium plus plutonium)

Radioactive shipments in the fuel cycle which are not listed are not significantly affected by recycle.

Table IV G-2 Shipments of Waste Materialsl (1975-2000) Under Various Modes of Fuel Management

TYPE OF SHIPMENT	PROBABLE MODE OF TRANSPORT	QUANTITY SHIPPED (CU FT)			NO. OF SHIPMENTS			ESTIMATED AVERAGE SHIPPING DISTANCE (MILES)	TOTAL SHIPPING DISTANCE (MILES)		
		NO U OR Pu RECYCLE	WITH U RECYCLE	WITH U + Pu RECYCLE	NO U OR Pu RECYCLE	WITH U RECYCLE	WITH U + Pu RECYCLE		NO U OR Pu RECYCLE	WITH U RECYCLE	WITH U + Pu RECYCLE
HIGH LEVEL WASTE FROM REPROCESSING PLANT TO FEDERAL REPOSITORY	RAIL	1,940,000**	261,000***	254,000	28,600	3,740	4,390	1,500	85,800,000*	11,200,000*	13,200,000*
LOW LEVEL ALPHA WASTE FROM REPROCESSING PLANT TO FEDERAL REPOSITORY	RAIL	—	5,090,000	5,090,000	—	11,300	11,300	1,500	—	34,000,000*	34,000,000*
LOW LEVEL ALPHA WASTE FROM MOX FUEL FABN. PLANT TO FEDERAL REPOSITORY	TRUCK	—	—	131,000	—	—	440	1,500	—	—	1,320,000*
	RAIL	—	—	523,000	—	—	740	1,500	—	—	2,220,000*
TOTAL					28,600	15,100	16,900		85,800,000	45,200,000	50,700,000

* Includes return trip for empty containers.

** Shipment of irradiated fuel assemblies to Federal repository from storage facility.

*** Includes shipments of Pu to Federal repository.

Radioactive shipments in the fuel cycle which are not listed are not significantly affected by recycle.

Table IV G-3 Dose Estimates for Shipments of Fuel Material (1975-2000) Under Various Modes of Fuel Management

TYPE OF SHIPMENT	MODE OF TRANSPORT	TRANSPORT WORKERS						GENERAL PUBLIC (ONLOOKERS AND PEOPLE ALONG THE ROUTE)					
		NO U OR Pu RECYCLE		WITH U RECYCLE		WITH U + Pu RECYCLE		NO U OR Pu RECYCLE		WITH U RECYCLE		WITH U + Pu RECYCLE	
		PERSON-REM	NO. OF PEOPLE	PERSON-REM	NO. OF PEOPLE	PERSON-REM	NO. OF PEOPLE	PERSON-REM	NO. OF PEOPLE	PERSON-REM	NO. OF PEOPLE	PERSON-REM	NO. OF PEOPLE
UO_2 TO MOX FUEL FABN. PLANT	TRUCK	—	—	—	—	1	240	—	—	—	—	0.1	784,000
MOX FUELS RODS TO UO_2 FUEL FABN. PLANT	TRUCK	—	—	—	—	135	565	—	—	—	—	22	788,000
UNIRRADIATED FUEL ASSEMBLIES TO REACTOR	TRUCK	28	64,700	28	64,700	510	64,700	61	9,670,000	61	9,670,000	84	9,670,000
IRRADIATED FUEL ASSEMBLIES TO FUEL REPROCESSING PLANT OR STORAGE	TRUCK	4,580	86,000	4,580	86,000	5,620	106,000	800	4,990,000	800	4,990,000	980	5,060,000
	RAIL	65	26,000	65	26,000	70	27,800	340	9,510,000	340	9,510,000	370	9,520,000
PuO_2 TO STORAGE OR OTHER USES	TRUCK	—	—	—	—	110	2,690	—	—	—	—	21	1,880,000
TOTAL		4,670	177,000	4,670	177,000	6,450	202,000	1,200	24,200,000	1,200	24,200,000	1,480	27,700,000

Radioactive shipments in the fuel cycle which are not listed are not significantly affected by recycle.

Table IV G-4 Dose Estimate for Shipments of Waste Material (1975-2000) Under Various Modes of Fuel Management

TYPE OF SHIPMENT	MODE OF TRANSPORT	TRANSPORT WORKERS						GENERAL PUBLIC (ONLOOKERS AND PEOPLE ALONG THE ROUTE)					
		NO U OR Pu RECYCLE		WITH U RECYCLE		WITH U + Pu RECYCLE		NO U OR Pu RECYCLE		WITH U RECYCLE		WITH U + Pu RECYCLE	
		PERSON-REM	NO. OF PEOPLE	PERSON-REM	NO. OF PEOPLE	PERSON-REM	NO. OF PEOPLE	PERSON-REM	NO. OF PEOPLE	PERSON-REM	NO. OF PEOPLE	PERSON-REM	NO. OF PEOPLE
HIGH LEVEL WASTE FROM REPROCESSING PLANT TO FEDERAL REPOSITORY	RAIL	160	62,900	21	8,230	24	9,670	840	2,480,000	110	2,110,000	130	2,120,000
LOW LEVEL ALPHA WASTE FROM REPROCESSING PLANT TO FEDERAL REPOSITORY	RAIL	—	—	62	24,900	62	24,900	—	—	330	2,220,000	330	2,220,000
LOW LEVEL ALPHA WASTE FROM MOX FUEL FAB'N. PLANT TO FEDERAL REPOSITORY	TRUCK	—	—	—	—	105	1,150	—	—	—	—	13	2,060,000
	RAIL	—	—	—	—	4	1,630	—	—	—	—	22	2,070,000
TOTAL		160	62,900	83	33,200	195	37,400	840	2,480,000	440	4,330,000	495	8,470,000

Radioactive shipments in the fuel cycle which are not listed are not significantly affected by recycle.

To and from MOX fuel fabrication plant	5 routes east of Mississippi River
Fuel assemblies to and from reactors	12 routes east of Mississippi River
PuO_2 to storage and other uses	8 routes east of Mississippi River
High level and low level wastes	3 routes east of Mississippi River
	2 routes west of Mississippi River

The number of routes was chosen on the basis of estimated number of plants of each type and the limited number of alternate routes in the railroad and highway network. Routes in the East were emphasized since the population density is higher in that area and the person-rem exposure is thereby conservatively projected. The site locations were assumed to be of the same general distribution as today's sites. If nuclear parks are established, there would be a reduction in transportation mileage and a corresponding reduction in radiological effects due to transportation.

Shipments associated with uranium and plutonium recycle for the 26-year period would be accompanied by about a 3 percent decrease in total exposure to a sector of the general public (onlookers and people along the route) and about a 38 percent increase in total exposure to transportation workers as compared to no-uranium or plutonium recycle case. Shipments associated with the uranium-recycle-only case would be accompanied by about a 19 percent decrease in total exposure to the general public (onlookers and people along the route) and about a 2 percent decrease in total exposure to transportation workers over the no-uranium or plutonium recycle case. These dose estimates are strongly influenced by the assumption that in about the year 2000 the DOT dose limit of 2 mrem/hr will be reached in the cab of each truck transporting irradiated fuel or plutonium-bearing materials. The assumption of 2 mrem/hr in the cab of the truck is considered conservative because it would be anticipated that shielding would be installed in the packages or truck cab to lower the dose rate. Use of dose rates in truck transportation typical of current UO_2 fresh fuel shipments would diminish the estimated doses by a factor of three to five. However, the relative changes in exposure would remain about the same even if different assumed dose rates were used in the analysis.

For shipments of unirradiated MOX fuel rods and assemblies, the assumed DOT maximum dose limits are conservative by a factor of two, based on actual experience.

The radiological impact for the integrated 26-year period (1975-2000) is given below. The values have been taken from Tables IV G-3 and IV G-4. The resultant doses are less than the normal variation in background dose and represent no significant radiological impact.

AVERAGE DOSE ESTIMATES PER PERSON FOR 26-YEAR PERIOD

	No U or Pu Recycle	With U Recycle	With U and Pu Recycle
Dose Estimates, mrem/person			
Transport Workers	20	23	28
General Public*	0.08	0.06	0.05

(Average natural background radiation exposure is 100 mrem per person per year or 2600 mrem per person for the 26-year period.)

Should the plutonium oxide be blended with the uranium oxide at the reprocessing plants, additional exposure to the transportation workers and members of the general public would result. This would be due to an increased number of shipments with the maximum assumed DOT radiation levels. Quantitative estimates of the number of shipments and associated radiation exposure are discussed in CHAPTER IV Section L, paragraph 3.2.

Accidents

The probability of an accident occurring in transportation is small--about one accident per million vehicle miles--and decreases with increased severity of the accident to about one extra severe accident (point at which the package containment may be breached) per 50 trillion vehicle miles, and one extremely severe accident per 10 million-million vehicle miles.[2] Based on these data and the increment of vehicle miles for the 26-year period attendant on widescale use of recycle plutonium, the frequency of an extra severe transportation accident ascribable to recycle of uranium and plutonium is 4.1×10^{-5}/yr. This frequency corresponds to a recurrence period between extra severe transportation accidents of about 20,000 years (see Section G-5.0).

Because of package design and quality assurance, the probability of a breach in the containment of a package involved in an accident is small and is related to the accident severity. Due to regulatory limits on contents of packages and the nature and form of the unirradiated and irradiated nuclear fuel and solid radioactive waste, the amount of radioactivity that would be released if such a breach were to occur is unlikely to be large. Although the consequences could be serious with regard to property damage, no deaths would be expected as a direct result of the release of radioactivity. It is concluded that the radiation risk to the environment in transportation accidents is small.

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*Assumed general public change homes every 10 years (new people living along transportation routes every 10 years).

Alternatives

Alternatives have been considered to present transport methods, i.e., routing, escorts, decay times, lower package radiation levels, and more stringent accident damage tests. These alternatives have been found not to increase safety significantly.

1.0 INTRODUCTION

1.1 Purpose

This section presents an analysis of the predicted environmental impact from changes in the quantities and types of radioactive material shipments in the fuel cycle caused by the recycle of plutonium in light water-cooled reactors. This is part of an overall evaluation of the total environmental impact of potential widescale use of mixed oxide fuel that NRC is performing prior to reaching a decision as to licensing such use.

1.2 Basic Approach to Assessment

An assessment of the potential environmental impact of transporting fresh and irradiated fuel and solid radioactive wastes for nuclear power plants (LWR's) fueled by enriched uranium has already been made and documented in WASH-1238.[2] In addition, the effects of transportation of radioactive materials for the uranium fuel cycle were analyzed in WASH-1248.[1] In these studies, the environmental impact of transportation of radioactive materials for the overall uranium fuel cycle was found to be small (3.4×10^{-3} mrem/person/reactor/year).

The proposed widescale use of fuel made from mixed plutonium and uranium oxides to replace enriched uranium oxide represents an incremental change of an existing transportation industry operating under a set of regulations. This assessment is directed toward analyzing the differences in the environmental impact of radioactive materials transportation caused by the change in fuel type.

The detailed analysis is based on a total industry annual LWR fuel charge (as heavy metal) of 13,500 metric tons (MT) and a fuel discharge of 10,250 MT as projected for the nuclear power industry in about the year 2000 under the low growth scenario.[3] The large disparity between the mass of fresh fuel charged and discharged occurs because of the large number of new reactors being started annually in the projected rapidly growing industry. For conservatism, the quantities of high level waste considered to be shipped about the year 2000 are based on the fuel to be reprocessed that year, representing a commitment to make such shipments within the succeeding ten years. Actual shipments of high level waste will correspond to the lesser quantities of fuel reprocessed about 5-10 years earlier, when significantly smaller amounts of fuel are reprocessed.

The mixture of reactor types and other important parameters in the analysis have been extrapolated from available data. One-third of all reactors using mixed oxide (MOX) fuel are assumed to be BWR's and two-thirds PWR's. See CHAPTER IV, Section C, paragraph 4.1.4. The packaging designs for shipments are assumed to be similar to existing designs, with comparable heat load limits. Distances for the added or changed transportation steps are from 200 to 1,500 miles based on current and predicted facility locations.[5] As with the other parameter values, average mileages assumed have been taken in the upper range to ensure the conservatism of calculated impact effects. Also, any future barge shipments have not been considered in this section because data reported in WASH-1238 show that any such future shipments will reduce the environmental impact.

1.3 Overview of Added and Modified Transportation Steps

Large scale use of uranium and plutonium recycle to replace enriched uranium for LWR fuel affects the transportation of radioactive materials by (1) requiring added transportation steps and (2) modifying effects of a transportation step already a part of the uranium fuel cycle. The seven distinctly new shipment steps in the LWR fuel cycle with uranium and plutonium recycle are:

- Shipments of uranium oxide from the uranium fuel fabrication plant or the UF_6 production plant to the mixed oxide fuel fabrication plant
- Shipments of mixed oxide fuel rods from the mixed oxide fuel fabrication plant to the uranium oxide fuel fabrication plant
- Shipments of alpha wastes from the mixed oxide fuel fabrication plant to the Federal waste repository
- Shipment of irradiated fuel from the reactor to the fuel reprocessing plant
- Shipment of plutonium oxide from the fuel reprocessing plant to storage or to the mixed oxide fuel fabrication plant
- Shipment of high level waste (HLW) from the fuel reprocessing plant to the Federal repository
- Shipment of low level alpha waste from the fuel reprocessing plant to the Federal repository

In addition to requiring the seven added shipment steps listed above, the use of recycled plutonium affects one transportation step already a necessary part of the fuel cycle:

- Shipment of unirradiated fuel from the uranium fuel fabrication plant to the reactor

It should be noted that no credit has been taken for any savings in the transportation steps that occur in the uranium feed steps prior to the UO_2 fuel fabrication step. The recycle of uranium and plutonium would decrease all such operations and the associated transportation steps by about 23 percent. See CHAPTER III, Appendix A.

The modified impacts result primarily from increases in the levels of gamma and neutron radiation and the rate of heat generation in the affected steps as a result of changes in fuel composition. The flow rate of plutonium from the reprocessing plant and at other points in the fuel cycle is also increased by recycle. The introduction of plutonium recycle increases the concentrations of some isotopes (plutonium-238, plutonium-240, plutonium-241, plutonium-242, americium-241, americium-243, curium-242, and curium-244) that affect the levels of radiation and the rate of heat generation in the spent fuel.

The recycle of recovered uranium and plutonium from spent LWR fuel assemblies reduces the volumes of uranium-bearing materials in the transportation steps prior to uranium fuel fabrication, i.e., mining, milling, UF_6 production, and enrichment. These flow reductions cause reductions (see CHAPTER IV, Section F, Summary) in the environmental effects of the affected process and transportation steps, but credit is not taken for these transportation savings in this assessment as a further step to assure that a conservative analysis is provided.

2.0 REGULATORY STANDARDS AND REQUIREMENTS

Commercial packaging and transport of radioactive materials are regulated at the Federal level by the Nuclear Regulatory Commission (NRC), the Department of Transportation (DOT), and the U.S. Postal Service. Certain aspects--such as limitations on gross weight of trucks and transportation not subject to DOT, NRC, or the Postal Service regulations--are regulated by the States. Most States have adopted regulations which require the shipper to conform to the packaging, labeling, and marking requirements of the DOT to the same extent as if the transportation were subject to the rules and regulations of that agency.

Most shipments of radioactive material move in routine commerce and on conventional transportation equipment. Shipments are therefore subject to the same transportation environments, including accidents, as nonradioactive cargo. Although a shipper may impose some conditions on the carriage of his shipment, such as speed limitations and the provision of an escort, most of the conditions to which his shipment is subjected are not under his control. Protection of the public and transport workers from radiation during the shipment of radioactive materials is achieved by adherence to regulations promulgated by NRC through a combination of limitations on the contents according to the quantities and types of radioactivity, and standards and criteria for package design. Primary reliance for achieving safety in radioactive materials transport is placed on package design and integrity; achieving an acceptable level of safety in transport does not require dependence on special routing. However, special routings are used to bypass some bridges and tunnels to avoid possible interference with the flow of traffic, should an accident occur.

Primary reliance for safety in the transport of radioactive material is placed on the packaging. As indicated, packaging must meet regulatory standards established by the NRC, DOT, Postal Service, and the States, according to the type and form of material, for containment, shielding, nuclear criticality safety, and heat dissipation. Standards provide that the packaging shall prevent the loss or dispersal of radioactive contents, retain radiological shielding efficiency, assure nuclear criticality safety, and provide adequate heat dissipation under normal conditions of transport and under specified accident damage test conditions (the design-basis accident). The contents of the package also must be limited so that the standards for external radiation levels, external temperature, internal pressure, and containment are met.

Protection from external radiation is provided by limitations on the radiation levels on the outside of packages of radioactive materials, and by stowage and segregation provisions. The number of packages in a single vehicle or area is limited, to control the aggregate radiation level and to provide nuclear criticality safety. Minimum separation distances from people and undeveloped film are specified for loading and storing packages to keep the exposure of persons and film to a minimum.

Specific aspects of the regulations (10 CFR Part 71) are discussed in Appendix A to this chapter.

3.0 DESCRIPTION AND EVALUATION OF THE POTENTIALLY ADDED TRANSPORTATION STEPS DUE TO URANIUM AND PLUTONIUM RECYCLE FOR NORMAL CONDITIONS OF TRANSPORTATION

3.1 Introduction

As previously noted, the environmental impact of potential industrywide plutonium recycle in LWR's and attendant changes in radioactive materials transportation is being assessed in terms of incremental effects on the total industry as of about the year 2000 in detail and extrapolated for the 26-year period (1975-2000) based on total quantities being transported for the two different time frames. Such incremental effects for normal conditions of transport are divided into those resulting from new steps in transportation, the subject of this section, and the transportation step that is modified as a result of plutonium recycle, the subject of Section 4.0. The potential impacts of transportation accidents for both the added and the modified transportation steps are discussed in Section 5.0.

Recycle of plutonium in LWR's requires seven added types of shipments as a part of the fuel cycle shipments of (1) unirradiated uranium oxide (UO_2) to the mixed oxide (MOX) fuel fabrication plant, (2) MOX fuel rods from the MOX fuel fabrication plant to the UO_2 fuel fabrication plant, (3) low level alpha wastes from the MOX fuel fabrication plant to a Federal repository, (4) irradiated fuel from the reactor to the reprocessing plant, (5) plutonium oxide from the fuel reprocessing plant to the mixed oxide fuel fabrication plant, (6) high level wastes from the reprocessing plant to the Federal repository, and (7) low level alpha wastes from the reprocessing plant to the Federal repository.

The shipment of unirradiated UO_2 to the MOX fuel fabrication plant may originate at several different points (UO_2 fuel fabrication plant, UF_6 production plant, etc.).

This section considers that shipments originate at the UO_2 fuel fabrication plant. If the shipment originates at the UF_6 production plant, the total number of shipments will not be increased from that already considered in WASH-1248.[1]

It is assumed that the MOX fuel rods are shipped from the MOX fuel fabrication plant to the UO_2 fuel fabrication plant. These MOX rods are then incorporated into fuel assemblies at the UO_2 fuel fabrication plant.

Low level alpha wastes are generated at the MOX fuel fabrication plant and transported to a waste storage facility. It is anticipated that these alpha-bearing wastes will be stored at a Federal waste repository.

The text which follows presents (1) a description of the package designs for the required transport steps, (2) a description of the anticipated transport conditions for each of the types of shipment, and (3) an evaluation of the environmental effects of each.

3.2 Packaging Descriptions

3.2.1 Unirradiated UO_2

Typically for enrichments of less than one percent, UO_2 is packaged in 55-gallon steel drums which have a capacity of about 0.38 MT.

3.2.2 MOX Fuel Rods

The mixed oxide fuel rods are assumed to be transported in packages similar to those used for UO_2 fuel assemblies--in metal containers which support the fuel along its entire length during transport. Examples of fuel element shipping packages are shown in Figures IV G-2, IV G-3, and IV G-4. These packages may have to be modified slightly to include some neutron shielding in order to meet DOT requirements pertaining to external radiation dose levels for normal conditions of transport. The fuel packages are assumed to contain the equivalent of 2 PWR (500 kg U each) or 2 BWR (200 kg U each) fuel assemblies each.

3.2.3 Low Level Alpha Waste

It is assumed that the solid alpha wastes generated at the MOX fuel fabrication plant would be placed in 55-gallon steel drums. It is further assumed that these drums would be placed inside a steel cargo container, 8 ft by 8 ft by 20 ft, and two cargo containers could be put inside an ATMX-500 or -600 type rail car (shown in Figure IV G-5) that would have a total waste capacity of 1,000 cubic feet. Alternatively, the drums may be loaded in an overpack such as the Super Tiger (shown in Figure IV G-6). These assumptions, with respect to both the drums and the outer packaging, are consistent with current and anticipated future practice in the handling of such wastes by the licensees and prime contractors of the Energy Research and Development Administration.

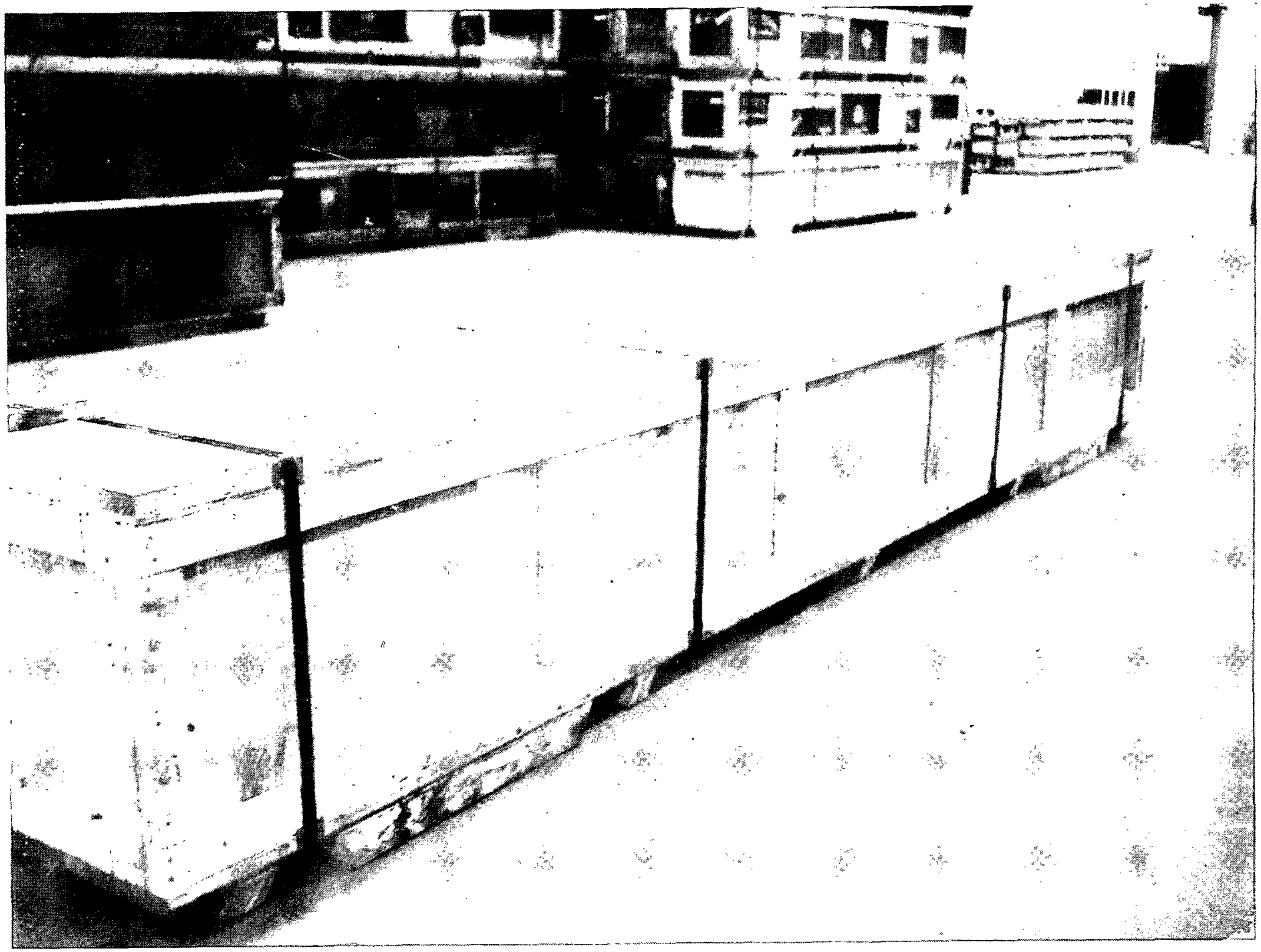

Figure IV G-2 BWR Fuel Element Shipping Container

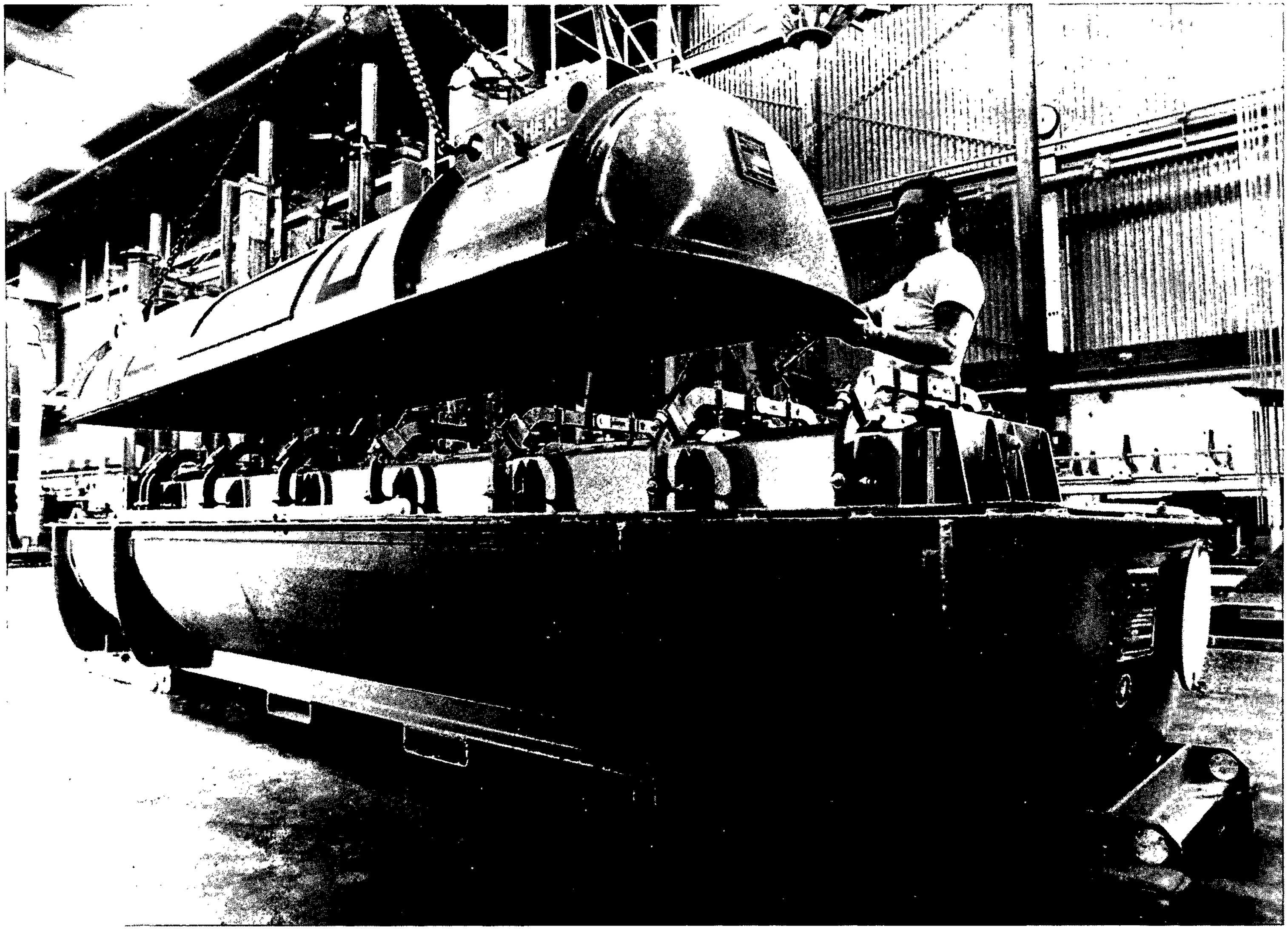

Figure IV G-3 PWR Fuel Element Shipping Container

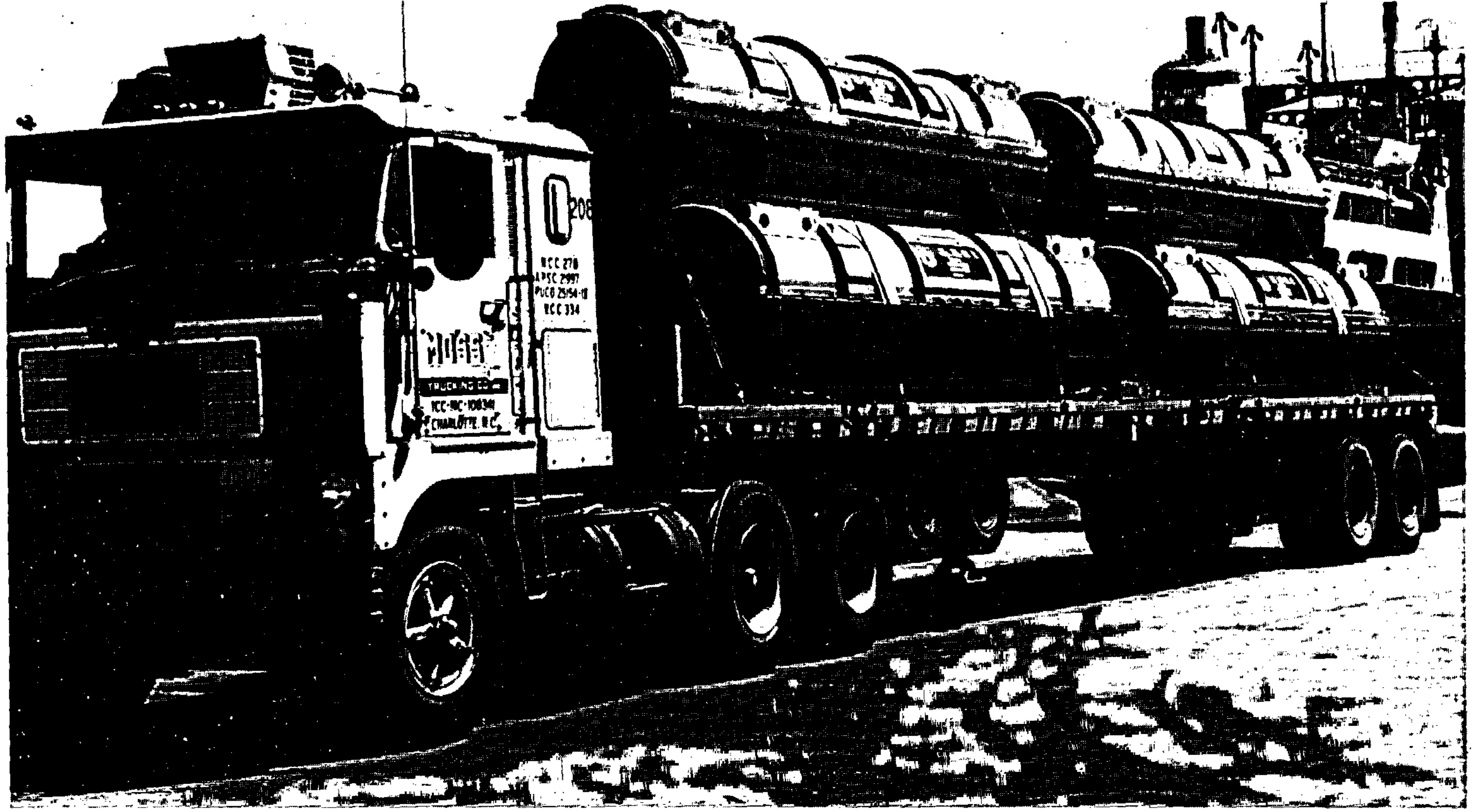

Figure IV G-4 Truckload of Unirradiated PWR Shipping Containers

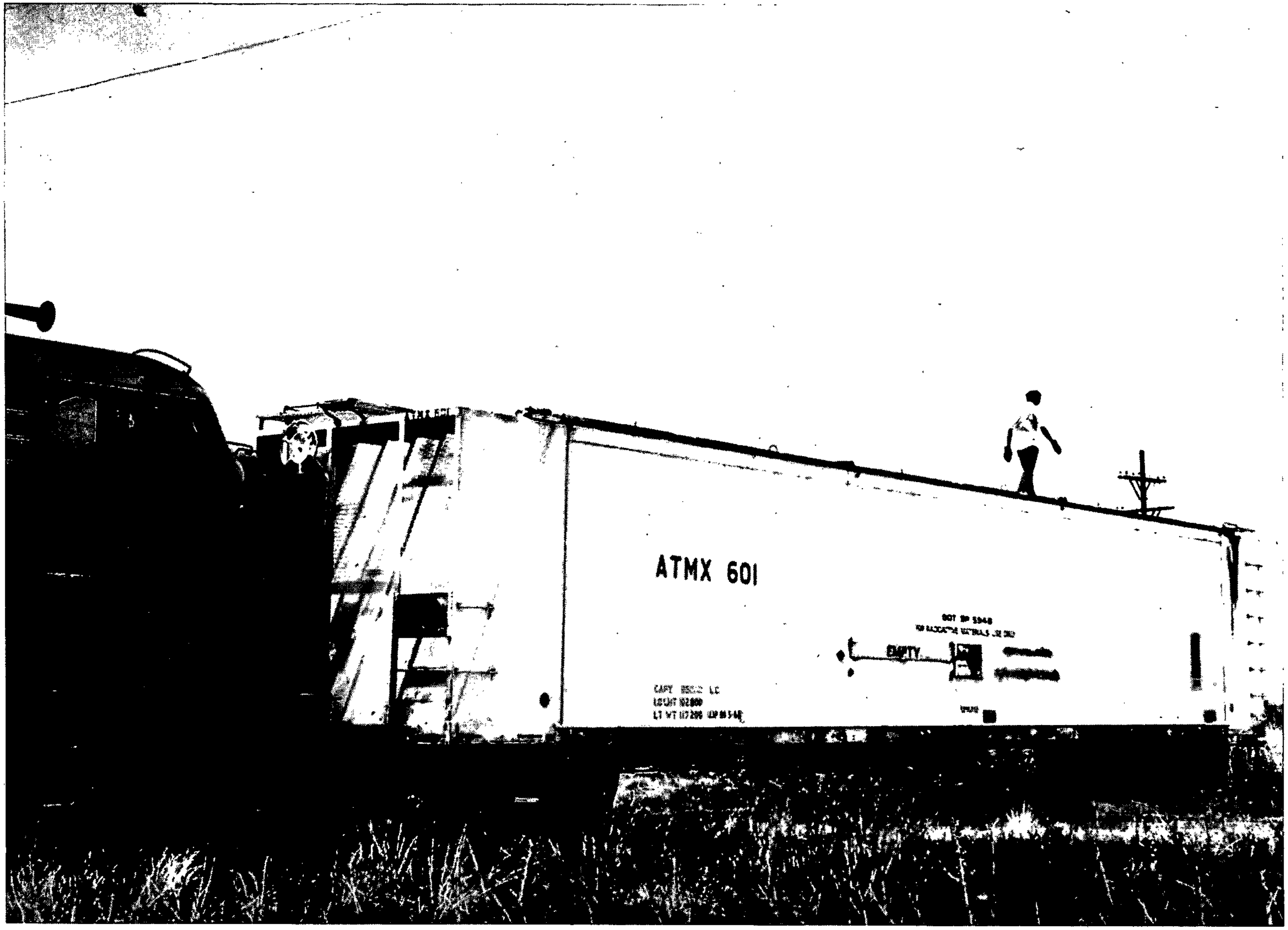

Figure IV G-5 Overpack (ATMX Rail Car)

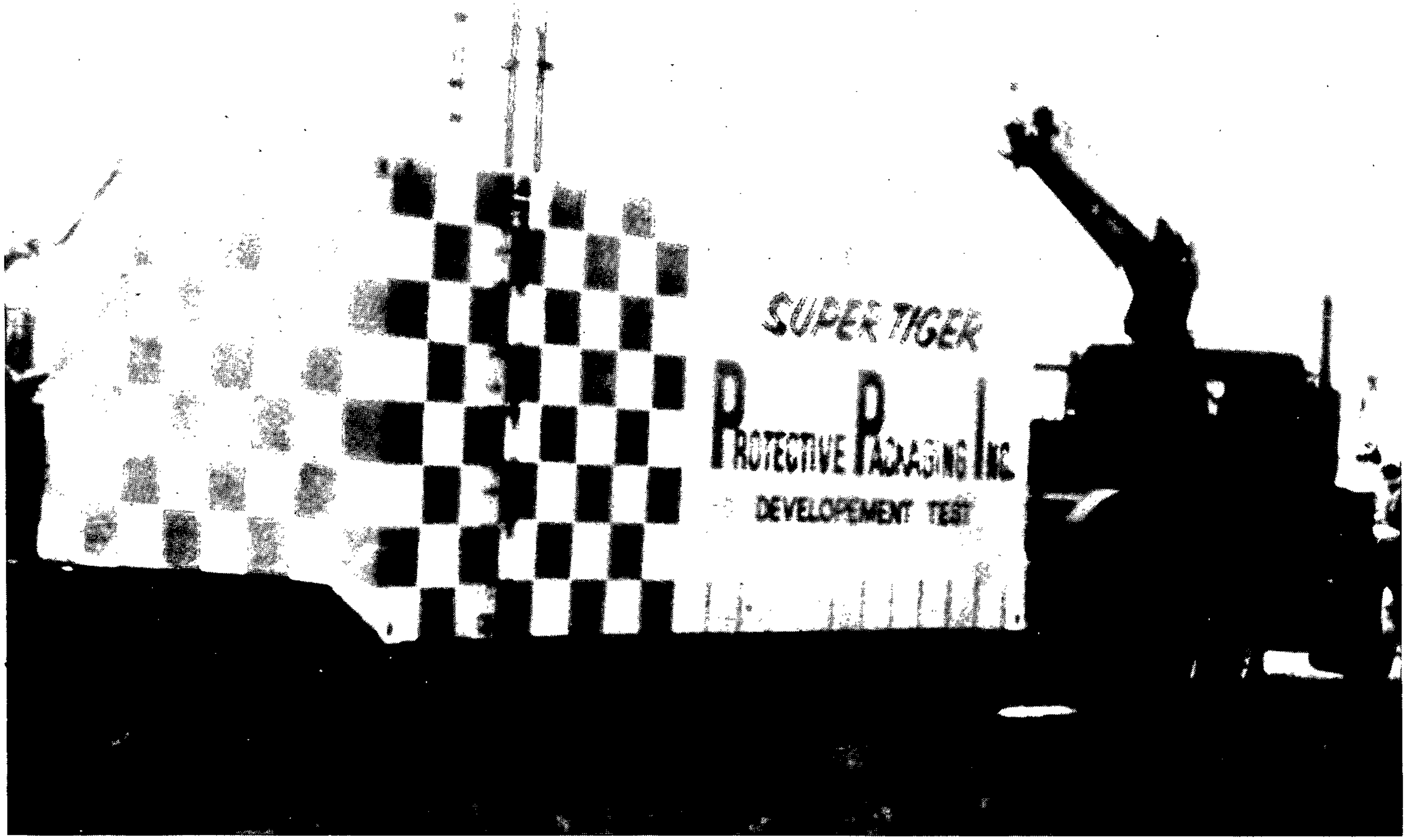

Figure IV G-6 Overpack (Super Tiger)

3.2.4 Irradiated Mixed Oxide Fuel Assemblies

Casks for shipping irradiated mixed oxide fuel assemblies are expected to be similar in design to casks for irradiated uranium oxide fuel assemblies, subject to the constraints of an overall weight limit of approximately 100 MT and the need for massive shielding and high heat dissipation capacity. A typical cask,[7] shown in Figure IV G-7, was designed to have a rating of about 330,000 Btu/hr, which corresponds to a capacity of 10 PWR or 24 BWR assemblies based on a 72,000 Btu/hr heat load per metric ton of heavy metal after reactor discharge and 120 days cooling time. The typical cask without appurtenances, such as a forced cooling system, is a cylinder of composite steel, lead, and uranium construction about eight feet in diameter by 21 feet long with a loaded weight of approximately 90 MT.

The irradiated mixed oxide fuel assemblies will raise the average heat generation rate in all irradiated fuel by 10 to 20 percent[6] over uranium fueled assemblies; the mixed oxide assemblies will have neutron radiation levels that are about two orders of magnitude higher than the levels for irradiated UO_2 fuel.[8] The increase in radiation level will require additional neutron shielding around the fuel cask and a reduction in the number of assemblies per shipment (from 10 to 8 PWR or from 24 to 20 BWR assemblies per rail cask, or from 2 to 1 PWR or from 3 to 2 BWR assemblies per truck cask, of the assumed designs) to take account of the increased heat generation rate per assembly. As noted, design changes are not expected to change the gross characteristics of the irradiated fuel casks, such as weight, overall size, and heat dissipation capacity.

3.2.5 Plutonium Oxide Packaging

Present plutonium oxide packages contain a few kilograms of the oxide contained in sealed metal cans within an inner, gasketed steel container supported within an outer steel drum of 10 to 110 gallon capacity. The gasketed steel cylinder is supported within the steel drum by thermal and shock insulating material such as cane fiberboard, vermiculite, or foamed phenolic plastic. Mass limits of up to 4.5 kg plutonium per package are defined in present package designs primarily by heat dissipation requirements.[9] The NRC regulations (10 CFR Part 71) have been revised to require that plutonium in excess of 20 curies be shipped as a doubly contained solid after June 17, 1978.

In the Safeguards Supplement to this statement, a conceptualized semi-trailer Integrated Container-Vehicle (ICV) was assumed. The ICV consists of a cylindrical steel secondary pressure vessel containing a number of primary pressure vessels loaded with plutonium oxide. The primary pressure vessels would carry four canisters, each holding 18 kg of plutonium. Seven primary pressure vessels would be carried in the vehicle, giving a payload of about 500 kg of plutonium oxide. It is expected, and therefore, assumed in this analysis, that licensees will be using a transport vehicle of this type and capability.

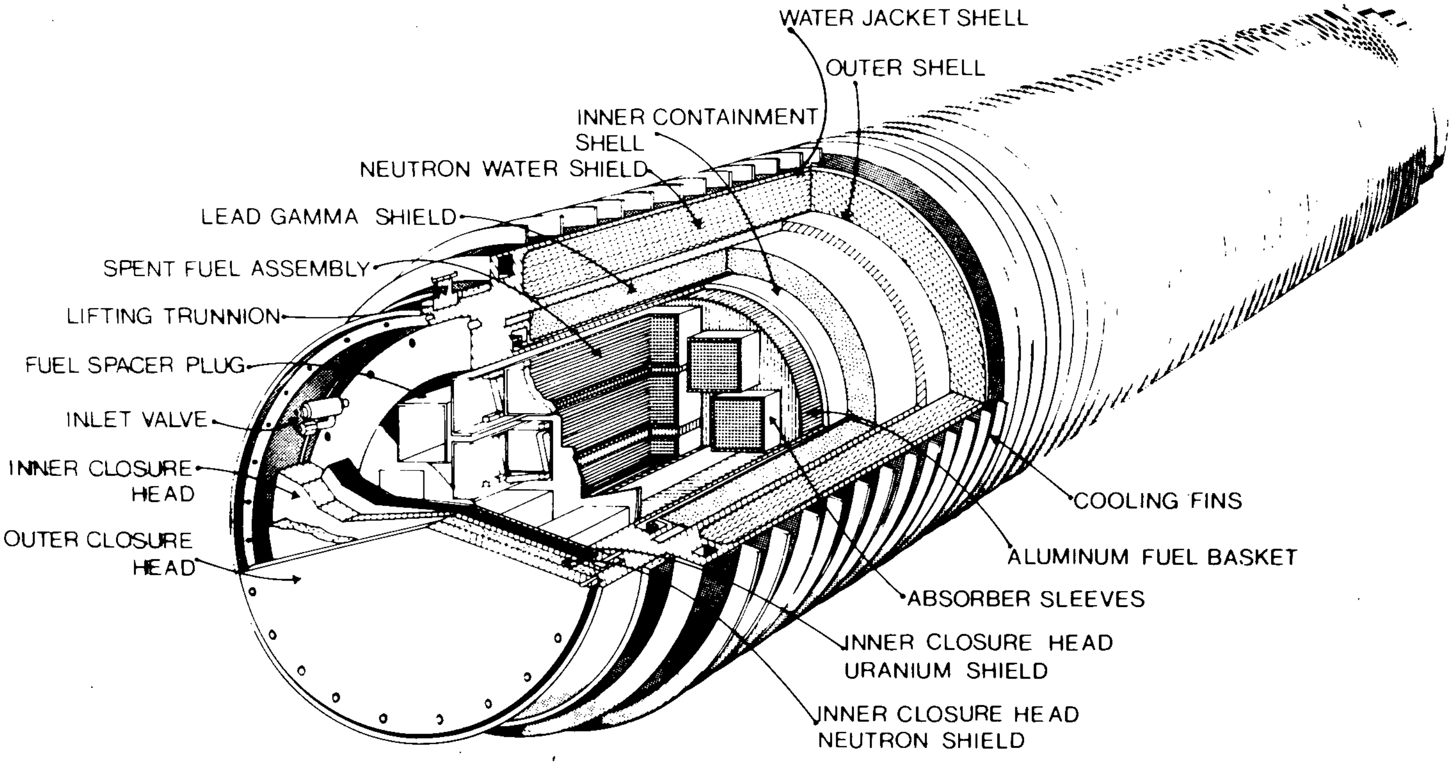

Figure IV G-7 Typical Cask (NLI-10/24)

NL INDUSTRIES INC.
NUCLEAR DIVISION
WILMINGTON. DEL.
NLI 10 24 RAIL CASK

3.2.6 High Level Waste (HLW) Packaging

The solidified fission product waste from fuel reprocessing must be shipped for long-term management at a Federal waste repository. Such shipments must meet limits for size, weight, containment, and heat generation rate for the individual waste containers that will be established for the repository. These requirements are expected to result in casks similar to those used for irradiated fuel assemblies. Each cask is expected to hold about 60 to 75 cubic feet of solidified waste and to weigh 110 MT. The corresponding heat load of about 150,000 Btu/hr is only about one-half that of the cask which was assumed for shipping irradiated fuel, and it thus appears possible that future waste casks may be designed to be capable of dissipating heat comparable to that of existing spent fuel casks.[10]

A canister of solidified high level waste will contain the waste from about 3.2 metric tons of fuel processed, based on about 2 cubic feet of solidified waste per metric ton and a canister of about 1 foot inside diameter by 10 feet long (6.28 cubic feet per canister). A canister containing waste from LWR fuel without plutonium recycle, aged for 10 years, will release about 3.4 kilowatts of heat. The average heat from wastes with plutonium recycle, assuming uniform mixing, will be 10 to 20 percent higher.[6] The high level waste generated in the year 2000 (from 10,000 metric tons of heavy metal processed), by present regulations (10 CFR Part 50, Appendix F), would have to be solidified by the year 2005 and would fill about 3,100 canisters. The solidified high level waste must be shipped by the year 2010, 10 years after separation of the waste from the recovered uranium and plutonium.[11]

In this analysis it has been assumed that high-level waste from spent mixed oxide fuel elements would be uniformly mixed with the waste from spent enriched UO_2 fuel elements. In the highly unlikely event that reprocessing of spent mixed oxide fuel would be done separately from the reprocessing of spent enriched uranium fuel--and the wastes segregated--heat released from the segregated mixed oxide wastes, aged 10 years, may be about 3 times the average released from uniformly mixed waste. Heat released from such segregated waste would be about 13 kilowatts per canister. The number of such canisters per shipment, based on a 45 kilowatt per shipment with the assumed cask design, may, therefore, be limited to about four canisters per shipment. This compares to the assumed capacity of the rail cask of 12 canisters per shipment. Thus, if the waste from reprocessing of mixed oxide fuels is segregated, the number of shipments for this waste (about 10% of the total waste from reprocessing) would be about 3 times that for this fraction of waste if plutonium recycle were not used. This would result in an increase of about 10 to 20 percent in total shipments of solidifed waste from reprocessing, with plutonium recycle, assuming 10 years storage prior to shipping.

The effect of recycle on HLW shipments is thus a function of the cask design, the degree of wastes at the reprocessing plant, and the decay period before shipment. Since longer decay periods could be used to reduce the heat evolution rate of waste containers, the estimate of a 10 to 20% increase in HLW shipments ascribable to plutonium recycle is deemed conservative.

In the case of no uranium or plutonium recycle it is assumed that irradiated fuel assemblies will be held for 10 years at fuel storage pools in the same physical location as fuel reprocessing plants for long-term management. The fuel would be shipped by the year 2010 to a Federal waste repository. Shipping casks are assumed to have a capacity of 10 PWR or 24 BWR assemblies, as for present irradiated spent fuel shipping casks.

3.2.7 Low Level Alpha Waste

It is assumed that the solid alpha wastes generated at the reprocessing plant would be placed in 55-gallon steel drums. The further assumption is made that these drums would be placed inside a steel cargo container, 8 ft by 8 ft by 20 ft, and two cargo containers could be put inside an ATMX-500 or -600 type railcar (shown in Figure IV G-5) that would have a total waste capacity of 1,000 cubic feet.[12] This assumption is consistent with current and anticipated future practice in handling such wastes by licensees and prime contractors of the Energy Research and Development Administration.

Cladding hulls are assumed to be compacted to an average of 70% theoretical density (2.1 cu ft per metric ton of heavy metal) and placed in disposable steel canisters that would have a volume of 3.5 cu ft each. Thirty-six disposable steel canisters would be placed in a shipping cask that would have a shield equivalent to 5 inches of lead. The cask may have a length of 12.5 feet, a diameter of 7.5 feet, and may weigh 94 tons.

3.3 Transport Conditions

3.3.1 Unirradiated UO_2

It is assumed that the shipments of UO_2 from the fuel fabrication plant to the MOX fuel fabrication plant will be made by exclusive-use truck with approximately 40 drums (holding 0.38 MT UO_2 per drum) loaded per vehicle. The resulting net weight per vehicle is thus about 15.2 MT of UO_2. Shipments are transported an average distance of 200 miles and require a transit time of about one day. A total of about 190 shipments is projected for about the year 2000 to meet the industry's needs for uranium oxide (2,525 MT as heavy metal) to be blended with plutonium oxide.

3.3.2 MOX Fuel Rods

Each shipment of MOX fuel rods from the MOX fuel fabrication plant to the UO_2 fuel fabrication plant is expected to be made by truck.

A typical shipment of MOX fuel rods is estimated to consist of about six packages of PWR fuel rods or 16 packages of BWR fuel rods, each package containing sufficient rods for two fuel assemblies. The present PWR fuel packages weigh up to 4,000 kg and the BWR packages weigh up to 1,300 kg. Shipments are assumed to be transported 200 miles with a transit time of about one day. A total of about 440 shipments is projected in the year 2000, to transport 2,650 MT (as heavy metal) of MOX fuel.

3.3.3 Low Level Alpha Waste

The anticipated quantities of alpha waste generated at the mixed oxide fuel fabrication plants and the characteristics of these wastes are based on industrial experience and the planned procedures for waste treatment. Waste treatment procedures will include incineration of combustibles, solidification of liquid wastes, and mechanical compaction. All these processes create dense, heat resistant, unreactive solids.

It is projected that 58,800 cu ft of material will be shipped by rail and 14,700 cu ft of material will be shipped by truck in about the year 2000. Rail shipments may be about one hundred 55-gallon drums per (ATMX) railcar. Truck shipments may be 42 drums (55-gallon) enclosed in an overpack (Super Tiger). All shipments are estimated to take 7.5 days over 1,500 mile routes to a Federal repository.

3.3.4 Irradiated Mixed Oxide Fuel Assemblies

Because of the weight of casks for irradiated fuel, most shipments are expected to be made by rail. The necessary addition of neutron shielding to casks for irradiated mixed oxide fuel assemblies will tend to increase cask weight and add to the impetus for rail transport. However, many nuclear power plants (approximately 10% to 12%)[14] do not have rail service to the plant site. For this reason these plants are restricted to highway shipments of lighter, smaller capacity casks. It is estimated that 40 percent of the fuel material will be shipped by truck. The estimated average distance from the nuclear power plant site to the fuel reprocessing plant is 1,000 miles by rail and 500 miles by truck. This journey would require an average transit time of about five days by rail and 2.5 days by truck.

For a mature industry, a total of approximately 7,510 irradiated fuel shipments per year will be required by the year 2000 to transport 10,250 MT of heavy metal from all reactors in the fuel cycle. About 6,270 shipments would be required without plutonium recycle; the additional 1,240 shipments are ascribable to plutonium recycle.

3.3.5 Plutonium Oxide

Following present practice all shipments of plutonium oxide from the fuel reprocessing plant are expected to be made by exclusive-use truck. Typically, 500 kg of plutonium oxide (440 kg Pu) are loaded per vehicle. By the year 2000, with uranium recycle, there would be 170 shipments to transport 75 MT of plutonium from the reprocessing plant to a storage facility, assumed to be 300 miles away. With uranium and plutonium recycle, 280 shipments would be required to transport 124 MT of plutonium to storage or to the mixed oxide fuel fabrication plant.

Industry planning indicates that the mixed oxide fuel fabrication plants will probably be located near the fuel reprocessing plants. Thus the plutonium shipping distance should average no more than 300 miles and transit time about 1.5 days.

3.3.6 High Level Waste

The expected 110 MT gross weight of each HLW cask requires that all HLW shipments be made by rail. Shipments of HLW are assumed to be transported 1,500 miles because of the possible siting of the waste repository west of the Mississippi River and the probable location of the fuel reprocessing plants in the eastern part of the United States, near the bulk of the nuclear power plants. Present regulations require shipment of solidified wastes to a repository within 10 years after separation of the fission products, so that wastes from the 10,250 MT of fuel to be reprocessed by about the year 2000 may be shipped in the following decade or sooner. Therefore, the fuel to be reprocessed at that time represents a commitment to make about 320 HLW shipments without plutonium recycle, or about 380 shipments with recycle no later than the year 2010.

For the no uranium or plutonium recycle case, the irradiated fuel elements are assumed to be shipped as high level waste over 1,500 mile routes to a western repository. Thus the 10,250 MT of spent fuel would represent a commitment to make about 1,700 HLW shipments no later than the year 2010.

3.3.7 Low Level Alpha Waste

Shipments are anticipated to be made by rail and may consist of about one hundred 55-gallon drums per overpack (ATMX railcar) or one hull cask per rail shipment. The shipments are assumed to be transported 1,500 miles with a transit time of about 7.5 days.

3.4 Effects on the Environment for Normal Conditions of Transport

3.4.1 Heat

The comparison of projected heat release data for added transportation steps with recycle is shown in Table IV G-5.

It can be seen from the typical values given in Table IV G-5 that the heat load for each type of added shipment will have no significant impact on the environment, since no shipment discharges more than 330,000 Btu/hr. The listed rates of heat generation may be compared with the 180,000 Btu/hr released by a 100 hp truck engine operating at full power.[2]

3.4.2 Weight and Traffic Density

The number of packages per vehicle will be within legal weight limitations, and hence no significant impact is predicted from vehicle weights.

It is estimated that in the year 2000 the industry will require 1,940 additional shipments due to the added transportation steps, which would necessitate an additional total vehicle distance of approximately 1,090,000 miles. As noted in paragraph 2.12, the total truck miles traveled on U.S. highways in 1974 was estimated to exceed 55 billion. The increase in mileage is less than 2×10^{-5} of total truck travel and

Table IV G-5

COMPARISON OF HEAT GENERATION RATES IN SHIPMENTS FOR ADDED TRANSPORTATION STEPS WITHOUT AND WITH RECYCLE

	Heat Generation Rate (Btu/hr/shipments)		
Material Transported	No U or Pu Recycle	With U Recycle	With U & Pu Recycle
Unirradiated UO_2			
0.38 MT UO_2/drum, 40 drums/shipment	--	--	negligible
MOX Fuel Rods			
Rods equivalent to 12 PWR or 32 BWR assemblies/shipment	--	--	12,000
Low Level Alpha Waste			
1,000 cu ft/shipment from mixed oxide fuel fabrication plant	--	--	<250
Irradiated Fuel Assemblies			
10 PWR or 24 BWR assemblies/shipment w/o Pu recycle or 8 PWR or 20 BWR assemblies/ship w/Pu recycle	330,000	330,000	330,000
Plutonium Oxide			
0.50 MT/shipment	--	32,000	32,000
High Level Waste**			
10 PWR or 24 BWR assemblies/shipment w/o U or PU recycle or 9 to 12 canisters/shipment	330,000	150,000	150,000
Low Level Alpha Waste			
1,000 cu ft/shipment (126 cu ft for hulls) from reprocessing plant	--	<250	<250

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*The heat dissipation capability of the irradiated fuel cask is limited, and thus the number of shipments is increased in the U and Pu recycle case to provide for the higher heat evolution rate. (The heat dissipation capability of the NLI-10/24 cask is based on the applicant's original Safety Analysis Report.)[7]

**The heat dissipation capability of the high level waste cask is limited, and thus the number of shipments is increased to provide for the higher heat evolution rate in the recycle waste. High level waste cask design may have sufficient capacity to avoid this increment.

thus too small to have a measurable effect on the environment from the increase in traffic density.

3.4.3 Radiation Dose During Transport

The bases used to estimate radiation doses to transport workers and to members of the general public, incurred under normal conditions of transportation, are described below. Since the radioactive material shipments discussed are added shipments as a result of plutonium recycle, the attendant doses are likewise entirely ascribable to recycle.

Doses to domestic animals and unexposed photographic film in transit are expected to be negligible. Exposure of domestic animals or pets is unlikely because the radioactive material considered in the additional fuel shipments is transported in exclusive use vehicles. Unexposed film is not likely to be exposed in these shipments because the weight and nature of the material and packaging usually preclude the loading of other freight on the car or truck.

3.4.3.1 Unirradiated UO_2 Shipments

Each shipment of UO_2 to the MOX fuel fabrication plant is assumed to require two drivers. Each driver is assumed to spend four hours in the truck cab and about 0.5 hour outside the truck cab for the 200 mile trip. The radiation level inside the cab of the truck transporting the uranium is assumed to be 0.02 mrem/hr above background and 0.40 mrem/hr at 3 feet from the side of the truck. Each driver transporting the uranium oxide would receive an average dose of 0.28 mrem per shipment. Assuming a driver makes 15 trips per year, each driver would receive a dose of 4.5 mrem/yr. The cumulative annual dose to all drivers would be 0.11 person-rem for 190 shipments.

Members of the general public (onlookers) might be exposed at en route truck stops for fuel and eating. Trucks carrying radioactive material are required to be placarded on both sides and front and rear as "RADIOACTIVE." A member of the general public who spends 3 minutes at an average distance of 3 feet from the truck or railcar might receive a dose of as much as 0.02 mrem. If two persons, on the average, were so exposed during each shipment, the cumulative annual dose to such onlookers for the 190 shipments by truck would be about 0.008 person-rem.

Approximately 300,000 persons who reside along the five 200 mile routes over which the unirradiated UO_2 is transported might receive a cumulative annual dose of about 0.007 person-rem. This dose is based on an estimated radiation level of 0.1 mrem/hr at 6 feet from the vehicle, one day in transit, and an exposed population density averaging 300 persons per square mile along the route. See Appendix D of WASH-1238[2] for the detailed calculations.

3.4.3.2 MOX Fuel Rod Shipments

Each shipment of MOX rods to the UO_2 fuel fabrication plant is assumed to require two drivers. Each driver is assumed to spend 4 hours in the cab of the truck and about 0.5 hour outside of the cab for the 200 mile trip. The radiation level inside the cab of a truck transporting MOX fuel rods is assumed to be 2 mrem per hour above background and 16 mrem per hour at 3 feet from the side of the truck (corresponding to DOT maximum limits under normal conditions of transport). Each of two drivers would receive an average dose of 16 mrem per shipment. Assuming a driver makes 15 trips per year, each driver would receive a dose of 0.24 rem/year. The cumulative annual dose to all drivers would be 14.1 person-rem for 440 shipments.

Members of the general public (onlookers) might be exposed at en route truck stops for fuel and eating. Trucks carrying radioactive material are placarded on both sides and front and rear as RADIOACTIVE. A member of the general public who spends 3 minutes at an average distance of 3 feet from the truck or railcar might receive a dose of as much as 0.8 mrem. If two persons, on the average, were so exposed during each shipment, the cumulative annual dose to such onlookers for the 440 shipments by truck would be about 0.7 person-rem.

Approximately 300,000 persons who reside along the five 200 mile routes over which the MOX fuel rods are transported might receive a cumulative annual dose of about 1.6 person-rem if MOX fuel rods are transported by truck. In this case, the regulatory radiation level limit of 10 mrem/hr at 6 feet from the vehicle was used to calculate the integrated dose to persons in an area between 100 feet and 1/2 mile on both sides of the shipping route. It was assumed the shipment would take one day and the population density would average 300 persons exposed per square mile along the route. See Appendix D of WASH-1238[2] for the detailed calculations.

3.4.3.3 Low Level Alpha Waste

For the truck shipments, it is assumed that two truck drivers might spend four hours in the cab and perhaps 30 minutes outside the truck at an average distance of 3 feet from the packages per day for 7.5 days. Assuming the radiation level in the cab (due to neutrons and gamma photons) is 2 mrem/hr and the level at 3 feet from the package is 16 mrem/hr, each truck driver might receive as much as 120 mrem during each shipment. If the same two truck drivers were used for, say five shipments, each could receive as much as 0.6 rem. The cumulative annual dose to all drivers for 50 shipments would be about 12 person-rem.

Garagemen or train brakemen might be expected to spend from one to 10 minutes each in the vicinity of the cab of the truck or the railcar during the 1,500 mile trip, for an average exposure of about 0.5 mrem per shipment. With 11 different garagemen or brakemen involved along the route, the cumulative dose for 50 truck shipments during the year is estimated to average about 0.28 person-rem and for 83 rail shipments 0.46 person-rem.

Despite the fact that railcars carrying irradiated fuel shipments are placarded on both sides and trucks on both sides and the front and rear as RADIOACTIVE, members of the general public (onlookers) might receive radiation exposure at en route truck stops for fuel and eating and at railroad stations. A member of the general public who spends 3 minutes at an average distance of 3 feet from the truck or railcar might receive a dose of as much as 0.8 mrem. If 16 persons, on the average, were so exposed during each shipment, the cumulative annual dose to such onlookers for the 50 shipments by truck would be about 0.64 person-rem and 1.1 person-rem for the 83 rail shipments.

Approximately 1,580,000 persons who reside along 1,500 mile (three routes each for truck and rail shipments east of the Mississippi River and two routes each for the 800 miles west of the Mississippi River) routes over which the low level alpha waste is transported might receive a cumulative annual dose of about 2.3 person-rem (0.87 person-rem from truck shipments and 1.44 person-rem from rail shipments). The regulatory radiation level limit of 10 mrem/hr at 6 feet from the vehicle was used to calculate the integrated dose to persons in an area between 100 feet and 0.5 mile on both sides of the shipping route. It was assumed the shipment would travel 200 miles per day and the population density would average 300 persons exposed per square mile along the route east of the Mississippi River and 100 persons west of the river. See Appendix D of WASH-1238[2] for the detailed calculations.

3.4.3.4 Irradiated Mixed Oxide Fuel Assemblies

As indicated in paragraph 4.3.2, 40 percent of irradiated fuel shipments from the reactor site to the fuel reprocessing plant or storage are assumed to involve truck transportation while the balance will be entirely by rail. Considering the very limited fuel assembly capacity (1 to 3 assemblies) of casks for irradiated fuel suitable for truck transport and the increased need for neutron shielding of mixed oxide assemblies (which increases the difficulty of meeting legal weight limitations for over-the-road transport), it is unlikely that there will be any significant increase in truck use. In this study it is assumed that such truck shipments of fuel will move over an average distance of 500 miles. The 1,000 mile average total distance for rail shipments from the reactor to the fuel reprocessing plant is based on the locations of the reprocessing plants now planned or built. Along with the majority of reactor sites, they are located in the eastern part of the United States.[5,15] It is assumed that storage facilities, if required, would have the same general location distribution.

Each truck shipment of irradiated fuel assemblies from the reactor to the reprocessing plant or storage is assumed to require two drivers, each of whom spends 10 hours in the cab and 30 minutes outside the cab for the 500-mile trip. The radiation level inside the cab of the truck is assumed to be 2 mrem per hour and 16 mrem per hour three feet from the surface of the truck based on the DOT maximum radiation levels. Each driver would receive an average dose of 28 mrem per shipment: composed of 20 mrem while occupying the cab and 8 mrem while outside the cab at an average distance of 3 feet from the side of the truck. If the same driver is used for 5 shipments per year, he would receive 0.14 person-rem. The cumulative annual dose to all drivers would then be 280 person-rem for 5,020 shipments without plutonium recycle and 346 person-rem for 6,170 shipments with plutonium recycle.

Garagemen might be expected to spend from one to 10 minutes around the cab of the truck. Each of 5 garagemen could be exposed to about 0.5 mrem. The cumulative dose to all garagemen for 5,020 shipments without Pu recycle would be 13 person-rem; for 6,170 shipments with Pu recycle it would be 15 person-rem.

For rail shipments, train brakemen would be expected to spend from one to 10 minutes each in the vicinity of the car during the trip, for an average exposure of about 0.5 mrem per shipment. With an average of 10 different brakemen involved along the route, the cumulative annual dose for 1,250 shipments without plutonium recycle would be 6.3 person-rem and with plutonium recycle (90 additional shipments) 6.7 person-rem.

Irradiated fuel shipments are transported as full loads. Since the casks are not handled en route, under normal conditions there would be no routine exposure of the carrier's freight handlers, either by truck or by rail.

In-transit storage of these casks is unlikely except while mounted on the vehicle (truck or rail) at truck stopover points, in terminal yards, or in railroad switchyards. There will be little, if any, across-the-dock handling of these casks outside the nuclear power plant and the fuel storage or recovery plant sites.

Members of the general public (onlookers) are normally excluded from loading and unloading operations, but some exposures might occur at en route truck stops and at railroad terminals. Railcars carrying irradiated fuel shipments are placarded on both sides and trucks on both sides and the front and rear as RADIOACTIVE. A member of the general public who spends 3 minutes at an average distance of 3 feet from the truck or railcar might receive a dose of as much as 0.8 mrem. If 5 persons, on the average, were so exposed during each truck shipment and 10 persons for each rail shipment, the cumulative annual dose to such onlookers for UO_2 fuels for 5,020 shipments by truck would be about 20 person-rem and for 1,250 shipments by rail, about 10 person-rem. The cumulative annual dose to onlookers for UO_2 and MOX fuels for 6,170 shipments by truck would be about 24 person-rem and for 1,340 shipments by rail, about 11 person-rem.

Approximately 1,800,000 persons who reside along the twelve 500-mile truck routes and approximately 2,880,000 persons who reside along the twelve 1,000-mile rail routes over which the irradiated fuel is transported might receive a cumulative annual dose of 46 person-rem for 5,020 truck shipments and 18 person-rem for 1,250 rail shipments without plutonium recycle and 56 person-rem for 6,170 truck shipments and 20 person-rem for 1,340 rail shipments with plutonium recycle. In this case, the regulatory radiation level limit of 10 mrem/hr at 6 feet from the vehicle was used to calculate the integrated dose to persons in an area between 100 feet and 0.5 mile on both sides of the shipping route. It was assumed the shipment would travel 200 miles per day and the population density would average 300 persons exposed per square mile along the route for distances not exceeding 700 miles per trip and 100 persons per square mile beyond 700 miles. See Appendix D of WASH-1238[2] for the detailed calculations.

3.4.3.5 Plutonium Oxide

Dose predictions for shipping plutonium oxide from the reprocessing plant are based on assumptions similar to those used in assessing the radiation effects (see paragraph 4.4.3.1) of unirradiated fuel shipments as regards numbers of drivers (2) per vehicle, dose rates (2 mrem/hr) en route, average truck speed (50 mph) and trips (5) per driver per year. These bases, combined with the average transport distance of 300 miles, result in predicted dose rates per truck driver of 20 mrem per shipment (6 hours in cab and 0.5 hour 3 feet from the side of the truck). The cumulative annual dose to all drivers would be 11 person-rem for 280 shipments with Pu recycle.

Servicing the truck may require three garagemen, spending from one to 10 minutes around the cab of the truck. They could be exposed to about 0.5 mrem. The cumulative annual dose to all garagemen for 280 shipments with Pu recycle would be 0.42 person-rem.

Members of the general public (onlookers) are normally excluded from loading and unloading operations, but exposures might occur at en route truck stops for fuel and eating. Trucks are placarded on both sides and the front and rear RADIOACTIVE. Members of the general public are unlikely to remain near a truck more than a few minutes. If a person spends 3 minutes at an average distance of 3 feet from the truck, the dose would be about 0.8 mrem. If 3 persons, on the average, were so exposed, the total annual dose to such onlookers would be about 0.7 person-rem with plutonium recycle.

The radiation level at 6 feet from a vehicle loaded with packages of plutonium oxide is assumed to be no more than 10 mrem/hr. Assuming that the vehicle travels 200 miles per day, and the mean population density along the route is 300 persons exposed per square mile, then for a trip of 8 routes of 300 miles one way, and 280 trips per year with plutonium recycle, the cumulative annual dose to approximately 720,000 persons in an area along the routes between 100 feet and 0.5 mile on either side of the vehicle would be about 1.5 person-rem with plutonium recycle. See Appendix D of WASH-1238[2] for detailed calculations.

3.4.3.6 High Level Waste

Dose predictions for shipping HLW by rail from the reprocessing plant to a repository are based on assumptions similar to some of those used in assessing the radiation effects (see paragraph 4.4.3.2) of irradiated fuel shipments by rail. Similar assumptions as regards population density, train speed, and dose rates were coupled with the average transport distance of 1,500 miles (700 miles east and 800 miles west of the Mississippi River) to provide estimated total annual doses to train brakemen (11 brakemen per shipment) of 1.8 person-rem for the 330 shipments of HLW per year with uranium recycle only and 2.1 person-rem for the 380 shipments with plutonium recycle. The annual doses to the general public were calculated to be 9.9 person-rem (16 onlookers receive 4.2 person-rem, per shipment, persons along the route receive 5.7 person-rem) with uranium recycle only and 12 person-rem (16 onlookers receives 4.9 person-rem per shipment; persons along the route receive 6.6

person-rem) with uranium and plutonium recycle. The waste shipments with uranium recycle only include the shipments of waste plutonium recovered from the irradiated uranium fuel.

In the case of no uranium or plutonium recycle, the assumptions for the shipment of irradiated fuel elements to the Federal repository would be the same as in the preceding paragraph. The total annual doses to train brakemen is 9.4 person-rem for the 1,700 shipments. The annual doses to the general public were calculated to be 52 person-rem (onlookers receive 22 person-rem; persons along the route receive 30 person-rem).

3.4.3.7 Low Level Alpha Waste

Train brakemen would be expected to spend from one to 10 minutes each in the vicinity of the car during the trip, for an average exposure of about 0.5 mrem per shipment per brakeman. With 11 different brakemen involved along the route, the cumulative annual dose for 980 shipments with or without plutonium recycle is estimated to average about 5.4 person-rem.

Members of the general public (onlookers) might be exposed at railroad stations. Railroad cars carrying radioactive material shipments are placarded on both sides as RADIOACTIVE. A member of the general public who spends 3 minutes at an average distance of 3 feet from the railcar might receive a dose of as much as 0.8 mrem. If 16 persons, on the average, were so exposed during each shipment, the cumulative annual dose to such onlookers for the shipments by rail would be about 13 person-rem without plutonium recycle and about 13 person-rem with plutonium recycle.

Approximately 790,000 persons who reside along the 1,500 miles (3 routes east of the Mississippi River and 2 routes for the 800 miles west of the Mississippi River) of routes over which the low level alpha waste is transported might receive a cumulative annual dose of about 17 person-rem without plutonium recycle and about 17 person-rem with plutonium recycle. In this case, the regulatory radiation level limit of 10 mrem per hour at 6 feet from the vehicle was used to calculate the integrated dose to persons in an area between 100 feet and 0.5 mile on both sides of the shipping route. It was assumed the shipment would travel 200 miles per day and the population density would average 300 persons (100 persons west of the Mississippi River) exposed per square mile along the route. See Appendix D of WASH-1238[2] for the detailed calculations.

4.0 DESCRIPTION AND EVALUATION OF TRANSPORTATION STEPS AFFECTED BY URANIUM AND PLUTONIUM RECYCLE FOR NORMAL CONDITIONS OF TRANSPORTATION

4.1 Introduction

In addition to requiring the seven new types of shipments described in the foregoing section, the use of recycled plutonium would potentially modify the effects of one transportation step already a necessary part of the uranium fuel cycle: unirradiated fuel from the uranium oxide fuel fabrication plant to the reactor. The modified impacts result primarily from increases in levels of gamma and neutron radiation and the rate of heat generation in the affected shipments as a result of changes in fuel composition.

This section presents (1) a description of packaging designs for the modified stream, (2) a description of the anticipated transport conditions for the shipment, and (3) an evaluation of the environmental effects.

4.2 Packaging Description

4.2.1 Unirradiated Mixed Oxide Fuel Assemblies

Packaging for transporting unirradiated mixed oxide fuel assemblies from the UO_2 fuel fabrication plant to the reactor is described in paragraph 3.2.2.

4.3 Transport Conditions

4.3.1 Unirradiated Mixed Oxide Fuel Assemblies

In keeping with present practice for unirradiated fuel assemblies, essentially all shipments of mixed oxide assemblies from the UO_2 fuel fabrication plant to the reactor are expected to be made by exclusive-use truck. The present practice of shipping six packages of PWR assemblies (12 assemblies) or 16 packages of BWR assemblies (32 assemblies) per truck is expected to continue with mixed oxide assemblies, although the individual packages may be heavier because of additional neutron shielding. Shipments of fuel assemblies are assumed to involve an average transport distance of 1,000 miles. (The average distance from the fuel fabrication plant to 83 reactors at 55 sites was determined to be approximately 970 miles, with distances ranging from 25 to 3,000 miles.)[13] Packages would be loaded on the truck at the fuel fabrication plant by the shipper, transported by the carrier directly to the nuclear power plant and unloaded by plant personnel, with no intermediate off-loading, storage, or intervehicular transfers en route. No other shipments would be loaded on the vehicle except by the shipper himself. Average transit time would be about five days.

Based on full loads, 2,200 shipments are required to transport the 13,500 MT (heavy metal) of mixed oxide fuel projected in the year 2000. The same number of enriched fuel assembly shipments would be required with or without plutonium recycle, since the mixed oxide assemblies may be considered as replacement for an equal number of enriched uranium fuel assemblies that would otherwise be required.

4.4 Effects on the Environment for Normal Conditions of Transport

4.4.1 Heat

As shown by Table IV G-6, the heat discharged by the modified shipment in the fuel cycle is unchanged by plutonium recycle or is too small to have a significant impact on the environment.

The listed rates of heat generation may be compared with the 180,000 Btu/hr released by a 100-hp truck engine operating at full power.[2]

Table IV G-6

COMPARISON OF HEAT GENERATION RATES IN SHIPMENTS FOR MODIFIED TRANSPORTATION STEP WITHOUT AND WITH RECYCLE

	Heat Generation Rate (Btu/hr/shipment)		
Material Transported	No U or Pu Recycle	With U Recycle	With U & Pu Recycle
Unirradiated Fuel Assemblies			
12 PWR or 32 BWR assemblies/shipment	Negligible	Negligible	12,000

4.4.2 Weight and Traffic Density

The impact on transportation from changes in material characteristics resulting from recycle is lessened by adjusting the radioactive materials contents per package and numbers of packages per shipment to meet heat dissipation capabilities and legal weight limitations, respectively, as described in paragraph 2.11. As indicated in Tables IV G-1 and IV G-2, the industrywide increase in shipments is too small to have a measurable effect on the environment from the increase in normal traffic density.

4.4.3 Radiation

The bases used to estimate radiation doses to transport workers and to members of the general public under normal circumstances of transportation are described below. As discussed in paragraph 3.4.3, exposures of domestic animals or unexposed photographic film are expected to be negligible; therefore, no further mention of such exposures will be included.

4.4.3.1 Unirradiated Fuel Assemblies

The radioactivity in a package of unirradiated UO_2 fuel is estimated to be about 0.5 to 2.0 curies. On the basis of data from NRC licensees and ERDA contractors, the radiation level at the outside surface of a truck containing a load of unirradiated UO_2 fuel would be about 1 mrem/hr, and at 3 feet from the surface of the truck about 0.1 mrem/hr. From actual experience, the level in the cab of a truck would be about 0.01 mrem/hr above the natural background.[2]

However, to assess the effects of MOX fuel assembly transport conservatively, the radiation level is assumed to be 2 mrem per hour inside the cab of a truck transporting MOX fuel and 16 mrem per hour three feet from the surface of the truck, based on DOT maximum radiation levels.

Each shipment of fuel assemblies from the fuel fabrication plant to the reactor site is assumed to require two drivers and to take 20 hours in the cab and about one hour outside the cab for the 1,000-mile trip.

Each driver transporting UO_2 fuel would receive an average dose of 0.3 mrem per shipment: 0.2 mrem while occupying the cab and 0.1 mrem while outside the cab at an average distance of 3 feet from the side of the truck.

Each driver transporting MOX fuel would receive an average dose of 56 mrem per shipment: 40 mrem while occupying the cab and 16 mrem while outside the cab at an average distance of 3 feet from the side of the truck.

If the same driver is used for 5 shipments per year he would receive 0.28 rem. The cumulative annual dose to all drivers would then be 1.3 person-rem without plutonium recycle and 67 person-rem with plutonium recycle.

Garagemen might be expected to spend from one to ten minutes around the cab of the truck. An average of 8.5 garagemen could be exposed (per trip) to about 0.002 mrem without Pu recycle or 0.5 mrem with Pu recycle. The cumulative annual dose to all garagemen for 2,200 shipments without Pu recycle would be 0.04 person-rem; with Pu recycle (587 MOX assembly shipments) it would be 2.5 person-rem.

Members of the general public (onlookers) are normally excluded from loading and unloading operations, but exposures might occur at en route truck stops for fuel and eating. Trucks are placarded on both sides and the front and rear as RADIOACTIVE. Members of the general public are unlikely to remain near a truck more than a few minutes. If a person spends 3 minutes at an average distance of 3 feet from the truck, the dose would be about 0.005 mrem from UO_2 fuels and 0.8 mrem from MOX fuels. If 10 persons, on the average, were so exposed, the total annual dose to such onlookers would be about 0.11 person-rem without plutonium recycle or 4.8 person-rem with plutonium.

The radiation level at 6 feet from a vehicle loaded with packages of unirradiated UO_2 fuel will likely be no more than 0.1 mrem/hr, assuming that the vehicle travels 200 miles per day and the mean population density along the route is 300 persons per square mile for distances not exceeding 700 miles and 100 persons per square mile beyond 700 miles. For a trip of 1,000 miles one way, 12 shipping routes, and 2,200 shipments per year, the cumulative annual dose to approximately 2,880,000 persons in an area along these routes (between 100 feet and 0.5 mile on either side of the vehicle) would be about 0.32 person-rem. See Appendix D of WASH-1238[2] for detailed calculations.

The radiation level at 6 feet from a vehicle loaded with packages of unirradiated MOX fuel will be no more than 10 mrem/hr, assuming that the vehicle travels 200 miles per day, and the mean population density along the route is 300 persons per square mile for distances not exceeding 700 miles and 100 persons per square mile beyond 700 miles. For a trip of 1,000 miles one way, 12 shipping routes, and 587 shipments per year of MOX fuel and 1,613 shipments per year of UO_2 fuel, the cumulative annual dose to approximately 2,880,000 persons in an area along these routes (between 100 feet and 0.5 mile on either side of the vehicle) would be about 8.7 person-rem. See Appendix D of WASH-1238[2] for detailed calculations.

5.0 TRANSPORTATION ACCIDENTS

An earlier study[2] of available statistics on accidents in transportation is the source of the accident probabilities in Table IV G-7 and the categories of accident severity in Table IV G-8. Consideration of the regulatory standards and requirements for package design and quality assurance, results of tests, and past experience support a conclusion that type B packages are likely to withstand all except very severe, highly unusual accidents. In Table IV G-8, the regulatory standards for impact speed and fire duration imposed by 10 CFR Part 71 are noted. Experience gained in studying the package designs for NRC approval indicates that if package failure to retain contents occurs, it would occur only in a very severe, highly unusual accident. For comparison, such an accident is taken in the context of Table IV G-7 to mean an extra severe or an extreme accident. Other studies[3,4,6,27] support this conclusion.

To estimate the potential for radiological effect from transportation accidents due to plutonium and uranium recycle compared to no recycle of these materials, the frequency of extra severe accidents resulting from consideration of the differences in the number of shipment miles for these two modes of fuel management is projected over the period 1975 through 2000 in Table IV G-9. This number essentially expresses the potential, since the accident rate for extremely severe accidents is negligible in comparison to that of extra severe accidents, and since no radioactive releases are assumed for accidents less severe than this category (see Table IV G-7). The results are expressed as the number of extra severe accidents that might be expected over this 26-year period. Since no transportation step can be considered more important than another, the resultant frequency of extra severe accidents for all transportation steps in which differences in transportation activity occur for these two modes of fuel management is the sum of frequencies corresponding to each step, or 4.1×10^{-5} extra severe accidents for the 26-year period. Expressed more simply, this frequency corresponds to one extra severe accident in 20,000 years.

Even if an extra severe accident occurs, package failure and subsequent release of radioactive contents do not occur necessarily but only with some probability. Estimates of the release probability have been discussed elsewhere for particular package designs.[4,14,26] In Reference 4, the frequency of release is estimated by fault tree analyses for plutonium dioxide shipments in the 6M container, as once in 220 accidents of all severities. In Reference 14, fault tree analyses were used to assess the release frequency of typical package designs for shipment of spent fuel, plutonium, high level waste, and noble gaseous waste. The results for spent fuel may be summarized as follows: one out of 100 accidents of all severities will result in a small release of radioactivity, one out of 500 in a medium release, and one out of 10,000 in a large release.[26]

Table IV G-7

ACCIDENT RATES FOR TRUCK AND RAIL
ACCIDENT SEVERITY CATEGORIES[2]

Accident Category	Severity	Accident Rate per vehicle mile	
		Rail	Truck
1	Minor	7.3×10^{-7}	1.3×10^{-6}
2	Moderate	7.9×10^{-8}	3×10^{-7}
3	Severe	1.5×10^{-9}	8×10^{-9}
4	Extra Severe	1.3×10^{-11}	8×10^{-13}
5	Extreme	1.2×10^{-13}	2×10^{-14}

Table IV G-8

ACCIDENT SEVERITY CATEGORIES*

Vehicle Speed at Impact (mph)	Fire Duration (hr)	Accident Category	Severity
0-50	0	1	Minor
0-30	<1/2		
50-70	0	2	Moderate
30-70	<1/2		
0-30	1/2-1		
>70	0-1/2	3	Severe
30-70	1/2-1		
0-50	>1		
>70	1/2-1	4	Extra Severe
50-70	>1		
>70	>1	5	Extreme

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*Reference 2. Note that Type B packages are required to withstand impact corresponding to a velocity of about 30 mph and to withstand a 1,475°F fire of 1/2 hour duration according to 10 CFR Part 71.

Table IV G-9

SUMMARY OF INCREMENTAL EXTRA SEVERE ACCIDENT* FREQUENCIES WITH RADIOACTIVE LOADS, ASCRIBABLE TO ADDED SHIPMENTS WITH U AND Pu RECYCLE

Material Shipped	Mode	Added Shipment Miles** For 26-Year Period	Projected Extra Severe Accidents For 26 Year Period
UO_2	Truck	360,000	2.9×10^{-7}
Mixed Oxide Fuel Rods	Truck	845,000	6.8×10^{-7}
Irradiated Fuel Assemblies	Truck	7,050,000	5.6×10^{-6}
	Rail	1,050,000	1.4×10^{-5}
PuO_2	Truck	807,000	6.5×10^{-7}
Alpha Waste from MOX Plant	Truck	660,000	5.3×10^{-7}
	Rail	1,110,000	1.4×10^{-5}
Alpha Waste From Reprocessing Plant	Rail	17,000,000	2.2×10^{-4}
High Level Waste	Rail	36,300,000 (decrease)	4.7×10^{-4} (decrease)
Total Increment		7,420,000 (decrease)	4.1×10^{-5}***

*This category of severity chosen because packages are designed to withstand extra severe accidents and accidents of less severity would not be expected to breach the packaging. Mileages and hence frequencies correspond to all shipments of the listed materials predicted for the 26 year period.

**Mileages with radioactive load; does not include return trips with empty containers.

***Corresponds to a decrease of one extra severe accident per 20,000 years.

Further details on the performance of specific shipping container types under various postulated unusual conditions and details of the resulting environmental impacts are given in the discussion which follows. The predicted performance of the packaging is based on a variety of information: test data for a variety of packaging,[16-19] engineering analyses performed in packaging design and licensing review[20-22] and analyses in other environmental statements.[1,2] The packaging and materials previously analyzed are projected to be similar to or identical with those that would be involved in plutonium recycle; hence plutonium recycle operations would introduce no new accident types that have not previously been the subject of analysis.

Before discussing the behavior of packaging in accidents of unusual severity such as severe fires, consideration is given to problems arising from equipment failure incident to normal use or from human errors. These problems, which do not involve accidents in the ordinary sense, are identified and discussed below.

5.1 Special Problems

Even without accidents there is the possibility of environmental effects in transportation, arising from errors in preparing the packaging for shipment or from malfunctioning of packaging components in use. As noted in paragraph 2.0, regulatory requirements on packaging design, fabrication and use include provisions to guard against human errors or equipment malfunctions. These requirements include quality assurance programs applicable to packaging, review of designs by the NRC staff, and tests prior to first use of packages for fissile material and large quantities with respect to shielding and heat dissipation. There are also requirements that fissile and large quantity packages be tested before each use as to proper assembly, proper closing, presence of neutron absorbers, and proper temperature and pressure control performance.

5.1.1 Leakage Of Coolant

Of the radioactive materials packaging in current use in the fuel cycle, casks for irradiated fuel are the only type that generally require primary liquid coolant for heat dissipation. There is little likelihood of leakage of coolant from a cask, under other than extra severe accident conditions, because of the rugged, leaktight designs needed to meet packaging regulations, and the procedures the shipper is required to follow to ensure leaktightness when preparing the cask for shipment.

The consequences of a leak depend on the amount of radioactive material that could be released. A cask must be held at the point of origin until pressure, temperature, and leaktightness tests have been made. Any major leak would be discovered at the point of origin and corrected.

Information on seals and leak detection[2] indicates that leakage of liquid at a rate of 0.001 cc/sec or about 80 drops an hour is about the smallest that can be detected by visual observation of a container with an extensive seal length. It is expected that leakage at a rate exceeding 0.001 cc/sec would moisten a large enough area to be visible or would drip and could be detected and corrected at the reactor site. Leakage at a smaller rate from a large heated cask might be evaporated as rapidly as it is released. Some fraction (perhaps 1%) of the radioactivity in the released liquid might be dispersed in the form of an aerosol. The exposure to people from such releases would be extremely small because of the low levels of radioactivity permitted in the coolant and the very low leakage rate.

Thus, the NRC regulations limit the contamination level in the primary coolant under normal conditions to 10^{-7} curies/cc of Group I (plutonium), 5×10^{-6} curies/cc for Group II (strontium and mixed fission products), and 3×10^{-4} curies/cc for Groups III and IV radionuclides (cesium and uranium).[23] Based on 0.25% of the rods being perforated, it is estimated that there would be about 1 μCi/cc of gross fission product activity in the cask coolant. Experience reported by the Savannah River[24] processing plant indicates that the activity in water-filled casks ranges from 10^{-5} to 1 μCi/cc and that the activity is primarily cesium-137.

During a 5-day trip, an undetected cask leak of 0.001 cc/sec would release 430 cc or, at the gross fission product concentration derived above, about 400 μCi of activity. That contamination would be expected to be retained on the surface of the cask and bed of the truck or railroad car.

5.1.2 Improperly Closed Packages

It can, of course, be postulated that a package of almost any type could be shipped without proper closure. The frequency of such an occurrence would be expected to be very small for large irradiated fuel shipping casks, but might be significant for packages shipped in large numbers with multiple containers per shipment, such as packages of alpha wastes or plutonium. The NRC regulations governing transportation of radioactive materials (10 CFR Part 71) have been revised to require that plutonium in excess of 20 curies be shipped as a doubly contained solid after June 17, 1978. The two containers must withstand the normal and accident conditions specified in 10 CFR Part 71. In view of the solid form of the material and the double containment, there should be no release of plutonium oxide as a consequence of human error in preparing the package for shipment. It should also be noted that the two nested plutonium oxide containment vessels will be supported within a closed outer drum, which would also act as a barrier to the release of plutonium oxide under normal conditions. The probability of improper closure is further reduced by the requirements for quality assurance and package tests discussed earlier in this section.

Nevertheless, in the shipment of a large number of drums of solid wastes, it is possible, as a result of human error, that some of the drums may not be properly closed--about one in 10,000 packages according to estimates. Since, as described in paragraph 3.2.3, drums or other types of packages are usually shipped with a large protective overpack, the probability of a release from the sealed overpack to the environment is very low. If a package were to open within the overpack or railcar, the solid form of the waste material makes it probable that contamination of the inside of the outer container and of the other waste containers would be limited. No significant radiation exposures would be likely to result, since the car or overpack would be opened at the Federal waste repository under controlled conditions. Cleanup costs might amount to several hundreds of dollars.

5.2 Unirradiated Mixed Oxide Fuel Rods or Assemblies

The packagings for unirradiated fuel are designed to retain the fuel under accident conditions. Prototype packages of the two main types have performed satisfactorily when tested.[17-19] Future modifications would not be permitted to reduce this capability to withstand accidents.

Fuel rods are designed and constructed to withstand internal and external pressures of 1,000 to 2,000 pounds per square inch, anticipated in operation of the reactor. The fuel rod construction is such that there is little likelihood of release of the pellets or of smaller aggregates of mixed oxide. Fuel rods of this type have been tested by being dropped thirty feet onto concrete on end, on the side, and at an angle of 45°, without rupture of the cladding or loss of contents.[25]

The pelletized form of the mixed oxide and the fact that it is encapsulated helps to ensure retention of the radioactive material even under accident conditions. If some extremely severe accident were to cause the release of some pellets from a package, the material is expected to be in a recoverable form. A pellet may be crushed or shattered, but the particle size would most likely fall predominantly in the non-respirable (greater than 10 microns) range. Typically, ceramic materials of this type (pellets are fired at or above 1,650°C and are washed free of fines following grinding to size) do not form a significant fraction of fine particles when subject to simple crushing or fracturing. The particle size distribution would limit the area of contamination to the immediate vicinity of the ruptured package.

The packaging is also designed to prevent criticality under normal and severe accident conditions. An accident that could lead to accidental criticality would require release of several fuel elements from more than one package. After release from the packages, some of the fuel elements must be assembled in a close array and moderated, for example, by being submerged in water; accidental criticality in air is not possible. Considering the requirements for package design and the controls exercised during transport, the probability of such an accident is very small.

Therefore, the impact on the environment from radiation or radioactive material release in transportation accidents involving unirradiated mixed oxide fuel is considered to be negligible.

5.3 Irradiated Mixed Oxide Fuel Assemblies

Casks for irradiated reactor fuel must be so designed that in the event of an accident there would be no release of radioactive material or increase in radiation levels outside of the package, above levels specified in the regulations. As noted in the introduction to this section and in paragraph 1.1 of Appendix A, NRC reviews the details of cask designs and analyses to ensure that regulatory requirements are fulfilled. Fulfillment of the design requirements should ensure that the casks survive most accidents that are less severe than Category 4 (Table IV G-8) without any radiological consequences. The discussion which follows is therefore addressed primarily to postulated accident conditions equal to or more severe than Category 4, with correspondingly low probabilities of occurrence (Table IV G-7).

5.3.1 Coolant Leakage

The impact of possible coolant leakage was discussed in paragraph 5.1.1.

5.3.2 Accident Conditions

The packaging and relevant characteristics for irradiated fuel previously analyzed[2] for the uranium fuel cycle are similar to those involved in plutonium recycle. Thus, recycle introduces no new accident types not previously analyzed. In the unlikely event that a cask of irradiated mixed oxide fuel is involved in an extra severe accident, the environmental impact should be about the same as already analyzed.[2] The following is therefore limited to the applicable essentials of the existing analysis.

Each cask is so designed and constructed there is little probability of its being breached in an accident. The form of the nuclear fuel is such that, should a breach occur, releases of radioactivity are unlikely. Furthermore, those releases that might occur are likely to be limited to gases and liquid coolant present in the cavity. The uranium, actinides, and most of the fission products would remain in the oxide pellets. Some of the gases and most of the volatile and semivolatile actinides and fission products released from the oxide pellets would be retained by the cladding in the void spaces of the fuel rods.

The total amount of the important gases, actinides, and gross fission products in MOX blended PWR fuel, cooled 150 days after irradiation to 33,000 megawatt days per metric ton, are given in Tables IV G-10 and IV G-11. The important activities in the void spaces of the fuel rods are shown in Table IV G-12.[2] A direct comparison between Table IV G-12 and Table 7 of WASH-1238[2] shows very little difference between the MOX and UO_2 activity in fuel rod void space.

The amount of radioactivity released in an accident is related to the number of fuel rods that are perforated. Penetration of the cladding would release some of the gases and gross fission products from the rod void spaces into the cask cavity and coolant. In the absence of severe impact, it is believed conservative to assume that 0.25% of the fuel rods may be perforated. If ten percent of the rods were ruptured, the radioactivity released would probably not exceed 1.5×10^3 Ci of ^{85}Kr, 0.02 Ci of ^{131}I and 2.2×10^2 Ci of volatile and soluble fission products. Because of the cask design and quality control and the nature, form and physical properties of the fuel

Table IV G-10

RADIOACTIVITY OF IRRADIATED MOX FUEL (CURIES PER METRIC TON OF HEAVY METAL)

Material	Cooling Period (days)			
	90	150	365	3650
Fission Products	6.2×10^6	4.4×10^6	2.3×10^6	3.1×10^5
Actinides (Pu, Cm, Am, etc.)	3.9×10^5	3.6×10^5	3.1×10^5	1.9×10^5
Total	6.6×10^6	4.8×10^6	2.6×10^6	5.0×10^5

Table IV G-11

PREDOMINANT FISSION PRODUCTS IN GASEOUS FORM INCLUDED IN THE RADIOACTIVITY OF IRRADIATED MOX FUEL (CURIES PER METRIC TON OF HEAVY METAL)

Radionuclide	Cooling Period (days)			
	90	150	365	3650
Krypton-85	1.0×10^4	1.0×10^4	9.6×10^3	5.5×10^3
Xenon-131m	100	3	Negligible	Negligible
Iodine-131	390	2	Negligible	Negligible

Table IV G-12

IRRADIATED MOX FUEL ROD VOID SPACE ACTIVITY

Type of Radioactive Material	Total Inventory 150 Day Cooling Time (Ci/MT)	Fraction of Activity in Void Space of Fuel Rods* (Percent)	Activity in Void Spaces (Ci/MT)
^{85}Kr	1.0×10^4	30	3.0×10^3
^{131}I	2.0	2	4.0×10^{-2}
Other Fission Products	4.4×10^6	0.01**	440
Actinides (Pu, Am, Cm)	3.6×10^5	Essentially none	Negligible***
^{131}Xe	3.0	2	0.1***
^{129}I	4.0×10^{-2}	30	Negligible***
^{3}H	515	1	8***

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*Realistic gap activities in terms of percent of total inventory based on references 20 through 32, WASH-1238.[2]

**A conservative (high) value estimated on the basis of leaching the outer 1.2×10^{-5} inch from the surface of the mixed oxide fuel.

***Due to the small amounts present, the dose contribution from ^{131}Xe, ^{129}I, ^{3}H, and the actinides may be neglected, compared to the doses from the other radionuclides.

assemblies, the probability of such a release is very small. Other estimates of accidental releases of radionuclides are given elsewhere.[2,6,14,30] Essentially no observed data exist for this parameter. The analyses upon which the estimates are based all seem to be conservative in that conditions assumed for the cask, its fuel contents, its coolant contents, and the accident parameters seem to be more severe than what would actually be realized. The numbers given above represent upper bounds for these estimates.

The fission products that would be most readily released from the fuel elements are those sufficiently volatile. The elements remaining in the fuel after 150 days cooling time and which would be most likely to escape are, in order of decreasing volatility, Kr, I, Cs, and Te.[14,31] Elements which are volatile under highly oxidizing conditions include Ru and Tc, and elements of low volatility include Sr, Ba, and Sb. Conditions in which Ru and Tc could be released do not seem credible in transportation accidents.[30]

Some controversy has arisen about the estimated release of Cs. Based on experimental data, regulatory requirements, and postulated accident conditions, a serious accidental release of Cs is estimated to not exceed 45 Ci.[27,32]

5.3.3 Extended Fire

Involvement of a cask in a fire lasting as long as 4 or more hours could cause loss of some neutron shielding and, if lead is used, some reduction of gamma shielding might be expected. An extended fire, which is not essentially different from a design basis accident, would not be expected to have any significant effect on the contents due to the large mass of the typical cask and its slow response to external thermal disturbances.

Both wet and dry fuel cooling schemes are used, depending on the cask design, but all designs contain provisions to prevent massive containment loss due to overpressure in extended fires. According to regulations, only a limited amount of radioactivity is permitted to escape in the form of released gases or contaminated coolant--exceeding neither 0.1 percent of the total radioactivity of the package contents nor specified activities for each of the transport groups of radionuclides, for example 0.01 curie of Group I, which includes the most radiotoxic actinides such as plutonium and americium (10 CFR Part 71.36). The environmental effects of much larger releases, such as those discussed in the previous paragraph, have been discussed in Reference 2 and have been found small.

5.3.4 Submersion In Water

If a cask is accidentally dropped into water during transport, it is unlikely to be adversely affected by hydrostatic pressure unless the water is much deeper than that of typical U.S. inland waterways. Most fuel is loaded into casks under water, and thus immersion would have no immediate effects. As the water would remove the heat, overheating would not occur. Each cask is required by 10 CFR Part 71.32(b), to be designed to withstand an external pressure equal to the water pressure at a depth of approximately 57 feet, and most designs will withstand a much greater external pressure. If a cask were to collapse due to excessive pressure in deep water, only the small amount of radioactivity in the cask coolant and gases from perforated elements in the cask cavity are likely to be released. The direct radiation would be shielded by the water. About 10 meters of water--the depth of most storage pools--would be ample shielding for radiation from exposed fuel elements.

Based on structural integrity requirements of the regulations, sinking of a cask in deep water would not result in serious radiological consequences. The most likely mechanism for loss of containment from external water pressure would be through failure of the pressure relief valves. This could result in an inflow of water and subsequent release of some of the contaminated coolant and most of the radioactive gases present in the cask cavity. In the absence of a severe impact, the total activity released might be on the order of 30 curies, most of which would be krypton-85 gas. The vast quantities of water available at the depth at which such a failure might occur would provide sufficient dilution so that it is unlikely there would be any radiation exposure or significant environmental impact.

The fuel elements, which contain most of the radioactive material, provide excellent containment. In an operating reactor, the fuel elements are under water at elevated temperatures and pressures on the order of 1,000 to 2,000 pounds per square inch.[28] Thus exposure to water pressures at depths of 600 to 1,200 meters (corresponding to hydrostatic pressures of 850 to 1,700 pounds per square inch) should have no substantial effect on the fuel elements themselves.

Except under very unusual circumstances in which the cask could not be located or was submerged in extreme depths, the cask probably could be recovered with normal salvage equipment. If the cask and elements were not recovered, there would be a gradual release of radioactive material over a long period of time, probably in the range of several hundred years. Considering the extremely low probability of occurrence, the substantial reduction in radioactivity due to radioactive decay, and the dilution that would be available, there would be no significant environmental impact from this gradual leaching and diffusion of the radioactive material.

5.3.5 Radiological Consequences

The radiological dose absorbed by the surrounding population from transportation accidents involving spent fuel shipments has been estimated in several studies.[2,6,14,26,30] The upper bound of the dose to a typical member of an emergency crew responding to an accident is considered to be represented by the dose field at a point 50 meters downwind from the cargo, which is considered as a point source at ground level emitting radioactive particles and radiation. The radioactive plume is described by a Gaussian distribution of radioactivity in space. Assuming the releases given in paragraph 5.3.2 for Kr and I and one percent of the release for volatile fission products are dispersed in an aerosol, corresponding dose conversion coefficients as given in Table IV G-13, and probability weighted weather conditions,[2] the external whole body dose to an individual adult with no shielding is estimated to be about 1.4 rem. The estimated internal doses are about 0.1 rem to the thyroid (adult), about 4 rem to the lungs, and about 0.6 rem to the gastrointestinal tract. These internal doses are conservative estimates in that no protective measures are assumed to be employed.

TABLE IV G-13

DOSE CONVERSION COEFFICIENTS

Radioactive Material Released	Dose	K $(\frac{\text{rem/sec}}{\text{Ci/m}^3})$
^{85}Kr	Skin - due to submersion in the radioactive cloud	0.053*
^{131}I	Thyroid - due to inhalation	
	Adult	320*
	Child	480*
Gross Fission Products	Whole Body - due to inhalation and immersion	0.12**
	Lung - due to inhalation	110*
	Gastrointestinal Tract - due to inhalation	16**

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*Reference 2.

**Reference 14.

The exposed population is assumed to be uniformly distributed in the area bounded by the isopleth corresponding to a concentration of radioactivity of 10^{-6} (Ci/m^3) (Ci released/sec). This area is roughly equivalent to that of a sector of a circle with angle of 22.5 degrees and extending from a radius of 50 meters to a radius of about 20 miles. If an accident occurs on a route east of the Mississippi River, the average population density is assumed to be 300 people/mi^2, yielding an estimated external whole body population dose of about 0.5 person-rem. If the accident occurs on a route west of the Mississippi River, the average population density is assumed to be 100 people/mi^2, yielding an estimated external whole body population dose of about 0.2 person-rem. This estimate is also applicable to much larger population densities. Thus for a population density of 10,000 people/mi^2, the estimated external whole body population dose is 20 person-rem. The estimated internal population doses are approximately given by 0.0001P for the adult thyroid, 0.0002P for the child thyroid, 0.005P for the lungs, and 0.0007P for the gastrointestinal tract, where P is the population density in people per square mile.

5.3.6 Conclusion

Considering the low probability that a shipment of irradiated fuel would be involved in an accident, the requirements for package design and quality assurance, the nature and form of the irradiated fuel, and the controls exercised over the shipment during transport, it is concluded that the radiation risk to the environment from irradiated fuel in transportation accidents is small.

5.4 Solid Low Level Alpha Waste

The likelihood of leakage of radioactive material from a package of solid waste is small because of the solid form of the material and the physical protection afforded by the protective overpack in which the waste drums are shipped, e.g., the ATMX railcar or the "Super Tiger" type container.

5.4.1 Improperly Closed Packages

The possible impact of improperly closed packages of alpha waste was discussed in paragraph 5.1.2.

5.4.2 Accident Conditions

Packages used for waste are so designed and constructed, and the solid form in which the waste is shipped is such that, in the event a shipment of solid waste is involved in an accident, it is unlikely that the radioactive material would be released. Both the ATMX railcar[12] and the Super Tiger[29] have been designed to withstand the accident test series specified in 10 CFR Part 71. Very little offgassing would result even in an extended fire, because processes of waste volume reduction will include incineration of combustibles, concretion of liquid wastes, and compaction favoring formation of dense, unreactive solids.

5.4.3 Conclusion

Because of the package design, quality assurance, and nature and form of the waste, a release is unlikely in an accident. But even if a release occurs, the form of the material, as noted in the foregoing paragraph, makes cleanup possible without any serious radiation exposures.

Therefore, the radiation risk to the environment from solid radioactive alpha waste in extra severe transportation accidents is small (no expected injuries or loss of life); however, cleanup following an accident exceeding the design capability of the package could be expensive.

5.5 Plutonium Oxide

NRC regulations governing transportation of radioactive materials (§71.42, 10 CFR Part 71) have been revised to require that plutonium in excess of 20 curies per package be shipped as a solid after June 17, 1978. In addition, plutonium in excess of 20 curies will be packaged in an inner and an outer container which meet the normal and accident conditions specified in 10 CFR 71. Reactor fuel elements, metal or metal alloy, and other plutonium-bearing solids that the Commission determines suitable are exempt. This study assumes that shipments of plutonium from the reprocessing plant will be in the oxide form meeting these revised criteria.

5.5.1 Improperly Closed Packages

The possible impact of improperly closed packages of plutonium oxide was discussed in paragraph 5.1.2.

5.5.2 Accident Conditions

Since the package will be doubly sealed, and both the inner and outer containment vessels will meet the accident conditions specified in 10 CFR Part 71, the probability that there will be any release of radioactive material from the package following any credible accident is not considered significant. A recent study indicated that a release from such a container would occur no more often than once in 220 transportation accidents.[4] The risk to the public from such release was estimated in this study to be significantly smaller than the risk of being struck by falling meteorites.

5.5.3 Conclusion

No release of radioactive material is expected from normal or accident conditions of transport. The accident conditions could result in slight increases in radiation levels from the package, but the frequency of such an occurrence should be so low as to have no significant effect on the exposure levels of transport workers or the general public (see Table IV G-7).

5.6 High Level Waste

As noted in paragraph 4.4.3.4, the structural and containment features of casks for transporting high level wastes will be similar to those of casks for irradiated fuel. Furthermore, high level wastes will be packaged in completely sealed steel canisters that are in turn enclosed in the shipping cask so that two levels of containment will be provided. The conclusions with respect to the ability of irradiated fuel casks to withstand unusual accidents, discussed in paragraph 5.3.2, will thus be applicable to the high level waste casks.

It should be noted that plutonium recycle would not have a significant effect on those characteristics of high level waste which are important to possible environmental impact under unusual accident conditions. Trained personnel equipped to monitor the area and competent to act as advisers are available through an intergovernmental radiological assistance program. Radiological Emergency Assistance Teams are dispatched in response to calls for emergency assistance.

6.0 ANALYSIS OF POSSIBLE ALTERNATIVES AND ADDITIONS TO TRANSPORTATION METHODS

Under normal conditions it has been shown the effects to the environment due to transportation are small: no significant change in traffic density or heat loads;

radiological impact is also not significant; transportation workers are exposed to 20%-30% of natural background radiation and general public 0.05%-0.08% of natural background radiation. Although the probability of accidents is small, the potential consequences of credible accidents are serious; therefore alternatives and actions are continually being considered as noted below.

6.1 Routing

The probable routing of shipments of unirradiated and irradiated nuclear fuel and solid radwastes is indicated in some environmental reports for individual nuclear power plants. It is not intended that the shipments be restricted to these routes, since the safety standards of NRC and DOT do not rely on restriction of routing to assure safety in transport.

The regulations of the States impose controls on weights of loads on roadways and bridges. In some cases municipalities and bridge, tunnel, and turnpike authorities place restrictions or exclusions on travel at specific periods of the day or night and over certain sections of routes. These restrictions may affect the choice of routes.

Requiring shipment of radioactive material over routes that avoid centers of population would reduce the radiological consequences of accidents in which a release of radioactivity or direct radiation exposure of persons in the area is involved. The dose would be smaller by restriction of the number of people in the affected area. The risk from accidents, however, involves both frequency and consequences. If the number of miles traveled is increased by the special routing restriction, the frequency of accidents will be increased unless the probability of an accident is smaller for the "special route," since the number of accidents is assumed proportional to the number of miles traveled.

At present, truckers carrying hazardous goods are required by DOT to avoid congested places so far as practicable. Truck routes usually are chosen to expedite movement, and for that reason usually avoid congested areas. Carriers use interstate highways whenever possible, and interstate highways avoid centers of population in most cases. Although the use of divided highways and routes around population centers may reduce the probability of an accident, the severity of those accidents which do occur could be increased because of the generally higher speed of the vehicle.

There are no specific regulatory requirements with regard to routing of hazardous material shipments by rail. Extra severe rail accidents usually involve high speeds and frequently occur because of faulty roadbeds or equipment. Roadbeds connecting centers of population are used more frequently than off-the-main-line roadbeds and generally are better maintained for that reason. Furthermore, accidents occurring inside city limits are unlikely to be as severe as those outside the city limits since speeds are restricted somewhat and emergency equipment is more readily available. For these reasons, it appears that for rail shipment the frequency of severe accidents may be greater for shipments made on routes chosen to avoid centers of population than on "main line routes" between population centers.

6.2 Escorts for Accident Emergencies

Escorts, in separate vehicles or cars, could be required to accompany shipments for accident emergency purposes. They could be equipped to monitor the area and take corrective action in case of an accident. Escorts could assist in control of any accident but probably could not reduce the effects of immediate releases, such as releases of noble gases and iodine. It does not appear likely that a requirement that escorts accompany a shipment for accident emergency purposes can be justified in view of the low probability of a severe accident in which an escort would be effective.

To be effective, escorts would have to be provided for each major shipment of radioactive material. Although an escort in a separate vehicle might mitigate the consequences of some accidents and reduce the already small probability of the shipment vehicle being involved in an accident, the escort vehicle itself has an accident liability at least equal to that of the shipment vehicle. Because injuries occur in 13% of all motor vehicle accidents and because less than 0.5% of all motor vehicle accidents are extra severe, the increased risk of injury due to escort vehicles appears to outweigh the small reduction in risk of consequences from extra severe shipment vehicle accidents.

6.3 Longer Storage of Spent Fuel

The amount of radioactivity and decay heat in the irradiated fuel can be reduced by holding the irradiated fuel in the storage pool at the reactor for long periods of time.

For purposes of heat removal during shipment, the radioactive decay that takes place in irradiated fuel during the first 90 days after removal from the reactor is considered important. During that time, most of the ^{131}I decays to low values, the short half-life noble gases are significantly reduced and other short-lived radionuclides decay so that the overall amount of heat generated is greatly reduced. The difference in radioactivity inventory and decay heat between 90 days and 150 days is not considered to be significant for shipment. Therefore, shipment any time after 90 days of cooling time is within the scope of this analysis. Shipment after less than 90 days cooling time would require reexamination and balancing of the added risk and potential benefit.

By storing the fuel for a full year instead of 150 days, the radioactivity and decay heat could be reduced by a factor of 2, and storage for 10 years would result in a further reduction by a factor of 10. Storage beyond 150 days gains little in terms of reducing the inventory compared to the required increase in storage capacity for the nuclear power plant, fuel inventory costs, and additional precautions necessary to assure that the risk is not greater because of the extra fuel on hand. On balance, it does not appear that storage beyond 150 days is warranted.

6.4 Lower Radiation Levels Outside Packages

It is possible to design and build heavier packaging with additional shielding or--by reducing the amount of radioactive material in a package--to reduce the radiation levels outside the package. For most container designs, additional shielding would be added to the outside of the present shielding to avoid reducing the capacity of the container. It is concluded that the fractional increase in the weight of the container due to the added shielding might be more than the fractional increase in shielding thickness, thus both shipping and container costs would be unfavorably affected. Shipping costs increase as the ratio of container weight to contents weight. Additional shielding also increases the initial cost of the container.

The weight of casks as now designed is approaching the limits of the available handling and transport facilities. Extra package weight means a smaller number of packages per vehicle, which would mean more shipments. More shipments would be required if the content of present packages were reduced. Increasing the number of shipments increases the frequency of accidents and thereby increases the impact on the environment.

Taking into account the costs associated with additional shielding, weight limitations of available facilities and equipment, and the present state of the technology, it is concluded that the radiation levels associated with present designs of casks are as low as practicable.

6.5 More Stringent Accident Damage Test Criteria

Radiological risk due to accidents involving packages of radioactive material might be reduced by imposing more stringent accident damage test criteria on package designs.

Experience and estimated probabilities and consequences of accidents indicate the radiological risk in transport accidents that result from packages which meet the present accident damage test criteria is small.[2] Increasing the severity of the test conditions would probably require heavier or larger packaging to meet the criteria. Extra weight of packaging would reduce the ratio of radioactive contents to package weight. Larger and heavier packages, in most case, would mean a smaller number of packages per vehicle. The reduced ratio and fewer packages per vehicle would increase the number of shipments required to be shipped. Increasing the number of shipments would increase the number of accidents involving such shipments.

Because the radiological risk is so small, imposing more stringent test criteria can achieve only a relatively small reduction in that risk. An increase in the number of accidents in which shipments of radioactive materials are involved tends to offset that advantage. The overall risk from both radiological and common (i.e., nonradiological) causes is proportional to the number of accidents. The risk from common causes, although small,[2] is greater per accident than is the risk from radiological causes.

Changes in the accident damage test criteria for radiological safety do not appear to be warranted in view of the small radiological risk as evaluated in this report. Considering the small overall risk in accidents and the present balance of radiological vs common cause risks, it is concluded that the present accident damage test criteria provide control over the radiological risk to a level that is as low as practicable.

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CHAPTER IV
Section G
APPENDIX A
REGULATORY STANDARDS AND REQUIREMENTS

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1.1 Packaging Standards and Requirements

The packaging standards and criteria are found in the regulations of NRC (10 CFR Part 71) and regulations of DOT (49 CFR §§ 170 through 179).

Present criteria provide assurance that packaging designed to meet such standards can be carried on all modes of transport considered here and will withstand conditions likely to be encountered in normal and accident conditions. As written, the criteria specify tests of packaging which can be carried out either analytically or experimentally. These criteria, which were first published by the International Atomic Energy Agency in 1964,[1] have been adopted in many international and national transportation regulations and serve as the basis for the regulatory standards and criteria of the United States. They were based on a detailed analysis of normal and accident conditions in transport and nearly 20 years of experience in shipping many types of radioactive materials.

Packaging must function in the normal transportation environment with a high degree of reliability. Systems selected to achieve the basic design functions--containment, shielding, heat dissipation, and nuclear criticality safety--must provide a high degree of inherent safety under normal conditions and have a high tolerance for malfunctions, off-normal conditions, and accidents. Each shipping container is required to be checked routinely to assure that the "as built" high quality is maintained throughout its lifetime.[2]

The type of packaging is specified in DOT regulations, 49 CFR 173, according to the type and quantity of radioactive material. See Table IV G(A)-1.

The forms of radioactive materials are divided into two broad classes: (1) "special form," which is a massive, nonfriable, solid material or material confined in a high integrity capsule of inert material (not generally involved in the fuel cycle) and (2) "normal form," which applies to all radioactive materials (covers most of the materials used in the fuel cycle) which are not "special form." Normal form radioactive materials are subdivided into seven groups of radionuclides based primarily on radiotoxicity. Package limits for the seven transport groups and "special form" are shown in Table IV G(A)-1.

Exempt 10 CFR Part 71.7 and type A 10 CFR Part 71.4(g) quantities of radioactive materials are not normally found in the fuel cycle.

TABLE IV G(A)-1 QUANTITY LIMITS AS RELATED TO PACKAGE REQUIREMENTS

TRANSPORT GROUP	EXAMPLES	EXEMPT QUANTITY (CURIES)	TYPE A PACKAGE (CURIES)	TYPE B* PACKAGE (CURIES)
I	^{239}Pu, ^{242}Cm, ^{252}Cf	10^{-5}	0.001	20
II	^{210}Bi, ^{90}Sr, ^{210}Po	10^{-4}	0.05	20
III	^{235}U, ^{238}U, ^{131}I	10^{-3}	3	200
IV	^{76}As, ^{14}C, ^{45}Ca	10^{-3}	20	200
V	NOBLE GASSES, ^{85}Kr	10^{-3}	20	5,000
VI	^{37}Ar, ^{133}Xe, ^{85}Kr UNCOMPRESSED	10^{-3}	1,000	50,000
VII	TRITIUM – AS A GAS OR IN LUMINOUS PAINT	25	1,000	50,000
SPECIAL FORM	^{60}Co RADIOGRAPHY SOURCE, Pu-Be NEUTRON SOURCE	10^{-3}	20**	5,000

*A Large Quantity is defined as any quantity in excess of a type B quantity.

**Except that for ^{252}Cf, the limit is 2 Ci

Standards for evaluation and testing of adequacy with respect to normal conditions specified in NRC and DOT regulations include exposure to temperatures ranging from -40°F to 130°F; wetting of all surface except the bottom for 30 minutes, dropping of package through a 4-foot free fall while wet; exposure to vibration as normally encountered in transport and to external pressure of 0.5 atmosphere.

Quantities exceeding type A quantities must be shipped in type B packaging. Type B packaging must be designed to withstand normal transport conditions without loss of contents or shielding efficiency and to suffer not more than a specified loss of contents or shielding efficiency if subjected to a specified sequence of accident damage test conditions. That damage test sequence includes (1) a free fall from a height of 30 feet onto a flat, essentially unyielding surface, with the package striking in the orientation which does the most damage; (2) a free fall from a height of 40 inches onto a 6-inch-diameter steel plunger that is long enough, and with the package in the orientation, to do maximum damage; (3) heat input from exposure for 30 minutes to a fire or other radiant environment having a temperature of 1,475°F and an emissivity of 0.9; and (4) for fissile material, immersion in water to a depth of 3 feet for 8 hours. These test conditions make up the design basis accident for Type B packages. Package designs which meet the criteria under these test conditions are considered to provide adequate protection to the public and to operating personnel in transportation accidents.

Large quantities (most of the material involved in the fuel cycle) must be shipped in type B packaging which provides for adequate dissipation of heat. In addition, there must be no loss of contents at an external pressure of 25 psig, which is approximately equivalent to immersion in water to a depth of 57 feet.

With respect to heat dissipation, the regulations require the package to be designed so that the temperature rise due to decay heat will not adversely affect the package or the contents and will not cause excessive pressure. The accessible surface of the package must not exceed a temperature of 180°F.

1.2 Nuclear Criticality Safety

Fissile material (uranium-233, uranium-235 and some plutonium isotopes) in quantities exceeding 15 grams per package or--in homogeneous hydrogenous solutions and mixtures--quantities exceeding 500 grams of ^{233}U or Pu ($H/^{233}U$ or Pu $< 7,600$) or 800 grams of ^{235}U ($H/^{235}U < 5,200$) per package requires some control in transport to assure safety from accidental criticality. This is provided by assuring that the contents of each package of fissile material are subcritical when delivered to a carrier and that the package is so designed that it will remain subcritical under all conditions likely to be encountered in transport, including accidents. In addition, the contents must be limited or the package must be designed so that the number of packages which are likely to be accumulated in one vehicle area will be subcritical under all conditions likely to be encountered in transport, including accidents and handling errors.

The NRC regulations specify the conditions for evaluating the adequacy of design of a package for fissile material, including form and geometry of the contents and moderation and reflection.

The package design must be evaluated against the accident damage test conditions discussed for type B packages. See paragraph 1.1.

A package for fissile material must be so designed and constructed and its contents so limited that the numbers of such packages as listed in Table IV G(A)-2 can be shown to be subcritical in a moderated and reflected array according to the Fissile Class (I, II, or III) to which the package is assigned (10 CFR Part 71.4(d)).

TABLE IV G(A)-2

CRITICALITY LIMITATIONS ON PACKAGES CONTAINING FISSILE MATERIAL

Fissile Class	Criticality of Number of Packages Under Normal Conditions	Criticality of All Packages as Damaged in Accident Conditions
I	Any number	250 packages
II	5 times the allowable number*	2 times the allowable number*
III	2 times the allowable number*	The allowable number*

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*The allowable number is the number of the same type of package to be allowed in one shipment.

The controls for transport vary according to the Fissile Class. Fissile Class II packages are controlled by the carrier as to allowable number on a vehicle or in one handling or storage area. This is done by the simple system of assigning a number called a transport index to each package and requiring the carrier not to allow the accumulation of more than 50 transport indexes on a vehicle or area. This system has been applied to limiting the accumulated radiation level since the year 1948.

For Fissile Class III, the shipment must be made in exclusive-use vehicles (the consignor loads the shipment and the consignee unloads the shipment) or with an escort provided by the shipper to assure that the shipment is kept separated from other fissile material, or by some other procedure specifically approved by DOT.

Fissile Class I packages do not require limitations on the number of packages in an area or vehicle for nuclear criticality safety.

In some cases physical properties limit the number of packages in a shipment. For example, one irradiated fuel cask is shipped on a truck or railcar and the irradiated fuel cask is shipped in an exclusive-use vehicle because of weight limitations on the vehicle even though some designs might meet Fissile Class I requirements. For unirradiated nuclear fuel, the allowable number of packages for Fissile Class II in the case of one design of PWR package is 20. However, because of the size and weight of each package, only six can be loaded on one truck.

1.3 Packaging Design Review

NRC reviews and issues approvals for designs of commercial packages for shipping type B, large quantities, and for fissile materials. DOT regulations authorize the use of type B, large quantities, and fissile materials packages based on NRC or Energy Research and Development Administration evaluations of the package designs. Based on these evaluations, DOT issues approvals for packages to be used for import or export except for shipments which conform to the regulations of the Canadian Transport Commission which require no further approval.[3]

Applicants requesting approval of a packaging design must provide a detailed analysis of that design to demonstrate that it meets the packaging standards and criteria. The demonstration may be made by quantitative assessment, tests of models of packaging details or mock-ups representing the methods of construction used, extrapolation from test results for similar designs or designs employing similar construction features, actual tests of samples of packaging made to the design, or other evidence.

1.4 Quality Assurance and Control

The possibility that a package will be constructed or used in a manner not in accordance with the design, is minimized through the regulatory requirements for quality assurance and various observations and tests before each shipment.

Under the DOT regulations, each fabricator of specification containers must register with and is subject to inspection by DOT. The regulations specify certain tests that must be carried out on such containers.

NRC requires licensees who wish to fabricate casks to describe their quality assurance program at the time they apply for approval of the design. In addition, NRC requires that packages for fissile material and large quantities be tested prior to first use with respect to shielding and heat dissipation and prior to each use as to proper assembly, proper closing, ability to maintain proper temperature and pressure, and presence of neutron absorbers, if required.

1.5 Radiation Level Limitations

External radiation exposure of transport workers and the general public in the transportation of packages of radioactive material is controlled during transport by several different methods.

Exposure from radiation emitted by individual packages of radioactive material is limited by DOT regulations in two ways: a surface radiation limit of no more than 200 mrem/hr to limit direct exposure to the person handling the package and a limit on radiation at 3 feet from the surface of the package of no more than 10 mrem/hr in order to limit the radiation level to which persons and property in the vicinity would be exposed.

If a package is shipped in a closed truck or railcar under the "exclusive use" condition (which means it is loaded by the consignor with no other material and unloaded by the consignee), the radiation limits are as follows: the radiation level at 3 feet from the surface of the package is limited to 1,000 mrem/hr provided the radiation level does not exceed 200 mrem/hr at the surface of the vehicle, 10 mrem/hr at 6 feet from the outside surfaces of the vehicle, and 2 mrem/hr in either the driver's compartment or other normally occupied positions in the truck or railcar.

As a simple indicator of the radiation dose rate from an individual package, DOT regulations define one "transport index" (TI) as equal to 1 mrem/hr at 3 feet from the surface of the package. The regulations specify limits for aggregations of packages in terms of the sum of transport indexes. The number of packages stored or handled in one area or loaded on one car or vehicle must be so limited that the sum of their transport indexes does not exceed 50. This prevents a large aggregation of packages, each with a significant radiation level, from producing a much higher radiation level than desirable.

Simple tables of minimum separation distances from people and unexposed film are specified for packages of radioactive materials in storage and on vehicles, in terms of the sum of the transport indexes in each group of packages.

Whether there is one package or a large number of packages in a vehicle or a location, the transport worker or carrier is required to read each TI, add the total number of TI's present, determine from the tables in the regulations the distance those packages must be kept from film and continuously occupied areas, and assure that those separation distances are provided.

The transport index system has also been adapted for limiting aggregations of packages containing fissile radioactive materials to assure nuclear criticality safety. The shipper determines--in accordance with specific criteria laid down in NRC regulations--a transport index figure which is to be assigned to the fissile material package. For shipping, the shipper assigns to each package of fissile material the nuclear safety TI as calculated, or the radiation level TI, as described earlier, whichever is the higher. The transport worker, as is the case for radiation levels, adds the TI's in any one vehicle or location, and thus keeps the amount of fissile material in all types of packages within safe limits. The TI assigned to individual packages of fissile material for nuclear safety reasons takes into account that--in cases other than exclusive-use shipments--2 to 5 times the permitted total TI in a collection of packages may be safely accumulated.

It should be noted that mixing nuclear-safety TI with radiation-level TI in the course of transport increases the margin of safety for both since they are not synergistic.

1.6 Surface Contamination Levels

DOT regulations require that there be no significant removable surface contamination on the external accessible surfaces of packages when they are shipped. Levels of removable contamination on the surfaces are determined by a "wipe test." The regulations consider the level is "not significant" if the activity on the "wipe" does not exceed 10^{-11} Ci/cm^2 for beta-gamma emitters and 10^{-12} Ci/cm^2 for alpha emitters.[4] Any fixed contamination of the surface is limited by the external radiation level limitations discussed in the previous paragraphs.

1.7 External Temperature

DOT regulations limit the temperature at any accessible surface of the cask to not more than 122°F at any time during transport, except that for full-load or exclusive-use shipments the temperature may be 180°F.

1.8 Warning Labels

Each package of radioactive material is required by DOT regulations to be provided on two opposite sides with a distinctive warning label. Each of three label formats bears the unique trefoil radiation symbol and alerts personnel that the package may require special handling. If the background color of the label is all white, the radiation is minimal and nothing special is required. If, however, the background of the upper half of the label is yellow, a radiation level requiring consideration may exist at the outside of the package, and an indication of what controls must be exercised is related to the transport index concept discussed above. Fissile Class III, special permit, and large quantity packages must bear a yellow label with three stripes, and the rail or highway vehicle in which it is carried must be placarded.

1.9 Placards

A truck or rail car carrying any package labeled with a radioactive yellow-III label must be placarded on the outside. The placard for rail cars bears the distinctive trefoil symbol and, for trucks, the word RADIOACTIVE in letters large enough to catch the eye. This is to advise freight handlers of the presence of radioactive material inside the vehicle, or to indicate the presence of special types of shipments (a Fissile Class III package, a special permit package, or a large source package); and to warn passers-by and emergency crews that radioactive material shipments are in the vehicle. Marking or placarding is intended to warn against remaining in the vicinity of the vehicle unnecessarily, thus reducing exposures. Also, the placard will alert emergency crews to the need for appropriate precautions in case such vehicles are involved in accidents. Cars and trucks carrying carload or truckload lots of radioactive materials, packages with significant external radiation levels or containing large quantities of radioactive material or Fissile Class III shipments are required to be marked with a "RADIOACTIVE" placard.

1.10 Shipper's Certification

Before delivering a package to a carrier for transport, the shipper must determine that there is no "significant" loose radioactive contamination on the outside of the package, that the radiation levels on the surface of the package and at 3 feet from the package meet the specified regulatory levels, and that marking and labeling are in compliance with requirements. The shipper must also certify in writing on the shipping papers that the radioactive materials are properly classified, described, packaged, marked, and labeled and are in proper condition for transport according to the applicable DOT regulations.

1.11 Weight and Traffic Density

State highway weight restrictions limit the gross weight of trucks for routine shipments so that the gross weight of a cask is limited to about 25 to 30 tons. Shipments of casks within this range may be allowed in most States under a special overweight permit.[5] Since repetitive shipments of overweight loads may cause breakup of the roadway, States often prescribe routing and in some cases restrict the period during which the truck can travel. Some irradiated fuel shipping casks may require overweight permits.

Rail shipments of 50 to 100 tons of other commodities, such as coal, are routinely handled, so rail shipments of casks of comparable weights would offer no unusual loading for rail facilities.

With respect to traffic density, the average number of round trip truck shipments of nuclear fuel, solid radwaste, and empty packagings is estimated to be about 200 per year for a typical reactor and involves a total of about 155,000 truck miles (assuming all shipments by truck for comparative purposes). The number of shipments and miles travelled are small compared to the present traffic densities and miles travelled by trucks for all purposes (in excess of 55 billion miles per year). In the event that some shipments are made by rail (because of the larger load capabilities of railcars), the total number of shipments would decrease.

As an indication of the traffic flow, an average of 43,500 motor vehicles per day traveled over one section of Interstate 5 between San Diego and Los Angeles in the year 1971. According to the Federal Highway Administration, the average number of trucks per day on any given section of U.S. highway generally varies from about 100 to 10,000. The total number of combination truck (semi-trailers) miles traveled in the year 1974 is estimated to be over 55 billion.

1.12 Changing the Standards and Requirements

Safety of radioactive material transport is assured not only through design standards for packaging, but also by quality assurance programs to assure conformance with approved designs, to correct problems and to help assure continuing satisfactory performance over the lifetime of the package. Despite the best possible design practices, assurance of reliable and predictable operations of the packaging and the transportation equipment, measures to reduce the already low probability of accidents, and provisions to mitigate the consequences of accidents which may occur, radiological incidents will occur. Such events must be reported and will be investigated. If evidence becomes available that accepted guidelines are being exceeded or the public is being unduly exposed or its health and safety impaired, action can and will be taken in a timely manner. The regulatory requirements, codes, standards, specifications and criteria applicable to the designs of packages, loading patterns, protective measures, and quality assurance practices for transportation of radioactive material can be modified, should the need for changes become evident.

The probability of leakage due to human error can be reduced by increased control over the preparation of packages for shipment. Two actions are already in progress: the DOT is amending its regulations to require that shippers carry out certain examinations and test procedures on packages prior to shipment[6] and the NRC is considering expanding its quality assurance requirements applicable to packages used by its licensee-shippers.[7]

1.13 Capacity for Coping with Accidental Releases

The consequences of an accident involving radioactive material are mitigated by the procedures which carriers are required to follow. These procedures include segregation of packages and materials from persons; immediate notification of the shipper and DOT in case of an accident, fire, or leaking package; and a requirement that vehicles, cars, building areas, and equipment involved in any release of radioactivity not be placed in service again until surveyed and, where necessary, decontaminated.

Trained personnel, equipped to monitor the area and competent to act as advisers, are available through an Intergovernmental Radiological Assistance Program.[8] Radiological Emergency Assistance Teams are dispatched in response to calls for emergency assistance. This assistance has been made available in the few transportation accidents involving radioactive materials shipments that have occurred in recent years.[9] Should a major release occur, this assistance could help reduce the impact of the release.

1.14 Cleaning up After an Accident

In a transportation accident involving radioactive material, DOT, NRC and the shipper would be informed immediately and would be available to advise on the need for and extent of cleanup activities. Also the Radiological Emergency Assistance Teams provided as part of the Intergovernmental Radiological Assistance Program[10] are made available on request to assist in advising as to methods of cleanup and the extent to which cleanup may be necessary. In addition, various States have formed groups and programs to address post-accident actions: Southern Interstate Nuclear Board, Western Interstate Nuclear Board, and others.

Decontamination of the accident site, if required, would be the functional responsiblility of the carrier; control during cleanup and setting of acceptable levels of decontamination for release would be the responsibility of the regulatory authorities involved (State agencies, NRC, EPA, DOT). The DOT regulation given in 49 CFR § 175.655(j)(3), requires the equipment and facilities not be placed in use again until decontaminated to a level not to exceed 0.5 mrem/hr and no significant removable contamination.

In general, the area would be decontaminated to the lowest practicable levels. If, for example, the radioactive material were in solid, nondispersible form such as fuel pellets, it could be recovered easily with little or no decontamination required. However, if the material were liquid or powder, decontamination would probably include removal of some soil or, if released on a concrete surface, removal of some of the surface. Monitoring of the decontaminated surfaces with appropriate instruments would likely be necessary to determine the effectiveness of the cleanup measures.

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GESMO

CHAPTER IV

ENVIRONMENTAL IMPACT DUE TO THE IMPLEMENTATION OF PLUTONIUM RECYCLE

SECTION H

RADIOACTIVE WASTE MANAGEMENT

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SUMMARY

General

This section assesses the effects from radioactive waste management for three alternative modes of operating the light water reactor (LWR) fuel cycle: no recycle, recycle of uranium only, and recycle of both uranium and plutonium.* Waste management activities are assessed from the point of view of determining the relative effects of these three options in terms of the environmental impact. Radioactive wastes generated in all segments of the nuclear fuel cycle during the years 1975 through 2000 are considered in the assessment.

The protection of Man and his environment from the radioactive wastes generated by the LWR industry requires the planning and implementation of a waste management program. A complete discussion of the current Energy Research and Development Administration (ERDA) waste management program and the underlying technical information is beyond the scope of the present study. The objective of this study is to assess the environmental impacts associated with recycle of plutonium in LWR's. Work is underway in ERDA to develop methods for isolating or containing the radioactive wastes generated in the nuclear fuel cycle to ensure that possible risks to Man's environment are maintained at acceptable levels. Present plans are to dispose of high level wastes, plutonium-bearing wastes, and transuranium-contaminated wastes in geologic formations of predictable stability such as salt beds or shale. Nontransuranium wastes would continue to be buried at suitable surface sites.

The Industry Today

The existing waste management facilities supporting the LWR industry include mine and mill tailings at retired facilities, six commercially operated land burial sites and two reprocessing plants not currently in operation at which a portion of the current LWR spent fuel is being stored. Some liquid high level waste is retained at one of the reprocessing sites. Spent fuel is also retained at the reactor sites. It is expected that commercial handling of high level and transuranium wastes will be based on ERDA (AEC) programs. A brief review of ERDA's ongoing programs in the area of waste management indicates the type of waste disposal facilities expected to exist in the future.

ERDA Waste Management Programs

In support of closing the nuclear fuel cycle, ERDA has initiated a broad range of development programs designed to provide for the safe disposal of radioactive wastes

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*For a description of the three fuel cycle options, see CHAPTER I, Section 2.0.

(see Alternatives for Managing Wastes from Reactors and Post Fission Operations in the LWR Fuel Cycle, ERDA-76-43, May 1976). The ERDA program is being implemented by the Office of Waste Isolation, Union Carbide Nuclear Division. It is beyond the scope and intent of this analysis to discuss all of the aspects of the overall ERDA waste management program and attention is directed to that portion of the program aimed at providing disposal facilities for the high level and transuranium wastes and how such operations may be affected by these alternative modes of operating the LWR industry. The initial goal of the ERDA program is the identification of geologic structures and sites suitable for the construction of pilot plants for further demonstration of the practicability of waste disposal in geologic formations. While a low level effort is continuing to examine the feasibility of storage in engineered surface structures, the major emphasis is upon geologic disposal for the ultimate disposition of high level and transuranium wastes. For that reason and based on the results of waste disposal tests in bedded salt (Project Salt Vault: A Demonstration of the Disposal of High Activity Solidified Wastes in Underground Salt Mines, ORNL-4555, April 1971), a model geologic repository in bedded salt has been assumed for this analysis in order to assess the three fuel cycle options referred to above.

Also under development are various liquid-to-solid, volume reduction, and encapsulation processes that will provide a stable, chemically inert waste product suitable for emplacement in a repository. A particular choice among the various processes has not been made for this analysis because of the variety of processes being considered and the relative insensitivity of the fuel cycle options to the use of a particular waste solidification operation. Consequently, it is judged that the choice of any given waste solidification process will not have an appreciable effect on the relative desirability of any of the fuel cycle options considered.

The Future Industry

To support the LWR industry, by about the year 2000 an additional five land burial sites for nontransuranium wastes will be required. Interim pool storage of spent fuel assemblies will be provided at the reactors, the two inactive reprocessing plants and, if spent fuels are reprocessed, the five reprocessing plants that will be required by the year 2000. Interim surface storage for high level and transuranium wastes will be provided at the reprocessing plants and mixed oxide fuel fabrication plants. By year 1990, two geologic repositories will be required for the disposal of the solidified high level wastes and the transuranium wastes generated at the reprocessing plants and mixed oxide fuel fabrication plants, or for the spent fuel assemblies in the event of the no recycle option.

Waste Generation

Table IV H-1 summarizes the cumulative waste volumes in the year 2000 resulting from the operation of the LWR industry during the years 1975 through 2000 for the three options of no recycle, uranium recycle only, and recycle of both uranium and plutonium. It shows that with the exception of mill tailings, the waste volumes are not significantly different for the three fuel cycle options. However, a substantial reduction in the amounts of mill tailings generated (and long term waste management requirements and radon releases) result from recycling uranium or both uranium and plutonium.

Table IV H-1

CUMULATIVE VOLUMES OF WASTE INVENTORY
BY THE YEAR 2000

(Cubic Meters)

Type of Waste	Fuel Cycle Option		
	No Recycle	U Recycle	U + Pu Recycle
Mill Tailings	7.8×10^8	6.9×10^8	5.9×10^8
Spent Fuel	55,000*	6,000***	6,000***
High Level	**	6,500[†]	6,500[†]
Transuranium	**	76,500[††]	96,500
Hulls and Hardware (Transuranium)	**	52,000	52,000
Reactor Waste (Nontransuranium)	3.8×10^6	3.8×10^6	3.8×10^6
Other Nontransuranium	310,000	300,000	223,000
Chemical	179,000	183,000	159,000

*400,000 spent fuel assemblies.

**The spent fuel constitutes these wastes.

***37,000 spent fuel assemblies in pool storage awaiting processing; not a waste.

[†] Volume of high level waste in 37,000 canisters.

[††] Includes plutonium waste.

Because the plutonium is recycled in LWR fuel, recycle of uranium and plutonium results in the accumulation of the least amount of plutonium (about 17 MT as opposed to more than 1,000 MT) in the wastes for the three fuel cycle options considered despite the fact that plutonium-bearing wastes will be generated at the mixed oxide fuel fabrication plants. With uranium recycle only, the plutonium recovered from reprocessing the spent fuel is considered a waste material and is disposed of in a manner similar to the high level waste. With no recycle, the spent fuel assemblies contain all of the plutonium generated in the fuel; these spent fuel assemblies constitute the high level waste.

Environmental Impact

The largest volume of waste generated in the fuel cycle is the impounded solid tailings at the uranium mills. Tailings are stored in the vicinity of the mills which are presently located in remote regions of the western United States. With no uranium or plutonium recycle, the volume of these wastes generated in the years 1975 through 2000 would be about 780 million cubic meters. With uranium and plutonium recycle, the volume of these wastes will be reduced by about 24%, and with recycle of uranium only by about 12%.

The largest quantity of wastes expected to be consigned to licensed commercial burial grounds is that generated during reactor operations. The content and quantity of these wastes is not anticipated to vary substantially between the three fuel cycle options. The volume of wastes generated at all LWR's for the years 1975 through 2000 will be 3.8 million cubic meters and will require about 540 acres of burial ground.

Surface lands permanently committed as a result of mining and milling are substantially reduced by uranium and plutonium recycle over no recycle. However, subsurface acreages in geologic facilities required for the disposal of wastes generated by uranium and plutonium recycle (1,100 acres) are about 12% greater than with no recycle. See Table IV H-2.

Table IV H-2

LAND PERMANENTLY COMMITTED TO DISPOSAL OF RADIOACTIVE WASTE FOR THE YEARS 1975 THROUGH 2000

	Acres of Land Permanently Committed		
	No U or Pu Recycle	U Only Recycle	U and Pu Recycle
U_3O_8 Production			
Mining	11,000	9,500	7,800
Mill Tailings	16,500	14,600	12,600
Total	27,500	24,100	20,400
Burial Grounds			
Reactor Operation	543	543	543
Fuel Cycle Facilities*	70	69	55
Total	613	612	598
Federal Repositories**			
Transuranium Waste	†	50††	65
High Level Waste***	†	830	1,000
Hulls and Hardware	†	35	35
Spent Fuel***	970	___	___
Total	970	915	1,100

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*Includes chemical wastes buried onsite.

**These are the subsurface acreages required for disposal of the wastes. The surface facilities of the two Federal repositories required in the year 2000 occupy 500 acres.

***Based on 115,000 MTHM reprocessed or disposed of as spent fuel.

†The spent fuel constitutes these wastes.

††Does not include 170 acres required for plutonium waste.

The estimated total radiological and nonradiological releases to the atmosphere for the years 1985 through 2000 from the operation of two model geologic repositories are shown in Table IV H-3. During normal operation of a model bedded salt repository, the release of small amounts of nonradiological pollutants and trace quantities of radionuclides should have only negligible effect on the environment. The radiological annual 50-year bone dose commitment to an individual located at the site perimeter would be about 0.3 mrem and to the general U.S. population about 0.5 to 0.6 person-rem for all fuel cycle options. These doses are insignificant when compared to those received from natural background radiation. The thermal effects from underground storage of high level wastes are expected to take place very slowly over a long time period so that the net effects at the surface would be almost unnoticeable. The overall environmental impact from the operation of a geologic repository is judged to be approximately the same whether there is no recycle, only U is recycled, or both U and Pu are recycled.

Table IV H-3

ESTIMATED TOTAL RELEASES FOR THE YEARS 1975 THROUGH 2000 FROM THE OPERATION OF TWO MODEL GEOLOGIC REPOSITORIES*

Material	Total Release
High level waste particles	0.189 Ci
Alpha waste particles	0.81 Ci
^{85}Kr (spontaneous fission)	0.397 Ci
^{3}H (spontaneous fission)	0.024 Ci
^{222}Rn (natural sources)	24.3 Ci
^{220}Rn (natural sources)	1.08 Ci
H_2 (corrosion, radiolysis, electrolysis)	144.7 MT
He (alpha decay)	0.07 MT
HCl (brine decomposition)	45.8 MT
CO_2 (diesel exhaust)	39,470.0 MT
CO (diesel exhaust)	25.2 MT
NO_2 (diesel exhaust)	41.2 MT
SO_2 (diesel exhaust)	34.5 MT
CH_2O (diesel exhaust)	0.38 MT
Soot (diesel exhaust)	0.025 MT
Salt particles	0.061 MT

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*Assumes one repository operates for 16 years (1985-2000) and the other operates for 11 years (1990-2000) resulting in a total of 27 repository-years of operation.

The greatest radiological impact from the wastes of the nuclear fuel cycle results from the long term release of ^{222}Rn from the retired tailings piles. The annual releases of ^{222}Rn from the tailings produced in the period 1975 through 2000 (after retirement of the tailings piles) is estimated at 420,000 curies for no recycle, 376,000 curies for uranium recycle only, and 327,000 curies for recycle of both uranium and plutonium. The highest population doses (50-year dose commitment) for the United States population from these releases for all pathways are received by the total body, kidney, liver, and lung. For example, the annual total body dose commitment for the no recycle option is 47,000, for the uranium recycle option 43,000, and for the uranium and plutonium recycle 37,000 person-rem. For the kidney the dose commitments would be

180,000, 170,000, and 140,000 person-rem, respectively. On a gross basis, the population doses indicate that the effects of radon from the tailings piles would be highest for the no recycle option, would be reduced by 11% if uranium only recycle were adopted, and would be reduced by an additional 12% (to a total of 23%) for the uranium and plutonium recycle option.

The model tailings pile used in this assessment, when retired following stabilization after the year 2000, is estimated to release 7,500 curies of ^{222}Rn per year. The annual 50-year dose commitment from the ^{222}Rn release at 500 meters from the pile, for the inhalation pathway, is received by the bone (0.12 rem), kidney (0.29 rem) and lung (0.17 rem). For the total diet pathway, the highest dose commitment is received by the total body (0.19 rem), bone (0.57 rem), liver (0.13 rem), and kidney (0.64 rem). These dose commitments continue for many thousands of years beyond year 2000.

The dose commitment from the model tailings piles can be placed in perspective by comparing them to the dose commitments received from natural background radon. For example, in one assessment, a lung dose commitment of 60 mrem/yr was estimated at a distance of 0.5 mile. In this case, the radon concentration at 0.5 mile from the stabilized model tailings deposit was 5 times the average background of 0.41 pCi/liter measured in three of the four milling cities by the Public Health Service; at 1 mile it was 1.5 times background; at 5 miles it was only 1.15 times background; and at 50 miles the radon from the tailings pile was indistinguishable from natural background radon.

Accidents

Considering the type and integrity of the facilities that will be designed for waste handling and disposal applications, little environmental impact from accidents is expected. The maximum credible operational accident at a waste repository is postulated to involve a major rupture of a high level waste canister during handling in the waste receiving cell. Such an accident, involving the average mix of solidified high level waste from uranium and plutonium recycle, would increase the 50-year bone dose commitment at the site boundary by a factor of 2 over that resulting from a similar accident involving the high level waste from uranium recycle only (2.8 mrem to 5.6 mrem at 1,500 meters).

Conclusion

The result of this assessment is that there is no clear preference for any specific fuel cycle option based on radioactive waste management considerations. It should be noted, however, that the no recycle option eliminates plutonium handling, that either the uranium only or the uranium and plutonium recycle options reduce committed land or radiological releases (due to reduced ore requirements), and that the uranium and plutonium recycle option minimizes the actual quantity of plutonium which enters waste streams. These, however, are interdependent considerations and not of sufficient significance upon which to base a preferential judgment. The major conclusion of the assessment is that no waste management consideration is sufficiently significant to dictate a decision between the three fuel cycle options.

1.0 INTRODUCTION

The purpose of this section is to assess the differential impacts from management of radioactive waste for the LWR industry associated with three nuclear fuel cycle options: no recycle, recycle of uranium only, and recycle of both uranium and plutonium. The amounts of wastes expected to be produced in all segments of the fuel cycle for each option for the years 1975 through 2000 have been estimated and the potential effects on the environment and the population are assessed. These estimates and assessments were made using model facilities representative of the various fuel cycle operations and model waste management facilities that are based on ongoing ERDA research and development programs. The draft GESMO statement (WASH-1327) addressed a number of waste management alternatives. Since that time, the ERDA program has focused on geologic disposal of high level waste and transuranium wastes. Accordingly, this final statement assesses the environmental impacts using a model Federal geologic repository for disposal of such wastes. Waste management alternatives are not the issue of concern in this assessment and are treated elsewhere.[1]

Federal agencies with help from States, national laboratories, universities, and a number of citizens groups are presently formulating and assisting with implementing a national policy and program for the management of nuclear wastes. This work is ongoing and as yet final decisions have not been reached regarding the facilities which will be required for the management of radioactive waste materials in the future. Since there is no final decision, a model facility has been defined in this section that is representative of the developing policy and program. It is also important to note the recognized limitations of the waste management system described herein. The system described in this section provides a basis for comparing the impact of the fuel cycle options of interest and is illustrative and judged to be representative of any variations which may (in time) be implemented. It should be recognized that this waste management system, while representative of the present approach to waste management, is not presented as the best or optimum system or a dictum or directive to responsible agencies for the development of policy or major programs.

The protection of Man and the environment from radioactive wastes requires planning and the implementation of a waste management system. The two major protective measures under consideration are isolation and containment. A common objective exists between the two. Isolation takes advantage of factors that prevent the transmission of radionuclides to Man's environment. Typical isolation factors include: distance, the ion exchange capacity of soil, and the absence of water in the disposal area. Alternatively, if adequate barriers can be provided for containment for the length of time necessary for radioactive decay, wastes could be placed in protected locations in Man's environment. The word "barrier," as used here, includes a confining matrix such as silicate glass in which waste is incorporated, enclosures such as metallic casings, and manmade structures constructed of materials such as concrete.

Presently, plans are underway to determine the feasibility of disposal of high level wastes, plutonium-bearing wastes, and transuranium-contaminated waste in geologic formations in five areas of the United States.[2] Isolation in geologic formations is based on emplacement of the waste in salt beds, shale, or other geologic formations of predictable stability. Interim surface storage of the wastes could provide adequate containment until these facilities are in operation. Federal repositories would be operated by ERDA under NRC licenses.

Radioactive wastes generated during fuel reprocessing operations are classified into two categories: high level wastes and other-than high level wastes. High level liquid waste is defined in Appendix F, 10 CFR Part 50,[3] as the aqueous waste resulting from operation of the first cycle solvent extraction system or equivalent, and the concentrated waste of subsequent extraction cycles or equivalent in a facility for reprocessing irradiated reactor fuels. Over 99.9% of all nonvolatile radioactive isotopes discarded from the LWR fuel cycle will be present in the high level waste category. The other-than high level wastes can be divided further as "transuranium" and "nontransuranium" wastes, with subcategories of intermediate and low level based on the quantities of penetrating beta gamma radiation also present. Transuranium wastes in a proposed radiation protection standard are defined as wastes containing greater than some limit* of plutonium and other long-lived transuranium elements per gram of waste. The nontransuranium wastes contain less than some limit* of transuranium elements.

Appendix F, 10 CFR Part 50, provides that the inventory of high level liquid waste at a fuel reprocessing plant will be limited to that produced during the prior 5 years. High level liquid radioactive wastes will be converted to a dry solid in order to comply with this inventory limitation. The resulting solid waste will be placed in sealed containers and transferred to a Federal repository no later than 10 years following separation of fission products from the irradiated fuel.

The high level waste from the reprocessing of the spent fuel will contain essentially all of the nonvolatile fission products, the transplutonium elements, the neptunium, and about 0.5% of the uranium and plutonium that was initially in the spent fuels. Of all the wastes generated in the fuel cycle, spent fuel or high level waste is potentially the most difficult to handle in terms of isolation or containment because of the need for shielding and heat dissipation. The solidified high level waste, 5 years after reprocessing, generates 1.8 to 2.1 kW** of heat per metric ton of processed LWR fuel. It contains some 435,000 Ci of fission product activity and 2,000 to 12,000 Ci** of actinide activity per metric ton. One thousand years after reprocessing, the fission product activity will have decreased by a factor of 1,000 and the actinide activity by a factor of 20 to 40** with a corresponding decrease in heat generation.

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*The specified concentration limit has not been established but a limit of 10 nCi/g of waste was indicated in the proposed amendment to 10 CFR Part 20.

**The quantity of heat and radioactivity per metric ton of fuel depends upon whether uranium alone is recycled or both uranium and plutonium are recycled.

In the uranium recycle option, the plutonium present in the spent fuel is assumed to be a waste product and, because of its high biological hazard and long decay time, it is assumed that the plutonium will be isolated in Federal repositories along with the high level and other transuranium wastes. The waste plutonium* is assumed to be a nonpurified plutonium solid that is handled in a manner similar to that used for hulls and hardware and the solidified high level wastes. Because of the potential for nuclear criticality, the plutonium may have to be mixed with neutron absorbing material, or the size of the stored units and their spatial distribution carefully controlled.

The volume of transuranium and nontransuranium wastes generated in all segments of the fuel cycle is several hundred times the volume of the high level waste, but these wastes initially contain less than one-thousandth of the radioactivity present in the high level waste. The nontransuranium wastes from the LWR UO_2 fuel cycle will be buried in licensed commercial burial facilities. Amendments to 10 CFR Part 20 are under consideration which would prohibit the burial of transuranium wastes in near surface burial facilities, and would require that wastes containing transuranium elements be transferred to ERDA for storage or isolation in a Federal repository within 5 years after its generation.[4] Adoption of the proposed amendments would be consistent with present management of transuranium waste generated by ERDA license-exempt facilities, as prescribed by ERDA Manual Chapter 0511.[5] For the purposes of this study it is assumed that these amendments will be approved. A large fraction of the wastes generated at the mixed oxide fuel fabrication plants and reprocessing plants would be classified as transuranium wastes.

Section 2.0 discusses the sources and kinds of wastes generated by all segments of the supporting LWR nuclear fuel cycle industry for the three options of no recycle, uranium recycle only, and recycle of both uranium and plutonium. Section 3.0 addresses the disposal of these wastes in both surface and subsurface facilities and the environmental impact of these waste management activities. Section 4.0 is a summation section which assesses the incremental environmental effects on the waste management program supporting the LWR industry that are associated with the three fuel cycle options.

2.0 RADIOACTIVE WASTES FROM THE LWR INDUSTRY

Wastes containing radioactive isotopes are generated from all segments of the LWR industry. Wastes from the uranium feed chain contain only naturally occurring radioisotopes, but the wastes from other segments of the nuclear fuel cycle contain radionuclides produced by reactor operations. Wastes from present mining and milling operations contain naturally occurring concentrations of radioisotopes and are generally stored on the earth's surface at the mine or mill site. Those wastes that contain sufficiently low levels of reactor generated radioisotopes, or have concentrations of natural isotopes somewhat greater than those occurring naturally, are disposed of in licensed burial facilities. For this analysis, it is assumed that wastes containing significant amounts of transuranic radionuclides will be sent to Federal geologic repositories.

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*The total volume of the impure plutonium waste would be approximately 1% of the total estimated volume of transuranium wastes generated for this option.

The estimated annual waste production from the model facilities of the nuclear fuel cycle is shown in Table IV H-4. If both uranium and plutonium are recycled, the waste volumes from mining, milling and UF_6 production will be decreased by about 25% and the waste volumes from enrichment by about 14% from the requirements with no recycle. With uranium recycle only, the waste volumes would be reduced about 11% and the enrichment waste volumes would be about the same as that required with no recycle.

Nuclear fuel cycles in which only uranium is recycled generate wastes that are not greatly different from those which recycle both uranium and plutonium. The major differences are: high level wastes have increased concentrations of transuranium elements, particularly americium and curium; and wastes from the mixed oxide fuel fabrication plant constitute an additional transuranium waste if both uranium and plutonium are recycled. Adoption of the uranium only recycle option leaves essentially all of the plutonium recovered by fuel reprocessing to be managed as a waste, approximately 1,000 MT of impure plutonium solids having a total waste volume of about 1,000 m^3.

The following summarizes the sources and types of waste generated by each segment of the LWR industry, and discusses the planned or projected method of disposal. The volumes of the various wastes generated by the LWR industry for each of the recycle options are estimated for the period 1975 through the year 2000.

2.1 Uranium Mining

Uranium ore is obtained from both underground and open pit mining operations. Wastes from underground mines consist mainly of rock removed in creation of shafts for access and ventilation and of passageways to make the ore body accessible. Wastes from open pit operations consist largely of overburden removed to expose the ore body. Wastes from either operation are expected to contain only relatively small amounts of uranium. The mine wastes should have essentially the same background radioactivity typical of the region.

Current practice is to conduct mining operations so as to minimize the impact on the environment.[6,7] Mine waste from opening the first pit or from passageways and haulways in underground mines is deposited so that it is unobtrusive and blends with the contours of the surrounding countryside. As later pits are opened in the open pit operation, the overburden removed is used as backfill in the earlier pits. The sides of the final pit are contoured and the pit may be left to become a lake. Disturbed areas are covered with earth and natural vegetation is then established. The successful reclamation of an existing mine waste dump indicates that in areas with 12 to 14 inches of annual precipitation it should be possible to return much of the disturbed area to grazing.[8] In arid locations which receive only 6 to 10 inches of precipitation annually, the land was marginal for grazing before it was disturbed and its reclamation for limited grazing use may not be cost effective.

If neither uranium nor plutonium is recycled and assuming that the entire environmental impact of the model mine occurs before the end of the first year of ore

Table IV H-4

ANNUAL WASTE PRODUCTION FROM THE MODEL FUEL CYCLE FACILITIES*

Source	Annual Capacity	Type of Waste	Volume of Waste (cubic meters)	Radioactivity (curies)	Method of Disposal
Uranium Mine					
Underground	18.1 st U_3O_8	Mine waste	Not estimated	Not estimated	Spread to be unobtrusive, covered with earth, and vegetation established
Open pit	181. st U_3O_8	Mine waste	Not estimated	Not estimated	
Uranium Mill	1,050 st U_3O_8	Mill tailings	525,000	3,200	Buried onsite under 2-ft earth cover topped with rock or vegetation
UF_6 Production					
Dry process	15,000 MTU	Low level CaF_2; chemical waste	3,900	350	Buried onsite or in licensed burial facility
Wet process	15,000 MTU	Low level CaF_2 sludges; chemical waste	2,480	360	Buried onsite or in licensed burial facility
Enrichment					
Gaseous diffusion	8.75 MTSWU	Low level miscellaneous	100	Not estimated	Buried onsite or in licensed burial facility
Gas centrifuge	8.75 MTSWU	Low level miscellaneous waste	5,600	Not estimated	Buried onsite or in licensed burial facility
Fuel Fabrication					
Enriched uranium	1,500 MTU	Low level CaF_2; miscellaneous	1,225	Neg.	Buried onsite or in licensed burial facility
Mixed oxide	360 MTHM	Plutonium-bearing solids	260	7,200 (10.9 kg Pu)	Federal repository
Reactor Operation					
PWR	0.8 GWy(e)	Low level miscellaneous	440	1,900	Buried in licensed burial facility
BWR	0.8 GWy(e)	Low level miscellaneous	980	4,100	
		Spent fuel**	12.2	F.P.s, 13,100,000 α Tru[††] 180,000 β Tru[††] 2,270,000	Federal repository
Spent Fuel Storage	3,500 MTHM	Low level miscellaneous	21	Neg.	Buried in licensed burial facility
Fuel Reprocessing	2,000 MTHM	Nonradioactive solids	400	Neg.	Buried in licensed burial facility
		Calcined high level wastes	125	F.P.s, 870,000,000*** 876,000,000[†] α Tru[††] 22,200,000*** 4,540,000[†] β Tru[††] 1,400,000*** 900,000[†]	Federal repository
		Miscellaneous transuranium solids (i.e., hulls, hardware, Pu-bearing solids, miscellaneous lab waste)	2,300	Act.P.[††] 10,000,000*** 10,000,000[†] F.P.s 870,000*** 880,000[†] α Tru[††] 63,000*** 12,600,000[†] β Tru[††] 870,000*** 162,000,000[†]	

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*For a description of the model facilities see appropriate sections of CHAPTER IV.

**Where no U or Pu is recycled, the spent fuel assemblies constitute a "waste" from reactor operation; annual removal from 1,000 MWe PWR.

***Activities associated with reprocessing wastes when both uranium and plutonium are recycled; decayed 5 years.

[†] Activities associated with reprocessing wastes when only uranium is recycled; all plutonium considered waste; decayed 5 years.

[††] TRU - transuranium elements; Act.P. - activation products in hulls and hardware.

production, it is estimated that for the period 1975 through year 2000 a total of approximately 334,000 acres of land would be disturbed by mining operations; 11,000 acres of this total would be permanently committed open pit areas. The total volume of ore removed would be about 780,000,000 cubic meters.

Over the same 26-year period, uranium recycle would reduce the area disturbed by mining by approximately 45,000 acres (13%) and the permanently committed open pit area by approximately 1,500 acres (14%). The volume of ore mined is also proportionally reduced to 690,000,000 cubic meters.

Over the period 1975 through year 2000, plutonium and uranium recycle, compared to no uranium or plutonium recycle, would reduce the area disturbed by mining operations by approximately 97,000 acres (29%) and the permanently committed open pit area by approximately 3,200 acres (29%). The total ore volume is estimated at 590,000,000 cubic meters.

Effects of the three fuel cycle options, based upon these assumptions, are summarized in Table IV H-5.

Table IV H-5

PROJECTED LAND USE IMPACT OF URANIUM MINING FOR THE YEARS 1975 THROUGH 2000

	No U or Pu Recycle*	U Recycle Only	U and Pu Recycle
Area disturbed (acres)			
Underground mines	56,000	48,500	39,500
Open pit mines	278,000	240,500	197,500
Total	334,000	289,000	237,000
Area permanently committed (acres)			
Open pit mines	11,000	9,500	7,800

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*Based on the assumption that the entire land use impact of the model uranium mines occurs before or during the first year of ore production.

2.2 Uranium Milling

Leaching processes at the uranium mill recover a large fraction of the contained uranium but dissolve only a very small fraction of the finely ground ore. The mills now discharge these undissolved solids (consisting of sands and slimes and referred to as mill tailings) along with nearly uranium free process water and its contained chemicals to an impoundment area (tailings pond) near the mill. Natural evaporation is generally sufficient to dispose of the water and, as the pond evaporates, salts crystallize and become part of the solid tailings. The quantity of such tailings is large. If the average ore is assumed to contain 0.1% U_3O_8 in the period years 1975 through 2000, about 1.3 metric tons of sands and slimes would be generated for each kilogram of uranium recovered. On this basis it may be estimated that at the end of the year 2000 if no U or Pu is recycled some 1.55×10^9 metric tons of tailings will have been generated. This number is reduced to 1.38×10^9 if uranium is recycled and to 1.19×10^9 if uranium and plutonium are recycled. If the tailings are assumed to be piled to an average height of 38 feet, the piles at the end of year 2000 would occupy 16,500, 14,600, or 12,600 acres, respectively, if no recycle, uranium only, or uranium and plutonium recycle is used.*

The tailings contain about 75-85% of the radioactivity originally present in the ore. Principal activities are those of ^{230}Th (half-life, 83,000 years) in secular equilibrium with nine radioactive daughters including ^{226}Ra (half-life, 1,600 years) and ^{222}Rn. During the period years 1975 through 2000, an estimated 4.7 million curies of alpha plus beta activity would accumulate in mill tailings for the no recycle option, 4.2 million curies if uranium only were recycled, or 3.6 million curies if both uranium and plutonium were recycled.

A detailed generic statement of the environmental impact of mill tailings is planned.[9] The present assessment assumes that when the mills are decommissioned the tailings will be stabilized (covered with 2 feet of earth topped by rock or vegetation) against erosion by wind and water and that periodic inspection and maintenance will ensure the integrity of the cover.

Effects of the three fuel cycle options, estimated with the assumptions indicated above, are summarized in Table IV H-6.

2.3 Uranium Hexafluoride Production

Wastes from UF_6 production facilities contain large quantities of chemicals, because excess reagents are used to achieve a high uranium recovery, but they contain very low levels of radioactivity. The wastes comprise three separate classes: a low level CaF_2 ash from the fluid bed fluorination at the dry plants, a low level sludge from neutralized solvent extraction raffinate at plants using wet processes, and low level CaF_2 chemical wastes from treating scrub liquors at both types of plants. The CaF_2 ash is presently drummed and shipped offsite to a licensed burial facility. Low level sludges may also be drummed and shipped offsite to a licensed burial facility. The low level CaF_2 chemical wastes are now stored or buried onsite; however, by the year 2000, they may be recycled to recover fluoride. The ash from fluorination and the sludge from neutralized solvent extraction raffinate contain ^{226}Ra, which was present in the yellowcake feed entering the UF_6 plant, and are, accordingly, potential long term radon sources.

In the period 1975 through year 2000, the UF_6 production facilities make only a small contribution (Table IV H-7) to the total waste management responsibility from

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*These numbers represent the prorated share of area which the tailings generated through the year 2000 will occupy when at some later date the mill ceases operation and the tailings pile is stabilized.

Table IV H-6

WASTE MANAGEMENT PARAMETERS FOR URANIUM MILL TAILINGS FOR THE YEARS 1975 THROUGH 2000

(Half-life of ^{230}Th, parent of the major chain, is 83,000 years)

	Composite Model Tailings Pile*	No U or Pu Recycle**	U Recycle Only**	U and Pu Recycle**
Quantity of tailings (million metric tons)	27	1,550	1,380	1,190
Area of tailings (acres)	289[†]	16,500[†,††]	14,600[†,††]	12,600[†,††]
Principal radionuclides (curies)[§]				
^{238}U, ^{234}Th, ^{234m}Pa, ^{234}U, each	770	43,600	38,700	33,500
^{230}Th, ^{226}Ra, ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, ^{214}Po, ^{210}Pb, ^{210}Bi, ^{210}Po, each	8,000	451,000	403,000	350,000
Total Activity	83,000	4,680,000	4,190,000	3,630,000
^{222}Rn release from tailings after	7,500[§§]	420,000[††,§§]	376,000[††,§§]	327,000[††,§§]

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*3,500 metric tons of ore/day, 300 days/yr, 26 yr life for mill, 0.104% U$_3$O$_8$, 90.4% U recovery.

**Approximately 0.10% U$_3$O$_8$ in ore, and 90% U recovery.

[†] Average thickness of tailings, 38 ft. Derived from Reference 10, page 175, Case 1 assuming 85% acid leach solvent extraction--15% alkaline leach mills and 50% New Mexico--50% Wyoming sites.

[††] These numbers represent the prorated share of area which the tailings generated through the year 2000 will occupy when at some later date the mill ceases operation and the tailings pile is stabilized.

[§] One year after milling.

[§§] Derived from Reference 5, Waste Treatment Case 2, 2-foot earth cover, 4% moisture.

Table IV H-7

ESTIMATED WASTE VOLUMES AND LONG TERM ANNUAL RELEASED ACTIVITIES FROM URANIUM HEXAFLUORIDE PRODUCTION FOR THE YEARS 1975 THROUGH 2000

	No U or Pu Recycle	U Recycle Only	U and Pu Recycle
Low level wastes (cubic meters)	88,000	79,000	67,000
Chemical wastes (cubic meters)	179,000	160,000	136,000
Radon* release from low level wastes (curies/year)	4,200	3,800	3,200

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*These are the annual radon releases after the year 2000 from the waste accumulated in the 26-year period. The radon comes from the decay of ^{226}Ra, which is present as an impurity in the yellowcake feed to the UF$_6$ plant.

the operation of the nuclear fuel cycle. There would be about a 24% reduction in these wastes and the associated long term radon releases with uranium and plutonium recycle, and about an 11% reduction if uranium recycle is chosen compared with no recycle.

Adoption of either recycle option will lead to generation of similar wastes, as described in paragraph 2.9.4, from preparation of UF_6 from uranium recovered in the fuel reprocessing plants.

2.4 Uranium Enrichment

Wastes generated by the enrichment process are primarily those resulting from equipment cleanout and uranium recovery. Liquid wastes are impounded in holding ponds where most of the uranium is collected in sludges that are periodically removed and buried onsite. In addition to the wastes generated by the enrichment plants, approximately 35 million cubic meters of nonradioactive sludges (water and solids) result from the fossil fueled power plant operations supplying power for the enrichment processes. These are disposed of at the power plant sites.

The enrichment requirements for the period year 1975 through year 2000 for a nuclear industry in which uranium and plutonium are recycled would be 523 million separative work units (SWU), for uranium recycle only 613 million SWU, and for no recycle 608 million SWU. The increase in separative work requirements for uranium recycle over no recycle is due to the ^{236}U present in the recycled uranium. In the latter two modes of operation, the enrichment needs are assumed to be fulfilled by four diffusion plants and two centrifuge plants as opposed to four diffusion plants and one centrifuge plant where plutonium as well as uranium is recycled. A gas centrifuge enrichment plant is expected to generate large quantities of nonreusable parts and materials from failed machines. The annual waste from a model centrifuge plant is estimated to be about 56 times greater than that from a model diffusion plant for the same separative work output. Thus, the elimination of one of the centrifuge plants reduces the projected amount of waste to be generated from enrichment plants by a large percentage.

Over the period from year 1975 through year 2000, with no recycle or with only uranium recycle, it is estimated that enrichment plants would generate approximately 65,000 cubic meters of low level waste that would be shipped to a licensed burial facility. With plutonium and uranium recycle, over the same 26-year period, the volume of waste generated by enrichment plants would be reduced by approximately 43,000 cubic meters (66%) compared to recycle of uranium only or to no recycle of uranium or plutonium, due principally to the avoidance of the second centrifuge facility.

2.5 Uranium Fuel Fabrication

Wastes are generated from uranium fuel fabrication plants by the conversion of UF_6 to UO_2 and by the preparation of fuel elements from UO_2. The most significant waste is CaF_2 which is formed during the conversion operation at the rate of one metric ton for each metric ton of uranium processed. The uranium content of the CaF_2 is estimated to be about 0.01 μCi/gm. The current practice is to package or store the waste in bulk form onsite. Wastes produced during the preparation of fuel elements from UO_2 are inconsequential.

The fabrication of uranium fuels will result in the expected generation of approximately 189,000 metric tons of CaF_2 wastes during the period from year 1975 through year 2000 without recycle of uranium or plutonium, or with recycle of uranium only. This is equivalent in volume to about 146,000 cubic meters. In addition, the volume of other low level wastes--paper, rags, clothing, tools, etc.--is estimated to total about 7,650 cubic meters. This waste is shipped to licensed burial facilities for disposal.

The recycling of uranium and plutonium over the same 26-year period would reduce the CaF_2 waste generated by about 26,000 metric tons or 20,000 cubic meters. The volume of other low level wastes would not change significantly.

2.6 Mixed Oxide Fuel Fabrication

The mixed oxide fuel plants, which produce fuel rods containing uranium and plutonium, are an incremental operation resulting from plutonium recycle. Thus, the radioactive wastes generated at these plants must be considered as an additional environmental cost of plutonium recycle.

With plutonium recycle, the mixed oxide fuel fabrication plants are expected to generate approximately 18,500 cubic meters of waste containing about 15 metric tons of heavy metals over the period 1975 through the year 2000. Because the wastes are estimated to contain about 700 kilograms of plutonium, they are assumed to be transferred to a Federal geologic repository for disposal as transuranium wastes.

2.7 LWR Operations

The quantity of solid wastes produced by LWR's is expected to be the same for all three fuel cycle options. The radioactive composition of these wastes would not vary significantly with different fuel cycle options. Based on a review of reactor licensees' semi-annual operating reports through December 1975, a 1,000 MWe PWR produces annually about 323 cubic meters (11,400 ft^3) of spent ion exchange resin, filters, filter sludge, and evaporator bottoms, containing about 1,900 curies (0.17 curies/ft^3), and 116 cubic meters of dry and compacted solid waste containing less than 5 curies.[11] A 1,000 MWe BWR produces annually about 850 cubic meters (30,000 ft^3) of spent ion exchange resin, etc., containing 4,100 curies (0.14 curies/ft^3) and 133 cubic meters of dry and compacted solid waste containing less than 5 curies.[11] The estimated annual waste volume from all LWR's operating by the end of the year 2000 would be about 250,000 cubic meters of spent resin, etc., containing about 1.3 million curies, and 51,000 cubic meters of dry solid waste containing on the order of 2,500 curies.*

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*Based on an estimated 507 LWR's, of which two-thirds would be PWR's and one-third BWR's.

Because this waste does not contain more than traces of plutonium, it can be disposed of in licensed burial facilities and would constitute the bulk of buried waste volume resulting from operation of LWR's.

Over the period 1975 through the year 2000, the total low level waste volume from LWR's would amount to about 3,800,000 cubic meters, independent of fuel cycle option.

2.8 Spent Fuel Storage

Over the period 1975 through year 2000, LWR reactor operations would discharge spent fuel assemblies containing about 126,400 metric tons of heavy metal. These would be stored temporarily at water pool storage basins at reactors, independent spent fuel storage facilities, or fuel reprocessing plants. If no fuel is reprocessed, the spent fuel assemblies would constitute the major radioactive waste and would be transferred after 5 to 10 years of decay to a Federal geologic repository for disposal.

It is estimated that a spent fuel storage facility using water basins to store 3,500 metric tons of heavy metal as spent fuel would generate annually about 21 cubic meters of low level radioactive solids consisting of miscellaneous wastes and spent resin from pool water clean-up.

With no recycle of uranium or plutonium, over the period 1975 through year 2000, interim pool storage would total about 558,800 metric ton-years, with generation of about 3,350 cubic meters of low level waste that would be shipped to licensed burial facilities. By the year 2000, about 50,100 metric tons of heavy metal would remain in pool storage, and about 76,300 metric tons would have been sent to Federal geologic repositories.

Over the same 26-year period, with recycle of uranium only, interim pool storage would be reduced to 279,300 ton-years, generating about 1,680 cubic meters of low level waste.

The recycle of uranium and plutonium would reduce interim pool storage to 167,200 metric ton-years, generating about 1,000 cubic meters of low level waste.

2.9 Fuel Reprocessing

In the period 1975 through the year 2000, either recycle option would result in reprocessing of an estimated 115,000 metric tons (as heavy metal) of spent fuel. If neither uranium nor plutonium is recycled, no fuel is reprocessed and the fuel elements and their associated hardware constitute the major radioactive waste to be managed. If fuel is reprocessed, the bulk of the radioactivity from the non-volatile fission products and unrecovered actinides end up in the several types of wastes described below. The volume of such wastes is essentially independent of which recycle option is used. However, the isotopic composition of the radioactive wastes is altered if plutonium recycle is adopted. In the uranium recycle option, the plutonium becomes a waste to be managed.

2.9.1 High Level Wastes

High level liquid radioactive waste is defined in Appendix F, 10 CFR Part 50,[12] as those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels.* If neither recycle option is adopted, the spent fuel requires management in a manner similar to that for the solidified high level waste.

Fluidized bed calcination has been used, for about 10 years, to solidify high level wastes from reprocessing Federal research reactor fuels at the Idaho National Engineering Laboratory. Other waste solidification processes that are being developed through radioactive demonstration on an engineering scale involve heating the liquid high level waste to drive off volatile constituents, primarily water and nitrate ion, and produce a dense, relatively water insoluble material such as a ceramic oxide. This material, if not monolithic, might then be dispersed in an appropriate matrix.

Production of a glass, or perhaps of a ceramic oxide, would generally require additions of a considerable quantity of inert nonradioactive solids (more than 2 moles per mole of radioactive species) to incorporate the radioactive species into materials fusible at reasonable temperatures.

It is assumed that the solidified high level waste is cast, formed, or otherwise contained in a primary container and that this container is provided with an overpack or secondary shell as the outer canister. The reference design of the canister, including its overpack, is a right cylinder some 14 inches in diameter and 10 to 12 feet long. It is assumed to hold about 6.3 ft^3 of solidified waste. Since it is estimated[13,14] that each metric ton of fuel reprocessed, from either recycle option, will produce about 2 ft^3 of solidified high level waste, each canister will hold the waste from 3.14 metric tons of processed fuel.

It is projected that if fuel is reprocessed through the year 2000, the quantity of solidified high level waste will be 230,000 ft^3 (6,500 cubic meters). Solidification (assumed to be completed during the year 2005) will result in filling about 37,000 canisters with high level waste.

Should the option of recycle of uranium only be adopted, the volume of solidified high level waste would be essentially the same; but about 1,000 metric tons of plutonium would require disposal in Federal repositories. Up to now, neither technical nor regulatory programs have addressed the storage form or the canister size for management of plutonium compounds as waste. For purposes of this assessment, it is

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*The environmental impacts associated with the solidification and packaging of high level wastes are assessed against reprocessing plant operations in CHAPTER IV, Section E.

assumed that the plutonium is to be converted to PuO_2 of about 95% purity and encapsulated in overpacked containers each capable of holding 6 kg of the material.* If so, nearly 200,000 such containers would be required by about the year 2005 to hold the 1,000 cubic meters of plutonium waste generated by fuel reprocessing through the year 2000. In this study, it is assumed that these containers could be mixed with the hulls and hardware and the high level wastes for shipment to, and disposal at, the Federal geologic repositories.

If, on the other hand, neither recycle option is used, it is anticipated that the spent assemblies will be emplaced in individual canisters at the reactor site. If so, the much larger total of 400,000 such canisters, about 55,000 m^3, would be required to contain the spent fuel assemblies discharged through the year 2000.

2.9.2 Fuel Element Hulls and Hardware

As a first step in the reprocessing operation, structural components of fuel assemblies are removed and the fuel rods are chopped into short pieces to expose the UO_2 or mixed oxide fuel. Leaching of the fuel from these pieces with nitric acid produces the feed solution for the processing operation and leaves the cladding undissolved. These cladding pieces, called hulls, after washing with nitric acid solution are expected to contain less than 0.1% of the uranium and plutonium present in the fuel[14] along with small quantities of fission products and the products of neutron activation of the metal. Processing of fuels containing 115,000 metric tons of heavy metal during years 1975 through 2000 would be expected to produce about 52,000 cubic meters of such waste (in an uncompacted state). These wastes would contain about 1,000 kg of plutonium if only uranium were recycled or about 1,300 kg of plutonium if uranium and plutonium were both recycled. Because of the differences in neutron spectra in reactors with Pu recycle, activation in the cladding of mixed oxide fuels would be about 25% less than that of UO_2 fuel cladding. The volume of cladding waste would be essentially independent of the recycle option.

Reprocessing plants as presently designed[15,16] propose interim storage of these wastes in vaults or concrete containers. Because of the residual plutonium contained in these wastes, it is assumed that the hulls will be transferred to a Federal geologic repository after interim storage for a few years at the reprocessing plant.

2.9.3 Transuranium Wastes

Reprocessing plants operating under either recycle option will generate plutonium-bearing wastes of several types. These include laboratory wastes (small tools, gloves, etc.), chemical wastes from cleaning the off-gases and plutonium extraction solvent, wastes from the plutonium nitrate-plutonium oxide conversion section of the plant, and certain wastes from the process for preparing UF_6 from the recovered uranium. The volumes of these wastes (estimated to be near 78,000 cubic meters for the years 1975 through 2000 if no compaction or incineration is done) should be nearly independent of

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*For example, after separation from uranium, the plutonium might not be purified further. Rather, it could be reduced to a solid form and remotely packaged in containers approximately 4 inches O.D. by 24 inches long.

the choice of recycle option, but the plutonium content of most of these wastes would increase by about 27% (due to increased Pu throughput) if plutonium recycle is chosen over uranium only recycle.

By using the mean of values estimated for an existing plant design,[17] the volume of plutonium-bearing laboratory waste generated through year 2000 would total 54,000 cubic meters. The separations facilities would produce about 12,600 cubic meters of solidified aqueous wastes from the solvent and off-gas cleaning systems. These wastes would be expected to contain about 0.1% of the plutonium throughput--or about 1,000 kg of plutonium with the uranium only recycle or about 1,300 kg of plutonium if both plutonium and uranium are recycled.

The plutonium conversion facilities would generate transuranium wastes consisting largely of filters, gloveboxes, discarded process components, and solidified process wastes. It is estimated that these wastes would contain about 0.2% of the plutonium throughput. If spent fuel is reprocessed to recycle only uranium, about 6,000 cubic meters of such waste containing about 2,100 kg of plutonium would be expected over the years 1975 through 2000; in addition, about 1,000 metric tons of plutonium would probably be recovered in an impure state and converted to a solid form, about 1,000 cubic meters, to be disposed of (as indicated in paragraph 2.9.1) at a Federal geologic repository. If plutonium is recycled, the plutonium would be converted to a purified oxide and about 7,400 cubic meters of solid transuranium waste containing approximately 2,600 kg of plutonium would be generated through the year 2000.

Spent beds and fines from the conversion of uranium to UF_6 for reenrichment will contain small quantities of plutonium. From estimates of such wastes from a plant in the final design stage,[18] it may be inferred that about 3,700 cubic meters of such waste containing about 12 kg of plutonium would be produced through the year 2000. Neither the quantity of waste nor the quantity of contained plutonium would be changed appreciably by choice of the recycle option.

Though some portions of the laboratory waste may contain transuranium element concentrations sufficiently low to permit burial of the waste in a licensed facility, it is presumed for this study that all these relatively low level transuranic wastes will be sent to a Federal geologic repository. The problems of management of these wastes, in any event, appear to be unaffected by choice of the recycle option.

2.9.4 Chemical Wastes

An estimated 36 million metric tons (about 22,500 cubic meters) of chemical wastes (i.e., spent electrolyte, CaF_2 from treating fluoride scrub liquors, calcine discharges from the $UO_2(NO_3)_2$ to UO_2 conversion step, and K_2UO_4 muds) containing about 140 metric tons of uranium but negligible radioactivity, would be produced at reprocessing plants in the period 1975 through the year 2000. Neither the quantity of waste nor the quantity of contained radioactivity is expected to depend upon which of the recycle options is adopted. These wastes will probably be shipped to a licensed commercial facility for burial, though onsite burial at the reprocessing plant may be acceptable.

2.9.5 Transuranium Wastes from Decommissioning of Plutonium Handling Facilities

The wastes generated during the decommissioning of fuel reprocessing and MOX fuel fabrication plants will be shipped to a Federal geologic repository for disposal. The volume of these wastes are estimated to be about 10% of the total volume of transuranium wastes to be deposited in the Federal geologic repository. This incremental increase in volume is within the accuracy of the estimated waste volumes and does not constitute an additional impact on the repository.

3.0 DISPOSAL OF RADIOACTIVE WASTES FROM THE LWR INDUSTRY

The origins and the nature of the several wastes, assumptions as to the mode of their disposition, and the effects of the three fuel cycle options on quantities and characteristics of the wastes were discussed in Section 2.0. This section discusses possible future waste disposal practices and assesses the effects of the fuel cycle options with respect to waste disposal. The objective here is to assess the impact from the disposal of radioactive wastes based on disposal concepts that may evolve by the year 2000, rather than present a review of the ERDA waste management program or an extended discussion of waste management alternatives.

In the broadest sense, two categories of waste repositories are assumed. They differ with respect to their location in the biosphere. Surface or near surface repositories are assumed to be used for disposal of mining wastes, mill tailings and low level wastes whose content of transuranium elements is sufficiently low. It is assumed that repositories in terrestrial geological formations will be used for containing solidified high level wastes, and all other wastes having appreciable concentrations of transuranic radionuclides including plutonium, and spent fuel elements in the no recycle option. For purposes of this assessment, disposal implies final emplacement of the waste with retrieval being possible only with difficulty. Use of storage, with ready retrieval, as a waste management option is assumed in this assessment only to provide surge space or in-process accommodation, such as the interim storage of spent fuel, hulls and hardware, or high level waste for intervals to permit the decay of short-life radionuclides and to reduce the heat generation rates to levels compatible with geologic disposal.

3.1 Surface Repositories

It is assumed that burial in near surface repositories will be used for three types of low level waste. Such repositories include the dumps for wastes from uranium mining and the stabilized tailings piles from uranium milling operations; wastes in these repositories will contain only naturally occurring radioactive species found in uranium ores. In addition, near surface burial sites, generally similar to existing licensed commercial burial facilities, are assumed for nontransuranium wastes: wastes in which concentrations of transuranium elements are sufficiently low in accordance with proposed amendments to 10 CFR Part 20.

3.1.1 Mining Wastes and Mill Tailings

A large fraction of the mine waste contains radioactive materials at the natural background level typical of the region. The quantities of mining waste removed from near an ore body are estimated to be about 0.1 ton per ton of ore from underground mines and 1 ton per ton of ore from open pit mines.[19] This waste contains very little uranium. The ore body is surveyed as mining proceeds, and the waste is monitored on the trucks before it is dumped. This waste, which is a small fraction of the total, contains less than 100 parts per million of uranium.[19]

Uranium mills receive a large fraction of the radioactivity removed from the mine and about 75-85% of this radioactivity is transferred to the mill tailings pile as indicated in Table IV H-4. Accordingly, as described in paragraph 2.2, large quantities of natural radioactive materials are impounded as low level waste near the mill sites.

Environmental impacts of these surface repositories consist of the permanent commitment of land and potential releases of the naturally occurring radionuclides to the environs of the surface repository. Radioactive releases from the mining wastes (or from the decommissioned mine) and from the mill tailings repository might, in principle, occur through dispersal of airborne particulates, through leaching of surface wastes, or by emanation of radon from the wastes.

3.1.1.1 Mining Wastes

The effects, both of essentially irretrievable commitment of land and of potential long term releases of radioactivity, from a decommissioned mine, are expected to be small compared with the effects of the nearby tailings piles.

In the initial stages of development or operation of the mine, wastes are distributed on the ground and contoured so as to minimize dispersal by water erosion.[20,21] In subsequent stages, these wastes are returned as backfill for used sections of the mine or to fill the mined out open pits. The long term commitment of land results largely from the excavations; the final open pit generally remains, after contouring, to become a lake. The total land area essentially removed from public use by the mines and mine wastes for the years 1975 through 2000 is estimated to be 11,000 acres (17 square miles) for no recycle. Comparable areas are 9,500 acres (15 square miles) if uranium only is recycled and 7,000 acres (12 square miles) if uranium and plutonium are recycled.

The potential for radioactive releases from a decommissioned mine and its waste depends somewhat upon the characteristics of the specific mine, but in general, it appears to be small. Mine wastes should be a negligible source of windblown radioactive particulates after mining ceases, the ore handling areas are cleaned, the low grade ore piles (if any) are stabilized with an earth cover, and the pits are filled with overburden or water.

The possibility that leachings from decommissioned mines can contaminate local water supplies must be evaluated for the particular site geology. Open pit mines are usually located in valleys; natural erosion would be expected to carry sediments into the pits and to bury the activity as has happened in the large reservoirs of Lake Powell and Lake Mead.[22] However, as the open pit first fills with water small quantities of radioactivity might be leached from the small deposits of unmined ore.

Since thick earth covers or water are excellent barriers to the diffusion of radon,[23,24] release of radon from unmined ore deposits in a backfilled open pit mine (or in a pit that had become a lake) should be negligible. When an underground mine is decommissioned and the mechanical ventilation that swept radon from it is discontinued, most of the radon would decay within the mine and only a small fraction would be dispersed into the atmosphere. Radon releases from underground mine waste dumps are expected to be small in comparison with those from the nearby mill tailings piles.

3.1.1.2 Mill Tailings

Notice has been given that a detailed generic statement of the environmental impact of uranium milling operations and the management of mill tailings will be prepared.[25] In the current GESMO assessment, it is assumed that when the mills are decommissioned, the mill tailings (in piles averaging 38 feet in depth) would be stabilized against erosion by wind and water. After the tailings pond has evaporated or has been drained, the pile would be graded to provide gradual slopes and to eliminate depressions which might collect water. Side slopes are presumed to be stabilized with riprap or by dikes, and drainage ditches provided to prevent neighboring surface runoff from reaching the tailings. The tailings would then be covered by 2 feet of earth topped with 6 inches of crushed rock in arid regions or with vegetation in regions with sufficient rainfall. It is assumed that periodic inspection and maintenance will be done to ensure the integrity of the cover. The quantity of mill tailings to be generated in years 1975 through 2000 will ultimately (see paragraph 2.2) occupy 16,500 acres for no recycle, 14,600 acres if uranium only is recycled, or 12,600 acres if plutonium and uranium are both recycled.* These areas represent essentially permanent commitments of land which are much larger than those required by the mines that produce the ore.

As a temporary matter, windblown tailings particles might be dispersed to neighboring areas during the period after closing of the mill and while the tailings pile is drying prior to stabilization. It has been estimated[23] that annual releases of radioactivity during this 2- or 3-year period might equal or exceed the radioactivity in ore dust plus tailings dust released annually while the mill was active.

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*These tailings would, of course, not be in stabilized 38-feet deep piles at the end of year 2000. The model mills are postulated to have a 26-year life; model tailings piles would be retired in the period years 2001 to 2026, and would presumably be stabilized 2 to 3 years after such retirement. At the end of the year 2000 most of the tailings would still be saturated with water or submerged in tailings ponds at active mill sites.

The potential for long term underground migration of radioactive materials via seepage or leaching by natural waters can only be evaluated in terms of a specific site; however, in general, such migration is expected to be low from a properly sited tailings area. The tailings tend to seal the pond bottom, and many tailings areas are located on soils with high clay content that have good ion exchange properties and that tend to seal upon contact with acidic effluents. In the arid environment of most western uranium mills, rainfall is generally insufficient to penetrate beneath the soil cover. Accordingly, the driving force for migration of activity ceases when the mill closes, process liquid no longer enters the tailings area, and the pond water has evaporated.

Radon-222 gas, generated by the decay of the ^{226}Ra parent and ^{230}Th grandparent, will emanate from the stablized tailings pile. The rate of radon release, which will continue for a very long time, will depend upon the area covered by dry tailings, the radium content of the upper portion of the pile, the thickness of the earth cover and the moisture content of both tailings and cover.[23] The amount of total radon released increases very little as the depth of the pile increases beyond about 15 feet because most of the radon (half-life 3.8 days) generated deep within the pile decays before it can diffuse to the surface. If it is assumed that sands and slimes are uniformly distributed in the stabilized pile and that tailings and cover contain 4% moisture, it may be estimated that the tailings generated in the years 1975 through 2000 would, after stabilization, release 420,000 curies of ^{222}Rn per year if no recycle were practiced, 376,000 curies per year if uranium only were recycled, and 327,000 curies per year if both plutonium and uranium were recycled. Such release rates would not be realized until long after the year 2000. In the year 2000, most of the tailings would still be under water in tailings ponds at active mill sites and the radon release rate would be considerably lower.

The largest of these numbers is less than 0.2% of the radon released annually from the soil of the conterminous United States; about 0.23% if the tailing piles are not stabilized with 2 ft of earth cover.* This fact, along with the short half-life of ^{222}Rn and the usual location of the tailings piles far from regions of high population density, argues that the radiological effect of radon from the tailings piles on the U.S. population will be small compared with the effect of natural background radon.

Results of measurements of radon concentrations in air near mill tailings piles have been published.[26,27] These measurements suggest that at distances beyond 0.5 mile in the prevailing wind direction radon concentrations are not markedly above the

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*The natural release of radon from the ground in most of the United States appears to be near 1×10^{-12} curies per square meter per second.[26] If this value is assumed to represent the average rate, it is estimated that the conterminous United States (7.6×10^{12} square meters in area) released 2.4×10^{8} curies of radon per year. The 420,000 curies per year estimated to be the release rate from the ultimate tailings piles from the no recycle option for years 1975 through 2000 is 0.18% of this value. It should be noted that natural release rates from soils near uranium deposits may be as much as 100 times the average value noted here.

(relatively high) background concentrations of radon in these regions.[23] From measurements of radon concentration in air near the tailings piles at Grand Junction, Colorado and Salt Lake City, Utah, the Environmental Protection Agency[24] has estimated lung dose from radon to an individual at a 1 kilometer distance from the tailings piles of about 1.5 to 2 times the background dose.

The model tailings pile used in this assessment covers about 290 acres and is estimated to emit, when fully developed and stabilized, 7,500 curies of ^{222}Rn per year. The estimated annual individual 50-year dose commitments* from the ^{222}Rn from the model tailings pile at 500 meters from the pile are listed in Table IV H-8.

The highest dose commitments for the inhalation pathway are received by the bone (0.12 rem), kidney (0.29 rem) and lung (0.17 rem). For the total diet pathway, the highest dose commitments are received by the total body (0.19 rem), bone (0.57 rem), liver (0.13 rem), and kidney (0.64 rem). The estimated annual population dose commitments for the United States from ^{222}Rn released from retired tailings piles accumulated in the years 1975 through 2000 are listed in Table IV H-9* for the three fuel cycle options. The highest dose commitments from all pathways are received by the kidney, liver, total body, bone, and lung. For example, the kidney dose commitment for no recycle is 180,000 person-rem, 170,000 person-rem for uranium only recycle, and 140,000 person-rem for uranium and plutonium recycle. In fact all of the dose estimates for the three fuel cycle options follow the pattern dictated by the gross release of ^{222}Rn listed in Table IV H-6: 420,000 Ci for no recycle, 376,000 Ci for uranium only recycle, and 327,000 for uranium and plutonium recycle. On a gross basis, these values indicate that the effect of radon from the tailings piles would be highest for no recycle, would be reduced by 11% if uranium only recycle were adopted, and by an additional 12% (to a total of 23%) if the uranium and plutonium recycle alternative were used.

Sears et al.,[23] have estimated annual lung doses (using the "smeared lung" model) to individuals resulting from radon releases (8,400 curies per year) from model tailings piles at two particular sites using meteorological conditions specific to those sites. That study estimated that an exposed individual at 0.5 mile from the tailings pile in the prevailing wind direction would receive an annual lung dose of 60 to 100 millirem from emitted radon. Doses from emitted radon to maximally exposed individuals at 1 mile and 50 miles from these model tailings piles would be expected to be less than those at 0.5 mile by factors of about 5 and 1,000, respectively.

The dose commitment from the model tailings piles can be placed in perspective by comparing it to the dose commitment from natural background radon. For example, in the estimate by Sears et al.,[23] a lung dose of 60 mrem/yr was estimated at a distance of

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*These estimates are calculated dose commitments using the so-called "smeared-lung" model. If the dose were assumed to be localized in the bronchial epithelium, as is done by EPA, much larger values would be obtained. However, consistent use of either model would yield the same ratio of doses from emitted radon to doses from background radon.

Table IV H-8

ANNUAL INDIVIDUAL 50-YEAR DOSE COMMITMENT AT 500 METERS FROM MODEL TAILINGS PILE

(millirem)

	Plume	Ground Deposition	Inhalation	Total Diet
Total Body	**	0.82	7	190
G.I. Tract	**	0.82	16	3
Bone	**	0.82	120	570
Liver	**	0.82	51	130
Kidney	**	0.82	290	640
Thyroid	**	0.82	**	**
Lung	3.2	0.82	170	**
Skin	*	0.82	**	**

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*7,500 curies of ^{222}Rn released per year.

**0% contribution for source term considered.

Table IV H-9

ANNUAL POPULATION DOSE (50-YEAR DOSE COMMITMENTS: PERSON-REM) FOR THE UNITED STATES FROM ^{222}Rn RELEASED FROM RETIRED TAILINGS PILES ACCUMULATED IN THE YEARS 1975 THROUGH 2000*,**

	No U or Pu Recycle	U Recycle Only	U and Pu Recycle
Total Body	4.7 E+4	4.3 E+4	3.7 E+4
G.I. Tract	2.0 E+3	1.8 E+3	1.6 E+3
Bone	1.6 E+5	1.4 E+5	1.2 E+5
Liver	3.9 E+4	3.5 E+4	3.0 E+4
Kidney	1.8 E+5	1.7 E+5	1.4 E+5
Thyroid	1.3 E+2	1.2 E+2	1.0 E+2
Lung	1.6 E+4	1.4 E+4	1.3 E+4
Skin	1.3 E+2	1.2 E+2	1.0 E+2

Note: 4.7 E+4 = 4.7×10^4.

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*Assumes the total area of tailings piles in years 1975 through 2000 after stabilization with 2 ft of earth (Table IV H-6).

**Assumes 7.5 people/sq mile in the western United States and 160 people/sq mile in the eastern United States.

0.5 mile. In this case, the radon concentration at 0.5 mile from the stabilized model tailings deposit is 5 times the average background of 0.41 pCi of radon per liter of air measured in three of the four milling sites by the Public Health Service; at 1 mile it is 1.5 times background; and at 5 miles it is only 1.15 times background; and at 50 miles the radon from the tailings pile would be indistinguishable from background radon.

The EPA,[28] Pohl,[29] and Cohen[30] have attempted, by means of various assumptions as to meteorology, dispersion models, and radon dosimetry, to estimate the impact, in terms of health effects (incremental lung cancers), of the radon emitted by mill tailings. Cohen[30] has argued that, even if the assumptions as to the bronchial epithelium lung dose model are correct, the estimated number of health effects is high. In any event, the estimated numbers of health effects from radon releases from the tailings piles obtained using either model are a small fraction of those estimated on similar bases that would be due to natural background radon.

Pohl[29] has pointed out that the presence of ^{230}Th (half-life 83,000 years) in the tailings will ensure that the radon emissions will continue at an essentially undiminished rate for many thousands of years. He has, accordingly, postulated that the health effects of such emissions should be projected over a long time frame. Comey[31] has enlarged upon this theme to show that large numbers of cancer deaths will ultimately result from exposure to the radon released from the tailings piles. However, as pointed out by Cohen,[30] the accuracy of these estimated numbers for health effects is in question and, although some increased exposures to radon will occur, these exposures will be a small fraction of the exposure from naturally occurring (background) radon.

However, the important factor to note in the above assessment is the fact that the dose commitment (by an assumed mechanism) is highest for the no recycle option, the second lowest for the uranium only recycle option, and the lowest for the uranium and plutonium recycle option.

3.1.2 Nontransuranium Waste[32,33]

Nontransuranium wastes are defined for this assessment as low level waste materials that contain less than a specified concentration of plutonium or other long-lived transuranium radionuclides* and may also contain beta and gamma emitting radionuclides. These low level wastes are shipped to commercial licensed burial facilities for disposal. The proposed amendment to 10 CFR Part 20[34] would prohibit future burial of wastes containing more than a specified limit of transuranium materials in commercial burial facilities. A program to establish this limit is in progress by the NRC.[35] Nontransuranium wastes may include laboratory wastes, contaminated tools, plastic bags, paper towels, protective clothing, solidified liquid wastes, ion exchange resins, and filters.

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*The specified concentration limit for nontransuranium materials has not been established but an upper limit of 10 nCi/g of waste was indicated in the proposed amendment to 10 CFR Part 20.[34]

These low level wastes are generally compacted and sealed in 55-gal drums for shipment to the burial facility. The fuel cycle operations that normally produce nontransuranium wastes are UF_6 conversion, enrichment, uranium fuel fabrication, LWR reactors and spent fuel storage.

3.1.2.1 Burial Facility

Six commercial facilities are currently licensed to receive and bury nontransuranium wastes. Five sites are owned by states; the other site is owned by the Federal Government and is leased to a state. Wastes are buried on land owned by a state or the Federal Government to assure long term control of the site. Authorization to operate a commercial burial facility is granted after an analysis of the nature and location of facilities and examination of topographical, geographical, meteorological, and hydrological characteristics of the site to ensure that the migration of wastes is unlikely. No credit is taken for containment provided by the packages once they are buried. The purpose of the packages is to provide containment and ease of handling during interim storage and shipment and, in some cases, to minimize the exposure of operating personnel to radiation through the use of shielded shipping containers.

Commercial burial sites are located in sparsely occupied areas in the States of Washington, Nevada, Illinois, Kentucky, South Carolina, and New York. The general site areas are isolated to limit public access, and the burial areas are enclosed by fences. Access to the site is controlled by the site operator.

The model burial facility for nontransuranium wastes occupies a land area of 100 acres. The site characteristics are such that the groundwater level is well below the bottom of the deepest trench. The location is selected to have soil that will provide good ion exchange characteristics to minimize the movement of radioactive materials to the groundwater via leaching of the solid wastes by rainwater. It is assumed that there is no nearby use of groundwater or wellwater downgradient from the site. Water movement through the soil at the site and its vicinity is assumed to be slow and in a direction in which there is little or no land use. Buildings provide space for offices, a laboratory, temporary storage of packaged radioactive wastes, and other waste handling as may be necessary, such as solidification of liquids containing low concentrations of radioactivity.

The primary operations at a licensed commercial burial facility are receipt, temporary storage, and burial of packaged radioactive wastes. Packages must meet the requirements of the Department of Transportation (DOT) and the NRC for transportation of radioactive materials. Packages are normally buried as received. However, in some cases the primary package containing the waste is shipped in a reusable overpack or secondary container which may be required by DOT regulations for shipment of some materials. In those instances, the primary package may be removed from the reusable overpack before burial. The operator must maintain an environmental monitoring program that includes sampling of air, water, and vegetation to determine if radioactive materials are migrating to the offsite environment.

The model burial trench is 91 m long, 9 m wide, and 6 m deep, and has a volume of 4,900 m^3. Assuming a close packing of 55-gal drums in rows 10 wide and 5 deep, about 3,500 m^3 of waste can be buried in the model trench. Assuming 2 burial trenches per acre, a burial site would provide for disposal of about 700,000 m^3 of waste per 100 acres.

Packages in partially filled trenches are frequently covered over with earth to minimize access, provide shielding during placement of other packages in the trenches, and minimize entry of water into the trench. Temporary storage is minimized. Packages are placed in a trench and covered with earth as soon after receipt as possible. Completed trenches are maintained to minimize erosion, and marked to specify the contents.

The volume of nontransuranium low level waste produced annually will increase in proportion to the growth of power production and it is estimated that a total of about 300,000 m^3 will be produced in the year 2000. About 45 acres of commercial burial ground will be required for this waste. The total volume of transuranium waste accumulated for each fuel cycle operation in the period 1975 through year 2000, and the totals for each fuel cycle option, are listed in Table IV H-10. Each option shows a total low level waste accumulation of about 4 million m^3. The variation between the options is less than 3% which is not significant within the accuracy of the data. It is apparent that the major contribution to the total is the low level waste from the reactors, and that this waste volume is independent of the fuel cycle option. Thus, about 600 acres of commercial burial ground will be required for each option through the year 2000.

Table IV H-10

ESTIMATED VOLUMES OF NONTRANSURANIUM WASTES GENERATED IN THE FUEL CYCLE OPTIONS IN THE YEARS 1975 THROUGH 2000

(cubic meters)

	No Recycle	U Recycle	Pu and U Recycle
Enrichment	65,000	65,000	22,000
U Fuel Fab.	154,000	154,000	133,000
Reactors	3,800,000	3,800,000	3,800,000
Spent Fuel Storage	3,350	1,680	1,000
Reprocessing	-	23,000	23,000
TOTAL (rounded)	4,290,000	4,280,000	4,180,000

3.1.2.2 Environmental Impact of Burial Facilities

The environmental impacts associated with waste burial facilities are dust and noise during construction of the offices and laboratory, periodic excavation and filling of trenches, the presence of an exclusion fence around the area, the permanent commitment of land, and the possibility of the movement of small quantities of activity into the groundwater. The release of heat to the environment is not a problem because of the very low heat generation rate of nontransuranium wastes.

Monitoring wells maintained at each burial site are sampled periodically to check for migration of radioactivity. In addition, air and vegetation samples are taken around the site for analysis. Should any sample reveal an increase in the concentration of radioactive material above that measured before burial operations, and it is determined that the radioactivity originated in the burial ground, its significance must be analyzed, corrective actions developed, and the matter reported to the licensing agency. Several courses of actions may be taken if significant migration is detected. These include: a halt to burial operations, removal of the radioactive material from the burial area in which the radioactivity orginated, grouting of the site from which the radioactivity originated, other procedures depending on the mode of migration. Corrective actions are subject to approval by the licensing agency.

A movement of radioactive materials from commercial burial facilities is not expected during normal operations based on siting criteria submitted with license applications. However, recent surveys of commercial burial facilities,[36,37] have shown some movement of radioactive materials to the area surrounding one of the sites. An analysis by the NRC[35] indicated that "(1) although there has been, to date, no adverse effect on public health or safety from any of the existing sites, the operation and performance of the sites has not been entirely satisfactory; (2) although the AEC (and now the NRC) has had an active program for regulation of commercial burial sites - there is a need for NRC to upgrade that program...." A reassessment of the waste management regulation program by the NRC is in progress. Initial conclusions[35] indicate that "combinations of improved site engineering, waste management, and packaging and solidification of wastes can minimize migration from the site." The benefits and disadvantages of potential modifications of current practices have been presented in a survey by ERDA.[33]

3.1.2.3 Environmental Effects of Accidents at Commercial Burial Facilities

Accidental opening of a waste package prior to burial and release of the package contents could occur at a commercial burial facility. Normally, the scattered waste materials would be confined to the immediate vicinity of the accident within the fenced facility and the impact on the environment would be negligible. This is particularly true of wastes which have been solidified by incorportation in cement, urea formaldehyde or bitumen. The waste materials can be recovered and repackaged in drums. If it is assumed that the waste is readily dispersed and as much as one-third of the waste contained in a 55-gal drum is released to the nearby unrestricted area, the effect on the environment would be insignificant. The volume of a 55-gal drum is approximately 7.4 ft^3. Using the assumption that the specific activity of the waste is 0.2 Ci/ft^3, the total amount of dispersed radioactivity would be about 500 mCi.

If the waste were uniformly dispersed over an area of 1 acre, the concentration of the radioactive material would be about 11 $\mu Ci/ft^2$. This concentration of radioactive materials is low and would be approximately 3 times higher than that in a 1-cm-thick layer of uranium ore.* Dispersion over a greater area would result in a proportionally lower concentration. The dispersed material would be in the vicinity of the site and could be recovered and repackaged.

Another means of accidental release of radioactive materials from a burial facility would be by migration through the soil into the groundwater. A limited movement of radioactive materials has occurred at one commercial site (see paragraph 3.1.2.2) and the NRC is currently reviewing the operating criteria for commercial burial facilities. Based on the results of years of experience at the current commercial sites, large movements of radioactive materials are unlikely.

3.2 Geologic Repositories

Geologic isolation, which is the emplacement of wastes in naturally occurring formations or media, offers two options, viz., storage or disposal. Storage implies that wastes could be readily retrieved after emplacement while disposal implies they could be retrieved only with great difficulty. Six types of media have been identified as candidates for isolation of radioactive wastes: ice sheets, sea floor clays, limestone, salt (domes or beds), clay derived minerals (e.g., shale), and crystalline rocks (e.g., granite). Storage or disposal relies on the medium to serve as a stable barrier that keeps the wastes isolated from the biosphere. The relative merits of each medium are described in Reference 39. It is necessary that the formation in which the geologic facility is placed be free of groundwater, be located in an area of tectonic stability, and have the capacity to absorb radiation and diffuse heat without impairing integrity of the formation and without requiring uneconomical spacing of waste. In addition, the location of the facility should be such as to minimize any effects associated with conservation and use of valuable mineral resources. The geologic barrier can be augmented by selecting waste forms, containers, and engineered structures that contribute to the stability of the system.

Design of geologic isolation repositories is specific to the site, medium, and the characteristics of the wastes and containers to be emplaced. Factors considered in the design of waste handling facilities and emplacement configurations include size and shape of the waste containers; heat and radiation emissions from the waste; chemical, thermal, mechanical, and radiation interactions between the waste and the geologic medium; and requirements for operational safety. Reference 39 reviews these and other factors.

Disposal of radioactive wastes in suitable geologic formations on land is considered the most likely method to be used and is assumed herein. The study of geologic containment systems over the years has focused on use of deposits of bedded salt, but several alternatives have been considered.[39]

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*The hazard index of uranium ore is about 10 times higher than that of the accumulated nontransuranium wastes on a gross cure basis because of the long-lived alpha nuclides present in the ore.[38]

Since the most likely agent for dispersal of the radionuclides from deeply buried wastes is water, the most suitable geologic environment would be one that maximizes the integrity of the barrier separating the waste from circulating groundwater. Consideration of these factors led a committee of the National Academy of Sciences - National Research Council (NAS-NRC) to conclude that natural salt deposits represent the most promising geologic environment.[40] Subsequent reviews by NAS-NRC committees[41,42,43] have continued to endorse bedded salt as the most promising geologic medium for waste disposal.

Salt has a variety of properties that make it attractive. Unlike many other rocks, salt is free of circulating groundwater as attested by the existence of extensive salt formations that were deposited hundreds of million years ago. Thick salt deposits are relatively common geologic features in the United States. The property of rock salt primarily responsible for this impermeability is its high plasticity. When initially deposited, salt can be as porous and permeable as a coarse sandstone. As the newly formed deposit becomes slightly buried, the individual grains yield plastically, deform, and recrystallize. This collapses the pores and squeezes out the brine (although a small amount is trapped as intracrystalline inclusions, generally about 0.1 mm in diameter), resulting in the formation of a massive, virtually impermeable, polycrystalline rock. Many rock types respond to tectonic stresses and consequent deformation in a brittle manner and develop an interconnected pattern of fractures that eventually become channels through which groundwater circulates. Salt formations, because of their high plasticity, yield and flow while maintaining their massive, impermeable character. Even if a salt formation should fail by fracturing (as in faulting), the fracture will subsequently be healed by plastic deformation, once again preserving the imperviousness of the salt to circulating groundwater. This feature can be seen occasionally in salt mines.[44] In addition to desirable mechanical properties, salt has a thermal conductivity that is higher than that of many other rocks; this is advantageous in removing radioactive decay heat from the waste and in preventing an excessive rise in temperature. In addition, salt is relatively stable to radiation[45] and most salt deposits are located in geologically stable regions typified by slow and gradual deformations.

Nearly all work on methods of high level waste disposal in the United States has been done upon bedded salt deposits, though original considerations[40] also included salt domes or anticlines. As is the case with bedded salt, site selection for domes must be based on site specific characteristics. Domes result from tectonic instability and massive volumes of salt have penetrated the rock sequence above deeply buried source salt beds. Therefore, such domes are not usually protected from groundwater by thick impermeable rock sequences, and it would be necessary to establish in detail the groundwater flow as well as to demonstrate that the formation process (diaspirism) is not now active and that its reinitiation is highly unlikely.[45]

While much of the reported work on bedded salt disposal has been conducted in the United States, other countries have also found salt attractive. In West Germany, plans are far advanced for use of a salt mine for disposal of high level wastes.[46] However, their focus is in domed salt.

Of the deep continental geologic formations suitable for isolation of radioactive waste, bedded salt is used for repositories in this study. A model facility is described in the following paragraphs.

3.2.1 Model Geologic Repository

The model Federal repository for the disposal of high level and transuranium wastes is described briefly in the following. Detailed descriptions are presented in References 39 and 47. The model repository is located in bedded salt and the principal areas are designated as the high level mine for the high level wastes and alpha mine for the transuranium wastes. A conceptual drawing of the repository is shown in Figure IV H-1. Both high level and transuranium wastes contain long-lived transuranium radionuclides which require isolation from the environment for hundreds of thousands of years. In addition, the wastes contain shorter-lived fission products which require isolation for thousands of years. Consequently, similar repository characteristics are required for both types of waste. The high level wastes include spent fuels and solidified high level waste, and the transuranium wastes include solidified plutonium waste (from the uranium recycle option), fuel element hulls and hardware, and miscellaneous wastes from fuel reprocessing plants and mixed oxide fuel fabrication plants.

The criteria established for selecting a site for a waste repository include conservative geologic features to provide assurance of long term isolation and efficiency of operations. These criteria provide that the salt formation be of the bedded type, extending several tens of miles horizontally; be at least 200 ft thick; be located at a depth between 500 and 2,000 ft; and not be associated with significant known reserves of petroleum or other mineral resources. In addition, the criteria require that the site be in a zone of tectonic stability. Safety and radiation controls will conform to the NRC licensing requirements.

Thermal emissions from the waste will heat the surrounding mineral deposit and adjacent formations. A slight differential expansion of strata will probably occur. To minimize adverse effects from this expansion, the heat sources would be placed as far as possible from the shaft locations. Thermal effects that influence the allowable temperature rises and heat release rates include:

- Thermal stability of the waste
- Thermal stability of the formation
- Migration of water contained in the pores or small cavities in the formation
- Structural integrity of the entire formation

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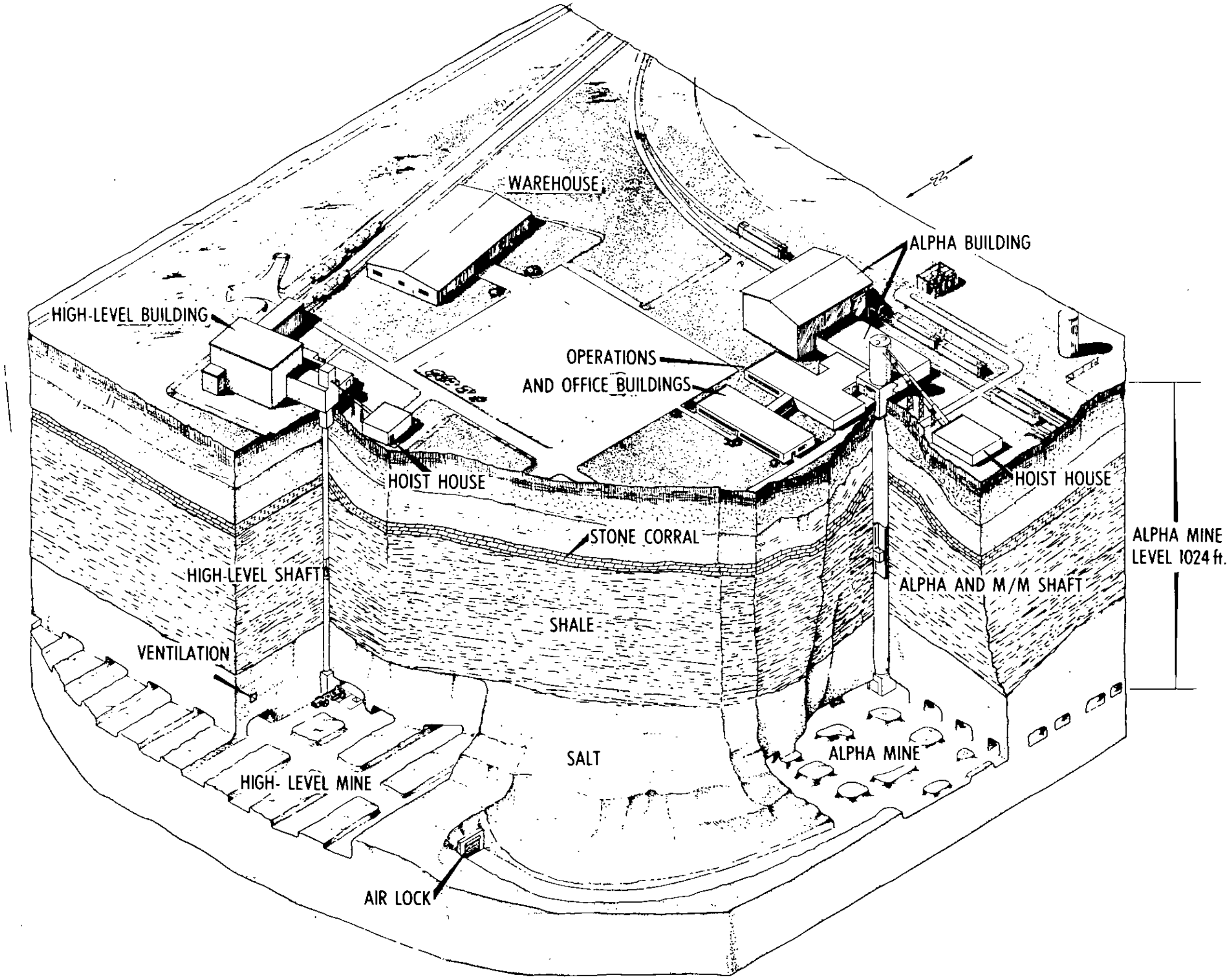

Figure IV H-1 Model Federal Repository for Transuranium (Alpha) and High Level Wastes (Artist's Concept)

- Temperature rise in any nearby fresh water aquifers
- Heating of the earth's surface
- Temperature increases beyond the boundaries of the disposal area

Establishment of thermal criteria that limit adverse thermal effects is dependent on the particular formation and is not a straightforward and simple procedure. Thermal criteria for bedded salt in the form of maximum permissible temperatures were developed by Cheverton and Turner.[48] For any emplacement geometry that affords a reasonable distribution of the heat, they found that a heat load as high as about 150 kW/acre (37 W/m^2) could be used as a design basis.

3.2.1.1 Transuranium Waste Repository

The transuranium repository complex consists of an alpha waste receiving building, mine operations building, administration building, hoist house, and a mine air supply fan house. The alpha waste receiving building contains the necessary facilities and equipment to unload, inspect, and transport the waste to the alpha man- and materials-(M&M) shaft for lowering into the mine. Except for container unloading and storage, all the transuranium waste handling areas are in a confined structure. The confined areas are constructed of reinforced concrete and are designed to permit safe shutdown of waste handling operations during or following an earthquake or tornado. Confined areas are maintained at a negative static air pressure with respect to the outside atmosphere to prevent outleakage of air. Exhaust air from confined areas of the building is passed through fans and HEPA filters located in a basement exhaust fan-filter room prior to discharge to the atmosphere through a stack. Access into confined areas of the building is through air locks.

Alpha M&M Shaft. The alpha and M&M (man and material) shaft, located between the alpha (transuranium) waste receiving building and the mine operations building is a multi-purpose shaft and is used for: lowering transuranium waste containers into the mine, access to the mine for men, materials, and equipment, and supply of fresh air to the mine.

Alpha Mine. The mine consists of rooms that are used for disposal of the transuranium waste containers. The waste containers will be lowered into the mine, transferred to the disposal rooms and placed in stacked rows, and salt will be used to backfill the rows and fill any unused space within the burial room.

3.2.1.2 High Level Waste Repository

The high level waste repository provides the facilities and equipment necessary to receive and handle packages of high level waste shipped to the repository in shielded shipping casks. Waste packages will consist of solidified high level wastes sealed in stainless steel canisters about 1 ft in diameter and 10 ft long. The packages are unloaded individually in shielded cell facilities, lowered to mine level through the high level shaft, and then transported and buried in the salt bed using a shielded transporter. The components of the high level facility include the high level waste receiving building, the hoist house, and exhaust fan filter house at surface level, and the high level shaft, the mine level receiving station, and the high level mine. The process waste system used to concentrate, solidify, and package any radioactive liquid waste originating at the repository will also be located within the high level facility.

The basic architectural and structural guidelines used for the alpha waste receiving building are also used for the high level waste receiving building. Differences between the handling of high level waste packages and alpha waste burial units result from the presence of penetrating radiation in the high level waste. Therefore, all handling operations involving waste packages or opened shipping casks containing waste packages must be conducted remotely behind massive shielding. In addition, the waste packages themselves represent significant heat sources. The areas within the high level waste receiving building are designed to permit safe shutdown of waste handling operations during or after an earthquake or a tornado. Exhaust air from confined areas of the building will be passed through fans and HEPA filters located in an adjacent exhaust fan-filter house.

The head frame for the high level shaft will be an integral part of the high level waste receiving building and is designed to provide radioactive confinement and biological shielding.

The high level waste receiving building contains approximately 18,300 ft^2 of building area, including the basement and the structure for a shielded recanning cell and transfer gallery. Equipment designed to recan high level waste packages that have lost their initial integrity during transport to the repository would be installed in this cell.

High Level Shaft. The high level shaft is used only to transport the waste packages from the high level waste receiving building to the mine. The bottom terminates in a shielded receiving station where the packages are transferred to a shielded transporter for delivery to the burial rooms.

High Level Mine. The mine consists of a conventional room and pillar excavation in which the waste packages are buried in vertical holes bored in the mine floor. After the package is in place, the space above the package is filled with salt to provide shielding during subsequent operations.

Process Waste System. The process waste system equipment in the high level facility is used to decontaminate any radioactive liquid wastes generated in the operation of the repository and reduce the wastes to a solid form suitable for disposal in the repository. There are no discharges of liquids containing radioactive materials from the repository.

Design Parameters. Several important design parameters for the disposal of wastes in repositories are listed in Table IV H-11. It is assumed that the high level waste containers will be about 12 inches in diameter by 10 ft long and will contain the solidified waste from 3.14 metric tons of fuel. The high level waste containers will have a maximum heat generation rate of 5 kW. This is an average number which allows for variations in cooling time and fuel cycle alternative. See Tables IV H-12 and IV H-13. The heat generation rates for the high level waste decayed ten years were used to determine the subsurface acreage required for waste disposal for the various fuel cycle options. The annual capacity of the repository is 360 m^3 for high level waste, 2,100 m^3 (15,000 fuel assemblies) for spent fuel, and 6,000 m^3 for transuranium waste. Operations in the first repository will be initiated in the late 1980's; a second repository will be required by about 1990 to handle the amounts of wastes that will be generated annually at that time. The second repository will be required about year 1990 for the uranium only recycle and the uranium and plutonium recycle options, and about year 1995 for the no recycle option. Many specifications, such as type of waste containers, have not been established for transuranium wastes at the present time.

3.2.1.3 Effluents During Normal Operations

Small quantities of radioactive and other materials are released to the atmosphere as a result of the normal operations of the repository. The only significant source for the release of radioactive materials is the gaseous effluent which will contain small amounts of radioactive particles after having passed through multiple HEPA filters. No liquid wastes are released from the repository. Solid radioactive wastes generated at the repository are packaged and buried in the repository. Aqueous wastes are recycled and any excess water will be evaporated and released to the atmosphere. Instrument systems are provided for continuous monitoring of radiation levels and concentrations of airborne materials in work areas, in stack effluents, and at the boundary of the site.

High Level Waste Receiving Building. Equipment and surfaces in the confined area of the high level receiving building will develop surface contamination, and a small fraction of the radioactive material will become airborne by resuspension. See Table IV H-14. The filtered (HEPA) effluent released from the stack of this facility may have concentrations estimated as no greater than 5×10^{-11} curies/cubic meter. The contained radioactivity will consist of oxides of mixed fission products and actinides.

Alpha Waste Receiving Building. Operations in the alpha waste receiving building are expected to result in the release of negligible quantities of activity to the building ventilation system. See Table IV H-14. Periodic decontamination procedures will be employed to assure that surface contamination levels do not build up to unacceptable levels. Ventilation air from this building will pass through HEPA filters before discharge to the atmosphere through a short stack.

Table IV H-11

DESIGN PARAMETERS FOR THE DISPOSAL OF WASTES IN REPOSITORIES

	HLW	TRU
Container size	12 in. diam x 10 ft	*
Container capacity	6.3 ft^3**	*
Container heat	5 kw***	*
Annual waste capacity	360 m^3[†] (2,060 m^3)[††]	6,000 m^3
Surface area committed	250 acres	

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*Final specifications have not been established for TRU wastes.

**The solidified waste from reprocessing 3.14 metric tons of fuel.

***Tables IV H-12 and IV H-13 indicate that the heat output per container varies from 3.3 kw to 6.5 kw, depending on cooling time and fuel cycle alternative.

[†] 2,000 containers.

[††] 15,000 spent fuel assemblies.

Table IV H-12

ESTIMATED HEAT OUTPUT IN SPENT FUEL AND HIGH LEVEL WASTE (5-YEAR DECAY)

Basis: 33,000 MWd/MTHM; 30 MW/MTHM

Spent Fuel and Waste Aged 5 Years

	Watts/MTHM		
Isotope	No Uranium or* Plutonium Recycle	Uranium Recycle** Only	Uranium and**,*** Plutonium Recycle
^{237}Np	0.010	0.018	0.017
^{238}Pu	93.4	3.62	6.87
^{239}Pu	10.0	0.050	0.058
^{240}Pu	14.9	1.11	0.28
^{241}Pu	3.44	0.017	0.025
^{242}Pu	0.041	0.0002	0.0004
^{241}Am	27.7	5.37	9.85
^{243}Am	0.67	0.66	2.15
^{242}Cm	0.80	0.54	1.41
^{244}Cm	71.4	68.7	365
Fission Products	1,870	1,690	1,680
Total	2,090	1,770	2,065
Watts per Canister[†]	950	5,550	6,500

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*Heat output of spent fuel.

**0.5% loss of Pu and U to waste at 160 days.

***On the basis that fuels processed are 11% mixed oxide; remainder is enriched UO_2 fuel.

[†] On the basis of 2 ft^3 of waste per metric ton, 6.3 ft^3 per canister or one PWR fuel assembly per canister.

Table IV H-13

ESTIMATED HEAT OUTPUT IN SPENT FUEL AND HIGH LEVEL WASTE (10-YEAR DECAY)

Basis: 33,000 MWd/MTHM

Spent Fuel and Waste Aged 10 Years

	Watts/MTHM		
Isotope	No Uranium or* Plutonium Recycle	Uranium Recycle** Only	Uranium and**,*** Plutonium Recycle
^{237}Np	0.010	0.018	0.017
^{238}Pu	89.4	3.52	6.64
^{239}Pu	10.0	0.050	0.058
^{240}Pu	14.9	0.14	0.43
^{241}Pu	2.71	0.013	0.020
^{242}Pu	0.041	0.0002	0.0004
^{241}Am	47.1	5.42	9.91
^{243}Am	0.66	0.66	2.14
^{242}Cm	0.265	0.26	0.83
^{244}Cm	58.3	56.7	302
Fission Products	1,040	1,010	992
Total	1,260	1,080	1,310
Watts per Canister^{+}	580	3,400	4,100

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*Heat output of spent fuel.

**0.5% loss of Pu and U to waste at 160 days.

***On the basis that fuels processed are 11% mixed oxide; remainder is enriched UO_2 fuel.

$^{+}$On the basis of 2 ft^3 of waste per metric ton, 6.3 ft^3 per canister or one PWR fuel assembly per canister.

Table IV H-14

ESTIMATED AIRBORNE CONCENTRATIONS OF RADIOACTIVE MATERIALS IN SURFACE FACILITIES AND AVERAGE ANNUAL RATE OF DISCHARGE OF THESE MATERIALS TO THE ATMOSPHERE

High Level Receiving Cell	
Concentration of airborne waste particles*	1×10^{-7} Ci/m^3
Concentration in filtered effluent**	5×10^{-11} Ci/m^3
Average annual discharge to atmosphere	0.007 Ci/yr
High Level Facility Confinement Zones	
Concentration of airborne waste particles*	2×10^{-11} Ci/m^3
Fraction of 10 CFR Part 20, Table I, Column 1	0.02
Concentration in filtered effluent**	1×10^{-14} Ci/m^3
Average annual discharge to atmosphere	0.9 μCi/yr
Alpha Waste Unloading and Transfer Rooms	
Concentration of airborne waste particles*	2×10^{-13} αCi/m^3
Fraction of 10 CFR Part 20, Table I, Column 1	0.2
Concentration in filtered effluent**	1×10^{-16} αCi/m^3
Average annual discharge to atmosphere	0.0009 α μCi/yr

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*These are the estimated maximum concentrations that will be maintained in these areas by routine decontamination operations. It has been found from experience that the air concentration is about 4×10^{-5} Ci/m^3 per Ci/m^2 of average surface contamination in work areas.

**Assumes a filter efficiency of 99.95%.

High Level Mine. Ventilation air from sections of the high level mine that have been exposed to waste canisters is blended with the air from mining operations and the alpha mine, and exhausted to the atmosphere through HEPA filters and a short stack. This air contains small concentrations of radioactive particles and gases, chemically noxious gases, and salt particles. Estimated concentrations of these materials in the mine air and the maximum average annual release to the atmosphere (after HEPA filtration) are presented in Table IV H-15.

Burial of waste containers below 7-1/2 ft of crushed salt in rooms, progressive backfilling of the utilized rooms and corridors, and eventually, backfilling of the entire mine and sealing of the shafts will provide assurance that once buried, virtually no radioactive particles can be transported from the wastes to the mine ventilation air and, thus, to the surface. It is estimated that all gases produced in the mine will eventually be discharged to the surface but that particulate activity will not be released after the mine is sealed and decommissioned.

The maximum concentration of radioactive particles in the mine ventilation air is estimated as 2×10^{-11} curies/cubic meter. This would result from resuspension of radioactive materials that may eventually deposit on surfaces exposed to the ventilation streams.

Table IV H-15

ESTIMATED AIRBORNE CONCENTRATIONS AND AVERAGE ANNUAL RATE OF DISCHARGE OF RADIOACTIVE AND NONRADIOACTIVE MATERIALS FROM UNDERGROUND FACILITIES

Material	High Level Mine* Concentration in Mine Exhaust Air	Alpha Mine* Concentration in Mine Exhaust Air	Total Average Annual Release to Atmosphere After HEPA Filtration
High level waste particles	2×10^{-11} Ci/m^3	-	6 μCi/yr
Alpha waste particles**	-	2×10^{-13} Ci/m^3	0.03 μCi/yr
^{85}Kr	2.3×10^{-11} Ci/m^3	5×10^{-14} Ci/m^3	0.014 Ci/yr
^{3}H	1.5×10^{-12} Ci/m^3	3×10^{-15} Ci/m^3	0.0009 Ci/yr
^{222}Rn	1×10^{-9} Ci/m^3	1×10^{-9} Ci/m^3	0.9 Ci/yr
^{220}Rn	5×10^{-11} Ci/m^3	5×10^{-11} Ci/m^3	0.04 Ci/yr
H_2 (corrosion, radiolysis, electrolysis)	90 ppm	20 ppm	4. scfm
He (alpha decay)	0.04 ppm	0.001 ppm	0.001 scfm
HCl (brine decomposition)	2 ppm	-	0.07 scfm
CO_2	1,100 ppm	300 ppm	50 scfm
CO	1 ppm	-	0.05 scfm
NO_2	1 ppm	-	0.05 scfm
SO_2	0.7 ppm	-	0.03 scfm
HCHO	0.02 ppm	-	0.0007 scfm
Soot	3 mg/m^3	-	2 lbs/yr
Salt particles	5 mg/m^3	5 mg/m^3	5 lbs/yr

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*These concentrations occur simultaneously only in the air exhaust tunnel, which is not normally occupied.
**Expressed in terms of alpha curies.

The rate of release of gases from the mine is determined by their rate of production, the rate of temperature rise, and the rate of convergence of the mine. It is assumed that the waste containers will not be perforated (by stress corrosion cracking) for at least 90 days after emplacement and that they will not corrode at a rate exceeding 10% per year. Disposal rooms will have been backfilled with crushed salt (leaving about 25 volume % air) before the containers begin to leak gas. Following perforation of the containers, gases (Xe, Kr, Rn, He, and H_2) produced by spontaneous fission and decay of actinides (principally ^{244}Cm) are expected to migrate slowly from the storage hole into the backfilled room with a half time of several days. These gases, as well as small volumes of H_2 and HCl from radiolysis and decomposition of the brine, will migrate from the room into active air corridors. Following release to the ventilation air stream, the gases will be discharged to the atmosphere through the mine exhaust. The time required for migration of gases through the backfilled room to the ventilation stream is such that ^{85}Kr and ^{3}H (half-lives 10.76 and 12.26, respectively) will be the only radioisotopes that are released in measurable concentrations.

It is estimated that the air discharged from the mine will contain radon isotopes, principally from natural sources, at concentrations of about 1×10^{-9} curies/cubic meter. This concentration is about 10 times the concentration that occurs naturally in the atmosphere but is only 3% of the permissible concentration for occupational exposure. See 10 CFR Part 20. These isotopes (^{222}Rn and ^{220}Rn) occur naturally in most subsurface deposits and are released in all mining operations.

Fumes from diesel equipment in the mine and an aerosol of salt will be discharged from the mine in the ventilation air. The total amount of purge air used to dilute the diesel exhaust gases is in conformance with Schedule 24 (Diesel Equipment for Nongassy, Noncoal Mines) of the U.S. Bureau of Mines. Practically all of the soot from the diesel exhaust and salt that has become airborne in the mining operations will be removed in the mine exhaust filters.

Alpha Mine. The air in the alpha mine may contain small concentrations of radioactive particles and gases, chemically noxious gases and salt particles. See Table IV H-15. Concentrations of radioactive particles (containing plutonium) in mine ventilation air will be maintained at less than 20% of the permissible concentrations for occupational exposure. The gaseous radioactivity will consist primarily of radon from natural sources.

3.2.2 Environmental Impact of Geologic Repositories

The land area committed for each bedded salt repository would be 250 acres and a total of 500 acres would be required for the two repositories envisaged to be in operation in the year 2000. However, exploration or exploitation of minerals in areas near the repositories would be preempted; thus, the relative mineral value of the permanent underground area of a repository must be a factor in selection of a repository site.[49] Following decommissioning of a repository and sealing of the mine, the site and certain subsurface mineral rights in a one-mile buffer zone bordering the repository would be maintained in the perpetual care of governments with permanent markers appropriately placed to warn of the potential hazard of drilling operations.

Very small concentrations of radioactive and other materials will be released to the atmosphere as a result of the normal operations of a repository. The concentrations of these materials in normal working areas and offsite will be well below concentrations that are considered acceptable for occupational exposure and for exposure of the general public, respectively.

The materials that will become airborne in small concentrations include solid waste particles resulting from resuspension of surface contamination; gases (e.g., ^{85}Kr, ^{3}H, and He) that result from decay of the waste; radon isotopes that occur naturally in the mine; nonradioactive gases (H_2 and HCl) that are formed by corrosion, electrolysis, radiolysis, and decomposition of brine; exhaust gases from the diesel-powered transporter; and salt particles resulting from mine operations. Sources for occasional releases to the cell air would be material from receipt of contaminated casks, decontamination of contaminated canisters, and miscellaneous cleanup operations. Since none of the operating areas will be occupied full time, the radiation exposure of workers would be limited to low levels. With the projected shielding, the annual radiation dose to approximately 300 workers is judged to be <30 person-rem. The air discharged from the mine will contain radon isotopes, principally from natural sources, at an estimated concentration of about 1×10^{-9} Ci/m^3. This concentration is about 10 times the concentration that occurs naturally in the atmosphere but is only 3% of the permissible concentration for occupational exposure. These isotopes (^{220}Rn and ^{222}Rn) occur naturally in most subsurface deposits and are released in all types of mining operations.

Estimated release rates and average offsite concentrations of radioactive and other materials resulting from full scale operation of two repositories are presented in Table IV H-16. Similar releases would continue beyond the year 2000 until the repositories were sealed. The average annual offsite concentration of airborne materials that originate in the repository is only a small fraction of 0.1% of the applicable standards for exposure (Table IV H-16). The inhalation hazards from the radioactive materials that originate from the radioactive material placed in the repository are small as compared with the radon that occurs naturally in the atmosphere. One repository is envisaged to begin operation in the late 1980's. A second would be required about the year 1990 for the option involving recycle of uranium only or recycle of uranium and plutonium. With no recycle, the second repository would be needed about the year 1995. Thus at the end of the year 2000, there would be up to 27 years of accumulated repository operation for each of the two recycle options but only 22 years for the no recycle option. The total releases from the repositories over 27 years of operation (the maximum values) are given in Table IV H-16; these values would be approximately 20% lower for no recycle.

By prorating the 0.007 Ci annual release among the radionuclides contained in spent fuel and high level waste after 5 years decay (see Table IV H-17) and by using

Table IV H-16

ESTIMATED RELEASE RATES AND AVERAGE ANNUAL OFFSITE CONCENTRATIONS OF RADIOACTIVE AND OTHER MATERIALS RESULTING FROM FULL SCALE OPERATION OF TWO REPOSITORIES

Material	Average Annual Rate of Release to Atmosphere	Total Release* for the Years 1975 through 2000	Average Annual Offsite Concentrations Resulting from Repository Effluents**	Percent of Applicable Standards for Exposure of the Public***
High level waste particles	0.007 Ci/yr	0.189 Ci	2×10^{-15} Ci/m^3	0.02
Alpha waste particles	0.03 μCi/yr	0.81 μCi	1×10^{-20} Ci/m^3	0.0001
^{85}Kr (spontaneous fission)	0.014 Ci/yr	0.378 Ci	4×10^{-15} Ci/m^3	0.000004
^{3}H (spontaneous fission)	0.0009 Ci/yr	0.024 Ci	3×10^{-16} Ci/m^3	0.0000004
^{222}Rn (natural sources)	0.9 Ci/yr	24.3 Ci	2×10^{-13} Ci/m^3	0.02
^{220}Rn (natural sources)	0.04 Ci/yr	1.08 Ci	1×10^{-14} Ci/m^3	0.0003
H_2 (corrosion, radiolysis, electrolysis)	4 scfm	144.7 MT	0.02 ppm	0.0002
He (alpha decay)	0.001 scfm	0.07 MT	0.000004 ppm	4×10^{-8}
HCl (brine decomposition)	0.07 scfm	45.8 MT	0.0003 ppm	0.006
CO_2 (diesel exhaust)	50 scfm	39,470 MT†	0.2 ppm	0.004
CO (diesel exhaust)	0.05 scfm	25.2 MT	0.0002 ppm	0.0004
NO_2 (diesel exhaust)	0.05 scfm	41.2 MT	0.0002 ppm	0.004
SO_2 (diesel exhaust)	0.03 scfm	34.5 MT	0.0001 ppm	0.002
CH_2O (diesel exhaust)	0.0007 scfm	0.38 MT	0.000003 ppm	0.00003
Soot (diesel exhaust)	2 lb/yr	0.025 MT	0.1 μg/m^3	0.007
Salt particles	5 lb/yr	0.061 MT	0.2 μg/m^3	0.001

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*Assumes one repository operates for 16 years (1985 through 2000) and the other operates for 11 years (1990 through 2000) to provide a total of 27 repository-years of operation, including initial pilot plant test period. Similar releases would continue beyond the year 2000 until the repositories were sealed.

**These are the maximum concentrations which result at the fencepost of the site.

***Based on one-third of limits in 10 CFR Part 20, Table II, Column 1 for radionuclides and threshold limit values (American Conference of Governmental Industrial Hygienists) for other materials.

†CO_2 is a normal constituent of air and is not considered a pollutant.

Table IV H-17

ESTIMATED AMOUNTS OF RADIOACTIVE MATERIALS IN SPENT FUELS AND HIGH LEVEL WASTES DECAYED 5 YEARS (Ci/MTHM)*

Principal radionuclides	Half-life	No U or Pu** recycle	Uranium*** recycle	U and Pu (11% MOX + 89% UO_2) recycle***,†
Volatile:				
^{3}H	12 y	4.03E+2		
^{85}Kr	10.4 y	8.21E+3		
^{129}I	1.6E+7 y	3.74E-2	4.9E-5	5.07E-5
Semi-Volatile:				
^{106}Ru	1 y	1.73E+4	1.27E+4	1.37E+4
Uranium:				
^{232}U	72 y	1.21E-2	1.56E-4	1.43E-4
^{234}U	2.5E+5 y	5.11E-2	7.14E-3	7.90E-3
^{235}U	7.1E+8 y	1.69E-2	8.62E-5	8.02E-5
^{236}U	2.4E+7 y	2.88E-1	2.34E-3	2.11E-3
^{237}U	6.7 d	1.99E 0	9.92E-3	1.58E-2
^{238}U	4.51E+9 y	3.13E-1	1.57E-3	1.56E-3
Neptunium:				
^{237}Np	2.14E+6 y	3.41E-1	6.24E-1	5.67E-1
^{239}Np	2.35 d	1.83E+1	1.80E+1	5.88E+1
Plutonium:				
^{238}Pu	86 y	2.82E+3	1.09E+2†††	2.07E+2
^{239}Pu	2.4E+4 y	3.23E+2	1.62E 0†††	1.85E 0
^{240}Pu	6.6E+3 y	4.78E+2	3.49E 0†††	9.02E 0
^{241}Pu	13 y	8.29E+4	4.05E+2†††	5.97E+2
^{242}Pu	3.9E+5 y	1.38E 0	6.87E-3†††	1.37E-2
Particulates:				
^{89}Sr	50.4 d	1.95E-5	2.28E-6	2.20E-6
^{90}Sr	28 y	6.86E+4	6.79E+4	6.49E+4
^{90}Y	64.8 h	6.86E+4	6.79E+4	6.49E+4
^{91}Y	59 d	4.23E-4	6.39E-5	6.22E-5
^{95}Zr	65 d	4.79E-3	8.66E-4	8.59E-4
^{95}Nb	35 d	1.06E-2	1.88E-2	1.86E-3
^{110m}Ag	249 d	2.49E+1	1.58E+1	1.79E+1
^{125}Sb	2.7 y	2.45E+3	2.18E+3	2.33E+3
^{127m}Te	105 d	1.45E-1	5.20E-2	5.34E-2
^{134}Cs	2.1 y	4.56E+4	3.90E+4	3.87E+4
^{137}Cs	30 y	9.61E+4	9.50E+4	9.53E+4
^{144}Ce	285 d	1.29E+4	8.71E+3	8.58E+3
^{147}Pm	2.7 d	2.90E+4	2.59E+4	2.60E+4
^{154}Eu	16 y	5.65E+3	5.50E+3	5.76E+3
^{155}Eu	1.7 y	1.11E+3	9.30E+2	9.82E+2
^{241}Am	458 y	8.30E+2	1.61E+2	2.95E+2
^{243}Am	7.5E+3 y	1.83E+1	1.80E+2	5.88E+1
^{242}Cm	162 d	2.16E+1	1.45E+1	3.82E+1
^{243}Cm	33 y	3.35E 0	3.27E 0	6.04E 0
^{244}Cm	18 y	2.04E+3	1.96E+3	1.04E+4
Carbon:††				
^{14}C	5730 y	5.54E-1		

Note: 4.03E + 2 = 403, or 4.03 x 10^{2}; 3.74E-2 = 0.0374, or 3.74 x 10^{-2}.

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*33,000 MWD/MTHM; 30 MW/MTHM.

**Radioactivity in spent fuel.

***High level wastes generated by reprocessing spent fuel after 160 days cooling.

†Combination of enriched uranium and mixed oxide fuel such that the mixed oxide fuel comprises 11 wt % of the total. The mixed oxide fuel is from a 1.15 SGR PWR based on its total fuel requirements over a 30-year life (includes initial uranium-burning years).

††Assumes 25 grams nitrogen/MTHM in fresh fuel.

†††With uranium recycle alone, all the plutonium recovered by reprocessing spent fuel is considered a waste and would be handled in a manner similar to the high level waste. These activities are only those associated with the solidified high level waste.

the gaseous releases given in Tables IV H-14, H-15, and H-16, the annual release rates given in Table IV H-18 were calculated. In this analysis the releases for the no recycle option are assumed to be the same as for uranium recycle only. The annual total body dose and organ doses (50-year commitment) to individuals from the gaseous effluent (at 500 meters) from a model bedded salt repository, operating with the releases given in Table IV H-18, are summarized in Table IV H-19. Population dose commitment estimates are also presented. Dose commitments to the organs of the individual at the fencepost and the population at large were slightly higher for the uranium and plutonium recycle option than for uranium recycle only, with the exception of the bone dose commitments which are slightly lower and the population liver dose commitments which are equal in both cases. The maximum increase in dose commitment due to plutonium recycle for any organ of the individual at the site perimeter was 27%; whereas the maximum increase was 22% for the general public. The annual individual total body dose commitment (0.05 mrem) is about 0.04% of the average annual dose commitment due to natural background[51] and approximately half the annual dose commitment (0.1 mrem) resulting from television viewing.[51]

Introduction of heat into the environment by a repository would be limited by control on the heat output of individual canisters emplaced and on the spacing of canisters in the formation. A projected complete inventory of waste for the repository might, over an estimated 800 years, produces a peak heat flux at the ground surface of no more than six or seven times the natural geothermal flux. The associated maximum increase in surface temperature has been estimated to be less than 0.1°F, which should not have any perceptible effect on surface climate, meteorology, or ecology. By comparison, the average annual solar flux reaching the ground is about 10,000 times greater than the geothermal flux.

There would be an extremely slow process of subsidence due to plastic deformation of the mineral to close the mined cavities, modified at first by thermal expansion and later by thermal contraction. It is estimated that, over several thousand years, this process would result in the development of a very broad, uniform, and shallow (about 3-1/2 ft) subsidence over the general area above the storage repository. The rate of this subsidence is estimated to approximate that of the ground above a producing mineral mine of equal size, but the magnitude of the subsidence is projected to be about one-half as much as that of a mine because of requirements for backfilling of rooms and tunnels at the time of closing and sealing of the repository. Thus, the rate of development of this shallow depression is judged to be so slow that the overlying rock would adapt to the deformation without fracturing or other adverse effects.

The repository would release no radioactive liquid effluents. Any contaminated liquid (e.g., from canister or equivalent decontamination) would be evaporated at the surface facilities and the residue handled as solid radioactive waste. All solid radioactive waste generated at the repository will be disposed of at the site.

Table IV H-18

SOURCE TERMS FOR 1-YEAR OPERATION OF A MODEL BEDDED SALT REPOSITORY

	Release Rate - (mCi/yr)	
Element	U recycle only*	U and Pu Recycle
Semi-Volatile:		
Ru-106	2.15E-1	2.88E-1
Uranium:		
U-232	2.64E-9	3.01E-9
234	1.21E-7	1.66E-7
235	1.46E-9	1.69E-9
236	3.95E-8	4.44E-8
237	1.68E-7	3.33E-7
238	2.65E-8	3.28E-8
Neptunium:		
Np-237	1.05E-5	1.19E-5
Np-239	3.04E-4	1.24E-3
Plutonium:		
Pu-238	9.12E-2	4.36E-3
239	5.44E-3	3.89E-5
240	8.01E-3	1.90E-4
241	1.36E+0	1.26E-2
242	2.30E-5	2.88E-7
Particulates:		
Sr-89	3.85E-11	4.63E-11
Sr-90	1.15E+0	1.37E+0
Y-90	1.15E+0	1.37E+0
Y-91	1.08E-9	1.31E-9
Zr-95	1.46E-8	1.81E-8
Nb-95	3.18E-8	3.92E-8
Ag-110m	2.67E-4	3.77E-4
Sb-125	3.68E-2	4.91E-2
Te-127m	8.78E-7	1.12E-6
Cs-134	6.59E-1	8.15E-1
Cs-137	1.60E+0	2.01E+0
Ce-144	1.47E-1	1.81E-1
Pm-147	4.38E-1	5.47E-1
Eu-154	9.29E-2	1.21E-1
Eu-155	1.57E-2	2.07E-2
Am-241	2.72E-3	6.21E-3
Am-243	3.04E-3	1.24E-3
Cm-242	2.45E-4	8.04E-4
Cm-243	5.52E-5	1.27E-4
Cm-244	3.31E-2	2.19E-1
Gases:		
H-3	9.00E-1	9.00E-1
Kr-85	1.40E+1	1.40E+1
Rn-220	4.00E+1	4.00E+1
Rn-222	9.00E+2	9.00E+2

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*Source terms for no recycle are expected to be similar to those for the U recycle case.

Table IV H-19

SUMMARY OF 50-YEAR DOSE COMMITMENTS TO AN INDIVIDUAL* AND A POPULATION** DUE TO 1-YEAR OPERATION OF A MODEL BEDDED SALT REPOSITORY

Organ	U Recycle Only***		U and Pu Recycle	
	Individual (rem)	Population (person-rem)	Individual (rem)	Population (person-rem)
Total body	4.7E-5	1.2E-1	5.1E-5	1.3E-1
G.I. tract	2.6E-5	2.7E-2	3.3E-5	3.3E-2
Bone	3.3E-4	5.8E-1	2.7E-4	5.3E-1
Liver	6.1E-5	1.2E-1	6.9E-5	1.2E-1
Kidney	1.4E-4	4.2E-1	1.5E-4	4.3E-1
Thyroid	1.7E-5	1.4E-2	2.1E-2	1.7E-2
Lung	4.3E-5	5.2E-2	4.6E-5	5.4E-2
Skin	1.7E-5	1.4E-2	2.1E-5	1.7E-2

NOTE: 4.7E-5 = 4.7×10^{-5}

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*Dose to individual is at site boundary 500 meters from point of release.

**Dose to U.S. population integrated from 7.5 persons/sq mi in the West to 160 persons/sq mi in the East, over a distance of 2,000 miles.

***Doses for no recycle are expected to be similar to those for the uranium recycle case.

3.2.3 Environmental Effects of Accidents at Geologic Repositories

Several improbable accidents that could result in the dispersion of radioactive materials have been considered to provide estimates of the maximum consequences. All radioactive materials will be handled as tightly sealed packages of solid materials within confinement zones of the repository. The only significant mechanism for sudden and acute release of radioactive materials from the repository must involve rupture of a package and passage of airborne particulates or gases through a series of High Efficiency Particulate Air (HEPA) filters. These filter systems will be physically isolated from operating areas; preceded by a series of roughing filters; followed by differential-pressure-sensitive, quick-operation check valves; and tested routinely to provide protection against pressure surges, plugging, and deterioration. The abnormal occurrences or events that could take place at the model repository depend on the fuel cycle operating mode (no recycle, uranium recycle, and uranium and plutonium recycle) under which the repository is operating.

Uranium and Plutonium Recycle or Uranium Recycle. The maximum credible accident in a waste repository is postulated to involve a major rupture of a waste canister dropped during handling or inspection in the waste receiving building, and the resulting release of dispersible solids from the canister. Probability of dropping a canister is low--about one of every 10,000 canisters handled--and of those dropped, one in

1,000 may release waste.[52] It is assumed that such an accident could occur in the waste receiving cell, with waste containing the most radioactivity and waste which is in the most dispersible form: a granular calcine derived principally from reprocessing plants. On release from the canister, a small fraction of the waste would be suspended in the air in approximately 400 m^3 of cell space. On the basis of six air changes per hour, the contaminated cell air could be exhausted with a half period of 7 minutes following the accident. The air would pass in series through the receiving cell and exhaust system HEPA filters, and would be discharged to the atmosphere at an effective stack height of 28 m. Under the postulated conditions, 0.02 mg of waste per cubic meter of air might escape through the HEPA filters,[53] thus resulting in a total release of 8 mg.[54] The calculated impacts of radioactive releases resulting from such an accident with uranium recycle alone and both uranium and plutonium recycle are summarized in Table IV H-20, for distances of 1,500 m, the assumed boundary of the exclusion area.

If the waste is in the form of a monolithic glassy solid, the solids would not be dispersed to a significant extent and the accident would not cause a significant radiological effect.

A criticality accident during the handling of plutonium waste canisters in the high level receiving building would have approximately the same consequence as a criticality accident in a fuel fabrication plant. See CHAPTER IV, Section D, paragraph 5.3.

The rupturing of a container of Zircaloy fuel element built hulls such that the contents are emptied onto the floor of the high level receiving cell was also considered. Experiments have shown that it is virtually impossible to ignite this material.[55] Thus, it is assumed that there would be no serious effect as a result of this accident.

No Recycle. The maximum credible accident for the no recycle option is the rupture of a spent fuel assembly in the waste receiving building. The upper limit release of gases would be the complete release of krypton and tritium. A single fuel assembly contains 500 kg of heavy metals (HM). The dispersion of a significant amount of solids from the monolithic fuel is not expected. It was concluded that the environmental effect of this accident would be negligible based on the analysis of a similar incident at a fuel reprocessing plant.[56] The reprocessing plant will release all of the krypton and tritium from 2,000 MTHM annually and it was concluded that the effect of these releases from 500 kg of fuel would be negligible.

Loss of Geologic Containment. The most complete study of geologic containment failure mechanisms and their consequences was made for a waste repository in bedded salt of the Delaware Basin in Southeast New Mexico.[44] The main conclusion of that study was that a serious breach of containment of a waste repository, either by natural events or human action, is an extremely remote possibility that is a much smaller risk than many others acceptable to society and of such small magnitude to be beyond the limit of human experience. Once the waste has been placed in such a

Table IV H-20

CALCULATED 50-YEAR DOSE COMMITMENT (MILLIREM) RESULTING FROM THE MAXIMUM CREDIBLE ACCIDENT 1,500 METERS FROM FACILITY*

Organ	U Recycle	U and Pu Recycle
Whole body	1.7	1.9
G.I. tract	1.2	1.2
Bone	2.8	5.6
Thyroid	1.8	1.9
Lungs	1.6	1.8
Kidneys	1.7	2.4
Liver	1.6	2.8

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*Stable meteorological conditions (E stability category) wind speed 1 m/sec, 1,500 m from center of facility.

configuration and the mine sealed, only the most extreme of natural events have any potential for release of radioactivity from the disposal zone. In addition, a sealed repository would be sabotage-proof if there were no armed intervention and temporary occupation of the area. Even the surface burst of a 50-megaton nuclear weapon could not breach the containment.

4.0 SUMMARY OF INCREMENTAL ENVIRONMENTAL EFFECTS FROM WASTE MANAGEMENT

Table IV H-21 summarizes the source and type of waste generated by each segment of the LWR industry, the volumes of waste generated in the period 1975 through year 2000 and the land area permanently committed to the diposal of these wastes for the three fuel cycle options of no recycle, uranium recycle only, and recycle of both uranium and plutonium. The volume of nontransuranium waste generated by LWR's is the same for the three fuel cycle options: 3,800,000 m^3 of waste requiring about 543 acres of burial ground.

The spent fuel and reprocessing wastes shown in Table IV H-21 are based on the disposal schedules given in Table IV H-22. Table IV H-22 shows the annual generation of spent fuel and reprocessing waste during the period 1975 through year 2000. By the year 2000 for the option of no recycle, 76,300 MTHM of spent fuel would have been sent to a Federal geologic repository and 50,000 MTHM of spent fuel would remain in pool storage. If uranium only is recycled, or both uranium and plutonium are recycled, 115,000 MTHM would have been reprocessed by the year 2000 and 11,200 MTHM as spent fuel would be in a pool storage awaiting reprocessing.

Table IV H-21

WASTE MANAGEMENT PARAMETERS FOR THE NUCLEAR FUEL CYCLES FOR THE YEARS 1975 THROUGH 2000

	Volume of Waste, m^3			Acres of Land Permanently Committed			
Source and Type of Waste	No U or Pu Recycle	U Recycle Only	U and Pu Recycle	No U or Pu Recycle	U Recycle Only	U and Pu Recycle	Remarks
Uranium mining	-----------Not Estimated-----------------			11.000**	9,500**	7,800**	Average mix: 60% ore from mining and 40% from open pit mines
Uranium mill tailings	780,000,000	690,000,000	590,000,000	16,500	14,600	12,600	Composite model with 85% acid leach, SX--15% alkaline leach and 50% New Mexico--50% Wyoming sites
UF_6 production waste							Industry mix 59% dry process and 41% wet process; waste process; waste buried onsite or in licensed burial ground†
Low level waste	88,000	79,000	67,000	12.6	11.3	9.6	
Chemical waste	179,000	160,000	136,000	25.6	22.9	19.4	
Enrichment wastes	65,000	65,000	22,000	9.3	9.3	3.1	Total of four gaseous diffusion plants; gas centrifuge plant on-line in 1990-1995. Waste buried onsite or in licensed burial ground†
Fuel fabrication							Buried onsite or in licensed ground†
Enriched uranium	154,000	154,000	133,000	22	22	19.3	
Mixed oxide	0	0	18,500	0	0	***	Waste Federal repository***
Reactor operation							
Misc. low level waste	3,800,000	3,800,000	3,800,000	543	543	543	Buried in licensed burial ground†
Spent fuel	55,000	0	0	***	0	0	Spent fuel to Federal repository***
Spent fuel storage	3,350	1,680	1,000	0.5	0.2	0.1	Buried in licensed burial ground†
Fuel reprocessing							
Non-radioactive solids	0	23,000	23,000	0	3.3	3.3	Buried in licensed burial ground†
Calcined high level	0	6,500	6,500	0	***	***	To Federal repository***
Misc. transuranium	0	76,500	78,000	0	***	***	To Federal repository***
Hulls and hardware		52,000	52,000	0	***	***	

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*Before compaction or incineration.

**Based on assumption that the entire environmental impact of the model mines occurs during the first year of ore production.

***Permanent geologic disposal at Federal repository; above ground facilities for each repository occupy about 250 acres; 2 repositories to be established by the year 2000.

†Eleven licensed commercial burial grounds required by year 2000; each site occupies about 100 acres.

Table IV H-22

ANNUAL GENERATION OF SPENT FUEL OR REPROCESSING WASTES IN THE YEARS 1975 THROUGH 2000 (MTHM)*

Year	No Recycle	U-Recycle	U and Pu Recycle
1978	0	0	500
1979	0	0	1,000
1980	0	0	1,500
1981	0	0	1,500
1982	0	0	1,750
1983	0	0	2,000
1984	0	0	2,250
1985	0	0	2,250
1986	0	1,350	2,850
1987	3,500	3,400	3,550
1988	3,500	5,840	4,250
1989	3,500	7,440	4,250
1990	3,500	7,850	4,850
1991	3,570	8,250	5,550
1992	4,100	8,240	6,250
1993	4,610	8,250	6,850
1994	4,930	8,250	7,550
1995	5,470	8,250	8,250
1996	6,130	8,250	8,250
1997	6,670	9,250	9,250
1998	7,150	10,250	10,200
1999	7,800	10,250	10,200
2000	8,400**	10,250***	10,200***

---

*For the two fuel recycle options, these are the metric tons of heavy metal reprocessed. The amounts of each category of waste generated per metric ton reprocessed may be estimated from the data of Table IV H-21 assuming that 115,000 MTHM would have been reprocessed in the years 1975 through 2000.

**In the year 2000, the MTHM as spent fuel at a Federal repository and in pool storage would be 76,300 and 50,000, respectively.

***In the year 2000, 11,200 MTHM as spent fuel would be in pool storage awaiting reprocessing.

Tables IV H-23 and IV H-24 are corollaries of Table IV H-21 showing more clearly the effect of fuel cycle option on waste generation and the nonradiological impact of the waste management activities. The former shows the cumulative volumes of waste in the year 2000 for the three fuel cycle options and the latter summarizes the acres of land permanently committed to the disposal of radioactive wastes generated over the 26-year period through the year 2000 for the three fuel cycle options. The acres of land permanently committed to disposal of LWR waste in Table IV H-24 were estimated based on the following assumptions:

Table IV H-23

CUMULATIVE VOLUMES OF WASTE INVENTORY IN THE YEAR 2000 (m^3)

Type of Waste	Fuel Cycle Option No Recycle	 U Recycle	 U + Pu Recycle
Milling Tailings	7.8×10^{8}	6.9×10^{8}	5.9×10^{8}
Spent Fuel	55,000*	6,000***	6,000***
High Level	b	6,500†	6,500†
Transuranium	b	76,500††	96,500
Hulls and Hardware (Transuranium)	b	52,000	52,000
Reactor Waste (Nontransuranium)	3.8×10^{6}	3.8×10^{6}	3.8×10^{6}
Other Nontransuranium	310,000	300,000	223,000
Chemical	179,000	183,000	159,000

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*400,000 spent fuel assemblies.

**The spent fuel constitutes these wastes.

***37,000 spent fuel assemblies in pool storage awaiting processing; not a waste.

†Volume of high level waste in 37,000 canisters.

††Includes plutonium wastes.

Table IV H-24

LAND PERMANENTLY COMMITTED TO DISPOSAL OF RADIOACTIVE WASTE FOR THE PERIOD 1975 THROUGH YEAR 2000

	Acres of Land Permanently Committed No U or Pu Recycle	 U Only Recycle	 U and Pu Recycle
U_3O_8 Production			
Mining	11,000	9,500	7,800
Mill Tailings	16,500	14,600	12,600
Total	27,500	24,100	20,400
Burial Grounds			
Reactor Operation	543	543	543
Fuel Cycle Facilities*	70	69	55
Total	613	612	598
Federal Repositories**			
Transuranium Waste	†	50††	65
High Level Waste***	†	830	1,000
Hulls and Hardware	†	35	35
Spent Fuel***	970		
Total	970	915	1,100

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*Includes chemical wastes buried onsite.

**These are the subsurface acreages required for disposal of the wastes. The surface facilities of the two Federal repositories required in the year 2000 occupy 500 acres.

***Based on 115,000 MTHM reprocessed or disposed of as spent fuel.

†The spent fuel constitutes these wastes.

††Does not include 170 acres required for plutonium waste.

- Burial grounds have two trenches per acre, each holding about 3,500 m^3 of low level waste.

- Each geologic stratum can store about 1,500 m^3 of transuranium waste, or aged fuel element hulls and hardware, per acre.

- Each geologic repository for high level waste, or spent fuel assemblies, is limited to a power density of about 150 kW per acre.

- The power density of the high level waste and spent fuel assemblies are those resulting after 10 years' decay (see Table IV H-13).

Table IV H-23 shows that with the exception of mill tailings the cumulative waste volumes in the year 2000 for the operation of the LWR industry during the years 1975 through 2000 are not significantly different for the three options of no recycle, uranium recycle only, and recycle of both uranium and plutonium. However, a substantial reduction in the quantity of mill tailings generated (and long term waste management requirements and radon releases) result from uranium and plutonium recycling.

Lands permanently committed as a result of mining and milling are substantially reduced by uranium and plutonium recycle over no recycle (see Table IV H-24). Subsurface acreages required for the disposal of the wastes generated by uranium and plutonium recycle are about 12% greater than with no recycle.

The volumes of wastes generated in fuel reprocessing when both uranium and plutonium are recycled will not necessarily be greater than those from uranium recycle only (see Table IV H-21); however, the rate of heat emission in the high level wastes resulting from reprocessing the average mix of enriched uranium and mixed oxide fuels will be about 17% higher after 5 years' decay and 21% higher after 10 years' decay than in the high level wastes from uranium oxide fuels because of the increased amounts of transuranium elements.

Recycle of uranium and plutonium results in the accumulation of the least amount of plutonium (about 17 MT as opposed to about 1,000 MT) in the wastes for the three fuel cycle options because the plutonium is recycled as fuel in LWR's (see Table IV H-25). With uranium recycle only, the plutonium recovered from reprocessing the spent fuel is considered a waste material and is disposed of in a manner similar to the high level waste and fuel assembly hulls and hardware. With no recycle, the spent fuel assemblies constitute the waste and contain all the plutonium generated in the fuel.

The greatest radiological impact from the radioactive wastes of the nuclear fuel cycle results from the long term release of ^{222}Rn from the retired tailings piles. The annual releases of ^{222}Rn from the tailings produced in the period 1975 through year 2000 (after retirement of the tailings piles) is estimated at 420,000 curies for no recycle, 376,000 curies for uranium recycle only, and 327,000 curies for recycle of both uranium and plutonium. The annual 50-year total body dose commitment to the United States population from these releases for all pathways are: 47,000 person-rem for the no recycle option, 43,000 person-rem for the uranium recycle option, and 37,000 person-rem for the uranium and plutonium recycle option. The annual kidney dose commitment would be 180,000, 170,000, and 140,000 person-rem, respectively. On a gross basis, the population doses from the release of radon from tailings piles would be highest for no recycle, would be reduced by 11% if the uranium only recycle were adopted and reduced by an additional 12% (to a total of 23%) if the uranium and plutonium recycle were adopted.

With the exception of the volume of mill tailings, the waste volumes are not significantly different for the three fuel cycle options. The overall environmental impact from the operation of geologic repositories will be approximately the same for the three fuel cycle options. Thus, it is concluded that no radioactive waste management consideration is sufficiently significant to influence a decision between the three fuel cycle options.

Table IV H-25

TOTAL PLUTONIUM ACCUMULATED IN THE WASTES FROM THE LWR INDUSTRY IN THE YEARS 1975 THROUGH 2000 (MT)*

Waste Category	No Recycle	U Recycle	U and Pu Recycle
Spent fuel elements	1,022	0	0
Mixed oxide fuel fabrication	0	0	0.7
High level**	0	6	10.7
Miscellaneous transuranium	0	1,036	3.9
Hulls and hardware	0	1	1.3
Total	1,022	1,043	17

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*Based on 115,000 MTHM reprocessed or disposed of as spent fuel.

**Decayed 5 years; after 5 years' decay the plutonium concentrations in the high level wastes has increased by a factor of 1.19 in wastes from uranium recycle and 1.63 in wastes from uranium and plutonium recycle over the concentrations obtaining when the wastes were first generated. After 25 years' decay, the increase factors are 1.40 and 2.74, respectively.

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GESMO

CHAPTER IV

ENVIRONMENTAL IMPACT DUE TO THE IMPLEMENTATION OF PLUTONIUM RECYCLE

SECTION I

STORAGE OF PLUTONIUM

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STORAGE OF PLUTONIUM

SUMMARY

Spent fuel from the projected nuclear industry power generation from the year 1975 through 2000 will contain about 1,000 MT of plutonium. If this spent fuel is not reprocessed, there will be no need for any commercial plutonium storage facilities. If it is reprocessed and the plutonium recycled as fuel in LWR's, ample plutonium storage capacity will be available as long as recycle is deferred for no more than 2 years after plutonium separation at the reprocessing plant. If this deferral is greater than 2 years, the industry will have to expand planned storage capacity or store excess plutonium in a separate commercial plutonium storage facility. This section addresses the impacts of such a separate facility, assuming that recycle may be delayed for as much as 7 years after reprocessing begins.

Assuming that reprocessing begins in 1978 and that plutonium recycle in LWR's may be delayed for as much as 7 years, a commercial plutonium storage facility would be needed by about 1980. Thus the environmental impacts of the commercial plutonium storage industry from 1980 through 2000 are the total impacts from the commercial plutonium storage industry during the GESMO study period--from 1975 through 2000.

There is now no commercial plutonium storage facility, and no detailed plans for an installation have been developed to date. Therefore, for the purpose of assessing the impact of plutonium storage, a model commercial plutonium storage facility was postulated. The model plant is assumed to be designed and constructed to maintain integrity under conditions that would be imposed by natural phenomena and include provisions for criticality prevention, radiation and contamination control, fire prevention, heat removal, and security against theft or diversion of plutonium. The hardened warehouse structure, which might be located in a 10-acre exclusion area on a 250-acre site or collocated at a reprocessing plant site, is projected to have a storage capacity of 75 MT of plutonium oxide. The storage units are assumed to be shipping containers containing about 6 kg of plutonium oxide each.

The environmental impacts of the plutonium storage industry that might result from the deferral of recycle are small and would have a comparatively minor effect on a decision regarding recycle timing.

Table IV I-1 summarizes the environmental effects of the commercial plutonium storage industry from the year 1980 through 2000 based on the postulated model storage facility.

Table IV I-1
ENVIRONMENTAL EFFECTS OF THE PLUTONIUM STORAGE INDUSTRY
(1980 Through 2000)

Type of Effect	Industry from 1980-2000
Radiological Impact	
Radioactivity in Effluents	None
Dose Commitment to Offsite Individuals	None
Occupational Exposure:	
Maximum Individual	3.1 rem/yr
All Personnel	410 person-rem
Chemical Effluents	None
Use of Natural Resources	
Manpower Use	1,100 man-years
Land Use	5,000 acre-years
Water Use	3×10^7 gallons
Power Use	320 Kw
Thermal Effluent	4.0×10^{11} Btu

1.0 PLUTONIUM STORAGE IN THE FUEL CYCLE

1.1 Plutonium Storage Plans and Methods

Spent fuel from the projected nuclear power generation from the year 1975 through 2000 will contain about 1,000 MT of plutonium.[1] If this spent fuel is not reprocessed, there will be no need for any commercial plutonium storage facilities. If spent fuel is reprocessed and the plutonium is recycled as fuel in LWR's, plutonium storage capacity will be required at reprocessing plants, at plutonium nitrate to oxide conversion facilities, and at mixed oxide fuel fabrication plants. In the event that recycle is delayed for an extended period of time after reprocessing resumes, reprocessing plants could enlarge onsite storage capacity, or separate storage facilities independent of these plants could be constructed.

1.1.1 Plutonium Storage in Reprocessing Plants

All current plans for plutonium storage at reprocessing plants have been based on storing plutonium as a nitrate solution, although it appears in the near future that provisions will be made for storing the oxide.

It has been the plan for reprocessors to store plutonium as a nitrate solution for several reasons. As it is separated from uranium and fission products, and in the final product stream from the separations plants using the Purex process, plutonium is in the nitrate solution form and can be directly transferred to storage tanks. If the plutonium needs any further purification or processing (e.g., the removal of ^{241}Am), it can be easily returned to ion exchange columns or other process equipment since only solution transfers would be required.

Except for the AGNS reprocessing plant,[2] which has increased its nitrate solution storage capacity from about 4,000 to about 8,000 kg of plutonium to accommodate an expected delay in plutonium conversion in the 1970's, all future reprocessing plants are assumed to have a nitrate solution storage capacity of 4,000 kg of plutonium.

Thus for purposes of this assessment it is assumed that the first reprocessing plant will provide a nitrate solution storage capacity of about 8 MT of plutonium and each plant thereafter will provide a nitrate solution storage capacity of 4 MT of plutonium. Impacts associated with plutonium storage at reprocessing plants are included in CHAPTER IV, Section E.

1.1.2 Storage at Plutonium Nitrate to Oxide Conversion Facilities

Based on current NRC regulations, which require shipment of plutonium in a solid form after June 1978,[3] it is projected that, with the resumption of reprocessing of commercial spent enriched uranium LWR fuel, reprocessing plant licensees will most likely make the addition of a plutonium nitrate to oxide conversion step to their process a first priority if the recycle of plutonium as fuel in LWR's is permitted.

If reprocessing resumes in 1978, it is judged that one such conversion step could possibly be in operation by late 1979 and that it will be designed with an initial plutonium oxide storage capacity of about 15 MT; it is also assumed that the storage capacity can be readily expanded to about 30 MT if needed.[2]

Thus, for purposes of this assessment, plutonium conversion facilities are assumed to be designed to have an initial plutonium oxide storage capacity of 15 MT and be easily expandable to 30 MT. Impacts associated with plutonium storage at plutonium oxide conversion facilities are included in CHAPTER IV, Section E.

1.1.3 Plutonium Storage at MOX Fuel Fabrication Plants

Based on current NRC regulations, which require the shipment of plutonium in a solid form after mid-1978,[3] it is assumed that MOX fuel fabrication plants will receive and store plutonium in the oxide form. It is judged that each MOX plant will have a design plutonium oxide storage capacity of about 1,000 kg.[4] Impacts associated with plutonium storage in MOX fuel fabrication plants are included in CHAPTER IV, Section D.

1.1.4 Plutonium Storage at Commercial Storage Facilities

Based on the NRC regulation cited above, it is assumed that commercial storage facilities will also receive and store plutonium as the oxide. These facilities would only be needed if the recycle of plutonium as fuel in LWR's is delayed for more than 2 years after reprocessing resumes.

A commercial plutonium storage facility is needed when plutonium production exceeds the storage capacity available at reprocessing plants.

Thus the storage capacity of such a facility is dependent on the delay in plutonium recycle after reprocessing resumes. For purposes of this assessment, these delay times are assumed to be 2 and 7 years.

2.0 PLUTONIUM STORAGE REQUIREMENTS FOR VARYING RECYCLE TIMING CONDITIONS

Figure IV I-1 shows the plutonium storage requirements for the prompt and deferred recycle cases. These requirements are based on a "low growth" industry model with the assumption that reprocessing will resume in 1978.[1]

Superimposed on the curves representing storage requirements for the various recycle cases is a step function graph representing the total annual plutonium storage capacity available at reprocessing plants and nitrate to oxide plutonium conversion facilities.

The total annual plutonium storage capacity of reprocessing plants and nitrate to oxide plutonium conversion facilities is based on the following assumptions: one reprocessing plant may be in operation from 1978 through 1985 with a plutonium nitrate solution storage capacity of 8 MT; a plutonium nitrate to oxide conversion facility will be added to the reprocessing plant process in late 1979 with a storage capacity of 15 MT of plutonium oxide; the oxide storage capacity of the conversion plant will

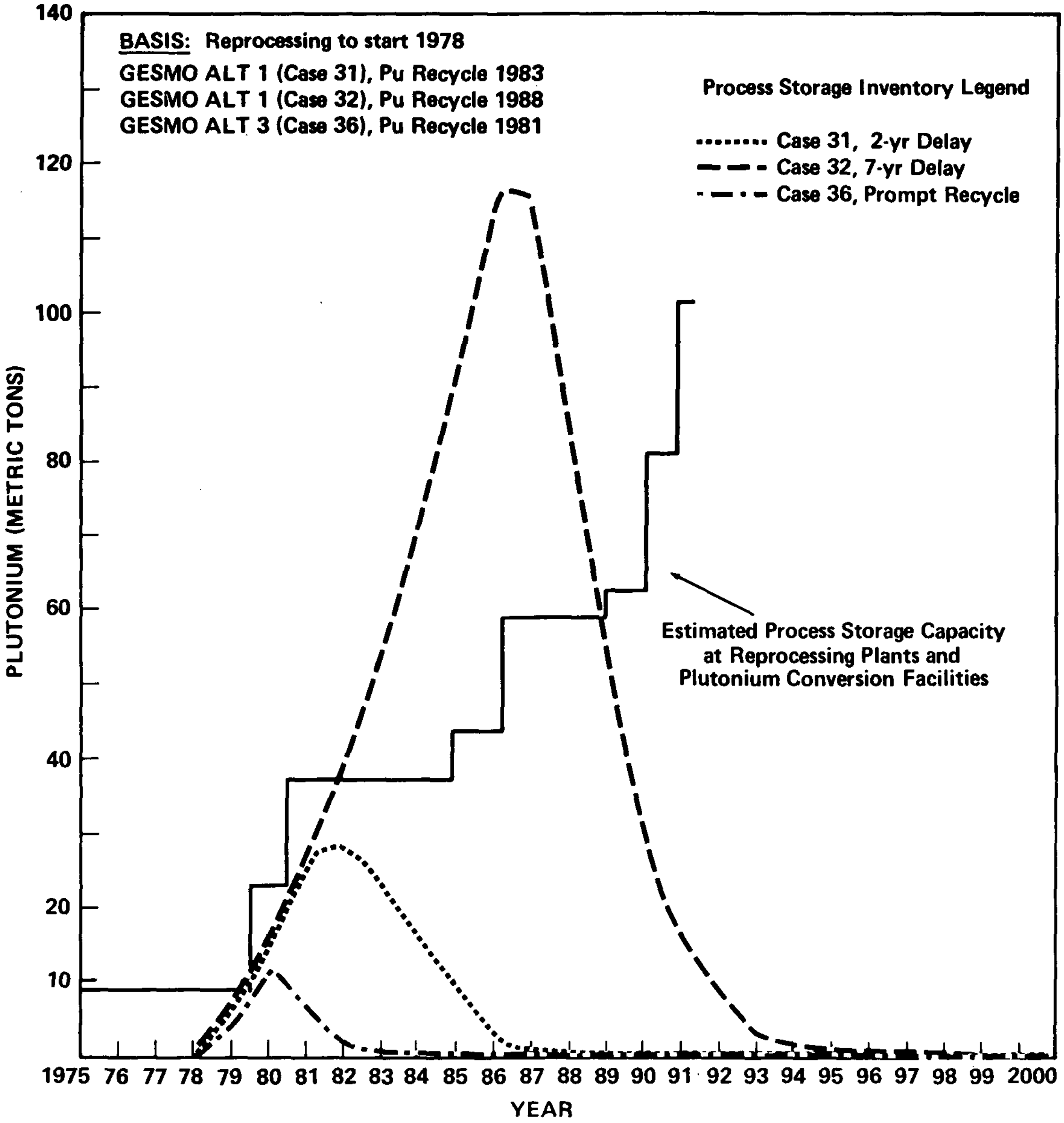

Figure IV I-1 Plutonium Storage Requirements be increased by an additional 15 MT by late 1980; 4 MT of plutonium (as nitrate solution) storage capacity will be made available when the second reprocessing plant comes on line in about 1985; and in mid-1986, too late to alleviate storage requirements during the peak storage year in the 7-year delay case, an additional 15 MT of plutonium (as oxide) storage could be provided at the second plutonium nitrate to oxide conversion facility and the first MOX fuel fabrication plant.

2.1 Storage Requirements for Prompt Recycle of Plutonium

Figure IV I-1 indicates that, if reprocessing begins in 1978 and plutonium is promptly recycled as fuel for LWR's starting in 1981, ample plutonium storage capacity will be available at reprocessing plants and plutonium nitrate to oxide conversion facilities.

2.2 Storage Requirements for Deferred Recycle of Plutonium

2.2.1 Two-Year Deferral of Plutonium Recycle

Figure IV I-1 indicates that a delay in recycle from 1981 (prompt recycle) to 1983 (2-year delay) would result in the need for plutonium storage capacity in addition to that projected to be provided at fuel reprocessing plants. Under these postulated conditions the storage needs would peak in about early 1982 at about 30 MT. The 8 MT of plutonium (as nitrate solution) storage capacity available at the reprocessing plant and the 15 MT plutonium (as oxide) storage capacity provided by the plutonium conversion facility would therefore have to be supplemented by an additional 7 MT of storage capacity. It is judged that storage for the 7 MT excess will be provided by enlarging the storage capacity of the plutonium conversion facility.

Thus there can be ample plutonium storage capacity at reprocessing plants and plutonium conversion facilities if plutonium is promptly recycled in 1981 or if recycle is delayed 2 years until 1983.

2.2.2 Seven-Year Deferral of Plutonium Recycle

Figure IV I-1 indicates that a further deferral in recycle to about 1988, assuming reprocessing to begin in 1978, would cause the total plutonium storage requirement to increase, peaking at 117 MT in the year 1986, the initial year that plutonium oxide would be shipped to MOX fuel fabrication plants for manufacture into LWR fuel. The 42 MT combined plutonium storage capacity projected for reprocessing plants and plutonium conversion facilities for the year 1986 would have to be augmented by an additional storage capacity of about 75 MT to accommodate the total amount of plutonium produced during the 1976-86 period. Seventy-five MT of additional storage could be accommodated by providing significant increases in the plutonium conversion facility storage capacity or by constructing an offsite storage facility.

For purposes of this assessment, it is assumed that a separate offsite commercial plutonium oxide storage facility with a capacity of 75 MT of plutonium is constructed in about 1980 to satisfy this storage requirement. Thus the environmental impacts resulting from the commercial plutonium storage industry shown in this chapter to occur from the year 1980 through 2000 are the total impacts from the commercial plutonium storage industry during the GESMO study period--from 1975 through 2000.

3.0 MODEL COMMERCIAL PLUTONIUM OXIDE STORAGE FACILITY

The model plutonium storage facility assumed for purposes of this evaluation occupies a 10-acre exclusion area within a 250-acre site.

The plant complex includes a central warehouse structure containing facilities for loading and unloading shipments, packaging and decontamination cells, security facilities, and utilities for essential safety functions. This structure will be designed and constructed to maintain integrity under conditions postulated to be imposed by natural phenomena. The design of the central warehouse structure will also include provisions for criticality prevention, radiation and contamination control, fire prevention, heat removal, and security against theft or diversion of plutonium. Facilities needed for maintenance shops, utilities not essential to safety, plant administration, and other activities in which radioactive material is not involved could be located in the hardened structure but for economic and safeguards reasons will most likely be located in buildings of conventional construction separate from the warehouse structure.

3.1 Storage Space Requirements

The configuration and floor area of the storage facility structure are dependent on the method used to ship and store plutonium.

3.1.1 Requirements for Current Shipping Practices

If current shipping practices are used, 6 kg lots of plutonium oxide will be packaged in steel canisters, the canisters will be placed in specially designed shipping containers--similar to 55-gallon drums[5]--and the containers will be shipped by truck in lots of 40 to the storage facility.

In the "drum" shipment concept, each shipping container would occupy about 4 sq ft of floor area standing upright and about 12 sq ft of floor area if placed lengthwise, or 1 to 2 sq ft of floor area per kilogram of plutonium oxide. If the canisters were removed from the shipping container for storage, each would occupy about 2 sq ft of floor area if placed lengthwise, or about 0.33 sq ft per kilogram of plutonium oxide.

It is likely that the canisters may be removed from the shipping containers for storage so that the shipping containers may be reused. The storage of unpackaged canisters will require significantly less floor space than the storage of shipping containers. It is expected that an integrated storage facility would provide storage capacity for both storage modes; however, for conservatism, storage floor area requirements are based on the assumption that all plutonium is stored in shipping containers that are placed lengthwise in the storage area. Further, no consideration is given to possible reduction in floor area requirements that might be realized by stacking the containers.

3.1.2 Requirements for Future Shipping Practices

For safeguards considerations, a conceptualized semitrailer Integrated Container Vehicle (ICV) is proposed for the shipment of plutonium oxide. The ICV consists of a cylindrical steel secondary pressure vessel containing seven primary pressure vessels. Each primary pressure vessel would contain four canisters, each holding 18 kg of plutonium oxide. In the ICV option it is expected that the seven primary pressure vessels (each 1.3 feet in diameter by about 4 feet long) would be removed from the secondary pressure vessel and stored as units. This method of storage would require about 0.25 sq ft of floor area per kilogram of plutonium oxide. For further discussion of this concept see the safeguards supplement.

A storage facility designed for the "drum" concept would provide ample storage space for storing the primary pressure vessels in the ICV concept.

3.2 Storage Facility Capital Costs

Storage could be accommodated in a single modular structure or in several structures of appropriate hardened design. For purposes of this assessment, a single structure is assumed.

The capital required to construct a storage facility was estimated in 1973 by the Atomic Industrial Forum to be about $215 per sq ft of floor area.[6] Considering escalating building costs, it is judged that a reasonable estimate of capital costs, in 1975 dollars, for a storage facility in the late 1970's would be about $250 per sq ft of floor area.

As indicated previously, it has been assumed that shipping containers containing about 6 kg of plutonium oxide will be stored lengthwise in the facility without stacking. The containers are about 2 feet in diameter and 6 feet long and occupy about 12 sq ft of floor area, resulting in an area requirement of 2 sq ft/kg or 2,000 sq ft per metric ton of plutonium oxide.

On the assumption that 50% of the floor area in the facility is occupied by aisleways, access spaces, shielding and other safety related equipment, a total of 4,000 sq ft of floor area per metric ton of plutonium oxide is required.

Thus it is conservatively estimated that a commercial storage facility capable of storing 75 MT of plutonium oxide would require about 300,000 sq ft of floor area, about 7 acres, and cost approximately $75 million in 1975 dollars.

3.3 Plutonium Characteristics of Deferred Recycle Material

The type of plutonium considered in the deferral option is that initially produced from uranium-238 during a single cycle through a reactor. It has greater isotopic fractions of plutonium-239 and lesser fractions of plutonium-238, plutonium-240 and plutonium-241 than the industry average plutonium used for dose calculations for prompt recycle. See CHAPTER IV, Section D, paragraph 1.2. Because of the lower plutonium-238 content, heat generation will be about 80% and the specific activity about 90% of that for the prompt recycle plutonium materials. After about 5 years of storage, however, an increase in the specific activity and heat generation will result from the formation of americium-241 because of the decay of plutonium-241 making the heat and specific activity characteristics of the plutonium about equal for the prompt recycle and the deferral options. Thus, regardless of the option chosen--prompt or delayed recycle--it is assumed that no significant changes in the design or operation of plutonium storage facilities will be required due to the characteristics of the plutonium.

3.4 Refrigeration and Power Requirements

Plutonium oxide generates heat at about 68 Btu/kg-hr (about 20 W/kg) because of the radioactive decay of its isotopes.[7] Thus air cooling is considered adequate for stored plutonium oxide. If the cooling system air flow is interrupted, however, the temperature of stored plutonium oxide can increase slowly. The temperature in the immediate vicinity of a 10-kg mass of plutonium oxide--confined with no heat removal provisions--may reach about 350°F in about 36 hours. For this reason, redundant components are provided in storage facility air cooling and ventilation systems.

Since plutonium oxide generates about 20 W/kg, or about 1,137 Btu/min-MT and 1 ton of refrigeration will absorb 200 Btu/min,[8] about 5.7 tons of refrigeration will be required to cool each metric ton of stored plutonium oxide.

Thus in 1986, the peak storage year, the 75 MT of stored plutonium oxide would require about 428 tons of refrigeration for cooling, or about that required to cool 150 to 175 private homes during the summer months. The power requirement for this amount of cooling is estimated to be about 320 kW.

It is likely that the ventilation system of the facility would be designed so that the heat generated by the storage would provide the bulk of the facility heating requirement during the winter months.

3.5 Manpower Requirements

The storage operation is not labor intensive. It is judged that a supervisor, a clerk, six operators, and two maintenance personnel per shift on a three 8-hour shift basis would be adequate to staff the facility. The total work force would thus be about 40 persons exclusive of safeguards manpower requirements. This work force may be supplemented in the years 1985-1990, when the facility experiences the peak storage activity.

3.6 Occupational Exposure

The storage facility is designed, constructed, and operated using the radiation protection philosophy of "as low as reasonably achievable."[9]

The major portion of the occupational radiation exposure resulting from plutonium storage facility operations is judged to result from the container handling operations--that is, the movement of containers in and out of storage.

Department of Transportation (DOT) regulations limit radiation dose rates of radioactive shipments to 10 mrem/hr 3 feet from the surface of the shipping container. Plutonium oxide shipping containers are designed to meet DOT specifications for radiation levels.[10] It is estimated that the neutron-gamma dose rate at 3 feet from such a container holding about 6 kg of first recycle plutonium oxide would be about 3 to 5 mrem/hr.

However, for purposes of this evaluation, occupational exposures are based on the following conservative assumptions:

- Shipments are received by truck in 40-drum lots, each drum containing about 6 kg of plutonium oxide.
- Unloading and storing 40 drums takes 1 day and requires six handling personnel.
- The neutron-gamma dose rate is at the DOT transportation limit of 10 mrem/hr 3 feet from the container.[11]
- The average neutron-gamma dose rate to handling personnel during the loading, unloading and placement operations is about 3 mrem/hr.
- The average neutron-gamma dose rate in the storage area is 1 mrem/hr.
- Personnel in occupations other than handling spend 50% of their time in the storage facility.
- The balance of the work time spent at the facility by handling and support personnel is spent in service facilities separate from the warehouse structure.

Under these assumptions, the handling of each metric ton of plutonium oxide requires the unloading or loading and placement of about four truckloads of shipping containers during which six handling personnel would be exposed to an average neutron-gamma dose rate of 3 mrem/hour. During the 4 days, the individual dose would be about 100 mrem, or 100 mrem/MT. The total dose to the six handling personnel would be about 0.6 person rem/MT.

Table IV I-2

PLUTONIUM OXIDE STORAGE REQUIREMENTS WITH 7-YEAR RECYCLE DELAY

Year	Quantity of Plutonium Oxide (MT) Placed or Removed from Storage	Cumulative in Storage
1978	2	2
1979	4	6
1980	9	15
1981	10	25
1982	13	38
1983	15	53
1984	18	71
1985	22	93
1986	24	117
1987	-2	114
1988	-31	83
1989	-30	53
1990	-22	31
1991	-15	16
1992	-6	10
1993	-9	1
1994	<-1	<1
1995	<-1	<1
1996	0	0
1997	0	0
1998	2	2
1999	1	3
2000	<1	3

Table IV I-2 summarizes the amount of plutonium sent to or removed from storage annually and the cumulative amount in storage each year from mid-1981 through the year 2000. The peak storage activity will occur in 1988, when 31 MT of plutonium will be removed from storage.

Thus the maximum annual individual occupational dose during the life of the storage facility is estimated to be about 3,100 mrem, or about 60% of the annual individual occupational dose permitted by 10 CFR Part 20 regulations. Since handling operations are expected to be evenly distributed among four six-operator shift groups this estimate is likely to be high by a factor of at least 3.

For the 20-year period 1981 through 2000, the total occupational dose to personnel from handling 75 MT of plutonium oxide into and out of the storage facility would be about 90 person-rem. Review of Table IV I-2 shows that the 75 MT of stored plutonium oxide could be removed from the facility as early as late 1990. For conservatism, however, it is assumed that some plutonium will remain in storage until the year 2000 and the dose rate in the facility will remain constant at 1 mrem/hr resulting in a total 20-year occupational dose to the remaining facility personnel of about 320 person-rem.

Thus the total facility occupational radiation dose for the 20-year period is conservatively estimated to be about 410 person-rem.

3.7 Accidents

The only handling operation involved is the placing and removing of shipping containers, designed to withstand transportation accidents,[12] in carefully analyzed, predetermined arrays. It is unlikely that a shipping container would be damaged to the extent that it would be contaminated on arrival at the facility or that any accident in the facility could rupture a shipping container. However, even though these events are unlikely, the plant cooling and ventilation systems include high efficiency particulate air filters and radiation monitors, and a decontamination cell is included in the facility design.

Storage facilities will also be constructed to withstand the effects of natural phenomena, with essentially nonflammable materials, and have no process needs for explosive quantities of chemicals or gases.

Based on the design of the storage arrays and containers and the nature of operations, the possibility of accidental criticality is judged to be remote.

For these reasons, storage facility accidents that would endanger the radiological health and safety of facility personnel or the public are considered sufficiently remote that they have not been further evaluated in this statement.

3.8 Radioactivity in Effluents

3.8.1 Airborne Effluents

3.8.1.1 Normal Operation

During normal operation, all plutonium is contained in shipping containers or sealed canisters. Thus, there are no radioactive airborne effluents.

3.8.1.2 Accidents

No radioactive airborne effluents will result from accidents at storage facilities. See paragraph 3.7.

3.8.2 Liquid Effluents

No radioactive liquid effluents will result from facility operation.

3.9 Exposure of the General Public

Since no radioactive effluents will be released from a plutonium storage facility, no radiation exposure to offsite individuals or populations will result from facility operation.

4.0 INCREMENTAL ENVIRONMENTAL EFFECTS

Prompt recycle of plutonium or a delay in recycle of up to 2 years after separation at the reprocessing plant has the effect of substantially decreasing plutonium storage requirements and thereby proportionally decreasing any environmental impact that could be ascribable to that storage. The incremental effects described in this section are based on a 7-year delay in recycle.

4.1 Radiological Impacts

Extra handling steps are required when plutonium is stored in an offsite facility. As occupational radiation exposure is judged to be proportional to the amount of plutonium handled, the incremental radiological impact of about 410 person-rem in 20 years would not be realized if plutonium is recycled within 2 years after separation at the reprocessing plant.

4.2 Manpower Impacts

About 200 construction personnel will be required for a period of about 1 year to construct the facility. About 40 or 50 personnel, exclusive of transportation and security personnel, will be required to operate the facility for the 26-year study period. The total manpower requirement would therefore be about 1,100 man-years.

4.3 Land Commitments

If plutonium recycle is delayed for 7 years after separation of plutonium at the reprocessing plant, about 250 acres of land would be committed to a storage site. The storage structure would occupy about 10 acres of the site land. The storage method for the plutonium oxide would provide essentially total confinement such that little or no decontamination should be necessary on decommissioning. It is expected that the structure and land will be returned to unrestricted use after the useful life of the facility. Land use, therefore, would be about 5,000 acre-years during a 20-year period.

4.4 Water Requirements

Small amounts of water--approximately 5,000 gal/day--will be used for sanitary purposes. Over the 20-year period water use would amount to 3×10^7 gallons.

4.5 Thermal Effluents

In the peak storage year, 1986, approximately 5×10^6 Btu/hr of decay heat will be released from the facility--the equivalent of the winter heat loss of 75 to 100 private homes. About 675 metric ton-years[13] of storage will be required during the 26-year study period at 1.14×10^3 Btu/min-MT resulting in a 20-year heat loss from decay heat of 4.0×10^{11} Btu.

4.6 Other Resource Commitments

Other resource commitments are the structural components (steel, concrete, glass) for the storage building and the power required for security, lighting, cooling and mechanical transfer. These are irreversible commitments and would not be committed if plutonium is recycled within 2 years after separation at the reprocessing plant.

REFERENCES

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GESMO

CHAPTER IV

ENVIRONMENTAL IMPACT DUE TO THE IMPLEMENTATION OF PLUTONIUM RECYCLE

Section J

RADIOLOGICAL HEALTH ASSESSMENT

CHAPTER IV

Section J

RADIOLOGICAL HEALTH ASSESSMENT

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Section J
RADIOLOGICAL HEALTH ASSESSMENT

SUMMARY

The previous sections of this CHAPTER IV have described the radiological impacts for each facility in the fuel cycle. Each section contains a radiological assessment of the population, occupational, and accidental modes of radiation exposure from that facility for each fuel cycle option. In Section J the radiological impacts from the facilities for each fuel cycle option and estimates of the risks from the radiation exposure in terms of health effects are summarized.

Comparisons of the different fuel cycle options indicated the U.S. population and occupational dose commitments are about the same for all three options. The major contributors to the environmental dose commitment are mining, milling, and electricity generation if there is no recycle. Spent fuel reprocessing adds another contributor to the environmental dose commitment if uranium only or uranium and plutonium are recycled. The increase due to reprocessing is offset by lower population dose commitments from mining and milling. Compared to natural background total body dose commitment for the period 1975 through 2000, no recycle is 1.2 percent of natural background. Recycle of uranium only is 1.3 percent and plutonium and uranium recycle is 1.2 percent of natural background.

The individual risk of the nuclear fuel cycle compared to the normal expectancy of the same type of health effect ranges from 0.004 percent to 0.09 percent of the current normal incidence. The maximum individual risks from the nuclear fuel cycle options would be about one percent of the risks from natural background radiation.

The estimated health effects from each fuel cycle option are listed in Table IV J-14 for the U.S. industry from 1975 through 2000. The no recycle case indicates about 1,100 cancers would be induced in the world population and the recycle of plutonium and uranium indicates about 1,200 cancers would be induced for the 26-year period. In view of the uncertainties in the dose estimations, these numbers are considered to be about the same. The radiological aspects of the fuel cycles do not provide a definitive basis for choice of a fuel cycle option.

1.0 INTRODUCTION

The previous sections of this chapter have described the radiological impacts for each facility in the fuel cycle. Each section contains a radiological assessment of the population, occupational, and accidental modes of radiation exposure from that facility for each fuel cycle option. Section J summarizes the radiological impacts from the facilities for each fuel cycle option and estimates the risks from the radiation exposure in terms of health effects. Details of the models and parameters used in the calculational methods are contained in Appendices to Section J. As part of this introduction, the following paragraphs 1.1 and 1.2 provide some general discussion of radiation sources and units before the more specific discussions of radiological impact.

1.1 Sources of Human Radiation Dose

The individual and population receives radiation doses from several sources. At present the largest contributor of radiation dose for most of the population is naturally occurring radiation. The second largest contributor is clinical medical use of radiation. Much smaller quantities of radiation dose are contributed by residual fallout from nuclear weapons testing and the presently existent activities of the nuclear fuel cycle. In the previous sections of CHAPTER IV, the individual doses calculated for normal operations and postulated accident conditions were compared with radiation doses from natural sources. The following paragraphs contain a brief discussion of the sources of natural radiation doses in order to provide more details about this common baseline of comparison.

Natural radiation constitutes the greatest source of ionizing radiation dose to the United States population today. At times referred to as natural background, these sources vary because of a number of factors that are influenced by location and living habits. Some of the more important factors are altitude, geologic features, geographic location and the type of dwelling. Variations in the natural sources of external and internal radiation exposure often exceed the individual exposures from manmade sources that receive more attention. Average dose equivalents from natural sources are estimated by various studies for whole body, gonads, and bone marrow. For whole body dose, the BEIR Report has estimated an average of 102 mrem/yr.[1] In a study by Oakley the average gonadal dose for the United States was estimated as 88 ± 11 mrem/yr at the 95% confidence level.[2] In 1972, UNSCEAR[3] estimated the average world gonadal dose as being 93 mrads/yr.[3] However, the population dose from natural sources varies from 40 to 300 mrem/yr for the United States according to location.[2]

The major components of the external natural radiation exposure are terrestrial radiation and cosmic rays. The terrestrial radiation dose varies with the area of the United States under consideration. The terrestrial radiation doses in the Atlantic and Gulf Coastal Plain average about 23 mrem/yr although doses in the area of the eastern slope of the Rocky Mountains average about 90 mrem/yr.[2] The cosmic ray component varies from 41 mrem/yr at sea level, to 44 mrem/yr at an altitude of 1,000 feet, to 162 mrem/yr at 10,500 feet altitude.[2] In addition to being a source of external radiation, cosmic rays interact with atmospheric components to form radionuclides that contribute to internal radiation exposure from natural sources.

Internal irradiation from natural sources is contributed by tritium, ^{14}C formed by cosmic ray interactions and the primordial radionuclides ^{40}K and ^{87}Rb, and the three radioactive decay series headed by ^{238}U, ^{235}U and ^{232}Th.

Doses from internally deposited naturally occurring radionuclides average 18 mrem/yr gonadal dose.[2,3] Potassium-40 contributes a major portion of this dose and the content in the body varies with diet and location. Other radionuclides contributing to the dose also vary with location. The air and diet contents of ^{238}U, ^{228}Ra, ^{226}Ra, ^{222}Rn, ^{220}Rn, ^{210}Pb and ^{210}Po are dependent on the type of geologic formations in an area.

In GESMO, 100 mrem/yr whole body dose, essentially as reported in BEIR, is used as the typical value for the radiation dose from natural sources.

1.2 Units of Radiation Dose

Throughout this document the word "dose" is usually used as a shortened term for dose equivalents which have units of rem. More commonly, the magnitude of radiation dose equivalents encountered have units of mrem (one thousandth of a rem). Other units that may be encountered are the Roentgen and rad. The definition of the Roentgen specifically limits its application to exposure to gamma or X-radiation. The rad is a measure of the energy absorbed in a substance by any of the ionizing radiations. It is only a few percent greater than the Roentgen, being rounded to the value of 100 ergs of energy absorbed per gram of absorbing material. For the same rad dose, alpha and neutron radiations are generally more effective in producing damage in biological systems. The difference in biological response to various radiations is calculated from the ratio of the rads of neutrons or alpha particles to the rads of gamma or X-rays for the same biological response. This ratio, for human dosimetry purposes, is the quality factor (Q). A distribution factor is considered for dose distribution in tissue. The product of rads, distribution factor and quality factor for a given radiation type is the amount of dose equivalents in rem. Once the doses from several different radiations are expressed in rem, the total dose equivalents to an organ or whole body can be obtained by summation.

The term "man-rem" is derived from the rem and in this report is called "person-rem." It is the summation of all the individual rem doses in a given population. It was devised to assess the exposure of a population for hypothetical evaluations and comparisons of the public health impact of radiological releases. Its validity is contingent upon the premise that the probability (or risk) of a biological effect from radiation is proportionate to the rem dose, regardless of its magnitude or rate of delivery. In this report the term person-rem is also applied to the summation of doses to specific organs in a population. The product of the person-rem for a given organ or system in a population and a risk estimator results in an estimate of the ill health (health effects) attributable to radiation. A complete discussion of the risk estimators and the associated uncertainties is in Appendix B of this section.

The term "dose commitment" as used in this report, is associated with a specified intake of a radionuclide and is defined as the total dose to the total body or to a reference organ, resulting from that intake, which will accrue during the remaining lifetime of the individual. This definition is intended to include the contribution of any radioactive daughters which are formed in the body as the parent nuclide decays. The exposed individual is assumed to be an adult (20 years of age) at the time of intake who will live to an age of 70 years. Thus "dose commitment" in the discussion and accompanying tables is to be interpreted as "50-year dose commitment" during the fortieth year of operation. These dose commitments for one year of model plant operation were integrated over the period 1975 through 2000.

2.0 RADIATION DOSES FROM FUEL CYCLE OPTIONS

2.1 Sources of Radiation Dose

The previous sections of CHAPTER IV have estimated the sources of radiation exposure for the nuclear facilities for each fuel cycle option. The model plants are discussed in Sections C, D, E, F, G, H, I, and K of CHAPTER IV. Control of the radiation exposure of individuals as the nearest neighbor to the plant or as a worker in the plant is incorporated into the design and operation of each plant or facility in the nuclear fuel cycle. From a practical standpoint, full retention is not generally possible. Thus, small quantities of radiation and radioactive materials are released and added to that which naturally occur in the environment. Table IV J-1 is an index of the CHAPTER IV data for the estimated source terms, occupational dose, maximum individual dose, and accident dose for each part of the fuel cycle.

2.2 Population and Occupational Radiation Dose Commitment Estimates

Dose commitments attributed to the low growth projection for nuclear power demand for the U.S. population for total body, bone, thyroid and lung are listed in Tables IV J-2 through IV J-5 integrated over the period 1975 through 2000. Each table indicates the estimated dose commitment contribution from mining, milling, the reactor, reprocessing, and the other components of the fuel cycle for each option. Included in the value for "other" dose commitments are conversion of yellowcake to uranium hexafluoride, enrichment, fuel fabrication, spent fuel storage, transportation, and waste management. The estimated dose commitments for these facilities and the other organs not included here are given in Appendix E, Tables IV J(E)-1 through IV J(E)-8.

The contribution to the dose commitments from plutonium is listed in Tables IV J-2 through IV J-5 for the uranium only recycle and plutonium and uranium recycle fuel cycle options. The dose commitment to the population for the recycle of only uranium is from fuel reprocessing effluents. For the plutonium and uranium recycle option, the plutonium dose commitment is the sum of releases from fuel reprocessing, mixed oxide fuel fabrication, and waste management. The latter contributes negligible amounts of plutonium to the summation of dose commitments. The methodology for summation of the dose commitments is described in CHAPTER VIII.

Table IV J-1

INDEX OF CHAPTER IV DATA

Fuel Cycle	Industry	Emission	Person-rem		Maximum	
Component	Capacity	Curies	Public	Occupational	Individual	Accidents
Mining	Table III-1, 2, 3	§ IV F-1.6.2	Table IV F-4	§ IV F-1.6.3	Table IV F-4	§ IV F-1.6.4
Milling	Table III-1, 2, 3	Table IV F-6	Table IV F-7	§ IV F-2.3.3	Table IV F-7	§ IV F-2.3.4
UF_6 Conversion	Table III-1, 2, 3	Table IV F-9	Table IV F-11	§ IV F-3.3.3	Table IV F-10	§ IV F-3.3.4
Enrichment	Table III-1, 2, 3	§ IV F-4.4.2	Table IV F-13	§ IV F-4.4.3	Table IV F-13	§ IV F-4.4.4
UO_2 Fuel Fabrication	Table III-1, 2, 3	Table IV F-16	Table IV F-17	§ IV F-5.3.3	§ IV F-5.33	§ IV F-5.3.4
MOX Fuel Fabrication	Table III-1, 2, 3	Table IV D-1	Table IV D-2	Table IV-D-2	Table IV D-2	Table IV D-2
LWR Power Generation	Table III-1, 2, 3	§ IV C-5.1	§ IV C-5.2	§ IV C-5.2	§ IV C-5.2	§ IV C-5.4
Irradiated Fuel Storage	Table III-1, 2, 3	Table IV K-6, K-7	§ IV K-2.4	Table IV K-8		§ IV K-3.0
Fuel Reprocessing	Table III-1, 2, 3	Table IV E-1	Table IV E-9, E-10	§ IV E-3.4	Table E-12, E-13	Table IV E-16
Transportation	Table III-1, 2, 3	§ IV G-4.4.3	Table IV G-3, G-4	Table IV G-3, G-4	§ IV G-4.4.3	§ IV G-5.0
Waste Management	Table III-1, 2, 3	§ IV H-3.1.2, 3.2.2	§ IV H-3.2.2	§ IV H-3.2.2	§ IV H-3.2.2	§ IV H-3.1.2, 3.2.2

Table IV J-2

TOTAL BODY DOSE COMMITMENT TO U.S. POPULATION
MILLIONS OF PERSON-REM
U.S. INDUSTRY 1975-2000

Facility	No Recycle	U Recycle	Pu and U Recycle
Mining	3.0	2.6	2.3
Milling	0.58	0.52	0.45
LWR	0.31	0.31	0.31
Reprocessing	-	1.1	1.1
Other	0.045	0.046	0.045
Total	3.9	4.6	4.2
Pu Contribution	-	0.00079	0.0014

Table IV J-3

BONE DOSE COMMITMENT TO U.S. POPULATION
MILLIONS OF PERSON-REM
U.S. INDUSTRY 1975-2000

Facility	No Recycle	U Recycle	Pu and U Recycle
Mining	9.7	8.7	7.5
Milling	1.9	1.7	1.4
LWR	1.1	1.1	1.1
Reprocessing	-	2.6	2.6
Other	0.14	0.14	0.13
Total	13	14	13
Pu Contribution	-	0.034	0.062

Table IV J-4

THYROID DOSE COMMITMENT TO U.S. POPULATION
MILLIONS OF PERSON-REM
U.S. INDUSTRY 1975-2000

Facility	No Recycle	U Recycle	Pu and U Recycle
Mining	0.0073	0.0065	0.0057
Milling	0.0015	0.0014	0.0012
LWR	0.48	0.48	0.49
Reprocessing	-	1.9	1.9
Other	0.00048	0.00049	0.00013
Total	0.49	2.4	2.4
Pu Contribution	-	0.000002	0.00003

Table IV J-5

LUNG DOSE COMMITMENT TO U.S. POPULATION
MILLIONS OF PERSON-REM
U.S. INDUSTRY 1975-2000

Facility	No Recycle	U Recycle	Pu and U Recycle
Mining	0.91	0.81	0.70
Milling	0.18	0.16	0.14
LWR	0.30	0.30	0.31
Reprocessing	-	1.2	1.1
Other	0.0020	0.0019	0.0022
Total	1.4	2.4	2.3
Pu Contribution	-	0.00072	0.0013

Occupational dose commitment estimates for mining, milling, reactors, fuel reprocessing and the other components of the fuel cycle are contained in Tables IV J-6 through IV J-9 for whole body, bone, thyroid and lung exposures. Table IV J-10 lists a summary of the total dose commitments for the U.S. population, occupational workers, and the contribution to foreign populations integrated over the 26-year period between 1975 and 2000. The additional data for other organs and the dose commitments summed for facilities listed as "Other" are in Appendix E, Tables IV J(E)-1 through IV J(E)-8.

Table IV J-6

TOTAL BODY DOSE COMMITMENT TO OCCUPATIONAL WORKERS
MILLIONS OF PERSON-REM
U.S. INDUSTRY 1975-2000

Facility	No Recycle	U Recycle	Pu and U Recycle
Mining	1.2	1.1	0.94
Milling	0.56	0.50	0.44
LWR	2.3	2.3	2.3
Reprocessing	-	0.072	0.15
Other	0.14	0.096	0.086
Total	4.1	4.0	3.9

Table IV J-7

BONE DOSE COMMITMENT TO OCCUPATIONAL WORKERS
MILLIONS OF PERSON-REM
U.S. INDUSTRY 1975-2000

Facility	No Recycle	U Recycle	Pu and U Recycle
Mining	1.7	1.6	1.4
Milling	2.3	2.0	1.8
LWR	2.3	2.3	2.3
Reprocessing	-	0.072	0.15
Other	0.16	0.15	0.15
Total	6.4	6.1	5.7

Table IV J-8

THYROID DOSE COMMITMENT TO OCCUPATIONAL WORKERS
MILLIONS OF PERSON-REM
U.S. INDUSTRY 1975-2000

Facility	No Recycle	U Recycle	Pu and U Recycle
Mining	1.2	1.1	0.94
Milling	0.21	0.19	0.16
LWR	2.3	2.3	2.3
Reprocessing	-	0.072	0.15
Other	0.071	0.065	0.081
Total	3.8	3.7	3.6

Table IV J-9

LUNG DOSE COMMITMENT TO OCCUPATIONAL WORKERS
MILLIONS OF PERSON-REM
U.S. INDUSTRY 1975-2000

Facility	No Recycle	U Recycle	Pu and U Recycle
Mining	6.5	5.8	5.0
Milling	4.8	4.3	3.8
LWR	2.3	2.3	2.3
Reprocessing	-	0.072	0.15
Other	2.1	2.1	1.8
Total	16	14	13

Table IV J-10

SUMMARY OF DOSE COMMITMENT FROM FUEL CYCLE OPTIONS
MILLIONS OF PERSON-REM
U.S. INDUSTRY 1975-2000

	No Recycle	U Recycle	U and Pu Recycle	Difference U and Pu Recycle Less No Recycle
Total Body				
Total U.S. Population (Environmental)	3.9	4.6	4.2	0.3
Total Occupational	4.1	4.0	3.9	-0.2
Total	8.0	8.6	8.0	0
Additional to Foreign Populations	0.21	0.91	0.89	0.7
Bone				
Total U.S. Population (Environmental)	13	14	13	0
Total Occupational	6.4	6.1	5.7	-0.7
Total	19	20	18	-1
Additional to Foreign Populations	1.1	3.3	3.2	2.1
Thyroid				
Total U.S. Population (Environmental)	0.49	2.4	2.4	1.9
Total Occupational	3.8	3.7	3.6	-0.2
Total	4.3	6.1	6.0	1.7
Additional to Foreign Populations	0.21	0.91	0.89	0.7
Lung				
Total U.S. Population (Environmental)	1.4	2.4	2.3	0.9
Total Occupational	16	14	13	-3
Total	17	17	15	-2
Additional to Foreign Populations	0.21	1.3	1.2	1

The dose commitment for the U.S. and foreign populations from each model plant or facility in the nuclear fuel cycle has been estimated on an annual basis in terms of person-rem per GWe-yr. In Appendix E, Tables IV J(E)-9 through IV J(E)-16 list the dose commitments for each nuclear facility. The annual population dose commitments were calculated from the integrated person-rem for the 1975-2000 period and equivalent production capacity of each facility in GWe-yr for the same period. Table IV J(E)-17 contains the 26-year production capacity and the equivalent GWe-yr for the production capacity of each facility.

2.3 Comparison of Dose Commitments

Several comparisons of the dose commitments calculated for the fuel cycle options considered will be discussed in the following paragraphs. However, a comparison with natural radiation dose commitment for the period 1975 to 2000 indicates the projected dose commitments for any of the fuel cycle options is a small fraction of that which all individuals receive naturally. For the 26-year period, the dose commitment from natural background for whole body exposure is estimated to be 650 million person-rem for the U.S. population. The nuclear fuel cycle dose commitment with no recycle is 1.2 percent of the natural background. Recycle of uranium only is 1.3 percent and plutonium and uranium recycle is 1.2 percent of the natural background.

In all fuel cycle options, the major contributors to the environmental whole body dose commitment are mining, milling, and electricity generation.[4] Addition of fuel reprocessing adds another major contributor to environmental whole body dose commitment if uranium alone or plutonium and uranium are recycled. The increase due to reprocessing is offset by lower population dose commitments from uranium mining and milling. Figure IV J-1 illustrates the changes of environmental whole body dose commitment. Also included in Figure IV J-1 is the occupational dose commitment for each fuel cycle option.

The occupational whole body dose commitment from each fuel cycle option is about equal to the population total body dose commitment. For the plutonium and uranium recycle option, the occupational whole body dose commitment is about 93 percent of the environmental dose commitment. The ratio of occupational to population dose commitment varies from 0.34 for kidney to 5.7 for lung if plutonium and uranium are recycled. Appendix E, Table IV J(E)-18 lists the ratios for the different dose commitments for the organs considered. The occupational dose commitments would decrease for uranium recycle alone or for plutonium and uranium recycle because less uranium is mined and milled.

The percent changes in dose commitment for the 1975 through 2000 period with no recycle as a base case are listed in Table IV J-11. The major changes in dose commitments brought about by implementation of uranium recycle only or plutonium and uranium recycle are decreases in total occupational and increases in the dose commitments to foreign populations. In addition the thyroid and lung dose commitments for the U.S. population increase. The increases are due to the additional fuel reprocessing step. For the supporting uranium fuel cycle, the decreased demands for uranium account for the decreased dose commitments. Dose commitments for reactor operations and waste management are generally the same for all of the fuel cycle options. For transportation, the recycle of plutonium and uranium recycle about double the dose commitments based on added operations. Fabrication of plutonium fuels adds a small additional dose commitment.

The contribution of plutonium dose commitment to the total dose commitment is extremely small. The total U.S. population whole body dose commitment for recycle of uranium only is 4.6 million person-rem. Plutonium contributes 0.00079 million person-rem, or is 0.02 percent of the total. For the plutonium and uranium recycle option, the plutonium contributes 0.03 percent. For the bone dose commitment plutonium contributes 0.24 percent in the uranium recycle option and 0.48 percent in the plutonium and uranium recycle option.

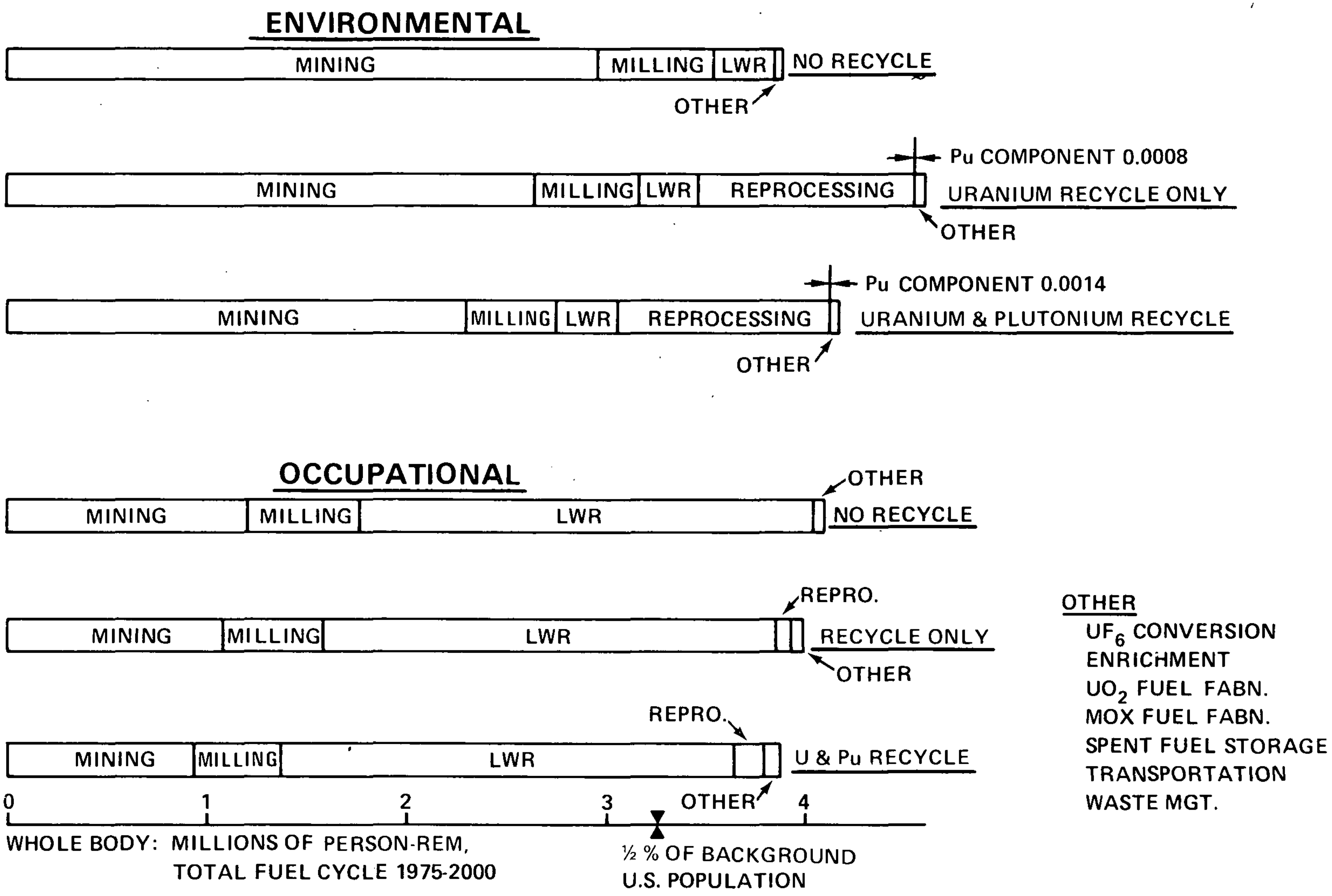

Figure IV J-1. Comparison of U.S. Environmental and Occupational Whole Body Dose Commitments

Table IV J-11

CHANGE IN DOSE COMMITMENT
NO RECYCLE AS BASE CASE
U.S. INDUSTRY 1975-2000

	U Recycle		U and Pu Recycle	
	U Recycle Less No Recycle	U Recycle Percent Change	U and Pu Recycle Less No Recycle	Percent Change
Total Body				
Total U.S. Population	0.7	18	0.3	6.9
Total Occupational	-0.1	-2.4	-0.2	-4.9
Total	0.6	7.5	0	0
Additional to Foreign Populations	0.7	330	0.7	330
Bone				
Total U.S. Population	1	7.7	0	0
Total Occupational	-0.3	-4.7	-0.7	-11
Total	1	3	-1	-5.3
Additional to Foreign Populations	2.2	200	2.1	190
Thyroid				
Total U.S. Population	1.9	390	1.9	390
Total Occupational	-0.1	-2.6	-0.2	-5.3
Total	1.8	42	1.7	40
Additional to Foreign Populations	0.7	330	0.7	330
Lung				
Total U.S. Population	1	71	0.9	64
Total Occupational	-2	-12	-3	-19
Total	0	0	-2	-12
Additional to Foreign Populations	1	500	1	500

2.4 Mitigation of Dose Commitment

In consideration of the overall fuel cycle, technologies are under development that may lead to decreases in dose commitment. These technologies are related to the separation and retention of gaseous radionuclides from fuel reprocessing. The reduction of dose commitments by 90 percent retention of tritium, ^{14}C, and ^{85}Kr is discussed in CHAPTER IV, Section E, Reprocessing Plant Operations.

The contribution of uranium mill tailings to dose commitments is considered in CHAPTER IV, Section F. The contribution to dose commitment by inactive tailings is not included in GESMO. Uranium milling and tailings management practices are to be addressed in a separate generic environmental statement.[5] The statement will address management alternatives for inactive tailings.

3.0 IMPACT DUE TO RADIATION RISKS

The impact of radiation on individuals and populations may be expressed in terms of adverse health effects. The hypothesis employed in this report about radiation risks assumes that the health effects due to radiation are linearly proportional to dose based on an extrapolation of existing high dose response data to low doses and dose rates.

Radiation injury is observed at doses in the order of hundreds of rem administered in minutes or less. Background, in contrast, is in the order of tenths of a rem per year. Theoretical nearest neighbor exposures are identified in the sections of this chapter describing the respective steps in the fuel cycle. The exposures are well below regulatory limits.

3.1 Individual Risks from Radiation

Radiation protection standards have been promulgated in terms of limits of exposure to individuals.[6] For populations, the standard for the average individual has been adjusted to accommodate the expected variations within a population in a manner that would limit individuals above the norm for the population to the maximum individual limit.[8]

The BEIR Report[1] indicates radiation injury risks that might be anticipated from statistical incidences in populations exposed at dose rates which were great enough to produce injury. NCRP Report No. 43[8] cautions against extrapolating risks from high dose rate risks to low dose rate risks without realizing that the validity of such an extrapolation has not been established.

However, in an effort to describe the radiation impact, this analysis has used extrapolated risk estimators for health effects to provide an assessment of the nuclear fuel cycle options. Before discussing the population risks, the individual risks of the normally expected incidence of the health effects identified as possible radiation injury are compared with the expected incidence per person-rem in Table IV J-12. The risk per person-rem in Table IV J-12 is a composite of all ages in the U.S. population. An individual in late life would experience less risk than the composite, and conversely the risk of the young would be somewhat greater.

A comparison of the individual risk for the health effects considered is contained in Table IV J-13 for the normal expectancy, the contribution by the total nuclear fuel cycle from 1975 to 2000, and natural radiation exposure for 70 years of life. The individual risk of the nuclear fuel cycle compared to the normal incidence of the same type of health effect ranges from 0.004 percent to 0.09 percent of the current normal incidence. The individual risk from the nuclear fuel cycle options would be about one percent of the risks from natural background radiation. However, the variations of individual risks from natural radiation would be greater than one percent due to differences between locations caused by geologic formations, altitude, and the types of building materials selected.

Table IV J-12

RISK COMPARISON BETWEEN NORMAL EXPECTATION AND EXPECTATION PER PERSON-REM

	Normal Expectation per U.S. Individual[9]	Expectation per Person-rem**
Total* risk of cancer mortality	1.2×10^{-1}	1.4×10^{-4}
Risk of lung cancer mortality	2.7×10^{-2}	2.2×10^{-5}
Risk of bone cancer mortality	6.3×10^{-4}	6.9×10^{-6}
Risk of thyroid cancer mortality	3.5×10^{-4}	1.3×10^{-5}
Risk of thyroid cancer & benign nodules	3.2×10^{-2}	3.3×10^{-4}
Risk of specific genetic defects	1×10^{-2}	1.6×10^{-4}
Risk of defects with complex etiology	4×10^{-2}	1.0×10^{-4}
Total genetic defect risk	5×10^{-2}	2.6×10^{-4}

---

*Includes leukemia

**Appendix B, this section

Table IV J-13

COMPARISON OF INDIVIDUAL RISKS FROM NORMAL INCIDENCE, LWR POWER INDUSTRY 1975-2000, AND NATURAL RADIATION

	Normal Expectation per U.S. Individual	Individual Risk from Nominal Background 70 year Life Span at 0.1 rem/yr 7 rem	Individual Risk to Average U.S. Resident Dose Commitment* from Total LWR Power Industry 1975-2000 *rem
Total Cancer Mortality	1.2×10^{-1}	9.5×10^{-4}	4.8×10^{-6}
Lung Cancer Mortality	2.7×10^{-2}	1.6×10^{-4}	1.5×10^{-6}
Bone Cancer Mortality	6.3×10^{-4}	4.8×10^{-5}	5.6×10^{-7}
Thyroid Cancer Mortality	3.5×10^{-4}	9.4×10^{-5}	3.0×10^{-7}
Total Thyroid Cancer & Benign Nodule Incidence	3.2×10^{-2}	2.3×10^{-3}	7.6×10^{-6}
Total Genetic Defects	5×10^{-2}	7.7×10^{-4}	2.9×10^{-6}
Specific Genetic Defects	1×10^{-2}	4.7×10^{-4}	1.8×10^{-6}
Complex Etiology Defects	4×10^{-2}	3.0×10^{-4}	1.1×10^{-6}

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*Total Body 0.034, Lung 0.068, Bone 0.081, Thyroid 0.023.

3.2 Impact of Radiation on Populations

Based on the risk estimators in Appendix B of this section, the impact on the U.S. population, occupational workers, and foreign populations is estimated in terms of health effects. The estimated health effects are listed in Table IV J-14.

The data in Table IV J-14 are integrated over 26 years of nuclear power generation based on the low growth projection without FBR. For that period, the estimates indicate the recycle of plutonium and uranium would contribute about 100 additional cancer mortalities and about 200 total genetic defects to the world population. In the U.S. population, the health effects attributable to plutonium and uranium recycle would be about 20 cancers. These estimates are based on absolute risk estimators containing uncertainties of at least one order of magnitude.

The number of health effects estimated are from exposures of very large populations to very small dose commitments. Because of the large populations included in the person-rem calculations it is possible to calculate numerous health effects from any source of radiation. For example, the natural background dose commitment for the U.S. population is estimated as 6.5×10^8 person-rem. For the 26-year period 1975-2000, the number of cancers estimated from natural background is 90,000. The estimated error in the average natural background dose is about 10 percent.[2] The error in the estimated cancers from natural background would be about ± 9,000. The error in the cancers would be part of the expected 10,000,000 cancers in the normal mortality.

The estimated error in cancers from natural background introduces an uncertainty much larger than the estimated cancers from the fuel cycle options. Table IV J-15 contains a comparison of the cancer mortality expectancies in the U.S. population from normal incidence, natural background and the cancer incidence for the no recycle and plutonium and uranium recycle cases. Because of the large uncertainty, the small differences in the estimated health effects provide little basis for selection of a fuel cycle option.

3.3 Uncertainties and Conservatisms

In preparation of the separate sections of CHAPTER IV, the estimates of releases of radioactive materials to the environment contain some conservatism. Where estimates of uncertain future values were made, the upper bound estimate was usually chosen. The health effects estimates based on absolute risk were also upper bound estimates. The final health effects estimates include the conservatisms of all steps in the computations and thus, the final risks are thought to be conservative.

4.0 ACCIDENT CONSIDERATIONS

The consequences of an accident in the respective steps of the nuclear fuel cycle are essentially the same whether fissile materials are recycled in nuclear fuel

Table IV J-14

ESTIMATED HEALTH EFFECTS U.S. INDUSTRY 1975-2000

Type of Health Effect	No Recycle				U Recycle					U and Pu Recycle				
	Occ.	U.S. Non Occ.	Foreign	OPTION Total	Occ.	U.S. Non Occ.	Foreign	OPTION Total	Difference Between Options	Occ.	U.S. Non Occ.	Foreign	OPTION Total	Difference Between Options
Bone Cancer Deaths	45	90	6.9	140	42	97	23	160	20	39	90	22	150	10
Benign and Malignant Thyroid Nodules	1,300	160	69	1,500	1,200	800	300	2,300	800	1,200	800	300	2,300	800
Thyroid Cancer Deaths	51	6.6	2.8	60	50	32	12	94	34	48	32	12	92	32
Lung Cancer Deaths	360	31	4.7	390	330	53	29	420	30	290	51	27	370	-20
Total Cancer Deaths	550	530	28	1,100	540	620	120	1,300	200	530	570	120	1,200	100
Specific Genetic Defects	650	620	33	1,300	630	730	140	1,500	200	620	660	140	1,400	100
Defects with Complex Etiology	410	390	21	820	400	460	91	950	130	390	420	89	900	80
Total Genetic Defects	1,100	1,000	54	2,100	1,000	1,400	170	2,400	300	1,000	1,100	230	2,300	200

Table IV J-15

COMPARISON OF CANCER MORTALITY EXPECTANCY
(1975-2000)

Expected Total Cancer Expectancy in U.S. Population	10,000,000
Estimated Cancers from Natural Background for 1975 through 2000	90,000
Estimated Uncertainty in Cancers from Uncertainty Natural Background	±9,000
Estimated Cancers in U.S. Population from No Recycle*	1,080
Estimated Cancers in U.S. Population from Plutonium and Uranium Recycle*	1,100
Difference Between Plutonium and Uranium Recycle and No Recycle for U.S. Population	20

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*Consideration of the world population would increase these estimates to 1,110 and 1,210 respectively. The difference between plutonium and uranium recycle and no recycle is 100 cancer mortalities.

or not. Characterization of accidents and the radiological impact that major accidents would impose on the environment are presented in CHAPTER IV, Sections C, D, E, F, G, H, I, and K for the various fuel cycle facilities. Reprocessing is an added step in fuel cycle that could involve accidents. However, it reduces the mining, milling, uranium hexafluoride conversion, and enrichment activities. Mixed oxide fuel fabrication is an added step but replaces uranium oxide fuel fabrication.

Experience with a variety of accidents has been gained in the operation of government owned plants. See CHAPTER II. This has resulted in the design and operation of such plants with preventive measures that will nearly eliminate the occurrence of accidents and will mitigate any effects from the higly unlikely events. The low frequency of expected accident occurrence that could have any significant offsite effects during the 26 years reduces the contribution to population doses from accidents below that included in the evaluation of normal operational performance. Therefore, the judgment of the staff is that the conservative evaluations of normal operations already include provisions for any accidental doses.

5.0 PLUTONIUM CONSIDERATIONS

The radiological exposure of individuals and populations by plutonium contributes a fractional percentage of the total dose from the fuel recycle option. Appendix C of this section provides an abstract of the body of knowledge that has been developed about plutonium in the environment and in biota.

The evaluations of plutonium exposures in this analysis used standard NRC methods. The "hot particle" hypothesis which was advanced as a petition for rule making by the Natural Resources Defense Council, Inc., dated February 14, 1974 (Docket No. PRM-20-5) was denied, and the denial was published in the Federal Register on April 12, 1976. The following quote from the Federal Register summarizes the petition.[10]

> "The Natural Resources Defense Council (NRDC) petitioned the Atomic Energy Commission (AEC) to establish specific health protection standards for 'hot particles,' defined by NRDC as particles containing 0.07 picocuries or more of alpha radioactivity and yet sufficiently small to be inhaled and deposited in the lung. The petition contained the following requests:
>
> "1. Stay approvals for new construction or operation of facilities involving 'hot particle' materials, and stay approvals for increase in quantity of 'hot particle' materials for previously approved operations, until the petitioner's requests for modification of associated standards are resolved.
>
> "2. Establish, for occupational exposure, a maximum permissible lung particle burden of two 'hot particles,' and for non-occupational exposure a maximum permissible lung particle burden of 0.2 (average) 'hot particle;' add concentration values to 10 CFR Part 20 for all alpha-emitting radionuclides which could form 'hot particles,' as defined by NRDC, each value to be a factor of 115,000 smaller than the value given for the radionuclide when not in 'hot particle' form.
>
> "3. Establish, for unrestricted areas, a maximum permissible surface contamination level of one 'hot particle' per square meter.
>
> "4. Amend 10 CFR Part 100 by adding a site criterion guide of 10 'hot particles' deposited in the lung during a two-hour exposure under accident conditions.
>
> "5. Convene public hearings to determine as-low-as-practicable regulations for materials in 'hot particle' form. In denying the petition the Nuclear Regulatory Commission (NRC) denies all five of these requests. This follows from the fact that the NRC finds that scientific evidence does not support the technical positions upon which the NRDC petition is based. This technical position is stated by the NRDC in the corollary to the 'hot particle' hypothesis."

The complete Denial of Petition of Rule Making from the Federal Register is included as Appendix D to this section.

REFERENCES

1. U.S. National Academy of Sciences - National Research Council, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," Report of the Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR Report), Washington, 1972.

2. D. T. Oakley, "Natural Radiation Exposure in the United States," U.S. Environmental Protection Agency, Washington, 1972.

3. United Nations, Scientific Committee on the Effects of Atomic Radiation, (UNSCEAR), "Ionizing Radiation Levels and Effects," Vol. 1, Levels, United Nations, New York, 1972.

4. E. E. Pochin, "Estimated Population Exposure from Nuclear Power Production and Other Radiation Sources," Nuclear Energy Agency, Organization for Economic Co-operation and Development, Paris, 1976.

5. Federal Register, "Uranium Milling: Intent to Prepare a Generic Environmental Impact Statement," Washington: 41 FR 22430, June 3, 1976.

6. National Council on Radiation Protection and Measurement, "Basic Radiation Protection Criteria," NCRP Report No. 39(71).

7. Federal Radiation Council, "Background Material for the Development of Radiation Protection Standards," FRC Report No. 1(60) and Report No. 2(61).

8. National Council on Radiation Protection and Measurement, "Review of the Current State of Radiation Protection Philosophy," NCRP Report No. 43(75).

9. U.S. Nuclear Regulatory Commission, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975, Appendix VI, p. 9-38, Table VI 9-9 and p. I-11, Table VI I-1.

10. U.S. Federal Register, "Natural Resources Defense Council (Docket No. PRM-20-5) Denial of Petition for Rule Making," Washington: 41 FR 15371, April 12, 1976.

CHAPTER IV

Section J

Appendix A

DOSE CALCULATION METHODOLOGY

CHAPTER IV
Section J
Appendix A
DOSE CALCULATION METHODOLOGY

1.0 INTRODUCTION

All dose conversion calculations have been made following the recommendations of the International Commission on Radiological Protection, ICRP, unless noted otherwise. Calculations of internal doses were performed using the dosimetric data given by ICRP Committee II[1] and ICRP Publication 6.[2] For radioiodines the biological half-life given in ICRP 6 were used. Table IV J(A)-1 tabulates the updated radioiodine dose factor expressed in the form of concentrations yielding the maximum permissible organ dose of ICRP Committee II, such that a comparison can be made to the maximum permissible concentrations. More recent information contained in ICRP Publication 10[3] was also considered. For those cases in which the more recent data of ICRP 10 caused a significant change (a factor of 2) in dose factors from those implied by ICRP 2 and 6, updated dose factors were prepared. These updated dose factors are shown in Table IV J(A)-1 as maximum permissible concentrations. A few isotopes which occur as daughter products in actinide decay chains are not listed in the ICRP publications. For these isotopes, dose factors were taken from the latest Oak Ridge data contained in the INREM code.[4] These isotopes and the effective maximum permissible concentrations implied by the INREM data are also shown in Table IV J(A)-1. The only isotope for which the internal dose calculations were not made on the basis of dose factors was carbon-14. Carbon-14 dose calculations were made by assuming that all body carbon reaches the isotopic equilibrium of ^{14}C in the air. Doses were then made on the basis of the maximum permissible body burden given in ICRP II.

In summary, except for the specific isotopes listed in Table IV J(A)-1, internal dose calculations have been made using dosimetric data of ICRP Publications 2 and 6. All data on breathing rates, organ masses, etc. for man are those of the standard man of ICRP 2.

2.0 DOSE DEFINITIONS

All doses were calculated to represent a 50-year dose commitment which would be received by an individual or a population from spending one year living in the area of concern. The isotopic concentration levels in the environment used in the dose calculations were those which would exist during the final year of plant lifetime. A 40-year plant operational lifetime was assumed for calculating buildup of long-lived activity in the environment.

Table IV J(A)-1

DOSE FACTORS THAT DIFFER FROM THOSE OF ICRP 2 AND 6

Isotope	Organ of Reference	Maximum Permissible Concentrations Water ($\mu Ci/cm^3$)	Air ($\mu Ci/cm^3$)
^{89}Sr	Bone	3.4×10^{-4}	2.9×10^{-8}
	Total Body	2.7×10^{-3}	2.2×10^{-7}
^{90}Sr	Bone	3.3×10^{-6}	2.7×10^{-10}
	Total Body	2.7×10^{-5}	2.3×10^{-9}
^{132}Te	Thyroid	1.6×10^{-4}	2.1×10^{-8}
^{226}Ra	Bone	1.3×10^{-6}	1.1×10^{-10}
INREM DOSE FACTORS[4]			
^{243m}Pa	Total Body	43	1.9×10^{-6}
	Bone	54	2.5×10^{-5}
	Liver	120	5.0×10^{-6}
	Kidney	31	
	Lung		3.2×10^{-7}
^{234}Pa	Total Body	193	8.5×10^{-6}
	Bone	236	1.1×10^{-5}
	Liver	525	2.2×10^{-5}
	Kidney	139	6.1×10^{-6}
	Lung		1.4×10^{-6}
^{237}U	Total Body	0.57	2.4×10^{-6}
	Bone	1.3	5.4×10^{-6}
	Kidney	0.13	5.8×10^{-7}
	Lung		2.9×10^{-7}
UPDATED IODINE MODELS[1,2]			
^{129}I	Total Body	5.8×10^{-4}	8.3×10^{-3}
	Thyroid	5.2×10^{-6}	7.5×10^{-10}
^{131}I	Total Body	1.8×10^{-3}	2.6×10^{-7}
	Thyroid	2.2×10^{-5}	3.1×10^{-9}
^{133}I	Total Body	8.1×10^{-3}	1.2×10^{-6}
	Thyroid	8.6×10^{-5}	1.2×10^{-8}

The calculated dose represents the dose which would be received by an individual or a population which moved into a plant vicinity during the final year of plant operation, resided in the vicinity for one year, and then moved. In this sense the dose represents the dose received from one year's activity (breathing, eating, etc.). However, the doses include a future integration over the next 50 years of life* from any isotope taken into the body which will continue to reside in the body even after intake has stopped. Thus, the calculated internal doses are 50-year dose commitments which result from a one year exposure.

3.0 INDIVIDUAL AND POPULATION DOSE BASIS

All individual doses from atmospheric releases have been calculated using the average isotopic airborne concentration appropriate to the spatial point being considered. Because siting experience for many of the facilities examined here was very limited, individual calculations were made for the case of an assumed person residing at the immediate site boundary. In these calculations, typical yearly average dilution rates were used for each type facility considered to scope the doses that could be received by an adult individual utilizing the immediate plant environs. See Table IV J(A)-2.

Table IV J(A)-2

GENERALIZED ATMOSPHERIC DISPERSION FACTORS

Meters from Release Point	Annual Average $\bar{\chi}/Q$ Ground Level Release	Annual Average $\bar{\chi}/Q$ 100m Stack Release
100	6.4×10^{-5}	-
300	1.4×10^{-5}	-
500	5.4×10^{-6}	-
1,000	1.5×10^{-6}	1.6×10^{-8}
1,300	1.0×10^{-6}	3.7×10^{-8}
2,500	3.3×10^{-7}	6.7×10^{-8}

Individuals other than adults could receive higher or lower doses depending upon the released radionuclide inventory and existing pathways. In order to present a common basis for comparison, an adult individual was hypothesized to obtain all his food, to breathe, and to reside at one sample location for each facility. Since an adult can be exposed to all food pathways considered, the adult dose will be less

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*In fact, the forward integration over time is already included in the maximum permissible concentrations (MPC's) used by ICRP. The MPC's are used to imply a yearly dose, but the yearly dose is that dose which would be reached after 50 years of continuous uptake. It can be shown that the concentration in the body in the 50th year from a constant unit of uptake over 50 years is numerically equal to the 50-year dose commitment integral resulting from a unit uptake in the first year. Thus, the yearly dose implied by the ICRP MPC's can be used directly to derive 50-year dose commitment values.

sensitive to variations in radionuclide inventory. This decreased sensitivity will result in a more balanced presentation of potential radiological impact. Certainly, in the case of an actual site, the specific pathways and most sensitive individuals would have to be evaluated and the results compared to any applicable regulations. However, for a generic, non-site specific study, the described approach serves to represent approximate impacts for a general site. In addition, comparison of these doses for several recycle modes will indicate the relative radiological impact.

Individual doses from most radioisotopes were based on the airborne concentration in the receptor region and calculated from the standard pathways modeling described subsequently in greater detail. However, doses from noble gases, tritium, and carbon-14 were calculated as special cases. In the case of noble gases, the dose results solely from plume submersion and direct intake by breathing. A semi-infinite plume submersion model was used for the calculations. Doses from carbon-14 were calculated by assuming all carbon in the body reaches the same equilibrium composition of the carbon-14 prevailing in the air at the receptor point. This is a maximum specific activity the body could obtain. The specific activity of ^{14}C in the atmosphere at the receptor point was determined by dividing the ^{14}C activity (in Ci/m^3) by a typical value of 0.16 carbon/m^3 which exists (as CO_2) in air.[5] Doses from tritium were calculated on the basis that concentrations of tritium in plants will reach a limiting concentration of one-half of the concentration in the moisture in air.[6] A value of 8 g water/m^3 in the atmosphere[7] was used to determine the specific activity of tritium in atmosphere water.

Population doses for a given facility can be calculated by summing concentration values times population density over the entire affected population. For generic facilities, the actual population densities and concentration values are not known. Therefore, population doses were estimated using average population densities. These averages represent higher than actual siting population densities.

3.1 Atmospheric Pathway Modeling

The potential exposure pathways which were evaluated for atmospheric releases were inhalation and dietary intake contributions to internal doses and plume submersion and ground plane irradiation contributions to external doses.

3.1.1 Inhalation

Inhalation doses include two different components; inhalation of material soon after release and inhalation of deposited material after resuspension. Annual average concentrations were estimated at the points of interest and individual doses were calculated. For inhalation doses from resuspended material, the resuspension concentration of material from 40 years of continuous deposition was calculated. Based on measurements of Pu resuspension,[8] a resuspension coefficient, K, was estimated to be 10^{-5}/m initially and to decay with a 50-day half-life until it reaches a steady value of 10^{-9}/m. Using this resuspension coefficient and allowing for buildup of daughter products, the resuspension concentration, C^i_{Res}, in air is given by:

$$C^i_{Res} = C^\circ_a \times 10^{-5} \times V_g \times \prod_{k=1}^{i} \lambda_k \sum_{\ell=o}^{i} \frac{1}{\lambda_\ell + \lambda_s} \frac{1}{\prod_{\substack{m=o \\ m \neq \ell}}^{i} (\lambda_m - \lambda_\ell)} +$$

$$C^\circ_a \times 10^{-9} \times V_g \times \prod_{k=1}^{i} \lambda_k \sum_{\ell=o}^{i} \frac{(e^{-\lambda_\ell t_1} - e^{-\lambda_\ell t_2})}{\prod_{\substack{m=o \\ m \neq \ell}}^{i} (\lambda_m - \lambda_\ell)}$$

C^i_{Res} = Resuspension Concentration of Isotope i in air, Ci/m^3

C°_a = the original parent air concentration, Ci/m^3

V_g = the deposition velocity (0.01 m/sec)

λ_k = the radiological decay constants of parents and daughters, sec^{-1}

t_1 = the time for the resuspension coefficient to reach 10^{-9}/m (5.74×10^7 sec)

t_2 = the assumed plant lifetime (40 years)

The parent nuclide is denoted with an index value of 0. When the index is equal to 0, the empty product $[\prod_{k=1}^{o}]$ is equal to unity.

The expression is rather complex, but in the limiting case where a long-lived isotope is deposited and the original isotope is of primary concern, the limiting resuspension concentration will be 0.64 of the initial plume concentration.

In the above formula, deposited isotopes are depleted only by radioactive decay. The decrease in the resuspension coefficient with time is accounted for by the agglomeration of the radioisotopes into larger particles, but not by any removal of the deposited materials. Removal of deposited material would reduce the actual dose.

3.1.2 Dietary Intake

Dietary doses were calculated for intake of vegetation, milk and meat. The individual intake of each type of food used in the calculation was:[9]

Vegetation - 400 grams/day
Milk - 350 grams/day
Meat - 250 grams/day

Isotopes were assumed to be deposited directly on vegetation as well as deposited on soil and taken up by plant roots. Transfers of isotopes from soil to plants and from plants to animals were based on relative concentration data of stable elements in the pathway of concern. The transfer factors used in the calculations are given in Table IV J(A)-3.[10,11,12]

Concentrations of isotopes on the soil assumed buildup of the isotope from continuous deposition over a facility lifetime (40 years). Also included was ingrowth of radioactive daughter products. The following equation was used to estimate the buildup of isotopes on the soil:

$$C_s^i = C_a^o V_g \left[\prod_{j=1}^{i} \lambda_j\right] \sum_{h=o}^{i} \frac{1 - e^{-\lambda_h t}}{\lambda_h \prod\limits_{\substack{p=o \\ p\neq h}}^{i} (\lambda_p - \lambda_h)} \tag{2}$$

C_s^i = Concentration of isotope i on soil, Ci/m^2

C_a^o = the concentration of the parent isotope in air, Ci/m^3

λ = the radiological decay constant per second

V_g = the deposition velocity, m/sec

Since no loss of radioisotopes by weathering or other removal mechanisms is considered, the calculated results are expected to be conservative.

The concentrations of radioisotopes in vegetation which result from buildup of radioisotopes in the soil were determined by assuming a 20 cm soil depth for mixing in the root zone and a soil density of 1.5 g/cm^3, i.e., mixing layer density thickness of 30 g/cm^2. The expression used for estimating the concentration of isotopes in vegetation resulting from deposition onto soil is:

$$C_{s\to v}^i = C_s^i \times U_i/(3 \times 10^5) \tag{3}$$

$C_{s\to v}^i$ = Concentration of isotope i in vegetation, Ci/g - veg

U_i = the transfer factor given in Table IV J(A)-3 for transfer of isotopes from soil to plants, Ci/g - veg ÷ Ci/g - soil

The numerical constant 3×10^5 is the product of the mixing layer density thickness and the conversion of m^2 to cm^2, i.e., $3 \times 10^5 = 30$ g/m^2 $\times 10^4$ m^2/m^2.

Concentrations of isotopes directly deposited on vegetation reflect an effective 13-day weathering removal half-life from plant leaves in addition to the radiological half-life.[13] Deposition onto both soil and vegetation are treated assuming the full original airborne concentration (i.e., deposition of isotopes on the soil was not depleted to account for the isotopes deposited on vegetation before they reach the soil). Therefore, material weathered from the plants to the soil has already been considered. Of the amount directly deposited on vegetation, 30 percent was assumed to be captured by the plant.[13] For plants which will be used as processed foodstuffs,

Table IV J(A)-3

STABLE ELEMENT TRANSFER DATA[10]*

	B_{iv}	$F_{m(Cow)}$	F_f		B_{iv}	$F_{m(Cow)}$	F_f
Elem	Veg/Soil	Milk(D/L)	Meat(D/KG)	Elem	Veg/Soil	Milk(D/L)	Meat(D/KG)
H	4.8E 00	1.0E-02[6]	1.2E-02	Sb	1.1E-02	1.5E-03	4.0E-03
He	5.0E-02	2.0E-02	2.0E-02	Te	1.3E 00	1.0E-03	7.7E-02
Li	8.3E-04	5.0E-02	1.0E-02	I	2.0E-02	6.0E-03[25]	2.9E-03
Be	4.2E-04	1.0E-04	1.0E-03	Xe	1.0E 01	2.0E-02	2.0E-02
B	1.2E-01	2.7E-03	8.0E-04	Cs	1.0E-02	1.2E-02[11]	4.0E-03
C	5.5E 00	1.2E-02	3.1E-02	Ba	5.0E-03	4.0E-04[11]	3.2E-03
N	7.5E 00	2.2E-02	7.7E-02	La	2.5E-03	5.0E-06	2.0E-04
D	1.6E 00	2.0E-02	1.6E-02	Ce	2.5E-03	6.0E-04[11]	1.2E-03
F	6.5E-04	1.4E-02	1.5E-01	Pr	2.5E-03	5.0E-06	4.7E-03
Ne	1.4E-01	2.0E-02	2.0E-02	Nd	2.4E-03	5.0E-06	3.3E-03
Na	5.2E-02	4.0E-02[11]	3.0E-02	Pm	2.5E-03	5.0E-06	4.8E-03
Mg	1.3E-01	1.0E-02	5.0E-03	Sm	2.5E-03	5.0E-06	5.0E-03
Al	1.8E-04	5.0E-04	1.5E-03	Eu	2.5E-03	5.0E-06	4.8E-03
Si	1.5E-04	1.0E-04	4.0E-05	Gd	2.5E-03	5.0E-06	3.6E-03
P	1.1E 00	2.5E-02	4.6E-02	Tb	2.5E-03	5.0E-06	4.4E-03
S	5.9E-01	1.8E-02	1.0E-01	Dy	2.5E-03	5.0E-06	5.3E-03
Cl	5.0E 00	5.0E-02	8.0E-02	Hd	2.5E-03	5.0E-06	4.4E-03
Ar	6.0E-01	2.0E-02	2.0E-02	Er	2.5E-03	5.0E-06	4.0E-03
K	3.7E-01	1.0E-02	1.2E-02	Tm	2.5E-03	5.0E-06	4.4E-03
Ca	3.6E-02	8.0E-03[11]	4.0E-03	Yb	2.5E-03	5.0E-06	4.0E-03
Sc	1.1E-03	5.0E-06	1.6E-02	Lu	2.5E-03	5.0E-06	4.4E-03
Ti	5.4E-05	5.0E-06	3.1E-02	Hf	1.7E-04	5.0E-06	4.0E-01
V	1.3E-03	1.0E-03	2.3E-03	Ta	6.3E-03	2.5E-02	1.6E 00
Cr	2.5E-04	2.2E-03	2.4E-03	W	1.8E-02	5.0E-04	1.3E-03
Mn	2.9E-02	2.5E-04	8.0E-04	Re	2.5E-01	2.5E-02	8.0E-03
Fe	6.6E-04	1.2E-03	4.0E-02	Ds	5.0E-02	5.0E-03	4.0E-01
Cd	9.4E-03	1.0E-03	1.3E-02	Ir	1.3E 01	5.0E-03	1.5E-03
Ni	1.9E-02	6.7E-03	5.3E-03	Pi	5.0E-01	5.0E-03	4.0E-03
Cu	1.2E-01	1.4E-02	8.0E-03	Au	2.5E-03	5.0E-03	8.0E-03
Zn	4.0E-01	3.9E-02	3.0E-02	Hg	3.8E-01	3.8E-02	2.6E-01
Ga	2.5E-04	5.0E-05	1.3E 00	Tl	2.5E-01	2.2E-02	4.0E-02
Ge	1.0E-01	5.0E-04	2.0E-01	Pb	6.8E-02	6.2E-04	2.9E-04
As	1.0E-02	6.0E-03	2.0E-03	Bi	1.5E-01	5.0E-04	1.3E-02
Se	1.3E 00	4.5E-02	1.5E-02	Po	1.5E-01	3.0E-04	1.2E-02
Br	7.6E-01	5.0E-02	2.6E-02	At	2.5E-01	5.0E-02	8.0E 00
Kr	3.0E 00	2.0E-02	2.0E-02	Rn	3.5E 00	2.0E-02	2.0E-02
Rb	1.3E-01	3.0E-02	3.1E-02	Fr	1.0E-02	5.0E-02	2.0E-02[12]
Sr	1.7E-02	8.0E-04[11]	6.0E-04	Ra	3.1E-04	8.0E-03[11]	3.4E-02
Y	2.6E-03	1.0E-05	4.6E-03	Ac	2.5E-03	5.0E-06	6.0E-02[12]
Zr	1.7E-04	5.0E-06	3.4E-02	Th	4.2E-03	5.0E-06	2.0E-04[12]
Nb	9.4E-03	2.5E-03	2.8E-01	Pa	2.5E-03	5.0E-06	8.0E 02
Mo	1.2E-01	7.5E-03	8.0E-03	U	2.5E-03	5.0E-04	3.4E-04[12]
Tc	2.5E-01	2.5E-03	4.0E-01	Np	2.5E-03	5.0E-06	2.0E-04[12]
Ru	5.0E-02	1.0E-06	4.0E-01	Pu	2.5E-04	2.0E-06	1.4E-05[12]
Rh	1.3E 01	1.0E-02	1.5E-03	Am	2.5E-04	5.0E-06	2.0E-04[12]
Pd	5.0E 00	1.0E-02	4.0E-03	Cm	2.5E-03	5.0E-06	2.0E-04[12]
Ag	1.5E-01	5.0E-02	1.7E-02	Bk	2.5E-03	5.0E-06	2.0E-04[12]
Cd	3.0E-01	1.2E-04	5.3E-04	Cf	2.5E-03	5.0E-06	2.0E-04[12]
In	2.5E-01	1.0E-04	8.0E-03	Es	2.5E-03	5.0E-06	2.0E-04[12]
Sn	2.5E-03	2.5E-03	8.0E-02	Fm	2.5E-03	5.0E-06	2.0E-04[12]

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*2.0E 02 is 2.0 x 10^{-2}

Note: Superscript numbers refer to references.

a plant density of 2,300 grams vegetation/m^2 was used to define plant uptake from direct deposition.[13] For grasses on pastureland used to define part of a cow's uptake, a vegetation density of 440 grams/m^2 vegetation was used.[14] These numbers are typical of average agricultural and pasturelands. The expression for the concentration of isotopes in processed food material is:

$$C_v^i = C_a^i \, x V_g / 7700 \, (\lambda_w + \lambda_r^i) \qquad (4)$$

where

C_v^i = concentration of isotope i in vegetation, Ci/g - veg

C_a^i = the isotopic concentration in the air, Ci/m^3

λ_w = the weathering decay constant, 6.17×10^{-7}, sec^{-1}

λ_r^i = the radiological decay constant, sec^{-1}

The constant 7700 is the ratio of the vegetation density and the fractional capture, i.e., 7700 = 2300 g/m^2/0.3.

Deposition of resuspended material on vegetation is accounted for by including resuspension concentrations in the airborn isotopic concentrations (i.e., C_a^i including C_{res}^i of equation 3 plus C_a^o, if appropriate). The concentration of isotopes in pastureland vegetation, C_{pv}^i, is given by a similar expression:

$$C_{pv}^i = C_a^i \times V_g / 1500 \, (\lambda_w + \lambda_r^i) \qquad (5)$$

C_{pv}^i = Concentration of isotope i in pastureland vegetation, Ci/gm. veg.

The constant 1500 is the quotient of the pasture land density and the fractional capture, i.e., 1500 = 440 g/cm^2/0.3. The total dietary human uptake from vegetation is thus the sum of $C_v + C_{s \rightarrow v}$ times the assumed diet of 400 g.veg/day. A decay time of 7 days was assumed in transfer of vegetation foodstuffs from the field to their ultimate use by the consumer.

For animal uptake of vegetation which might result in transfer of isotopes to man from meat or milk, a total animal uptake value of 50 kg per day was assumed. To maintain a high productivity, animals are generally offered feeds such as harvested forages and grains to supplement or replace the pasture intake.[14]

Milk animals were assumed to obtain 80% of the daily intake from pasture and 20% of the daily intake from harvested forages and grains. Beef animals were assumed to obtain 50% of the daily intake from pasture and 50% from harvested forages and grains. No radioactive decay time was considered between deposition on pastureland

and animal consumption, but a 2 month delay was assumed between harvest of forages and grains and their consumption by the animal.

The total human uptake of isotopes from the meat and milk pathways was calculated from the total daily animal uptake of isotopes given above using the meat and milk transfer factors given in Table IV J(A)-3. Delay times between consumption of vegetation by an animal and ultimate consumption of meat or milk from that animal of one day for milk and 7 days for meat were used.

3.1.3 Plume Submersion

Plume submersion doses were calculated on the basis of a semi-infinite cloud model. Total body and organ doses were calculated on the basis of the total energy deposited with 5 cm of tissue shielding.[15] Lung doses included both the external dose as well as the dose from inhaled material followed the treatment of Snyder.[16] All gamma and beta energy from inhaled material is deposited in the total lung mass with no attenuation. Skin doses were calculated for gamma and beta energies using a 7 mg/cm^2 dead skin layer attenuation.[15] The dose conversion factors used for plume submersion are shown in Table IV J(A)-4.

Table IV J(A)-4

PLUME SUBMERSION DOSE FACTORS

		Dose = Rem per Ci-sec/m^3	
Isotope	Whole Body	Lung	Skin
^{39}Ar	9.2×10^{-5}	5.3×10^{-4}	3.3×10^{-2}
^{41}Ar	3.1×10^{-1}	4.4×10^{-1}	3.1×10^{-1}
^{83m}Kr	2.4×10^{-6}	9.4×10^{-5}	6.5×10^{-4}
^{85m}Kr	3.7×10^{-2}	3.7×10^{-2}	8.9×10^{-2}
^{85}Kr	5.1×10^{-4}	1.1×10^{-3}	4.3×10^{-2}
^{87}Kr	1.9×10^{-1}	1.9×10^{-1}	2.5×10^{0}
^{88}Kr	4.7×10^{-1}	4.7×10^{-1}	6.1×10^{-1}
^{89}Kr	5.3×10^{-1}	5.3×10^{-1}	9.2×10^{-1}
^{90}Kr	5.0×10^{-1}	5.0×10^{-1}	8.1×10^{-1}
^{131m}Xe	2.9×10^{-3}	3.3×10^{-3}	2.1×10^{-2}
^{133m}Xe	8.0×10^{-3}	8.4×10^{-3}	4.3×10^{-2}
^{133}Xe	9.3×10^{-3}	9.6×10^{-3}	2.2×10^{-2}
^{135m}Xe	9.9×10^{-2}	9.9×10^{-2}	1.4×10^{-1}
^{135}Xe	4.4×10^{-2}	5.8×10^{-2}	1.3×10^{-1}
^{137}Xe	4.5×10^{-2}	4.9×10^{-2}	4.4×10^{-1}
^{138}Xe	2.8×10^{-1}	2.8×10^{-1}	4.4×10^{-1}

3.1.4 Ground Plane Irradiation

Ground plane irradiation dose calculations for most isotopes were made following the treatment used in the Oak Ridge EXREM III[17] code. This consists of determining the gamma fluence at 100 cm above an assumed infinite groundplane. Buildup of long-lived activity on the ground from 40 years of continuous deposition, including ingrowth of daughter products, was treated using equation (2). No beta doses from ground plane irradiation were treated because vegetation on the ground, clothing, and the travel distance in air all combine to make this dose contribution very small. The total body dose factors used in the calculations are shown in Table IV J(A)-5. Most of these dose factors were taken from the EXREM data compilation. However, the few isotopes listed in Table IV J(A)-5 that were not included in the EXREM data were taken from other sources.[13]

3.1.5 Special Consideration of ^{222}Rn from Uranium Mines and Mill Tailings Piles

The radon emanation from a source was considered to be unaccompanied by the radioactive decay descendants (daughters). The daughter product ingrowth was calculated as the ^{222}Rn traveled from the point of release to the receptor. For individual doses, the hypothetical receptor is located 500 meters from the mine and 1,300 meters from the mill tailings piles. Ingrowth of daughter products is estimated using a transport rate of 2 meters/sec.

Plume submersion doses attributable to ^{222}Rn daughters and resuspension and ground plane irradiation from deposited ^{210}Pb and its descendants were also evaluated.

3.2 Hydrological Pathway Modeling

The pathways to man analyzed for radionuclides released to waterways were consumption of drinking water, fish, irrigated crops, beef, and milk contributions to internal doses and shore line use contribution to external dose.

3.2.1 Drinking Water Dilution

For maximum individual dose estimations, the radioactive effluent was assumed to undergo a prompt dilution from an average river flow of 200 cfs. This flow value was used to calculate the radionuclide concentration in the immediate discharge region. Maximum doses to individuals consuming 700 liters/year of drinking water from this location were calculated directly from the average radionuclide concentration after allowing for a 12-hour delay.

3.2.2 Fish Consumption

Based on the annual average concentration in the discharge region, the quantity of radionuclides in fish flesh was estimated using the biological accumulation factors[21] listed in Table IV J(A)-6. Thus,

$$C_i^f = 5.6 \times 10^{-9} \times Q_i \times (BAF)_i \qquad (6)$$

Table IV J(A)-5

GROUND SHINE DOSE FACTORS

Nuclide	Dose Factor Rem per Ci-sec/m^2	Nuclide	Dose Factor Rem per Ci-sec/m^2
^{3}H	0	^{99m}Tc	5.2×10^{-4}
^{14}C	0	^{99}Tc	0
^{22}Na	9.2×10^{-3}	^{103}Ru	2.0×10^{-3}
^{24}Na	1.4×10^{-2}	^{106}Ru	0
^{51}Cr	1.9×10^{-4}	^{103m}Rh	1.2×10^{-5}
^{54}Mn	3.2×10^{-3}	^{110m}Ag	1.1×10^{-2}
^{56}Mn	3.1×10^{-3}	^{125}Sb	2.0×10^{-3}
^{55}Fe	5.9×10^{-6}	^{127m}Te	8.4×10^{-5}
^{59}Fe	4.7×10^{-3}	^{127}Te	1.7×10^{-5}
^{58}Co	3.6×10^{-3}	^{129m}Te	2.3×10^{-4}
^{60}Co	9.8×10^{-3}	^{129}Te	3.3×10^{-4}
^{65}Zn	2.3×10^{-3}	^{131m}Te	5.4×10^{-3}
^{69m}Zn	1.7×10^{-3}	^{132}Te	8.0×10^{-4}
^{69}Zn	0	^{129}I	1.8×10^{-4}
^{86}Rb	3.5×10^{-4}	^{131}I	1.6×10^{-3}
^{89}Sr	0	^{132}I	9.1×10^{-3}
^{90}Sr	0	^{133}I	2.7×10^{-3}
^{91}Sr	2.9×10^{-3}	^{135}I	6.8×10^{-3}
^{90}Y	0	^{134}Cs	6.1×10^{-3}
^{91m}Y	2.2×10^{-3}	^{136}Cs	8.1×10^{-3}
^{91}Y	1.0×10^{-5}	^{137}Cs	2.3×10^{-3}
^{92}Y	4.4×10^{-4}	^{140}Ba	8.8×10^{-4}
^{95}Zr	2.9×10^{-3}	^{140}La	9.1×10^{-3}
^{97}Zr	7.1×10^{-4}	^{141}Ce	3.2×10^{-4}
^{95}Nb	2.9×10^{-3}	^{143}Ce	1.3×10^{-3}
^{97}Nb	2.7×10^{-3}	^{144}Ce	8.9×10^{-5}
^{99}Mo	7.0×10^{-4}	^{143}Pr	0
		^{147}Nd	6.2×10^{-4}
^{147}Pm	0	^{232}U	1.9×10^{-7}
^{148}Pm*	7.9×10^{-8}	^{233}U	2.5×10^{-5}
^{151}Sm	1.8×10^{-8}	^{234}U	1.5×10^{-5}
^{153}Sm	8.6×10^{-7}	^{235}U	9.0×10^{-4}
^{154}Eu	5.1×10^{-3}	^{236}U	1.0×10^{-5}
^{155}Eu	2.9×10^{-4}	^{237}U	1.2×10^{-3}
^{210}Pb	7.2×10^{-5}	^{238}U	1.0×10^{-5}
^{212}Pb	6.4×10^{-4}	^{237}Np	4.2×10^{-4}
^{210}Bi	0	^{239}Np	7.9×10^{-4}
^{212}Bi	4.8×10^{-4}	^{238}Pu	1.0×10^{-5}
^{224}Ra	4.2×10^{-5}	^{239}Pu	4.3×10^{-6}
^{226}Ru	3.0×10^{-5}	^{240}Pu	9.0×10^{-6}

Table IV J(A)-5 (Continued)

Nuclide	Dose Factor Rem per Ci-sec/m^2	Nuclide	Dose Factor Rem per Ci-sec/m^2
^{228}Ra	2.1×10^{-4}	^{241}Pu	0
^{228}Ac	2.9×10^{-3}	^{242}Pu	8.6×10^{-6}
^{228}Th	5.3×10^{-5}	^{241}Am	2.4×10^{-4}
^{230}Th	1.9×10^{-5}	^{242}Am	1.8×10^{-4}
^{231}Th	4.1×10^{-4}	^{243}Am	2.1×10^{-4}
^{232}Th	1.1×10^{-5}	^{242}Cm	9.2×10^{-6}
^{234}Th	5.3×10^{-5}	^{243}Cm	7.5×10^{-5}
^{233}Pa	1.0×10^{-3}	^{244}Cm	5.7×10^{-5}
^{234m}Pa	1.4×10^{-6}		
^{234}Pa	1.2×10^{-2}		

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*Dose factors for these isotopes from other data,[13] the remaining factors from EXREM data.[17]

Table IV J(A)-6

BIOACCUMULATION FACTORS[21]

(pCi/kg per pCi/liter)

Element	Fish	Invertebrate	Plant
H	9.0×10^{-1}	9.0×10^{-1}	9.0×10^{-1}
He	1.0	1.0	1.0
Li	5.0×10^{-1}	4.0×10^{-1}	3.0
Be	2.0	1.0×10^{1}	2.0×10^{1}
B	2.2×10^{-1}	5.0×10^{1}	2.2
C	4.6×10^{3}	9.1×10^{3}	4.6×10^{3}
N	1.5×10^{5}	1.5×10^{5}	1.3×10^{4}
O	9.2×10^{-1}	9.2×10^{-1}	9.2×10^{-1}
F	1.0×10^{1}	1.0×10^{2}	2.0
Ne	1.0	1.0	1.0
Na	1.0×10^{2}	2.0×10^{2}	5.0×10^{2}
Mg	5.0×10^{1}	1.0×10^{2}	1.0×10^{2}
Al	1.0×10^{1}	6.3×10^{1}	4.2×10^{-2}
Si	2.5	2.5×10^{1}	1.3×10^{2}
P	1.0×10^{5}	2.0×10^{4}	5.0×10^{5}
S	7.5×10^{2}	1.0×10^{2}	1.0×10^{2}
Cl	5.0×10^{1}	1.0×10^{2}	5.0×10^{1}
Ar	1.0	1.0	1.0
K	1.0×10^{3}	8.3×10^{2}	6.7×10^{2}
Ca	4.0×10^{1}	3.3×10^{2}	1.3×10^{2}
Sc	2.0	1.0×10^{3}	1.0×10^{4}
Ti	1.0×10^{3}	3.0×10^{3}	5.0×10^{2}

Table IV J(A)-6 (Continued)

Element	Fish	Invertebrate	Plant
V	$1.0x10^{1}$	$3.0x10^{3}$	$1.0x10^{2}$
Cr	$2.0x10^{2}$	$2.0x10^{3}$	$4.0x10^{3}$
Mn	$4.0x10^{2}$	$9.0x10^{4}$	$1.0x10^{4}$
Fe	$1.0x10^{2}$	$3.2x10^{3}$	$1.0x10^{3}$
Co	$5.0x10^{1}$	$4.0x10^{2}$	$2.0x10^{2}$
Ni	$1.0x10^{2}$	$1.0x10^{2}$	$5.0x10^{1}$
Cu	$5.0x10^{1}$	$4.0x10^{2}$	$2.0x10^{3}$
Zn	$2.0x10^{3}$	$1.0x10^{4}$	$2.0x10^{4}$
Ga	$3.3x10^{2}$	$6.7x10^{2}$	$1.7x10^{3}$
Ge	$3.3x10^{3}$	$3.3x10^{1}$	$3.3x10^{1}$
As	$1.0x10^{2}$	$4.0x10^{1}$	$3.0x10^{3}$
Se	$1.7x10^{2}$	$1.7x10^{2}$	$1.0x10^{3}$
Br	$4.2x10^{2}$	$3.3x10^{2}$	$5.0x10^{1}$
Kr	1.0	1.0	1.0
Rb	$2.0x10^{3}$	$1.0x10^{3}$	$1.0x10^{3}$
Sr	$3.0x10^{1}$	$1.0x10^{2}$	$5.0x10^{2}$
Y	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Zr	3.3	6.7	$1.0x10^{3}$
Nb	$3.0x10^{4}$	$1.0x10^{2}$	$8.0x10^{2}$
Mo	$1.0x10^{1}$	$1.0x10^{1}$	$1.0x10^{3}$
Tc	$1.5x10^{1}$	5.0	$4.0x10^{1}$
Ru	$1.0x10^{1}$	$3.0x10^{2}$	$2.0x10^{3}$
Rh	$1.0x10^{1}$	$3.0x10^{2}$	$2.0x10^{2}$
Pd	$1.0x10^{1}$	$3.0x10^{2}$	$2.0x10^{2}$
Ag	2.3	$7.7x10^{2}$	$2.0x10^{2}$
Cd	$2.0x10^{2}$	$2.0x10^{3}$	$1.0x10^{3}$
In	$1.0x10^{5}$	$1.0x10^{5}$	$1.0x10^{5}$
Sn	$3.0x10^{3}$	$1.0x10^{3}$	$1.0x10^{2}$
Sb	1.0	$1.0x10^{1}$	$1.5x10^{3}$
Te	$4.0x10^{2*}$	$1.0x10^{5}$	$1.0x10^{2*}$
I	$1.5x10^{1}$	5.0	$4.0x10^{1}$
Xe	1.0	1.0	1.0
Cs	$2.0x10^{3}$	$1.0x10^{2}$	$5.0x10^{2}$
Ba	4.0	$2.0x10^{2}$	$5.0x10^{2}$
La	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Ce	1.0	$1.0x10^{3}$	$4.0x10^{3}$
Pr	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Nd	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Ph	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Sm	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Eu	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Gd	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Tb	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$

Table IV J(A)-6 (Continued)

Element	Fish	Invertebrate	Plant
Dy	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Ho	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Er	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Tm	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Yb	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Lu	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Hf	3.3	6.7	$1.0x10^{3}$
Ta	$3.0x10^{4}$	$6.7x10^{2}$	$8.0x10^{2}$
W	$1.2x10^{3}$	$1.0x10^{1}$	$1.2x10^{3}$
Re	$1.2x10^{2}$	$6.0x10^{1}$	$2.4x10^{2}$
Os	$1.0x10^{1}$	$3.0x10^{2}$	$2.0x10^{2}$
Ir	$1.0x10^{1}$	$3.0x10^{2}$	$2.0x10^{2}$
Pt	$1.0x10^{2}$	$3.0x10^{2}$	$2.0x10^{2}$
Au	$3.3x10^{1}$	$5.0x10^{1}$	$3.3x10^{1}$
Hg	$1.0x10^{3}$	$1.0x10^{5}$	$1.0x10^{3}$
Tl	$1.0x10^{4}$	$1.5x10^{4}$	$1.0x10^{5}$
Pb	$1.0x10^{2}$	$1.0x10^{2}$	$2.0x10^{2}$
Bi	$1.5x10^{1}$	$2.4x10^{1}$**	$2.4x10^{1}$**
Pd	$5.0x10^{2}$	$2.0x10^{4}$	$2.0x10^{3}$
At	$1.5x10^{1}$	5.0	$4.0x10^{1}$
Rn	$5.7x10^{1}$	1.0	1.0
Fr	$4.0x10^{2}$	$1.0x10^{2}$	$8.0x10^{1}$
Ra	$5.0x10^{1}$	$2.5x10^{2}$	$2.5x10^{3}$
Ac	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Th	$3.0x10^{1}$	$5.0x10^{2}$	$1.5x10^{3}$
Pa	$1.1x10^{1}$	$1.1x10^{2}$	$1.1x10^{3}$
U	2.0	$6.0x10^{1}$	$5.0x10^{-1}$
Np	$1.0x10^{1}$	$4.0x10^{2}$	$3.0x10^{2}$
Pu	3.5	$1.0x10^{2}$	$3.5x10^{2}$
Am	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Cm	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Bk	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Cf	$2.5x10^{1}$	$1.0x10^{3}$	$5.0x10^{3}$
Es	$1.0x10^{1}$	$1.0x10^{2}$	$1.0x10^{3}$
Fm	$1.0x10^{1}$	$1.0x10^{2}$	$1.0x10^{3}$

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*ORNL - Private Communication

**Derived from data in Bower, H.J.M. Trace Elements in Biochemistry, New York, Academic Press (1966).

where

C_i^f = concentration of radionuclide i in fish flesh, μCi/g

Q_i = release rate of radionuclide i, Ci/yr

$(BAF)_i$ = biological accumulation factor for radionuclide i, Ci/gm per Ci/cc

The constant 5.6×10^{-9} reflects the conversion of the release rate to μCi/sec and the flow rate to cc/sec, $5.6 \times 10^{-9} = 10^6$ μCi/c x yr/3.15×10^7 sec x sec/200 ft^3 x ft^3/2.83×10^4 cc.

In estimating maximum doses to individuals, it was assumed that an adult consumed 20 kg/year of fish caught in the discharge region. A decay time of 24 hours was assumed between catching and consuming the fish.

3.2.3 Irrigated Vegetation

Individuals can also receive doses from consuming food crops irrigated with water containing released radionuclides. Radioactive material in vegetation results from deposition onto the plant foliage and from uptake from the soil of activity deposited on the ground. The concentration of any radionuclide except tritium in vegetation is given by:

$$C_i^v = C_i^w \, I \quad \frac{0.25\,[1-\exp\,(-\lambda_{Ei}\;t_e)]}{Y_v\;\lambda_{Ei}} + \frac{B_i^v\,[1-\exp\,(-\lambda_i\;t_b)]}{P\lambda_i} \quad \exp\,(-\lambda_i t_h) \qquad (7)$$

where

C_i^v = concentration of radionuclide i in edible portion of crop v, pCi/kg

C_i^w = concentration of radionuclide i in irrigating water, pCi/ℓ

I = irrigation rate, l/m^2/hr

λ_{Ei} = effective removal constant for radionuclide i from crops, hr^{-1}; $\lambda_{Ei} = \lambda_i + \lambda_w$ where λ_w is the removal constant for loss by weathering (removal half-life of 13 days) and λ_i is the radioactive decay constant

C_i^f = concentration of radionuclide i in feed for beef cattle <u>only</u>, pCi/kg

t_e = time that crops are exposed to contaminated water, hr

Y_v = agricultural productivity, kg (wet weight)/m^2

B_i^v = transfer factor for radionculide i from soil to edible plant parts, pCi/kg (wet weight of plants) per pCi/kg (dry soil)

t_b = midpoint of the soil exposure time, hr

P = effective "surface concentration" of soil, kg (dry soil)/m^2. By assuming a uniform mixing of all radionuclides in a plowlayer of 15 cm, P is equal to 240 kg/m^2

t_h = time interval between harvest and consumption of food, hours

(for tritium,

$C^v = C^w$.)

3.2.4 Beef and Milk Pathways

The consumption of milk and meat produced by animals consuming contaminated forage and drinking water can result in doses to man. For this evaluation, it was assumed that meat animals consumed contaminated forage and water. Dairy animals were assumed to be consuming contaminated drinking water only. The radionuclide concentration in animal products is proportional to the animal's intake of the radionuclide in feed and water. Thus, for all radionuclides except tritium,

$$C_i^A = S_i^A \, [C_i^f + 60 \, C_i^w] \times 10^3 \qquad (8)$$

where

C_i^A = concentration in animal product, pCi/l or pCi/kgm

S_i^A = transfer factor for radionuclide i (listed in Table IV J(A)-3), pCi/cc milk per pCi/day intake or pCi/kg meat per pCi/kgm veg

C_i^f = concentration in feed, pCi/kgm

C_i^w = concentration in water, pCi/l

The factor of 60 is the animal's daily water intake (l/d).

For tritium,

$$C^A = 3k \, C^W \, (w + 60) \qquad (9)$$

where

C^A = tritium concentration in milk or meat, pCi/l or pCi/kg

k = reciprocal of body water volume (0.0041 l^{-1} for beef cattle and 0.0028 l^{-1} for dairy cattle)

C^w = tritium concentration in water, pCi/l

w = water intake rate via fresh forage (28 l/day for beef cattle <u>only</u>)

The factor of 3 is the inverse of the water elimination rate (day).

3.2.5 External Exposure

External exposures to individuals utilizing the riverbank for recreational purposes for 70 hours/year were also evaluated using the following expression:

$$S_i = 20\ T_i\ C_i^w\ [1 \quad \exp\ (-\lambda_i t)] \tag{10}$$

where

S_i = shoreline surface contamination of radionuclide i, pCi/m^2

T_i = radiological half-life of radionuclide i, day

C_i^w = concentration in river of radionuclide i, pCi/l

λ_i = radiological decay constant for radionuclide i, hour^{-1}

t = length of time sediment is exposed to contaminated water, hours (assumed to be midpoint of facility operating life - 20 years)

The factor 20 is an experimentally derived constant. Individual doses are calculated from the shoreline surface concentrations using the dose factors listed in Table IV J(A)-5.

4.0 POPULATION DOSES

4.1 Atmospheric Pathway

Expressions for calculating population dose can be derived as follows:

At any time, t, the amount of radionuclide (i) in the environment is:

$$Q_i\ (t) = 3.2 \times 10^{-8}\ \dot{Q}_i\ (1-e^{-\lambda_i t})/\lambda_i \tag{11}$$

where

Q_i (t) = the amount of radionuclide i in the environment at time t, Ci

$\dot{Q}_i$ = the release rate of radionuclide i, Ci/yr

λ_i = the radioactive decay constant for radionuclide i, sec^{-1}

The constant 3.2×10^{-8} is the conversion from years to seconds, i.e., 3.2×10^{-8} yr/sec. Populations can be directly exposed to radionuclides deposited onto the ground. The surface concentration, C_{Ai}, over an arbitrary area A is:

$$C_{Ai}\ (t) = 3.2 \times 10^{-8}\ \dot{Q}_i\ (1-e^{-\lambda_i t})/A\lambda_i \tag{12}$$

where

C_{Ai} (t) = the areal concentration of radionuclide (i) at time t, Ci/m^2

A = the area of interest, m^2

Total body doses to individuals standing on the contaminated surface were calculated using the radionuclide dose constants found in Table IV J(A)-5 with following expression:

$$D_{Gi} = K_{Gi} \cdot C_{Ai}(t) \tag{13}$$

where

K_{Gi} = dose factor for radionuclide i, mrem/yr per Ci/m^2

The population dose is then:

$$T_{Gi} = P \cdot D_{Gi} \tag{14}$$

where

P = total number of exposed people

Since $P/A = \rho$, where ρ is the population density in units of people per square meter, the population dose can be expressed as

$$T_{Gi} = 3.2 \times 10^{-8}\ \rho\ K_{Gi}\ Q_i\ (1-e^{-\lambda_i t})/\lambda_i \tag{15}$$

The total population dose from all radionuclides is the sum of the population dose from each individual radionuclide.

For the inhalation exposure pathways the airborne concentration, χ, of depositable radionuclides, can be related to the ground concentration, C_A, as follows:

$$C_{Ai}(t) = \chi_i U_g (1-e^{-\lambda_i t})/\lambda_i \tag{16}$$

$$\text{or } \chi = C_{Ai}(t)\lambda_i/U_g(1-e^{-\lambda_i t}) \tag{17}$$

where

V_g = deposition velocity, m/sec

C_A = surface concentration, Ci/m^2

λ = decay constant appropriate to radionuclide, per sec

The above considerations of population doses were modified when considering population doses from food uptake. The total exposed population from fallout on food crops is determined not by the density of people in the area of the food crop but by the number of persons that can be fed by the crops grown on a unit area. Values of agricultural productivity were determined by assuming the agricultural land area east of the Mississippi could support the entire U.S. population at the per capita diet[9] given below:

Vegetation = 400 g/day
Meat = 250 g/day
Milk = 350 g/day

On this basis, the following agricultural productivity figures were computed:

Vegetation = 100 kg/day square mile
Meat = 65 kg/day square mile
Milk = 90 kg/day square mile

Thus, the implied number of people supported per unit area of agricultural land is:

Virtual people per unit area, vegetation = $\frac{100}{.4}$ = 250 people/square mile

Virtual people per unit area, meat = $\frac{65}{.25}$ = 260 people/square mile

Virtual people per unit area, milk = $\frac{90}{.35}$ = 260 people/square mile

The above number of virtual people were used in calculating dietary doses, rather than the average population density of 160 people/square mile used for other population pathway doses.

Since uranium mines and mills are primarily located in the Colorado plateau, population doses were calculated using modified assumptions. Population doses were estimated by assuming a distance of 2,000 miles to the east coast and an initial population density of 7.5 people/mile^2. The population density was increased exponentially so that it reached 160 people/mile^2 at the east coast. Food production values and the associated virtual populations were varied using the same proportionality.

Production and subsequent deposition of radon daughters was accounted for in the model.

The above considerations for population doses imply deposition of the airborne release and are not valid for airborne releases of noble gases or other isotopes dispersed in gaseous form. Population doses from noble gases, tritium, and carbon-14 were thus calculated on a different basis.

Short-lived noble gases were dispersed to the atmosphere without deposition. Radioactive decay limits the spread of these gases. The population dose, after integrating along a 1,500 mile plume path length, is given by:

$$\text{Population dose} = KQ\rho\,(1 - \exp(-\lambda r/v))/\lambda L \qquad (18)$$

where

λ = radioactive decay constant, per sec

r = path length, m

v = average wind speed, m/sec

L = height of the assumed vertical air mixing, m

Other parameters were defined earlier.

Population doses from krypton-85, tritium, and carbon-14 were calculated in two steps. The first component of the population dose occurs immediately after release and was determined using equation (18) above. For tritium and carbon-14, the dose constant, K, and the population density reflect the appropriate pathways. For krypton-85, only the submersion pathway is operative.

The second part of the population dose occurs some time after release. Population doses from krypton-85 and carbon-14 were calculated by diluting the curies released into the world's atmosphere. The volume of the world's atmosphere is 3.8×10^{24} cc.[5] The world population of 5×10^9 people was assumed to be exposed to the radionuclide concentration after 40 years of buildup. Tritium was treated similarly except that it was diluted by the earth's circulating water volume[22] of 2.7×10^{22} cc.

For krypton-85 and tritium, the only removal mechanisms operating were radiological decay. For carbon-14, environmental removal mechanisms accounted for an observed removal half-life of about 6 years.[5,23]

4.2 Hydrological Pathway

For radionuclides released to the liquid environment, it was assumed that the plant was situated at the beginning of a large watershed where the initial river flow rate was 200 cfs.

The total length of the river was assumed to be 500 miles with 7 population centers distributed along its length. People at these centers were furnished drinking water, fish, irrigated foods, and milk and meat from animals consuming irrigated feed and water.

Dilution from the point of discharge was determined by the ratio of the flow rate at the point of interest and the flow rate at the discharge. The first city was assumed to be 15 miles downstream from the plant site with a population of 4,300. The size of subsequent population centers was assumed to increase in proportion to the river flow. Water usage parameters[24] were used to describe the river area. Table IV J(A)-7 lists the assumed characteristics of the river and the associated populations.

Table IV J(A)-7

CHARACTERISTICS OF REFERENCE FACILITY WATER SOURCE

Reach (miles)	Average Distance (miles)	Dilution Factor	Transit Time (hours)	Population Drinking Water (people)	Edible Fish Catch (kgm/yr)	Shoreline Usage (hours)	Swimming and Boating Usage (hours)	Irrigated Vegetation (kgm/yr)	Irrigated Beef Cattle Feed (kgm/yr meat)	Irrigated Milk Cow Feed (kgm/yr milk)
0-30	15	1.1	11	4,300	37	2,800	860	110,000	30,000	62,000
30-60	45	1.8	33	7,000	60	7,300	2,300	180,000	47,000	98,000
60-100	80	3.5	58	18,000	160	24,000	7,400	470,000	120,000	250,000
100-200	150	9.9	110	130,000	1,100	200,000	62,000	3,300,000	860,000	1,800,000
200-300	250	26	180	330,000	2,900	530,000	160,000	8,600,000	2,200,000	4,600,000
300-400	350	50	260	630,000	5,500	1,000,000	310,000	16,000,000	4,200,000	8,700,000
400-500	450	81	330	1,100,000	9,000	1,700,000	530,000	27,000,000	6,900,000	14,000,000
Totals	-	-	-	2,100,000	19,000	3,500,000	1,100,000	56,000,000	14,000,000	30,000,000

REFERENCES

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CHAPTER IV

Section J

APPENDIX B

HEALTH RISKS FROM IRRADIATION

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HEALTH RISKS FROM IRRADIATION

1.0 RISK CHARACTER OF RADIATION PATHOLOGY

Soon after X-rays were discovered in 1895, it became apparent that excessive exposure to ionizing radiation could be injurious. However, below those dose levels that are known to produce specific acute or short term effects in all exposed individuals, the fraction of exposed individuals manifesting some form of chronic or long term injury will depend on the magnitude of the dose administered, as well as upon the dose rate. Hence, the response of an individual has a probabilistic character, and the chance that one may be affected can, therefore, be expressed in terms of a risk per unit dose of radiation exposure.

The expression of this risk must also identify the period of time over which the probability of a syndrome's occurrence is being considered, because, characteristically, there is a lapse of time between irradiation and the manifestation of symptoms. Although the delay, or latent period, may increase with lower doses and with protraction of the administered dose, no allowance is made for such variation in this discussion, and each specific cancer type has been characterized by its own latency. See Table IV J(B)-1.

The duration of risks considered in GESMO are generally those of either the full lifetime or the first 30 years post-latency, whichever occurs first. In addition, in any population with a spectrum of ages, the age at exposure must also be taken into account in the calculation of risks, since there are some recognized significant age differences in radiation sensitivity.

Table IV J(B)-1

RISK ESTIMATORS DERIVED FROM THE BEIR AND RASMUSSEN REPORTS

	Occurrence per million person-rem	Latent period (yr)	Duration of risk (yr)
Total risk of cancer mortality	135	0-15	10-30
Risk of lung cancer mortality	22.2	15	30
Risk of bone cancer mortality	6.9	10	30
Risk of thyroid cancer mortality	13.4	10	30
Risk of thyroid cancer & benign nodules	331.5	10	30
Risk of specific genetic defects	158	--	--
Risks of defects with complex etiology	100	--	--
Total genetic defect risk	258	--	--

1.1 Unobservability of Small Theoretical Risks

The syndromes or pathological effects produced by irradiation are not unique, but also occur with an appreciable probability in populations that are not exposed to man-made sources of radiation. This absence of uniqueness makes it impossible to determine if any particular manifestation in any given individual might have been caused by radiation, especially when the exposure may have occurred at low total doses and dose rates.

At doses in the range where increased incidence of syndromes can be distinguished from normal incidence by epidemiological study, there is some evidence that for the low LET* radiations the risk of cancer induction, for example, diminishes more rapidly with decreasing dose than would be linearly projected. This suggests that there may be a threshold for radiation exposure; that is, a dose below which the risk of injury is zero.

For this environmental statement, however, it is postulated that radiation produced syndrome risks are proportional to the dose received by individuals, commensurate with the risks at dose levels which have been clinically observed to produce those syndromes.

2.0 RISK QUANTIFICATION

The quantitative estimation of pathological risks employs data derived from medical, accidental, or occupational exposure of humans to various radiation sources. Studies of expected pathological risks from irradiation have been summarized by a National Academy of Sciences-National Research Council (NAS-NRC) committee[1] and by the United Nations Scientific Committee on the Effects of Atomic Radiation.[2] Both reports have arrived at comparable risk estimates of the expected mortality from radiation-induced cancer, but the NAS-NRC document, referred to as the BEIR report, presents these estimates in a form that is more readily applied to the estimation of risks in populations that include the normal age distribution. Additional data and considerations were applied to these risk estimates in the preparation of the Rasmussen Report.[3] The upper bound risk factors pertinent to a continually exposed population have been selected therefrom for cancer risk estimates in this statement.

2.1 Cancer Risk Estimators

The Rasmussen Report[3] (Appendix VI Section G1.3) presents cogent reasons for using the "absolute risk" values of the BEIR report instead of the "relative risk" estimates. The choice of a 30-year post latent period of risk or the remaining life expectancy, whichever is shorter, for chronic exposure of a population with a defined age distribution, and some modifications of the BEIR report risk coefficients due to recent considerations, are explained in the Rasmussen Report, Appendix VI, Sections 9.3.2.2 and 9.3.2.3. Tables IV-9-4 and IV-9-8 ibid., provide the basis for the

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*LET: Linear Energy Transfer

cancer risk estimators in Table IV J(B)-1 of this appendix. The latent period and duration of risk are also given in the table. The zero latency and 10-year risk duration values refer only to leukemias that might occur in children who were exposed before birth.

Though these risk estimates are the upper bound estimates given in the Rasmussen Report,[3] higher estimates can be developed by use of the "relative risk" model along with the assumption that risk prevails for the duration of life. This would produce risk values up to sevenfold greater than those used in GESMO. That would place an upper limit to the range of our uncertainty. The lower limit would range from zero risk to values near those in Table IV J(B)-1, depending upon total dose, dose rate and radiation energy or LET.

A mortality rate of 10% for thyroid cancer is used in Table IV J(B)-1. Clinical data indicates that ^{131}I irradiation of the thyroid from deposition within the organ is less effective by more than an order of magnitude in producing either benign or malignant nodules than irradiation of the organ by X-rays from an external source. In GESMO, the risk per rem contributed by radioiodine in the thyroid is taken to be 10% of the risks indicated in Table IV J(B)-1.

2.2 Genetic Risk Estimators

The BEIR report is also employed to derive the genetic risks that might be attributed to irradiation of the gonads. Genetic risks are entirely translated from experience with external irradiation of laboratory animals. However, the available limited human experience with external irradiation is consistent with these observations.[1]

The genetic risks include the full spectrum of genetic defects seen in the United States and other Western nations. Their effects upon the carrier may range from a lethal action at or near birth to minor metabolic consequences that may be nearly undetectable. The genetic spectrum ranges from dominant single gene mutants, whose effects may be categorically recognized, to subtle genetic contributions to disease conditions that are predominantly of environmental or nongenetic origin. As a consequence, it is not appropriate to compare or equate estimates of genetic risk directly with the cancer risks. The latter are health consequences where case incidence and case mortality are substantially one-to-one. (This is the situation for the lung and bone cancer risks set forth here, though it is not the situation for thyroid and other cancers that are known to have a low risk of mortality.)

The disparity between genetic and cancer risk is further clarified by noting that the genetic estimate incorporates two distinctly different types of genetic defects. The first relates to categorical or specific genetic conditions usually attributed to single genes, while the second type of genetic disability concerns the diseases of complex etiology, such as congenital anomalies and constitutional or metabolic diseases that have an ill-defined genetic component.

The BEIR report indicates an uncertainty of risk estimation over a tenfold range for both types due to uncertainty in the value of the mutation rate doubling-dose. The second type of defect has an additional tenfold uncertainty attributed to the lack of precise knowledge of the magnitude of the genetic component of the defects with complex etiology.

In GESMO the geometric mean is used to simplify comparisons in tables, but the reader should bear in mind that a range of risk uncertainty extends a factor of 3.16 above and below this for specific genetic defects and a factor of 10 for defects with complex etiology. These genetic risks are expressed in Table IV J(B)-1 as probabilities of genetic defect per million person-rem of exposure to the parent population per generation.

3.0 CONSERVATISM OF RISK ESTIMATORS

These estimates of cancer and genetic risk, as previously noted, are all basically derived from risk values developed in the BEIR report. That report assumed the dose-response relationship was linear and passed through the origin. In other words, no threshold or indifference dose was assumed. Although it was recognized that the minimum risk could well be zero, arguments were also presented to support the nonthreshold contention. The linear non threshold concept was considered to be in the public interest for risk assessment. Unfortunately, no evaluation has yet been made of the potential economic cost that might be due to the linear assumption, as compared to other assumptions.

The risks for cancer mortality were generally extrapolated from human experience at relatively high dose rates and total doses. The risks are considered to be maximum estimates for the low LET radiations, such as conventional X or gamma radiation, as these radiations have generally shown a twofold to tenfold reduced effectiveness at low dose rates (0.1 rad per hour or less). On the other hand, 5 MeV alpha particles, due to their high ionization density, are not expected to demonstrate the same protraction effect. Therefore, there may be less conservatism built into the risk estimates following exposures to the transuranic elements.

The genetic risks are derived from mutation rate studies carried out on mice that had been exposed to low LET radiation delivered at low dose rates. The experimental data show a linear dose response curve, through the origin, so that no intrinsic factor of conservatism is incorporated in the genetic risks as noted for the cancer risk estimates following a low LET exposure. A quality factor (Q) of 10 is used to adjust rad doses upward to allow for the higher mutagenic potential of high LET radiation. A recent ICRP report[4] notes that mutation rate studies with fast neutrons, a high LET radiation, when compared to gamma radiation, have generally shown an RBE of about 20 at low dose rates and about 5 at high dose rates. As the base line genetic risks used in this report are developed from low dose rate data, the Q of 10 might be considered low by a factor of two, but no official position has been taken on this issue by the International Commission on Radiological Protection.

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CHAPTER IV

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APPENDIX C

PLUTONIUM IN THE ENVIRONS AND IN FAUNA

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PLUTONIUM IN THE ENVIRONS AND IN FAUNA

1.0 FALLOUT PLUTONIUM

Weapons testing has released an estimated 320 kCi of ^{239}Pu (about 5 tons).[1] This is highly dispersed so that relatively low concentrations are measurable in environmental media, including just perceptible amounts of ^{239}Pu in man. Compared to the activity concentrations and doses from fallout fission products, ^{239}Pu does not contribute greatly to the total dose to man.[2] On the basis of a projected mixed oxide fuel cycle release of 24 mCi of alpha activity via transuranium nuclides with half lives exceeding 1 year per GWe-yr, the 320 kCi of fallout ^{239}Pu is equivalent to 1.3×10^7 GWe-yr of power generation. This is over 3,220 times the energy expected to be generated by LWR's between 1975 and 2000.

Plutonium fallout measurements provide some direct evidence on which to judge the recent behavior of ^{239}Pu in the environment. Estimates of environmental transfers in the short term are, therefore, more soundly based than the health consequences estimates. However, more measurements of environmental plutonium are required to reduce the large number of assumptions and extrapolations involved in estimating the consequences of the long persistence of ^{239}Pu in the environment.

Currently available data from observations of the behavior of fallout plutonium in the environment support the following conclusions. Plutonium is primarily an inhalation hazard, and it appears that the greatest fraction of the inhalation intake occurs during the original deposition. For periods of 10 to 20 years after deposition, inhalation prior to deposition is clearly the predominant single route of entry into man. Inhalation of radioactivity resuspended by winds from the soil surface contributes to the inhalation intake, but the additional amount, though somewhat uncertain, is apparently lower, partly because deposited activity soon begins distributing itself to greater depths in soil, thus reducing the amount of surface radioactivity which is available to be resuspended.

In comparison with the inhalation pathway, uptake from soil by plants is quite low, and subsequent absorption of plutonium by the gastrointestinal tract in animals and man is many orders of magnitude lower than that for most other elements. The generally low solubility of plutonium compounds may be considered the prime factor limiting its availability. As with other radionuclides, plutonium may have reduced availability with time after deposition due to downward movement in soil and to a greater or lesser extent as a consequence of fixation by soil minerals or other physical and chemical changes that can occur.

1.1 Plutonium-239 in Air and Deposition Amounts

The concentrations of fallout ^{239}Pu in surface air in New York have been measured by the Health and Safety Laboratory since 1965.[3] By comparing with the ^{90}Sr measurements, it is possible to reconstruct the ^{239}Pu concentrations in air during the earlier fallout years.[4] Estimates of ^{239}Pu deposition can be obtained from the air concentrations by assuming that the same relationship holds between air concentrations and deposition as for ^{90}Sr. The average annual deposition-air ratio is 5.1×10^5 m/yr (1.6 cm/sec). The cumulative deposit of ^{239}Pu on soil can be further obtained by adding the yearly depositions, radioactive decay being insignificant.

The ^{239}Pu fallout estimates for New York are listed in Table IV J(C)-1. A check on the related values of air concentration, deposition and cumulative deposit is available from a New York area soil, sampled in 1970.[5] The ^{239}Pu content was found to be 2.6 mCi/km^2, in agreement with the estimated cumulative deposit listed in Table IV J(C)-1.

1.2 Depth Distribution in Soil

Radioactivity deposited on the soil surface eventually becomes distributed with depth. The downward movement is a result of many processes, both physical and biological. Traditionally, vertical movement of elements in soil profiles has largely been associated with their solubility or leaching rates. However, in the case of plutonium compounds (predominantly oxides) which are very insoluble in water, physical processes such as movement of submicron plutonium particles by mass transport of percolating rainwater through soil profiles may play a more important role than leaching.

The mechanisms of transport have not been thoroughly evaluated but most likely will vary according to climate and soil characteristics. The greatest difficulty in interpreting data on the movement of plutonium in soils is that little or no information is presented other than the concentration at various soil depths. For example, pH, texture, slope or the concentrations of plutonium in specific horizons that are characterized by the soil forming properties are seldom stated. Information of this nature can help in identifying processes responsible for the vertical distribution in soil profiles.

From the present information on the distribution of plutonium in soil profiles, which is very limited, the vertical movement of plutonium appears to be intermediate between that of radio strontium and cesium, Table IV J(C)-2 and Figure IV J(C)-1.

Table IV J(C)-1

FALLOUT ESTIMATES FOR ^{239}Pu - NEW YORK

Year	Surface Air (fCi/m^3)*,**	Estimated Deposition (mCi/km^2)***	Estimated Cum. Deposit (mCi/km^2)***
1954	.14	.072	.072
55	.18	.093	.17
56	.23	.12	.29
57	.23	.12	.40
58	.32	.16	.56
59	.45	.23	.79
60	.081	.041	.82
61	.13	.063	.89
62	.63	.32	1.21
63	1.68	.62	1.83
64	.91	.41	2.24
65	.33	.14	2.38
66	.12	.054	2.43
67	.051	.038	2.47
68	.080	.040	2.51
69	.063	.058	2.57
70	.065	.033	2.60
71	.060	.026	2.63
72	.031	.021	2.65

*fCi = 10^{-15} Curies.

**Data before 1965 were not measured, but were reconstructed by comparison with ^{90}Sr measurements.

***These estimates are based on the ^{90}Sr average annual deposition air ratio 5.1×10^5 m/yr.

Table IV J(C)-2

FALLOUT ^{239}PU AND ^{90}SR DEPTH DISTRIBUTION IN SOIL IN 1970

Brookhaven, N.Y.[5]

	% Pu	% SR
0- 7 cm	57	42
7-11	27	30
11-15	11	15
15-21	3	8
21-25	1	2
25-30	1	3

Frilford, Berks, U.K.[5]

	% Pu	% Sr
0- 5 cm	55	43
5-30	45	57

Denver, Colo.[6]

	% Pu
0- 5 cm	81
5-20	19

Waynesville, Ohio[6]

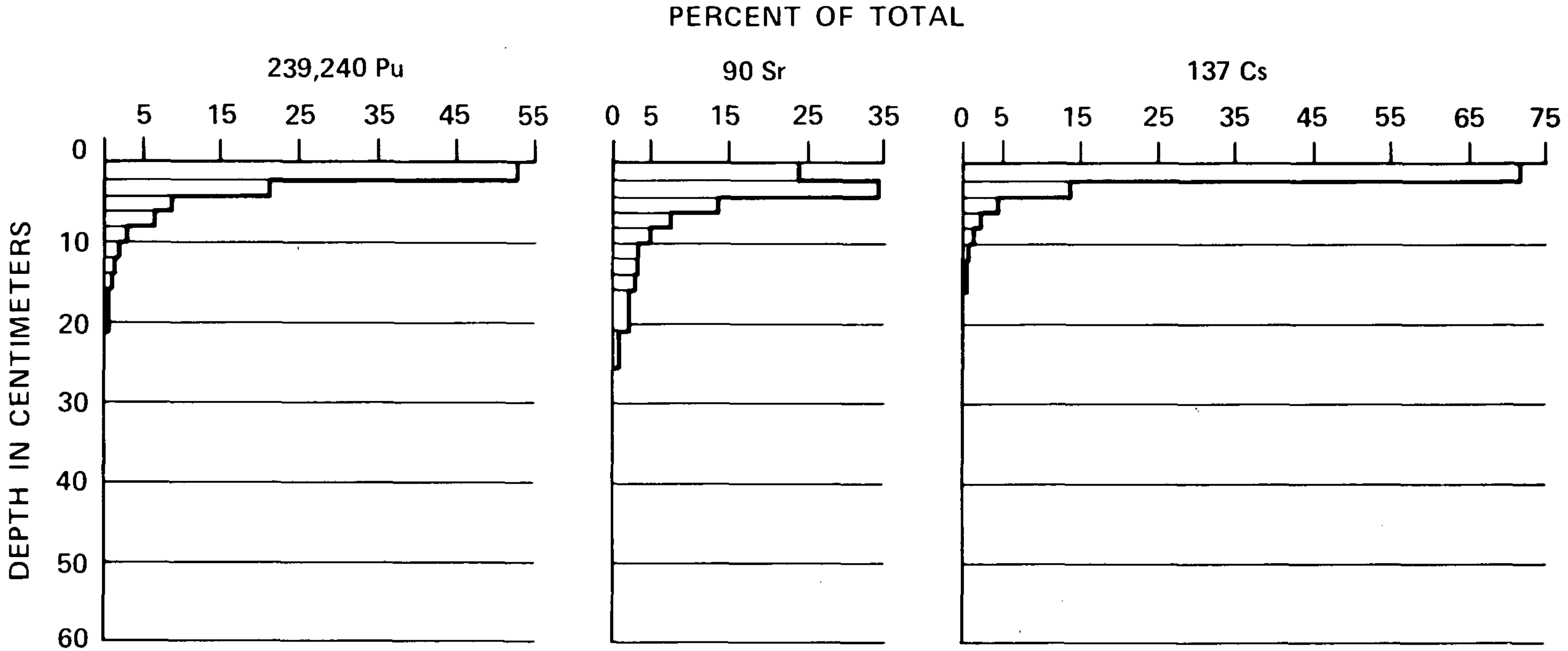
	% Pu
0- 6 cm	47
6-20	53

New York City[6]

	% Pu
0- 5 cm	81
5-20	19

Bennett, Colo.[6]

	% Pu
0-10 cm	92
10-20	8



(This figure is taken from testimony by McDonald E. Wrenn, EPA Plutonium Standards Hearing, Washington, D.C., December 10 - 11, 1974)

Figure IV J(C)-1 Depth Distribution of ^{90}Sr, ^{137}Cs, $^{239,240}Pu$ in Sandy Loam Soil, North Eastham, Mass., October 1972

1.3 Resuspension

Resuspension does not appear to have been a phenomenon of major consequence as far as fallout plutonium is concerned. The cumulative deposit of ^{239}Pu in soil obtained from air concentration measurements has not been overestimated even though resuspended material would have been included in the measured air concentrations. The air deposition ratios have shown no systematic changes, as would be expected if resuspension from the accumulating deposit in soil were playing a role. Most of the air activity is known (from inventories) to be coming from the stratosphere instead of from the soil surface.

Presently there is no completely general model which may be used to predict the concentrations of resuspended air activity with due regard to the geometrical configuration of the land surface, the particle characteristics of the deposited radioactivity and the physical and chemical parameters of host soil, vegetation cover, and meteorological conditions. These highly variable factors and others related to land use, such as the disturbance of soil surfaces by human activity, have to be taken into account in applying the following to the geographic region around any specific fuel cycle facility.

However, there have been many measurements of resuspended radionuclides in the vicinity of ground-deposited particulate debris.[7,14] Many of these studies have specifically examined the resuspension of ^{239}Pu in natural environments. Other radionuclides and nonradioactive materials have also been studied. The results of these experiments can be expressed in terms of a resuspension factor, K, defined as:[10]

$$K = \frac{\text{Resuspended Air Activity (per unit volume)}}{\text{Deposited Ground Activity (per unit area)}}$$

Values for K from a number of experiments have been tabulated.[7,8] Since K is a crude index that neglects several variables influencing the resuspension process, the total range of measured values is large; values range from 10^{-2} to 10^{-13}/m. However, the very large values are associated with resuspension from laboratory floors and the very low values are from aged Pu deposits discussed below. Most of the values of K, as inferred from measurements in material environments soon after the contaminating event, are in the range 10^{-4} to 10^{-7}/m. Stewart[8] suggested that a value of 10^{-6}/m be used under quiescent conditions and a value of 10^{-5}/m be used under conditions of moderate activity. Kathren[15] recommended the use of 10^{-4}/m as a conservative value for setting standards for PuO_2 surface contamination. Langham[10] suggested a value of 10^{-6}/m as a reasonable average value to use in considering the potential hazard of contaminated areas. Thus it appears that 10^{-5}/m is a reasonable, still conservative, value for estimating the initial resuspension effect.

There is evidence from several experiments that the value of K decreases with time after deposition.[10,11,13] This is generally interpreted as an indication that the deposited activity is weathering into the soil surface or otherwise becoming less available for resuspension. A useful approach to evaluating the additional exposure that the resuspension process contributes to the inhalation pathway is to calculate the ratio of integrated exposure via resuspension to the integrated exposure from the initial air activity.

The first item in calculating this ratio is the integrated air exposure, Ψ (Ci-sec/m^3) due to cloud passage. The fraction of this material that will be deposited upon the ground is $V_g\Psi$, where V_g is the deposition velocity (m/sec). The magnitude of $V_g\Psi$ describes the areal source term (Ci/m^2) which may be used to predict the level of resuspended air activity.

The real situation is complicated by the decrease of the resuspended air concentration with time. This can be accounted for by letting the factor K be a function of time, denoted by K(t). (An alternate approach is to define a portion of the deposited material as being available for resuspension and letting that amount decrease with time while K is a constant. Mathematically, the results are the same.)

The resuspended air activity as a function of time is therefore $K(t)V_g\Psi$, and the ratio, R, of the integrated resuspended air activity to the integrated air activity of the original cloud is equal to

$$R = V_g \int K(t)\, dt$$

It has frequently been assumed that

$$K(t) = K_o e^{-0.693t/T_{1/2}}$$

This expression fits the available data reasonably well at early times with times ($T_{1/2}$) of 35 to 70 days.[10,11,13] However, these results are for relatively short term measurements. The longest study of the decrease of K with time extended to only eleven months following the initial deposition.[11]

The value which the resuspension factor approaches at long times can be estimated from measurements made above aged contaminated soil and from consideration of other tracers, such as ^{238}U. From recent ^{239}Pu measurements made at the Nevada Test Site in an area contaminated 17 years previously, resuspension factors of 10^{-9} and 10^{-10}/m are obtained.[16,17] Measurements of ^{239}Pu in the vicinity of the Rocky Flats plant[17] several years after deposition indicate a resuspension factor of 10^{-9}/m.

The use of ^{238}U as a more general tracer to indicate the resuspension effect avoids the complications of interpreting large concentration gradients, and also the uncertain relationships of terrain characteristics in the measured areas (mostly arid) and in more densely inhabited (mostly humid) regions. The natural availability of ^{238}U far exceeds any additional weapons debris that might have been introduced. Hamilton[18] presented representative air and soil concentration measurements from samples obtained at Sutton in the United Kingdom.

From the soil profile, the ^{238}U concentration to 2 feet in depth was about 1 ppm, except in the leached surface soil zone, where the concentration was 0.8 ppm. Assuming a soil density of 1.5 g/cm^3, this corresponds to 0.012 gram of uranium/m^2 to a depth of 1 cm. The mean ^{238}U concentration in air measured during a 20-month period was 0.6 x 10^{-10} g/m^3. The effective resuspension factor is thus:

$$0.6 \times 10^{-10} / 0.012 = 5 \times 10^{-9}/m.$$

Since the airborne particulate matter contained considerable material of industrial origin, such as smoke and dusts from building materials, even this estimate of the resuspension factor appears to be an overestimate.

Sedlet[19] reports the following ^{238}U data for Argonne, Illinois: "120 x 10^{-6} pCi/m^3 in air and 1.5 pCi/g in surface soil." From these measurements a resuspension factor of 5 x 10^{-9}/m is obtained.

Discounting material of industrial origin in air, it appears from the ^{238}U data that a realistic estimate of the resuspension factor of aged radioactive material in surface soil lies between 10^{-8} and 10^{-10}/m. This is in agreement with the ^{239}Pu field measurements. An intermediate value of 1 x 10^{-9} is therefore used in estimating long term inhalation potential for the relatively large, well-vegetated regions of the United States.

The time dependent function of early resuspension indicates that the aged situation is achieved within 2 years after initial deposition. An estimate of the ratio of inhalation intake of resuspended material to direct inhalation intake of initially depositing airborne material can be obtained by assuming an initial resuspension factor of 10^{-5}/m, reduced exponentially with an intermediate half-time of 50 days until the aged value of 1 x 10^{-9}/m is achieved (approximately two years). It is then assumed that this value remains constant even though the deposited transuranium elements may not remain at or near the surface, and thus continue to reduce the availability for resuspension.

1.4 Food Chain Contamination

Transuranium elements dispersed to the environment may be ingested by man as a consequence of mobilization along any of a large number of potential food chain routes. There is little direct experimental data on transuranium element behavior via many of these routes. This appendix considers the data on food chain mobilization of transuranium elements for predicting ingestion by man.

1.4.1 Routes Via Plants to Man

The route via plants appears to be the most direct and therefore the most probable route by which transuranium elements in the soil might reach man. Experimental data on plant uptake of these elements has been recently reviewed.[20,21,22] In laboratory studies employing a variety of soil types, chemical treatments and compounds of plutonium, americium, and curium, the concentration factors (dry plant/dry soil) have ranged from approximately 2 to 10^{-5} or less. The actinides, americium and curium, generally show the higher values, and are of particular concern because of the assumed greater absorption of these elements (compared to plutonium) from the gastrointestinal tract. Concentration factors of 2×10^{-3} to 3×10^{-3} were reported for ^{241}Am in beans and barley grown on several soils and in nutrient solution;[23] values of 1.4×10^{-3} for ^{241}Am, and 2.2×10^{-3} for ^{244}Cm were reported for tumbleweed, and lower values for cheatgrass.[24] However, these latter values were based on a minimum estimate of contaminated soil and may be up to a factor of 25 higher. Concentration factors up to 1.8 were obtained with DTPA chelates of americium.[25,26]

Two studies have shown increasing uptake of Pu with time following soil contamination.[27,28] In one of these there was a sevenfold increase in the plutonium concentration in clover during a 5-year cropping sequence.[28] Some of this increase may be attributed to continuing development of the root system as the crop became established; some might also be due to solubilization of the plutonium by organic complexing, perhaps associated with microbial decomposition products of root material. This latter effect may be accentuated by the presence of higher concentrations of Pu in the roots than in the shoots of certain plants. Whether a long term continuing increase in transuranium element availability might result from reactions with soil organic matter or from microbial transformations is not known.

A factor operating to decrease transuranic availability to certain plants with time is the downward movement in the soil. The model used assumes that this process will not extend beyond a 20 cm depth, but data on fallout plutonium in soil indicates movement below this depth with activity already present below 20 cm at some locations.[28,29] This would tend to decrease availability to shallow-rooted plants, but not to deeper-rooted plants such as alfalfa. Furthermore, increased mobility in soil may be a reflection of increased plutonium solubility or plant translocatability, both of which may, in turn, reflect increased potential for plant uptake.

From the fallout plutonium data one can estimate an overall concentration factor between plant derived food and soil for plutonium. Table IV J(C)-3 summarizes data from large samples of foods obtained in New York City in 1972.[30] The plutonium concentration in the plant derived portion of the diet averaged 3.7×10^{-15} Ci/kg. For the same year, the cumulative plutonium deposit on New York soil was estimated as 2.65 mCi/km^2, or 2.65×10^{-13} Ci/cm^2.[31] If plutonium is distributed uniformly through a 20 cm depth of density 2.0 (this assumption is conservative since most measured values approximate 1.5 or less) a plutonium concentration of 6.6×10^{-12} Ci/kg soil can be calculated. Comparing the average concentration in food with the concentration in soil gives a concentration factor of 6×10^{-4}. Note that this concentration factor is based on "market weight" of food and would be larger by perhaps fivefold if converted to a dry weight basis. Furthermore, this value has an upper limit of 1.3×10^{-3} if the highest plutonium concentration in plant derived foods (bakery products) is considered.

Table IV J(C)-3

PLUTONIUM IN NEW YORK FOODS--1972[30]

Food Item*	Plutonium Concentration (10^{-12} Ci/kg)	Per Capita Annual Intake Food (kg/year)	 Pu (10^{-12} Ci/yr)
Plant-Derived			
Bakery Products	0.0085	44	0.37
Whole Grain Products	0.0060	11	0.066
Fresh Fruit	0.0051	59	0.30
Dry Beans	0.0049	3	0.015
Fresh Vegetables	0.0043	48	0.21
Root Vegetables	0.0035	10	0.035
Flour	0.0028	34	0.095
Rice	0.0015	3	0.004
Potatoes	0.0013	38	0.048
Macaroni	0.0012	3	0.004
Canned Vegetables	0.0009	22	0.019
Fruit Juice	<0.0003	28	<0.007
Canned Fruit	<0.0002	11	<0.002
Total Plant Derived Diet	0.0037**	314	1.17 - 1.18
Animal-Derived			
Shellfish	0.011	1	0.011
Poultry	0.0033	20	0.066
Meat	0.0026	79	0.20
Fresh Fish	0.0016	8	0.013
Eggs	0.0012	15	0.019
Milk	<0.0003	200	<0.064
Total Animal Derived Diet	0.0010**	323	0.31 - 0.37
Total Diet			1.5

*Food categories and quantities are based on Department of Agriculture food purchase surveys.
**Values are calculated from total per capita annual intake of food and of plutonium.

On the other hand, an uncertain fraction of the fallout plutonium in some foods may be due to external contamination rather than to root uptake.[30]

1.4.2 Routes Via Animals to Man

The absorption of transuranium elements from the gastrointestinal tract of animals is very limited. Fractional absorption of 2×10^{-5} or less is reported for plutonium nitrate or oxide in the pig,[32] and values of 3×10^{-4} for the citrate and 3×10^{-5} for the oxide can be estimated from recent data on the cow.[33] Considering this very low absorption it would seem that a food chain which included an animal between the plant and man would be an insignificant contributor to plutonium intake by man. It is therefore surprising that the animal derived diet estimated in Table IV J(C)-3 contains about one-fifth of the plutonium in the total diet. This is perhaps explained by direct inhalation or ingestion of plutonium by the meat-animal during grazing. Grazing would not explain the high poultry value, since these chickens, for the most part, never set foot on the ground. Recent data suggest that the chicken may absorb a higher fraction of ingested plutonium than most other animals.[34] Data for gastrointestinal absorption of americium and curium are available only for the rat, with maximum reported fractional absorption values of about 10^{-3}.[35]

1.4.3 Aquatic Food Chains

Although liquid effluents constitute a minor fraction of estimated releases of transuranium elements, some of the material released to the atmosphere will fall on rivers, lakes and oceans, and over long periods of time some leaching of these elements from the soil and transport to rivers and oceans will undoubtedly occur. Of some concern in this regard is the remarkable affinity for transuranium elements of certain algae, seaweed and invertebrates.[36] Thus, concentrations of plutonium in certain shellfish may be a few hundred times higher than the water in which they live; even higher factors have been observed for algae and seaweed. Shellfish for example, showed the highest plutonium concentration of any food sampled in the New York City study.[30] While not a significant factor in the total context, concentrations of transuranium elements along food chains in aquatic environments are considered in specific siting reviews.

2.0 HUMAN DATA

Over the years a number of workers in the nuclear industry have received exposures that have resulted in detectable plutonium deposition. Some exposures have led to deposition at or above the maximum body, or lung burden established by the ICRP.[37] However, no known serious health consequences have been caused in man by exposure to plutonium. The scanty evidence of effects at the cellular level such as the histologically evidenced effects surrounding a plutonium wound[38,39] and possibly increased chromosomal aberrations[40] following accidental exposure appear to be special cases. Consideration must, therefore, be limited to the kind and magnitude of exposures that have occurred without evidence of effect. Such exposures arise from two principal sources; the accidental exposure of plutonium workers, and the worldwide exposures to plutonium fallout from atmospheric testing of nuclear weapons and other devices. In addition, a small number of humans have been injected with plutonium with no apparent adverse effects.[46]

2.1 Follow-up Studies of Plutonium Deposition Cases from Occupational Exposure

Personnel exposures date to the period shortly after the discovery of plutonium three decades ago. One group of persons who experienced such exposure at Los Alamos Scientific Laboratory (LASL) has been studied at intervals since 1945 and is of sufficient interest to be described in some detail.

The group consists of 25 male subjects who worked with plutonium during World War II under very crude working conditions judged by today's standards. Twenty-one of these men had their latest complete physical examination at LASL in 1973.[41] In addition to physical examinations and laboratory studies (complete blood count, blood chemistry profile, and urinalysis), roentgenograms were taken of the chest, pelvis, knees, and teeth. Chromosomes of lymphocytes cultured from peripheral blood and cells shed from the pulmonary tract were also studied. Urine specimens assayed for plutonium yielded calculated body burdens which ranged from 0.005 to 0.42 μCi. These estimates of body burden were generally higher than earlier estimates based on radioassay of urine samples performed many years ago, perhaps reflecting uncertainties in the model used to estimate body burden from excretion data.

To date none of the medical findings in the group can be attributed to internally deposited plutonium. Except for the ailments that one would expect in a group of men, who are mostly in their early fifties, all subjects examined were in remarkably good health and all are actively working.

In view of the relatively small number of persons with sizeable depositions, it seems unlikely that precise statistical estimates of the toxic effects will ever be made.

An estimated 16,000 persons have been employed in operations that might lead to significant plutonium exposure. About 3,000 of these have shown some measurable evidence of internal plutonium deposition. Efforts are being made to obtain the maximum amount of information from these exposed persons. The U.S. Transuranium Registry[42,43] was established for this purpose in 1968 and is operated by the Hanford Environmental Health Foundation. The Registry seeks to identify potentially exposed workers, obtain their health physics and medical records, and their permission for post-mortem sampling. The accumulating autopsy data is in general agreement with data on the distribution of plutonium in experimental animals, and thus supports the extrapolation of animal toxicity data to man.

The collection and analysis of tissues from the general nonregistry worker population is conducted by the Pacific Northwest[44] and Los Alamos Scientific Laboratories.[50] Tissue analysis has been completed on about 376 autopsies at PNL, all on individuals who died in the Richland area. LASL has obtained tissues from 512 autopsies in several geographic areas. In addition to members of the general population, these autopsies include workers, 160 at Hanford and 75 at Los Alamos, some of whom were exposed to plutonium in their work. Since these employees were not enrolled in the Registry prior to death, they are not included in the Registry statistics.

A more direct human experiment was conducted at the close of the Second World War when increasing plutonium production led to a critical need for human plutonium excretion data. Excretion rates differed in dogs and rats and thus were of no help in estimating excretion rates in humans. As a result, 18 hospital patients with supposedly terminal illnesses were injected with tracer doses of plutonium. Sixteen patients received a single intravenous injection of plutonium either as ^{239}Pu (IV or VI) citrate or nitrate; one patient received a single injection of ^{238}Pu and ^{239}Pu as $PuO_2(NO_3)_2$; and one patient received a single intramuscular injection of ^{238}Pu(VI) nitrate. Doses ranged from 0.004 to 0.169 μCi/kg. The original data have been reviewed and reanalyzed by Durbin in Stover and Jee, 1972.[45]

Despite experimental deficiencies these data have been used to help set standards for plutonium workers and are among the few human data available. The patients had a variety of pathological conditions including cardiovascular disease, hypertension, hemophilia, Cushing's syndrome, osteoporosis, adenoma, Addison's disease, cirrhosis of the liver, and others, some of which could have affected excretion and distribution of plutonium within the body. No long term followup was planned; however, eight patients lived at least 8 years and three are still alive (Science, 191: p. 1030, 1976).[46] There is no evidence that the plutonium contributed to the course of diseases already present or caused the appearance of new diseases in any patient. No conclusions concerning induction of liver or bone cancer can be drawn from this study because of the relatively low doses used, the long latency period for cancer at low doses compared to survival times of patients in this study, and the small number of persons involved.

No effects that are attributed to Pu have been observed in any of the human population studies. However, the studies are obviously incomplete in view of the brief period of follow-up (30 years maximum) and the small number of humans definitively studied to date.

2.2 Plutonium Levels in the General Population Resulting from Plutonium Fallout

As a result of fallout from nuclear weapons testing plutonium is present in extremely small quantities in various organs of the human body in the general population. An estimated 320 kCi (about 5 tons) of plutonium has been deposited on the earth's surface from weapons testing of which 250 kCi has been deposited on the northern hemisphere and 16 kCi on the United States.[47,48] Estimates of human organ burdens and doses have been made by Bennett,[49] employing the ICRP lung model and the data from New York City plutonium air concentrations.

Direct measurements of organ burdens from fallout plutonium have been made in autopsy samples, although the levels are so low as to tax analytical capabilities. Many of the earlier values reported were highly uncertain. The best available data are those from the Los Alamos Scientific Laboratory on samples from Colorado and New Mexico.[50] These determinations employ larger tissue samples than have usually been available and utilize improved analytical procedures.

The measured organ burdens are compared in Table IV J(C)-4 with calculated organ burdens based on the New York plutonium air concentration and the ICRP lung models. The agreement is quite good, except for the lymph nodes. The low measured value in lymph nodes may reflect a greater solubility of fallout plutonium than assumed in the model. Proportionately larger amounts of plutonium are observed in the lymph nodes of experimental animals and plutonium workers exposed to insoluble plutonium oxide.

No measured effect on man has been linked to the data for fallout plutonium.

Table IV J(C)-4

FALLOUT PLUTONIUM IN MAN

(COLORADO AND NEW MEXICO, 1970-71)[49,50]

Organ	No. of Samples	Plutonium in Organ (pCi) Measured	Plutonium in Organ (pCi) Calculated from New York Air Concentrations*
Lung	96	0.3	0.3
Bone	96	1.4	0.9
Liver	88	1.4	0.8
Lymph Nodes	73	0.03	0.6
Kidney	73	0.1	
Total		3.2	2.6

*Calculated by Bennett using lung model.[49]

3.0 ANIMAL DATA

As deduced from studies with experimental animals, the organs in which plutonium is retained most tenaciously, and in highest concentration, are bone, liver, lung, and lymph nodes. In all of these organs tumor formation has been observed as a result of plutonium deposition--most significantly in bone and lung, only rarely in lymphatic tissue.

The findings of studies on the toxicity of plutonium in experimental animals are briefly reviewed herein. There is no firsthand knowledge of plutonium effects in man, and it is necessary to infer what effects might occur from the accumulated information on effects in animals. In this document, there is no attempt to extrapolate animal toxicity data directly to man; instead, the cancer risk estimates as derived for radiation, generally, in the BEIR report[51] were employed. However, the animal data are used to indicate which tumor types are of concern, i.e., which are the effects that are most likely to occur.

Acute toxicity is conventionally expressed in terms of an LD_{50} dose; i.e., the dose required to kill 50% of the animals within some specified period of time, usually 30 days. For intravenously injected tetravalent plutonium-239 citrate, in rats, the $LD_{50/30}$ is about 70 μCi/kg. A similar value was observed for mice. A somewhat lower value, 20 μCi/kg, was observed in dogs, but this was with hexavalent plutonium. The $LD_{50/30}$ for inhaled plutonium in rats and dogs was not very different from these values for injected plutonium. These values will vary somewhat depending on the compound administered, or the valence state. Male rats show a somewhat higher sensitivity than females. Plutonium-239 is somewhat more toxic, per μCi, than plutonium-238, due to a different distribution in the animal, occasioned by the much greater mass of plutonium-239 required to deliver the same μCi dose. But these are minor variations--the significant point is that acute death requires an internally deposited dose of probably 10 μCi/kg as a minimum estimate, which translates to 700 μCi for a 70 kg man. This amounts to about 10 mg of plutonium-239, or about 40 μg of plutonium-238.[52]

Long term effects will occur at very much lower exposure levels than those required to produce acute death, and it is these long term effects that are the only concern at the very low exposure levels that might result from use of mixed oxide fuels in LWR's. Studies of acute toxicity have been useful, however, to magnify effects which could not easily be studied at lower exposure levels; they give leads as to what effects should be looked for in long term studies at lower levels; and they may help researchers understand the mechanisms by which plutonium exerts its effects.

The study of long delayed effects of a toxic substance in experimental animals is a more complex problem than the study of acute effects, not only because of the longer duration and more extensive controls required in a lifetime study, but because of the variety of endpoints that may need to be evaluated to quantify effects. For acute toxicity, the conveniently quantifiable endpoint is death. The corresponding parameter for the study of long term effects is life shortening, but life shortening turns out, in many experiments, to be a less sensitive measure of effect than other measurable parameters. For instance, the incidence of neoplasia has been shown to be significantly increased in experimental groups that show no significant decrease in life span. The incidence of neoplasia does, in fact, appear to be the most sensitive measure of long delayed effect of internally deposited plutonium.

3.1 Disposition of Transuranium Elements in the Body

Transuranium elements may enter individuals from the reservoirs in the environment via three pathways.[53] Transuranium elements incorporated in food may be ingested and absorbed from the gastrointestinal tract, while those dispersed in air may be either deposited on the skin and absorbed, or inhaled and deposited in the respiratory tract. Significant entry through a wound is possible only where heavy contamination prevails.

3.1.1 Gastrointestinal Tract Absorption

Plutonium is not readily absorbed from the gastrointestinal tract. The gastrointestinal tract absorption of plutonium as percent of administered dose is given in Table IV J(C)-5 for newborn rats and adult rats.[53] The most readily absorbed compound was plutonium nitrate; least readily absorbed was plutonium oxide.

3.1.2 Absorption Through Intact Skin

Data from animal experiments indicate that intact skin is an effective barrier to the entry of plutonium.[53] The data in Table IV J(C)-6 reveal that the greatest absorption occurred for rat skin exposed to $Pu(NO_3)_4$ in 10 N HNO_3 for 5 days. All other compounds were absorbed to a significantly lesser extent.

Table IV J(C)-5

GASTROINTESTINAL TRACT ABSORPTION OF PLUTONIUM IN RATS*

		Percent Administered Dose	
Radioisotope	Compound	Newborn	Adult
^{238}Pu	Nitrate	2	0.03
^{239}Pu	Nitrate	0.3	0.003
^{239}Pu	Chloride	-	0.007
^{239}Pu	Oxide	-	0.0001

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*Taken from Bair Report[53]

Table IV J(C)-6

ABSORPTION OF PU THROUGH INTACT SKIN*

^{239}Pu Compound	Animal Species	Duration of Exposure	Percent Absorbed
$Pu(NO_3)_4$ in 10 N HNO_3	rat	1 hour	0.05
Pu-tributyl phosphate in CCl_4	rat	15 min	0.04
$Pu(NO_3)_4$ in 0.1 N HNO_3	rat	5 days	0.1-0.3
$Pu(NO_3)_4$ in 10 N HNO_3	rat	5 days	1-2
$Pu(NO_3)_4$	rabbit	14 days	0.15
Pu citrate	swine	10 days	0.25

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*Taken from Bair Report[53]

3.1.3 Lung Retention

Data from animal studies yield values on retention half-times for plutonium compounds in the pulmonary region of the lung. The pulmonary retention half-times for organic complexes of plutonium, plutonium nitrate, and fluoride range from less than 100 days to approximately 300 days in rats and dogs.[53] The plutonium dioxide retention half-times range from 200 to 500 days in rats and from 300 to 1,000 days in dogs.[53] The range of half-time retention values for dogs is largely due to extensive experimentation with a variety of plutonium oxides with different particle size characteristics. The studies with $^{238}PuO_2$ in dogs indicate a much shorter lung retention half-time than was observed for studies with $^{239}PuO_2$. This difference appears to be due to instability of $^{238}PuO_2$ particles, possibly caused by radiolysis in tissue fluids.[53]

The physical properties of the inhaled particle also affect the retention of plutonium in the lung. Plutonium oxide prepared by calcining the oxalate at 1,000°C was retained in the lung with a half-time of from 650 to 950 days while plutonium oxide calcined at 350°C was retained with a half-time of from 300 to 400 days.[53] The particle size also affected the retention half-time in that the high temperature fired oxide with a smaller particle size was retained with a half-time of from 400 to 500 days; the high temperature fired oxides with a larger particle size were retained with half-times up to 900 days. In addition, the retention of $^{238}PuO_2$ in the small particle size was less than for comparable sized $^{239}PuO_2$.

Plutonium compounds deposited in the lung are not uniformly distributed and may deposit unequally among the lobes of the lung or among portions of the lung lobes. Studies conducted on rats and dogs show that plutonium is present in the lung in both particulate and nonparticulate form immediately following inhalation exposure.[53] This nonuniform deposition is evidenced by the presence of both alpha stars and single tracks in autoradiographs taken immediately after inhalation exposure. Autoradiographs taken among dogs exposed to inhaled $^{239}PuO_2$ reveal a relatively diffuse distribution of plutonium particulates throughout the entire lung.

A fraction of the plutonium deposited in the lung may be dissolved and absorbed into the blood.[53] Macrophages may engulf the remaining particles and aggregates. The phagocytized plutonium particles are rapidly localized in cells as witnessed by electron micrographs taken of plutonium dioxide particles localized in the lung cells of rats following inhalation exposure. The alveolar macrophage appears to be capable of transporting the particles and aggregates from the alveoli to the ciliated epithelium lining the bronchioles where the phagocytic cells can be removed from the lung in the mucous blanket that ciliary action propels up the respiratory passage. Plutonium removed by this route is swallowed and excreted in the feces. Further aggregation occurs when soluble and insoluble transuranium elements are not immediately cleared from the lung.[53]

Particles of plutonium in the lung are transported via lymphatic vessels and are collected in the thoracic lymph nodes. Autoradiographs of lung tissues from dogs several weeks and months after inhalation of plutonium dioxide show alpha stars concentrated in subpleural areas apparently in lymphatic vessel.[53] It is suggested that the particles themselves become immobilized in the scar tissue in these regions.

There is evidence to suggest that biological and physical forces act on particles deposited in the lung. Evidence argues against the concept that particles remain static indefinitely, except for those particles that become immobilized in scar tissue. Movement of the deposited particles may be seen as an attempt by the lung to expel the particles and may compensate for the nonuniformity of radiation exposure from the particles.[53]

3.2 Translocation

Translocation of plutonium from the lungs of beagles after inhalation of $^{239}Pu(NO_3)_4$ results in an accumulation of approximately 30% of the plutonium in bone and approximately 10% in the liver. The fraction of the plutonium remaining in the lung decreased to 40% or less of the amount deposited in the lower respiratory tract.[53] A small percentage of the plutonium was deposited in spleen, lymph nodes, and other soft tissue, and the remaining fraction was excreted in urine and feces.

Experimental data have shown that the lymphatic system accounts for a large fraction of the plutonium cleared from the lung.[53] An 11-year study with beagle dogs reveals that after 5 years the lung and thoracic lymph nodes each account for 30% of the $^{239}PuO_2$ initially deposited in the lower respiratory tract. At the end of the 11-year study the fraction of plutonium left in the lung was approximately 10% and the thoracic lymph nodes had accumulated 40%. Approximately 10% of the translocated plutonium was found in the liver, approximately 5% in bone, and approximately 7% in abdominal lymph nodes.

The relative concentration of plutonium in the tissues of dogs 7 to 9 years after inhalation of $^{239}PuO_2$ was highest in the thoracic lymph nodes and next highest in the abdominal lymph nodes. The thoracic lymph nodes had a relative concentration over 1,000 times greater than the concentration in either the lung or the liver. Relative concentrations in the lung were approximately 5 times the concentration in the spleen and approximately 10 times the concentration in the bone. The average radiation dose followed the same relationship as the relative concentration, indicating that the lymph nodes received the highest radiation exposure.

Experimental data have also shown that $^{238}PuO_2$ may be cleared from the lung much more rapidly than $^{239}PuO_2$. Data have shown that the translocation of ^{238}Pu from the lung may be greater than that of ^{239}Pu. A comparison of distribution of plutonium in the tissues of beagles 5 years after inhalation of $^{238}PuO_2$ or $^{239}PuO_2$ revealed that only 10% of the body burden of ^{238}Pu was in the lung while 46% of the ^{239}Pu remained in the lung. Three times as much ^{239}Pu accumulated in thoracic lymph nodes as ^{238}Pu and 12 times as much ^{238}Pu accumulated in bone as ^{239}Pu. Thus the data indicate that the behavior of $^{238}PuO_2$ may differ significantly from that of $^{239}PuO_2$.

The translocation of plutonium compounds from the lung may differ considerably depending mostly on the in vivo solubility of the particular compound. Translocation of the relatively soluble hydrated oxide of ^{238}Pu was mainly to bone in dogs 30 days after inhalation of plutonium oxide aerosols. The more stable oxides of ^{238}Pu and $^{239}PuO_2$ showed little translocation from the lung to other tissues.

The translocation of plutonium absorbed from the gastrointestinal tract, through the skin, or from wounds is similar to that observed for the respiratory tract. The proportion of translocation may vary because plutonium deposited in a wound or in the lung provides a reservoir for absorption into the blood stream.[53] Absorption from the skin during the interval prior to decontamination or from the gastrointestinal tract until the tract is cleared will be of short duration, except in cases of continuous exposure.

3.2.1 Deposition in Gonads

The deposition of plutonium in the gonads of experimental animals was studied because of a concern about possible genetic effects. Approximately 0.05% of the plutonium in the circulating blood may deposit in testes and approximately 0.01% in ovaries.[53] Approximately half of the plutonium in the testes appears to be associated with germinal tissue. At present there is no basis for assuming that testes or ovaries should be considered critical organs. Effects in these organs have been observed only following plutonium doses much higher than the doses that would have resulted in other evidences of toxicity.

3.2.2 Cross Placental Transfer

Experimental data indicate that the placenta is an effective barrier to the transfer of plutonium. Table IV J(C)-7 gives the experimental data on transfer of plutonium to the fetus as well as the placenta and placental membrane in the rat. In all cases transfer is very low and extremely high levels of contamination would have to be present in the maternal animal before appreciable quantities of plutonium would occur in the fetus.

Table IV J(C)-7

CROSS PLACENTAL TRANSFER OF TRANSURANIUM ELEMENTS IN RATS*

Time of Injection		Percent Ingested Dose per Gram		
(Day of Ingestion)		^{238}Pu	^{239}Pu	^{239}Pu**
15	Fetus	0.01	0.01	0.0002
	Placenta	0.22	0.40	0.01
	Membranes	3.80	1.93	0.03
19	Fetus	0.01	0.01	0.0004
	Placenta	0.51	0.91	0.02
	Membranes	4.30	3.20	0.07

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*Taken from Bair Report[53]

**Polymeric

4.0 BIOLOGICAL EFFECTS

Experimental data have shown that the distribution of plutonium in the body is dependent on the route of entry, the chemical compound and the particular radioisotope.[53] This section deals with the resultant biological effects which depend on the radiation exposure and the radiation sensitivity of each tissue. The tissues of primary concern are the blood, bone, liver, lung and lymphatic system. The effects examined are those that occur at low doses.

4.1 Blood

The major effects seen in blood cells will result from irradiation of the hematopoietic tissue in which the plutonium is deposited or from irradiation of blood circulating through tissues containing deposits of plutonium. Most of the hematogenic effects observed after deposition of plutonium in laboratory animals have occurred at doses above those that have resulted in cancer. In experiments with dogs the erythrocyte levels were reduced only after high dose of either ^{238}Pu or ^{239}Pu were administered by intravenous injection or inhalation. White blood cells show transient reductions following both intravenous injection and inhalation. Leukemia was not a common finding in laboratory animal experiments despite the fact that plutonium deposits in bone and lymph nodes.[50,51] The most consistent response was lymphopenia and it was seen after injection and inhalation of plutonium, and in current experiments with dogs it is seen as the most sensitive indicator of biological effect. Lymphopenia has been shown to occur at dose levels of inhaled $^{239}PuO_2$ that have not shown cancer. Although the health effects of a lymphopenic condition are not known, the possibility of a relationship between this effect and lymph node pathology, decreased immunological competence, and the pathogenesis of plutonium-induced lung tumors cannot be ruled out.[53]

4.2 Bone

Probably the most informative experiment on the toxic effects on plutonium in bone is the beagle dog study, still in progress at the University of Utah. This experiment was initiated in 1952 and was designed to compare the long term effects of intravenously injected plutonium and radium. The comparison with radium is of particular interest because much more is known about the toxicity of radium in human beings. If a ratio between the toxicity of radium and plutonium can be established in experimental animals, then this same ratio might be applied with some confidence to man.

Table IV J(C)-8 shows the current status of the plutonium injected animals in this experiment.[54] For the initially injected groups, all animals are now dead. In all of these groups, there was a very substantial incidence of osteosarcoma. With decreasing dose, the time to tumor appearance increases until, in the lowest dose group, the average lifespan is not significantly different from that of the controls--though the incidence of osteosarcoma is certainly significantly higher than in controls. When it became evident that effects were being seen in the lowest exposure group of the originally planned experiment, additional groups were exposed, extending to a low group of 0.0006 μCi/kg, which is the concentration equivalent to the occupationally permissible body burden for man. These animals were injected beginning in 1964 and nearly all of them are still alive.

Table IV J(C)-8

BONE SARCOMAS IN UTAH BEAGLES

Injected Dose (μCi/kg)*	Number of Sarcomas	Number of Deaths	Sarcoma Dogs Years to Death	Sarcoma Dogs Rads to Skeleton
2.9	7	9	4.1	4900
0.9	12	12	3.6	1300
0.3	12	12	4.5	600
0.1	10	12	7.2	310
0.05	9	13	8.5	190
0.016	4	13	9.9	78
Controls	0	12	11.5 (all dogs)	

---

*Additional studies are in progress at dose levels of 0.016, 0.006, 0.002, and 0.0006 μCi/kg

A number of long term studies in rodents have also pointed to osteosarcoma as the most sensitive indicator of the toxicity of injected plutonium. Figure IV J(C)-2 summarizes this data, including the Utah dog data.[55] It is clear that the dogs are more sensitive to bone tumor induction by plutonium than are either rats or mice. From these kinds of data, May and Lloyd, assuming a time independent linear dose response relationship, have calculated an increased incidence per rad of 0.38% for beagles, 0.10% for mice, and 0.06% for rats.[56] Since one rad of alpha irradiation of bone is equivalent to 50 rem, the increased incidence per rem is 1/50th of these numbers, or about 4×10^{-5} as an average for these three species. It is no doubt fortuitous that this number is within a factor of two of the maximum risk estimate of bone tumors per rem as derived from the BEIR report.[51] This agreement does nothing, of course, to support the absolute value of these risk estimates, since the linearity of the dose response curve assumed in both estimates is by no means established in the very low dose range of concern in general population exposures.

Of more interest than absolute incidence figures, and their uncertain extrapolation to man, is the finding in the Utah studies that plutonium-239 is five to ten times more toxic than radium-226, on the basis of the same total energy delivered to bone.[57] This difference is attributable to the more hazardous localization of plutonium on bone surfaces, whereas radium is distributed more uniformly throughout bone. The cells from which bone tumors originate are located near bone surfaces. Present occupational exposure limits for plutonium were established more than 20 years ago, in relation to the human radium toxicity data, on the assumption that plutonium was five times as toxic as radium, this assumption being based on limited rodent data available at the time. The Utah results seem consistent with that earlier judgment.

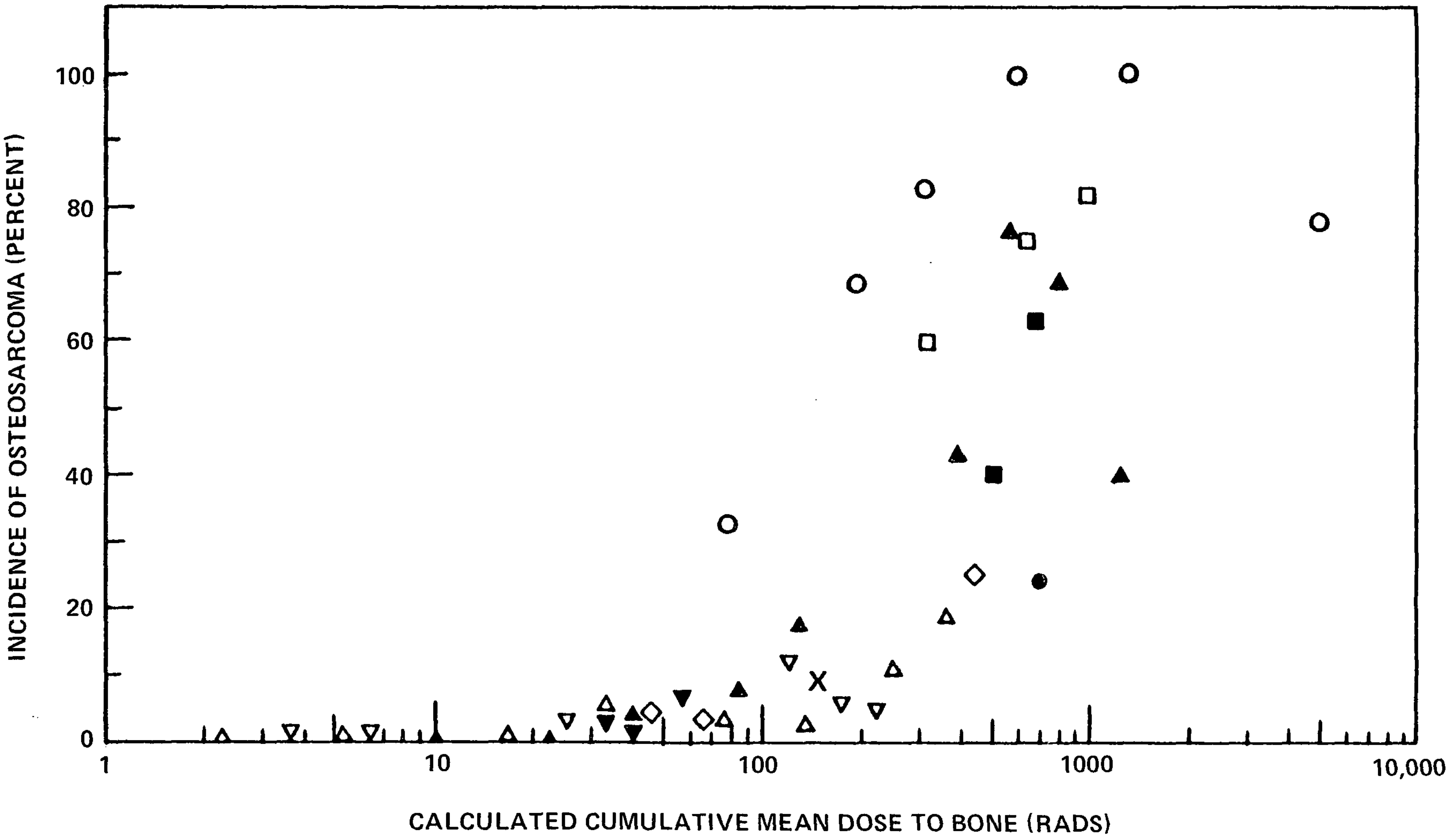

Figure IV J(C)-2 Plutonium - Induced Osteosarcoma in Experimental Animals

Inhalation of $^{239}PuO_2$ has not led to the development of osteogenic sarcomas in experimental animals.[53] Osteogenic sarcomas have been found in dogs after inhalation of $^{238}PuO_2$. Experimental data are given in Table IV J(C)-9. The sarcomas occurred as a result of translocation of ^{238}Pu from the lung to bone which subsequently received a higher radiation dose. A secondary finding in one of the dogs was lung cancer, although another dog had leukemia and fibrosarcoma as well as osteosarcoma.

4.3 Liver

In terms of plutonium content and radiation dose received, liver is in the same class with bone and lung. However, the liver seems less radiosensitive than bone and lung. Malignant liver tumors were the primary cause of death in two of 96 plutonium dogs at risk in the Utah experiment.[57] Small, benign bile duct tumors were incidental findings at autopsy in eight other dogs, but these were also seen in controls at a somewhat lower incidence. The liver tumors showed a typically long latent period, longer than the bone tumors, which might explain the lower incidence of liver tumors. This also suggests that at lower dose levels, and lower incidences of bone tumors, liver tumors might become proportionately more important. The longer lifespan of man might also favor the ultimate development of liver tumors.

4.4 Lung

In the case of inhaled plutonium, acute death results from pulmonary edema, hemorrhage, and necrotic destruction of the functional tissue of the lung.[52,58] The inhalation of relatively soluble plutonium compounds such as organic complexes, plutonium nitrate, and $^{238}PuO_2$ have resulted in primary lung cancer in rodents, rabbits, and dogs in addition to the bone cancer already mentioned.[53] Lung cancer has also occurred in rats after inhalation of $^{238}Pu(NO_3)_4$ and in beagle dogs, baboons, and rodents after inhalation of $^{239}PuO_2$. In the case of inhaled plutonium, the toxicity of inhaled plutonium may be illustrated with data from an experiment conducted at Battelle Pacific Northwest Laboratory, involving the inhalation of PuO^2 by beagle dogs.[59] This experiment was initiated in the late 1950's; it involved some 65 dogs, the last of which died in 1973. Of the 21 dogs that survived more than 4-1/2 years post exposure, 20 had malignant lung tumors; tumor incidence figures are therefore of little help in interpreting these results. Life shortening, however, showed a dose effect relationship, as indicated in Figure IV J(C)-3. Each point in this figure represents a dog. Most of the dogs died early, within a year or two, with symptoms of respiratory distress occasioned by severe pulmonary fibrosis. There appears to be about a 3-year minimum latent period before a tumor can develop; a longer latent period is associated with smaller doses. A line fitted to these points intersects the normal life span of the beagle dog at a deposition of about 5 nCi/g. This extrapolation is very uncertain, however. The best line drawn through the closed circles (the tumor-bearing animals) would produce a steeper slope. Clearly, more data is needed from dogs exposed at lower levels. Experiments to obtain this data are in progress.

Figure IV J(C)-4 is a summary of experimental data on plutonium induced lung cancer.[55] These are rat and mouse data, except for the squares, which represent

Table IV J(C)-9

OSTEOSARCOMA IN DOGS AFTER INHALATION OF $^{238}PuO_2$[*]

			Plutonium Distribution (1 Percent of Body Burden)				
$^{238}PuO_2$	Survival Time (Months)	Terminal Body Burden (μCi)	Lungs	Thoracic Lymph Nodes	Liver	Bone	Osteosarcoma
Calcined at 350°C	23-70	2.6-3.0	17	9	23	47	5/8[**]
Crushed microspheres	22-76	0.2-3	20	12	16	24	4/8[***]

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[*] Taken from Bair[53]

[**] One lung tumor

[***] One myelogenous leukemia and one fibrosarcoma

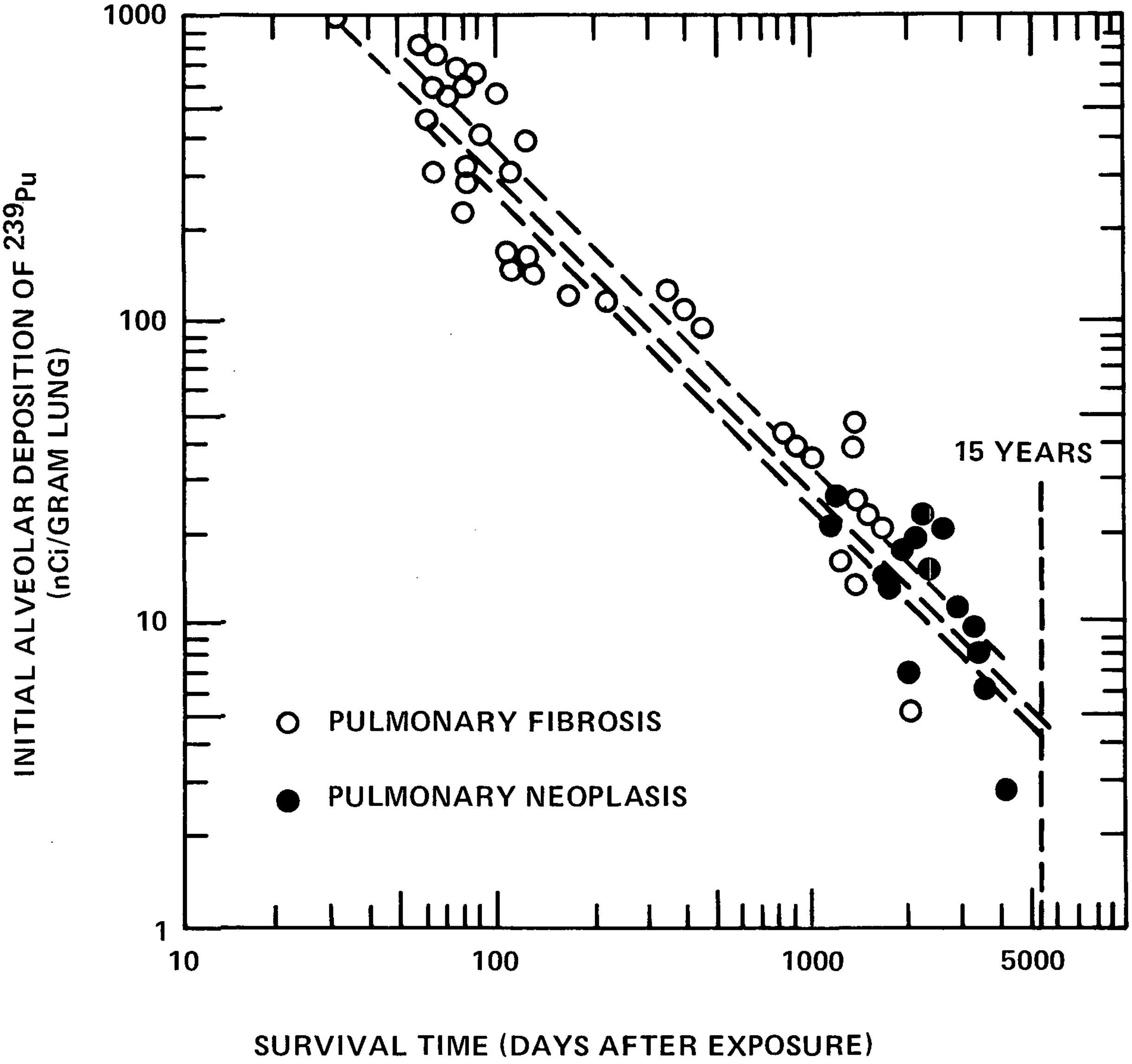


Figure IV J(C)-3 Survival Time of Beagles as a Function of PuO_2 Deposition

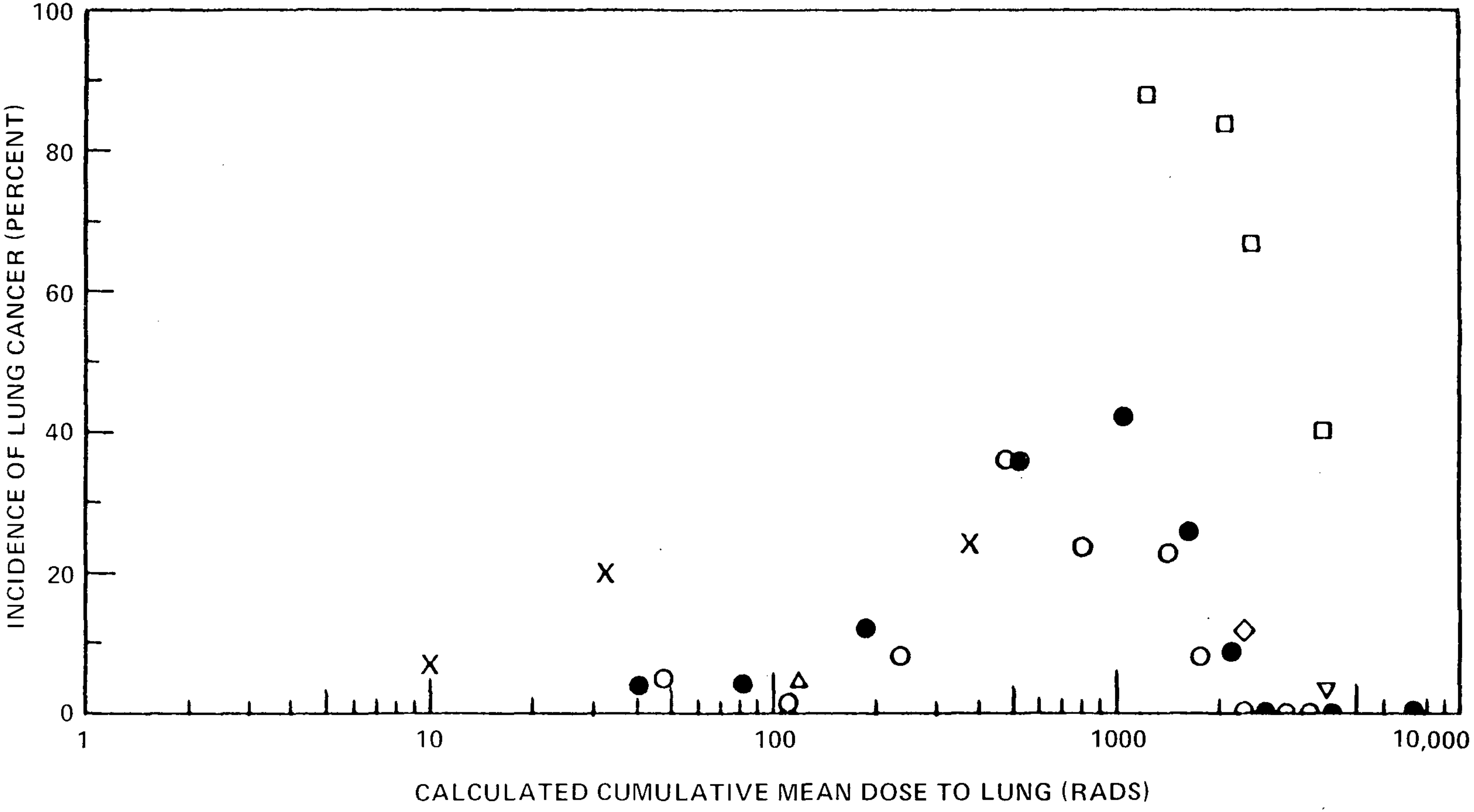


Figure IV J(C)-4 Plutonium - Induced Lung Cancer in Experimental Animals the beagles just discussed. Again, the dog seems to be appreciably more susceptible than rodents. However, there are no data on dogs below about 1,000 rads.

4.5 Lymph Nodes

Lymph nodes draining the lung, or sites of intramuscular plutonium deposition, may accumulate plutonium concentrations many times higher than concentrations seen anywhere else in the body. Primary cancer of lymphatic tissue has not occurred in dogs studied for 11 years following inhalation of $^{239}PuO_2$ or in rodents studied in life span experiments after inhalation of plutonium compounds. Metastasis to mediastinal lymph nodes and lymphatics occurred among dogs that had primary cancer but only one dog had a possible malignant lymphoma and it was confined to the mesenteric and mandibular lymph nodes. The data lead to the conclusion that the lymph nodes are not especially susceptible to the carcinogenic action of alpha radiation from plutonium.[53]

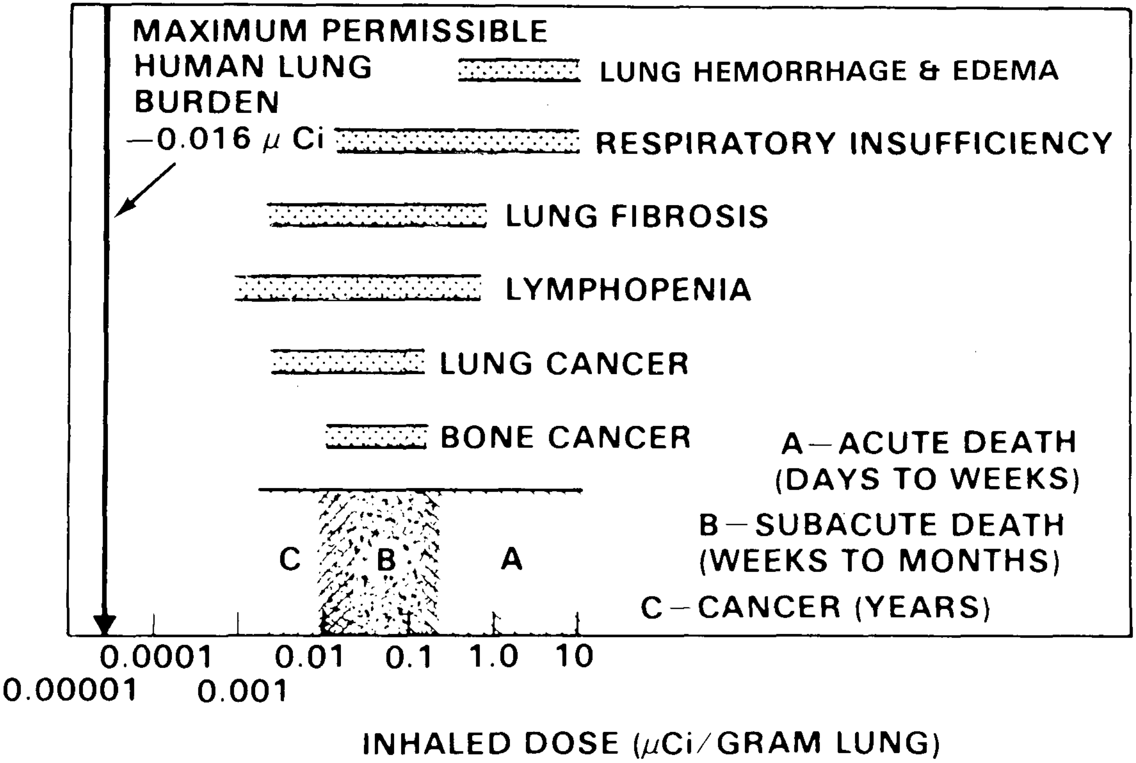
4.6 Relative Population Sensitivities

Another special area of concern is that related to possible effects on the unborn or the newborn. These are life stages that are apt to be most sensitive to radiation effects. Knowledge relative to the transfer of plutonium across the placenta and via milk secretion, and of the metabolism of plutonium in the fetus and infant, does not suggest that irradiation of the fetus or infant would be a critical factor in the exposure of populations to plutonium.[60,61] There are, however, few experimental data on toxic effects at minimal dose levels. Acute lethality data in the rat show no marked differences between adult, weanling, and newborn.[60] An experiment studying long term effects in rats exposed either prenatally, as newborns, weanlings, or adults, is still in progress. Preliminary data show no indication of an enhanced sensitivity of the very young. Injection of pregnant female rats at several times during gestation resulted in no significant effects on fetal mortality after doses less than about 5 μCi/kg, a relatively high dose.[60]

Perhaps the most pertinent data relating to the toxicity of plutonium in children are not plutonium data at all, but data on effects of ^{224}Ra in human children. Because of the short 3.6-day half-life of this radium isotope, it does not have time to become distributed throughout bone in the manner typical of radium, but irradiates the bone surface in a manner similar to plutonium. Data from 900 patients treated for tuberculosis, ankylosing spondylitis, and other diseases, showed that ^{224}Ra was approximately twice as effective in producing bone tumors in children as in adults.[62]

4.7 Status of Current Research

Present knowledge of the biological effects of inhaled plutonium is summarized in Figure IV J(C)-5. The dose levels for the various biological effects are all shown relative to the maximum permissible lung burden of 0.016 μCi/g lung set for occupational exposures.



(See Reference 53)

Figure IV J(C)-5 Observed Biological Effects of Inhaled Plutonium

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CHAPTER IV
Section J
APPENDIX D
PLUTONIUM "HOT PARTICLE" HYPOTHESIS

The "hot particle" hypothesis was advanced by the Natural Resources Defense Council, Inc. (NRDC) in a petition for rule making on February 14, 1974. The hypothesis suggested plutonium was much more hazardous than the analyses being used for radiation exposures indicated. The petition for rule making and the "hot particle" hypothesis were considered by NRC and all parts of the petition denied.

This Appendix contains the Denial of Petition for Rule Making published in the Federal Register, Vol. 41, No. 71 on April 12, 1976. The first page of this notice includes the five requests of the NRDC petition relating to hot particles as follows:

- Stay of approvals
- Establishment of exposure maxima
- Establishment of surface contamination levels
- Amendment 10 of CFR Part 100
- Holding of public hearings

[Docket No. PRM-20-5]

NATURAL RESOURCES DEFENSE COUNCIL

Denial of Petition for Rule Making

Notice is hereby given that the Nuclear Regulatory Commission has denied a petition for rule making submitted by letter dated February 14, 1974, by the Natural Resources Defense Council, Inc., 1710 N Street NW., Washington, D.C. A notice of filing of petition, Docket No. PRM-20-5, was published in the FEDERAL REGISTER on March 28, 1974 (39 FR 11450). Interested persons were invited to comment on the petition. Six letters were received opposing the petition, and two letters were received which supported it. The supporting letters, from the West Michigan Enivornmental Action Council, Inc. and from the Citizens' Action Group for Safe Energy Sources, petitioned the Nuclear Regulatory Commission to take the same action as requested by the Natural Resources Defense Council. These petitions have also been denied.

The Natural Resources Defense Council (NRDC) petitioned the Atomic Energy Commission (AEC) to establish specific health protection standards for "hot particles," defined by NRDC as particles containing 0.07 picocuries or more of alpha radioactivity and yet sufficiently small to be inhaled and deposited in the lung. The petition contained the following requests:

1. Stay approvals for new construction or operation of facilities involving "hot particle" materials, and stay approvals for increase in quantity of "hot particle" materials for previously approved operations, until the petitioner's requests for modification of associated standards are resolved.

2. Establish, for occupational exposure, a maximum permissible lung particle burden of two "hot particles," and for non-occupational exposure a maximum permissible lung particle burden of 0.2 (average) "hot particles;" add concentration values to 10 CFR Part 20 for all alpha-emitting radionuclides which could form "hot particles," as defined by NRDC, each value to be a factor of 115,000 smaller than the value given for the radionuclide when not in "hot particle" form.

3. Establish, for unrestricted areas, a maximum permissible surface contamination level of one "hot particle" per square meter.

4. Amend 10 CFR Part 100 by adding a site criterion guide of 10 "hot particles" deposited in the lung during a two-hour exposure under accident conditions.

5. Convene public hearings to determine as-low-as-practicable regulations for materials in "hot particle" form.

In denying the petition the Nuclear Regulatory Commission (NRC) denies all five of these requests. This follows from the fact that the NRC finds that scientific evidence does not support the technical position upon which the NRDC petition is based. This technical position is stated by the NRDC in the corollary to the "hot particle" hypothesis, as discussed below.

By letter dated March 27, 1975, the NRDC requested that the NRC conduct quasi-adjudicatory hearings in connection with the NRDC petition. However, public hearings were conducted by the Environmental Protection Agency on December 10–11, 1974, and January 10, 1975, which included the subject of standards for protection against plutonium and other transuranic elements. The "hot particle" question was addressed during those hearings, and very little pertinent information was presented beyond that presently available in the open literature. The Commission also had the benefit of meetings with the petitioner's consultants and others knowledgeable in the field. The Commission believes that the supporting information cited in the petition, and the large body of available information in the open scientific literature, provide an adequate basis for a thorough examination of the merits of the petition. In the light of this, and in the light of the fact that examination and cross examination as in a formal hearing are not likely to produce additional useful scientific information in this complex, scientific field, the Commission does not believe that holding of formal hearings would be in the public interest.

The sections which follow contain: (A) background information concerning the question at issue, (B) a discussion of the formulation of the NRDC hypothesis and its corollary, (C) a critical analysis of the hypothesis and its corollary, (D) the conclusions of the NRC, (E) a discussion of the basis for existing standards for insoluble plutonium, and (F) a summary of ongoing work which will be important to the NRC in its future considerations of radiological protection standards for insoluble plutonium.

A. BACKGROUND

1. *Spatial Distribution of Dose.* An important issue involved in this petition is the spatial distribution within the lung of radiation dose due to deposted alpha-emitting particles. Such particles irradiate immediately surrounding tissues intensely, but may leave other more distant tissues unirradiated. The radiobiological issue is whether, for a given quantity of radioactive material in the lung, the risk of cancer is greater for discrete particles distributed nonuniformly in the lung tissues or for material that is distributed uniformly throughout the lung. Present recommendations of the National Council on Radiation Protection and Measurements (NCRP) and the International Commission on Radiological Protection (ICRP), present guidance to Federal agencies issued by the Federal Radiation Council (now incorporated in the Environmental Protection Agency), and present NRC standards, are based upon the premise that nonuniform distribution is not more hazardous that uniform distribution. The petitioner takes the position that nonuniform distribution can be much more hazardous and that special, extremely restrictive standards are needed to limit exposure to alpha-emitting particles such as those containing plutonium-239.

2. *Current NRC Standards.* The NRC's current standards for protection against radioactive material, implicitly including materials in "hot particle" form, are specified in 10 CFR Part 20, "Standards for Protection Against Radiation." The particular standards to which the petition is addressed are given in 10 CFR Part 20 as limiting concentrations of radioactive materials in air for occupational exposure and limiting concentrations for radioactive materials in effluents to unrestricted areas, and provisions for limiting quantities of radioactive material in air or water.

In its first memorandum to the President (25 FR 4402, May 18, 1960) the Federal Radiation Council (FRC), pursuant to Section 274h of the Atomic Energy Act, recommended that Federal agencies use radioactivity concentration guides consistent with the Radiation Protection Guides given in the same memorandum. The Radiation Protection Guide for the lung was 15 rems per year, occupational. The concentration values for insoluble nuclides listed in 10 CFR Part 20 were at that time, and are still, based on a dose rate of 15 rems per year to the lung. With regard to nonoccupational exposure, the FRC recommended in the memorandum that protection guides in use by the Federal agencies be continued. These recommendations were approved by the President as guidance to Federal agencies (25 FR 4402, May 18, 1960). The occupational and nonoccupational concentration values in 10 CFR Part 20 were consistent with this guidance when it was issued. Subsequently, all functions of the FRC were transferred to the Administrator of the Environmental Protection Agency (EPA) by Reorganization Plan No. 3 of 1970 (35 FR 15623, October 6, 1970). EPA has not altered the guidance issued in the FRC's first memorandum to the President, and therefore the NRC's regulations remain consistent with guidance to Federal agencies pursuant to Section 274h of the Atomic Energy Act.

3. *Action Taken Due to Petition.* Although the standards in 10 CFR Part 20 are consistent with FRC guidance, upon receipt of the NRDC petition it was determined by the AEC that the results of pertinent research programs and the status of scientific evidence should be reevaluated. Scientific personnel most closely associated with relevant research programs were requested by the AEC to perform a study of current radiobiological evidence. The results of this study have been published in a report entitled "A Radiobiological Assessment of the Spatial Distribution of Radiation Dose from Inhaled Plutonium," by W. J. Bair, C. R. Richmond and B. W. Wachholz, WASH-1320, dated September 1974 (see Section C-4 below). Copies of this report may be obtained from the Superintendent of Documents, U.S. Government Printing Office, Washington, D.C. 20402 (price $1.10).

Because the AEC had for many years looked to the NCRP for authoritative guidance on radiation protection standards, upon receipt of the petition the AEC joined the EPA, which had received an identical petition, in requesting the NCRP to provide its views on the adequacy of existing radiation protection standards as related to radioactive particles deposited in the lungs, with particular emphasis given to the technical questions raised in the petition. EPA and the AEC made a similar request to the National Academy of Sciences (NAS). The resulting NCRP Report No. 46, "Alpha-Emitting Particles in the Lung," was issued July 1, 1975 (see Section C-5 below). A report from the NAS is expected in 1976.

After the AEC was abolished and its regulatory functions vested in the NRC, the NRC staff completed a review of available information and data bearing on the petition. This review included WASH-1320, many of the references cited in that report, and other references cited in this denial. The following document was also reviewed:

> Tamplin, A. R. and Cochran, T. B., "A Critique on the Biophysical Society's Draft Comments on 'Radiation Standards for Hot Particles'." NRDC, December 1974.

This document is available for inspection in the NRC's public document room, file PRM-20-5.

With regard to two instances of human hand exposure to plutonium discussed in the NRDC petition, Dr. C. C. Lushbaugh and Dr. Neil Wald were asked for their medical opinions. Their replies have been placed in the public document room and are discussed below.

On January 9, 1975, members of the AEC (now NRC) staff met with Thomas B. Cochran of NRDC to discuss the petition, and on January 30, 1975, the staff met with the authors of WASH-1320 for the same purpose. Minutes of these meetings, which were useful in elucidating the issue of the petition but which did not add substantive new information are available in the public document room.

B. NRDC Position

1. *Hypothesis and Corollary.* In reports written in connection with the petition[1, 2] the authors provide a hypothesis for cancer induction as caused by the irradiation of tissue. According to this hypothesis, if the dose to a critical tissue mass is sufficiently large, there is a high probability of tumor production. Thus the hypothesis restates a generalized, widely accepted conclusion on the biological effects of radiation. Also developed is a corollary to the hypothesis. According to this corollary, if the human lung is irradiated by an immobile, alpha-emitting particle of sufficient activity, a lesion will develop; if the lesion develops in a particularly susceptible type of tissue, the carcinogenic risk is high. Thus the corollary, on which the petition is based, is concerned directly with cancer as caused by lesions in critical lung tissues and is concerned only with radiation doses sufficiently high to cause such lesions. The corollary does not deal with particles of insufficient activity to cause a lesion, and the authors make no recommendations regarding such particles. Similarly, no distinction is made between lesion-forming particles of varying activity. The corollary assumes that the same probability of causing cancer is associated with all particles that can be deposited in the lung and that can cause a lesion.

The existence of a particularly susceptible type of tissue in the lung is not addressed in the NRDC report; the authors assume that such tissues are present. It appears to the NRC from the NRDC supporting documents that these critical tissues would be located in the deep lung. For small particles within the size range given in NRDC's "hot particle" definition, the deep lung tissues are much more heavily irradiated than tissues in upper respiratory passages where particle removal is relatively rapid.

2. *Quantification of Corollary.* In order to quantify this corollary, as is necessary in the development of a standard for personnel protection, the threshold dose or dose rate to form a lesion in the critical tissue would have to be determined, and an estimate of the cancer risk per lesion would have to be made. A large portion of the NRDC report is devoted to such quantification. For the threshold dose, 1000 rems was adopted by NRDC, based primarily on experiments involving irradiation of rat skin.[3 4 5] In these experiments there was little carcinogenic response below doses of 1000 rems. With respect to the rate of dose administration, NRDC selected 1000 rems in one year and justified the one-year period by estimating that the epithelial cell turnover time in the lung is about one year.

This selection of a threshold dose and time period permitted the NRDC to estimate the minimum quantity of activity necessary to cause a lesion—0.07 picocuries. Thus a "hot particle" was initially defined as containing 0.07 picocuries or more of alpha radioactivity and yet sufficiently small to be inhaled and deposited in the lung. (The definition was later changed to 0.14 picocuries as discussed subsequently under Subtitle 9, Human Inhalation Exposure.) This definition presumes the particle to be immobile for one year.

3. *Risk Estimate and Proposed Standards.* Quantification of the corollary also required a risk estimate, i.e., the cancer risk per lesion. For this estimate the NRDC again used data from the rat skin irradiation experiment mentioned previously. D. P. Geesaman,[6] in his study of the rat skin data, concluded that the risk probability is 10^{-3} to 10^{-4}. The NRDC selected the approximate midpoint of this range, viz, 5×10^{-4}, or one tumor per 2000 lesions. Since the only particles under consideration are those which cause lesions, this risk can also be expressed as $n(1/2000)$ cancers per n particles. To determine an appropriate occupational value for n, i.e., the permissible number of particles in a worker's lungs, the NRDC concluded that the risk from n particles should be no greater than the risk from the occupational external dose limit of five rems per year to the whole body. This risk can be estimated from risk factors reported by the NAS to be approximately 1/1000.[7] The NRDC equated these risks

$$n(1/2000) = 1/1000$$

to obtain two particles as the appropriate value for n.

Since a "hot particle," as defined by the NRDC, must contain at least 0.07 picocuries, the minimum activity permissible in the lung in "hot particle" form would be 0.14 picocuries. The present occupational limit is 16,000 picocuries for all forms of alpha-emitters. The NRDC concludes that new standards for materials in "hot particle" form should be established, and that these new standards should be a factor of 16,000 0.14 (or about 115,000) lower than the current standards for such materials in insoluble form.

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[1] "Radiation Standards for Hot Particles," by A. R. Tamplin and T. B. Cochran, February 14, 1974.

[2] Tamplin, A. R. and Cochran, T. B., "NRDC Supplemental Submission to the Environmental Protection Agency Public Hearings on Plutonium and Transuranium Elements," February 27, 1975.

[3] Albert, R. E., Burns, F. J., and Heimbach, R. D., "The Association Between Chronic Radiation Damage of the Hair Follicles and Tumor Formation in the Rat," Radiation Research, 30, 1967.

[4] Albert, R. E., Burns, F. J., and Heimbach, R. D., "The Effect of Penetration Depth of Electron Radiation on Skin Tumor Formation in the Rat," Radiation Research, 30, 1967.

[5] Albert, R. E., Burns, F. J., and Heimbach, R. D., "Skin Damage and Tumor Formation from Grid and Sieve Patterns of Electron and Beta Radiation in the Rat," Radiation Research, 30, 1967.

[6] Geesaman, D. P., "An Analysis of the Carcinogenic Risk from an Insoluble Alpha-Emitting Aerosol Deposited in Deep Respiratory Tissue," UCRL-50387 and Addendum, 1968.

[7] "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," Advisory Committee on Biological Effects of Ionizing Radiations, National Academy of Sciences—National Research Council, 1972.

4. *Extrapolation of Risk Factor to Human Lungs.* The basis for the corollary is the postulate that lung cancer can be caused by lesions located within sensitive, or critical, human lung tissues. The foundation for this postulate is the series of rat skin irradiation experiments referred to above. These experiments involved electron irradiation of relatively large areas of skin on a large number of rats.[3,4,5]

These irradiations resulted in a high incidence of skin tumors. It was noted by the experimenters that the tumors formed primarily within hair follicles, and that the tumors were correlated with the disruption of the hair follicles in a rough proportion of one tumor per 2000 atrophied follicles. The NRDC corollary, as mentioned previously, suggests that the human lung also has a particularly sensitive tissue that can be disrupted by an alpha-emitting particle, and that the cancer risk due to such disruption is also 1/2000.

C. *Analysis of the NRDC "Hot Particle" Corollary.* The NRDC petition to establish specific health protection standards for "hot particles" raises the issue of the health effects of certain radioactive materials in the human lung. The issue as *viewed by NRC relates to the effects of* these materials in the lung as discrete, insoluble, and immobile particulates on the one hand, or as materials distributed uniformly within the organ on the other hand. Central to the issue is whether the biological evidence presently available supports continued use of the NRC's present standards for insoluble, alpha-emitting nuclides in particulate form, or whether the "hot particle" corollary, as provided by the petitioners, can be supported sufficiently by this evidence to form the basis for new health protection standards in the NRC regulations. The hypothesis as most recently stated by Tamplin and Cochran is:[1]

> When a critical tissue mass is irradiated at a sufficiently high dose, the probability of tumor production is high.

The corollary is:[2]

> When a critical tissue mass in the lung is irradiated by an immobile particle of sufficient alpha activity the probability of a lesion developing approaches unity, and the probability of this lesion developing into a tumor is high.

Evidence supporting the plausibility of the hypothesis can be obtained from studies of tumor incidence of rat skin subjected to ionizing radiation. A discussion of this experimental work is provided in the following paragraphs.

1. *Irradiated Rat Skin Experiments.* Albert and co-workers[3] irradiated defined areas of rat skin of the Sprague-Dawley strain with single exposures of electrons having maximum penetration of 0.5, 1.0 and 1.5 millimeters. They observed that in the non-ulcerogenic dose range the ratio of the number of tumors to atrophic hair follicles was between 1/2000 to 1/4000. When tumor incidences and atrophic hair follicles were related to dose as a function of depth below the skin surface, coincident incidence-dose curves were found at depths of 0.27 millimeters.[3] It was noted by the investigators,[3] who observed that atrophic hair follicles diminished near the margin of the irradiated areas, that:

> these observations strongly suggest that the pathogenic mechanisms for the development of both irreparable hair follicle damage and skin tumors depend on both the dose and the amount of skin irradiated.

Radiation experiments were also carried out on rat skin using grid and sieve patterns[5] of dose delivery. It was concluded for the non-uniform radiation patterns that both chronic hair follicle damage and tumor formation were reduced by these patterns of dose delivery within a limited dose range.

In the experiments described above, a tumor response curve was observed that was closely proportional to a hair follicle atrophy response curve. These curves appeared to exhibit a threshold (i.e., the biological response appears to begin) at about 1000 rads delivered at 0.27 millimeters beneath the skin surface. A maximum response was observed at about 2000 rads delivered at this depth, followed by a rapidly decreasing response at doses greater than 2000 rads.

On the basis of the above, it can be inferred that enhanced tumor incidence for the skin of the Sprague-Dawley rat strain results from hair follicle damage (atrophy) caused by the irradiation of relatively large areas of the skin. The evidence suggests that a dose threshold for enhanced adnexal (i.e., follicle or sebaceous) cancer incidence may exist at about 1000 rads when measured at a depth below the skin of 0.27 millimeters and that the incidence curve passes through a maximum at about 2000 rads and then diminishes with increasing energy deposition. The experiments further suggest that tumor formation occurs in the ratio of 1/2000 to 1/4000 to hair follicle atrophy and is dependent upon the amount of skin irradiated in addition to the number of atrophied follicles.

A study was conducted by Passonneau, et al.,[8] in which the tumor incidence of rat skin was measured versus the activity of Sr-90/Y-90 sources in the form of beads or plates. The results of this study, as summarized by Bair, et al.,[9] clearly indicate that the efficiency for tumor production, in tumors per microcurie, increased with increasing uniformity of irradiation (i.e., from high activity beads to flat plate sources).

However, the data provided by Passonneau, et al., have been analyzed by the NRC in another manner. Based upon estimates of those actual areas subject to doses exceeding 1000 rads by both particles containing Sr-90/Y-90 or by flat plates with uniform Sr-90/Y-90 activity distribution, the number of tumors produced per unit area of rat skin actually irradiated to 1000 rads or more is nearly constant, indicating that nonuniform irradiation was as hazardous as uniform irradiation. Although no estimates were obtained of hair follicle damage as a consequence of these studies, the work of Passonneau, et al., appears to be consistent with the work of Albert, et al., referenced earlier. These data contribute the only evidence for the existence of a "critical tissue mass" in animals or man contained in submittals to the NRC by the NRDC.

2. *Critical Tissues in Human Lung.* Critical tissues from the standpoint of cancer origination have also been indicated to exist in human lungs.[7] These critical tissues constitute the basal cell layer of the bronchial epithelium. In the bronchial region of the lung, the residence time of particles is short because they are trapped in mucus, moved to the pharynx by action of the epithelial cilia, and are then swallowed. The deep lung regions of interest to the "hot particle" question (i.e., regions of lower particle mobility) are the respiratory bronchioles, the alveolar ducts and alveolar sacs. The NRC has no evidence that indicates the existence of tissue that might be described as "critical" or of "critical tissue mass" within these regions of the lung.

The corollary of the NRDC is apparently provided support only by experiments conducted on rat skin. The postulate by the NRDC of the existence of "critical tissue mass(es)," located in the deep lung, is not supported by available information and is considered to be highly speculative. Experience with uranium miners indicates that critical tissues probably do exist in the respiratory epithelium of the human bronchus (an upper region of the lung) in which tumors may originate more probably than in other cells in the lung following irradiation by the short-lived daughter products of Rn-222. However, the NRDC petition deals with particles lodged immobile in the deep region of the lung in which there is no evidence of critical tissue masses.

3. *Difference in Response Between Rat Strains.* In the initial experiments conducted by Albert and his co-workers concerning the irradiation of rat skin[10] and discussed by Healy, et al.,[11] Y-91 was the source of irradiation and two strains of rats, the Holzman and the Sprague-Daw-

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[8] Passonneau, J. V., Brues, A. M., Hamilton, K. A., and Kisieleski, W. E., "Carcinogenic Effects of Diffuse and Point Source Beta Irradiation On Rat Skin: Final Summary," ANL-4932:31, 1952.

[9] Bair W. J., Richmond, C. R., and Wachholz, B. W., "A Radiobiological Assessment of the Spatial Distribution of Radiation Dose from Inhaled Plutonium," U.S. Atomic Energy Commission, WASH-1320, September, 1974.

[10] Albert, R. E., Newman, W., and Altshuler, B., "The Dose-Response Relationships of Beta-Ray—Induced Skin Tumors in the Rat," Radiation Research, 15, 1961.

[11] Healy, J. W., Richmond, C. R., and Anderson, E. C., "A Review of the Natural Resources Defense Council Petition Concerning Limits for Insoluble Alpha Emitters," LA-5810-MS, Los Alamos Scientific Laboratory, November, 1974.

ley, were used. The Holzman strain is considered to be similar to the Sprague-Dawley strain, but the Holzman rats in these experiments were considerably older than the Sprague-Dawley rats. The dose-response curves of the two strains were observed to differ quite markedly for the principal type of tumor that resulted. In the case of the Sprague-Dawley strain, the onset of tumor response appears to be well-defined and begins at about 2000 rads dose delivered to the skin surface. For the Holzman strain, a less well-defined but discernible threshold appears near the same dose delivery value as the Sprague-Dawley strain. However, the response at the maximum for the Sprague-Dawley strain is greater than that of the Holzman strain by about a factor of five. It is not evident that this striking difference in response is related to strain or age at irradiation. Such differences suggest, however, that the extrapolation of irradiation response characteristics to similar tissue within an animal species is highly uncertain. The validity of the extrapolation of irradiation response characteristics of a particular tissue and species to a dissimilar tissue of a different species greatly compounds the uncertainties.

4. *Difference in Response Between Rodent Species*. Further evidence of differences that can occur between species is provided in the work of Albert, et al.,[12] and discussed by Healy, et al.,[11] in examining the tumor response of mouse skin to irradiation. The authors confirmed that while under certain conditions the rats exhibited adnexal tumors in response to skin doses, this outcome was rare in mice. Furthermore, the total number of tumors produced in mice under these conditions was only 15% to 20% of the total produced in rats. The decreased frequency of adnexal tumors and atrophied hair follicles in mouse skin relative to rat skin can be attributed to a greater lethal sensitivity of mouse hair follicles to radiation than rat hair follicles. (It appears that the mouse hair follicles may have been destroyed in this experiment.) This conclusion indicates the difficulties that can be encountered by attempting to impose the characteristics of one species onto another. The characteristic behavior of the skin of Sprague-Dawley rats to radiation has no known relevance to the behavior of the human lung other than the general observation that cancer can be induced in either type of tissue as a consequence of irradiation.

5. *Partial Irradiation of "Critical Tissue Mass"*. A further element of the NRDC corollary is that it could be assumed that irradiation at high levels of dose of only a portion of a "critical tissue mass" would result in a high probability for tumor production. However, as reported by Albert, et al.,[12]

> the development of both irreparable hair follicle damage and skin tumors depend upon both the dose and the amount of skin irradiated.

Further studies of rat skin tumor induction with ionizing radiation[13] indicated that upon using alpha particles and protons, no tumors were produced where the ranges of the particles extended to about 0.15 millimeters below the skin surface. The investigators found that no tumors or atrophied hair follicles were observed for irradiation depths of 0.3 millimeters under alpha particle irradiation unless the entire hair follicle was substantially irradiated. The significance of these findings, according to Albert is:[14]

> This observation suggests that even though the critical cell population is located at 0.3 mm, that there are recovery mechanisms that block tumorogenesis when only parts of the 'critical architectural unit of tissue' is irradiated. What these recovery processes might be is not understood. Nevertheless, this result does not support the contention that a single plutonium particle positioned next to a 'critical architectural unit' such as the hair follicle, will produce a tumorogenic risk of the magnitude assumed by Tamplin and Cochran.

On the basis of the above, "critical tissue mass(es)" in rat skin for which there is evidence, requires substantial irradiation of the entire structure before hair follicle ("critical tissue mass") tumors are induced. Thus, experimental support for the corollary of the NRDC is restricted to conditions where a "critical tissue mass" is entirely irradiated. It should be noted that Tamplin and Cochran, in their development of the 1/2000 risk factor, apparently did not take the recovery mechanisms reported by Albert[14] into account.

(NOTE: As pointed out earlier, the evidence for a "critical tissue mass" is supported only by radiation experiments involving rat skin. The corollary presumes the existence of "a critical tissue mass in the lung," sufficiently small to be entirely irradiated, but not destroyed, by a "hot particle." The NRC knows of no evidentiary support for this speculative assertion. However, in the discussion of the corollary which follows, the existence of such a "critical tissue mass in the lung" is hypothetically assumed.)

6. *Particle Immobility*. As provided in the corollary given by the NRDC, the source of radiation for "critical tissues mass in the lung" must be an immobile particle to satisfy the requirements of the corollary. Although this may be a necessary condition to aid in establishing the validity of the corollary, its relevance to inhaled particulates of insoluble plutonium in the lungs does not appear to be substantial. As provided in the report of Bair, et al.,[9] in the upper lung particles are efficiently and rapidly removed, principally by mucociliary mechanisms. In the lower lung, particles are subjected to gradual dissolution, followed by absorption into the blood or removal by macrophages; these appear to be the prime mechanisms for plutonium transport. *Evidence is available, however, to* indicate that some plutonium particles can be immobilized in scar tissue in the lung. Bair, et al.,[9] summarize the knowledge of particle mobility in the lung in the following statements:

> Although the kinetics are unknown and even a qualitative description is still rather primitive, there is ample evidence that plutonium deposited in lung is subjected to biological and physical forces. This argues against either particles or aggregates of plutonium remaining static indefinitely, except for the plutonium that becomes immobilized in scar tissue. To the contrary, while the rates may be low, movement of plutonium within lung tissues, by several mechanisms, certainly occurs, as the lung attempts to expel the plutonium and other foreign material. The migration of deposited plutonium particles in lung is recognized in the USSR as at least partially compensating for the nonuniformity of the radiation exposure from plutonium particles and justifying acceptance of the concept of averaging the radiation dose over the entire lung mass. (A reference of Bair, et al., is deleted here.)

Since all particles are not immobile in the lung, the probability of particle immboility should be considered in the estimate of risk.

7. *Alpha Induced Lesions in Rat and Hamster Lungs*. The corollary to the hypothesis of the NRDC states that under specific lung irradiation and tissue conditions "the probability of a lesion developing approaches unity, and the probability of this lesion developing into a tumor is high." As applied to alpha-emitting particles in the lung, the NRDC states:[15]

> If a particle deposited in the deep respiratory tissue is of such activity as to expose the surrounding lung tissue to a dose of at least 1000 rems in 1 year, this particle represents a unique carcinogenic risk. The biological data suggest that such a particle may have a cancer risk equal to 1/2000.

The petitioners do not explicitly define a lesion, but they assume[1] that lung tissue with a mass of 65 micrograms surrounding an alpha-emitting particle, that receives an average dose of 1000 rem or more per year, will have a probability of essentially unity for the development of a lesion, and that such a lesion would constitute a cancer risk of 1/2000.

Lesions have been observed surrounding plutonium-238 oxide particles with diameters ranging from 122 to 207 micrometers lodged in the blood vessels of rat lungs by intravenous injection.[16] The

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[12] Albert, R. E., Burns, F. J., and Dermott, P., "Radiation-Induced Hair Follicle Damage and Tumor Formation in Mouse and Rat Skin," *J. Nat'l Cancer Inst.*, 49(4), 1972.

[13] Heimbach, R. D., Burns, F. J., and Albert, R. E., "An Evaluation by Alpha-Particle Bragg Peak Radiation of the Critical Depth in the Rat Skin for Tumor Induction," Radiation Research, 39, 1960.

[14] In "Plutonium and Other Transuranium Elements: Sources, Environmental Distribution and Biomedical Effects," U.S. Atomic Energy Commission WASH-1359, December 1974.

[15] Tamplin, A. R., and Cochran, T. B., "The Hot Particle Issue: A Critique of WASH-1320 as It Relates to the Hot Particle Hypothesis," Report of the Natural Resources Defense Council, Inc., Washington, D.C., November, 1974.

[16] Richmond, C. R., Langham, J., and Stone, R. S., "Biological Response to Small Discrete Highly Radioactive Sources, II. Morphogenesis of Microlesions in Rat Lungs from Intravenously Injected $^{238}PuO_2$ Microspheres," Health Physics, Vol. 18, 1970.

alpha dose rate to tissue within 40 micrometers of a 180-micrometer particle (the average diameter of these particles was 175 micrometers) was about 6×10^6 rems per hour and the photon dose delivered to a distance of 300 micrometers was estimated to be about 830 rads per day. The authors described a lesion found in one rat lung as similar to that reported by Lushbaugh, et al.,[17] describing a plutonium lesion found in the palmar dermis of a plutonium worker. Richmond, et al.,[18] have described the experimental results further. They state:

> Microlesions caused by exposure of rat lung tissue to high specific-activity $^{238}PuO_2$ microspheres for 1–211 days were examined histologically. The huge radiation dose rates ($\sim 10^6$ rad/hr for alpha particles and $\sim 10^6$ rad/hr for photons) caused surprisingly little change in the lung structure except in the immediate area of the particle. The lesion progresses from a highly cellular to an acellular, collagen surrounded state and appears to be limited in size after several months.

In the experiment, no cancers developed in the animals. Of the 38 animals under study, 32 were sacrificed 120 to 400 days after particle implantation, and six died of natural causes.

A significant study with hamsters has been conducted at Los Alamos Scientific Laboratory which allows direct testing of the corollary of the NRDC hypothesis. In this experiment, implantation of plutonium particles was carried out by intravenous injections, as in the previously described experiments with rats.[18] In the course of this study lesions were observed. This work has been summarized by Healy, et al.,[11] as follows:

> In an experiment currently in progress,* uniform-sized microspheres (10-μm-diameter) of ZrO_2 are used with intermixed PuO_2 to provide particles of differing activities, and these are introduced into the lungs of hamsters by the above technique. In the first study in this experiment, 8 groups of 60 animals each were injected with 2000 such particles, with the plutonium content of each particle ranging from 0.07 to 59.4 pCi.
>
> Essentially all of the animals have now died, with only two lung cancers observed. (Three other cancers in the exposed animals occurred in organs other than the lung.) The dose rates to the lungs of those animals, when calculated as the average dose rate to the lung, ranged from 13 rads per year (130 rems per year) to 12,000 rads per year (120,000 rems per year). This is a range over which one would expect high tumor incidence and, in fact, premature death from pulmonary inefficiency if the material had been distributed homogeneously. Since the survival curves of the individual groups did not differ from those of the controls and the total tumor incidence was low, one can only conclude that the DF (Distribution Factor) for plutonium in particulate form must be less than one. In the continuation of this study, some 1900 hamsters have received 1.6×10^5 microspheres.* As of October 1974, the minimum time of exposure has been 60 weeks, which is comparable to or longer than the tumor induction times observed by Little, et al., in their experiments with more uniformly distributed ^{210}Po. In fact, only three lung tumors (including the two observed in the first study) have, as yet developed from the microsphere exposures. While this study is as yet incomplete, the very low tumor incidence again indicates a low effectiveness of the particles in inducing lung cancers as compared to more homogeneously distributed alpha emitters, as well as the failure of the Geesaman hypothesis to correctly forecast the results of this experiment.

In describing some of the effects observed in their experiments, Richmond and Sullivan[18] discuss changes in lung tissues surrounding the immobile particles with the statement:

> There has been no increase in frank tumors observed within the past year; however, the epithelial changes described above could be considered as precursors of peripheral adenomas.

These observations are interpreted by Tamplin and Cochran[3] in their statements: "These experiments strongly support the proposal that a single particle imbedded in tissue is capable of eliciting a carcinogenic response. The killing of cells and the development of a lesion surrounding the particle is the suggested mechanism of carcinogenesis (an injury mediated mechanism)." They state further:

> Although no tumors appeared in association with the microspheres in the animal experiments, the description of the lesions is suggestive of an incipient tumorogenic response. Richmond, et al., state that they could be considered as precursors of peripheral adenomas and their description is consistent with that of developing bronchiolo-alveolar carcinoma. It is reasonable to propose that the induction period for a frank tumor by this mechanism is longer than the life span of rats and hamsters.

Tumors have been induced during numerous experiments in the rat by plutonium through a variety of exposure means (see, for example, Table III–A, p. 14, reference 9), and in the Syrian hamster[19] high tumor incidences have been observed with short induction times for exposures to particulate and more uniformly distributed Po-210. If it is assumed on the basis of this limited evidence that the period of tumor induction in the hamster does not exceed the life span of the animal and that the estimates of probability for tumor induction by Cochran and Tamplin were correct (i.e., the probability for lesion production approaches unity and the probability of cancer induction per lesion is 1/2000), the number of tumors to be expected in the 1150 hamsters having lived their lives or sacrificed* would be about 2900. In reality, three primary lung tumors were observed in all of the exposed animals. Thus, the relation between lesions and assumed cancer induction as proposed by the NRDC is not supported by this evidence. There are no data on the period of tumor induction by the specific mechanism proposed by Cochran and Tamplin for radioactive particles. However, as stated above, experiments reported in reference 19 demonstrated that exposure to alpha radiation produces pulmonary neoplasms in Syrian hamster lungs with high efficiency and short induction times.

8. *Human Hand Exposure.* In terms of the risk of cancer induction in man from exposure to particulate plutonium, Cochran and Tamplin[1] cite two instances of human hand exposure to plutonium as being potential or actual causes of cancer. The first, a report of Lushbaugh and Langham, describes the results of examinations of a lesion that developed from plutonium imbedded in the palm of a machinist. Lushbaugh and Langham state in their report:[17]

> The autoradiographs showed precise confinement of alpha-tracks to the area of maximum damage and their penetration into the basal areas of the epidermis, where epithelial changes typical of ionizing radiation exposure were present. The cause and effect relationship of these findings, therefore, seemed obvious. Although the lesion was minute, the changes in it were severe. Their similarity to known precancerous epidermal cytologic changes, of course, raised the question of the ultimate fate of such a lesion should it be allowed to exist without surgical intervention * * *

The information contained in this quotation and an estimate of puncture wounds involving plutonium that had occurred at approximately the time of publication of the Lushbaugh and Langham report led Tamplin and Cochran to conclude:[1]

> Therefore, this wound data would suggest that insoluble plutonium particles could offer a risk of cancer induction in man that is even greater than 1/1000 per particle.

This conclusion is not sustained by the information cited. The AEC contacted Dr. Lushbaugh,[20] requesting his views as to whether his report supported the NRDC's conclusions that: (1) a single Pu-239 particle is capable of inducing cancer; and (2) a risk of cancer may be greater than 1/1000 per particle. The entire response of Dr. Lushbaugh to this inquiry dated September 10, 1974, is reproduced below:

> In reference to your letter of August 16, 1974, I should point out that earlier this year I worked with Dr. Bruce Wachholz of Bio-Medical Programs, DBER, Germantown Headquarters, on the initial stages of a document recently numbored WASH–1320; entitled, *A Radiobiological Assessment of the Spatial Distribution of Radiation Dose from Inhaled Plutonium Particles;* and authored by W. Bair, C. Richmand, and B. Wachholz. Although I have not seen this paper in its final form as it is at this moment still being

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[17] Lushbaugh, D. C. and Langham, J., "A Dermal Lesion from Implanted Plutonium," Archives of Dermatology, 86, October, 1962.

*Refer to original document for references given.

[18] Richmond, C. R., and Sullivan, E. M., (eds.), "Annual Report of the Biomedical and Environmental Research Program of the LASL Health Division for 1973," Los Alamos Scientific Laboratory Report LA–5633–PR, May, 1974.

[19] Little, J. G., Grossman, B. N., and O'Toole, W. F., "Factors Influencing The Induction of Lung Cancer in Hamsters by Intratracheal Administration of ^{210}Po," in: Radionuclide Carcinogenesis, (C. L. Sanders, R. H. Busch, J. E. Ballou, and D. D. Mahlum, eds.), CONF–720505:119, AEC Symposium Series No. 29, USAEC, 1973.

[20] Letter from L. Rogers to C. C. Lushbaugh, M.D., dated August 16, 1974.

> printed. I am certain that it contains an attempt to answer the question of whether or not Mrs. Langham's and my article in Archives of Dermatology (1962) supports the contention of Dr. Tamplin and Mr. Cochran that a single particle of Pu-239 is capable of inducing cancer and that the risk of cancer from such a particle is 1 per 1000. We believe that these conclusions cannot be derived *from the histopathologic observations* we reported in this case report nor in the other cases we subsequently published along with it in the Annals of the New York Academy of Science.
>
> In the petition from the Natural Resources Defense Council to which you refer, one can see that the authors apparently do not know the difference between a precancerous cellular change and a cancer. While it is true that the term "precancerous change" contains the implication that a cancer follows it, this is not always the case because precancerous changes are reversible and reparable. In fact when a lesion showing pre*cancerous changes is removed surgically*, the surgeon knows from this diagnostic impression given him by the pathologist that the lesion he removed is not a cancer and that he does not have to worry about it further. My object in using the term "precancerous" to describe the cytologic appearance of some of the epithelial cell nuclei around the plutonium particles in the skin of the case in *Arch. Dermatol.* was to point out that in spite of the amazingly huge dose of alpha radiation over a period longer than 4 years a cancer had *not* developed and that one could at most only call the changes precancerous.
>
> *In reviewing this case in the Annals of the* New York Academy article, we attempted to show that the strictly localized injury caused by the plutonium particles was developing in such a fashion (like a pimple) that the particles would have been shed in time along with a small amount of pus-like material as the pimple "ripened" and drained spontaneously. Dr. Tamplin in his arguments assumes that fibrosarcomas in rat skin are equateable with the minimal changes we described in the skin of this man. Of course, they are not. The statement that it is "clear" on the basis of this one human case that plutonium can cause skin cancer in man is false. If this case and others like it show something of radiobiologic importance, they show only that the development of cancer from plutonium exposures of human tissues must be much more difficult to obtain than cancers in rodent tissues, since no human cancers have ever been seen or reported following plutonium exposure of human beings. Logically, if there is no observed plutonium-induced human cancer case, the one per thousand per particle level of cancer risk for plutonium exposure has no basis in fact and amounts to only a conjecture on the part of the authors of the NRDC petition.

The interpretation of his use of the term "precancerous" provided by Dr. Lushbaugh is shared by Peterson,[21] who cautions with regard to precancerous changes in the alimentary tract:

> such entities have been called "premalignant" or "precancerous" but these terms convey a precursor relationship that is not proved in most cases and is not understood in others. For instance, adenomas of the colon are thought to be "precancerous" but that they actually develop from benign into malignant tumors is unproved; Plummer-Vinson syndrome is known to be followed frequently by carcinoma of the esophagus, but the "precancerous" relationship of this lesion is not understood. Unfortunately, these terms stem from post hoc observations, and their use may be misleading.

On the basis of the foregoing, the association of risk of cancer induction based upon observations of lesions described as "precancerous" is speculative, and such observations should not be used in quantitative estimates of risk.

The second instance of human hand exposure to plutonium cited by Cochran and Tamplin involves the case of a freight handler who "developed an infiltrating soft tissue sarcoma on the left palm which eventually resulted in his death."[1] The AEC contacted Dr. Neil Wald, who was a consulting physician in the case, to obtain his medical opinion as to whether "there is an overwhelming medical probability that his cancer was induced by plutonium"[1] as stated by Cochran and Tamplin. Dr. Wald advised the AEC that he remains in agreement with the data and conclusions drawn in his consultation report concerning the absence of any evidence to support the claim of a relationship between the exposure incident and the subsequent development of neoplastic disease. Dr. Wald's letter and consultation report are on file in the NRC public document room.

9. *Human Inhalation Exposure.* There is limited human experience which is relevant to the "hot particle" question. Perhaps the most relevant case of human exposure to plutonium inhalation as well as the best documented study relating to the "hot particle" issue has been reported by McInroy, et al.[22] Investigators who examined pulmonary lymph nodes of a plutonium worker (case 7–138) killed in an automobile accident, determined the plutonium particle size distribution in the lymph nodes by emulsion track techniques, and estimated the number of plutonium particles associated with size classes contained within the observed particle size distribution.

Cochran and Tamplin[23] have examined these results. Using the parameters for plutonium oxide and the calculational methods for inhalation provided by the ICRP,[24] they estimated that at the time of the worker's death the number of "hot particles" of $^{238}PuO_2$ with activities greater than 0.07 or 0.14 picocuries was 20,000 and 1600 "hot particles" respectively. There was no evidence of cancer in the lungs of the deceased worker. In the event that 20,000 particles were present in the lungs for over one year, and allowing sufficient time for cancer induction (26 years since first exposure), the probability of one or more lung tumors being present at death would be essentially unity using the tumor probabilities proposed by the NRDC. Cochran and Tamplin[23] then suggest that the minimum activity of a "hot particle" should be adjusted upward from 0.07 to 0.14 pCi.

The NRC has also reviewed the work of McInroy, et al.[22] Based on the tabulated number of particles estimated by the authors to exist in the lymph node under discussion, the total number of particles was calculated by the NRC staff to be 306,000.

McInroy, et al., suggest that the deceased worker (Case 7–138) suffered his principal exposure during his first eight years of work (1947–1955). Assuming this, further assuming that he experienced plutonium dioxide inhalation at a uniform rate during this period, and using the parameters and models of the ICRP,[24] the NRC staff calculates that about 52,000 "hot particles" containing 0.07 picocuries or more were present in the lungs at the end of the eight-year exposure period, and about 14,000 remained in his lungs at death. The number of "hot particles" defined to be represented by activities of 0.14 picocuries or more in the lung are calculated to be 9300 at the end of the exposure period and 2500 at death. The residence half-time in the pulmonary region given by the ICRP for plutonium dioxide is 500 days[24]. Under this assumption, in the case of "hot particles" defined to contain 0.07 picocuries or more activity it is estimated that 32,000 particles remained in the lung for more than one year. In the case of "hot particles" defined to contain 0.14 picocuries or more activity the estimate is 5700 particles remaining over one year. Using the tumor probability estimates of Cochran and Tamplin[15], the probabilities for cancer for the two cases would be 99.99999% and 94.2%, respectively. In either case, the NRC finds that evidence provided by the study of this worker provides support for the adequacy of present standards but no support for the corollary as advanced by the NRDC.

Additional studies of relevance to human exposure to plutonium have been reported by Hempelmann and co-workers[25, 26] and reviewed by Blair, et al.,[9] and by Healy, et al.[11] The studies summarize the results of 27 years of observations of 24 individuals exposed to plutonium in several chemical forms during Manhattan Project operations. No lung cancers had been observed in these persons through the latest examinations reported. It has been estimated[9, 11] that the

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[21] Peterson, M. L., "Neoplastic Diseases of the Alimentary Tract," in *Textbook of Medicine*, eleventh edition, Cecil-Loeb Publishers, Philadelphia and London, 1963.

[22] McInroy, J. F., Stewart, M. W., and Moss, W. D., "Studies of Plutonium in Human Tracheobronchial Lymph Nodes," LA-UR-74-1454 (Preprint), Los Alamos Scientific Laboratory, undated.

[23] Letter from T. B. Cochran and A. R. Tamplin to R. B. Minogue, Nuclear Regulatory Commission, dated February 4, 1975.

[24] "The Metabolism of Compounds of Plutonium and Other Actinides," ICRP Publication 19, International Commission on Radiological Protection, Pergamon Press, adopted May, 1972.

[25] Hempelmann, L. H., Richmond, C. R., and Voelz, G. L., "A Twenty-Seven Year Study of Selected Los Alamos Plutonium Workers," LA-5148-MS, Los Alamos Scientific Laboratory, January, 1973.

[26] Hempelmann, L. H., Langham, W. H., Richmond, C. R. and Voelz, G. L., "Manhattan Project Plutonium Workers: A Twenty-Seven Year Follow-Up Study of Selected Cases," Health Physics, 25, November, 1973.

total initial plutonium burden was about 10 microcuries summed over the lungs of all these men. The burden of "hot particles" (plutonium activity per particle of 0.07 picocuries or greater) was estimated at 4.0×10^{5} particles per man. Based upon the lung cancer probability estimates of Cochran and Tamplin,[1] approximately 5000 lung tumors should have been observed in these men. Under the later assumption of Cochran and Tamplin[23] that "hot particles" must exhibit an activity of 0.14 picocuries or greater, the estimated minimum number of "hot particles" in the lungs of each of the Manhattan workers is 8.4×10^{4}. The expected number of lung tumors, based upon the NRDC cancer induction estimates, would then be approximately 1100. As noted earlier, no lung cancers have been observed in these men and the NRC considers this human experience as supporting evidence that its present standards for insoluble plutonium have a radiobiologically sound basis.

The NRDC has examined[2] the Manhattan Project worker data from a somewhat different view than Healy, et al.[11] The NRDC assumes that the distribution of plutonium particles in the lungs of the Manhattan Project workers may be inferred to be the same as that reported by McInroy, et al.[26] The particle size classes that NRDC provides (Table I, Reference 2) do not strictly conform to the distribution reported by McInroy, et al. However, using this distribution and the tumor probability estimate of Cochran and Tamplin, the NRC staff has estimated that the number of lung tumors would exceed 2800 for "hot particles" defined as containing 0.07 picocuries of plutonium activity or more, and would exceed 250 for "hot particles" defined as containing 0.14 picocuries of plutonium activity or more.[2] These values may be compared with the observation that no lung tumors have been observed in the 24 Manhattan Project workers.

A study has been conducted[27] to evaluate lung burdens of plutonium dioxide in persons exposed at an AEC contractor facility, Dow Chemical, Rocky Flats Division, in 1965. The NRDC observes[2] that while no lung cancers have appeared in the 25 persons exposed, the time required for the induction of cancer might exceed 10 years. Thus there do not yet appear to be any definitive conclusions that can be drawn from the Rocky Flats results from the standpoint of their providing support or refutation of the corollary of the NRDC hypothesis.

D. CONCLUSION

The Nuclear Regulatory Commission denies the petition of the Natural Resources Defense Council to establish specific health protection standards for "hot particles." The denial is based on the NRC's finding that its present standards for long-lived, alpha-emitting radionuclides in insoluable form are, with respect to the spatial distribution of dose, radiobiologically sound and that the NRDC corollary to the hypothesis describing an injury-mediated mechanism of carcinogenic response to alpha-emitting particles is speculative and not supported by the body of scientific data and knowledge on this subject. Consequently, the NRDC position does not provide a sufficient scientific basis for changing or supplementing existing radiation protection standards.

In Section C above, the NRC has outlined its examination of the carcinogenic response mechanism which was hypothesized by the NRDC from a plausibility argument which the NRDC based on selected portions of the considerable body of knowledge on this subject. That is, the corollary to the hypothesis is shown in this analysis to be based on a pattern of arbitrary interpretations of selective portions of the available information.

The tests which the NRC has applied in evaluating the NRDC petition are: (a) whether existing radiobiological evidence indicates that present standards in question should be modified as requested; (b) whether the corollary to the NRDC hypothesis is supported by the body of relevant knowledge; and (c) whether the corollary is a valid interpretation of the supporting data cited by its authors. The NRC finds that the corollary fails to satisfy any of these tests.

The NRDC has stated[15] that given two hypotheses—(1) the Cochran and Taplin model, (2) and the uniform dose model—the responsible regulatory agency must make the prudent choice and select the more conservative of the two as the basis for radiological protection standards. The NRC agrees in principle, if the two hypotheses are generally supported by the body of knowledge. That is not the case in this instance.

The uniform dose model is examined in Section E, below. The NRC concludes on the basis of its examination of the body of knowledge that the uniform model remains an acceptable basis for radiological protection standards for insoluble plutonium.

E. DISCUSSION OF EXISTING STANDARDS

The preceding discussion has dealt specifically with the question of special standards for protection against the inhalation of insoluble, alpha-emitting particles of specified physical characteristics, which may be capable of forming lesions in the lung which may in turn induce cancer. In this section the question of the adequacy of existing NRC standards for protection against all insoluble, alpha-emitting particles is considered, irrespective of the mechanisms for adverse biological effects.

1. *Present NRC Standards.* The present NRC standards for protection against insoluble, alpha-emitting radionuclides are given in 10 CFR Part 20. For plutonium-239 these standards specify that no occupationally exposed individual may be exposed to concentrations exceeding 4×10^{-11} microcuries per milliliter of air, averaged over a 40-hour week. Under equilibrium conditions, this level of exposure will deliver about 15 rems per year averaged over the entire lung mass. The 15 rems per year limit has been recommended by the ICRP, the NCRP, and the FRC. The regulations further specify that insoluble plutonium-239 in effluents to unrestricted areas cannot exceed 1×10^{-12} microcuries per milliliter of air when averaged over one year. This level of exposure could deliver about 1.5 rem per year to the lung, i.e., the limit recommended by the ICRP. Similar standards are given for other insoluble, alpha-emitting radionuclides. In the development of these standards it was assumed that the dose is uniformly distributed throughout the entire lung mass; thus uniform and nonuniform dose are treated in the same manner.

2. *Position Taken By Other Organizations.* Organizations such as the ICRP, NCRP, FRC, NAS, National Radiological Protection Board (UK), the Biophysical Society, and the Medical Research Council (UK) have considered the question of whether nonuniform dose is more hazardous than uniform dose. Their conclusions are that the uniform dose assumption is adequately conservative. Below are statements to this effect from these organizations:

> The general opinion which emerged from the discussion was that the carcinogenic effect per unit volume is probably considerably less for the irradiation of small masses of tissue than for large.[28]
>
> On the basis of general considerations and of some experimental data and clinical experience the Task Group were of the opinion that, for late effects, the same radiation energy absorption might well be less effective when distributed as a series of hot spots, than when uniformly distributed. Thus, with particulate radioactive sources within a tissue, a mean tissue dose would probably introduce a factor of safety. * * *[29]
>
> It is therefore concluded that the current NCRP practice of averaging over the lung the absorbed dose from particulate alpha-emitting radionuclides is a defensible procedure when employed in conjunction with appropriate dose limits.[30]
>
> * * * it may be inferred that a higher localized dose from alpha particles was not more cancerogenic than the same mean tissue dose delivered more uniformly to critical cells.[7]

It is noted that the basis of ICRP recommendations is the average radiation dose to an organ and not the number of radioactive particles in the organ. This dosimetric basis of radiological protection has been established for many years by observations of humans and experimental work with animals. A better evaluation than that offered by Cochran and Tamplin would be needed for

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[27] Mann, J. R. and Kirchner, A. R., "Evaluation of Lung Burden Following Acute Inhalations of Highly Insoluble PuO_2," Health Physics, 13, 1967.

[28] McMurtrie, G. E. (Secretary), "Permissible Doses Conference held at Chalk River, Ontario (Sept. 1949)," Report RH-10, May, 1950.

[29] "Radiosensitivity and Spatial Distribution of Doses, Reports Prepared by Two Task Groups of Committee 1, of the International Commission of Radiological Protection," ICRP Publication 14, Pergamon Press, Oxford, 1969.

[30] "Alpha-Emitting Particles in the Lung" NCRP Report No. 46, Washington, D.C., 1975.

this system to be set aside in favor of the hot particle concept.[31]

> The use of the data of Albert et al. on rat skin tumors induced by fast electrons to estimate the risk from hot particles seems unjustified on four grounds. (i) The rat data involved a single dose, whereas the lung irradiation being considered is chronic. (ii) Cochran and Tamplin do not cite data showing that nonuniform irradiation by beta and alpha particles is less effective than uniform radiation. (iii) Previous experiments cited by the Albert group showed no tumor production by 0.3 MeV electrons, external alpha particles and protons. (vi) The hair follicle seems to be the sensitive structure for radiation induced cancer in the skin. No similar structure has been identified in the lung, nor is there any estimate of the probability of a hot particle being close to such a structure.[32]
>
> *In summary, therefore, there is at present* no evidence to suggest that irradiation of the lung by particles of plutonium is likely to be markedly more carcinogenic than when the same activity is uniformly distributed.[33]

The organizations which have recommended the use of the uniform dose assumption have reviewed considerable data in their decision-making process. The studies considered most important by the NRC staff are discussed below as they relate to the uniform dose assumption.

3. *Hamster Experiments.* Richmond and Sullivan[18] and Richmond and Voelz[34] have reported the partially completed results of experiments, previously discussed, in which large numbers of plutonium particles were implanted in the lungs of hamsters by intravenous injections. According to the summary of these experiments reported by Bair, et al.,[9] approximately 560 hamsters each received 2000 particles (0.07 to 59.4 picocuries), 485 received 6000 to 1,000,000 particles, and a large number of additional hamsters received 50,000 to 900,000 particles (some containing as little as 0.015 picocuries). About 2000 animals were involved in these experiments. Bair, et al.,[9] report that 1150 of these animals have lived their full life spans or have been sacrificed, with only three primary lung tumors observed. These results indicate a very low risk for nonuniform lung dose due to plutonium in particulate form.

Little, et al.,[19] exposed hamster lungs to alpha radiation from Po-210. The size of the particles was varied, thus resulting in a range in the degree of dose uniformity. The incidence of lung cancer was lower for the less uniform dose. The authors concluded:

> * * * in the dose range studied, alpha radiation is cancerogenic when a lower but relatively uniform dose is delivered to a large volume of lung tissue than when a similar amount of radioactivity is distributed nonuniformly such that the primary effect is to deliver much higher radiation doses to relatively small tissue volumes.

These results indicate that plutonium standards based on uniform dose distribution would be conservative for particles in the lung. This experiment also revealed that hamster lungs develop cancer in a relatively short period, as compared with their life span, following alpha irradiation. This information lends support to the usefulness of the data reported in references 18 and 34.

4. *Special Study on the Spatial Distribution of Lung Dose (Bair, et al., Reference 9).* As previously mentioned, a study was recently conducted relative to the question as to whether, for a given quantity of radioactive material in the lung, the risk of cancer is more properly characterized by assuming that the material is concentrated nonuniformly in discrete particles or by assuming that the material is distributed uniformly throughout the lung. This study was conducted by personnel most closely associated with pertinent research programs. The results were published in reference 9. Two of the conclusions from reference 9 are reproduced below:

> Available experimental data indicate that averaging the absorbed alpha radiation dose from plutonium particles in lung is radiobiologically sound.
>
> After thirty years experience with plutonium in laboratory and production facilities, there is no evidence that the mean dose lung model on which occupational radiation protection standards for plutonium are based is grossly in error or leads to hazardous practices. Currently available data from occupationally exposed persons indicate that the nonhomogeneous dose distribution from inhaled plutonium does not result in demonstrably greater risk than that assumed for a uniform dose distribution. Thus, empirical considerations lead to the conclusion that the nonuniform dose distribution of plutonium particles in the lung is not more hazardous and may be less hazardous than if the plutonium were uniformly distributed and that the mean dose lung model is a radiobiological sound basis for establishment of plutonium standards.

5. *NCRP Report No. 46.* The NCRP report on this subject, quoted above, concludes that the dose-averaging procedure that was used to derive current standards is defensible. This conclusion is based on observations in experimental animals, on observations in man, and on a theoretical analysis showing that the number of cells at risk is much greater per unit quantity of activity when the activity is distributed uniformly in the lung.

6. *Human Experience.* Hempelmann, et al.,[35, 36] discuss several workers who were exposed to insoluble plutonium particles about 30 years ago during the Manhattan Project. It is estimated that they were exposed to levels of plutonium considerably exceeding the present NRC standards. Several of these persons still retain body burdens in excess of the presently permissible level. None of the workers have suffered any illness attributable to the exposures, which can be taken to indicate a low risk associated with the levels of exposure permitted by the NRC standards.

7. *Summary.* In summary, the uniform dose model is generally recognized by the scientific community and supported by experimental evidence as a conservative basis for standards for personnel protection. The NRC finds, in agreement with the recommendations of the organizations quoted, that available data support the use of the uniform dose assumption as an appropriately conservative approach. That is, the available data indicate that while the biological risk from a uniform lung dose of 15 rems per year is low, an equivalent dose delivered in a nonuniform manner is at least as low. *Therefore, standards for insoluble,* alpha-emitting radionuclides, as based on a uniform dose assumption, are believed to be adequately conservative.

F. FURTHER CONSIDERATIONS

The NRC conclusions cited in Section D do not obviate the need for continuing review of developments in the field. The Commission will reconsider its determination if warranted by any of several considerations. These may include new guidance to Federal agencies from EPA, new recommendations from the NCRP, ICRP or NAS, or new data from observations of exposed personnel or from the results of ongoing or future animal experiments. The Commission will continue to follow closely any new information that becomes available, and consideration will be given to the modification of standards as necessary to reflect advances in radiobiological knowledge.

Extensive studies on inhalation hazards are being continued by the Energy Research and Development Agency. The most relevant ongoing studies in this program are discussed below.

1. *Pacific Northwest Laboratory.* Pacific Northwest Laboratory is conducting polydisperse aerosol studies with transuranium radionuclides in dogs and rodents which range from short-term experiments to determine the kinetics and dosimetry aspects to long-term (lifetime) experiments to help define the risks associated with inhalation of radionuclides.

Of major interest are the long-term beagle experiments in which animals are given an exposure to polydisperse plutonium oxide aerosols at various levels, from levels that overlap previous beagle experiments down to lower levels which provide an initial overall average lung dose of 15 rems per year. Experiments are being performed with both ^{239}Pu and ^{238}Pu to define quantitative differences between the two plutonium isotopes. The ^{239}Pu exposures were administered in 1971 and 1972 and the ^{238}Pu exposures were administered in 1973 and 1974. There are 130 animals including 20 controls in the ^{239}Pu studies and the same number in the ^{238}Pu studies.

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[31] Dolphin, G. W., "Hot Particles," National Radiological Protection Board, Harwell, UK, 1974.

[32] Science and Technology Advice and Information Service Committee, "Report on Radiation Protection Standards for Hot Particles of Plutonium and Other Antinides," Biophysical Society, draft dated November, 1974.

[33] Committee on Protection Against Ionising Radiation, "The Toxicity of Plutonium," Medical Research Council, 1975.

[34] Richmond, C. R. and Voelz, G. L., (eds.), "Annual Report of the Biological and Medical Research Group (H-4) to the USAEC," Division of Biology and Medicine, Los Alamos Scientific Laboratory Report, LA-4923-PR, 1972.

2. *Lovelace Foundation.* Lovelace Foundation is conducting monodisperse aerosol studies with transuranium radionuclides in hamsters and dogs and preliminary studies with shetland ponies. Experiments are designed to determine deposition, retention, mobility, dosimetry and correlation to pathological observations of various physical and chemical forms of monodisperse particulates in animal lungs.

Of particular interest to the "hot particle" question are beagle experiments with monodisperse $^{238}PuO_2$ particles similar to the PNL studies of beagles with polydisperse particles. The exposures initiated in 1973 are scheduled to be completed in 1975.

Experiments are also being conducted with other alpha and beta emitting radionuclides in various chemical and physical forms.

3. *Los Alamos Scientific Laboratory.* Los Alamos Scientific Laboratory is continuing studies in which particulate materials are transported by the circulatory system and lodged in hamster lungs following intravenous injection. Results to date [18] after three years of exposure indicate minimal to no effects. Experiments are being extended to larger numbers of particulates in an attempt to provide some experimental overlap with the results of Little [19] who obtained lung tumor incidence in hamsters after polonium exposures. A collaborative program involving experiments by Little with plutonium particles and Los Alamos experiments with polonium is being initiated.

4. *Human Exposures.* Studies [22] of tissues of the Los Alamos worker whose lymph nodes contained particulate plutonium are being extended to include other portions of the lung.

A number of personnel have been exposed to insoluble particles of plutonium and other transuranic elements in connection with the operation of the AEC's national laboratories. The results of medical examinations for these personnel are considered by the NRC to be the best possible source of direct information regarding the adequacy of its standards for the protection of personnel against such particles. These results are being closely followed.

Copies of the petitions for rulemaking and of the Commission's letters of denial are available for public inspection in the Commission's Public Document Room at 1717 H Street NW., Washington, D.C.

Dated at Washington, D.C. this 7th day of April 1976.

For the Nuclear Regulatory Commission.

SAMUEL J. CHILK,
Secretary of the Commission.

[FR Doc.76-10523 Filed 4-9-76;8:45 am]

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CHAPTER IV

Section J

APPENDIX E

SUPPLEMENTAL DATA

Table IV J(E)-1

TOTAL BODY PERSON-REM COMMITMENT FROM USA INDUSTRY OVER 1975-2000 PERIOD

	FUEL RECYCLE OPTION					
	None		U Only		Pu + U	
Fuel Cycle Component	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	2.97×10^6	1.21×10^6	2.65×10^6	1.08×10^6	2.30×10^6	9.36×10^5
Milling	5.79×10^5	5.62×10^5	5.18×10^5	5.02×10^5	4.49×10^5	4.36×10^5
UF_6 Conversion	4.20×10^4	4.48×10^3	3.87×10^4	4.00×10^3	3.42×10^4	3.45×10^3
Enrichment	7.63×10^1	3.40×10^3	1.30×10^2	3.50×10^3	1.11×10^2	2.70×10^3
UO_2 Fuel Fabrication	2.51×10^3	5.05×10^4	2.57×10^3	5.05×10^4	2.23×10^3	4.37×10^4
MOX Fuel Fabrication	0	0	0	0	2.96×10^2	2.46×10^4
LWR Power Generation	3.06×10^5	2.27×10^6	3.06×10^5	2.27×10^6	3.10×10^5	2.27×10^6
Irradiated Fuel Storage	2.83×10^1	1.12×10^4	1.41×10^1	4.00×10^3	8.46	3.35×10^3
Fuel Reprocessing	0	0	1.08×10^6	7.20×10^4	1.07×10^6	7.74×10^4
Transportation	1.50×10^3	5.40×10^3	1.60×10^3	5.60×10^3	2.00×10^3	8.10×10^3
Waste Management	5.90	6.0×10^3	4.50	5.90×10^3	5.20	5.20×10^3
Industry Total (U.S.)	3.90×10^6	4.11×10^6	4.60×10^6	3.99×10^6	4.17×10^6	3.81×10^6
Option Total	8.01×10^6		8.59×10^6		7.98×10^5	
Foreign Additional from U.S.A.	2.1×10^5		9.12×10^5		8.90×10^5	

Table IV J(E)-2

G.I. TRACT PERSON-REM COMMITMENT FROM USA INDUSTRY OVER 1975-2000 PERIOD

	FUEL RECYCLE OPTION					
	None		U Only		Pu + U	
Fuel Cycle Component	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	1.16×10^5	1.21×10^6	1.04×10^5	1.08×10^6	9.10×10^4	9.36×10^5
Milling	2.24×10^4	2.10×10^5	2.00×10^4	1.88×10^5	1.74×10^4	4.40×10^5
UF_6 Conversion	5.61×10^3	2.73×10^3	5.01×10^3	2.44×10^3	4.30×10^3	3.30×10^3
Enrichment	6.65×10^3	1.70×10^3	7.35×10^3	1.80×10^3	6.20×10^3	2.70×10^3
UO_2 Fuel Fabrication	2.72×10^3	4.98×10^4	2.79×10^3	4.98×10^4	2.41×10^3	4.40×10^4
MOX Fuel Fabrication	0	0	0	0	1.78×10^1	2.46×10^4
LWR Power Generation	3.05×10^5	2.27×10^6	3.05×10^5	2.27×10^6	3.09×10^5	2.27×10^6
Irradiated Fuel Storage	2.83×10^1	1.12×10^4	1.41×10^1	5.59×10^3	8.46	3.35×10^3
Fuel Reprocessing	0	0	1.59×10^6	7.20×10^4	1.62×10^6	7.80×10^4
Transportation	1.50×10^3	5.40×10^3	1.60×10^3	5.40×10^3	2.00×10^3	8.10×10^3
Waste Management	7.90	6.0×10^3	7.20	5.90×10^3	7.80	5.70×10^3
Industry Total (U.S.)	4.59×10^5	3.76×10^6	2.03×10^6	3.67×10^6	2.05×10^6	3.60×10^6
Option Total	4.22×10^6		5.70×10^6		5.65×10^6	
Foreign Additional from USA	2.10×10^5		9.12×10^5		8.90×10^5	

Table IV J(E)-3

BONE PERSON-REM COMMITMENT FROM USA INDUSTRY OVER 1975-2000 PERIOD

	FUEL RECYCLE OPTION					
	None		U Only		Pu + U	
Fuel Cycle Component	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	9.69×10^6	1.74×10^6	8.67×10^6	1.55×10^6	7.52×10^6	1.35×10^6
Milling	1.87×10^6	2.28×10^6	1.68×10^6	2.04×10^6	1.45×10^6	1.77×10^6
UF_6 Conversion	1.03×10^5	5.00×10^4	9.16×10^4	4.70×10^4	7.70×10^4	3.80×10^4
Enrichment	9.20×10^2	3.10×10^4	9.40×10^2	3.20×10^4	8.02×10^2	2.40×10^4
UO_2 Fuel Fabrication	4.06×10^4	6.61×10^4	4.17×10^4	6.61×10^4	3.60×10^4	5.72×10^4
MOX Fuel Fabrication	0	0	0	0	1.40×10^4	2.46×10^4
LWR Power Generation	1.10×10^6	2.27×10^6	1.10×10^6	2.27×10^6	1.10×10^6	2.27×10^6
Irradiated Fuel Storage	2.83×10^1	1.12×10^4	1.41×10^1	5.59×10^3	8.46	3.35×10^3
Fuel Reprocessing	0	0	2.65×10^6	7.20×10^4	2.61×10^6	7.80×10^4
Transportation	1.50×10^3	5.40×10^3	1.60×10^3	5.40×10^3	2.00×10^3	8.10×10^3
Waste Management	9.00×10^1	6.00×10^3	4.30×10^1	5.90×10^3	5.10×10^1	5.70×10^3
Industry Total (U.S.)	1.28×10^7	6.45×10^6	1.42×10^7	6.08×10^6	1.28×10^7	5.63×10^6
Option Total	1.93×10^7		2.03×10^7		1.84×10^7	
Foreign Additional from USA	1.06×10^6		3.31×10^6		3.20×10^6	

Table IV J(E)-4

LIVER PERSON-REM COMMITMENT FROM USA INDUSTRY OVER 1975-2000 PERIOD

Fuel Cycle Component	FUEL RECYCLE OPTION					
	None		U Only		Pu + U	
	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	2.40×10^6	1.21×10^6	2.15×10^6	1.08×10^6	1.86×10^6	9.36×10^5
Milling	4.49×10^5	2.10×10^5	4.02×10^5	1.87×10^5	3.49×10^5	1.63×10^5
UF_6 Conversion	7.20×10^2	5.30×10^3	6.24×10^2	4.90×10^3	5.50×10^2	4.21×10^3
Enrichment	2.00×10^1	3.40×10^3	2.19×10^2	3.50×10^3	1.87×10^2	2.70×10^3
UO_2 Fuel Fabrication	5.19	4.95×10^4	5.21	5.00×10^4	4.51	4.29×10^4
MOX Fuel Fabrication	0	0	0	0	1.37×10^3	2.46×10^4
LWR Power Generation	3.10×10^5	2.27×10^6	3.10×10^5	2.27×10^6	3.15×10^5	2.27×10^6
Irradiated Fuel Storage	2.83×10^1	1.12×10^4	1.41×10^1	5.59×10^3	8.46	3.36×10^3
Fuel Reprocessing	0	0	1.09×10^6	7.20×10^4	1.08×10^6	7.80×10^4
Transportation	1.50×10^3	5.40×10^3	1.60×10^3	5.40×10^3	2.00×10^3	8.10×10^3
Waste Management	1.20×10^1	6.00×10^3	5.80	5.90×10^3	9.80	5.70×10^3
Industry Total (U.S.)	3.34×10^6	3.76×10^6	4.13×10^6	3.67×10^6	3.61×10^6	3.54×10^6
Option Total	7.10×10^6		7.80×10^6		7.15×10^6	
Foreign Additional from USA	2.10×10^5		9.12×10^5		8.90×10^5	

Table IV J(E)-5

KIDNEY PERSON-REM COMMITMENT FROM USA INDUSTRY OVER 1975-2000 PERIOD

	FUEL RECYCLE OPTION					
	None		U Only		Pu + U	
Fuel Cycle Component	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	1.13×10^7	1.74×10^6	1.01×10^7	1.55×10^6	8.75×10^6	1.35×10^6
Milling	2.13×10^6	2.61×10^5	1.91×10^6	2.34×10^5	1.66×10^6	2.03×10^5
UF_6 Conversion	1.19×10^4	1.34×10^4	1.07×10^4	1.19×10^4	9.10×10^3	9.80×10^3
Enrichment	5.22×10^2	8.50×10^3	3.00×10^3	8.50×10^3	2.56×10^3	6.60×10^3
UO_2 Fuel Fabrication	6.55×10^3	5.35×10^4	6.77×10^3	5.35×10^4	5.86×10^3	4.63×10^4
MOX Fuel Fabrication	0	0	0	0	1.23×10^3	2.46×10^4
LWR Power Generation	3.03×10^5	2.27×10^6	3.03×10^5	2.27×10^6	3.07×10^5	2.27×10^6
Irradiated Fuel Storage	2.83×10^1	1.12×10^4	1.41×10^1	5.59×10^3	8.46	3.35×10^3
Fuel Reprocessing	0	0	1.10×10^6	7.20×10^4	1.09×10^6	7.80×10^4
Transportation	1.50×10^3	5.40×10^3	1.60×10^3	5.40×10^3	2.00×10^3	8.10×10^3
Waste Management	9.40×10^1	6.00×10^3	4.40	5.90×10^3	7.40×10^1	5.70×10^3
Industry Total (U.S.)	1.38×10^7	4.36×10^6	1.34×10^7	4.21×10^6	1.18×10^7	4.01×10^6
Option Total	1.82×10^7		1.76×10^7		1.59×10^7	
Foreign Additional from USA	2.1×10^5		9.12×10^5		8.90×10^5	

Table IV J(E)-6

THYROID PERSON-REM COMMITMENT FROM USA INDUSTRY OVER 1975-2000 PERIOD

	FUEL RECYCLE OPTION					
	None		U Only		Pu + U	
Fuel Cycle Component	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	7.30×10^3	1.21×10^6	6.53×10^3	1.08×10^6	5.67×10^3	9.36×10^5
Milling	1.54×10^3	2.10×10^5	1.38×10^3	1.87×10^5	1.19×10^3	1.63×10^5
UF_6 Conversion	4.84×10^1	4.40×10^3	4.32×10^1	4.00×10^3	3.70×10^1	3.30×10^3
Enrichment	1.80×10^1	3.40×10^3	1.73×10^1	3.50×10^3	1.48×10^1	2.70×10^3
UO_2 Fuel Fabrication	4.64	4.95×10^4	4.66	4.95×10^4	4.04	4.29×10^4
MOX Fuel Fabrication	0	0	0	0	5.59	2.46×10^4
LWR Power Generation	4.80×10^5	2.27×10^6	4.80×10^5	2.27×10^6	4.93×10^5	2.27×10^6
Irradiated Fuel Storage	2.83×10^1	1.12×10^4	1.41×10^1	5.59×10^3	8.46	3.35×10^3
Fuel Reprocessing	0	0	1.93×10^6	7.20×10^4	1.93×10^6	7.80×10^4
Transportation	1.50×10^3	5.4×10^3	1.60×10^3	5.40×10^3	2.00×10^3	8.10×10^3
Waste Management	3.20	6.0×10^3	2.90	5.90×10^3	3.00	5.70×10^3
Industry Total (U.S.)	4.90×10^5	3.76×10^6	2.42×10^6	3.67×10^6	2.43×10^6	3.54×10^6
Option Total	4.25×10^6		6.09×10^6		5.97×10^6	
Foreign Additional from USA	2.10×10^5		9.12×10^5		8.90×10^5	

Table IV J(E)-7

LUNG PERSON-REM COMMITMENT FROM USA INDUSTRY OVER 1975-2000 PERIOD

	FUEL RECYCLE OPTION					
	None		U Only		Pu + U	
Fuel Cycle Component	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	9.07×10^5	6.49×10^6	8.12×10^5	5.80×10^6	7.10×10^5	5.10×10^6
Milling	1.75×10^5	4.84×10^6	1.57×10^5	4.34×10^6	1.36×10^5	3.76×10^6
UF_6 Conversion	9.70×10^2	3.47×10^4	8.77×10^2	3.09×10^4	7.30×10^2	2.63×10^4
Enrichment	4.85×10^2	7.50×10^4	4.40×10^2	7.50×10^4	3.70×10^2	5.80×10^4
UO_2 Fuel Fabrication	1.41×10^2	1.95×10^6	1.46×10^2	1.96×10^6	1.26×10^2	1.70×10^6
MOX Fuel Fabrication	0	0	0	0	3.11×10^2	2.86×10^4
LWR Power Generation	3.03×10^5	2.27×10^6	3.03×10^5	2.27×10^6	3.07×10^5	2.27×10^6
Irradiated Fuel Storage	6.12×10^1	1.12×10^4	3.06×10^1	5.59×10^3	1.83×10^1	3.36×10^3
Fuel Reprocessing	0	0	1.16×10^6	7.20×10^4	1.14×10^6	7.80×10^4
Transportation	1.50×10^3	5.40×10^3	1.60×10^3	5.40×10^3	2.00×10^3	8.10×10^3
Waste Mangement	4.90	6.00×10^3	3.30	5.90×10^3	3.90	5.70×10^3
Industry Total (U.S.)	1.39×10^6	1.57×10^7	2.43×10^6	1.45×10^7	2.29×10^6	1.30×10^7
Option Total	1.71×10^7		1.69×10^7		1.53×10^7	
Foreign Additional from USA	2.10×10^5		1.26×10^6		1.22×10^6	

Table IV J(E)-8

SKIN PERSON-REM COMMITMENT FROM USA INDUSTRY OVER 1975-2000 PERIOD

	FUEL RECYCLE OPTION					
	None		U Only		Pu + U	
Fuel Cycle Component	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	7.30×10^3	1.21×10^6	6.53×10^3	1.08×10^6	5.67×10^3	9.36×10^5
Milling	1.54×10^3	2.10×10^3	1.38×10^3	1.87×10^5	1.19×10^3	1.63×10^5
UF_6 Conversion	3.76×10^1	1.28×10^4	3.36×10^1	1.15×10^4	2.88×10^1	9.90×10^3
Enrichment	1.83×10^1	1.13×10^4	1.18×10^2	1.14×10^4	1.00×10^2	9.00×10^3
UO_2 Fuel Fabrication	4.63	4.95×10^4	4.65	4.95×10^4	4.03	4.29×10^4
MOX Fuel Fabrication	0	0	0	0	5.59	2.46×10^4
LWR Power Generation	3.13×10^5	2.27×10^6	3.13×10^5	2.27×10^6	3.14×10^5	2.27×10^6
Irradiated Fuel Storage	2.39×10^3	1.12×10^4	1.20×10^3	5.59×10^3	7.17×10^2	3.35×10^3
Fuel Reprocessing	0	0	6.62×10^6	7.20×10^4	6.38×10^6	7.80×10^4
Transportation	1.50×10^3	5.40×10^3	1.60×10^3	5.40×10^3	2.00×10^3	8.10×10^3
Waste Management	3.20	6.00×10^3	2.90	5.90×10^3	3.00	5.70×10^3
Industry Total (U.S.)	3.25×10^5	3.78×10^6	6.94×10^6	3.69×10^6	6.70×10^6	3.61×10^6
Option Total	4.11×10^6		1.06×10^7		1.03×10^7	
Foreign Additional from USA	2.60×10^5		2.62×10^7		2.50×10^7	

Table IV J(E)-9

TOTAL BODY PERSON-REM COMMITMENT PER GWy(e)

Fuel Cycle Component	FUEL RECYCLE OPTION					
	No Recycle		Recycle U Only		Pu + U Recycle	
	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	628.	256.	570.	232.	507.	206.
Milling	122.	119.	111.	108.	99.0	96.1
UF_6 Conversion	9.51	0.984	8.64	0.893	7.82	9.789
Enrichment	0.0166	0.695	0.0282	0.695	0.0247	0.713
UO_2 Fuel Fabrication	0.578	11.6	0.600	11.8	0.526	10.3
MOX Fuel Fabrication	0	0	0	0	0.0584	4.85
LWR Power Generation	75.8	562.	75.8	562.	76.8	562.
Irradiated Fuel Storage	0.0092	3.65	0.0044	1.73	0.0026	1.02
Fuel Reprocessing	0	0	366.	24.4	354.	25.5
Transportation	0.111	0.359	0.117	0.365	0.160	0.836
Waste Management	0.00222	0.353	0.00177	0.337	0.00205	0.324
Industry Total (U.S.)	836.	955.	1,130.	942.	1,045.	933.
Option Total	1,790.		2,070.		1,980.	
Foreign Additional from USA	52.		290.		272.	

Table IV J(E)-10

G.I. TRACT PERSON-REM COMMITMENT PER GWy(e)

	FUEL RECYCLE OPTION					
	No Recycle		Recycle U Only		Pu + U Recycle	
Fuel Cycle Component	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	24.5	256.	22.4	232	19.9	206.
Milling	4.73	44.4	4.30	40.4	3.84	35.9
UF_6 Conversion	1.23	0.600	1.12	0.545	0.967	0.478
Enrichment	1.44	0.413	1.60	0.391	1.40	0.312
UO_2 Fuel Fabrication	0.626	11.5	0.652	11.6	0.569	10.2
MOX Fuel Fabrication	0	0	0	0		4.85
LWR Power Generation	75.6	562.	75.6	562.	76.6	562.
Irradiated Fuel Storage	0.00921	3.65	0.00437	1.73	0.00256	1.02
Fuel Reprocessing	0	0	538.	24.4	536.	25.5
Transportation	0.111	0.359	0.117	0.365	0.167	0.874
Waste Management	0.00296	0.353	0.00279	0.337	0.00311	0.335
Industry Total (U.S.)	108.	879.	644.	874.	639.	872.
Option Total	987.		1,520.		1,510.	
Foreign Additional from USA	52.0		290.		272.	

Table IV J(E)-11

BONE PERSON-REM COMMITMENT PER GWy(e)

	FUEL RECYCLE OPTION					
	No Recycle		Recycle U Only		Pu + U Recycle	
Fuel Cycle Component	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	2,050.	368.	1,860.	333.	1,660	298.
Milling	395.	482.	361.	439.	320.	390.
UF_6 Conversion	22.6	11.4	20.5	10.4	18.0	9.35
Enrichment	0.190	5.78	0.206	5.86	0.179	4.86
UO_2 Fuel Fabrication	9.34	15.2	9.75	15.5	8.50	13.5
MOX Fuel Fabrication	0	0	0	0	2.64	4.85
LWR Power Generation	272.	562.	272.	562.	272.	562.
Irradiated Fuel Storage	0.0092	3.65	0.0044	1.73	0.0026	1.02
Fuel Reprocessing	0	0	897.	24.4	864.	25.5
Transportation	0.111	0.359	0.117	0.365	0.167	0.874
Waste Management	0.00334	0.353	0.00374	0.337	0.0202	0.335
Industry Total (U.S.)	2,750.	1,450.	3,420.	1,390.	3,150.	1,340.
Option Total	4,200.		4,810.		4,490.	
Foreign Additional from USA	263.		1,025.		971.	

Table IV J(E)-12

LIVER PERSON-REM COMMITMENT PER GWy(e)

	FUEL RECYCLE OPTION					
	No Recycle		Recycle U Only		Pu + U Recycle	
Fuel Cycle Component	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	507.	256.	462.	232.	410.	206.
Milling	94.9	44.4	86.4	40.2	76.9	35.9
UF_6 Conversion	0.153	1.19	0.139	1.08	0.115	0.963
Enrichment	0.00448	0.695	0.0475	0.695	0.0416	0.557
UO_2 Fuel Fabrication	0.00119	11.4	0.00112	11.6	0.00106	10.1
MOX Fuel Fabrication	0	0	0	0	0.270	4.85
LWR Power Generation	120.	562.	120.	562.	78.0	562.
Irradiated Fuel Storage	0.00921	3.65	0.00437	1.73	0.00256	1.02
Fuel Reprocessing	0	0	369.	24.4	357.	25.5
Transportation	0.111	0.359	0.117	0.365	0.167	0.874
Waste Management	0.00464	0.353	0.00226	0.337	0.00388	0.335
Industry Total (U.S.)	722.	880.	1,040.	874.	923.	874.
Option Total	1,600.		1,910.		1,800.	
Foreign Additional from USA	52.0		290.		272.	

Table IV J(E)-13

KIDNEY PERSON-REM COMMITMENT PER GWy(e)

	FUEL RECYCLE OPTION					
	No Recycle		Recycle U Only		Pu + U Recycle	
Fuel Cycle Component	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	2,390.	368.	2,170.	333.	1,930.	298.
Milling	450.	55.2	411.	50.3	366.	44.8
UF_6 Conversion	2.61	2.94	2.39	2.66	2.02	2.40
Enrichment	0.113	1.63	0.651	1.63	0.570	1.34
UO_2 Fuel Fabrication	1.51	12.3	1.58	12.5	1.38	10.9
MOX Fuel Fabrication	0	0	0	0	0.243	4.85
LWR Power Generation	75.1	562.	75.1	562.	76.1	562.
Irradiated Fuel Storage	0.00921	3.65	0.00437	1.73	0.00256	1.02
Fuel Reprocessing	0	0	372.	24.4	361.	25.5
Transportation	0.111	0.359	0.117	0.365	0.167	0.874
Waste Management	0.00353	0.353	0.00172	0.337	0.00294	0.335
Industry Total (U.S.)	2,920.	1,010.	3,030.	989.	2,740.	977.
Option Total	3,930.		4,020.		3,720.	
Foreign Additional from USA	52.0		290.		272.	

Table IV J(E)-14

THYROID PERSON-REM COMMITMENT PER GWy(e)

	FUEL RECYCLE OPTION					
	No Recycle		Recycle U Only		Pu + U Recycle	
Fuel Cycle Component	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	1.54	256.	1.40	232.	1.25	206.
Milling	0.325	44.4	0.297	40.2	0.262	35.9
UF_6 Conversion	0.0106	0.984	0.00965	0.893	0.00834	0.789
Enrichment	0.00511	0.695	0.00376	0.695	0.00330	0.557
UO_2 Fuel Fabrication	0.00107	11.4	0.00109	11.6	0.000954	10.1
MOX Fuel Fabrication	0	0	0	0	0.00110	4.85
LWR Power Generation	195.	562.	195.	562.	122.	562.
Irradiated Fuel Storage	0.00921	3.65	0.00437	1.73	0.00256	1.02
Fuel Reprocessing	0	0	653.	24.4	639.	25.5
Transportation	0.111	0.359	0.117	0.365	0.167	0.874
Waste Management	0.00733	0.353	0.00113	0.337	0.00118	0.335
Industry Total (U.S.)	197.	880.	850.	874.	763.	873.
Option Total	1,077.		1,724.		1,636.	
Foreign Additional from USA	52.0		290.		272.	

Table IV J(E)-15

LUNG PERSON-REM COMMITMENT PER GWy(e)

	FUEL RECYCLE OPTION					
	No Recycle		Recycle U Only		Pu + U Recycle	
Fuel Cycle Component	Environmental	Occupational	Environmental	Occupational	Environmental	Occupational
Mining	192.	1,372.	175.	1,247.	155.	1,111.
Milling	37.0	1,020.	33.8	933.	30.0	829.
UF_6 Conversion	0.216	7.62	0.196	6.90	0.173	6.01
Enrichment	0.105	14.0	0.0936	14.0	0.0837	11.7
UO_2 Fuel Fabrication	0.0324	449.	0.0341	458.	0.0297	401.
MOX Fuel Fabrication	0	0	0	0	0.0614	5.64
LWR Power Generation	75.1	562.	75.1	562.	76.1	562.
Irradiated Fuel Storage	0.0199	3.65	0.0095	1.73	0.0055	1.02
Fuel Reprocessing	0	0	393.	24.4	377.	25.5
Transportation	0.111	0.359	0.117	0.365	0.167	0.874
Waste Management	0.00183	0.353	0.00128	0.337	0.00153	0.335
Industry Total (U.S.)	305.	3,430.	677.	3,250.	639.	2,980.
Option Total	3,740.		3,930.		3,360.	
Foreign Additional from USA	52.		407.		386.	

Table IV J(E)-16

SKIN PERSON-REM COMMITMENT PER GWy(e)

Fuel Cycle Component	FUEL RECYCLE OPTION No Recycle Environmental	 No Recycle Occupational	 Recycle U Only Environmental	 Recycle U Only Occupational	 Pu + U Recycle Environmental	 Pu + U Recycle Occupational
Mining	1.54	256.	1.40	232.	1.25	206.
Milling	0.325	44.4	0.297	40.2	0.262	35.9
UF_6 Conversion	0.00826	2.81	0.00750	2.57	0.00658	2.17
Enrichment	0.00398	2.45	0.0256	2.47	0.0223	2.00
UO_2 Fuel Fabrication	0.00107	11.4	0.00109	11.6	0.00095	10.1
MOX Fuel Fabrication	0	0	0	0	0.00110	4.85
LWR Power Generation	77.6	562.	77.6	562.	77.8	562.
Irradiated Fuel Storage	.778	3.65	.372	1.73	0.217	1.02
Fuel Reprocessing	0	0	2,240.	24.4	2,110.	25.5
Transportation	0.111	0.359	0.117	0.365	0.167	0.874
Waste Management	0.00121	0.353	0.00113	0.337	0.00118	0.335
Industry Total (U.S.)	80.4	883.	2,320.	878.	2,190.	876.
Option Total	963.		3,200.		3,066.	
Foreign Additional from USA	54.4		8,850.		8,250.	

Table IV J(E)-17

LOW GROWTH PROJECTED LWR ELECTRICITY PRODUCTION INDUSTRY COMPOSITION 1975-2000

		Magnitude of Projected Industry 1975-2000					
		No Recycle		U Only Recycle		Pu + U Recycle	
Fuel Cycle Component	Units of Measure	Capacity	GWy(e) Equivalent	Capacity	GWy(e) Equivalent	Capacity	GWy(e) Equivalent
Mining	U_3O_8 short tons	1.77×10^6	4,732	1.58×10^6	4,651	1.37×10^6	4,536
Milling	U_3O_8 short tons	1.60×10^6	4,732	1.43×10^6	4,651	1.24×10^6	4,536
UF_6 Conversion	U Metric Tonnes	1.21×10^6	4,551	1.08×10^6	4,479	9.16×10^5	4,374
Enrichment	MT-SWU	6.08×10^5	4,603	6.13×10^5	4,607	5.23×10^5	4,490
UO_2 Fuel Fabrication	U Metric Tonnes	1.89×10^5	4,346	1.86×10^5	4,277	1.63×10^5	4,237
MOX Fuel Fabrication	Health Metal Tonnes	0	0	0	0	2.53×10^4	5,060
LWR Power Generation	Electricity GWy	4.04×10^3	4,036	4.04×10^3	4,036	4.04×10^3	4,036
Irradiated Fuel Storage	Heavy Metal Tonne-yrs.	5.59×10^5	3,072	2.79×10^5	3,224	1.67×10^5	3,299
Fuel Reprocessing	Heavy Metal Tonnes	0	0	1.15×10^5	2,954	1.15×10^5	3,021
Transportation	Shipment Miles	1.84×10^8	3,146	2.94×10^8	3,480	3.19×10^8	3,331
Waste Management	Acres	4.71×10^4	3,857	4.32×10^4	3,770	$3,82 \times 10^4$	3,638

Table IV J(E)-18

RATIO OF OCCUPATIONAL TO ENVIRONMENTAL PERSON-REM
OF USA LWR INDUSTRY, 1975-2000 (Pu + U RECYCLE OPTION)

Fuel Cycle Component	Total Body	G. I. Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Mining	.41	10.	.18	.50	.15	165.	7.1	165.
Milling	.97	9.4	1.2	.47	.12	137.	28.	137.
UF_6 Conversion	.10	.49	.52	8.4	1.2	95.	35.	330.
Enrichment	29.	.22	270.	13.	2.3	170.	140.	90.
UO_2 Fuel Fabrication	20.	18.	1.6	9,500.	7.9	11,000.	13,000.	11,000.
MOX Fuel Fabrication	83.	1,400.	1.8	18.	20.	4,400.	92.	4,400.
LWR Power Generation	7.3	7.3	2.1	7.2	7.4	4.6	7.4	7.2
Irradiated Fuel Storage	400.	400.	400.	400.	400.	400.	180.	4.7
Fuel Reprocessing	.14	.094	.058	.14	.14	.079	.13	.024
Transportation	5.2	5.2	5.2	5.2	5.2	5.2	5.2	5.2
Waste Management	164.	110.	17.	87.	110.	280.	220.	280.
Industry Total (U.S.)	.93	1.8	.44	1.0	.34	1.5	5.7	.54

GESMO

CHAPTER IV

ENVIRONMENTAL IMPACT DUE TO THE IMPLEMENTATION OF PLUTONIUM RECYCLE

SECTION K

EXTENDED SPENT FUEL STORAGE

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SUMMARY

General

Distribution of Stored Spent Fuel

The nuclear power growth projections outlined in CHAPTER III and the spent fuel discharge rates and projected reprocessing plant operations outlined in CHAPTER IV, Section E, predict that the backlog of spent fuel in storage and available for reprocessing will build to a maximum of about two times the annual reactor discharge rate during the early 1980's. If a reprocessing industry becomes a viable part of the LWR fuel cycle, this backlog, as a percentage of the discharge rate, will then gradually be reduced during the 1990's to about the then current annual reactor discharge rate. It is expected that most of the spent fuel awaiting reprocessing will be stored at the originating power plant sites. Thus, concern over spent fuel storage capacity is limited to existing power plants, and then only if reprocessing is significantly delayed beyond the projected schedule.

License application modifications for enlarging the spent fuel storage capacity at existing nuclear power plants indicate that the spent fuel storage capacity at several installations could be expanded as much as 200% providing storage on site for an average of about 10 years. Some plants that have been operating for a number of years already have an appreciable tonnage of spent fuel in storage. Spent fuel in storage in some pools may limit modifications and additional spent fuel storage capacity may have to be provided offsite. However, such plants are few in number and are not expected to change the basic situation of extended storage at power plants until the spent fuel can be shipped either to a reprocessing plant or to a Federal repository for permanent disposal.

If reprocessing plants are built as projected in Section E (assuming the prompt recycle of uranium and plutonium, Alternative 3), the distribution of spent fuel in storage is expected to be about 1 year's accumulation at the originating power plant, plus one-half year's accumulation at reprocessing plants. If reprocessing does not become available on the schedule assumed, i.e., if it is delayed a few years or, as in Alternative 6, if it is not established by 2000, it is projected that most of the spent fuel will be held in storage at power plants for about 10 years. On these terms, most present and committed power plants (through 1980 startup) could build up an inventory of 50% to 60% of their projected potential capacity, with modified storage pools. In the event that reprocessing is significantly delayed beyond the schedule in CHAPTER IV, Section E, it is assumed that post-1980 power plants will be built with increased spent fuel storage capacity. If spent fuel is treated as a waste, shipments to an assumed permanent repository are projected to start with 3,500 metric tons (MT) in the year 1986, increasing to an annual rate of about 8,400 metric tons by the year 2000. This option is discussed in CHAPTER IV, Section H.

If the Federal repository is not developed by the early 1980's, it appears that some form of supplementary spent fuel storage facilities may have to be built to augment the storage capacity of the nuclear power plants projected to be installed by 1985. Such incremental capacity needs are projected to total 20,000 MT by 1995 plus an annual addition of about 3,200 MT until about 2015 and then gradually decreasing as the reactors involved are taken out of service.

Environmental Impact of Stored Spent Fuel

The environmental impact of extended spent fuel storage represents a very small part of the total impact of the nuclear industry. Dissipation of decay heat is less than 0.3% of the total waste heat dissipated by nuclear power plants. A small fraction of the fission products, possibly on the order of 1% of the ^{85}Kr contained in spent fuel, could escape from defective fuel elements while in storage. Radioactive solid wastes are generated by storage pool water treatment systems. The volume of such wastes attributable to extended fuel storage is on the order of 2% of the total generated by the nuclear power plant at which the fuel is stored.

26-Year Integrated Environmental Impact

The integrated 26-year impact (1975-2000) projected for extended spent fuel storage results from dissipation of negligible amounts of decay heat, small atmospheric releases of ^{85}Kr, some low level solid wastes requiring disposition, and a small incremental increase in occupational dose. The totals are shown in Table IV K-1.

Sabotage Potential

Sabotage potential is covered in the Safeguards Supplement to this statement.

Accident Potential of Extended Spent Fuel Storage

In the analyses of the Maine Yankee and Point Beach modifications for spent fuel storage, the NRC staff considered criticality, cask drop consequences, mechanical and structural considerations, potential accidents during new storage rack installation, release of radioactive materials and direct radiation. On the basis of these analyses for Maine Yankee and Point Beach Plants, it was concluded that extended spent fuel storage at these two reactor sites poses no potential accident conditions not already addressed in the licensee's safety analysis reports. On the basis of these two conclusions and the continuing reviews of subsequent plans for expanding spent fuel storage at other nuclear power plants, it leads to the conclusion that extended spent fuel storage in appropriately designed and constructed facilities poses no new potential accident conditions.

Table IV K-1

EXTENDED SPENT FUEL STORAGE - 26-YEAR INTEGRATED ENVIRONMENTAL IMPACT

	With Reprocessing*	Without Reprocessing**
Heat Dissipation (Btu) x 10^{12}	100	400
^{85}Kr (curies)	2×10^5	6×10^5
^{85}Kr (person-rem; world wide)	35	110
Solid wastes (cu ft)***	3.3×10^4	1.2×10^5
Occupational Dose (person-rem)†	3.3×10^3	1.2×10^4††

Note: No other environmental value factors are significantly affected.

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*Both uranium only and uranium and plutonium recycle.

**No recycle--spent fuel disposal.

***Average of 1 Ci/cu ft and 200 cu ft per 1,000 MT/yr.

†At 20 person-rem per 1,000 MT/yr.

††Without reprocessing, there is more fuel in storage and the impact of storage of spent fuel, per se, is greater for this case.

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1.0 INTRODUCTION

1.1 Status of Spent Fuel Reprocessing Requirements Versus Capacity

On the basis of the nuclear power growth projections shown in CHAPTER III, spent fuel discharges through the year 2000 are projected to exceed somewhat the capacity to reprocess such fuel. This will result in the buildup of a backlog of fuel in storage and available for reprocessing as shown in Table IV K-2.

Table IV K-2

PROJECTED BUILDUP OF SPENT FUEL BACKLOG IN THE UNITED STATES* MTHM

Year	Discharged This Year	Reprocessed This Year	Backlog At End of Year
1975	650	0	1,200
1980	1,450	1,500	3,600
1985	3,200	2,250	4,800
1990	5,850	4,850	7,700
1995	8,700	8,250	9,400
2000	11,100	10,250	11,200

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*Based on Table IV E-5.

1.2 Projected Disposition of Stored Spent Fuels

GESMO is concerned with the widescale recycle of plutonium in LWR's in the period 1975-2000. A separate study, covering the period 1975 through 1985, will be published as the Generic Environmental Impact Statement on LWR Spent Fuel Storage. This second impact statement is being made to evaluate the near term environmental implications of extended spent fuel storage. The figures on spent fuel discharges over the next 10 years may vary from those shown in Table IV K-2. Because of contemplated changes in new plant starts, the GEIS figures could be lower than those shown above, but this is not considered significant to the longer term view of GESMO.

1.2.1 Recycle of Uranium and Plutonium (Reprocessing Assumed)

Over the next 10 to 15 years, provisions must be made to store the spent fuel backlog until the recycle decision has been made and the reprocessing industry becomes firmly established. In the long range a 12-month backlog of spent fuel awaiting reprocessing is considered to be a reasonable condition for industry operation by about the year 2000.

For the near term, reactor owners are taking steps to expand onsite spent fuel storage capacity as the need arises at individual plants. In general, the spent fuel storage facilities at power plants were conservatively designed. With replacement of present storage racks by racks with closer spacing, capacity can be increased by 100% to 200% at most LWR installations without compromising safety. With a few exceptions, provisions can be made to store spent fuel at existing power plants for about 10 years. It is expected that the exceptions, primarily the older reactors, can be accommodated by storage at a "storage only" facility such as the General Electric Company's Morris Operation, or at storage pools at existing and near future reprocessing plants, specifically the Nuclear Fuel Services (NFS), Allied-General Nuclear Services (AGNS), and the proposed Exxon plants.

Table IV K-3 shows a projected distribution of spent fuels in storage at reactors and reprocessing plants plus storage only facilities through the year 2000.

New reprocessing plants expected to come on stream starting about 1985 are projected to have a nominal spent fuel storage capacity in excess of 6 months' feedstock supply for these plants. Reprocessing plants are highly capital intensive, and it is likely that their owners will strive for the maximum base load. Exxon, for example, plans to provide spent fuel storage capacity of 7,000 MT for a plant with a capacity of approximately 2,000 MT/yr.

Nationwide, in late 1975, the 850 MT in the spent fuel storage pools of the 36,000 MWe of installed nuclear power plants represented about 25% of the storage. With modified storage racks, the total capacity theoretically could be increased to about 14,000 metric tons. On this basis, one would predict that the pools would be only about 10% full. In practice, however, it becomes increasingly difficult to modify a storage pool as the amount of spent fuel in storage is increased. On the

Table IV K-3

PROJECTED DISTRIBUTION OF STORED SPENT FUEL (WITH REPROCESSING) MTHM

Year	At Reactors*	At Reprocessing Plants and Storage Only Facilities	Total
1975	850	300	1,150
1980	2,050	1,500	3,550
1985	2,850	2,000	4,850
1990	3,000	4,700	7,700
1995	2,000	7,400	9,400
2000	1,800	9,400	11,200

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*Does not include spent fuel awaiting decay before shipping.

same basis, it is expected that the 70,000 MWe of installed capacity projected by 1980 would have a potential onsite spent fuel storage capacity for about 27,000 metric tons. Though this capacity would be distributed among all of the nuclear utilities, is dispersed geographically, and represents all types of reactors, it is estimated that 20,000 metric tons of storage capacity could be made available for the through-1980 class of reactors. It is expected that post-1980 power plants will have adequate spent fuel storage capacity to meet whatever the disposition requirement may be.

If the reprocessing industry does not develop as projected, then sometime during the late 1980's provision will have to be made for the long term storage of spent fuels or their ultimate disposition at a Federal repository.

1.2.2 No Recycle (No Reprocessing)

A projection of the disposition of spent fuels in storage over the next 26 years, for the case in which reprocessing services are not available, is presented in Table IV K-4. This table is based on the following assumptions:

- Presently projected pre-1981 reactors are assumed to have a total of about 20,000 metric tons of potentially available spent fuel storage capacity.

- Post-1980 reactors are projected to have greater than 10 years' spent fuel storage capacity on site.

- Existing licensed nonreactor storage pools are assumed to be used as storage pools to accommodate spent fuel storage capacity imbalances in the industry.

- Spent fuels will be stored for 10 years on site and then shipped to a long term storage or ultimate disposal facility.

On this basis, the present and near future (through 1980) reactors would reach a maximum onsite inventory of spent fuel in storage of about 10,000 metric tons, about 40% of their projected potential capacity of 27,000 metric tons with new racks. Non-reactor storage facilities could serve as a backup to meet short term needs. For the post-1980 reactors, the figure of 39,000 metric tons, shown for the year 2000, represents about 2.8 years' inventory of spent fuel in storage on site for an installed generating capacity of 470,000 MWe.

Table IV K-4

PROJECTED DISTRIBUTION OF STORED SPENT FUEL WITH NO REPROCESSING
MTHM

	In Storage At			Shipped to Repository*	
Year	Pre-1981** Reactors	Post-1980 Reactors	Storage only Facilities	Annual	Cumulative
1975	1,300	-	300	-	-
1980	6,550	-	1,000	-	-
1985	10,000	6,600	1,000	-	-
1990	10,000	11,700	1,000	3,500	17,500
1995	10,000	25,200	1,000	5,500	40,200
2000	10,000	39,000	1,000	8,400	76,350

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*See CHAPTER IV, Section H for a discussion of treating spent fuel as high level waste and projected shipping schedules.

**These quantities could be reduced by an expansion of storage only facilities such as the GE Morris Operation.

If reprocessing is not available by about 1985, it is assumed that reactors built after that date will have provisions for longer term storage on site. The need for storage only facilities, therefore, is based on providing long term storage for approximately the first 155,000 MWe to come on line. The timing and cumulative capacity for such storage only facilities is reflected in the last column of Table IV K-4 with additions after 1995 at the rate of about 4,650 metric tons per year until 2015, based on an average annual discharge rate of 30 MT/1,000 MWe and a 1985 installed capacity of 155,000 MWe.

2.0 ENVIRONMENTAL IMPACT OF EXTENDED SPENT FUEL STORAGE

2.1 Overall Impact

Extended spent fuel storage is thus seen to be located primarily at power plants. It is this pattern of development that is considered significant from the standpoint of potential environmental impact.

As of early 1976, 17 utilities had notified the NRC of their plans for modifying their spent fuel storage facilities. Formal notices on nine of these had been published in the Federal Register; action had been completed on four.* The environmental impact of extended spent fuel storage reported herein is based on the detailed evaluations of the applications covering modifications of the spent fuel storage facilities for the Maine Yankee Atomic Power Station (License Modification Application, Docket 50-309) and the Point Beach Nuclear Units 1 and 2 (License Modification Application, Dockets 50-266 and 50-301).

These specific licensing actions plus the detailed environmental review to which each was subjected, indicate that similar modifications can be made at other existing and near future power plants, resulting in equally insignificant environmental impacts.

2.2 Heat Dissipation

Relative to increased heat load, the Maine Yankee modification represents an addition of storage capacity for 3 cores, equivalent to 285 metric tons, with a maximum heat generation of 18 x 10^6 Btu/hr (about 0.3% of the total plant waste heat discharge which is in excess of 5,000 x 10^6 Btu/hr). With this figure as a reference (although believed to be high by about a factor of 5 times), stored spent fuel, aged from 1 to 10 years, represents a heat load of less than 75 x 10^6 Btu/hr per 1,000 metric tons. The heat load represented by spent fuel in storage is summarized in Table IV K-5.

Table IV K-5

HEAT LOAD REPRESENTED BY THE
EXTENDED STORAGE OF SPENT FUELS - 10^6 BTU/HR
(Based on 75 x 10^6 Btu/hr per 1,000 MT)

Year	With Reprocessing*	Without Reprocessing**
1975	90	90
1980	270	490
1985	365	1,320
1990	575	1,700
1995	705	2,710
2000	840	3,750

*Based on Table IV K-3.
**Based on Table IV K-4, less quantities shipped to repository. Without reprocessing there is more spent fuel in storage and hence a higher heat load from this source.

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*Maine Yankee Atomic Power Station
Point Beach Nuclear Units 1 and 2
Indian Point 2
Oconee Nuclear Station, Unit 3

To put these figures in perspective, the year 2000 totals of 840 x 10^6 Btu/hr for the reprocessing case and 3,750 x 10^6 Btu/hr for the no processing case represent about 17% and 75%, respectively, of the waste heat from one power station of the size of Maine Yankee. These are negligible nationwide effects.

2.3 Radioactive Solid Wastes

Radioactive wastes generated by the storage pool water cleanup system are to some extent proportional to the quantity of spent fuel in storage. The NRC staff study estimated, in the case of Maine Yankee, increase in stored fuel capacity by a factor of more than 200% would result in an increase of approximately 2% in the amount of solid radioactive waste generated by the plant. This is about 200 cu ft per year per 1,000 metric tons of stored spent fuel.

This type of radioactive waste is currently shipped to commercial facilities for disposal, where approximately 1 acre of space is used for each 50,000 cu ft of waste. The commitment of space is summarized in Table IV K-6. Other kinds of contaminated solid materials, such as protective clothing, small tools and equipment, are also packed in drums and disposed of off site. The contribution of this sort of waste due to increased spent fuel storage onsite is negligible because fuel storage is a static situation requiring no increase in the storage pool operations.

For perspective, the figures in Table IV K-6 might be compared to the total wastes of this type generated by reactors, which is in the order of 7,700 to 13,000 cu ft per year per 1,000 MWe.

Table IV K-6

POOL WATER TREATMENT WASTES GENERATED BY
EXTENDED SPENT FUEL STORAGE - CUMULATIVE FIGURES
(Based on 200 cu ft per 1,000 MT of Fuel Stored)

	With Reprocessing*		Without Reprocessing**	
Year	Solid Wastes Generated (cu ft)***	Required Land (Acres)	Solid Wastes Generated (cu ft)***	Required Land (Acres)
1975	225	0.0045	225	0.0045
1980	3,375	0.066	4,375	0.087
1985	7,425	0.15	16,850	0.33
1990	13,950	0.27	36,550	0.72
1995	22,900	0.45	67,100	1.35
2000	33,425	0.66	111,750	2.22

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*Based on Table IV K-3 totals

**Based on Table IV K-4 totals, less quantities shipped to repository. Without reprocessing more fuel will be in storage, and hence more pool water treatment wastes will be generated.

***Average about 1 Ci/cu ft

2.4 Radioactivity in Airborne Effluents

From its evaluation of expanded spent fuel storage at the Maine Yankee and Point Beach nuclear power stations, the NRC staff concluded that krypton-85 was the only radioactive material likely to be released as gaseous effluents in additional quantities as a direct result of increased spent fuel in storage. Because krypton-85 has a 10.7 year half-life, its release rate is a function of time in storage. For the relatively short average storage time (3 years) at Point Beach, the estimated release rate was 1.86 Ci/yr per MT of fuel stored. For the longer average storage time at Maine Yankee (about 10 years), the comparable figure was 0.7 Ci/yr per MT. For the purposes of this assessment, 1 Ci/yr per MT of fuel stored is used.

Calculated krypton-85 release rates from extended spent fuel storage are shown in Table IV K-7. A comparison with the figures in Table IV E-1 makes it clear the quantities shown in Table IV K-7 are negligible when compared to those associated with reprocessing.

Table IV K-7

RELEASE OF ^{85}Kr ACTIVITY FROM EXTENDED SPENT FUEL STORAGE - Ci/yr
(Based on 1 Ci/yr per MT of Fuel Stored)

Year	With Reprocessing*	Without Reprocessing**
1975	1,150	1,150
1980	3,550	6,550
1985	4,850	17,600
1990	7,700	22,700
1995	9,400	36,150
2000	11,200	50,000

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*Based on Table IV K-3 totals

**Based on Table IV K-4 totals, less quantities shipped to repository. Without reprocessing more fuel will be stored, hence more ^{85}Kr activity will be released from this source.

2.5 Occupational Exposure

As part of the analysis of the expansion of the Maine Yankee spent fuel storage capacity, the NRC staff estimated the increased occupational dose commitment involved. Based on the concentrations of radioactive cesium and cobalt in the pool water, this study indicated that the person-rem dose commitment for the plant as a whole is not likely to be increased by more than approximately one percent as a result of extended spent fuel storage. For this plant, this would be about 4 person-rem per year.

Based on this finding it is conservatively projected that extended spent fuel storage might result in an occupational person-rem dose commitment of about 20 person-rem per year per 1,000 MT of spent fuel stored. The figures are shown in Table IV K-8.

Table IV K-8

OCCUPATIONAL EXPOSURE - PERSON-REM DOSE COMMITMENT FOR EXTENDED SPENT FUEL STORAGE

Year	With Reprocessing	Without* Reprocessing
1975	25	25
1980	70	130
1985	100	350
1990	155	455
1995	190	720
2000	225	1,000

*Without reprocessing, more fuel will be stored and hence will be the source of greater occupational exposure from spent fuel storage.

3.0 ACCIDENT POTENTIAL OF EXTENDED SPENT FUEL STORAGE

In both the Maine Yankee and Point Beach spent fuel storage pools the reactor owners proposed to obtain increased storage capacity by the use of storage racks with a smaller center-to-center distance and the incorporation of fixed neutron poison materials. In the analysis of these modifications, the following areas were reviewed: criticality, cask drop consequences, mechanical considerations, release of radioactive materials, and direct radiation.

3.1 Criticality

Criticality evaluations (of the potential of defined geometric arrays supporting a nuclear chain reaction) were made by the licensee and by the NRC staff. The Maine Yankee design was found to have a k_{eff}* of 0.775; the Point Beach design had a "worst case" k_{eff} of 0.876--both of these are well below the acceptance criterion of 0.95. Hence, the proposed increase storage density does not significantly increase the potential for criticality. The potential for criticality will be evaluated in all future applications of this kind and the criterion of a k_{eff} of less than 0.95, to allow for calculational uncertainties, will be satisfied.

3.2 Cask Drop

Safety evaluations were made of the likelihood and consequences of a fuel shipping cask drop at Maine Yankee on March 26, 1975. It was found that the conditions of the existing Maine Yankee fuel storage pool are equally applicable to the new design and that there is no increase in the probability or consequences of a fuel cask drop accident with the proposed modifications of spent fuel storage conditions.

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*k_{eff}: when k_{eff} equals 1.000, the system would be just critical.

The NRC evaluation of the postulated spent fuel shipping cask drop accident at the Point Beach reactors is scheduled to be completed during 1976. The owners of the Point Beach reactors do not plan to ship any fuel off site for approximately 3 years, and this evaluation will be completed before then.

Based on the work done to date, there is no apparent reason to believe that modifications of spent fuel storage racks, per se, would alter the essential conditions of a cask drop accident at either plant.

3.3 Mechanical Considerations

In both the Maine Yankee and Point Beach modifications, new storage racks will meet the same design and construction criteria as the existing racks. The increased loading on the pool floors in both cases has been found to be well within the static and seismic loading capability of the existing pools. Future storage modifications of this kind will be required to meet the same criteria.

The decay heat removal system for the Maine Yankee pool is adequate to keep pool water temperature below the design limit of 154°F under the projected most adverse storage conditions. Point Beach will require future installation of an additional cooling system. Until then, it will operate under restrictions that limit decay heat input to the spent fuel pool water. Installation of additional decay heat removal capacity, where needed in future modifications of this type, is not considered of major significance.

Spent fuel pools contain a large volume of water that, even when full of spent fuel, would take many hours to heat to the boiling point in the event of a failure of the cooling system. At Maine Yankee, 7.8 hours would be required under the most adverse conditions to raise water temperature from 154°F to boiling. At Point Beach, 11.4 hours would be needed to raise the temperature from 150° to 200°F. These time spans are considered adequate for emergency actions, such as a hookup to the plant water purification or the fire protection systems, if necessary. All nuclear power plants are required to have an assured water supply to cover such an emergency. Hence, increased spent fuel in storage in these pools does not significantly change the effectiveness of their cooling systems.

3.4 Storage Rack Installation

Restrictions have been placed on installation procedures at Maine Yankee that forbid movement of new storage modules over spent fuel in storage. This precludes the possibility of any damage to spent fuel elements that could result from a construction accident, such as dropping a fuel storage rack onto spent fuel assemblies. All fuel assemblies at Point Beach have been removed from the pool in which the new storage racks will be installed.

3.5 Release of Radioactive Materials

Radioactive materials are released to storage pool water from surface films on fuel elements (primarily cobalt-60), and from fuel with cladding defects (primarily cesium-134 and cesium-137). These materials and trace quantities of others that may be present, are removed by continuous circulation of the water through a filtration and ion-exchange purification system. An increase in the quantity of spent fuel in storage in a pool is not expected to change the composition of the pool waters nor pool area radiation levels, though it may require more frequent servicing of the water purification system and result in the generation of about 2% more low level solid wastes as discussed in paragraph 2.3. There is no evidence that this in itself represents any increase in the accident potential of the plant as a whole.

There is also no evidence to indicate that a small evolution of krypton-85 from spent fuel in storage, as discussed in paragraph 2.4, will increase the accident potential of the overall plant.

3.6 Direct Radiation

Installation of new pool storage racks in most cases would require removal of existing racks, which would have to be decontaminated prior to disposal. However, in view of the radiation protection procedures routinely followed by reactor operators, this decontamination operation is considered relatively minor from a radiation exposure standpoint.

Because all storage pools are designed with adequate water over the stored fuel, an increase in the amount of fuel stored would not affect the above water radiation levels; these are primarily a function of contamination in the pool water rather than of the quantity of fuel in storage.

As discussed in paragraph 2.5, the more frequent servicing of the pool water treatment system required by increasing the amount of fuel in storage could result in a small increase in the occupational exposure of plant personnel. However, the slightly increased frequency of this operation should not significantly affect the potential for accidents from this source.

4.0 SABOTAGE POTENTIAL

A thorough treatment of this subject is presented in the Safeguards Supplement.

GESMO

CHAPTER IV

ENVIRONMENTAL IMPACT DUE TO THE IMPLEMENTATION OF PLUTONIUM RECYCLE

SECTION L

BLENDING OF PLUTONIUM AND URANIUM AT REPROCESSING PLANTS

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BLENDING OF PLUTONIUM AND URANIUM AT REPROCESSING PLANTS

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Section L
BLENDING OF PLUTONIUM AND URANIUM AT REPROCESSING PLANTS

SUMMARY

One of the safeguards alternatives for plutonium recycle is blending plutonium and uranium into a mixed oxide master blend prepared immediately after final purification of plutonium at the spent fuel reprocessing plant. To date, all MOX blending has been done at fuel fabrication plants, and current plans for new reprocessing and fuel fabrication plants reflect continuation of this practice. To perform a first stage blending at the reprocessing plant would involve changes in processes and flowsheets plus changes in transportation requirements. The effects of these changes are assessed with respect to health, safety, environmental and economic impacts. The assessment is based on blends containing 30%, 10% and 1% (or less) PuO_2 concentrations. It is assumed that reprocessing plants would send recovered uranium back to the enrichment plants and would utilize essentially unenriched uranium for blending with PuO_2, except for the very dilute PuO_2 blends, which would require slightly enriched uranium.

The very dilute blend (1% or less PuO_2) was proposed by nuclear consultant Karl Puechl,[1] who recommended that all plutonium recovered from LWR spent fuel be mixed uniformly with all the uranium used to make new fuel for LWR's. This would have the result that all LWR fuel would contain plutonium at very low concentrations, estimated to be 1% or less. With the plutonium content so low, slightly enriched uranium would have to be used for blending to attain the fissile atom content required for LWR fuel. It was Puechl's contention that the low concentrations of plutonium in this type of mixed oxide fuels would allow them to be processed in UO_2 fuel fabrication plants with only minor modifications and without causing unacceptable health, safety and environmental effects. The analysis made for this statement does not support this view. Because slightly enriched UO_2 is much less radiotoxic than plutonium, the requirements for its confinement and shielding in UO_2 fuel fabrication plants are much less stringent than for plutonium handling operations as in MOX fuel manufacture. It is calculated that if used in the standard UO_2 fuel fabrication plants the very dilute mixed blends would cause health, safety and environmental effects that are unacceptable. For example, during normal operation, the annual 50-year dose commitment to the lung of an individual worker could be in the range of 1,650 to 45,000 rem, which exceeds present standards by 110 to 3,000 times. If new facilities were provided to handle all LWR fuel in accordance with the standards for plutonium containing materials, the fuel cost is estimated to approximately double.

Mechanical blending of PuO_2 and UO_2 is considered to be the lower cost method, because it involves no extra chemical reprocessing steps and generates no liquid or gaseous process wastes that require treatment and disposal. It is calculated on the basis of rough estimates of cost that preparing the 30% and 10% blends at the reprocessing plants would increase MOX fuel costs less than 5%.

The health, safety and environmental effects of performing the first stage of blending at the reprocessing plant are calculated to be negligible. The effects at the fuel fabrication plant are also negligible except for the case of processing very dilute mixed oxide blends in a standard UO_2 fuel fabrication plant.

1.0 INTRODUCTION

In the safeguards supplement to GESMO, in the subsection titled "Costs and Effectiveness of Safeguards Alternatives," one of the options considered is the blending of plutonium with uranium into a mixed oxide "master blend" immediately after the separation and purification of plutonium at the spent fuel reprocessing plant. To perform the blending at this stage would involve changes in processing facilities and flowsheets plus changes in transportation requirements. The effects of these changes upon the health, safety, environmental and economic aspects of plutonium and uranium recycle are evaluated in this section.

Planning for the LWR fuel cycle has been based on shipping PuO_2 from the reprocessing plants to new fuel fabrication plants specially designed with gloveboxes and other features for handling plutonium and mixed oxide fuel materials. Because widescale use of mixed oxide fuels in LWR's has not been authorized by the NRC, no production scale MOX fuel fabrication plants have been built. All mixed oxide fuel used to date in the development and testing programs has been produced in research and development facilities. It is estimated that the new, specially designed MOX fuel fabrication plants would produce about 13% of the LWR fuel between 1975 and 2000 and that 87% of the LWR fuel would contain UO_2 only and would be produced in conventional UO_2 fuel fabrication plants much like those in operation today.

If PuO_2 is blended with UO_2 at the reprocessing plants, the blends will be shipped only to the MOX fuel fabrication plants. The UO_2 fuel fabrication plants that produce 87% of the LWR fuel will not be involved with the blending and will continue their present type of operations with no change other than technological improvements.

2.0 BLENDING CONCEPTS

2.1 Number of Stages of Blending

Blending of PuO_2 with UO_2 to make a mixed oxide suitable for nuclear fuel fabrication could theoretically be accomplished in a one-stage operation. However, nuclear fuel specifications may call for different concentrations of fissile plutonium in certain fuel elements, according to the customer's planned usage of the fuel. Because of the requirements for meticulous control to meet fuel specifications for plutonium concentration and uniformity of mixing, fuel fabricators unanimously prefer to do their own final blending to meet their customer's specifications. Therefore the blending option is expected to involve two stages of blending. This first stage of blending, performed at the reprocessing plant, would be designed to produce a "master blend" of standard concentration. The second stage will be "custom blending" by the fuel fabricator at the MOX plant to achieve the exact specifications called for by his customer. A 2,000 MT/yr reprocessing plant handling spent fuel during the period 1979-2000 would recover about 3 to 27 MT/yr of PuO_2. With this they would blend from 63 to 6,600 MT/yr of UO_2 according to the concentration of Pu desired in the blend.

2.2 Uranium for Blending

Since uranium and plutonium both will be recovered and purified at the reprocessing plant, it is possible that they could be recombined in the first stage of the blending operation. However, the recovered uranium will vary in enrichment from as low as 0.8% ^{235}U content to perhaps 1% or 2% enrichment for fuel discharged before reaching full burnup. Mixing batches of recovered uranium to average the isotopic abundances would give a more uniform fissile material content but would be wasteful of the separative work that had been done on the uranium of higher enrichments. Therefore, economic considerations may favor keeping recovered uranium batches separate and returning them to the enrichment plant for upgrading to the enrichments desired for future use. The recycled uranium will supply at most only about 22% of the feed needed by the enrichment plants, which will continue to use natural uranium feed as well. With respect to the uranium which will be blended with plutonium to make the master blend, the optimum selection for uniformity of the blends and for economy would be essentially natural uranium, or very low enriched uranium for the very dilute blends. It is assumed that the fuel reprocessor will return the recycle uranium to the enrichment plant and will use either natural or slightly enriched uranium for blending with the plutonium, according to the requirements of the blend.

2.3 Plutonium Concentration in the Master Blend

The industry's plans, and all of the reprocessing plants built to date, provide for the shipment of plutonium without blending at the reprocessing plant. Therefore, the base case for comparison with other options will be the shipment of plutonium to the fuel fabrication plant with no blending. Under this option the entire blending operation to reach the customer's exact specifications would be performed at the fuel fabrication plant, as it has been in the past.

To be interchangeable with the standard uranium only fuel elements, mixed oxide fuel for LWR's will normally contain plutonium concentrations in a range that provides 3% to 4% fissile plutonium, although lower concentrations can be used if offset by increased uranium-235 content. Thus, the initial blending of plutonium oxide powder and uranium oxide powder can be done with some latitude in concentrations as long as the total fissile atom content is above the final concentration to which MOX will be custom blended at the fuel fabrication plant to meet the customer's specifications. For standard LWR fuel, a master blend of mixed oxide containing at least 10% PuO_2 is indicated. A 30% blend would provide the fuel fabricator with greater flexibility. It is assumed that either of these blends would go through a second custom blending at the fuel fabrication plant to be adjusted to the exact fissile atom content specified by the user. The Puechl blending concept would result in very dilute blends containing 0.12% to 1.0% plutonium, depending upon the amount of plutonium available and the quantity of LWR fuel needed. These three blends (30%, 10% and 0.12% to 1.0%) have been selected for analysis because their concentrations encompass the range of greatest interest and their health, safety, environmental and economic effects typify those to be encountered in the blending of mixed oxides for nuclear fuel use.

2.3.1 Blend Containing 30% PuO_2

The preliminary industry reaction to possible use of 30% blends is summarized as follows:

- Additional facilities would be required at the reprocessing plant.
- MOX fuel fabricators indicate they can work satisfactorily with master blends as their starting materials.
- Additional specifications and acceptance tests would be required for master blends.

Since no detailed plans have been developed for performing blending operations, only rough estimates have been made of the capital and operating costs required to handle such blends at the reprocessing and fuel fabrication plants. As the facilities are presently designed, nearly all of the changes required for a 30% master blend would be at the reprocessing plant and would add perhaps $30,000,000 in incremental costs to the total annual costs of the reprocessing and fuel fabrication industries in the year 2000.* This would add about one percent to the cost of reprocessing and fuel fabrication (UO_2 and MOX) in the year 2000.

2.3.2 Blend Containing 10% PuO_2

For LWR fuel, a lower plutonium concentration in the master blend would be desirable. With first recycle plutonium a 5% Pu blend would be adequate for most fuel elements, but some might require concentrations as high as about 8%. With second and third recycle plutonium, the total fissile atom content of the plutonium is lower and master blend concentrations might have to go as high as 12% to allow blending to the desired fissile atom content. For the LWR fuels a nominal 10% master blend has been chosen as representative and has been used as the basis for analysis. It will be shown that the selection of 5% or 12% blend concentrations would cause only insignificant differences in the health, safety and environmental effects. Economic effects would change by a larger amount, but not enough to alter the main conclusions.

Several fuel fabricators have indicated the acceptability of blends having about 10% plutonium concentration; however, one fabricator expressed reservations about any blending, particularly in the 10% range, at reprocessing plants because this might limit the use of his uranium fluoride-to-oxide conversion facilities (because the conversion of the UO_2 used in the blend might be done elsewhere). However, with only 13% of the total LWR fuel requirement estimated to be met by MOX fuel, this does not seem likely to be a major problem. Other industrial reservations about use of 10% blends were concerned with potential quality control problems and economic aspects. The quality control concern is that the 10% blends prepared at the reprocessing plants do not allow sufficient dilution during custom blending by the fuel fabricator to

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*See Safeguards Supplement, CHAPTER VI.

assure feed material with proper particle size, sintering, and dissolution properties. This is particularly a concern when fabricators use mechanical mixing, but would be of little concern if blended powders are dissolved for coprecipitation processing.

A preliminary cost estimate indicates that the total annual costs for the LWR fuel fabrication industry to adopt the 10% blend would be roughly $50,000,000 in about the year 2000. This represents an estimated increase of about 1.5% to 2% of the costs of reprocessing and fuel fabrication (UO_2 and MOX).

2.3.3 Very Dilute Blends - 0.12 to 1.0% PuO_2

A very dilute blend has been suggested by nuclear consultant Karl Puechl,[1] who recommended that all of the plutonium formed in LWR power reactors be recovered from the spent fuel and mixed uniformly with all of the uranium used to make new fuel for LWR's. In this blending option, all LWR fuel would contain a small amount of plutonium and all recovered plutonium from fuel reprocessing would be promptly blended with uranium. The Puechl blending option was designed to reduce the risks of theft and misuse of plutonium and also to provide a very dilute mixed oxide fuel that could be fabricated in the UO_2 fuel fabrication plants without the expensive facilities normally specified for processing plutonium. It was Puechl's contention that such low concentration mixed oxide fuels could be processed in UO_2 facilities with only minor modifications to the facilities without causing unacceptable health, safety and environmental effects.

In the Puechl blending option, mixing all available plutonium with all the UO_2 required to supply the total industry needs for LWR fuels would give plutonium concentrations in the range of 0.12% to 1.0% depending on the quantity of plutonium available and the total amount of LWR fuel on order. Using the backlog of plutonium that has been accumulated in stored spent fuels while no reprocessing was being done would give an initial PuO_2 content of about 0.6% in all new LWR fuel. As the backlog disappeared and as LWR fuel needs increased, the average PuO_2 content might drop as low as 0.12%. With increased quantities of Pu being produced near the year 2000, average PuO_2 concentrations would be about 1%. In all Puechl blends, enriched uranium would have to be used to adjust the total fissile atom content to the level specified by the customer.

Assessments of the effects of handling the dilute MOX blend in present UO_2 fuel fabrication facilities indicate that the licensing standards for health, safety and environmental criteria cannot be met without extensive modifications. A rough estimate of the incremental capital and operating costs to implement the Puechl option in LWR fuel manufacturing facilities appropriate for handling plutonium is $1.5 billion per year. This would approximately double the estimated fuel costs.

2.4 Blending Methods

The usual form of the purified plutonium product from spent fuel reprocessing is plutonium nitrate in aqueous solution. It is customary to precipitate plutonium from this solution as the oxalate, which can be converted to PuO_2 by calcining. Milling and screening of the calcined oxide will produce PuO_2 powder of the desired particle size.

Ceramic grade UO_2 powder is mechanically blended with the PuO_2 to produce a mixed oxide of the desired concentration. This mixture would not be processed further at the reprocessing plant, but would be shipped in this form to the fuel fabrication plant. Because all reprocessing plants are expected to provide plutonium nitrate to oxide conversion facilities to comply with the new rule that plutonium can only be shipped in solid form, the only additional operations and facilities required to accomplish the blending at the reprocessing plant are those involved in the mechanical blending operation itself. It would be necessary to add UO_2 and PuO_2 receiving and storage vessels, a mechanical blending vessel, and mixed oxide storage vessels all of which must be designed to prevent criticality. The facilities for packaging the mixed oxide and loading the shipping containers will be little changed from those that would have been provided for unblended plutonium except that larger capacities will be required because of the UO_2 diluent.

Mechanical blending creates no liquid or gaseous wastes; it does involve processing steps in which plutonium and uranium particles could become airborne. These would be handled effectively by the standard ventilation and HEPA filter systems which assure virtually complete removal of airborne particles (see paragraph 3.1.2).

An alternate blending method is the coprecipitation of plutonium and uranium from aqueous solution. To the plutonium nitrate product solution from reprocessing operations, one could add uranyl nitrate in appropriate quantities to make a blend of the desired concentration. From this solution, plutonium and uranium could be coprecipitated by the addition of ammonia to form plutonium hydroxide and ammonium diuranate. Under carefully controlled conditions the coprecipitated solids will contain uranium and plutonium uniformly mixed. After filtration and drying, the precipitate is calcined in air to convert the uranium and plutonium to oxides. Subsequent reduction by passing hydrogen gas, usually mixed with nitrogen, over the oxides while they are still in the furnace converts both to the stable dioxide form that is desired for nuclear fuel fabrication. Formed in this way, the mixed oxide is more nearly a solid solution of uranium and plutonium, and each particle contains atoms of both elements in the desired concentration ratio.

This method of blending by coprecipitation creates a large volume of liquid waste filtrate plus gaseous wastes from the drying and calcining steps. Furthermore, it can involve the extra expense of dissolving the uranium in nitric acid to prepare the initial nitrate solution for coprecipitation. Alternatively, it may be possible to purchase uranyl nitrate solution from a wet process uranium conversion plant. Shipping uranyl nitrate would be more expensive than shipping the oxide because of the added volume and weight.

The added complexities would increase the costs for the coprecipitation method of blending, even though yielding a mixed oxide of somewhat better quality for fuel use. Accordingly, the analysis of environmental impacts of blending will be based on the lower cost mechanical blending option, which yields a product of acceptable quality.

3.0 HEALTH, SAFETY AND ENVIRONMENTAL EFFECTS OF BLENDING

With mechanical blending, there will be no liquid wastes, no significant change in airborne wastes, and no appreciable increase in occupational dose commitments to reprocessing plant workers because of the added blending step. The specific health, safety and environmental impact changes have been estimated and are discussed below.

3.1 First Stage Blending at the Reprocessing Plant

In a model reprocessing plant of 2000 MT/yr capacity the annual plutonium output could be expected to range from about 3 MT in the late 1970's to about 24 MT in the year 2000. The full 24 MT output would yield about 27 metric tons per year of plutonium dioxide for blending. To make a 30% master blend, the required quantity of uranium dioxide would be 63 metric tons (55.5 MTU) per year. For a 10% blend, the UO_2 requirements would be 243 MT (214 MTU) per year. The very dilute blend proposed by Puechl would require about 2,700 metric tons of UO_2 (2,400 MTU) per year in the year 2000 (1% blend) or up to about 6,600 metric tons of UO_2 (5,800 MTU) per year in earlier years when the plutonium concentration in the final blend might be as low as 0.12% (this would occur only in 1980 because of comparatively less plutonium available to be blended with all the uranium needed to meet LWR fuel requirements). It may also be noted that in the Puechl blend, the uranium would have to be slightly enriched to attain the fissile atom content needed for LWR fuel. Table IV L-1 summarizes the annual feed requirements for blending.

Table IV L-1

FEEDS FOR BLENDING AT A REPROCESSING PLANT
METRIC TONS PER YEAR

	30%	10%	Very Dilute
PuO_2	27	27	27
Natural UO_2	63	243	--
Low Enriched UO_2	NA	NA	2,700-6,600

The facilities for the very dilute blend would require capacities 30 to 70 times those for the 30% blend and 10 to 25 times those for the 10% blend. Thus, the capital investment and the production costs would be higher for the very dilute blends.

3.1.1 Occupational Dose Commitments for Blending

Occupational dose commitments incurred in the blending facilities for each blending option will vary in proportion to the work force. The estimated work force at a single blending facility is a function of the quantity of feeds for blending and each blending option is shown in Table IV L-2.

Table IV L-2

ESTIMATED WORK FORCE AT MODEL BLENDING FACILITIES

	Reprocessing Plant with No Blending	Additional Employees Required for the Blending Option		
		30%	10%	Very Dilute
No. of employees at a single facility	675*	26	31	125

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*From CHAPTER IV, Section E, paragraph 3.4, "Occupational Exposure."

The occupational dose commitments incurred in blending facilities for the entire LWR industry for the 26-year period and for the year 2000 are presented in Tables IV L-3 and IV L-4, respectively.

Table IV L-3

OCCUPATIONAL DOSE COMMITMENTS INCURRED IN BLENDING FACILITIES FOR LWR INDUSTRY OVER 26-YEAR PERIOD (PERSON-REM)

	Reprocessing Plants with No Blending	Additional Dose Commitment From Blending**		
		30%	10%	Very Dilute
Total Body	78,000*	1,500	1,800	5,750
Internal (Lung)		240	290	900

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*From CHAPTER IV, Section E, Table IV E-8.
**From CHAPTER IV, Section D, paragraph 4.4, "Occupational Exposure."

Table IV L-4

OCCUPATIONAL DOSE COMMITMENTS INCURRED IN BLENDING FACILITIES FOR LWR INDUSTRY IN THE YEAR 2000 (PERSON-REM)

	Reprocessing Plants with No Blending	Additional Dose Commitment From Blending**		
		30%	10%	Very Dilute
Total Body	6,750*	160	190	600
Internal (lung)		25	30	95

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*From CHAPTER IV, Section E, paragraph 3.4, "Occupational Exposure."
**From CHAPTER IV, Section D, paragraph 4.4, "Occupational Exposure."

3.1.2 Environmental Effects of Airborne Particles from Blending

For the four blending options (no blending at the reprocessing plant, blending to 30% PuO_2, blending to 10% PuO_2, and blending to very dilute Pu contents), all reprocessing operations through the preparation of PuO_2 would be the same. Since the production of MOX requires blending to a specified end concentration of Pu in the fuel, whether the blending is done entirely at the fuel fabrication plant (as has been the case in the past) or is partly done at the reprocessing plant, the choice of options would not be affected by uranium supply costs. The quantities of uranium that would be required would be the same, but blending at the reprocessing plant would involve added costs for transporting UO_2 first to the reprocessing plant and then from the reprocessing plant to the fuel fabrication plant.

For estimating purposes, it is assumed that the blending of PuO_2 and UO_2 will be done first by mechanical blending and then by jet milling (a method presently preferred in the industry) in which jet streams of airborne PuO_2 and UO_2 particles impinge violently inside a blending chamber that is designed to retain particles but allow air to escape via a bag filter, all of which are inside a confinement enclosure. Based on industry experience, it is assumed that the quantity of material released will be 0.10% of the material charged to the blender. It is assumed that all escaped particles are drawn out of the mixing vessel through a filter that removes 95% of the particles and then into the process ventilation duct where there are two stages of HEPA filters, each of which removes 99.95% of the particles encountering it. The quantity of plutonium released in all blending options annually is calculated as .3 milligrams/year which is 170 μCi of Pu/year.

The uranium releases in the case of the Puechl method in which the largest amount of uranium is processed are calculated to be 29 milligrams/year airborne uranium release, which is .086 microcuries/year.

The annual airborne releases of plutonium and uranium from the blending operations at the reprocessing plant are tabulated in Table IV.L-5 in comparison with the expected release from reprocessing operations.

Table IV L-5

INCREASE IN AIRBORNE RELEASES BROUGHT ABOUT
BY BLENDING AT THE REPROCESSING PLANT (microcuries per plant)

	Airborne Releases From Reprocessing Without Blending*	Additional Airborne Releases from Blending		
		Blending to 30% Pu	Blending to 10% Pu	Blending to 1% Pu
U isotopes** μ Ci/yr	2,690	0.002	0.008	0.086
Pu isotopes** μ Ci/yr(alpha)	70,000	170.0	170.0	170.0

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*Values from Table IV E-8

**Average specific activity for isotopic distribution expected after one recycle = 0.585 Ci/g of alpha activity for Pu and 3 x 10^{-}[illegible] Ci/g for U.

Dose commitments to the maximally exposed individual at the reprocessing plant site boundary (500 meters from point of release) would be determined by the plutonium releases and therefore would be the same for each blending option. Table IV L-6 shows the calculated values.

Table IV L-6

AVERAGE CLOSEST THEORETICAL RESIDENT ANNUAL DOSE COMMITMENTS FROM AIRBORNE EFFLUENTS FROM A MODEL PLUTONIUM AND URANIUM BLENDING FACILITY AT A REPROCESSING PLANT*

Organ	Dose Commitment to Closest Theoretical Resident (Millirem)
Whole Body	0.12
Bone	5.7
G.I. Tract	0.005
Lung	0.13
Liver	0.6
Kidney	0.5
Skin	0.003

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*Based on dose commitments given in Table IV D-10.

The additional whole body dose commitment of 0.12 millirem amounts to 1.6% of the average annual whole body dose commitment of 7.5 mrem that is received by the closest theoretical resident as a result of normal operation of the reprocessing plant (see Table IV E-12).

The estimated total dose commitments to the total U.S. population for the entire LWR PuO_2 blending industry are projected in Table IV L-7 for the 26-year period.

Table IV L-7

TOTAL U.S. POPULATION INTEGRATED DOSE COMMITMENTS FOR THE PLUTONIUM AND URANIUM BLENDING INDUSTRY

Organ	26-Year Period (Person-Rem)
Whole Body	6.8
Bone	328
G.I. Tract	0.4
Lung	7.2
Liver	31.5
Kidney	37
Skin	0.1

These dose commitments are insignificant in comparison to the approximately 20 million person-rem to the whole body from natural background radiation that will be received annually by the U.S. population in the year 2000. It should be noted that the population dose commitment from the total LWR blending industry would be the same for all blending options because the plutonium throughput would be the same. Also, although the uranium throughput does vary with the blending option employed at the reprocessing plant, the dose commitments are not appreciably changed because the doses from uranium are orders of magnitude less than for plutonium (see Table IV L-5).

3.2 Transportation of MOX Blends to Fuel Fabrication Plants

With or without blending at the reprocessing plant, recovered uranium would be transported to the enrichment plant and plutonium would be transported to the fuel fabrication plant. The changes introduced by blending at the reprocessing plant are that some or all of the UO_2 for blending would be shipped to the reprocessing plant rather than to the fuel fabrication plant and that the material shipped from the reprocessing plant to the fabrication plant would be mixed oxide rather than PuO_2.

There is no reason to estimate that the shipping distance from the UO_2 supplier to the reprocessing plant is different from the distance to the fabrication plant; so shipping the UO_2 to the reprocessing plant instead of to the fuel fabrication plant involves no difference in environmental impact. The increase in shipments of UO_2 to the reprocessing plant will be offset by a corresponding decrease in shipments to the fuel fabrication plant. However the shipment of 10% and 0.12-1.0% uranium and plutonium blends from the reprocessing plant to the fabrication plant does represent an increased transportation requirement--increased by the amount of UO_2 in the blend.

The quantities of MOX to be shipped to fuel fabrication plants from each reprocessing plant were estimated in paragraph 3.1 to be about 90 MT/yr for the 30% blend, 270 MT/yr for the 10% blend, or 2,700 to 6,600 MT/yr for the very dilute blends. The transportation effects of shipping MOX in comparison with shipping PuO_2 are given below in Table IV L-8.

Table IV L-8

TRANSPORTATION DOSES FOR SHIPMENTS FROM A REPROCESSING PLANT TO FUEL FABRICATION PLANTS

		MOX		
	No Blending (Ship PuO_2)	30% Blend*	10% Blend	1.0-0.12% Blend
Number of Shipments per year	56	56	595	1,880-6,900
Annual Dose Estimates, person-rem				
Transport Workers	2	2	25	11-40
General Public	0.4	0.4	5	2-7

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*The Integrated Container Vehicle has sufficient volume to transport the larger 30% blend without increasing the quantity of shipments. The limiting item is the quantity of plutonium, which is the same in both cases.

These radiation doses constitute less than 1% of the exposure received by transportation workers and general public as a result of all shipments in the nuclear fuel cycle. Since the total dose estimates for transportation of materials involved in the nuclear fuel cycle (see Table IV G-3) are low in comparison to permissible levels, the transportation doses from shipping MOX blends are insignificant. Details of the basis for estimating transportation doses are given in CHAPTER IV, Section G.

3.3 Fuel Fabrication Plants

With master blends prepared at the reprocessing plants, the fuel fabrication plants will have less UO_2 to procure and handle; however, the actual custom blending operation will yield the same total quantity of MOX, and the work involved in adding UO_2 and mixing to assure homogeneity will be about the same as if the entire blend had been made at the fabrication plant. This, in terms of health, safety and environmental effects, represents a duplication of the mixing done at the reprocessing plant but with additional UO_2; therefore it can be expected to release virtually the same quantities of plutonium and uranium oxides. However, it was calculated that the quantities released by blending operations at the reprocessing plant were insignificant, and so the effect of repeating essentially the same blending steps at the fuel fabrication plant will also be insignificant.

After the second stage of blending has been completed to the customer's specifications, all the mixed oxide is at the specified fuel concentration, regardless of whether the starting material was a 30% or a 10% blend, and the remaining process steps are unchanged from present methods. If the Puechl blends (1% Pu or less) were processed in the UO_2 fuel fabrication facilities, as Puechl has proposed, the changes would be very significant, in fact unacceptable according to calculations based on actual UO_2 plant operating data. These effects are discussed in the following paragraphs.

3.3.1 Fuel Fabrication Facilities for MOX Fuels

Although selection of the 30% blend or the 10% blend will have little economic effect on fuel fabrication (compared to unblended PuO_2 as a starting material), use of the very dilute blend as Puechl has proposed could have a significant effect. The 30% or 10% blends would be sent to fuel fabrication plants specifically designed to handle MOX fuels, while most (87% average for 1975-2000) LWR fuels would contain only UO_2 and would continue to be made in plants that handle only UO_2 fuel. The MOX fuel plants would be required to be designed to withstand earthquakes, tornadoes, floods, and other natural phenomena and still confine the plutonium. The choice of pure PuO_2 or a 30% or 10% blend as a starting material would not affect the quantity of the final Pu concentration of mixed oxide fuel that would be fabricated for recycle to the LWR's and therefore would not affect the environmental impact of the MOX fuel fabrication operations. However, use of the Puechl concept of blending all available Pu into the uranium used in making all LWR fuel would have the result that all fabricated LWR fuel would be mixed oxide fuel.

A key factor in assessing the economic viability of the Puechl concept is the added degree of health and safety protection that NRC may judge to be necessary for uranium fuel fabrication plants to be allowed to process mixed oxide fuel containing 0.12% to 1.0% plutonium. The present UO_2 fuel fabrication plants are not required by NRC to be designed to withstand the effects of tornadoes, floods or other natural phenomena as the plutonium handling facilities are. Although the dose commitment resulting from a tornado can be made very small through appropriate design, the cost of the facility would be greatly increased. For that reason, an economic assessment of Puechl's proposal must consider whether the UO_2 fuel fabrication plants would then have to be designed to withstand stresses caused by natural phenomena. The requirements for shielding and confinement are also important. These considerations are discussed in the following paragraphs.

3.3.2 Dose Commitments Resulting From a Tornado

An assessment was made of radioactive material released to the environment in the event of natural phenomena occurrences at fuel fabrication plants. The assessment was based on a low enriched UO_2 fuel fabrication plant processing 5 MT/day of mixed oxide fuel containing about 1.0% plutonium. An average alpha specific activity of 0.537 Ci/gm of plutonium was used. This specific activity is based upon plutonium isotope compositions predicted for first recycle plutonium, as shown in Table IV D-6 of this environmental statement. The greatest impacts upon the environment from natural phenomena are judged to result from a tornado or earthquake involving the powder processing part of the plant. For this size plant (5 MT/day), the inventory of powder in process was judged to be 50 MT.

It was postulated that a tornado would entrain 5 MT of mixed oxide, all of which would be in respirable form, i.e., particle sizes up to 10 micrometers diameter. The tornado funnel was modeled as a cylinder 50 meters in diameter and 1,000 meters high traveling 5 kilometers at a speed of 30 meters per second.

The 50-year dose commitment to the lung of a maximally exposed individual in the passing funnel, taken at the site boundary, would be approximately 1,300 rem. Assuming a population density of 100 people per square mile in the comparatively isolated vicinity of the site boundary, about 10 persons would receive such an exposure in a 5 km path. For comparison, under current radiation safety standards the annual 50-year dose commitment to the lung of an individual worker is 15 rem.

The 50-year lung dose commitment to the entire U.S. population would be about 26 million person-rem. This is based on a population density of 400 people per square mile away from the immediate vicinity of the plant (average east of Mississippi River). This same exposed U.S. population would receive an annual 50-year dose commitment of approximately 20 million person-rem from natural background.

3.3.3 Dose Commitments Resulting From an Earthquake

The effect of earthquakes was addressed in a recent study by the Battelle Pacific Northwest Laboratories.[2] Based on the PNL study, it was postulated that an earthquake, which exceeded stresses that the building was designed to withstand (the Design Basis Earthquake (DBE)), could result in the destruction of two storage bins in the bulk storage area. Since each of these bins could contain up to 170 kg of mixed oxide powder (containing 1.0% Pu), the rupture of the two bins could result in the contents (340 kg of mixed oxide containing 3.4 kg of Pu) being ejected.

The time required to completely empty the bins was assumed to be 1,000 seconds. It was estimated that 0.12 of this material (40.8 kg of mixed oxide containing 408 gm of Pu) would become airborne. Based upon a mean particle diameter of 1 to 10 micrometers, it was estimated that all of the 40.8 kg of mixed oxide would be respirable and the deposition velocity would be 1.0 cm/sec. The 50-year dose commitment to a maximally exposed individual at the site boundary could be as high as 80 rem to the lung.

The 50-year dose commitment to the exposed U.S. population would be approximately 560 person-rem to the lung based on a population density of 400 people per square mile.

3.3.4 Structural Design Requirements

In view of the potential for large releases to the environment and high dose commitments to a maximally exposed individual from tornadoes and earthquakes, it is judged necessary that facilities for the manufacture of mixed oxide fuel containing about 1.0% plutonium should be resistant to natural phenomena.

3.3.5 Evaluation of Shielding Requirements

The surface dose rates were calculated assuming no shielding. The assessments were made for fuels containing only low-enriched UO_2, a standard LWR mixed oxide fuel, and a 1.0% Pu mixed oxide. The dose rates are listed below:

Fuel Material	Gamma Dose Rate (rem/hr)	Neutron Dose Rate (rem/hr)
UO_2	∿ 0.1	nil
Standard LWR Mixed Oxide Fuel (containing 3-4% fissile Pu)	6.1	∿ 0.04
1.0% Pu Mixed Oxide	0.8	0.008

Although it does not appear that neutron shielding would be required for 1.0% Pu mixed oxide, some gamma shielding would be required.

3.3.6 Evaluation of Confinement Requirements

One of the advantages claimed by Puechl in his discussion of the very dilute blend is that it could be handled in the fuel fabrication facilities designed for low enriched UO_2 or highly enriched UO_2. He expressed the view that very dilute mixed oxide fuel could be fabricated into fuel assemblies in conventional UO_2 fuel fabrication plants without the high degree of confinement normally required for plutonium-bearing fuels.

For this reason, an assessment was made of the occupational, individual, and population dose commitments that would result from fabricating 1.0% Pu mixed oxide under the confinement conditions normally provided for UO_2 fuels in the conventional UO_2 fuel fabrication plants. For comparison, the same assessment was made for the standard LWR mixed oxide fuel. Internal dose commitments were calculated for both mixed oxides based on the same concentration of oxide in the air that would be equivalent to half the maximum permissible concentration (MPC) for low (3% to 5%) enriched UO_2 or for 93% enriched UO_2. The UO_2 plants are generally designed to achieve this conservative degree of containment. If mixed oxide were being processed in the plant, air concentrations equal to half the MPC for uranium would be encountered at times.

The airborne concentration of 3.3% enriched UO_2 for 50% lung MPC (which is about the level that UO_2 fuel fabrication plants experience in dusty operations) is 1.67 x 10^{-11} gm/cm^3. With a 1.67 x 10^{-11} gm/cm^3 concentration of 1% Pu mixed oxide in air, the annual 50-year lung dose commitment to an individual breathing this air would be 45,000 rem. Obviously, the air concentrations which are permissible for 3.3% enriched UO_2 would not be acceptable for 1% Pu mixed oxides. Much greater degrees of confinement of the mixed oxide, such as installation of gloveboxes, would be required.

The airborne concentration of highly enriched (93%) UO_2 for 50% MPC is 6.25 x 10^{-13} gm/cm^3 (Ref. 3). If 1.0% Pu mixed oxides were allowed to reach this concentration in air, the annual 50-year dose commitment to an individual breathing the air would be about 1,650 rems; again, much too high. Therefore, the processing of Puechl blend mixed oxides even in the highly enriched UO_2 fuel fabrication plants could not be done without major modifications. The high occupational dose commitments described above show that the confinement required for processing UO_2 fuel would not be adequate for processing even very dilute MOX blends. Confinement would have to be improved by a factor of about 500.

3.3.7 Estimated Doses From Gaseous Effluents

The dose commitments to the average closet theoretical resident and the dose commitments to the U.S. population were calculated for UO_2 fuel fabrication plants processing the very dilute (1% Pu) mixed oxide. These calculations were based on an annual airborne release of 2 kg of mixed oxide. This fraction of throughput (1.33 x 10^{-6}) is typical of the losses of UO_2 particulates in gaseous effluents from the powder and pellet portions of a low enriched UO_2 fuel fabrication plant. The UO_2 plant has one bank of HEPA filters. For 1.% mixed oxide, the annual release passing one HEPA filter would result in 50-year lung dose commitments of about 8 rem to the average closest theoretical resident and 12,000 person-rem to the U.S. population.

Two or more stages of HEPA filters in series would be required for confinement control in 1.0% MOX fuel fabrication plants to reduce those exposures to 4 mrem to the average closest theoretical resident and 6 person-rem to the U.S. population.

3.3.8 Comparison of Environmental Impacts of LWR Fuel Fabrication Industry With Puechl Recycle Mode and With Standard Recycle Mode

A comparison was made of radioactive exposures that would result from the entire industry processing the 1.0% Pu mixed oxide in MOX fuel fabrication plants of the type designed for recycle of the standard LWR mixed oxide fuel (3% to 4% fissile Pu). The estimated annual 50-year occupational dose commitments and the annual 50-year dose commitments to the U.S. population are shown in Table IV L-9. The estimates for the standard LWR mixed oxide are presented in CHAPTER IV, Section D, and the estimates for the supporting UO_2 fuel fabrication industry are presented in CHAPTER IV, Section F-5.0.

The integrated occupational and U.S. population dose commitments covering the period 1975-2000 for the entire industry fabricating 0.12% to 1.0% Pu mixed oxide would be 113,540 person-rem (total body) and 14,000 person-rem (bone), respectively (Table IV L-9). This compares to integrated occupational and U.S. population dose commitments of 1,700,000 person-rem (lung) and 50,000 person-rem (bone), respectively, from the total LWR fuel fabrication industry using the 1.15 SGR recycle mode (both UO_2 and mixed oxide) over the 26-year period. The reason for the large reduction in U.S. population and occupational dose commitments shown for the 1.0% Pu recycle concept is that in this case all LWR fuel would be processed in mixed oxide fuel fabrication plants which have much higher degrees of confinement than the UO_2 fuel fabrication plants. For the standard LWR fuel, only 13% would be processed in MOX plants and 87% would contain UO_2 only and would be processed in UO_2 fuel fabrication plants from which particulate releases are generally higher, although well within acceptable limits. The 50-year dose commitments are shown in Table IV L-9.

A detailed discussion of the risks associated with radiological exposures is presented in Section J of CHAPTER IV.

4.0 ASSESSMENT OF BLENDING OPTIONS

The proposal to blend all available plutonium with all the uranium required to meet LWR fuel needs is not attractive. If the very dilute blends thus created are processed in UO_2 fuel fabrication plants, the health, safety and environmental effects are unacceptable. If special facilities are built to handle the fabrication of all LWR fuel with very dilute blends, the fuel cost will approximately double.

If safeguards considerations result in a requirement for blending mixed oxides at the reprocessing plants, blends having PuO_2 concentrations of 10% to 30% can be prepared with health, safety and environmental effects well below permissible limits. Very rough estimates of costs associated with blending indicate that fuel prices would probably increase less than 5% as a result of performing the first stage of blending at the reprocessing plants.

Table IV L-9

CUMMULATIVE DOSE COMMITMENTS FROM THE ENTIRE FUEL FABRICATION INDUSTRY OVER A 26-YEAR PERIOD FOR TWO BLENDING CONCEPTS (PERSON-REM)

50-Year Dose Commitment to Total Body
(Person-Rem)

	1.0% MOX Fuel				Standard MOX Fuel			
	Occupational		U.S. Population		Occupational		U.S. Population	
	Total Body	Organ	Total Body	Organ	Total Body	Organ	Total Body	Organ
UO_2 Fuel Fabrication	Not Applicable				43,800	1,693,000	2,200***	36,000***
Mixed Oxide Fuel Fabrication	114,000*	18,500*	300	14,000	25,000**	4,000	300**	14,000**
TOTAL	114,000	18,500	300	14,000	69,000	1,697,000	2,500	50,000

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*Based on same individual occupational dose commitment per MT of Pu throughput as standard, i.e., 3-4% fissile Pu, MOX fuel fabrication plant, CHAPTER IV, Section D, paragraph 4.4

**Table IV D-2

***Table IV F-20

REFERENCES

1. K. H., Puechl, "The Case for Low Concentration Plutonium Recycle," Nuclear Engineering International, September 1975, p. 687.

2. "Airborne Plutonium Releases Postulated for Serious Accidents in a Commercial Scale Mixed Oxide Fuel Fabrication Plant," PNL for Office of Standards Development, November 1974,

3. 10 CFR 20, Appendix B.