PROPOSED FINAL
ENVIRONMENTAL STATEMENT

**LIQUID METAL
FAST BREEDER REACTOR
PROGRAM**

VOLUME II

DECEMBER 1974

U.S. ATOMIC ENERGY COMMISSION

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ENVIRONMENTAL STATEMENT

LIQUID METAL FAST BREEDER REACTOR PROGRAM

VOLUME II

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U.S. ATOMIC ENERGY COMMISSION

DECEMBER 1974

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EDITORIAL NOTE

This Environmental Statement has been prepared in seven volumes, each containing one or more Sections, the titles of which are listed on the flyleaf. A Table of Contents is contained in each volume and a summary is provided in front of each Section. The outline and contents of the Statement generally follow the subject coverage required by the National Environmental Policy Act (NEPA) of 1969.

VOLUME I contains a summary of the entire Environmental Statement and background information on the U.S. energy economy, the LMFBR Program and the relationship between the two. It includes discussion of the past, current and projected uses of energy and its importance to society, and describes the role of electricity, including that produced by nuclear reactors, in helping to meet the Nation's energy requirements. This volume also summarizes the origins and history of the LMFBR Program and provides a brief discussion of the several experimental and special purpose fast reactors that have been built in the United States since the late 1940's. Volume I also reviews the fast reactor programs in other industrialized nations. A discussion of the current U.S. LMFBR program is offered which highlights the important program planning mechanisms, the key reactor plants now under design and construction, and the various supporting studies on LMFBR costs, technology, environmental impacts, and program planning. This volume lays the background for examination of the environmental characteristics of the LMFBR.

VOLUME II describes the direct environmental impact of each element of the LMFBR fuel cycle. It examines the power reactors, fuel fabrication plants and fuel reprocessing plants that make up the LMFBR fuel cycle and discusses for each the siting considerations, plant characteristics, effects on the environment from construction and normal operation, and environmental monitoring programs that together entail a complete environmental evaluation. Volume II also includes an evaluation of the potential environmental impacts of various types of accidents in the facilities comprising the LMFBR fuel cycle. In addition, this volume examines the transportation of radioactive materials between these facilities and the management of radioactive wastes produced in LMFBR activities, and analyzes in detail the properties of plutonium and its behavior in the environment, and the resulting health effects. Extensive supporting data are provided in the appendices to Volume II. The volume concludes with an examination of the related sociopolitical impacts of the LMFBR.

VOLUME III examines individually the various alternative technologies, nuclear as well as nonnuclear, that might be utilized in conjunction with or instead of the LMFBR to satisfy the Nation's future electric power requirements. The options considered include the further implementation of various types of nuclear power reactors such as

the already existing Light Water Reactor and High Temperature Gas-Cooled Reactor, as well as the development of alternative breeder reactors such as the Gas-Cooled Fast Reactor, Light Water Breeder Reactor and Molten Salt Breeder Reactor. The development of another potential nuclear energy system, controlled thermonuclear fusion, is also addressed. The possibilities of increased emphasis on the use of conventional fossil fuels, namely coal, oil and natural gas, and the development of unconventional fossil fuels such as oil shale and domestic tar sands are discussed next, followed by consideration of the further development of additional nonnuclear energy sources such as hydroelectric power systems, geothermal energy, solar energy, and other potential sources of power. Each option is examined as to the extent of its energy resource base, the research and development program that would be required (if any) to bring the option into commercial use, the environmental implications of its utilization and the costs and benefits associated with its use, in order to assess its capability for satisfying projected energy requirements. This volume also discusses the use of improved energy conversion and storage devices such as gas turbines, fuel cells and magnetohydrodynamics, and concludes with an examination of the various elements of a potential national effort in energy conservation to assess their capabilities for reducing projected energy demands and thereby replacing partially or entirely the need for additional power sources such as the LMFBR.

VOLUME IV provides a broad overview of the many implications of LMFBR program implementation, up to and encompassing a fully developed LMFBR power plant economy, including the secondary impacts, the unavoidable adverse environmental impacts, cumulative environmental impacts, and cost-benefit analyses, and also discusses alternative energy strategies. Under the heading of secondary impacts, it examines the national implications of the availability of electricity from LMFBRs, and the specific economic impacts of the LMFBR program. This volume also discusses the currently feasible alternatives and potential future alternatives for mitigating adverse environmental impacts of the LMFBR fuel cycle, and in this context analyzes the problems of safeguarding special nuclear material from potential diversion to unauthorized purposes. Also covered in Volume IV are the cumulative environmental effects of LMFBR operation to the Year 2020, the decommissioning of LMFBRs and fuel cycle facilities upon the completion of their useful life, the irreversible and irretrievable commitments of resources that will accompany implementation of an LMFBR economy, and an analysis of the costs and benefits of implementing the LMFBR Program.

VOLUMES V - VII contain copies of all formal comments received on the Draft Statement and copies of the AEC's replies. Where appropriate, these comments have been identified and discussed in the text, and are further identified by footnotes indicating the letter and page number in which the comment appears. Finally, Volume VII includes information concerning the public hearing held on April 25-26, 1974.

ALPHABETICAL LIST OF ABBREVIATIONS,
SYMBOLS AND ACRONYMS USED IN THE STATEMENT

A	ampere	cm ³	cubic centimeter
AC	alternating current	Co	cobalt
ACRS	Advisory Committee on Reactor Safeguards	Cs	cesium
ADU	ammonium diuranate	CO	carbon monoxide
AEC	Atomic Energy Commission	CO ₂	carbon dioxide
AEC-RL	AEC - Richland Operation	DC	direct current
AI	Atomics International	DF	decontamination factor
AMAD	activity median aerodynamic diameter	DOD	Department of Defense
ANL	Argonne National Laboratory	DOP	dioctyl phosphate
ANPO	Aircraft Nuclear Propulsion Office	DOT	Department of Transportation
ARHCO	Atlantic Richfield Hanford Company	DTPA	diethylenetriaminepentaacetic acid
AWSF	Alpha Waste Storage Facility	diam.	diameter
Ag	silver	EBR-I	Experimental Breeder Reactor - I
Am	americium	EBR-II	Experimental Breeder Reactor - II
As	arsenic	EEI	Edison Electric Institute
Ar	argon	EHV	Extra High Voltage
Avg.	average	EIAP	Environmental Impact Assessment Project
BEIR	Biological Effects of Ionizing Radiation (Committee)	EIS	Environmental Impact Statement
Btu	British thermal unit	EPA	Environmental Protection Agency
BRC	Breeder Reactor Corporation	Eu	euporium
BWR	boiling water reactor	°F	degrees Fahrenheit
B&W	Babcock and Wilcox	FBI	Federal Bureau of Investigation
Bi	bismuth	FEA	Federal Energy Agency
Br	bromine	FHA	Federal Housing Administration
CCD	counter-current digestion (ore leach process)	FFTF	Fast Flux Test Facility
CE	Combustion Engineering	FPC	Federal Power Commission
CEA	Commissariat à l'Énergie Atomique	ft	foot (feet)
CEQ	Council on Environmental Quality	ft ²	square feet
CIA	Central Intelligence Agency	ft ³	cubic feet
CF	Confinement factor	Fe	iron
CFR	Code of Federal Regulations	GAC	Gulf Atomic Corporation
CRBR	Clinch River Breeder Reactor	GAO	General Accounting Office
CTR	Controlled Thermonuclear Reactor	GCFR	Gas Cooled Fast Reactor
CWP	Coal worker's pneumoconiosis (black lung)	GE	General Electric Company
Cf	californium	GI	gastro-intestinal
cfs	cubic feet per second	GNP	gross national product
°C	degrees Centigrade	GW	gigawatt
Ci	curie	GWe	gigawatt electric
C	carbon (carbide)	GWT	gigawatt thermal
Ca	calcium	g	gram
Ce	cerium	gal	gallon
Cm	curium	gpd	gallon per day
cm	centimeter	gpm	gallon per minute
cm ²	square centimeter	H	hydrogen
		H-3	tritium
		HCDA	hypothetical core disruptive accident
		HEDL	Hanford Engineering Development Laboratory

HEPA	high efficiency particulate air (filter)	MPC	maximum permissible concentration
HEW	Health, Education and Welfare (Dept. of)	MSBE	Molten Salt Breeder Experiment
HF	hydrogen fluoride	MSBR	Molten Salt Breeder Reactor
HPOF	high pressure oil filled (cable)	MT	metric ton (tonne)
HTGR	High Temperature Gas Reactor	MTU	metric ton of uranium (metal)
HVDC	high voltage direct current	MUF	material unaccounted for
hr.	hour	MW	megawatt
I	iodine	MWd	megawatt-day
IAEA	International Atomic Energy Agency	MWe	megawatt electric
ICC	Interstate Commerce Commission	MWt	megawatt thermal
ICRP	International Commission on Radiological Protection	m	meter
ID	inside diameter	m ²	square meter
IRAP	Interagency Radiological Assistance Program	m ³	cubic meter
in	inch (es)	mCi	millicurie (10 ⁻³ curie)
K	potassium	μCi	microcurie (10 ⁻⁶ curie)
Kr	krypton	MCi	megacurie (10 ⁶ curies)
Kv	kilovolt	μm	micrometer
k eff	effective multiplication constant	mg	milligram
kCi	kilocurie (1000 curies)	min	minute
kg	kilogram	ml	milliliter
km	kilometer	mph	miles per hour
kV	kilovolt	Mn	manganese
kW	kilowatt	mrem	millirem
kWe	kilowatt electric	NASA	National Aeronautics and Space Administration
kWhr	kilowatt-hour	NAS-NRC	National Academy of Sciences - National Research Council
kWt	kilowatt thermal	NEPA	National Environmental Policy Act
LASL	Los Alamos Scientific Laboratory	NFS	Nuclear Fuel Services
LAMPRE	Los Alamos Molten Plutonium Reactor Experiment	NGSF	Noble Gas Storage Facility
LE	limit of error	NRDC	Natural Resources Defense Council, Inc.
LET	linear energy transfer	NRTS	National Reactor Test Station
LLL	Lawrence Livermore Laboratory	NSF	National Science Foundation
LNG	liquid natural gas	NSSS	nuclear steam supply system
LMFBR	Liquid Metal Fast Breeder Reactor	Nb	niobium
LOCA	loss of coolant accident	Np	neptunium
LSA	low specific activity	NO ₂	nitrogen dioxide
LWBR	Light Water Breeder Reactor	NO _x	oxides of nitrogen
LWR	Light Water Reactor	nCi	nanocurie (10 ⁻⁹ curie)
l	liter	OD	outside diameter
lb	pound	OMB	Office of Management and Budget
Max	maximum	OP	oxygen pressure process (ore leach process)
MeV	million electron volts	ORNL	Oak Ridge National Laboratory
MHD	magnetohydrodynamics	PCRv	prestressed concrete reactor vessel
MIT	Massachusetts Institute of Technology	PMC	Project Management Corporation
		PNL	Pacific Northwest Laboratory
		PSAR	Preliminary Safety Analysis Report
		P&W	Pratt and Whitney
		PWR	Pressurized Water Reactor
		pCi	picocurie (10 ⁻¹² curie)

ppb	parts per billion
ppm	parts per million
psi	pounds per square inch
psia	pounds per square inch, absolute
psig	pounds per square inch, gauge
Pb	lead
Pm	promethium
Pu	plutonium
PuO ₂	plutonium dioxide
°R	degrees Rankine
R	Roentgen
R&D	research and development
RSSF	Retrievable Surface Storage Facility
rpm	revolutions per minute
Rb	rubidium
Ru	ruthenium
SAR	Safety Analysis Report
SEFOR	South-East Fast Oxide Reactor
SIPI	Scientist's Institute for Public Information
SNG	synthetic natural gas
SNM	special nuclear material
SRE	Sodium Reactor Experiment
STP	standard temperature and pressure
SWU	separative work unit
scfm	standard cubic feet per minute
sec	second
sq. ft.	square feet
Sb	antimony
Sr	strontium
SO ₂	sulfur dioxide

SO _x	oxides of sulfur
TEG	thermoelectric generator
TFE	thermionic fuel element
TGLM	task group lung model (ICRP)
TVA	Tennessee Valley Authority
TVR	Tennessee Valley Region
Te	tellurium
Tonne	metric ton
U	uranium
UHV	ultra high voltage
UMRB	Upper Mississippi River Basin
UO ₂	uranium dioxide
U ₃ O ₈	black oxide of uranium
UF ₆	uranium hexafluoride
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
USAEC	United States Atomic Energy Commission
USGS	United States Geological Survey
W	watt
We	watt electric
WEP	water extended polyester
WLM	working-level month
wt. %	weight percent
w/o	without
Xe	xenon
Y	yttrium
yr	year
Zr	zirconium

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VOLUME II

SECTION 4

**ENVIRONMENTAL IMPACT
OF THE
LMFBR FUEL CYCLE**

SUMMARY FOR SECTION 4

The expected environmental effects of the LMFBF fuel cycle are discussed in Section 4, on the basis of model fuel-cycle facilities. Section 4.2 presents such information on the model 1000-MWe LMFBF power plant.

During construction of an LMFBF power plant and associated transmission lines, changes will necessarily be made in the topography of the site and related areas. These changes will result from excavation, grading, and the construction of roads, buildings, and related facilities. About 35 to 50 acres will be required for the plant buildings, plus an additional 1500 acres if an artificial cooling lake is to be built. Transmission lines will require another 20 to 50 acres/mile.

Once-through cooling at a rate of 1500 cubic feet per second (cfs) will produce a 15°F temperature rise across the condenser for a 1000-MWe LMFBF power plant. Evaporation due to the added heat will be 12 to 13 cfs. With other cooling systems, evaporative water loss may be up to 22 cfs. Spray canals and evaporative cooling towers will lose water due to drift alone of up to 0.05 cfs. These structures may be responsible for fogging and icing, but the effects will be limited to the plant vicinity.

Under normal operating conditions, radioactive liquid releases (tritium only) from an LMFBF power plant could result in a maximum dose rate of 0.001 millirem/year to an individual member of the population living within 50 miles of the plant, or a population dose rate of 0.47 man-rem/year. Doses primarily due to gaseous releases from leaks will be greatest within one mile of the plant, and the dose rate was estimated to be 0.012 millirem/year 1000 m from the plant. Within 50 miles, the population dose rate was estimated to be 0.10 man-rem/year from this source.

The environmental impacts of potential accidents in an LMFBF power plant are examined in Section 4.2.7. A spectrum of accidents of decreasing probability and increasing potential severity is addressed in the context of the AEC's defense-in-depth approach. Examples of specific accidents discussed in Section 4.2.7 include accidents that would have relatively minor consequences such as fuel failures, radioactive waste treatment system failures, and steam generator failures. In addition, the potential results of very unlikely events that could have more serious consequences such as large sodium fires, major heat transport system failures, and core disruptive accidents are considered.

The consideration in this Statement of very unlikely events in LMFBR power plants is not accompanied with rigorously derived quantitative estimates of the probability (probable frequency) of their occurrence. While methodology (probabilistic analysis) for assessing such probabilities has been under development in recent years, it requires certain input data that does not yet exist for the LMFBR, including detailed power plant design and experience with the performance of various types of components. When the requisite information is available, probabilistic assessments of risk will be made for the LMFBR power plants expected to be in operation. In the interim, a probabilistic assessment of hypothetical accidents for a conceptual 1000-MWe LMFBR has been reported. Some results of this study are briefly noted in Appendix II.G-1.

An assessment of accident risks in the types of commercial light water reactors in operation today has been conducted over the past two years. A draft report of this study was published in August 1974, and the summary portion is reproduced as Annex E to Section 4.2.7 for the information of the reader. Some of the conclusions are:

- (1) the only way that potentially large amounts of radioactive material can be released is by melting the fuel in the reactor core;
- (2) only about one in ten potential core melt accidents (among 100 operating reactors), occurring on the average of once every 17 centuries, might produce measurable health effects;
- (3) the consequences of potential reactor accidents are no larger, and in many cases much smaller, than those of non-nuclear accidents;
- (4) the likelihood of reactor accidents is much smaller than many non-nuclear accidents having similar consequences;
- (5) from the viewpoint of a person living in the general vicinity of a reactor, the likelihood of being killed in any one year in a reactor accident is one chance in 300,000,000. The chance of being killed in any of the non-nuclear accidents that occur each year is about one in 1600.

A reasonable expectation is that future risk assessments of LMFBR power plants will similarly confirm the value judgments (e.g., "very unlikely") currently being applied to potential accidents in conceptual LMFBR systems having an impact on public health and safety.

Fuel-Fabrication Plant

A comprehensive assessment of all potentially important exposure pathways indicates that radionuclide releases from normal operation of the LMFBF fuel-fabrication plant would likely result in low radiation exposures to individuals and to the population. The maximum individual adult total-body dose predicted to result from all exposure pathways is estimated to be 0.059 millirem/year and the population total-body dose is estimated to be 1.2 man-rems/year for the total population within 50 miles of the plant. Likewise, organ doses are low; the maximum calculated adult critical organ (bone) dose is estimated to be 1.0 millirem/year.

A number of potential accidents in fuel-fabrication plants were analyzed. The maximum potential total-body dose to an individual at the site boundary (1000 m from point of release) was calculated to be 4.2 millirems and would result from the accidental release of plutonium during a general fuel-fabrication plant fire. The corresponding dose to the population within 50 miles of the accident was calculated to be 0.51 man-rem. The accident providing the maximum potential damage to the environment due to the release of chemicals was the rupture of a cylinder of molten UF_6 . The accidental release of large quantities of UF_6 and its reaction product HF could cause localized effects on the environment. Effects on vegetation (foliar necrosis) and possibly on livestock (dental mottling, repression of food intake) would be expected at 1000 m. Sensitive plant species would be damaged, but there are insufficient data to predict the response of resistant perennial species (flowers and trees).

Fuel-Reprocessing Plant

The potential maximum individual total-body dose likely to result from exposure to the routine releases from the model LMFBF fuel-reprocessing plant under assumed typical meteorological conditions and population distribution is estimated to be 2.0 millirems/year. For the average individual within 50 miles, this dose is predicted to be 0.035 millirem/year. The calculated population dose of 35 man-rems/year for one million people living within 50 miles of the plant would be a very small fraction of the population dose from natural radiation and medical exposures.

A representative spectrum of potential accidents in the model LMFBF fuel-reprocessing plant was examined. These included: (1) fuel element rupture in the cell but outside the process equipment, (2) leakage of highly radioactive liquid from a process vessel to a cell, (3) a solvent fire in a cell, (4) nuclear criticality in a process vessel, (5) explosions rupturing various process vessels, and (6) sudden

release of gas from a Kr-85 storage vessel. The maximum doses to an individual at the plant boundary were calculated to be less than 1 millirem for all accidents. These doses are much lower than doses due to background radiation or medical exposures.

Transportation

Under normal operating conditions, heat from spent fuel casks is insignificant; for example, the heat generated by a spent fuel shipment is about one-fourth that generated by an automobile engine during turnpike driving. Transport workers and the general public will receive, respectively, exposures of 11.8 and 2.9 man-rem from all shipments of radioactive materials associated with the operation of one reactor and its supporting fuel cycle each year.

Nonradiological consequences involve injury or death to transport workers or the general public but at no higher frequency rates than those experienced in shipping similar non-nuclear cargo. In one year, 0.80 injuries and 0.077 deaths would be expected to result from all shipments required for a 1000-MWe LMFBR power plant and its associated fuel cycle.

An extremely severe accident involving the spent fuel cask was evaluated. Most likely releases involve radioactive gases such as krypton but might include cesium and strontium. The analysis indicated the short-time release would provide the highest doses both to individuals and the population as a whole, 4.3 rem whole-body dose* to an individual 50 meters away from the release and 2.1 man-rem to the population beyond 50 meters and within 50 miles of the accident. The dose due to natural background radiation for this population is about 130,000 man-rem.

Waste Management

The normal release of radioactive material from operation of the retrievable surface storage facility would be less than 0.001 Ci annually in air and water. Assuming the site boundary is at least five miles from the facility, the radiation exposure to an individual of the general public would be less than 0.001 millirem/year. The maximum credible accident is postulated to involve a major rupture of a waste canister as a result of being dropped during handling or inspection. A total of 0.01 Ci of waste (in particulate form) could escape to the atmosphere. Maximum

*This dose is less than the AEC accident guidelines (10 CRF 100--25-rem whole body dose).

radiation exposure (50-year bone dose) to an individual of the general public located at the site boundary would be approximately 4 millirems for this accident.

The normal operations involved in the burial of other-than-high-level radioactive wastes in solid form are not expected to result in any effluent releases to the offsite environment.* Accidental opening of a waste package prior to burial and release of the package contents could occur. Normally, the scattered waste materials would be confined to the fenced facility and the impact on the environment would be negligible. Recovery of the bulk of the scattered waste could easily be accomplished. Radiation surveys would be made and the area decontaminated.

Normal radioactive material releases from the noble gas storage facility would be very small and would be exclusively to the air. At the site boundary, radiation exposures from the facility would be less than 0.01 millirem/year for a member of the general public. The maximum credible accident in the noble gas storage facility is postulated to involve a major rupture of a single gas cylinder with the resultant rapid and total escape of the noble gas from the cylinder. Exposure to an individual of the general public for this case would be 0.13 millirem at the site boundary. This dose is much lower than doses due to background radiation and medical exposure.

Fuel Cycle

The doses due to normal operation of the model LMFB power plant and its associated fuel cycle are given in Tables 8.3-1 and 8.3-2. The doses are those due to one 1000-MWe LMFB power plant and its fuel cycle requirements for one year. Average doses due to natural background radiation, fallout from weapons testing (based on 1970 data), and medical uses (based on average 1970 diagnostic use) were included for perspective. Yearly average population doses due to the operation of one 1000-MWe LMFB power plant and its supporting fuel cycle are considerably smaller than these other doses, particularly the natural background and medical doses.

The maximum individual doses for the LMFB power plant shown in Table 8.3-1 are the site-boundary inhalation doses due primarily to the leakage of radioactive noble gases, while the annual population dose is primarily due to the ingestion of water containing tritium that diffuses through the cladding and piping of the LMFB power

*A conceptual alpha waste storage facility is also described in the Final Statement. This facility would be suitable for receiving and sorting the incoming waste shipments and would provide high integrity interim storage warehouses that could be expanded incrementally as needed. The concept is based on safely holding alpha wastes for a period of up to 100 years until a permanent disposal concept and site are developed.

plant and is discharged in the steam generator blowdown. The maximum individual doses for the fuel-fabrication plant releases are the total-body doses at the site boundary from all exposure pathways and all normal releases. The annual population dose is the total-body dose from all exposure pathways and all normal releases from the fuel-fabrication plant. The maximum individual dose for the fuel-reprocessing plant is the GI-tract dose* due to ingestion of crops grown at the site boundary (containing mostly Ru-106 released to the atmosphere from the reprocessing plant). The annual population dose is the total-body dose from all exposure pathways for atmospheric releases from the reprocessing plant. (There are no liquid releases from the reprocessing plant.) The maximum individual nonoccupational doses due to transportation are total-body doses due to direct radiation (truck shipments of high radiation level materials--12 min at 50 millirems/hr). The population dose for transportation is the total-body direct radiation dose due to all shipments of radioactive material for one 1000-MWe LMFBR power plant and its supporting fuel cycle for one year and is received by individuals located along the shipping routes.

Plutonium Toxicity

Estimates have been made of the total potential health consequences to this and all future generations which might be attributed to the release of plutonium and other transuranic elements during operation of the LMFBR fuel cycle. The estimates are based upon: (1) the amounts of transuranic elements expected to be released during operation of LMFBR power plants and associated fuel cycle facilities and (2) the fraction of the amount released to the environment that may find its way to man and the possible health consequences occurring from the incorporation of this material into man (Section 4.7, Appendix II.G).

The main source of transuranic elements is expected to be the releases from fuel reprocessing plants (all airborne from the stack). Releases from these plants are expected to be less than 0.4 millicurie of alpha-emitting transuranic activity for each 1000-MWe-year equivalent of fuel reprocessed, and airborne releases from all other components of the LMFBR fuel cycle are expected to be smaller by a factor of 10 or more.

A fraction of released transuranics will eventually enter man by two paths--inhalation and ingestion. An estimate was made of the cumulative fraction inhaled by all individuals before the released materials are deposited on the ground. After reaching the ground, the material might be resuspended in the air and inhaled; and

*The total-body dose is very small compared with this dose.

the fraction of the released alpha activity that, for all future years, might enter man in this way is conservatively estimated to be comparable to the amount inhaled before deposition. Most of the alpha activity would enter man within two years after release. Observations on global fallout from weapons testing indicate that the model and parameters used do not underestimate the resuspended transuranics that might enter man soon after release.

The ingestion estimate assumes that the released material is distributed uniformly across the surface of the United States. This assumption leads to an underestimate of the amount entering man for points of release upwind of land used for intensive agriculture, and to an overestimate of the amount for points of release upwind of lands used for other purposes. The estimates assumed that plant material achieves a concentration of transuranics equal to one-tenth of the concentration in the soil in which the plants grow, that there is no downward movement in the soil beyond the root zone, and that there is no runoff. Observations on the behavior of fallout plutonium indicate that these assumptions probably lead to an overestimation of the amount entering man through ingestion.

Estimates made of potential health consequences that might occur as the result of the entry of these exceedingly small quantities of alpha emitters into man over the many generations they will exist in the environment provide a substantial basis for believing that potential health consequences associated with the persistence and accumulation of transuranic elements released by the LMFBR fuel cycle would be small in comparison with many risks that are accepted by society.

Although a substantial data base exists, some of the information required to make precise predictions is not available. In the absence of complete data, conservative estimates have been made using the best available information. Therefore, there is a high probability that estimated potential health consequences are overestimated. There are no observations in man or in animals directly relevant to estimating potential health effects at the exceedingly low levels of exposure being considered. The estimates have had to assume that health consequences are directly proportional to the quantity and residence time of alpha-emitters in the organs of man and have probably led to overstatement of the maximum potential health consequences. The estimates are intended solely to provide some perspective on the long-term health consequences that might accrue to large populations as the result of the persistence and accumulation of transuranic elements released by the LMFBR fuel cycle.

4.1 INTRODUCTION

The breeder reactor power plant, fuel fabrication plant, fuel reprocessing plant, radioactive waste management, and transportation of fuel and radioactive material are the major elements of the LMFBR fuel cycle. This section describes the projected environmental effects that would directly result from construction and operation or other activities associated with model plants in the LMFBR fuel cycle. Each element of the fuel cycle is evaluated separately. Portions of the evaluations on a model plant basis in this section are used in the subsequent summation of effects in Sections 9 and 11. The model LMFBR fuel fabrication plant, the model LMFBR reprocessing plant, the model LMFBR power plant, and the waste-storage and disposal facilities are considered to be located at separate and independent sites, with the transportation system being the connecting link. Plutonium toxicity is discussed in Section 4.7 and Appendix II.G.

It is not possible at this early stage of the LMFBR Program to evaluate in a precise manner the environmental effects of the LMFBR fuel cycle operations. Many of the effects of normal operation and consequences of accidents or abnormal occurrences are strongly dependent on the site and on the specific facility design. Atomic Energy Commission regulations require that, prior to the issuance of a construction permit or an operating license, the applicant submit to the Commission a Safety Analysis Report (SAR) and an Environmental Report. The purpose of the SAR, which receives a rigorous review by the regulation staff, is threefold. First, it requires the applicant, from the earliest stages of design and site selection, to give careful consideration to assuring that the public health and safety are protected and that potentially hazardous situations are recognized and corrected. Second, the restrictions and limitations contained in the SAR and the accompanying Technical Specifications are formulated to bound the operation in such a way as to minimize any hazards which may arise. Finally, the SAR provides an evaluation of the consequences of various abnormal events, together with estimates of the probability that they may occur. Evaluation of the transportation operations in the LMFBR fuel cycle obviously does not involve specific sites. A safety analysis is required for each shipping container, and AEC regulations (10 CFR 71) prescribe the safety requirements for shipping.

4.1.1 Brief Description of LMFBR Fuel Cycle

The nuclear-fuel cycle for LMFBRs is shown in Figure 4.1-1. The initial feed materials consist of plutonium (obtained from the reprocessing of fuel from light water reactors) and depleted uranium (which results from the enrichment of the U-235

LMFBR NUCLEAR FUEL CYCLE
Figure 4.1-1

content of natural uranium). The plutonium would be converted to an oxide (PuO_2) at the reprocessing plant. The uranium, as uranium hexafluoride (UF_6), would be converted to an oxide (UO_2) at the fuel-fabrication plant. Plutonium dioxide and uranium dioxide would be combined at the fuel-fabrication plant and fabricated into mixed oxides for the core fuel. Uranium dioxide would also be fabricated into pellets for the axial and radial blankets of the reactor. After exposure in the reactor, the irradiated fuels would be stored at the reactor for up to 360 days. (A major objective of the current fuel recycle development program is to reduce the time between discharge of the fuel from the reactor and its return to the reactor as refabricated fuel elements.) After storage at the reactor, the irradiated fuel is shipped in shielded casks to the reprocessing plant, where the plutonium, uranium, and fission products would be chemically separated. The separated fission products would be shipped to a Federal waste-storage facility, and the plutonium would be recycled as fuel. The recovered uranium could be either stored or recycled into the mixed oxide or blanket UO_2 . Depleted uranium would be used for makeup for the uranium that is either converted to plutonium in the reactor, lost as scrap in the processes, or stored. A material flowsheet for this fuel cycle is given in Figure 4.1-2.

The model LMFBR power plant would have a thermal power output of about 2500 MWt and a net electrical output of 1000 MWe, corresponding to an overall plant efficiency of 40%. The reactor heat-generation system (Figure 4.1-3) consists of the reactor core, control rod drives, and reactor vessel and associated pumps, piping, instrumentation, and equipment. The heat transport system consists of sodium pumps, piping, intermediate heat exchanger, steam generators, turbine generator, and condenser. Heat generated in the reactor core is transferred by forced circulation of liquid sodium primary coolant to intermediate heat exchangers and then through an essentially nonradioactive secondary sodium coolant system to steam generators in which superheated steam is produced. This steam drives a turbine generator to generate electrical power. Waste heat is rejected either to a nearby river, lake, or ocean by once-through cooling or to the atmosphere in wet or dry cooling towers.

The model fuel-fabrication plant would fabricate five metric tons of uranium and plutonium oxide fuel per day or 1500 metric tons/year. This production capacity would supply the annual fuel requirements for about eighty 1000-MWe LMFBRs. In this plant, the oxides of uranium and plutonium would be combined and formed into dense solid pellets of mixed oxide. Some of the uranium oxide would be formed into dense solid pellets that do not contain plutonium; these are called blanket pellets. Both types of pellets would be loaded into stainless steel tubes, which would then

MATERIAL FLOWSHEET FOR 1000-MWe LMFBR

Figure 4.1-2

TYPICAL LMFBR POWER PLANT

Figure 4.1-3

be seal welded to contain the fuel material; the loaded tubes (fuel pins) would be assembled into fuel elements, which would be shipped to the reactor. The fabrication plant is assumed to be located near the center of a 1000-acre site in a rural area. An "average" meteorology is assumed (Appendix II.I) for the site, and a population of one million people is assumed to be uniformly distributed within 50 miles of the plant.*

The fuel-reprocessing plant would have the same production capacity, type of location, meteorology, and population distribution as that assumed for the fuel-fabrication plant. Spent fuel elements from the reactor would be received at the fuel-reprocessing plant, where they would be mechanically disassembled and the fuel and blanket material would be dissolved in nitric acid. The solution would be processed to separate uranium, plutonium, and the fission products. The plutonium would be converted from a nitrate to an oxide and shipped to the fuel-fabrication plant, while the uranium would be stored or converted to an oxide and shipped to the fuel-fabrication plant.

Transportation within the LMFBR fuel cycle includes shipments of unirradiated fuel material (such as UF_6 and UO_2) and irradiated material (such as spent fuel, PuO_2 and wastes from fuel-fabrication, reactor, and reprocessing plants). Many such shipments will be made by truck, with appropriate restrictions regarding shipping conditions (see Appendix II.M). Where heavy packages are involved (e.g., a spent fuel cask weighing about 150,000 lb), shipments will be made by rail.

Radioactive wastes generated during LMFBR cycle operations will be classified as high level and other than high level. Liquid high-level wastes are produced during fuel reprocessing and AEC regulations require that these wastes be converted to a stable solid material within five years after separation in the fuel-reprocessing step and that they be encapsulated and shipped to a Federal repository within ten years of production. Other-than-high-level wastes will be mostly solid and gaseous wastes. They include miscellaneous plant waste materials contaminated with low levels of radioactivity; solidified tritium, iodine, and sodium fuel cladding hulls; and bottled noble-gas fission products. Except for the noble-gas fission products and the alpha wastes, most wastes classified as other-than-high-level waste will be packaged for burial in a licensed burial facility. The technology and facilities

*The size of a site, the meteorological characteristics of the area, and the population density surrounding a site will probably vary over a substantial range when the actual plants are built in the future. Based upon locations of plants that are currently licensed for plutonium fabrication (see Table 4.3-1), these assumptions regarding future sites appear to be reasonable.

established for management and disposal of radioactive waste from the LWR nuclear power industry will be directly applicable to the management of radioactive waste generated in the LMFBR fuel cycle.

A retrievable surface storage facility (RSSF) will be used for storage of solidified commercial high-level radioactive wastes from the LWR and LMFBR fuel cycles.* The RSSF will comprise facilities for receiving and inspecting packaged wastes from LMFBR fuel reprocessing plants and facilities for safely storing these wastes. The facility will be designed to hold safely, for at least 100 years, all of the commercial high-level waste produced by both the LMFBR and LWR fuel cycles through the year 2000. A site for the RSSF has not yet been selected. For purposes of this Statement, the RSSF was assumed to be located within the Hanford reservation in the southeastern part of the State of Washington.**

The model land burial facility for beta-gamma wastes consists of about 100 acres of land located in a rural, sparsely settled area. The prime operations at this licensed commercial land burial facility are the receipt, temporary storage, and burial in trenches of packaged radioactive wastes. The site characteristics are such that the groundwater is well below the bottom of the deepest trench and the soil provides good ion exchange characteristics to minimize percolation of radioactivity. No significant environmental impacts are anticipated due to operation of the model land burial facility.

A conceptual noble gas storage facility (NGSF) for the noble gases recovered during fuel reprocessing was developed for use in this Statement. The facility would include areas for receiving, handling, and inspecting noble-gas cylinders and for transferring and storing the cylinders in the final storage area. The concept is based upon holding safely for at least 50 years all the noble gas waste produced by commercial nuclear reactors through the year 2020.

The environmental impacts of construction, operation, and postulated accidents for the various model plants in the LMFBR fuel cycle are summarized in Section 8. Detailed discussions concerning these impacts are presented in Section 4.2 for the

*A conceptual alpha waste storage facility is also described in Section 4.6. This facility would be suitable for receiving and sorting the incoming waste shipments and would provide high integrity interim storage warehouses that can be expanded incrementally as needed. The concept is based on safely holding alpha wastes for a period of up to 100 years until a permanent disposal concept and site are developed.

**See WASH-1539, Draft Environmental Statement--"Management of Commercial High Level and Transuranium Contaminated Wastes," September 1974 for additional information on the RSSF.

model LMFBR power plant (1000 MWe), Section 4.3 for the model fuel-fabrication plant (5 metric tons of uranium and plutonium per day), Section 4.4 for the model fuel reprocessing plant (also 5 metric tons/day), Section 4.5 for transportation of the fuel cycle (that required for one 1000-MWe power plant and associated fuel cycle for one year of operation), and Section 4.6 for radioactive waste management (same basis as transportation). Additional detailed information on these topics is presented in the appendices.

4.1.2 Availability of Plutonium and Uranium

The use of enriched uranium in the LMFBR fuel cycle would be required only if the plutonium available as initial fuel is insufficient for an expanding LMFBR economy. The sufficiency of plutonium is determined by the quantity produced as excess in the LWR economy, by the growth rate and timing of the LMFBR economy, and by the doubling time of the plutonium inventory due to the breeding gain in the LMFBRs. In the projected nuclear economy discussed in Section 11, sufficient plutonium is expected to be available during the period covered by this Statement.

The availability of depleted UF_6 in the current AEC stockpile is not considered in this Environmental Statement, although this material could be used for fuel for LMFBRs for central-station generation of electricity.¹

The quantity of depleted uranium projected to be produced as tails from enrichment of natural uranium for LWRs and HTGRs by the year 2000 is shown in Table 4.1-1.² The data of Table 4.1-2 show that sufficient depleted uranium would be available by the year 2000 to supply all LMFBRs projected to be built by the year 2020, with enough remaining at that time to provide the uranium refueling requirements of these LMFBRs for the next 565 years. In addition, depleted uranium production after the year 2000 would add to available stocks. Thus, depleted uranium is clearly forecast to be available in quantities substantially greater than needed.²

After reprocessing, the recovered uranium from both the core and the blanket of an LMFBR can be recycled or stored for later use. In Table 4.1-2, recycling is assumed in order to show more clearly the net consumption of uranium. Whether the discharged uranium will in fact be recycled or will be stored and replaced by drawing from depleted uranium stockpiles will be determined by considerations of process economics at the time.

As discussed in Section 11.2.3.3.7, introduction of LMFBR power plants is assumed to occur at a sufficiently rapid rate that they are net consumers of plutonium for about

Table 4.1-1
PRODUCTION OF DEPLETED UF₆ FORECAST
FOR THE YEARS 1973-2000

Year	Fabrication load, metric tons of U				UF ₆ , metric tons of U	
	LWR UO ₂	HTGR fissile	FBR		Conversion	Depleted ^a
			Mixed oxide	Blanket		
1973	640	0	0	0	6,004	5,364
1974	1,390	0	0	0	6,900	5,510
1975	1,770	0	0	0	11,190	9,420
1976	1,740	0	0	0	11,780	10,040
1977	2,400	0	0	0	14,450	12,050
1978	3,000	0	0	0	21,170	18,170
1979	3,900	1	0	0	23,980	20,079
1980	4,900	2	9	11	31,280	26,378
1981	5,600	3	0	0	34,300	28,697
1982	6,000	6	2	2	30,700	24,694
1983	5,200	3	3	3	34,600	29,397
1984	5,900	10	3	3	37,600	31,690
1985	7,100	14	4	3	48,200	41,086
1986	8,700	17	4	3	54,300	45,583
1987	9,300	20	42	51	59,400	50,080
1988	10,400	25	14	12	67,900	57,475
1989	11,500	29	75	86	74,100	62,571
1990	12,300	33	128	147	81,000	68,667
1991	13,300	38	200	230	88,500	75,162
1992	14,300	42	290	320	94,400	80,058
1993	15,300	47	410	450	101,700	86,353
1994	16,100	51	550	590	108,500	92,349
1995	17,200	55	740	790	114,700	97,445
1996	18,200	59	930	970	121,300	103,041
1997	18,900	63	1,170	1,190	127,000	108,037
1998	19,600	67	1,420	1,430	130,600	110,933
1999	20,100	70	1,570	1,520	133,800	113,630
2000	20,400	73	1,870	1,780	136,700	116,227
Total	275,140	728	9,434	9,591	1,806,054	1,530,186

^aDepleted UF₆ = (conversion UF₆) - (LWR UO₂) - (HTGR fissile).

Source: Forecasting Branch, Office of Planning and Analysis, USAEC, *Nuclear Power Growth 1974-2000*, WASH-1139 (74), February, 1974.

Table 4.1-2

DEPLETED URANIUM CONSUMPTION IN AN LMFBR ECONOMY

		Depleted Uranium (metric tons U)
U Throughput (Fuel and Blanket) per GWe-yr Generation		21.1 ^a
Net U Consumption per GWe-yr Generation (includes allowance for recycle losses @ 2% of throughput)		1.6 ^a
Installed LMFBR Capacity Year 2020	2,250 GWe ^b	
LMFBR Electric Generation Year 2020	1,650 GWe-yr ^b	
Cumulative LMFBR Electric Generation through 2020	27,700 GWe-yr ^b	
Net LMFBR U Consumption through 2020		36,320
Depleted U Accumulated by Year 2000 (Table 4.1-1)		1,530,186
Excess		1,493,855
Period Over Which Excess Would Support LMFBR Economy at Year 2020 Rate	565 years	

^aDerived from Table 10.1-1, Section 10.

^bFrom Table 9.1-15, Section 9.

GWe = gigawatt electric = 1000 megawatts = 1,000,000 kilowatts.

20 years following initial introduction. During this period, fewer LWRs are introduced than in the case where no breeders are introduced. The LWRs which are introduced, however, are mostly uranium-fueled producers of plutonium. As the breeders approach self-sufficiency and become net producers of plutonium (about 2005-2010), the number of uranium-fueled LWRs drops rapidly. However, relatively large numbers of plutonium-fueled LWRs could then be introduced, using the surplus plutonium produced by the breeders.

REFERENCES FOR SECTION 4.1

1. Federal Register 32: 289 (March 16, 1971).
2. U.S. Atomic Energy Commission, "Nuclear Power Growth 1974-2000," USAEC Report WASH-1139(74), February 1974.

4.2 LMFBR POWER PLANTS

4.2.1 Introduction

To assess the environmental impact of future LMFBR power plants, the concept of a model plant on a typical site has been adopted. The site, described in Section 4.2.2, is typical of sites presently used for light water reactor power plants. The model plant is a 1000-MWe LMFBR with characteristics similar to those described in the "1000-MWe LMFBR Follow-on Study Reports,"¹ modified where more recent information is available. In the absence of a detailed design, alternatives for various plant systems are described and ranges of parameters are presented. The model plant described in Section 4.2.3 is considered to be technically feasible using reasonable extrapolations of today's technology.

The environmental impacts expected from construction of an LMFBR will in general be no different from those incurred in construction of an LWR and not greatly different from that of construction of a fossil-fueled plant. In Section 4.2.4 these impacts are described. Means are available for controlling such impacts to within currently acceptable limits.

The environmental effects of routine operation of a 1000-MWe LMFBR are assessed and discussed in Section 4.2.5. This assessment covers the impacts on land, water, and air expected from the various alternative plant design features. The radiological impacts on man from radioactive materials released during normal operation of the plant are described in Section 4.2.5.3 for the typical site with an average population distribution.² This same section also includes a discussion of radiological effects on biota other than man. There are no environmental impacts resulting from routine operation of the plant that cannot be controlled to within currently acceptable limits. It is the policy of the AEC to require that operators of nuclear plants conduct a program of monitoring the environment surrounding the plant to detect unexpected effects. Generalized requirements for such a monitoring program for an LMFBR are presented in Section 4.2.6. In Section 4.2.7, analyses of the expected effects of a spectrum of potential accidents are presented.

AEC regulations require that peaceful uses of atomic energy not result in undue risk to the health and safety of the public. To meet this requirement, the AEC has established a regulatory process which involves a two-step procedure for licensing a nuclear power plant. The first step authorizes the utility applicant to construct the nuclear facility by virtue of issuance of a construction permit. The second step authorizes operation of the plant on issuance of an operating license. In

addition, the AEC conducts a continuing review and surveillance of operating nuclear power plants to assure compliance with conditions imposed in the operating license.

Each LMFBR power plant will be licensed according to this established procedure, which requires that the electric utility owner and operator of the plant submit to the AEC a Preliminary Safety Analysis Report (PSAR) and an "Applicant's Environmental Report--Construction Permit Stage." The PSAR must contain evidence that the proposed plant can be built and operated without undue risk to the health and safety of the public. The Environmental Report must include detailed analyses of potential environmental impacts that might result from construction and operation of the plant; it must also present considerations of other factors as required by NEPA.³ These documents are reviewed by the AEC, and suitability of the plant on the proposed site is judged from the standpoints of safety and environmental impact; the latter assessment is reported in an environmental statement as required in 10 CFR 51.

For each power reactor construction permit application submitted to the AEC, the AEC's regulation staff conducts a safety evaluation. The Advisory Committee on Reactor Safeguards (ACRS) also independently conducts a review of the application and makes recommendations to the AEC regarding the safety of the facility.

After these reviews are completed, a three-man Atomic Safety and Licensing Board conducts a public hearing, in accordance with the Administrative Procedure Act and AEC regulations, and full public participation is afforded. After the conclusion of the hearing, an initial decision is issued by the Board which is then subject to review by the Atomic Safety and Licensing Appeal Board and, in instances specified by AEC regulations, by the Commission itself. If the Licensing Appeal Board's decision is favorable, a construction permit is issued by the AEC's Director of Regulation. Later, when an application for an operating license is filed, the regulation staff and the ACRS again conduct a comprehensive safety review for the purpose of assessing the results of research and development completed during the period between reviews, updating previous analyses and conclusions, assuring that safety features have been properly incorporated into the as-built plant, and assuring that the facility has been constructed in compliance with the construction permit, the Atomic Energy Act, and its implementing regulations. Under present provisions of the Atomic Energy Act, an opportunity for hearing is afforded on the application for an operating license. As noted above, an updated Environmental Report is also filed by the utility when applying for an operating license, and an additional environmental statement is prepared by the AEC.

Although present regulations are directed primarily to light water reactor power plants, the AEC will promulgate regulations applicable to LMFBRs at the appropriate time. Applicants to build future LMFBR power plants will be required to meet the requirements and criteria set forth in such regulations before construction permits are granted.

4.2.2 Siting Considerations

4.2.2.1 Introduction

"Site" represents the real estate which is owned, leased, or otherwise controlled by the licensee for construction and operation of a nuclear power station. The terms "LMFBR power plant" or "nuclear power station" refer to the complex including one or more reactors and turbine generating units and auxiliary facilities, equipment, and structures necessary to maintain and operate the station.

Decisions relating to the siting of LMFBR power plants are expected to be guided by practices, guidelines, and criteria that will have been developed and established through experience gained in the construction and operation of nearly one full generation of large LWR nuclear power plants and siting of LMFBR demonstration power plants. It is not anticipated that major changes will be made in the guidelines currently in use.

The choice of a site for a specific LMFBR power plant will be determined by a number of economic, safety, and engineering factors as well as consideration of the acceptability of environmental impacts caused by the construction and operation of the plant. LMFBR siting must be responsive to the public's concern for the quality of the environment.

Factors of importance in selecting a specific site include: proximity of load centers; availability of access and transportation such as highway, rail, or waterway; adequacy of geological conditions for engineering and safety requirements; suitability of meteorological and hydrological conditions; and anticipated environmental impacts that would be caused by the construction and operation of the plant.

Responsibility for investigating prospective sites for a nuclear power plant and selecting and procuring ownership of a specific site normally rests with the utility. However, authority for approving construction and operation of the plant at the site rests with the AEC and other governmental bodies, both state and

Federal. Such authority is exercised through the issuance of permits and licenses following routine and extensive considerations and hearings by official persons, boards, and commissions. Requirements of such bodies are also discussed in this section.

Existing plant sites may also be used for construction of additional capacity in the form of LMFBFR plants if they are economically feasible and if they meet safety and environmental requirements. If insufficiently large, they might be expanded by acquisition of adjacent lands. Such new plants would require approval of governmental bodies even though they were located on existing approved sites. Pre-existing plants on a site could be either fossil- or nuclear-fueled, and assessment of acceptability of the additional plant will take into account the existing plant and its interrelationship with the operation of the proposed new plant on the basis of both safety and environmental impacts. The expanded use of existing transmission lines or addition thereto will also be considered before approval for the additional plant is given. For the purpose of this Statement, however, the site for the LMFBFR plant is taken as not previously used.

Selection of future sites for power plants may be expected to be associated with comprehensive land-use plans which may be developed for various areas of the country. Coordinated planning on the part of local and national governmental bodies and the utilities will probably be necessary to identify sites that best satisfy economic, safety, and environmental requirements.

Areas suitable for the location of LMFBFRs may be found in any part of the country. The availability of natural resources for the disposal of waste heat from the plant to the environment is of particular importance in assessing the suitability of proposed sites. Location of the plant along the shores of oceans, lakes, or rivers permits consideration of these bodies of water for plant cooling. However, assessment of environmental impacts on the use of such waters may dictate that alternative cooling methods, such as cooling towers, cooling ponds, or spray canals be used. In such cases, the site might also be located away from major water bodies. Dry cooling towers are not nearly as economical as once-through or evaporative cooling.⁴ However, by the use of dry cooling towers, sites could be located where only minimal quantities of water are available.

A number of studies have been made by various groups on the generalized problem of power plant siting. Among them is one that was sponsored by the Energy Policy Staff of the President's Office of Science and Technology. Cooperating in this study were

the U.S. Atomic Energy Commission, Federal Power Commission, U.S. Department of the Interior, U.S. Department of Health, Education, and Welfare (HEW), Rural Electrification Administration, and Tennessee Valley Authority. Also assisting were the National Association of Regulatory Utility Commissioners and the state utility commissions throughout the country in providing a survey of the important work of the states on these problems. The report,⁵ published in January 1969, recommends that plans for power plant siting consider the following:

- (1) compliance with safety criteria for nuclear plants as prescribed by AEC;
- (2) compliance with air pollution criteria and standards as established by the states and the National Air Pollution Control Administration of HEW;*
- (3) compliance with the water quality standards for thermal effects as established by the states and the Federal Water Pollution Control Administration (FWPCA) of the U.S. Department of the Interior;*
- (4) development of opportunities for public recreation at plant sites;
- (5) consideration of aesthetic values and attention to the appearance of power plant facilities and associated transmission lines;
- (6) recognition of rural development considerations in plant siting;
- (7) siting and lead-time requirements for reliability of service;
- (8) impact on defense preparedness of particular sites and power plant capacities;
- (9) the routing of associated transmission lines and the problems of rights-of-way at various alternative plant locations;
- (10) assuring that the plant will be of sufficient size to meet regional loads including mutually agreeable arrangements for meeting the bulk power needs of the small utilities;
- (11) prospects for combining power plants with other purposes such as desalting plants, industrial centers, and even new cities.

These are all considerations over and above such basic requirements as sufficient land, the availability of transmission lines, fuel, and the many other factors that every utility considers before deciding on the size, type, and location of a power plant.

*The Department of the Interior and Department of Health, Education, and Welfare functions indicated were transferred to the Environmental Protection Agency by Reorganization Plan No. 3 of 1970.

4.2.2.2 Site Requirements

4.2.2.2.1 Proximity to Load Centers

Nuclear power plants do not suffer economically by being located at a distance from the natural source of fuel. More important is their location relative to load centers and factors related to safety and environmental considerations. Although it has generally been the practice to locate steam-electric generating plants near load centers because of economic factors, most nuclear plants have been located some distance from population centers. However, as more experience is gained in the design, construction, and operation of nuclear plants, the use of locations nearer to population centers is expected to be possible,⁵ although restrictions and requirements stated below will dictate permissible locations.

4.2.2.2.2 Availability of Water

Of major importance in the selection of a power plant site is the need to dispose of large quantities of waste heat to the environment. For example, a single 1000-MWe LMFBR plant will reject up to five billion Btu/hr. Environmental considerations may be overriding and may dictate that such once-through systems may not be used. Alternative heat sinks must then be considered, such as cooling towers, cooling ponds, or spray canals.

4.2.2.2.3 Area Requirements

Present LWR nuclear power plants of 1000 MWe capacity require 100 to 400 acres of land area to meet construction and operational requirements. This excludes area for cooling lakes or ponds but includes cooling towers, switch yards, parking areas, roads, railroad spur, and other facilities. If a cooling pond is used for the dissipation of waste heat, an additional 1000 to 2000 acres will be required for each reactor plant unit, depending on specific site location and prevailing meteorological conditions.⁶ Area requirements for an LMFBR plant are not expected to be different from the above. Multiple units of the same capacity could be accommodated by the addition of area, not including cooling ponds or lakes, on the order of tens of acres per additional unit.

4.2.2.2.4 Geology and Seismology

Safety and engineering requirements dictate the consideration of geological factors in the selection of specific sites. Geological conditions must be such that a satisfactory foundation is available for the plant structure.

Site selection should consider the presence of faulting. Preferably, a site should have no known faults and be sufficiently removed from capable faults so as to preclude or minimize plant damage in the event of an earthquake. AEC regulations⁷ require that no nuclear power plant be located closer than one-fourth mile from the surface location of a known capable earthquake fault.

4.2.2.2.5 Meteorology and Hydrology

Knowledge of meteorological conditions at the reactor site is important in determining the behavior of radioactive materials which might be released in the operation of the LMFBR. Meteorological parameters at a prospective site must be identified on a seasonal and annual basis from data recorded either on the site or at nearby points. Such information is also necessary to assess the environmental impacts of cooling ponds and spray canals.

Hydrological studies are necessary to assure that adequate plant cooling water is available. This might be a considerable quantity where once-through cooling is utilized or a lesser quantity (although still significant) where cooling ponds, spray canals, or wet cooling towers are used.

4.2.2.2.6 Historical, Archeological, and Cultural Resources

Areas valued for their historic, scenic, cultural, or natural significance should be avoided in selecting a plant site. Statutory requirements such as the 1966 National Historic Preservation Act extend sanctioned protection of historic sites in the United States and provide for funding of state and local groups to study and identify such sites. The National Register of Historic Places published in 1969 identifies these sites. Sites are listed periodically, as in Part II of the Federal Register of February 28, 1973, and the list is updated by monthly notices in the Federal Register. National landmarks were identified in the Federal Register of January 29, 1972.

4.2.2.2.7 AEC Siting Criteria for Licensing

Nuclear power plants must be licensed by the AEC before they may be constructed or operated. Review requirements for such licensing are essentially confined to protecting the health and safety of the general public and to minimizing detrimental impacts to the environment.

Part 100 (Reactor Site Criteria) of Title 10, Code of Federal Regulations, and subsequent revisions and amendments thereto,⁷ is the primary regulation for evaluating the acceptability of proposed reactor sites on the basis of health and

safety. In regard to land use and population characteristics of a site, this Part requires an applicant for a construction permit to identify the following:

- (a) "Exclusion area" which is 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. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result.
- (b) "Low population zone" which is the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific area, or instructed to take shelter, on a timely basis will depend on many factors such as location, number and size of highways, scope and extent of advance planning, and actual distribution of residents within the area.
- (c) "Population center distance" which is the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents.

In 10 CFR 100 the AEC also has listed factors to be considered when evaluating sites. These are repeated as follows:

Factors considered in the evaluation of sites include those relating both to the proposed reactor design and the characteristics peculiar to the site. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in release of significant quantities of radioactive fission products. In addition, the site location and the engineered features included as safeguards against the hazardous consequences of an accident, should one occur, should insure a low risk of public exposure. In particular, the Commission will take the following factors into consideration in determining the acceptability of a site for a power or testing reactor:

- (a) Characteristics of reactor design and proposed operation including:
 - (1) intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;
 - (2) the extent to which generally accepted engineering standards are applied to the design of the reactor;

- (3) the extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental release of radioactive materials;
 - (4) the safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur.
- (b) Population density and use characteristics of the site environs, including the exclusion area, low population zone, and population center distance.
 - (c) Physical characteristics of the site, including seismology, meteorology, geology, and hydrology.
 - (1) Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," describes the nature of investigations required to obtain the geologic and seismic data necessary to determine site suitability and to provide reasonable assurance that a nuclear power plant can be constructed and operated at a proposed site without undue risk to the health and safety of the public. It describes procedures for determining the quantitative vibratory ground motion design basis at a site due to earthquakes and describes information needed to determine whether and to what extent a nuclear power plant need be designed to withstand the effects of surface faulting.
 - (2) Meteorological conditions at the site and in the surrounding area should be considered.
 - (3) Geological and hydrological characteristics of the proposed site may have a bearing on the consequences of an escape of radioactive material from the facility. Special precautions should be planned if a reactor is to be located at a site where a significant quantity of radioactive effluent might accidentally flow into nearby streams or rivers or might find ready access to underground water tables.
 - (d) Where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards.

4.2.2.2.8 Other Regulatory Requirements

In addition to AEC criteria for siting nuclear power plants, there are other Federal, state, and local standards which must be considered in determining the adequacy of prospective sites. Among other Federal agencies responsible in this matter are: the U.S. Environmental Protection Agency (EPA)⁸ which is responsible for controlling air and water pollution, drinking water quality, solid wastes, environmental radiation and noise; the Federal Power Commission⁹; the Corps of Engineers, U.S. Department of the Army¹⁰; the U.S. Department of the Interior⁵;

and the U.S. Department of Commerce, National Oceanic and Atmospheric Administration. State agencies also are concerned with many of the considerations discussed above, as well as zoning, public convenience and necessity, pollution control, wildlife, and recreation.⁵

The Federal Water Pollution Control Act Amendments of 1972 provide for the states to reduce and eliminate water pollution within the framework of a national program. If the states do not or cannot fulfill their obligations under the law, the U.S. Federal Government, through the EPA, is empowered and directed to take action. The utilities will be required to adhere to FWPCA criteria and to requirements of discharge permits issued by state or Federal agencies.

4.2.2.3 Transmission Requirements

Decisions related to the selection of an LMFBR plant site must take into consideration the attendant rights-of-way that will be required for the transmission lines from the plant. Economic and environmental impact comparisons will be made of the alternative routing available at each site under consideration. Impacts due to the transmission lines may be expected to vary depending upon the sites and routes.

The system must have adequate capacity to provide additional support for any load areas within a region requiring it in emergencies, to be able to transfer capacity and energy within a region and possibly between regions to meet local power shortages when they occur, and to exchange energy on a regional and interregional scale to achieve economies in capital and operating costs. In the early days of electric power system construction, generating plants were located adjacent to major loads and there was little long-distance transmission of electricity. As more efficient transmission became feasible through the development of high voltage systems, plants could be located at a distance from load centers when economic or other considerations dictated this. Present-day power plants of advanced design are utilizing transmission lines designed for 765-kV operation. Such a line requires a right-of-way 280 ft in width. A station with a capacity of 3000 MWe will require two such lines with a right-of-way width of 430 ft.⁵ A 500-kV system will require about two-thirds of this width, while a 1500-kV system will be about twice as wide.

Selection of routings for transmission lines will consider many of the same factors as for plant siting. Safety requirements are different so that AEC regulatory criteria do not apply in this case. However, environmental effects are reviewed by the AEC in its assessment of impacts.³ Such review includes that of impact on the

soils caused by construction and maintenance of transmission lines; effect on river and stream banks where the line crosses them; impact on terrestrial and aquatic ecosystems, particularly during the construction phase; economic impact that would result from the purchase of right-of-way land and possible changes to the use of the land; electrical radiation effects including electromagnetic interference with communication systems, induced voltages in paralleling systems (such as railroads), audible noise emission, and ozone concentration; and visual aesthetic values. This is in addition to factors common to plant site and transmission lines. Specific routing of the transmission lines will be very sensitive to the requirements of reviews. Changes to the routing may result if it is found that alternative routings present more acceptable impacts to the environment.

4.2.3 Model Plant

4.2.3.1 Basis for Model Plant Description

In 1967-1969, the AEC sponsored independent, parallel studies by five reactor manufacturers--Atomics International (AI), Babcock and Wilcox (B & W), Combustion Engineering (CE), General Electric (GE), and Westinghouse--with the prime objective of identifying the research and development needed for the practical realization of commercial electric power generation by LMFBR power plants in the period 1980-1990. Each of these "1000-MWe LMFBR Follow-on Studies"* included the selection and development of a preliminary, company-favored reference concept for a 1000-MWe LMFBR power plant to be used as the vehicle for identifying the research and development needed. The concepts were intended to represent the manufacturers' then-current thinking as to the plants they would commercially market in 1975-1985, reflecting advancements in technology that the manufacturers anticipated could be achieved by that time. The scope of study encompassed the entire plant for all manufacturers except CE, whose scope was limited to the reactor heat-generation system portion of the nuclear steam supply system. By way of introduction, a brief review of key features in the five manufacturers' plant concepts is presented.

AI, B & W, and GE used mixed-oxide fuel; Westinghouse and CE used carbide. All except AI specified fuel elements vented to the reactor coolant. AI used sealed elements but provided building space for the additional fission gas processing equipment that would be required for vented fuel.

*The name is derived from the fact that the 1967-1969 studies were a "follow-on" of more fundamental 1000-MWe reactor investigations conducted in 1963-1964.

For their reference primary system configurations, AI, CE, and Westinghouse chose the loop-type system arrangement. B & W and GE chose the pool-type arrangement.

Conventional containment structures were used by all manufacturers except GE. The GE design provided for a low leakage refueling hot cell in a confinement-type structure. Any leakage into the confinement structure would be discharged through a filter-scrubber system and up a stack.

All contractors used reheat in their steam cycle designs except Westinghouse, who used a nonreheat cycle based on the results of their trade-off study of reheat versus nonreheat systems. The concepts are described in the manufacturers' follow-on study reports.¹¹⁻¹⁵

The ensuing description of a model plant draws upon the 1000-MWe follow-on study reference concepts. However, it must be recognized that those studies were limited in scope and that much of the detailed information needed to conduct a precise environmental analysis of a commercial LMFBR power plant was not developed. The actual designs of commercial plants to be built beginning in the next decade will be strongly influenced by the experience gained from the Fast Flux Test Facility (FFTF) and demonstration plant projects. The results of ongoing and planned research and development programs, foreign plant experience, revised environmental and safety standards, and other complex economic and societal factors will also influence plant design. For these reasons it should not be assumed that the model plant represents a design goal; rather, it merely represents an envelope of the currently available design thinking for commercial plants for use as a reference in this Environmental Statement.

4.2.3.2 Model Plant Description

The model plant has a thermal power output of about 2500 MWt and a net electrical output of 1000 MWe. Heat generated in the reactor core is transferred by forced circulation of liquid sodium primary coolant to intermediate heat exchangers and then through a nonradioactive secondary sodium coolant system to steam generators in which superheated steam is produced. This steam drives a turbine generator to generate electrical power. Waste heat released by condensation of exhaust steam from the turbine is rejected either to a nearby river, lake, or ocean by once-through cooling or to the atmosphere in wet or dry cooling towers. A simplified flow diagram for the plant is shown in Figure 4.2-1 and the principal plant parameters are listed in Table 4.2-1.

MODEL PLANT FLOW DIAGRAM
Figure 4.2-1

Table 4.2-1
MODEL PLANT DESIGN PARAMETERS

Overall Plant	
Net electrical power	1000 MWe
Reactor thermal power	~2500 MWt
Reactor Heat Generation System	
Core configuration	Quasi-cylindrical
Fuel height	30-45 in.
Axial blanket height--upper and lower	12-18 in.
Number of fuel elements (pins/assembly)	150-350
Fuel element outside diameter	0.2-0.3 in.
Fuel element cladding thickness	0.01-0.2 in.
Fuel assembly duct configuration	Hexagonal
Fuel assembly width (across flats)	5.5-6.5 in.
Fuel assemblies	200-300
Blanket assemblies	120-140
Control absorber assemblies	15-30
Fuel	Mixed PuO ₂ -UO ₂
Fuel cladding (stainless steel)	Type 316 or 304
Blanket	Depleted UO ₂
Bonding	Helium
Control absorber	B ₄ C or Tantalum
Peak neutron flux	10^{15} - 10^{16} n/(cm ²)(sec)
Total fissionable inventory	1800-2800 kg
Breeding ratio	~1.2
Control drive type	Electromechanical
Reactor vessel diameter (loop-type system)	20-30 ft
Primary tank diameter (pool-type system)	~50 ft
Reactor vessel or primary tank height	45-50 ft
Refueling fraction: Fuel	1/3-1/2
(~1 year interval) Radial blanket	1/6-1/3
Heat Transport System	
Primary coolant system arrangement	Pool or loop
Number of coolant loops--primary/secondary	3/3
Reactor outlet temperature	1000-1050°F
Reactor inlet temperature	~800°F
Primary coolant flow (sodium)	75-100 x 10 ⁶ lb/hr
Secondary coolant hot-leg temperature	~1000°F
Secondary coolant cold-leg temperature	650-750°F
Secondary coolant flow per loop (sodium)	50,000-70,000 gpm
Primary and secondary cover gas	Argon
Main steam flow	6.9-7.4 x 10 ⁶ lb/hr
Main steam pressure	1450-2400 psig
Main steam temperature	900°F
Reheat pressure	500-550 psig
Reheat temperature	900-1000°F
Condensate pressure and temperature	1.5 in. Hg and ~90°F
Feedwater temperature	~500°F
Reactor Containment Building	
Diameter	225-250 ft
Height	100-130 ft
Leak rates (two-containment approach) ^a	
Inner containment (at 20 psi)	10%/day
Outer containment (at 10 psi)	0.5%/day
High bay atmosphere	Air
Heat transfer equipment vaults atmospheres	Nitrogen
^a Single-containment approach (at 10 psi).	0.1%/day

The safety of the plant is ensured through independent and diverse shutdown systems. These safety measures include a plant protection system that senses any abnormalities and automatically shuts down the plant when necessary; a low-pressure primary coolant system that incorporates a reactor vessel placed in a guard vessel to assure that the reactor core will be immersed in the sodium coolant even in the unlikely event of reactor vessel leakage; an auxiliary power source to maintain coolant flow in case of loss of power; and multiple containment of nuclear fuel fission products with the containment consisting of fuel cladding, a primary coolant system boundary, and a separate reactor system outer containment. Plant safety is discussed more fully in Section 4.2.7.

Nationally recognized codes and standards, such as those issued by the AEC (RDT Standards), the American National Standards Institute, the American Society of Mechanical Engineers, the Institute of Electrical and Electronic Engineers, and others would be used wherever appropriate in the design, manufacture, construction, and operation of the systems and components constituting the plant. These codes and standards represent the accepted practice in the engineering and construction industries for assuring a uniform, acceptable level of quality in any product, and as such, they provide further assurance that the plant will function in accordance with its design objectives.

4.2.3.2.1 Reactor Heat-Generation System

The reactor heat-generation system consists of the reactor core, control rod drives, and reactor vessel. The core is an array of fuel assemblies, which are vertically disposed and surrounded by one or two rows of radial blanket assemblies and probably one row of reflector assemblies. Control assemblies are interspersed with the fuel assemblies in a regular pattern. The core assemblies are supported upon a grid plate structure that also forms a part of the inlet plenum for the entry of coolant sodium to the assemblies, and they are surrounded by a core restraint system to restrict physical distortion of the core that could result from thermal bowing and radiation-induced swelling of core structural material.* Radial shielding is

*The potential magnitude of the effect of radiation-induced swelling of core structural materials, in particular austenitic stainless steel, and its impact on core design were not recognized in the LMFBR community until after the design phase of the 1000-MWe follow-on studies had been completed; therefore the follow-on reference concepts do not reflect this effect. It is now generally concluded that some form of active or passive core restraint system will be necessary to counteract swelling effects at least until such time as improved, swelling-resistant structural materials for core use are developed. Such a system is included in the design of the FFTF now under construction. This is discussed more fully in the cost-benefit analysis (Section 11).

provided between the core and reactor vessel wall to limit radiation damage to the vessel, radiation exposure of components adjacent to the vessel, and radiation streaming upward through the space around the vessel to the operating areas above. A typical reactor cross section is shown in Figure 4.2-2.

Each fuel assembly comprises a cluster of fuel elements or pins, each containing blanket material above and below the fuel material to serve as an integral axial blanket. The cluster of fuel elements is enclosed within an outer duct or can, and the spacing between elements for the flow of reactor coolant is assured through the use of spiral wire wraps around the elements or spacer grids. Most probably the outer duct is hexagonal, with the elements within the assembly and the assemblies within the core disposed in triangular arrays. Typical fuel assembly and fuel element designs are shown in Figure 4.2-3.

The reference fuel elements are sealed to contain all fission product gases within the elements. A gas collection plenum is provided in order to limit the pressure buildup in the element over its design life.

Radial blanket assemblies have the same outer duct configuration as the core fuel assemblies but contain fewer, larger diameter elements. In other respects, they are similar to the fuel assemblies. Control absorber assemblies are also similar except that the element clusters are smaller to allow clearance for vertical motion within the outer stationary duct.

The fuel material is mixed plutonium-uranium oxide.* Elements are helium filled to enhance heat transfer across the gap between fuel and cladding. Blanket material, both axial and radial, is depleted uranium oxide. The control absorber material is most probably boron carbide (B_4C), although other materials are also under consideration.

The design of the reactor vessel varies significantly depending upon whether the primary coolant system arrangement is of the pool or the loop type. In the pool-type configuration, the reactor and all components of the primary coolant system

*The assumption of mixed-oxide fuel for the model reactor plant is not meant to preclude the use of other fuel materials, such as plutonium carbides or nitrides, which are being developed for possible use in commercial LMFBRs. It is recognized that the safety, environmental, economic, and performance implications of such advanced fuels would have to be thoroughly investigated prior to any decision on their use. A discussion of the economic and resource incentives for advanced fuel development and eventual use is presented in Section 11.

TYPICAL REACTOR CROSS-SECTION

Figure 4.2-2

TYPICAL FUEL ASSEMBLY AND FUEL ELEMENT DESIGN
Figure 4.2-3

[intermediate heat exchangers, circulating pumps, valves (if any), and inter-connecting piping] are immersed in a large primary tank filled with sodium. In the loop-type configuration, the piping, intermediate heat exchangers, pumps, and valves are outside the reactor vessel. For a pool-type system, the primary tank is about 50 ft in diameter; for a loop-type arrangement, the reactor vessel is 20 to 30 ft in diameter.

The design of the tank or vessel upper closure head depends upon the refueling concept chosen for the plant; concepts considered to date are discussed in the ensuing section.

In all designs a guard vessel surrounds the reactor vessel or primary tank. The guard vessel prevents the core from being uncovered in the event of reactor vessel or primary tank rupture. A typical reactor elevation section is shown in Figure 4.2-4.

4.2.3.2.2 Core Component Handling System

The core component handling system has the function of handling fuel and other core assemblies from their receipt at the reactor plant through acceptance testing, storage, conditioning, and insertion into the reactor core; and during removal of the core assemblies from the reactor, the handling system provides for the storage for decay of postirradiation heat generation (commonly called "decay heat storage"), and the inspection, cleaning, and packaging of this material for shipment to reprocessing facilities.

A number of different basic design approaches have been proposed for commercial-size LMFBR power plants for the removal of spent core assemblies from the reactor and for their decay heat storage prior to shipment from the plant site. The principal differences between the approaches are in the location of the decay storage facilities and in the method of transferring assemblies between the core and storage positions. The prime goal of all approaches is to reduce the reactor down-time necessary for refueling operations to the minimum length of time consistent with plant safety in order to improve plant availability and consequently plant economics.

In-vessel Decay Storage--Under-Head Transfer

This approach has been proposed for use in conjunction with a pool-type primary system arrangement (Figure 4.2-5). In addition to the reactor, primary pumps, and intermediate heat exchangers, two rotatable drums containing receptacles for holding

TYPICAL REACTOR ELEVATION SECTION

Figure 4.2-4

In-vessel Transfer of Core Assembly between Core and Decay Storage

Transfer of Core Assembly between Primary Tank Decay Storage and Spent Fuel Cell

IN-VESSEL DECAY STORAGE--UNDER-HEAD TRANSFER REFUELING SYSTEM SCHEMATIC

Figure 4.2-5

spent core assemblies are immersed in the bulk sodium of the primary tank. Cooling is achieved by natural circulation of the bulk sodium through the assemblies.

Transfers of core assemblies between the reactor and in-vessel storage drums are effected under the sodium in the primary tank by an in-vessel transfer machine mounted in rotatable plugs in the reactor head.

In a typical refueling sequence, new core assemblies are brought into the new fuel cell from outside the containment building through double-valved transfer tubes. Here they are removed from shipping containers and inspected. They are then transferred by the ex-vessel transfer machine to the storage drums in the primary tank. This is done while the reactor is operating. At refueling, the reactor is shut down, the spent core assemblies selected for replacement are moved one by one from the core to the storage drums and are replaced by the new assemblies previously placed in the drums. Later, after a decay period, each spent core assembly is moved from its storage drum position to the spent fuel cell by the ex-vessel transfer machine and sealed in a shipping container.

Ex-vessel Decay Storage--Under-Head Transfer

This approach has been proposed for use in conjunction with a loop-type primary system arrangement. One or more sodium-filled spent fuel decay storage vessels are located near the reactor in the containment building below the operating floor. Several transfer positions are provided within the reactor vessel outside the core restraint system and shield. These transfer positions contain canisters open at the top end and filled with sodium. As in the previous system, an in-vessel and an ex-vessel transfer machine are used in the refueling operation. The in-vessel machine transfers core assemblies between core positions and the sodium-filled canisters in the transfer positions, always immersed in the bulk sodium in the vessel. The ex-vessel machine transfers the assembly-containing, sodium-filled canisters between the transfer positions in the reactor vessel and the storage positions in the decay storage vessels and maintains assembly cooling in transit.

The external decay storage vessels may contain rotatable drum storage racks or fixed racks. In the latter case an additional handling machine is provided in the tank to distribute the spent assemblies from the entry point to the various storage positions. The system (employing rotatable plugs) is shown schematically in Figure 4.2-6.

In-vessel Transfer of Core Assembly between Core and Transfer Canister

Transfer of Core Assembly in Canister between Reactor Vessel and Ex-vessel Decay Storage or Spent Fuel Cell

EX-VESSEL DECAY STORAGE - UNDER-HEAD TRANSFER REFUELING SYSTEM SCHEMATIC

Figure 4.2-6

Ex-vessel Decay Storage--Hot Cell Transfer

This approach may be used with either a pool-type or loop-type primary system arrangement. The salient difference between this and the previously described system is that the top of the reactor and external decay storage facility are enclosed within a shielded, inerted hot cell. In order to conduct refueling operations, the reactor vessel head and storage facility cover are removed entirely within the hot cell. The entire operation is shielded by the hot cell walls and roof. The in-vessel transfer machine is mounted above the open reactor vessel. As in the previous system, an in-vessel machine transfers assemblies between core positions and sodium-filled canisters in transfer positions within the reactor vessel, always immersed in the bulk sodium in the vessel. The ex-vessel machine transfers the assembly-containing, sodium-filled canisters between the transfer positions in the vessel and storage positions in the decay storage facility. This system, adapted to a pool-type reactor, is illustrated in Figure 4.2-7.

4.2.3.2.3 Heat Transport System

The heat transport system has the function of removing the heat generated by the reactor core and converting it to the rotational mechanical energy required by the generator to produce electric power. The overall system consists of a radioactive primary coolant system, a nonradioactive secondary coolant system, a steam generation system, and a steam plant system, the latter including the turbine that delivers the required mechanical energy to the electrical generator.

The primary coolant system consists of several redundant circulating loops that conduct sodium from the core exit plenum of the reactor vessel and circulate it through intermediate heat exchangers wherein the heat is transferred to the sodium of the secondary coolant system. The primary sodium then returns to the reactor vessel. In the secondary system, secondary sodium is heated in the intermediate heat exchangers and is circulated to the steam generation system. Three parallel primary loops and three secondary loops, one serving each primary loop, may be used, although six primary loops have been included in some concepts.

As mentioned earlier, two basic arrangements for the primary coolant system have been proposed--the pool-type and the loop-type configurations. These are depicted schematically in Figure 4.2-8. In the pool-type configuration, the reactor, intermediate heat exchangers, primary pumps, and interconnecting piping are all immersed in a large primary tank filled with sodium. In operation, sodium is

EX-VESSEL DECAY STORAGE--HOT CELL TRANSFER
REFUELING SYSTEM SCHEMATIC

Figure 4.2-7

POOL AND LOOP-TYPE PRIMARY COOLANT SYSTEM CONFIGURATIONS

Figure 4.2-8

drawn from the bulk content of the tank by the primary pumps and is forced through the reactor. It then flows by gravity through the intermediate heat exchangers and discharges back to the bulk sodium in the primary tank. The driving force for intermediate heat exchanger flow is the difference between the levels of sodium over the reactor and in the remainder of the primary tanks. With this configuration, the primary tank with its cover and the tubes and tube sheets of the intermediate heat exchangers constitute the primary coolant system boundary.

In the loop-type configuration, the primary pumps and intermediate heat exchangers are located external to the reactor vessel, so that the primary system boundary is extended to include the outer containers of these components and the interconnecting piping, as well as the reactor vessel. Either hot or cold leg pumps could be used in the primary system.

The primary loop piping is elevated and guard vessels are provided around the pump, intermediate heat exchanger and reactor vessel so that leaks in the primary piping or these components can not cause the sodium level in the reactor to drop below the minimum safe level. Thus, the loop nozzles would be covered, and continuous heat removal by sodium circulation through the loops could be permitted.

The use of sodium as the reactor coolant is one of the major distinguishing characteristics of the LMFBR. Sodium is an opaque lustrous, silvery metal that melts at 208°F and boils at 1621°F. The favorable thermodynamic and neutronic characteristics of liquid sodium have long been recognized. These include its metallic nature, low neutron cross section, high boiling point, extended liquid range, high thermal conductivity, and low vapor pressure at temperatures of interest for LMFBR application.¹⁶ The breeding process depends upon maintaining high energy neutrons and minimizing their capture in the system, and it is therefore necessary to use a coolant that does not slow down or capture neutrons to an appreciable extent. The property of low vapor pressure allows the use of high-temperature low pressure systems that afford more efficient thermal energy conversion.

One of the favorable aspects of the chemistry of sodium is its ability to accommodate fission product releases. Should a fuel cladding failure occur, gaseous radioisotopes (particularly xenon and krypton), which are not held by the sodium, would escape to the cover gas where they would be removed by gas purification and recovery systems. Some of the metals and halogens would normally plate out on metal surfaces or be removed by the sodium cold-trap purification system.¹⁷ This would leave the longer lived cesium isotopes and isotopes of alkali metals in

sodium solution as the principal fission product impurities in the sodium. Thus, accidental spills involving primary system sodium would not release large quantities of fission products, as most of these would have already been removed through plate-out or by the purification system, or would be held in solution in the sodium or in the gas purification system.

Sodium is generally nonreactive with a wide range of structural materials commonly used in system components. It does, however, react with most nonmetals, and precautions must be taken to prevent exposure of sodium to air and water. Sodium readily oxidizes in air, and when exposed to water it reacts vigorously to form hydroxides and hydrides and to release hydrogen gas, which in turn can react explosively with oxygen. Special design features are employed in LMFBF plants to prevent sodium-water reactions and to mitigate their effects in the unlikely event that such reactions should occur.

Sodium in the primary system becomes radioactive primarily through the generation of Na-24, a 15-hr half-life radioisotope that decays to stable magnesium. This is the predominant radioactive species found in the primary sodium coolant system during normal operation. Because of its short half-life, it decays within a few days and thus constitutes only a minor maintenance problem. An additional activation product is Na-22, which, although produced in much smaller quantities and of lower radiotoxicity than Na-24, has an appreciably longer half-life of 2.6 years.

Design precautions are taken to avoid contact of radioactive primary sodium with air or water. All equipment or pipelines containing radioactive sodium are enclosed in inert gas-filled steel lined cells that exclude or reduce the normal atmospheric level of oxygen to prevent rapid combustion of any exposed sodium. The highest construction and quality assurance standards are used in the fabrication of stainless steel piping and vessels containing sodium.

The primary sodium system is designed to operate at a lower pressure than the secondary system. Thus, should a leak develop in an intermediate heat exchanger between these two systems, any leakage would progress from the nonradioactive secondary system into the radioactive primary system; therefore, contamination of the secondary system with radioactive primary sodium is highly improbable. Finally, even though such leakage would not result in a radiological problem in the secondary system, the intermediate heat exchangers are designed to facilitate removal or replacement of faulty tubes.

Engineering design features are employed to minimize the possibility of leakage in steam generators that could result in reactions between sodium and water. The water and steam in the steam generation system interface only with nonradioactive secondary sodium coolant.* Leak detection and early sodium-water/steam reaction warning systems are designed to detect a leak early enough so that damage to the steam system and other secondary system components would be minimized by prompt plant shutdown and water or sodium dumping. Steam generator models have been tested in both sodium and water environments to assess leakage mechanisms and the extent of damage.¹⁸ Additional testing is currently under way. Nevertheless, it is recognized that some leaks may occur, and provisions are made to relieve the pressures resulting from sodium-water/steam reactions by venting through rupture disks. The reaction product relief system is designed to separate and retain the sodium and most of the particulate reaction products and to vent the hydrogen to the atmosphere. (Spontaneous hydrogen explosions do not take place unless the reactants are present in certain specific proportions and other conditions prevail. Following a sodium-water/steam reaction, should one occur, there would be little oxygen present, and several other reaction products would be present in addition to hydrogen.) Thus, adequate means are available to prevent damage from a potential sodium-water/steam reaction.

An inert cover gas is used over free sodium surfaces in the primary coolant system. The gas used is most probably argon, although helium has also been proposed. In the event of defects in the sealed fuel elements, noble gases such as xenon and krypton that escape from the fuel elements collect in the cover gas and are removed in the cover gas purification system. This system is described in Section 4.2.3.2.4.

The overall steam cycle is expected to be similar to that of modern fossil-fired steam-electric power plants. Figure 4.2-1 shows a system in which secondary sodium is used to reheat the steam after it has passed through the high-pressure portion of the turbine. Commercial LMFBR power plants may also be designed with steam-reheat systems, or with steam systems having no reheat. Typical flow diagrams for such systems are shown in Figures 4.2-9 and 4.2-10.

*In pool-type systems, the secondary sodium may be very slightly radioactive depending upon the design and location of the intermediate heat exchanger in the pool and upon the shielding provided between this component and the reactor.

TYPICAL FLOW DIAGRAM FOR LMFBR STEAM-REHEAT STEAM SYSTEM

Figure 4.2-9

TYPICAL FLOW DIAGRAM FOR LMFBR NONREHEAT STEAM SYSTEM

Figure 4.2-10

4.2.3.2.4 Radioactive Waste Processing Systems

The model plant contains a number of auxiliary systems to perform specific support functions:

- Liquid metal receiving and processing;
- Inert gas receiving and processing;
- Impurity monitoring, sampling, and analysis;
- Auxiliary heating and cooling;
- Radioactive waste processing;
- Service and utility functions.

Of these systems, the radioactive waste processing systems are perhaps the most important in terms of environmental considerations and are described in this section.

The LMFBR primary system is a completely closed system in which the sodium and cover gas are continually purified to maintain radioactivity at low levels. Four radioactive waste processing systems are provided:

- (1) gaseous waste system, subdivided into reactor cover gas and cell atmosphere purification systems;
- (2) liquid waste system;
- (3) solid waste system; and
- (4) sodium waste system.

The relationship of these systems to the reactor system is shown in Figure 4.2-11.

It is expected that commercial LMFBR power plants will be designed and operated such that there will be essentially no planned, continuous release of radioactivity to the environment, although some small leakage of radioactive gases through seals and diffusion of tritium through metal is anticipated. Intermittent releases from LMFBRs would include the discharge of steam generator blowdown water containing small quantities of tritium.

Tritium is formed in the fission process and is also generated in boron carbide control elements. Minor amounts of tritium may also be generated from lithium and boron impurities in the sodium coolant. Current information indicates that most of the tritium formed in the fuel elements diffuses through the stainless steel cladding into the coolant,^{19,20} along with tritium formed in boron carbide control

RELATIONSHIP OF RADIOACTIVE WASTE SYSTEMS TO REACTOR SYSTEMS

Figure 4.2-11

elements. Once in the coolant, tritium readily migrates into all parts of the reactor system that are at high temperatures, that is, through the intermediate heat exchanger walls to the secondary coolant system and subsequently through the steam generator walls to the steam system.

Experience at EBR-II²¹ indicates that most of the tritium leaving the core precipitates in the primary sodium purification system cold traps, and of the <10% of the tritium that diffuses through the intermediate heat exchanger tube walls to the secondary sodium system, most precipitates in the secondary sodium purification system cold traps. An amount <1% can be expected to diffuse through the steam generator tube walls to the steam system and be released to the environment as tritiated water. The expected behavior of tritium in a commercial LMFBR plant is shown graphically in Figure 4.2-12.

Calculations performed for FFTF²² indicate that some diffusion of tritium through piping and equipment walls into the inert gas-filled enclosures surrounding the primary coolant system can also be expected. It is intended that this tritium be removed in the cell atmosphere purification system and subsequently be disposed of as solid waste and packaged in drums for offsite disposal.

The tritium release from the model 1000-MWe LMFBR power plant (see Table 4.2-2) was calculated based on the following generally conservative assumptions:

- (1) Activation of control rods containing B-10 produces about 60 Ci/day.
- (2) All tritium produced in the fuel during fission (about 30 Ci/day) diffuses through the clad into the coolant.
- (3) The cold trap removal "efficiencies" were 73%.*
- (4) The total primary cold trap capacity was 220 gpm.
- (5) The total secondary cold trap capacity was 250 gpm.

These assumptions and the calculations model used²³ lead to a decontamination factor (DF) for tritium of about 186 and thus a total release of about 120 Ci/year of tritium. Use of lower cold trap "efficiencies" of 25% instead of 73% would lead to a reduced DF (about 35) and an increased release of tritium by about a factor of 5 (about 630 Ci/year). The failure of all the primary cold traps (or all the secondary

*Steady-state hydrogen concentrations in sodium coolant of 0.5 parts per million (ppm) and cold trap operating at 300°F (0.135 ppm hydrogen) produce this "efficiency."

PATHWAYS FOR TRITIUM RELEASE IN LMFBR POWER PLANTS

Figure 4.2-12

cold traps) would lead to an increase in the tritium release of about one order of magnitude, a rate leading to release of about 1200 Ci/year if uncorrected.

Table 4.2-2

POSTULATED RADIONUCLIDE RELEASES--1000-MWe
LMFBR POWER PLANT AT 80% CAPACITY FACTOR

Nuclide	Atmospheric Release Ci/year
H-3	60.0
Ar-39	80.0
Kr-85m	0.3
Kr-85	0.4
Kr-87	0.4
Kr-88	0.5
Xe-133	0.03
Nuclide	Liquid Release Ci/year
H-3	60.0

Gaseous Waste System

The gaseous waste system processes all gases that can become contaminated with gaseous or volatile radioactive species. These species include:

- (1) fission products escaping from failed fuel, notably the noble gases, halides and alkali metals;
- (2) Ar-39 generated from potassium impurities in the coolant; and
- (3) activated sodium aerosol and vapor.

Two separate systems will be utilized to control radioactive noble gases. These two systems are the reactor cover gas purification system and the cell atmosphere purification system.

Reactor Cover Gas Purification System

Radioactive effluent from the primary argon cover gas system is passed to the reactor cover gas purification system. A typical system (designed for FFTF)²² is depicted in Figure 4.2-13. The effluent is drawn through sodium aerosol traps (not shown) into a

REACTOR COVER GAS PURIFICATION SYSTEM FLOW DIAGRAM

Figure 4.2-13

vacuum tank, then through filters to remove particulate material, and is pumped to a compressor receiver where there is a delay prior to injection into the processing section. The gas is cooled and passed sequentially through a succession of vessels filled with granular charcoal. The xenon and krypton are adsorbed more strongly by the charcoal than is the argon carrier gas. At FFTF four such delay beds are expected to provide an effective delay of 2.8 days for krypton and 140 days for xenon.²² Thus, the radioactivity flowing from the last of the four charcoal-filled vessels is almost entirely due to Kr-85. The effluent from the final charcoal bed is passed to a fractional distillation column where xenon and krypton are stripped out and collected in a pool of liquid argon in the bottom of the column. Periodically, the xenon/krypton concentrate is drained to a heat exchanger in which the cryogenic liquid is vaporized into suitable pressure containers for storage and shipment from the site. The quantity of Kr-85 removed from the cover gas system is not expected to exceed 2000 to 6000 Ci/year.²⁴ This radioactivity, along with inactive krypton and xenon isotopes, could be stored in one standard 50-liter gas cylinder at a pressure of <100 psi.

No provision is made in the above processing system for removal of Ar-39. Because the reactor cover gas purification system is intended to be completely recirculating, escape of Ar-39 from the system can only occur through leaks in seals and valves, and there may be small quantities of these gases purged on a routine basis. The half-life of Ar-39 is 260 years.

Several alternative approaches to a reactor cover gas purification system are possible. For example, a series of delay tanks could be substituted for the charcoal delay beds.^{11,12,14,15}

Cell Atmosphere Purification System

The effluent from all inert gas-filled enclosures surrounding all primary sodium piping systems is passed to the cell atmosphere purification system.

The major radioactive species present in the inert gas-filled enclosures under normal conditions is tritium that diffuses through the primary coolant system walls. However, radioactive argon, krypton, and xenon may enter the cells through leaks in the reactor cover gas purification system and from the fuel handling cells.

A typical design of a cell atmosphere purification system is shown in Figure 4.2-14. Effluents from the various cells are drawn into a suction tank and through a filter by a compressor and forced into a compressor receiver tank. The gases are then

CELL ATMOSPHERE PURIFICATION SYSTEM FLOW DIAGRAM
Figure 4.2-14

passed through a tritium removal system.* Tritium could be removed by conversion to tritiated water by catalytic oxidation or by passing the gases over hot copper oxide. The water could then be absorbed in a suitable solid such as calcium oxide, which is then handled by the solid waste system. Gases other than tritium (e.g., noble gases that leaked into the cell atmosphere) would be delayed and filtered prior to release to the atmosphere. Estimated annual quantities released due to such leakage are given in Table 4.2-2.

Liquid Waste System

The liquid waste system processes all potentially contaminated liquids before the liquids leave the plant. The sources of the liquid waste streams are the fuel handling area, the sodium waste system, laboratory areas, laundry, maintenance, and other support systems. The quantity of liquid waste entering the system is expected to be between 200,000¹² and 40,000¹⁵ gal/year, which is primarily low-level. (The liquid waste system will not handle tritium-containing effluents from the steam system; this waste stream has been discussed previously.)

A flow diagram of a typical liquid waste system is shown in Figure 4.2-15. Wastes from each of the principal sources are collected in holdup tanks. The level of the radioactivity in these tanks is determined periodically. If no radioactivity is detectable, the contents of the tanks may be discharged to the condenser cooling water or to the plant sewage system. If radioactivity is present, the contents are first transferred to delay tanks to allow short-lived activity to decay and then to a decontamination system. Decontamination may be accomplished in ion exchange beds¹⁵ or by evaporation of the liquid to create a concentrated sludge.¹² The contaminated ion-exchange resins or sludges are handled by the solid waste system.

Solid Waste System

Solid waste will be generated by most other waste systems, fuel handling operations, and laboratory and maintenance operations. The waste will be in such forms as spent resins, sludges, filters, clothing, and tools.

The flow diagram of a typical solid waste system is shown in Figure 4.2-16. Filters and dry solids will be compacted, and moist resins will be dried and combined with other dry solids. Sludges will be mixed with concrete and cast into drums. Centrifugation may be employed in some cases to further concentrate the sludge before casting.¹² Tritium wastes will be converted to tritiated calcium hydroxide.

*It is expected that a tritium removal system would be developed for the reactor cover gas purification system, also.

LIQUID WASTE SYSTEM FLOW DIAGRAM

Figure 4.2-15

4.2-42

SOLID WASTE SYSTEM FLOW DIAGRAM

Figure 4.2-16

All solid waste will be packaged in drums. If 55-gal drums are used, the number is estimated to be between 135 and 270 per year.

Sodium Waste System

The purpose of the sodium waste system is to convert small amounts of metallic sodium waste, both radioactive and nonradioactive, into a less reactive form.* The principal sources of sodium waste at the reactor are spent cold traps. Other possible sources are sodium-contaminated hardware (equipment and spent fuel assemblies).

Because almost all of the tritium generated in the fission process is expected to precipitate out in the primary and secondary cold traps, the conversion of residual metallic sodium in spent cold traps requires special handling to minimize tritium releases to the environment. The procedure used will probably involve the following steps:

- (1) draining as much sodium as possible into the primary or secondary sodium system before disconnecting the cold trap;
- (2) deactivation of the contents with appropriate solvents with collection of all off-gases and liquids;
- (3) conversion of tritium hydride to tritiated water and routing of all liquids to the solid waste system for conversion to tritiated Ca(OH)_2 ;
- (4) storing the chemically deactivated trap until removal for burial, salvage, and further cleanup.

After metallic sodium wastes have been chemically deactivated they will be routed either to the liquid waste system or the solid waste system for further treatment, discharge, or shipment.

4.2.3.2.5 Instrumentation and Control

The plant instrumentation and control system consists of three basic parts--the protection system, data system, and control system. The protection system provides for the measurement of neutron flux density from startup to full power, coolant and component temperatures, system pressures, sodium flows and levels, gamma radiation, radioactive gases and particulates, and other parameters of interest or necessity. The protection system provides for safe reactor shutdown and containment isolation.

*Sodium waste may not be processed onsite for LMFBR power plants. A few special facilities may be built for this purpose. However, for this Statement, onsite processing was assumed.

The data system monitors many of the above mentioned parameters independent of the protection system, performs various calculations, and provides display information. A fuel cladding failure detection and location system may also be included.

The control system provides both direct operator and automatic control of the reactor and includes alarm functions from remote or local stations to the central control system. Redundant instrument sensors are provided throughout the plant for separation of protection and control functions.

4.2.3.2.6 Reactor Containment

As mentioned earlier, an important safety feature of LMFBR commercial plant concepts is the multiple containment of the fission products generated in the fuel elements during reactor operation. The barriers to fission product release are the fuel element cladding, the boundary of the primary coolant system, and the outer reactor containment.

The outer containment consists of a leak-tight cylindrical steel or steel-lined concrete building having a flat bottom and hemispherical or ellipsoidal dome. The containment building houses the reactor and entire primary coolant system, spent fuel handling and storage facilities, and sodium service systems related to the primary system.

Both single- and a double-containment approaches have been suggested in commercial plant concepts to date. In the single containment concept, the steel wall or steel liner of the building constitutes the containment and is designed for a leakage rate of 0.1% of the contained volume per day at 10 psi internal pressure. In the double-containment approach, there is a dome over the reactor, and the primary coolant system below the operating floor is enclosed in inerted, concrete-shielded steel-lined vaults. This inner containment system has a leakage rate of $\sim 10\%$ of the contained volume per day at 20 psi. The inner containment is surrounded in turn by the outer containment, which in this case is designed for a leakage rate of 0.5% per day at 10 psig. In concepts employing hot-cell refueling, the hot cell and its liner serve the same purpose as the reactor dome. Figure 4.2-17 illustrates a double-barrier containment system.

The possibility of using a confinement- rather than a containment-type building has also been considered. In this type of building, outleakage is passed through a filtration system prior to release.

DOUBLE-BARRIER REACTOR CONTAINMENT BUILDING

Figure 4.2-17

4.2.3.2.7 Nonradioactive Waste Treatment Systems

The major source of chemical wastes from an LMFBF is discharges from the water treatment systems. The waste chemicals from these systems are discharged to the environment via the blowdown from a closed-cycle cooling system or in the condenser cooling water discharge from an open-cycle system. Condenser cooling water is treated to control biological slimes that can seriously interfere with the transfer of heat and adequate flow of water. The most common chemical used for this purpose is chlorine, added either as gaseous Cl_2 or as a hypochlorite solution. Other biocides are used for special conditions, particularly when salt water is used for cooling as the fouling deposits are often more extensive and contain a wider range of organisms than in the case when fresh water is used.

All closed-cycle evaporative systems after a period of operation will concentrate dissolved salts; the salts are the residue from water evaporation. The most common salts found in natural waters and hence in closed-cycle cooling systems are the bicarbonates, sulfates, chlorides and nitrates of calcium, magnesium, sodium, and potassium. These salts have various degrees of solubility within the water with calcium carbonate the least soluble. Whenever the solubility of this salt is exceeded, it sometimes precipitates jointly with other salts to form sludges or solid surface deposits called scale. The solubility of calcium carbonate decreases with increasing temperature so that scaling tends to occur first and to the greatest extent in the warm condenser where surface scale could seriously impair performance.

There are various ways to control scale and sludge formation in a closed-cycle system. One typically used is to blow down system water at a relatively low rate and replace it with fresh makeup water having a low salt concentration. This maintains lower salt concentrations within the system. Another method is to replace some of the carbonate (and the bicarbonate in equilibrium with it) with sulfate by adding sulfuric acid. The resultant carbon dioxide is released to the atmosphere. Sometimes sequestering agents are also used to tie up metallic ions which might otherwise form insoluble carbonates.

Other chemicals may be used in closed-cycle systems to provide treatment for corrosion, biological fouling, and silt deposition.

In summary, there are many chemicals which, by virtue of system water treatment, might appear in the blowdown discharged to the environment. For the most part, however, the blowdown merely contains higher concentrations of the salts found in the makeup water to the system and possibly residual chlorine. If sulfuric

acid is used, there will be higher concentrations of sulfates and lower bicarbonate concentrations.

Very high purity water is required in the steam system of an LMFB power plant to minimize corrosion and scale formation. To maintain purity, a portion of the process water may be blown down and replaced with demineralized water. The demineralized water is produced from raw water through a series of water treatments which include clarification, filtration, and ion exchange. Clarification is the process of removing turbidity and sediment. An array of chemicals may be used in clarification such as alum, sodium aluminate, ferric sulfate, and others to coagulate finely divided and colloidal matter suspended in the water. The chemicals used for this purpose are called coagulants or flocculants, and the residues of their operation are settled out from the water. The wastes from these operations, called sludges, are most often dewatered in settling basins. Other methods can be used to remove the water from the sludges, such as incineration and centrifugation. Filtration is used after clarification, and the residue materials are treated in the same manner as for clarification.

Ion exchange (demineralization) is typically the last step in the water purification process and it results in a relatively large chemical waste stream. In ion exchange, dissolved compounds in the form of ions are removed from the water to be purified by first attracting them to electrically charged resinous surfaces where they become tightly held and replaced with other ions. Typically the ion exchange resins are hydrogen cation and hydroxide anion types where the hydrogen ions are used to replace the cations in solutions such as magnesium and calcium, and the hydroxide ions are used to replace anions such as bicarbonate and sulfate. After a period of operation the ion exchange resins become ineffectual and must be regenerated. Sulfuric acid is generally used to regenerate the cation resin and sodium hydroxide the anion resin. In the process of regeneration, the original ionic materials that were trapped by the resins are released and replaced with hydrogen and hydroxide ions; the resulting regenerative waste stream is then composed of a solution of the original ionic materials less the bicarbonates, which were converted to water and carbon dioxide, plus quantities of sodium sulfate, the reaction product of sulfuric acid and sodium hydroxide. These wastes are then normally discharged to the environment after diluting them with either the once-through condenser cooling water flow or the blowdown effluent from a closed-cycle cooling system. In the absence of a specific plant design, no accurate estimates can be given as to the amount of demineralization wastes an LMFB power plant may generate.

Steam system blowdown is another possible source of aqueous waste from the system. Since the water is chemically treated for scale and corrosion control, it is expected that some of the chemicals used for this purpose might also be in the blowdown. Some of these chemicals might be sodium phosphates for scale control, sequestering agents, sodium hydroxide or carbonate for pH control, or hydrazine for oxygen control. The blowdown may also contain some corrosion products and other dissolved solids.

Other sources of aqueous chemical wastes discharged to the environment include spent chemicals from major component cleaning operations, sampling and laboratory wastes, sanitary wastes, and laundry wastes.

Cleaning of all heat transfer surfaces must be done periodically to remove possible scale deposits on all major plant components. Strong acids are usually used for this purpose. The waste stream from the scale cleaning operations results in a mixture of acids, copper, iron, phosphates, and other materials. Good practice requires that these wastes be collected in storage basins where they can be treated by neutralization and sedimentation after which they can be discharged to the environment with suitable dilution.

Plant sanitary wastes can be handled either by packaged treatment systems or by discharging them to a municipal treatment facility if available.

Nonradioactive solid wastes include demineralizer resins, filters, replaced equipment components, shop and laboratory wastes, garbage, water intake debris, and other trash. The disposal of these wastes is most likely performed by a disposal agency licensed and regulated by local authorities.

4.2.4 Environmental Effects of Construction

During the period of time an LMFBR power plant and its transmission lines are under construction, land and water will be disturbed and modified where the permanent structures are to be located and where other areas are used for temporary access, storage of construction materials and equipment, and disposal of excavated and moved earth. The extent of dredging of water courses and the extent of clearing, leveling, and filling of land will depend on the particular site. Special precautions will be taken to minimize erosion, siltation, and destruction of biota during the construction operations and in the period before the alteration of disturbed areas is completed.

In order that construction impacts will not be excessive, careful consideration will be given to selection of the plant site and the location of the transmission-line corridors. General environmental criteria for effects on land relevant to construction impacts will apply.^{25,26}

In making estimates of future LMFBR construction impacts, comparisons were made with LWRs and with coal-fueled plants in order to benefit from the experience gained with these two alternative types of base-load power plants. The construction impact of the LMFBR power plant is expected to be similar to and not greater than that for coal or for LWR plants. In general, construction of LMFBR plants does not appear to have environmental disadvantages in comparison to alternative plant types.

4.2.4.1 Impacts on Land

4.2.4.1.1 Reactor Plant Construction

Because of basic similarities in the three alternative types of base-load steam electric plants (LMFBR, LWR, and coal), these three types of plants have similar construction impacts. Construction impacts of a power plant depend upon the characteristics of the site and its previous uses. Planning is required to control soil erosion during construction, when the ground is being disturbed by earth movement and by movement and storage of spoil. Control of erosion to acceptable levels can be achieved by several means such as limiting the steepness of inclines, and by providing runoff channels and settling areas to collect and settle surface runoff water before allowing the runoff (with entrained soil) to reach natural surface-water bodies. The methods used must consider the nature of the local soil and the climatology of the region.

Excavations for foundations of major structures often require extensive dewatering, in which groundwater entering the excavation is pumped out to the surface water. Depending on the local groundwater recharge, this dewatering may temporarily lower the water table in production wells in the vicinity or may affect flow gradients in the groundwater in other ways, thus affecting the quality of groundwater. Careful attention must be given to the condition of the water to be disposed of during the dewatering process. Pollutants that get into this water during the construction process, if not removed or neutralized, can produce serious consequences in the biotic community. Geological, hydrological, and soil studies of the site are required prior to construction to allow construction methods to be chosen so that operations can be carried out with minimal deleterious effects.

4.2.4.1.2 Transmission Line Construction

One mile of transmission line right-of-way involves about 21 acres of land (assuming a 175-ft average width). This area is about equal to that of a nuclear power plant building complex. Ten miles of right-of-way involve an area of about 200 acres, equal to the entire disturbed area of a plant site, but the required length of transmission line is often much greater, and the distance traversed is likely to involve diversity of land and vegetation that is far greater than that on the plant site. Thus, the potential for environmental impact of transmission line construction is of a magnitude comparable to, or greater than, construction at the plant site.

Where forested areas must be traversed, some loss of trees is to be expected. Clearing of vegetation can be minimized, and cover can be replaced where it is badly disturbed. Temporary cover must be established in areas subject to erosion during the time between clearing and actual construction of the facilities. Soil erosion and sedimentation of nearby waterways must be controlled by the use of runoff water management procedures, such as providing proper drainage slopes and settling basins. Utilities have adopted regulations for construction with regard to techniques to minimize the impact of necessary access during construction and to minimize erosion effects, especially effects that extend the impact beyond the corridor proper.^{27,28}

Large transmission lines have towers hundreds of feet apart, and lines can be strung between towers by various means, including the use of helicopters, without requiring clearing of natural vegetative growth between the towers. Highly selective cutting can be used for trees to retain maximum tree height with allowance for line sag and sway between towers. Herbicides are used mainly for economic reasons when clearing long rights-of-way, but this procedure is meeting increasing disfavor.

4.2.4.2 Impact on Water

The demand for water during construction is relatively small. For example, water is needed for cleaning equipment and for sanitary facilities. Effluents would require treatment before discharge. Changes in surface water quality by runoff from disturbed land areas are possible; however, procedures for controlling erosion will minimize such effects.

The major impact on water use and on the aquatic environment is expected from construction in wetlands or along shorelines. As a general rule, construction in wetlands should be avoided. Access to the source of cooling water for a plant requires the construction of intake and discharge structures with associated

dredging and installation activities. The turbidity, color, or temperature of the river or lake would not be expected to be altered except for short periods during dredging. Some temporary local impacts on water traffic or shorelines may occur if temporary breakwaters or barge landings are required during construction, but such effects would not normally be either significant or persistent.

4.2.4.3 Ecological Effects

Changes in the local ecology are expected during the disruptions accompanying the construction activities, with reversal of some changes and restoration to a new equilibrium upon completion of these activities. For birds and fish, permanent impacts can be ameliorated in some cases by providing bypass routes or feeding stations for migratory species. For trees and other vegetation, carefully controlled procedures can minimize effects during construction and maximize recovery.

Clearing of wooded land will result in a loss of wildlife habitat. During such clearing and construction, animals will seek shelter in adjacent wooded areas; however, there may be increased mortality among displaced animals. Some foraging species may be benefited by this activity as new shrubs and low brush develop from natural regeneration. The creation of such an "edge effect" also tends to encourage increased biotic variety and density. Construction impacts in the site area are expected to be less than those in the transmission line corridors because the plant very likely will involve much less total area than that required for the lines.

Increased suspended solids and siltation, from dredging operations or runoff from disturbed soils, can lead to very noticeable short-term aquatic impacts or to subtle long-term impacts, depending on the interactions between the various physical, chemical, and biotic components of the system.²⁹⁻³¹ These impacts can be minimized and held to acceptable levels by careful consideration of siting and construction methods.

Turbid water, besides being aesthetically displeasing, will often be avoided by fish, although fingerlings and adults often are quite resistant to high concentrations of suspended solids for short periods. Fish may seek cover and stop feeding because of difficulty in finding food in turbid water or even cease migrating. High suspended solids have been shown to depress growth, affect length/weight relationships, and increase the incidence of fin rot and thickened gill epithelia. It is possible that at high suspended sediment concentrations the potential to resist other stresses is reduced. Hiding places and food supplies may be destroyed either by siltation destroying weed beds and benthic organisms, or directly by

dredging. Spawning may be affected by siltation of spawning grounds. Egg mortality may increase by silt smothering eggs or diminishing the flow of interstitial water within the redds, thereby limiting the amount of available oxygen.

Dredging spoils or runoff with a high organic content can exert a high oxygen demand and lead to local depletion of oxygen in the sediments or even up into the water column if the water currents are such that there is no fresh supply of oxygenated water.

Because the benthic community structure is strongly dependent on the type of substrate available, it is not surprising that studies of the influence of siltation on benthic organisms have shown pronounced effects. The substrate, of course, is changed physically in a dredged area or by suspended solids settling out. The oxygen demand of organics in the silt can affect the degree of oxygenation of the sediments. Benthic productivity can be affected by turbidity causing a decrease in primary productivity by reduction of light penetration. Changes in bottom topography and current patterns can also lead to very noticeable impacts. If dredging were performed in an estuary resulting in changes in currents, salt water could intrude, leading to associated physical, chemical, and biological changes.

Potential problems with pollutants in dredged materials or land runoff getting into the water supply and biotic systems can be of significant concern. Whether biostimulatory (phosphates and nitrates) or toxic (heavy metals and pesticides), once such pollutants find their way out of the dredged spoils into the water and become available to biota (including man), the repercussions can be far-reaching and possibly irreversible.

Erosion and consequent siltation of waterways during and after construction can be kept to a minimum by several methods.^{32,33} These include: (1) keeping grading to a minimum, (2) limiting the steepness of slopes, (3) constructing diversion swales to dissipate runoff and siltation basins to provide places for silt in runoff to settle out, (4) leaving vegetation intact near waterways to provide a buffer strip to slow down runoff and let silt settle out, (5) careful scheduling so that the time that the ground is disturbed is kept to a minimum, (6) keeping heavy construction equipment out of sensitive areas during critical periods such as the spring thaw, (7) saving topsoil to replace over disturbed areas when construction is complete, and (8) proper replanting and seeding as soon as possible.

Considerations^{30,34} for analysis of environmental impacts from dredging include: (1) volume and type of material to be dredged (e.g., coarse sands and gravels settle out quickly and cause few turbidity problems; clean, inorganic silts have little potential for creating oxygen depletion problems, and polluted sediments have chemicals which can be freed into the water); (2) biotic potential of the area to be dredged and of the spoil area; (3) type of dredging and disposal equipment (suction dredging where light materials are removed by dustpans, hoppers, side-casters; mechanical dredging where either loose or hard compacted materials are removed by clamshell, bucket, dipper, or ladder dredges; suction/mechanical dredging where either loose or hard compacted materials are removed by cutterheads); (4) maintenance or new work (greatest disturbance and potential problems with new work); and (5) land or water disposal. The observance of careful construction practices can alleviate many impacts as work progresses in the site and transmission line areas. Those that do occur will not be of a lasting nature provided prudent measures are taken during the planning as well as construction phases.

4.2.4.4 Community and Socioeconomic Effects (Discussed in Section 5)

The construction of a two-unit LMFBR power station* is expected to take approximately six to eight years and have a peak construction labor force of about 1500 to 3500 or more workers, similar to current LWR reactor practice. The impacts of these workers and the accompanying socioeconomic activity tend to be low if few new residents come to the local communities as a result of construction and if these communities have a large population and are highly industrialized. Schools, hospitals, police, and other community services are required for the newcomers as well as residential accommodations. Any sizable increase in the community will entail higher service costs, and provision should be made in advance for dealing with this aspect to avoid hardship and conflict. Historical community values for residents and traditional visitors may be affected by a major influx of personnel and business activity. Whether such changes are adverse or beneficial depends on the situation and interests of the persons involved.

Increased traffic, noise, and possible dust and other emissions will constitute a temporary annoyance during construction. Local tax payments, which serve as an important long-term compensation for local impacts, tend to lag the actual construction period effects. The extent to which local residents or traditional visitors are inconvenienced by construction is a factor contributing to a source of complaint or objection to the plant facilities.

*Nuclear energy centers, which would contain five or more LMFBR power plants, are discussed in Section 7.2.6.4.

4.2.5. Environmental Effects of LMFBF Power Plant Operation*

4.2.5.1 Impacts on Land Use

Approximately 35 to 50 acres of land will be required for facilities associated with an LMFBF power plant; namely, the reactor buildings, turbine building, switchyard, parking lot, access roads, and cooling towers. For a two-unit station, as much as 100 acres might be used. As a minimum, an exclusion area of at least 400 acres is needed, and most LWR stations today are on sites even larger. The average area of present day LWR power stations is about 1160 acres with a range of from 84 to over 30,000 acres.³⁵ Closed-cycle cooling systems, particularly cooling ponds or impoundments, require much larger commitments of land than needed for once-through cooling systems.

The land not actually used for facilities can be put to several uses depending on the site and its location. Many utilities have erected visitors' centers to provide opportunities to educate interested individuals and groups in the scientific and engineering aspects of nuclear power. In some cases, picnic areas have been provided adjacent to the visitors' centers. Other recreational uses have been allowed on the peripheral lands; fishing, boating, nature trails, and wildlife preserves are examples of present practice. In other cases, these lands have been leased or rented to farmers for agricultural purposes. The trend is toward sharing the land with other users in a beneficial manner, yet in keeping with the requirements set forth in 10 CFR 100 for control of personnel in the exclusion area, and ability to take emergency protective measures in the "low population zone." There is no reason to believe that this trend will be reversed.

Careful surveys and examination of records can indicate the possibilities for uncovering archaeological or historical values on a proposed plant site. Consultation with local and state historical and archaeological societies and with governmental agencies charged with preservation of historic places also will be done for future LMFBF power plant sites. Significant archaeological and historic artifacts discovered in the process of plant construction will be properly identified and preserved. Interference with the preservation of presently known historic sites or with their enjoyment by the public is not expected.

*The environmental impact of such auxiliary systems as the fuel storage and handling system, the air conditioning system and the heating and ventilation system would generally be insignificant compared to the impacts associated with an LMFBF power plant's condenser cooling water system.

4.2.5.2 Impacts on Water Use

The LMFBFR plant will use, as present day power stations do, large amounts of water for various purposes. The water for the most part will not be consumed in the conventional sense of the word but merely used for transferring waste heat to the environment and dispersing chemical and radioactive wastes.

Most of the waste heat from a thermal power plant is released when turbine exhaust steam is condensed as part of the energy conversion process. The latent heat of condensation is transferred to cooling water for eventual rejection to the environment. Heat is also released in lesser quantity by the service water system that provides necessary cooling of various plant equipment and components. Chemicals are used in regenerating ion exchangers, in treating water used in the steam system and, in some cases, in treating the cooling water to suppress biological growths that affect heat transfer. The waste products from these processes are discharged via the cooling water stream. The only radioactive waste expected to be carried off in the cooling water from an LMFBFR power plant is tritium that diffuses into the steam system and is discharged with the blowdown as tritiated water.

4.2.5.2.1 Thermal Wastes

The 1000-MWe LMFBFR will have a thermal efficiency of nominally 40% and a thermal rating of 2500 MW. There will be miscellaneous thermal losses to air (called general plant losses) amounting to less than 1%, a value typical of present day nuclear facilities. The amount of waste heat rejected to water is then about 59% of the plant's thermal rating or about 1475 MW. Table 4.2-3 briefly summarizes and contrasts present commercial nuclear plants and future LMFBFR nuclear plants versus a modern fossil plant with respect to waste heat cooling requirements.

Once-Through Cooling

The amount of waste heat the condenser cooling water carries is simply the product of its mass flow rate through the condenser, its temperature rise across the condenser, and the specific heat of water. For most once-through systems operational at existing power plants, typical temperature rises run between 8° and 32°F.

A 10°, 15°, or a 20°F cooling water temperature rise seems appropriate for the 1000-MWe LMFBFR power plant. The corresponding flow requirements are 1,004,000 gpm (2240 cfs), 669,000 gpm (1490 cfs), and 502,000 gpm (1120 cfs), respectively. In addition, service water requirements for miscellaneous purposes throughout the plant may require an additional 20,000 to 30,000 gpm water flow.

Table 4.2-3
THERMAL ENERGY BALANCES (MW)

	Thermal Rating	Electrical Generating Capacity	Stack and Misc. Plant Losses	Condenser Waste Heat
Fossil	2500	(40%) 1000	(15%) 375	(45%) 1125
LWR	3030	(33%) 1000	(1%) 30	(66%) 2000
HTGR	2560	(39%) 1000	(1%) 26	(60%) 1540
LMFBR	2500	(40%) 1000	(1%) 25	(59%) 1475

One of the most readily observed physical impacts of discharging waste heat to a natural water body is the formation of a thermal plume. The thermal plume is that portion of the receiving water which is heated to some temperature above ambient. The spatial and temporal characteristics of a plume are a function of many variables which are dependent on plant, meteorological, topological, and hydrological factors. Even under the most ideal circumstances, reliable characterization of plumes from specific plant discharges at specific sites under known meteorological and hydrological conditions is still challenging the predictive state-of-the-art. There may be competitive uses of the receiving water within the potential reach of a thermal plume and, therefore, the possible interrelationship of the heated plume water with these other uses must be examined for potential impact. Generally, a well designed power plant discharge will either ensure that the plume never reaches a sensitive area or, if it does, that the plume water temperature is sufficiently reduced so that the other uses for the receiving water are not adversely affected.

In the process of dissipating plume heat to the atmosphere, some water is evaporated. The water lost from the watershed in this manner is often treated as a consumptive water use by the plant. The LMFBR power plant, if located in the Great Lakes region or the Northeast, will evaporate (consume) about 12 to 13 cfs of water while operating in a once-through mode at full power.

Cooling Pond System

The most obvious physical impact of a pond system is the utilization of land surface area for uses which might be devoted to many other purposes. The actual amount of water surface required for a given plant capacity is strongly dependent on meteorological conditions of specific sites. For purposes of assessment, however, present operating experience indicates that one to two acres of water surface are required per megawatt of generating capacity with the lower limit more typical of fossil

plants. The LMFBR is similar to a fossil facility with regard to its waste heat discharge and, therefore, a value of 1.5 acres/MWe is reasonable. As a first estimate, then, about 1500 acres would be required for a 1000-MWe LMFBR power plant cooling pond.

The volume of water needed for a pond should at least be equal to the volume utilized by the power plant in a 24-hr period. In actual practice, ponds are often found to contain several times this amount. If a value of four 24-hr periods is selected, the LMFBR operating at a 20°F temperature differential would require about 410 million ft³ (≈9400 acre-ft) of water in its pond. This would correspond to about a 6 ft average depth for the 1500 acre pond for the LMFBR power plant.

Seepage of water from the pond, depending on its extent, could under certain conditions raise the local water table elevation, which could have a detrimental impact on surrounding land use. Further, depending on the quality of the pond water and its rate of exfiltration, there is a possibility for contaminating ground water aquifers which may be in the vicinity. If seepage is a potential problem, it may be resolved by adding a suitable liner to the pond. Nominal seepage rates of 1 to 5 cfs/1000 acres are used as rules-of-thumb for unlined ponds.

Evaporative losses from a cooling pond must be made up with water taken from a local surface or groundwater supply. As in the case of once-through cooling, evaporative loss from a pond is dependent on many factors which makes an accurate rate determination impossible without specifying site and meteorological parameters. Evaporative loss computations were made for two pond sizes corresponding to 2 acres/MWe and 1 acre/MWe. Interpolation of the given values for a 1.5 acre/MWe loading resulted in an evaporation rate of 0.9 gal/kWhr (33 cfs).³⁶

The blowdown from a pond will generally be at a temperature higher than the receiving water body.

Potential beneficial uses of cooling pond systems include recreational and/or conservation usage, reserve water supplies for a plant under drought or low flow conditions, and use as storage basins for flood control and stream regulation.

Spray Pond or Canal Systems

The spray pond or canal requires much less water surface area to dissipate its waste heat load than a cooling pond. An estimate for the amount of area that would be required for the LMFBR might be obtained by proportionately sizing the requirements

for an existing nuclear station in Illinois.³⁷ This station has a spray canal system with a surface area of 60 acres for a 1600-MWe nominal generating capacity or about 0.04 acre/MWe. On this basis, the 1000-MWe LMFBR power plant would require about 40 acres for a spray canal and perhaps double this amount for a spray pond, which is somewhat less efficient. The LMFBR power plant spray pond is estimated to have a yearly average evaporation rate of roughly 16 cfs.

Infiltration will probably not be a problem because of the relatively small size of the spray system. However, due to the fact that open sprays are used, there will be localized drift with consequent water loss. Based on operational tests, such loss appears to be no greater than 0.01% of the spray water circulation rate (about 0.1 cfs).³⁸

Evaporative Cooling Tower Systems

Mechanical and natural draft towers, while they are physically dissimilar, are much alike from their water use standpoint. The primary impacts of wet towers from a water use point of view are those basically associated with evaporative and drift losses and blowdown water quality.

Evaporative losses from wet tower systems should be roughly equivalent to spray ponds (16 cfs). Drift represents a water loss from the cooling system and must be made up. Drift rates are usually stated as a fraction of the tower system circulatory flow. A nominal drift rate is 0.005% for mechanical draft towers and 0.002% for natural draft towers.³⁹ These two drift rates lead to small losses of the total flow requirement of the LMFBR power plant.

The blowdown from an evaporative tower carries with it about 1 to 2% of the waste heat from the plant.^{40,41} This is significantly less than if the plant were discharging to the receiving water on a once-through basis.

Dry Cooling Tower System

The dry tower operates totally on conductive and convective heat transfer to the atmosphere and, therefore, has no water requirements save for makeup for possible minor system leaks. In contrast to the other cooling systems, the dry tower would have the least water use impact. The primary reason for its current lack of general acceptance is that its operation is controlled by the ambient dry bulb temperature which results in a significantly higher cooling water temperature than if the wet bulb temperature were regulating. A higher cooling water temperature results in

higher turbine exhaust pressures which in turn reduce plant efficiency and increase fuel usage. Most, if not all, turbines manufactured in this country at present, because of design constraints, could not operate efficiently at the high back pressure a dry tower system might operationally give; however, compatible turbine designs are currently being developed. An HTGR direct cycle gas turbine power plant would be well suited to the use of dry cooling tower systems.

4.2.5.2.2 Chemical Wastes

The aqueous chemical wastes from a nuclear power plant generally enter the environment via the blowdown stream from a closed-cycle cooling system or the circulating cooling water stream from an open-cycle system. In actual practice, the chemical composition of these waste streams is as varied as the number of existing discharges. Nevertheless, the nature of some of these wastes can be categorized as to their origin and to a limited extent their chemical makeup. The major sources of the waste streams from a nuclear power plant are those originating from the condenser cooling system and the process water system. All other waste streams are minor compared to them.

Chlorine

Essentially the only significant chemical waste associated with an open-cycle system is that resulting from the chlorination of the cooling water. Chlorination of closed-cycle system cooling water is common practice.

Chlorination of cooling water is used to control biological slimes within a cooling system. The presence of certain bacteria and algae in natural waters leads to the preferential formation and attachment of biological slimes to system surfaces and particularly to heated surfaces with which the water comes in contact. In the presence of sunlight at locations such as sluices and other outside water-carrying structures, the slimes are predominantly algal, while bacterial slimes predominate in closed heat-transfer surfaces. In power plants it is essential that the slimes be removed since they would otherwise seriously interfere with the transfer of heat and adequate flow of water.

The removal is most commonly effected by the periodic addition of small concentrations of chlorine or a hypochlorite solution. In either case, hypochlorous acid (HOCl) is the primary active agent. This substance and the hypochlorite ion in equilibrium with it are the chlorine species present in greatest concentration in

an aqueous solution formed by the addition of either chlorine or a hypochlorite to water.

When ammonia (and ammonium ion) and certain organic compounds are present in water to which chlorine is added, the free chlorine reacts with them to form compounds called chloramines, in which the chlorine retains oxidizing potential and some defouling capacity. The rate of reaction depends on concentrations of the ammonia and hypochlorous acids and the pH.⁴²

Chemical defouling is accomplished by killing at least some of the organisms that largely comprise the slime. Unfortunately, agents that are toxic to slime bacteria are also toxic to other aquatic organisms, so it is desirable to manage defouling treatments to release as little of the toxic substance to natural water bodies as possible. Chlorine has a good combination of high toxicity to fouling organisms and a natural instability in water so that free chlorine persists only short periods before it decomposes to form chloride ion and free oxygen. The time of persistence is markedly reduced in the presence of light.⁴³ For example, under ultraviolet illumination conditions an initial free chlorine concentration would decrease by a factor of 10 in about 18 sec.⁴⁴ Reaction times are greater by a factor of the order of 50 to 200 in the dark.⁴¹ Chloramines, sometimes referred to as combined chlorine, also decompose in natural waters to form nearly innocuous chloride. Decomposition times are longer than for free chlorine, ranging from a few hours to a few days.

Optimum defouling by chlorine requires optimization of the free chlorine concentration and the frequency and duration of treatment. With operations optimized in this way, the chlorine levels in the discharge from typically operated once-through or closed cooling systems will show considerable variation, depending on the biological and chemical content of the natural water. The residual chlorine will be essentially entirely in the combined form.

An analysis of the harm caused aquatic biota by the discharge of chlorine was done by William A. Brungs, of the EPA's National Water Quality Laboratory,⁴⁵ who suggests the following criteria, not to be consistently exceeded, for the protection of freshwater aquatic life:

- (1) in areas receiving wastes treated continuously with chlorine, not to exceed 0.01 mg/l for the protection of more resistant organisms only, or not to exceed 0.002 mg/l for the protection of most aquatic organisms; and
- (2) in areas receiving intermittently chlorinated wastes, not to exceed 0.2 mg/l for a period of 2 hr a day for more resistant species of fish, or not to exceed 0.04 mg/l for a period of 2 hr a day for trout and salmon. If

free chlorine persists, more restrictive criteria are warranted. Alternate procedures or substitutes for chlorination should be investigated.

These criteria apply to levels of residual chlorine in receiving waters as a result of discharges under circumstances in which the chlorine is combined. Conformity with the criteria in discharges from power plants will assure no significant harm to fresh water biota. Slightly exceeding them in the discharge stream might cause some biological damage in a zone within which a combination of dilution and reaction between residual chlorine and various substances in the water reduces the residual chlorine at the periphery of the zone to be in conformity with the criteria.

Experience to date indicates that with normal operating procedures, residual chlorine in some plant discharges can be kept low enough for short enough periods to meet the criteria and to avoid excessive impact, whereas in other plants it may not be possible. In such cases, special methods or operations can be employed to prevent harmful chlorine discharges. These methods include such things as:

- (1) cessation of blowdown from cooling tower circuits for certain periods to allow decay of residual chlorine before discharge;
- (2) mixing with natural water to allow reaction between chlorine and "chlorine-demand" constituents before discharge;
- (3) adding a retention pond to allow decay of residual chlorine in cooling tower blowdown before discharge (sprays can be used to accelerate the decay);
- (4) the use of devices such as charcoal beds to accelerate the decay of residual chlorine before discharge;
- (5) adding dechlorination chemicals such as sulfite before discharge.

Considering the options available, it is possible (at a cost) to use chlorine for defouling without serious environmental impact for most installations, and perhaps for all.

Dissolved Solids Concentrations

All closed-cycle evaporative systems, after a period of operation, will concentrate dissolved salts; the salts are merely the residue from evaporation. A measure of the degree of salt concentration within a closed system is given by a parameter called the concentration factor (C), which is the ratio of the concentration of dissolved salts conserved within the system to their concentration in the makeup water. To minimize both the amount of makeup water required and the blowdown, C

must be maintained at a high value. If evaporation (E) and blowdown (B) are the only sources of water loss from a closed cycle system, a simple relationship can be used to equate the blowdown requirements to the evaporation losses in terms of C, namely:

$$B = \frac{E}{C - 1}$$

In actual operating practice, values for C range between 1.5 and 25 or more. Very low C values are used in salt water systems where blowdown salt concentrations have to be kept low for water quality considerations. Very high C values can be used with high quality makeup water and/or chemically treated system water. For purposes herein, a C value of 5 will be chosen to calculate the makeup and blowdown water requirements for the LMFBF. A value of 5 is consistent with average makeup water quality. Table 4.2-4 lists the makeup (M) and blowdown flow rates for the closed-cycle evaporative systems under these conditions.

Table 4.2-4
ESTIMATED LMFBF BLOWDOWN AND MAKEUP REQUIREMENTS

	1500-Acre Pond	Spray Pond or Wet Towers
C	5	5
E(cfs)	22	16
B(cfs)	5.5	4
M(cfs) $\approx$ E + B	27.5	20

The blowdown contains salts whose concentrations equal those in the makeup water plus an excess released from the evaporated water. For surface water the dissolved solid content might typically be in the neighborhood of 250 ppm.

Other Chemical Wastes

The process water system provides the requisite high purity water utilized in the steam loops of all nuclear reactors to minimize corrosion and scale formation within these loops. The high purity water is produced from raw water through a series of water treatments which include clarification, filtration, and ion exchange. The aqueous waste effluent from the process water system is primarily composed of a solution of sodium sulfate, the reaction product of sulfuric acid and sodium hydroxide, plus significant quantities of the dissolved and suspended solids and

salts originally contained in the raw makeup water (less the bicarbonates which were converted to water and carbon dioxide in the cationic ion exchange process). Sulfuric acid and sodium hydroxide are typically used for ion exchanger regeneration purposes. The wastes are discharged either to the blowdown or the once-through cooling streams and in this way are diluted before entering the environment.

4.2.5.2.3 Biological Effects of Thermal Discharges

The effects of a thermal discharge on the biota of a receiving body of water will depend on a number of factors, including the temperature of the receiving water, the temperature difference (Δt) between the receiving water and the discharge, the quantity and velocity of the water being discharged, the particular organism and the stage of its life cycle, and other physical, chemical, and biological characteristics of the receiving water. The flora of natural waters include bacteria, phytoplankton, periphyton, and larger aquatic plants. The fauna include fish, zooplankton (invertebrates, fish eggs, and larvae), and benthos (macroinvertebrates such as aquatic insect larvae, worms, and shellfish). For ease of discussion, the effects of heated discharges on these organisms can be divided into the following directly lethal effects and sublethal effects.

The potential lethal effects of the effluent will depend on its temperature, the species and life stages of the organisms subjected to the plume, their acclimation temperatures, and physiological condition, among other factors. Experience with thermal discharges and laboratory studies have indicated that adult fish and some larger macroinvertebrates will avoid lethal temperatures if they can sense a thermal gradient and are free to move away. Kills due to extreme temperatures are not expected due to LMFBR waste heat discharges under normal operation. A sudden increase or decrease in temperatures, however, could be lethal to populations present in the discharge areas at that time. The design of the discharge structure, whether on a river, estuary, lake, or ocean, should be such that fish will not become trapped in the discharge canal or in small embayments.

Most benthic organisms move slowly and are not only subject to immersion in the heated effluent but also to the scouring action of the discharge, which may have a greater adverse effect than the temperature. On a lake or river, elimination of the benthos due to the discharge will be a local effect in the area immediately adjacent to the outfall. At an estuary or ocean site, modification of populations of some salt water benthic species such as crabs, clams, and oysters may have economic effects.

Plankton and drift organisms carried through the hottest part of the plume may be killed, but if such temperatures are confined to a small area of the water, plankton populations as a whole will not be adversely affected.*

Thus, of utmost importance in the choice of the cooling method for an LMFBR power plant is to pay attention to the horizontal and vertical extent of the thermal plume. On a river or estuary, restriction of the 5°F isotherm to a minor portion of the cross-sectional area and volume of the flow (e.g., less than 25%) would provide, in most cases, an adequate zone of passage for fish and ensure that only a minor portion of the planktonic organisms will be subjected to lethal temperatures. On a lake or reservoir, restriction of the area and volume of the 5°F isotherm and the choice between a once-through or closed-cycle cooling system will depend on the biological activity existing at the particular site and the extent to which the lake or reservoir is and will be used as a heat sink. Presently, the technology is available for the design and construction of a discharge that will meet these criteria.

The thermal discharge at sublethal temperatures is expected to have its most significant impact on coastal and estuarine aquatic ecosystems, primarily due to effects on benthos and disruption of normal temperature cycles. No adverse effects are expected from a thermal discharge on a large river if plumes of 5°F or greater are limited to a minor fraction of the waterway and if discharge into a spawning area is avoided. On a lake or reservoir, no adverse effects are expected if temperatures outside the mixing zone are less than 3°F and, again, if discharge into a spawning area is avoided. The area and volume of the mixing zone will depend on specific site requirements.

Release of heated effluent from several sources along a lake or cold water stream, such that a major portion of the entire water body is significantly warmed, may cause a change in the composition of fish, benthic, and plankton populations. To prevent such conditions, it may be necessary to limit the number of heat sources or to install different cooling systems for some of them.

4.2.5.2.4 Biological Effects of Water Intake

The requirements for cooling and service water will vary from about 10,000 gpm to 700,000 gpm or more for a 1000-MWe LMFBR power plant depending upon mode of cooling and design specifications. All water taken into the plant must be screened to

*Other factors of importance include the location and type of discharge and the temperature rise across the condenser.

protect the pumps and the condenser tubes from being clogged or damaged by debris and aquatic biota present in the cooling water. Inasmuch as some loss of aquatic biota (mostly small fish) by impingement on the screens is inevitable, screening of large amounts of water at some of the existing power plants has resulted in fish kills of considerable magnitude. However, by proper siting of the intake and the selection of the appropriate screening system, fish impingement can be greatly curtailed. Considerable information now exists on guiding fish at water diversions and on screening them at water intakes. General reviews on this problem have been prepared.⁴⁶⁻⁵⁰

Siting and design features that will help in minimizing the rate of fish impingement can be incorporated in the LMFBR plants. Consideration can be given to various points listed below, and this list can be revised and/or extended as new information becomes available.

- (1) A preliminary survey of the site for intake is essential to determine the relative density of fishes and fish spawning areas at various locations.
- (2) Depths at which a high concentration of fishes occurs should be avoided.
- (3) Where the depth of maximum abundance varies with seasons, intakes with variable depth inlet should be considered to avoid withdrawing a large number of fish.
- (4) Intakes with very short forebays or with traveling screens flush with the shore of a water source are preferable; in such designs fish have more freedom of lateral movement and can be swept away from the screens by strong currents of the source water.
- (5) The primary emphasis in preventing fish impingement should be at the outermost structure of intake.
- (6) The velocity at the outermost inlet should be matched with the swimming capability of native fish species to allow enough margin for the fish to escape. Low intake velocities are preferable for designs using screens perpendicular to flow. The swimming performance of fish tested in a relatively small apparatus in the laboratory should be interpreted and extrapolated for field conditions with ample caution.
- (7) The skimmer wall results in higher intake velocity because of reduced cross-sectional area under the wall. Therefore, the depth of the skimmer wall should be kept to the minimum necessary.
- (8) Lateral escape ports for fish are of little practical value unless the intake has only one or two adjoining bays. Where the number of bays is

- large, the fish will be exposed to flows in front of several screens to escape laterally and thus may not be able to avoid impingement.
- (9) The recirculation of warmed water to avoid freezing of screens in winter should be very localized, and the volume should be as low as possible.
 - (10) The screening system should be carefully selected to meet the requirements of a particular site.
 - (11) Where fish removal or pumping systems are planned, they should be adequately tested under field conditions for the native species prior to installation.
 - (12) A velocity cap on offshore intakes is desirable in reducing fish intake but should not be considered as a total solution to the problem.
 - (13) Recirculation of discharged water should be avoided in winter.
 - (14) In addition to using an intake design determined to be effective through substantial testing and use, an alternative plan of action including installation of auxiliary devices for fish repulsion and intake design modifications should be planned in case large numbers of fish are collected at the screens in spite of other precautions.

4.2.5.2.5 Entrainment and Condenser Passage Effects

In all types of steam-electric power plants, including the LMFBR, the circulating water that cools the condensers contains small organisms that have passed through the intake screening devices. In current industry practice, the smallest mesh size commonly used in power plant intake screening devices is 3/8 in. Thus, any aquatic organisms small enough to pass through 3/8-in. openings are potentially subject to entrainment and passage through the pumps and condensers of a power plant's circulating water system.

Entrainable organisms include (a) phytoplankton; (b) zooplankton; (c) the meroplanktonic eggs and larvae of certain fish and invertebrates; (d) the accidental and transient plankters such as normally benthic invertebrates and normally demersal fish eggs and larvae; (e) the fry and juveniles of many species of fish; and (f) other groups such as the protozoa, bacteria, and aquatic fungi. Although the distribution of plankton in natural waters is patchy rather than uniform or random, a general assumption can be made that in most cases the quantity of aquatic microbiota entrained will proportionately reflect the rate of intake or makeup flow.

Mortality of entrained multicellular organisms is nearly total in evaporation cooling towers or spray canal cooling systems, due to high temperatures, prolonged residence in the system, recirculation through the condensers, and chemical conditions in such

systems. Regarding power plants which employ cooling lakes, the differences in species composition and quantities of organisms between makeup flow entrainment and return in the blowdown depend on several environmental factors and customarily are not objects of the studies of the entrainment effects. Consequently, the detailed studies of effects of entrainment and condenser passage relate to plants which employ once-through cooling.

During condenser passage, entrained organisms may be subjected to thermal shock due to heat transfer in the condenser; mechanical shocks and abrasion due to turbulence, cavitation, and collision with pump impellers and irregular surfaces in the system; pressure changes due to pumping and hydrostatic head differences; toxic chemicals, including chlorine or other biocides used for shock defouling of condenser tubes; and additional thermal stress, turbulence, and heavy predation in the discharge area.

A number of power plant studies provide descriptions of entrainment effects regarding diverse combinations of cooling system designs (Δt , times of exposure, biocide use, and types of discharges), types of aquatic ecosystems, and methods of investigation. These studies verify that in many instances, considerable damage to entrained organisms has resulted from thermal stress, as well as from other types of stresses and shocks listed previously. In other cases, however, measurable damage has been slight. By examining these studies in the aggregate, some general observations can be discerned regarding the causes and extent of entrainment damage. Such considerations are presented in Appendix II.H.

The most frequently reported major effect of condenser passage on phytoplankton, observed in comparisons of primary productivities in condenser discharge samples with those in condenser intake samples, has been the apparent stimulation of photosynthesis during months when ambient (intake) water temperatures are low, and the apparent partial inhibition of photosynthesis during warmer months. Although most such data exhibit considerable variability, this type of effect has been evident in a number of studies (e.g., studies conducted at the Chalk Point, Indian River, Indian Point, Crane, and Millstone Point power plants). Differences in photosynthetic rates dependent upon ambient intake temperatures have also been reported in studies conducted at York River, Waukegan, and Allen power plants.

4.2.5.2.6 Biological Effects of Chemical Discharges

LMFBR power plant chemical discharges to the aquatic ecosystem can be regulated in compliance with legal restrictions, regulatory guidelines (including criteria and recommendations), and generally accepted conclusions in the scientific literature.

Operation within the technical specifications imposed on an operating LMFB power plant are not expected to cause unacceptable adverse effects on the biota.

The major influence on the chemistry of the receiving water is the discharge of blow-down from closed cooling systems which has been concentrated manyfold (approximately five times). To date there is little indication that an increase in total dissolved solids has adversely affected biota at power plant sites. The potential for damage will increase when large deterioration of water quality occurs on a single receiving body of water due to multiple siting of power plants. This could be especially severe under low flow conditions. Where the makeup water source contains toxins such as copper or industrial chemicals, their concentration by evaporation inside the power plant may result in toxic effects at the outfall. This phenomenon will be predictable from water quality analyses done by the utility for preparation of the environmental report and necessary preventive action will be taken prior to operation.

The chemical discharge will interact with the thermal discharge in several ways. In the first place, increased temperature tends to increase most metabolic processes in cold-blooded animals. Presence of toxins would therefore be of greater potential damage in a heated plume than in the river itself. The interaction of temperature with pesticide toxicity in fish has been noted, for example.⁵¹ Furthermore, it is generally agreed that fish are attracted to thermal discharges at least under some ambient conditions⁵² and therefore would be exposed to a greater opportunity for damage than would be suggested by the concentration of toxins calculated on the assumption of complete mixing in the receiving water.

4.2.5.3 Radiological Effects on Biota and Man

During normal operation of an LMFB power plant, small quantities of radioactivity will be released to the environment. The release rates of Table 4.2-2 have been estimated based on expected LMFB power plant designs and operating practices. For the purpose of radiological evaluation, plant and site characteristics have been chosen from those characterizing current LWR sites.^{35,53} These are summarized as follows:

Meteorology: An averaged set of parameters from current LWR sites (see Table II.I-1).

Population: An average population based on current siting experience.³⁵

Site and Plant: Release due to leakage 10 m above ground level. Site boundary 500 m from release point.

Dilution of Liquid Discharges: A small river with average and low flow of 3000 cfs.

Other Characteristics: Distribution and types of biota, drinking water, and dietary populations, non-resident populations (swimmers, hunters, anglers, visitors, and boaters) as given in the following tables and discussions.

The releases of Table 4.2-2 were used to compute the maximum doses given in the following paragraphs.

Dose rates have been included in Table 4.2-5 for biota in the vicinity of the LMFBR power plant. These include phytoplankton, zooplankton, benthic organisms, terrestrial and aquatic plants, and local and migratory birds and mammals. The river littoral zone has been chosen to allow calculation of the maximum dose rates to be expected on land or at the terrestrial-aquatic interface. Doses in all other terrestrial areas will be lower than those given for the littoral zone. Doses in the river are applicable only to aquatic forms.

Table 4.2-5
MAXIMUM DOSES TO BIOTA IN THE VICINITY
OF THE LMFBR POWER PLANT

Organism	Effluent	Dose Rates (millirems/year)	
		River	Littoral Zone
Aquatic Plants	0.01	1.9(-3)	3.0(-2)
Aquatic Invertebrates	0.01	1.9(-3)	3.0(-2)
Aquatic Vertebrates	0.01	1.9(-3)	3.0(-2)
Terrestrial Plants			3.0(-2)
Terrestrial Invertebrates			3.0(-2)
Birds			3.0(-2)
Mammals			3.0(-2)

A diagrammatic representation of some of the pathways utilized in this evaluation is included in Figure 4.2-18. In addition, equilibration between geosphere,

4.2-70

EXPOSURE PATHWAYS FOR ORGANISMS OTHER THAN MAN

Figure 4.2-18

hydrosphere, and atmosphere was considered, as well as the various trophic levels in the biosphere.

Inspection of Table 4.2-5 shows that doses are, in fact, quite low for all of the biota in the area. At these dose levels no deleterious effects are to be expected.^{54,55}

The methodology above was then extended to man. Figure 4.2-19 presents some of the pathways to man. Direct doses to the human population via atmospheric dispersion of LMFBR power plant releases are given in Table 4.2-6. These doses were calculated using methods outlined in Appendix II.E and are the total-body doses due to releases of noble gases.

The maximum airborne doses are found in the zero to one mile sectors, with a maximum total dose rate of 0.012 millirem/year to the 512 permanent residents in this area. Hunters, anglers, park visitors, and other persons in the area temporarily will receive lesser doses with an annual dose markedly less than 0.012 millirem. The site boundary dose, at 500 m from the release point, was 0.029 millirem/year. The annual integrated dose commitment to the population over the 50-mile radius will be 0.10 man-rem.

Direct and indirect doses to man via the tritium releases, both liquid and gaseous, are given in Table 4.2-7. These include doses to permanent residents of the area (e.g., via public water supplies), to temporary residents, hunters, anglers, boaters, swimmers, and park visitors and to consumers of food produced in the area. The dose estimates include the waterborne tritium deposited from the gaseous tritium release, calculated as discussed in Section II.0.3 of Appendix II.0. The maximum cumulative, annual dose received by any member of the permanent population via tritium releases from the LMFBR power plant would be 0.001 millirem. The total annual population dose would be 0.47 man-rem for residents in the 50-mile radius.

Direct radiation from the LMFBR power plant will contribute less than 1 millirem/year at the fenceline. This dose drops off very rapidly with distance, however, so that the total annual population dose from this source will be much less than 1 man-rem.

The releases from the LMFBR power plant do not constitute a radiological hazard. The expected releases and resulting doses not only meet the requirements set forth

4.2-72

PATHWAYS TO MAN
Figure 4.2-19

Table 4.2-6

CUMULATIVE POPULATION, CUMULATIVE ANNUAL DOSE, AND AVERAGE
DOSE DUE TO AIRBORNE RELEASES FROM THE LMFBR POWER PLANT

Radius (miles)	Cumulative Population	Cumulative Dose (man-rems/year)	Average Annual Dose (millirems/year)
1	512	6.3(-3)	1.2(-2)
2	2,928	1.1(-2)	3.6(-3)
3	7,936	1.4(-2)	1.8(-3)
4	15,980	1.8(-2)	1.1(-3)
5	25,470	2.1(-2)	8.2(-4)
10	116,100	3.3(-2)	2.9(-4)
20	494,200	5.3(-2)	1.1(-4)
30	1,329,000	7.6(-2)	5.7(-5)
40	2,457,000	9.6(-2)	3.9(-5)
50	3,774,000	1.1(-1)	3.0(-5)

Total = 0.1 man-rem/year

Table 4.2-7

POPULATION DOSES DUE TO TRITIUM RELEASES^a
FROM THE LMFBR POWER PLANT

Pathway	Population Type	Population at risk (man-years/year)	Dose (millirem/year)
Tapwater	Residential		
Riverwater		2.0(5)	5.9(-4)
Groundwater		8.0(5)	1.8(-4)
Dietary	Residential	1.0(6)	1.5(-4)
Dietary	Sport	1.0(4)	4.9(-6)
Immersion	Recreational	1.0(4)	7.1(-8)
	Residential	1.0(6)	1.1(-7)
Inhalation	Recreational	1.0(4)	4.2(-6)
	Residential	1.0(6)	6.2(-5)

Total = 0.47 man-rem/year

^aBoth liquid and gaseous releases.

in 10 CFR 20 but even the much more stringent guidelines proposed in Appendix I of 10 CFR 50, currently a guide for LWRs.

Doses due to atmospheric releases were also calculated for more adverse meteorological conditions such as periods of high atmospheric stability, low wind speeds, and limited mixing depths. These conditions, given in Table 4.2-8, and the population densities used in making the calculations, given in Table 4.2-9, are representative of the worst 5% LWR site conditions. The maximum airborne dose was computed to be 0.07 millirem/year at the site boundary.

4.2.5.4 Impacts on the Terrestrial Environment

4.2.5.4.1 Once-Through Cooling

As indicated previously, the once-through mode of cooling is the most efficient and least expensive and also produces the smallest meteorological impacts. All of the power plant waste heat discharged in a large water body will eventually enter the atmosphere by radiation, evaporation, and conduction from the water surface. Observations at operating power plants indicate that, except for minor and localized increases in steam fog over the thermal plume on cold winter days, atmospheric modifications due to the waste heat will be insignificant, because the energy will be transferred slowly over a large area.⁵⁶ Heat and water vapor are added to the atmosphere as air moves from the land over a warm water surface. Owing to the local temperature excess, more heat will enter the atmosphere from the thermal plume than from the rest of the surface. Thus, air with a trajectory over a thermal plume will be somewhat warmer, more humid, and less stable as a consequence of thermal discharges. Observations confirm that an increase in the frequency and density of steam fog over the immediate plume area is the primary observable effect of thermal discharges into the large water bodies.⁵⁶

Steam fog is frequently formed when cold air passes over a warm water surface. In addition to very rapid evaporation, heat energy passes from the water to the atmosphere, creating a very unstable density stratification and considerable turbulence in the lowest several feet of air. The warm, humid air rises and mixes with the cooler air aloft; if conditions are proper, some of the water vapor in the mixture condenses out as steam fog due to the nonlinear relation between saturation vapor pressure and temperature. Further vertical mixing tends to re-evaporate the condensed water droplets. Thus, the process which creates steam fog also tends to eliminate it. Despite some research,^{57,58} the exact criteria for predicting the formation of steam fog are not known, because they involve a complex function of water and air temperature, humidity, and wind conditions. A known fact is that

Table 4.2-8

METEOROLOGICAL DATA--WORST CASE
(joint frequency distribution of wind speed
and stability class for a single sector^a)

Wind Stability	Wind Speed (m/sec)						
	0.8	2.5	4.5	6.9	9.5	12.5	>12.5
B	0.0054	0.0108	0.0075	0.0032	0.00080	0.00015	0.000078
C	0.0031	0.0072	0.0080	0.0041	0.00093	0.000024	0.000036
D	0.0072	0.0147	0.0116	0.0093	0.0047	0.00036	0.00041
F	0.0111	0.0214	0.0165	0.0085	0.0044	0.00031	0.00075

^aThis distribution applies to the two sectors containing about 40% of the 50-mile population.

Table 4.2-9

CUMULATIVE POPULATION, CUMULATIVE ANNUAL DOSE, AND AVERAGE
DOSE DUE TO AIRBORNE RELEASES FROM THE LMFBR POWER PLANT--WORST CASE

Radius (miles)	Cumulative Population	Cumulative Dose (man-rems/year)	Average Annual Dose (millirems/year)
1	512	5.8(-3)	1.1(-2)
2	3,000	1.1(-2)	3.7(-3)
3	10,000	2.1(-2)	2.1(-3)
4	26,800	3.0(-2)	1.1(-3)
5	51,000	3.9(-2)	2.6(-4)
10	322,800	8.9(-2)	2.7(-4)
20	3,517,000	3.6(-1)	1.0(-4)
30	6,564,000	5.3(-1)	8.0(-5)
40	7,288,000	5.4(-1)	7.4(-5)
50	7,758,000	5.5(-1)	7.1(-5)
Total = 0.6 man-rem/year			

steam fog can be present over thermal discharges and not over nearby unheated water. Observations of steam fog at existing power plants with surface discharges [Point Beach, Waukegan, and Lakeside (Toronto)] show that steam fogs over thermal discharges are thin and wispy, show a considerable amount of turbulence, and cover an area only slightly larger than the plume itself.⁵⁶ Visibilities within the fog area may momentarily drop off to tens of feet but are usually several hundreds of feet, and thus pose no problem to navigation or other uses of water. Thus, steam fog over a thermal plume differs in form, density, depth, and liquid water content from other types of fog, such as advection and radiation fogs. The intensity and frequency of steam fog and icing would be even smaller for plants with submerged, diffuser discharge systems, as the area of the plume and the air-water temperature difference are both reduced.

Onshore winds and subfreezing conditions may cause ice deposits on nearby land and vegetation. Observations at power plants having shoreline canal discharges show that the area of icing created by the steam fog moving inland will be extremely small, extending only 10 to 50 ft inland from the shore.⁵⁶ Due to the very small size of the droplets, the ice that forms is of very low density and is very fragile. Hence, this form of icing is much less destructive than other types, such as freezing rain.

Since the total quantity of heat put into the main water body by these power plants is very small compared to that exchanged by natural processes (solar radiation, evaporation, conduction, and long-wave radiation), no significant weather modifications are expected. In any event, such minor meteorological and climatological changes as might occur would be very difficult to isolate in the natural variability of weather elements.⁵⁶

4.2.5.4.2 Alternate Cooling Systems

By the year 2000, a reasonable assumption is that in some areas of the country, most, if not all, of the large inland sources of cooling water will not be available for once-through cooling. Thus, utilities will use alternative means of cooling and because of economics, these systems will be primarily evaporative or wet systems.

The areas of atmospheric effects of all alternative heat dissipation methods can be separated into four areas of concern: visible plumes, fogging and icing, drift and the effect on local (mesoscale) weather. Other items for concern are consumptive water use, noise, and chemical and thermal impacts of the blowdown.

For evaporative or wet cooling systems, the primary atmospheric effect will be generation of visible (water droplet) plumes due to the recondensation of the water vapor in passing through the cooling device where it mixes with cooler, drier ambient air. If this plume is released at or near ground level, or is brought to the surface by atmospheric eddies or dispersion, ground-level fog will be generated by the thermal discharges.

In areas where land is relatively inexpensive and sufficiently level, cooling ponds and/or spray canals may be used to dissipate the plant's thermal discharges. Experience at operational plants with cooling ponds indicates that the ponds are effective and reliable heat sinks.⁵⁹⁻⁶³ Further, ponds have a considerable thermal inertia; that is, the power plant waste heat need not be dissipated to the atmosphere at the same rate as it is produced, as is the case with cooling towers. There is no noise or drift associated with cooling pond operation.

The primary meteorological effect of cooling ponds is the generation of fog over and near the pond.^{56,60,63} Steam fog is created whenever the air above the pond is sufficiently cold and humid so that it cannot absorb the water vapor evaporated into it by the heated water surface.

The frequency, intensity, and inland penetration of pond-induced fogs are items of concern in pond site selection. Observations made at existing cooling ponds indicate that the steam fog is thin, wispy, and usually does not penetrate inland more than 100 to 500 ft.^{56,60,63-65} Because the water vapor is released over large areas, ponds are not a major source of dense fog, despite the release of the water vapor at ground level. The cooling pond should be located so that the induced fogs (and freezing fogs) do not affect roads and bridges; experience at a cooling pond in northern Illinois indicates that a buffer distance of 300 to 500 ft is sufficient.^{59,65} Some of the water droplets will be removed by vegetation and other surfaces as they move across the nearby land areas, causing a local increase in humidity and dew.

Natural steam fog is fairly common in much of the Nation due to the frequent passage of cold air masses over open water. Because of higher water temperatures, steam fog may form over the heated water in cooling ponds when conditions do not favor natural steam fog.

During periods of subfreezing temperatures, some of the droplets will freeze and create a layer of low-density rime ice on nearby (within a few hundred feet)

vegetation and structures. Observations at existing ponds indicate that this rarely, if ever, causes problems with plants, power lines, or navigation on the pond due to the nature of the rime ice deposited.^{56,60,63} Observations of ice deposited from freezing steam fog from any source, including cooling ponds and towers, show that this ice is similar to that deposited by natural freezing fog; that is, the ice is very light and friable. Thus, the rime ice deposited by the operation of evaporative cooling systems is quite different in its nature and effect on plants and structures from the glaze ice formed by freezing rain and drizzle.

Since spray canals concentrate the heat transfer to the atmosphere, the atmospheric changes due to the extra heat and water vapor will be enhanced. The visible plume created by the spray canal contains some drift droplets in addition to drops of condensed vapor. These drift droplets will tend to be larger than those produced by condensation and add considerably to the wetting and icing potential of the visible atmospheric plume.

In contrast with cooling towers and ponds which have been used for decades, there has been little operating experience with large spray cooling systems, especially in winter. However, experience at a power plant with a spray cooling canal in northern Illinois indicated no serious environmental or fogging problem after two seasons of operation.⁶⁵⁻⁶⁷ Experience with spray canals in Michigan⁶⁶ and New Hampshire⁶⁷ is similar. As with cooling ponds, the fogging and icing effects decrease with distance. It has been shown that a distance of 600 ft from the canal to public roads and switchyards is sufficient to preclude any hazardous conditions arising.⁶⁶

From the limited experience to date, a reasonable expectation is that spray cooling systems will create more severe icing conditions near the spray canal during winter than mechanical-draft cooling towers and cooling ponds, with drift being the primary cause of the difference.

Quantitative estimates of fog and icing potential from spray canals are not possible, in part, because the properties of the air downwind of spray units are unknown functions of ambient weather conditions, water temperature, and characteristics of the spray heads. For most wind conditions, the air will be in contact with the water from the spray for a shorter period than it would be in a cooling tower. Thus, the heat and moisture transferred to the air will be slower, and more air will be modified to cool a given plant load. Sprays are slightly noisier than cooling ponds, due to the pumps and falling water.

The use of wet cooling towers, both natural and forced draft, is a proven economical technique for dissipating waste heat generated by the power plant. If properly designed and located, little adverse impact will be incurred.

Since most atmospheric effects from cooling towers depend on local weather conditions, each site will need to be examined to determine the effect of operation of the proposed cooling system at that site. Observations to date indicate that properly designed and located towers will not create significant adverse effects.

Since a natural draft tower dissipates a large amount of heat and water vapor, there exists a potential for creating an inadvertent weather modification. The warm, saturated air leaving the tower will mix with cooler, drier ambient air; due to the nonlinear relation between saturation vapor pressure and air temperature, the mixture of these two bodies of air will frequently be supersaturated. The excess moisture will condense in the form of an elevated visible plume. Due to its vertical momentum and buoyancy, the visible plume will usually continue to rise well above the top of the tower.

Available observations near natural draft towers indicate that the plumes rarely, if ever, reach the ground. For example, Colbaugh et al.⁶⁸ report that there have been no cases of visible plumes reaching the ground during two years of operation of the Paradise, Kentucky, steam plant. No surface fogs or icing have been observed in four years of operation of the Keystone, Pennsylvania, power plant (1800 MWe). The same results have been reported in England^{69,70} and in the United States.^{62,64,67,68,71-74} These observers report that the visible plumes enter the atmosphere at heights of 100 m or more and usually evaporate completely. Hosler⁷⁵ reported one occasion on which the visible plume from a natural draft cooling tower did reach the ground; this is the only reported case in the literature. Nevertheless, contrary to actual observations taken at tower sites, many theoretical analyses predict frequent tower-induced ground-level fog.^{74,76}

Under certain weather conditions (low temperature, high humidity, moderate wind speed, and a stable atmosphere), the visible plume from a cooling tower may extend several miles. Colbaugh et al.⁶⁸ have measured plumes extending 16 km in Tennessee.

The length and other dimensions of the plume (such as its plume rise, width, depth, height of base above ground) will depend primarily on existing weather conditions--

air temperature, saturation deficit,* wind speed, and atmospheric stability. Because the ability of air to hold water vapor is quite low at low temperatures, the plumes will be most pronounced during the winter season.

Bierman et al.⁷⁷ have published the only climatology of plume lengths available. They measured the length of the plume from a cooling tower complex in Pennsylvania for 6 months in 1969 (January 31st through July). Pictures were taken in early morning, normally the time of day with the longest visible plumes. The plumes were found to have evaporated completely on 81.5% of all days during the period of study. Of these plumes, 87.3% disappeared within five stack heights or 1625 ft of the tower; only 2.6% extended more than 15 stack heights or 4875 ft. The plume merged with an existing overcast on 16.5% of all days. On the remaining days (2.0%), the plumes were classified as "special cases," such as cloud building.

Aynsley⁷⁸ has observed that cooling tower plumes can, if meteorological conditions are proper, create cumulus clouds. He concludes that this is a "rare occurrence," and that these man-made clouds only precede natural cloud formation. He discussed the possibility that a cooling tower plume could somehow trigger an existing atmospheric instability and create extra clouds and precipitation miles downwind of the release point. As the number and size of cooling towers on a given site increase, the probability of significant alteration of cloudiness and precipitation patterns will increase.⁷⁹⁻⁸¹ The state-of-the-art in cloud physics provides little certainty that there will be any increase in rainfall amounts due to cooling tower plumes.⁸⁰ There are at least four reported occurrences of snow showers or ice crystals being generated by cooling towers.⁶⁴ In all cases, the amounts were very small.

In an unpublished report, the Central Electricity Generating Board of Great Britain⁶⁹ reported its findings on the environmental effects of cooling towers. No measurable change in surface relative humidity was detected downwind. The visible plume sometimes persisted for a number of miles downwind, altering sunshine in the area. No drizzle was observed from the towers. Cumulus clouds were sometimes formed, but no cases of showers or precipitation being generated by the plume have been observed.

A small part of the cooling water (estimated to be as low as 0.002%) is carried into the plume without being evaporated. These droplets, called drift, contain whatever

*The saturation deficit is the water vapor content of saturated air (mass per unit volume) minus the actual content.

impurities are present in the makeup water and release these materials into the atmosphere when they evaporate. The effect of salt deposition from natural draft cooling towers on vegetation is expected to be small, even those using saline water for makeup.⁸²⁻⁸⁵

The fog potential from mechanical draft towers is much greater than for natural draft units for the following reasons:

- (1) Mechanical units release their water vapor at a much lower elevation (50 to 80 ft compared to 350 to 500 ft) where winds are weaker, the saturation deficit is less, and the surface nocturnal inversion frequently prevails.
- (2) The plumes are frequently trapped in building eddies.
- (3) Much higher entrainment rates are generated owing to smaller exit diameters, higher exit speeds, and additional turbulence created by the fan.

Although wet mechanical draft towers have been used to cool power plants for decades, there is very little quantitative data available on the vapor plumes which they generate. There are even fewer references to significant adverse impacts of cooling tower operation.^{59,61,63,67} Several studies have reported light rime icing from cooling tower operation.

Plumes from such towers usually rise and do not reach the ground. The towers should be located with their long axes parallel to the prevailing winter winds to maximize plume rise.

Experience at mechanical draft cooling towers indicates that the area affected by ground-level fog is of the order of one-fourth mile,^{59,62,86} and that for a properly located unit, no significant effects from fogging and icing will be generated at greater distances.

Because of higher exit speeds, the drift rate is higher than with natural draft units; a value of 0.004% is possible.⁸⁷ In any event, almost all of the drift that falls to the ground will do so within 300 to 500 ft of the towers.^{59,88}

As illustrated in Figure 4.2-20, current opinion based on field observations indicates that the drift deposition should occur mainly within the station boundaries, and consequently little impact is expected.⁸⁸

4.2-82

DRIFT FROM A SALT WATER COOLING TOWER

Figure 4.2-20

Of the potential effects of wet cooling methods that have been proposed, the possibility that salts contained in the cooling tower drift might damage vegetation has engendered much interest. Drift could be the source of damage from both direct and indirect effects. Salts deposited directly on vegetation can cause osmotic stress if sufficiently concentrated. Root uptake of salt can also cause osmotic stress in some species such as maples. The symptoms can range from "scorching" of the leaves to death of the plant. The border of dead vegetation along some heavily salted highways is symptomatic of such effects. Indirect effects may result from subtle changes in soil physics and chemistry. In this situation, there may be long-term shifts in species composition of the affected communities.

The natural vegetation upon which a set of observations is based must be considered. Ecologists have known for years that even in plant communities extensively exposed to salt, there is a wide range among species' ability to survive and compete.⁸⁹ Consequently, extrapolating observations from seaside vegetation to inland vegetation must be done with caution.

A modeling effort of a natural draft system employing sea water as a coolant led to the observation that even under worst case conditions, all drift droplets which reach the ground will do so in the first 1000 ft.⁸⁸ Furthermore, taking $60 \mu\text{g}/\text{m}^3$ as a plant damage threshold value⁸⁷, then the farthest from the source that damage would be expected is 2200 ft downwind. At this distance, drift is responsible for an increment of $13 \mu\text{g}/\text{m}^3$ of salt, which is equivalent to the additional salt blown and deposited by a wind increase of only 3 mph. Since natural wind fluctuates well over this, it is doubtful that even long-term damage is likely. This example is based on a 134,000 gpm system which is less than a third of the circulating water in many 1000-MWe stations; therefore, salt damage may extend further for such a station than indicated here. This will be partially counterbalanced by the factor of conservatism of the 0.004% drift figure used for the analysis compared with the 0.002% expected drift.⁹⁰ Whatever effects may take place should occur within several thousand feet of the site boundary and fall within the range of natural changes due to minor weather and climatic changes. Empirical data on drift from salt water cooling towers will be available within the near future.⁹¹

The effects expected of a system using a fresh water supply should be negligible, because not only is the absolute amount of salt released far less, but also chloride is not the predominant anion. While apparently large increments for sulfate may occur, the figure is still far less than that expected to cause damage to vegetation.⁹²

While little impact is expected from drift due to the constituents of the makeup water source, the addition of chemicals to the recirculating water may change this conclusion. Where substances such as zinc and chromium are added at concentrations at or near the upper level of common use,⁹³ vegetation may be markedly affected. However, these effects will probably not extend beyond the site boundary. Because these substances are released to the cooling water source in the blowdown, there is already a decline in their use because of water quality considerations. The changeover to nontoxic substances is expected to continue.

Because no water vapor is added to the atmosphere by dry towers, a frequent assumption is that such towers will create no significant atmospheric impact.^{59,94} However, such towers are known to create large vertical air currents which could create clouds and induce precipitation downwind of the plant. Large vertical currents are observed above an operational dry tower in Germany. There is no consumptive water usage and there will, of course, be no drift. However, noise levels for mechanical draft towers could be high.^{94,95}

4.2.6 Environmental Measurements and Monitoring Programs

4.2.6.1 Aquatic Monitoring

Among the requirements to be met by each utility that applies for a reactor operating license is to demonstrate that no unacceptable damage will occur to any aquatic ecosystem affected by operation of the plant. Regardless of the type of cooling system employed (once-through, evaporative towers, cooling lake, spray canals, or dry towers), the nonradiological impacts of an LMFBR on an aquatic ecosystem will be similar to, and in some cases less than, those of other types of steam-electric power plants of equal capacity.

In addition to preoperational aquatic studies, increased future emphasis is expected upon the periodic analysis and assessment of the results of operational aquatic studies and monitoring programs conducted at sites of individual LMFBRs as well as the results of pertinent topical and ecosystem studies. These post-licensing analyses would be performed independently by the utilities conducting the programs and by the AEC in periodic regulatory review of each operating LMFBR, in order to insure the identification of any changes in design or operation of individual plants that may at times become necessary in order to reduce or eliminate any unacceptable occurrences of damage to the aquatic ecosystem not foreseen or underestimated in preoperational impact assessment.

The main purposes of the preoperational aquatic environmental studies conducted by each utility at a proposed LMFBR site will be as follows:

- (1) To provide a physical, chemical, and biological characterization of the potentially affected aquatic ecosystem. Such characterization will need to be adequate to enable reasonably quantitative projections and assessments of the plant's direct and indirect effects on the affected aquatic ecosystem, as well as to enable assessment of the suitability of the plant's proposed site and design features.
- (2) To obtain baseline data for later comparison with results of operational monitoring and study programs.*

The main purposes of the operational aquatic environmental monitoring and study programs conducted at each LMFBR site will be as follows:

- (1) to identify and reasonably quantify specific effects of plant operation on the aquatic ecosystem;
- (2) to enable reasonably quantitative assessment of such effects; and
- (3) to enable the determination of any changes in plant design or operation that might become necessary in order to reduce or eliminate any unacceptable occurrences of environmental damage not foreseen or underestimated in preoperational impact assessment.

In any environmental study or monitoring program, many decisions must be made regarding specific methodologies, frequencies, and locations of sampling, as well as the scope and areas of principal investigative focus. These decisions are based on responsible scientific judgment and must have appropriate data analysis and interpretation.

not to be

The selections of sampling gear and measuring methods employed in the preoperational studies and operational monitoring at each LMFBR site will reflect the types of information needed and experimental questions asked; the characteristics of the aquatic ecosystem being studied; the state-of-the-art regarding ecological system study and related methodologies (which are likely to be different in the year 2000 from their present state), as modified by information available from previous studies; and the type of cooling system used and its potential impacts. As additional information becomes available about the ecosystem being studied or the kinds and

*Another purpose is to provide some guidance to the operational monitoring program. Important or potentially important parameters may be identified which should be monitored in greater intensity than other parameters.

significance of impacts that can reasonably be considered possible, the required aquatic monitoring and study programs will be modified to reflect such information. In some cases, this will result in new kinds of studies. In most, however, it will bring a reduction in scope.

4.2.6.2 Terrestrial Monitoring

The major terrestrial effect of a power plant is the large-scale habitat change brought about by construction activities. The changes in the biota to be expected by building an LMFBR power plant where the former land use was farmland or woodlot are straightforward and can be assessed and predicted by a preconstruction survey of the biota using conventional sampling techniques. Operation of the plant is not expected to have any additional effects. LWR stations currently being licensed contain in their technical specifications the requirement that monitoring programs be conducted to demonstrate the presence or absence of these effects. Consequently, by the year 2000, many of these potential impacts will have been demonstrated not to be of importance and, therefore, no monitoring program to investigate them will be necessary. However, given the present state-of-the-art, the assumption will be made that monitoring programs for all of the impacts discussed will be required where appropriate (e.g., a drift study for a natural draft cooling tower but not for a cooling pond).

The preoperational survey of the terrestrial biota will have proceeded through several stages by the time the LMFBR construction permit application is submitted to the AEC. A procedure in use by some consultants to utilities involves a stepped-up program from the early stages of examination of several candidate sites, to the more detailed analysis of the selected site, usually including at least a year's worth of data. This cumulative information is included in the applicant's environmental report as part of his license application.

The initial stages of this procedure involve qualitative assessment by literature review, aerial reconnaissance, and onsite analysis. The latter usually involves a survey of flora and fauna by visual observation of the biota and their sign, or sometimes by rough quantitative measurements as by several nights of small mammal trapping or vegetation analysis by a line intercept. Site selection is usually determined on information of this nature. At this point, a commitment to long-term sampling is made. This sampling is done for about a year prior to submittal of the construction permit application. This sampling program is not subject to prior review by the AEC regulatory staff. However, since these data form the baseline for subsequent assessments and provide essential guidelines to the staff in drawing up

the monitoring provisions of the technical specifications, the regulatory staff demands that responsible scientific judgment has been shown in the preconstruction sampling program. If, in the judgment of the staff, the data are inadequate, then the license application is not accepted at the time of the acceptance review, resulting in subsequent major delay to the applicant. Consequently, although the regulatory staff is not involved in this program prior to the acceptance review, its post hoc activities regulate the quality of the submittal.

The methodology employed during this period is similar to the ongoing survey requirements of a monitoring program. Many of the potential terrestrial effects can best be detected by examination of the flora. If the plume is blocking necessary sunlight, if drift is changing the soil physics or chemistry or is directly killing vegetation, or if increased humidity is causing plant diseases, then direct examination of individual plants coupled with long-term ecological analysis of plant communities should reveal this.

Experimental design is of basic importance. Test plots must be selected in areas where effects are expected. If a monitoring program is to detect drift damage, it is probably unwise to locate the test plots upwind of the most likely wind direction. Attention must be paid to the details of sampling frequency and replication. Suitable control plots must be set up outside of areas of possible plume influence. The program must be designed so that sufficient ancillary data are gathered so that causality can be determined. This will frequently require gathering associated physical, chemical, and meteorological information. Examination of individual plants will also be necessary, even in long-term studies, to differentiate between, for example, drift damage and disease damage.

The program must also be designed so that long-term change due to plant operation is distinguishable from natural change. Because of shifts in vegetation due to changing short-term climatic fluctuation, changes in ambient air quality, and also from natural succession, it is doubtful that a program of more than several years' duration will provide data suggestive of long-term effects, unless the program has previously indicated short-term effects.

Long-term study of vegetation will depend for its methodology on the nature of the vegetation and topography. In intensively developed or cultivated areas, there may be little natural vegetation for long-term analysis. In farming areas, the data collected for short-term effects may be all that is appropriate.

As part of the preconstruction and preoperational surveys, there will be vegetation sampling. As far as possible, the same plots and same sampling techniques should be maintained. Where the plots are small, sample plots are desirable where there is complete enumeration of all shrub and tree vegetation.

For larger plots, quadrat sampling is a highly useful method. A nested series where trees are sampled in the largest quadrat, shrubs the next largest, and herbaceous vegetation next is frequently employed. Phillips⁹⁶ suggests 100, 16, and 1 m² respectively for these plots. Actual sizes would be best determined in view of the local vegetation.

Under many situations, either a transect (a very long, slender quadrat) or a line intercept (sample of the vegetation that falls along a line, i.e., a transect without width) will provide suitable information. A variety of techniques now in common use, such as plotless methods (e.g., point-quarter method) and variable plot methods (e.g., Bitterlich method), are rapid and provide good data.

No matter what method for gathering data is chosen, as much information as possible should be derived from the program. The data customarily incorporated into the "importance value" by plant ecologists provide a minimum guide. This includes relative density, relative dominance, and relative frequency.

Monitoring of the animal community will generally not be necessary except where special conditions are present. If particular vertebrate species are of interest because they are endangered, protected, or of special value, then a specific suitable program will be required. This may range, for example, from periodic inspection of bog turtle habitats along transmission lines to analysis of breeding bird census data for raptors.

Insect monitoring may be of interest at a particular site. Terrestrial sampling of adult aquatic insects by standard techniques may be suitable for the examination of aquatic impacts such as the thermal plume changing emergence times and thereby affecting the availability of food organisms to fish. A detailed discussion of insect sampling should be consulted for a specific program.⁹⁷

Monitoring should be done in light of well formulated and testable hypotheses. The results must be responsibly interpreted and reported to the AEC regulatory staff so that an independent assessment can be made. In this way, the public interest is safeguarded. With the advent of the International Biological Program, great advances

have been made in data collection, analysis, and interpretation. Emphasis on ecosystem dynamics through varying modeling techniques has made the prediction and assessment of impacts more rational. Given the presumed state-of-the-art in the year 2000, monitoring should provide far more information at less cost than is the case today.

4.2.6.3 Radiological Environmental Monitoring Program

Regulations of the AEC^{98,99} require routine monitoring of radioactivity in the environment around nuclear power plants. Guidance for the design of an acceptable monitoring program is given in USAEC Regulatory Guide 4.1 published by the Directorate of Regulatory Standards¹⁰⁰ and in publications of the U.S. Environmental Protection Agency¹⁰¹ and the International Commission on Radiological Protection.¹⁰²

The purpose of such monitoring is to permit the assessment of radiation dose to the public from operation of the facility and to maintain surveillance for long-term buildup of radioactivity in the environment. Because the radioactivity released from LMFBR power plant is expected to be very low, the limits of detection may possibly preclude its observation in the environment, and environmental monitoring may serve chiefly to provide upper limit estimates of dose along critical pathways. Therefore, estimates of human exposure will depend on detailed isotopic analysis of releases before they become diluted, together with the use of suitable models for dispersion and movement of radionuclides through the environment. These models will, in turn, depend on site-specific data with regard to meteorology, hydrology, demography, land and water use, water chemistry, and local food chains.

The description of an LMFBR power plant monitoring program in detail without reference to a specific site is not possible. Nevertheless, a number of general features may be outlined. Basically, the radioisotopes of concern will be those of interest in an LWR power plant, since the fission product inventory will qualitatively be much the same. However, additional emphasis on analyses for plutonium and transuranium isotopes will be in order, because they will be present in greater amounts in the LMFBR power plant. Such analyses are generally time consuming and require considerable skill, but techniques of suitable sensitivity are presently available and in use.¹⁰³⁻¹⁰⁶ The use of sodium in the reactor coolant loops will require monitoring for its principal activation products--Na-22 and Na-24. However, the decay of these isotopes is accompanied by readily detectable gamma rays, and no particular analytical problems are presented.

The monitoring program will have two phases: a preoperational phase beginning at least two years before plant startup, and an operational phase beginning with plant startup and continuing throughout the life of the plant. The preoperational phase will have several objectives.

- (1) Measurements will be made of environmental radiation levels and fluctuations attributable to natural background, fallout, and other sources. Media analyzed will include air, water, soil, milk, sediment, and aquatic biota. Measurements will emphasize analyses for specific radionuclides, particularly those whose origins may later be subject to doubt.
- (2) Critical population groups, pathways, and radionuclides will be identified. This effort will depend greatly on parallel programs for the accumulation of meteorological, hydrological, and demographic data for the site, together with information on local dietary habits, land use, and trace element analysis of potential receiving waters.
- (3) Sampling and analysis techniques and procedures will be developed and evaluated.
- (4) Appropriate LMFBR power plant staff personnel will be trained in the use of these techniques and procedures.

The operational monitoring phase will begin essentially as a continuation of the preoperational phase. It will be particularly intensive during the first two or three years of plant operation with frequent evaluation of data to develop better understanding of important pathways and behavior of radionuclides in the local environment. This information will be used to make necessary adjustments in the program to achieve more efficient use of resources. For example, if a particular pathway is demonstrated to be insignificant, it will be deemphasized in favor of others of greater interest. A generalized schematic of pathways that will be considered is shown in Figure 4.2-19.

Sampling will focus chiefly on media impinging directly on man, such as air and water; on foods consumed directly by man, such as milk, fish and leafy vegetables; and on media such as aquatic plants, sediments, and benthos that may be sensitive indicators of the presence of radioactivity in the environment. Whenever practicable, samples will be analyzed for specific radionuclides to permit dose estimates to be made for man and important biota. In addition, all potentially radioactive effluents will be monitored at their point of release and analyzed in accordance with Regulatory Guide 1.21,¹⁰⁷ or its equivalent.

In Table 4.2-10 an example of a monitoring program for a general site is given, but it should not be regarded as definitive or necessarily complete. Media mentioned may be inappropriate at some sites, and others perhaps should be added. Sampling frequencies and locations should be adjusted in the light of local conditions. However, it does illustrate a program wherein the major pathways are examined. The detailed program for a specific plant will be subject to the approval of the AEC and will be included in the Technical Specifications, which are a part of the operating license for the plant.

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Table 4.2-10
MONITORING FOR OPERATING LWR POWER FACILITIES

Operation or sample type	Approximate number of samples and their locations	Collection frequency	Analysis type ^a and frequency
Air particulates	1 sample from the 3 locations of the highest offsite ground level concentrations 1 sample from 1-3 communities within a 10-mile radius of the facility 1 sample from a location greater than a 20-mile radius in the least prevalent annual wind direction ^d	Continuous collection—filter change as required	Gross long-lived β at filter change ^b Composite for gamma isotopic analysis, radiostrontium and plutonium analysis ^c quarterly
Air iodine	Same sites as for air particulates	Continuous collection—canister changes as required	Analyze weekly unless absence of radioiodine can be demonstrated
Direct radiation	2 or more dosimeters placed at each of the locations of the air particulate samples which are located at the 3 highest offsite ground level concentrations 2 or more dosimeters placed at each of 3 other locations for which the highest annual offsite dose at ground level is predicted ^e 2 or more dosimeters placed at each of 1-3 communities within a 10-mile radius of the facility ^f 2 or more dosimeters placed at a location greater than a 20-mile radius in the least prevalent annual wind direction ^d	Quarterly	Gamma dose quarterly
Surface water ^g	1 upstream 1 downstream after dilution (e.g., 1 mile)	Monthly (Record status of discharge operations at time of sampling)	Gross β , gamma isotopic analysis ^h monthly. Composite for tritium, plutonium and radiostrontium analysis quarterly ^c
Ground water	1 or 2 from sources most likely to be affected	Quarterly	Gross β , gamma isotopic analysis ^h and tritium quarterly
Drinking water	Any supplies obtained within 10 miles of the facility which could be affected by its discharges or the first supply within 100 miles if none exists within 10 miles	Continuous samples ⁱ	Gross β , gamma isotopic analysis ^h monthly. Composite for tritium, plutonium and radiostrontium analysis ^c quarterly
Sediment, benthic organisms and aquatic plants	1 directly downstream of outfall ^j 1 upstream of outfall ^j 1 at dam site downstream or in impoundments ^j	Quarterly	Gamma isotopic analysis and plutonium analysis
Milk	1 sample at nearest offsite dairy farm in the prevailing downwind direction 1 sample of milk from local dairy representative of milkshed for the area	Monthly	Gamma isotopic analysis and radiostrontium analysis monthly ^c
Fish and shellfish	1 of each of principal edible types from vicinity of outfall 1 of each of the sample types from area not influenced by the discharges	Semiannually	Gamma isotopic analysis semiannually on edible portions
Fruits and vegetables	1 each of principal food products grown near the point of maximum predicted annual ground concentration from stack releases and from any area which is irrigated by water in which liquid plant wastes have been discharged 1 each of the same foods grown at greater than 20 miles distance in the least prevalent wind direction	Annually (At harvest)	Gamma isotopic analysis annually on edible portions
Meat and poultry	Meat, poultry, and eggs from animals fed on crops grown within 10 miles of the facility at the prevailing downwind direction or where drinking water is supplied from a downstream source	Annually during or immediately following grazing season	Gamma isotopic analysis annually on edible portions
Quality control ^k	Samples as required for accurate sampling and analysis	Annually	Minimum frequency—annually
Soil	Same locations as for fruits and vegetables	Annually	Gamma isotopic analysis and plutonium analysis

^a Gamma isotopic analysis means identification of gamma emitters plus quantitative results for radionuclides that may be attributable to the facility.

^b Particulate sample filters should be analyzed for gross beta after at least 24 hours to allow for radon and thoron daughter decay.

^c Radiostrontium analysis is to be done only if gamma isotopic analysis indicates presence of cesium-137 associated with nuclear power facility discharges.

^d The purpose of this sample is to obtain background information. If it is not practical to locate a site in accordance with the criterion, another site which provides valid background data should be used.

^e These sites based on estimated dose levels, as opposed to ground level concentrations where the dose may be affected by sky shine, high plumes, or direct radiation from the facility being monitored.

^f These locations will normally coincide with the air particulate samplers used in the monitored communities.

^g For facilities not located on a stream, the upstream sample should be a sample taken at a distance beyond significant influence of the discharges. The downstream sample should be taken in an area beyond the outfall which would allow for mixing and dilution. Upstream samples taken in a tidal area must be taken far enough upstream to be beyond the plant influence when the effluent is actually flowing upstream during incoming tides.

^h If gross beta exceed 30 pCi/liter.

ⁱ Drinking water samples should be taken continuously at the surface water intake to municipal water supplies. Alternatively, if a reservoir is used, drinking water samples should be taken from the reservoir monthly. If the holding time for the reservoir is less than 1 month, then the sampling frequency should equal this holdup time. Increases in concentration of activation and/or fission products at these sources necessitate the analysis of tap water for the purpose of dose calculations. Additional analyses of tap water may be necessary to satisfy public demand.

^j See figure 6 for locations on a stream. For facilities located on large bodies of water, sampling sites should be located at the discharge point and in both directions along the shore line.

^k Analytical quality control will consist of participation in an AEC approved intercomparison program such as the Analytical Quality Assurance Program (Radiation) operated by the Environmental Protection Agency's National Environmental Research Center - Las Vegas and the documentation of analytical techniques, instrument calibration procedures and frequencies, and vendor and standardization procedures for all reference radionuclide sources employed for calibration or intercomparison purposes.

4.2.7 LMFBR Plant Accidents

4.2.7.1 Summary

Many of the characteristics of an LMFBR lead to a design that is intrinsically safe in many respects, and these characteristics can be exploited to enhance safety. On the other hand, the potential exists for accidents in an LMFBR, or any other reactor, and steps must be taken to prevent accidents and to limit the consequences of accidents.

A number of safety issues associated with LMFBRs have been identified. The LMFBR Program Plan summarizes these issues, provides a general assessment of the technology related to the various issues, and identifies areas where improvements in the technology are desired. An extensive research and development program is continuing on LMFBR safety for the purpose of assuring that a technologically sound base exists for the design, construction, and operation of commercial LMFBRs.

The LMFBR Program is being developed within a framework of safety design and evaluation that has evolved in this country over a period of years. Continued evaluation of specific regulatory criteria is expected as additional results from the program are obtained. Extensive experience is available; a number of sodium-cooled reactors have been or are being built at sizes up to 600 MWe. Information is available from earlier conceptual studies of 1000-MWe LMFBRs, results to date of the research and development program, and detailed safety evaluations of and experience from plants such as SEFOR, EBR-II, and the FFTF. This base is used in making projections of the potential for and consequences of accidents in commercial LMFBRs. There is considerable flexibility in the specific design features that may be employed to achieve the requisite degree of safety for commercial plants. It should be emphasized that a detailed design analysis takes place to assess specific safety aspects of a particular plant. In a Statement such as this, which does not refer to a specific design, there are, necessarily, only relatively general statements that can be made.

An important aspect of design is that extensive evaluations of postulated accidents are conducted before the nuclear plant is constructed, and design features are provided to assure that significant consequences of such events are avoided or safely accommodated. To this end, several general classes of faults or accidents are typically postulated (radioactive waste system failures, fuel handling accidents, fuel failures, steam-generator failures, pipe breaks). Capabilities exist (through such means as selection of conservative design features) to accommodate or prevent such events in large LMFBRs. However, technical uncertainties remain to be resolved

or minimized through continued research and development. Design conservatism can be utilized to accommodate any uncertainties not fully resolved.

A key aspect of LMFBFR safety is the capability of safety systems to respond, when required, to malfunctions and incidents so that serious accidents do not result. Rigorous standards for plant-protection-system performance already exist. Continued emphasis is being given to research and development efforts to assure that suitably reliable protection systems are available and proven for use in commercial LMFBFRs.

With the use of a system for removal and retention of radioactive gases and with the affinity of sodium for retaining fission products, fuel failures from both anticipated and unlikely faults would not be expected to result in a significant release to the environment.

Relatively large inventories of radioactive noble gases are retained in the gas processing systems, and major failures of these systems have been postulated. Such failures would result in releases; however, the gases would be recycled or retained within the cells housing the system. In either event, the potential releases do not constitute a hazard to the public.

Another accident that is usually postulated involves fuel handling. The transparency of water makes refueling in LWRs relatively straightforward. The opacity of sodium and its chemical reactivity with air and water require fuel handling methods more complex than those in LWRs. Redundant components and multiple containment barriers can be employed to minimize the potential for serious accidents and render any resultant releases relatively inconsequential.

Experience to date has shown that one or more sodium leaks, generally small in extent, may occur during a plant lifetime. Early detection and control of sodium leaks is a major requirement. A number of systems have been used in existing sodium-cooled reactors; others are under development. The AEC has concluded that means can be provided to adequately deal with leaks, and further research and development will provide an improved base for selection of optimum systems in commercial LMFBFRs.

Means to assure there is no propagation of local fuel failures into whole core accidents has been and continues to be an area of emphasis in the safety research and development program. Design features that have been used and are available

include built-in features to prevent blockages and instrumentation to detect and locate failures. From existing experience and results of the research and development program, some fuel failures will occur but will not result in significant releases of radioactivity. Efforts are under way to assure that situations do not arise that lead to extended core damage. Experience from existing and planned facilities and the results from future research and development will provide the base for decisions on the types of detection instrumentation to employ.

Steam-generator faults are examined; while failures have occurred, they have not resulted in any significant public safety hazards. Means are available or are being developed to assure that the consequences of steam-generator faults are safely accommodated.

In addition to the accidents identified above, nuclear power reactors are typically required to safely accommodate very severe accidents. These accidents are analyzed in establishing performance requirements for safety systems. Severe natural phenomena, large sodium fires, and failures of the heat transport system (such as a pipe break) are the events that typically dominate containment system design. The consequences of these events are strongly design dependent, although in all cases the AEC has concluded that features can be provided to accommodate them safely. Substantial research and development efforts are under way to reduce the technical uncertainties in the assessment of these accidents and consequently provide a base for selection of appropriate design features for commercial LMFBRs.

An area of investigation that has received widespread attention in LMFBRs is an accident that leads to substantial damage to the core. With a core that is not in its most reactive configuration and with sodium voiding producing another positive reactivity feedback mechanism, a substantial nuclear excursion in a fast reactor has been considered as at least a theoretical possibility deserving study and by some a clear requirement to be accommodated in design. Through the process of minimizing the possibility for a wide variety of accidents, efforts have and will continue to be made to prevent any potential fault from progressing to core disruption. There are opposing views regarding the likelihood of reducing the accident risk to such a low level that core disruptive accidents need not be considered in design. If a core disruptive accident is assumed to occur, substantial damage to the core and releases of fuel, fission products, and sodium to containment spaces could occur. Current assessments indicate that the consequences of core disruptive accidents would be safely contained within the reactor primary coolant boundary, although there are uncertainties that require resolution or improved understanding

if the resultant consequences are to be well characterized. A substantial research and development program to specifically investigate such accidents is under way. This program, together with other efforts on preventing or on providing early detection and control of possible events that could lead to core disruption, should provide a base for decisions on treatment of core disruptive accidents in commercial LMFBRs.

Conservative design concepts are used, and supplemental features are evaluated which provide capabilities for accommodating substantial mechanical loadings (crush columns), failures of the coolant boundary in such a way as to release large quantities of fission products, sodium, and fuel (double containment, air cleanup systems), or failure of all normal heat removal systems (ex-vessel post-accident heat removal concepts). If required, because of unfavorable results in the research and development program, such supplemental features could be employed and research is under way to develop effective systems for these functions so that they will be available if needed.

The AEC has concluded that, notwithstanding the uncertainties in this area, substantial capabilities can and will be provided to the extent necessary to accommodate the consequences of a range of core disruptive accidents. The ongoing research and development program will serve to develop a better understanding of the risks associated with such events and means to further minimize such risks.

In conclusion, a substantial base of safety-related information exists. Safety issues associated with LMFBRs have been extensively investigated, and there are means of accommodating potential risks so as to assure that LMFBRs are designed and operated safely. Research and development continues with the objective of providing a better understanding of potential accidents and their consequences, so as to provide a sound technological base for commercial LMFBRs. Design options are available and under investigation which would be used to achieve requisite levels of safety even if unfavorable results in the research program should occur.

4.2.7.2 Introduction

Section 4.2.3 describes typical LMFBR reactor plant features and briefly addresses safety-related considerations. This section draws upon and supplements Section 4.2.3 with the objective of providing perspective regarding the potential for accidents and for accidental release of radioactive materials from the general class of large commercial LMFBRs which could be designed, constructed, and operated over the next few decades. The approach employed is to (1) set forth a general description

of the requirements for safety that will have to be met in any specific commercial LMFBFR and (2) summarize analyses of postulated accidents, focusing upon the generic features of this class of reactors relative to their capability to assure that such accidents can be safely accommodated.

As noted earlier, firm designs for commercial LMFBFRs have not yet been developed. However, a fairly broad range of safety-related plant features were evaluated in the earlier 1000-MWe conceptual design studies.¹⁰⁸⁻¹¹⁴ The studies were of a scoping nature rather than final analyses. They have served to identify general features of large plants, to provide a base for general risk and safety assessments of large LMFBFRs,¹¹⁵ and to aid in defining long-range development needs. A substantial body of experience already exists, and will be extended, from the construction and operation of the sodium-cooled reactors that have been or are being built at sizes of up to 600 MWe.¹¹⁶⁻¹²⁰ Information from the 1000-MWe studies, together with experience from construction and operation of smaller plants, and results from the extensive ongoing research and development program^{121,122} will provide a base for ensuring that important safety-related factors of various design alternatives are well understood; these three activities therefore form an important step in establishing the capability of a specific large LMFBFR to meet necessary safety requirements.

The LMFBFR program to achieve this base is described in Section 3. The LMFBFR Program Plan¹²³ provides a general summary of the various issues and identifies areas where improvements in the technology are desired. As noted in the Program Plan:

Safety is a major technical aspect of the national development program just as public safety has been a fundamental concern in all the programs of the U.S. Atomic Energy Commission. The AEC carries on an extensive safety research and development program directed toward understanding the behavior of reactor systems, components and materials, as well as the basic laws governing their thermal, hydraulic, mechanical, chemical, and neutronic behavior, as they relate to safety. The output of this Safety program constitutes important input to other areas in the AEC reactor development program and is an important source of information to be used by private industry in the design and operation of commercial nuclear power plants. In addition, the Safety program provides the AEC with a safety-assessment capability with respect to its regulatory responsibilities. As larger and more advanced reactors such as the LMFBFR are built, improved safety criteria, analyses and evaluations are being developed to help ensure the safety of the new plants. The primary responsibility for the safety of a reactor plant and the technical justification for construction permits and operating licenses remains, of course, with the owner-operator...

Because safety is an integral aspect of every phase of any plant design and operation, each of the other Elements of the Program Plan has vital

safety-related activities. In addition to the factors intrinsic in safe reactor design, safety criteria for sodium-cooled fast breeder reactors must appropriately take into account several significant factors: the hypothetical supercritical transient; sodium voiding; sensitivity to dimensional changes in the core; use of a high-temperature, chemically reactive coolant; high power density; and high irradiation levels.

Essentially all the major industrial countries of the world are developing liquid metal cooled fast breeder reactors on a national priority basis, and each of these nations has committed significant resources to test facilities and demonstration plants. As a consequence, substantial technical experience is available from such fast reactors as EBR-II, RAPSODIE, DFR, and BR-60, which are currently operating, as well as from SEFOR and Fermi. A conservative approach has been used in providing safe designs and safety systems for fast reactors. Technology that already exists or is being developed assures that larger plants can be built and operated safely. Indeed, considerable progress has been made in improving the understanding of the mechanisms and dynamics of postulated accidents for fast reactors, and in defining the margins of safety. The extensive AEC base technology and development programs now underway will provide the additional detailed understanding needed to assure the success of the LMFBFR program objectives.

In this section on LMFBFR plant safety, a summary description of the regulatory requirements is provided, first to provide perspective of this framework and second to identify likely constraints on facility design and on the potential for, and consequences of, accidents. This description is followed by a brief summary of general safety-related characteristics of LMFBFRs and by an assessment of various accidents considered in LMFBFR safety analyses. Key safety issues are discussed generally in the context of an assessment of the design options available and likely to be used to accommodate postulated accidents or to prevent their occurrence. Necessarily this discussion involves some forecasting as to the research and development results and design evaluations accomplished in achieving the dual goals of improved performance and safety. For a more detailed review of the safety program, the reader is directed to refs. 121-123.

4.2.7.3 General Safety Requirements

The LMFBFR safety program is being developed within a framework of safety design and evaluation that has evolved in this country over a period of years. The overall licensing framework is briefly summarized in Section 4.2.1. As noted therein, present regulations are directed primarily to light water reactor power plants. Traditionally, the procedures for the reviews of the safety of fast reactors have been based on the same basic philosophy and practices as exemplified in ref. 124. Indeed, several early sodium-cooled reactors obtained licenses through these procedures (SEFOR and Fermi). An appreciation of this framework is essential in evaluating the steps that have been and are being taken to assure that

LMFBRs will be safe. To this end, a general summary of this framework (taken from ref. 125 and specifically referring to current LWRs) follows:

A principal aim of the AEC in reviewing the design, construction, and operation of nuclear power plants is the protection of the health and safety of the public. This objective is carried out through the safety review process by the regulatory staff, the Advisory Committee on Reactor Safeguards, and the Atomic Safety and Licensing Boards. It is backed up by an inspection program by the staff to assure that all permit and license conditions are followed during plant construction and operation. Nuclear plants are allowed to be built and operated only after careful and detailed reviews at both stages that show the plant design and method of operation meet prescribed safety standards and criteria.

The Commission's standards and criteria require that the plant design be conservative, that there are ample safety margins in operation, and that redundant components and systems are provided to compensate for the fact that no body of knowledge can ever be complete enough to reduce residual uncertainties to zero. Thus, although the operation of nuclear power plants is not completely risk-free, the general safety objective of the AEC is to assure that the risk from normal operation and from design basis accidents is reduced to an acceptable level and to assure that the likelihood of more severe accidents is extremely small.

This safety objective is achieved through a comprehensive quality assurance program for the design and construction of the plant, the provision of multiple safety systems and physical barriers to prevent the release of radioactivity, and by extensive testing and inspection of plant operation. Operating experience provides a practical test of whether the safety objective is being achieved. Although the operating experience to date does not provide a sufficiently broad statistical basis for a quantitative determination of the probability of occurrence of low-probability accidents, the fact that no member of the general public has ever been injured by radiation from an accident at a licensed nuclear power reactor provides some evidence of successful results. Further, the lessons of operating experience as applied to the design, construction, and operation of each new plant help assure an even greater degree of safety for new units.

The starting point in the AEC's multi-layered safety concept is to require that nuclear power plants be designed and built so that, with a high degree of reliability, they will operate without failures or malfunctions that could lead to accidents. This requirement involves designing the plant to conservative standards and engineering practices so that in all phases of operation there will be a large tolerance for operator errors, off-normal operation, and component malfunctions.

The design requirements for nuclear power plants are set forth in the General Design Criteria that form Appendix A to 10 CFR Part 50. Appendices E, G, H, and J of Part 50 also set forth other requirements for various aspects of design and operation, as does the proposed Appendix I. Paragraph 50.55a of Part 50 sets forth codes and standards for nuclear power plants. Quality assurance criteria for all phases of plant design, construction, and operation are set forth in Appendix B to Part 50. The criteria are implemented for a particular plant by the detailed quality assurance programs of the applicant. These programs are carefully reviewed by the staff and amended as required. Field reviews and inspections by the regulatory staff provide assurance that the criteria and detailed plans are followed and that the as-built structures,

systems, and components of the plant important to safety reflect all of those features that are needed for safe operation.

Considerations of quality assurance extend into all phases of plant design and construction. They include such things as the requirement that fuel, coolant, and core structural materials be selected such that in combination they will have inherent stability and safety characteristics as verified by data from experiments and by measurements made during plant startup and operation. Materials used in the safety-related portions of the plant must have properties that are well-known from prior measurement, not only in the as-built condition but also throughout service life. Safety margins in design are required, such as the provision of extra strength in structural members beyond that known to be needed for anticipated operating conditions. Such measures provide for changes of properties with time including those that may be due to radiation effects from operation of the reactor.

Plant systems are designed to allow continuous or periodic monitoring of components for signs of wear or incipient failure. Important safety systems are required to be redundant. The objective is to ensure that no failure of an individual system will deprive the plant of any needed function. In summary, the overall design aim is to assure that the plant is soundly and conservatively designed so that it can be built, tested, operated, and maintained in accordance with stringent quality standards and conservative engineering practices, and with a high degree of freedom from faults and errors.

The next level of safety is based on the belief that, in spite of the assurance offered by careful plant design and construction, it is necessary to anticipate that some incidents or malfunctions will occur during the service life of a nuclear power plant, and to provide measures to forestall or cope with such events. Examples of these measures include the reactor protection system with which we are concerned in the matter of ATWS. The protection system is provided to reduce reactivity quickly and reliably if plant conditions should require such action. Design features are included in all plants to limit the rate at which reactivity and reactor power can be increased by the movement of any element of the control system. The electrical power supply to plant systems, important to maintain safety margins, is provided through two physically independent circuits from the offsite electrical transmission network, and in addition, from an onsite electrical system which consists of two or more independent and redundant generators for emergency use. Another example of measures to deal with operating incidents is the provision of leak detection systems of various designs to provide sensitive indication of incipient failures of fuel cladding, or leakage from the primary coolant system, well before such leaks become operating or safety problems.

The third level of safety, unique to nuclear plants, is based on the belief that it is prudent to go beyond the levels previously mentioned and to incorporate additional systems and margins in the plant design to protect the public even in the event certain highly unlikely accidents should occur. To establish these additional margins, major failures of plant components and systems are postulated and the accident sequences that would follow therefrom are analyzed. A series of these postulated events is established as a set of design basis accidents, and safety systems are designed to control them.

The plant conditions considered in the full safety analysis, therefore, range from relatively trivial events with essentially no risk of release of radioactivity to design basis accident situations that, although very unlikely,

have potential for serious consequences. This range of accidents is categorized by classes as set forth in the environmental reports for nuclear plants. The various classes of incidents and accidents can be put into three major groups as follows.

1. Anticipated operating occurrences and other events leading to no significant releases of radioactivity (Class 1 and 2 accidents). This group of events includes such things as turbine trip, loss of electrical power from offsite sources, partial loss of feedwater flow or reactor coolant flow, and improper control rod withdrawal.
2. Events with the potential for small releases of radioactivity from the plant (Class 3 to 5 accidents). This group of events includes such things as a partial failure of the radwaste system, malfunctions leading to leakage of gaseous fission products from fuel rods into the reactor primary coolant, and steam generator tube failures.
3. Events that are postulated to establish performance requirements for safety systems for the plant (Class 6 to 8 accidents). The events in this group are considered design basis accidents and include refueling accidents, loss-of-coolant accidents, steam line break accidents, and rod drop or ejection accidents.

All of the events listed above in the three major groups of accidents have the common property that they are within the design basis envelope for a nuclear power plant. The potential consequences of all of these events, even the very unlikely and extreme ones, are well within the guidelines established by the Commission's regulations on siting (10 CFR Part 100). The reason for this relatively low level of possible consequences from the entire range of design basis accidents is that all of these events are provided for in the design of the plant. Special safety features, called engineered safety features, are provided to control them. Engineered safety features include such systems as emergency core cooling systems, containment sprays, containment atmosphere control systems, special filtering systems, and the various designs of reactor containment. The net effect of all of the engineered safety features in a plant is to provide a series of overlapping safety systems and multiple physical barriers to the escape of radioactivity outside the plant boundary, and thus to reduce the consequences of all design basis accidents to acceptable levels.

The envelope of design basis accidents does not include all events that are conceivable for a nuclear power plant. It is not necessary, nor even possible, to design nuclear power plants, or any other man-made device or system, for all conceivable eventualities that are physically possible. Thus, at the very low probability end of the spectrum of all possible events there is a residuum of conceivable accident sequences that could, if they occurred, lead to radiological consequences outside the plant boundary in excess of the Part 100 guidelines. These are the Class 9 accidents mentioned in AEC environmental impact statements. It is the judgment of the AEC that these events are of such low likelihood as to present in sum an acceptable risk in view of the benefits derived from the electricity produced by a nuclear power plant.

In attempting to deal with events that have a very low likelihood of occurrence, there is difficulty in establishing probability numbers that can be "proved." We have, in the United States, statistics that give reliable probability numbers for the likelihood of an individual being involved in an automobile accident in any given year. This is because there are, unfortunately, a very large number of such accidents each year and the statistical base for the

probability determination is more than sufficient for the purpose. However, for postulated events of very low likelihood, that might occur only once in a thousand years or more, and have not occurred thus far, there is simply no base of experience from which to obtain probabilities in a direct way. This is the situation with regard to accidents that fall outside the design basis envelope for a nuclear plant. Such accidents may involve sequences of failures, each one of which is, in itself, relatively unlikely. The determination of the probabilities of accidents more severe than the spectrum of design basis accidents, then, is necessarily a matter of judgment. Experience with various components and systems roughly similar to those in nuclear plants provides some guidance in making these judgments, where such components and systems exist in other areas of industrial technology. The regulatory staff uses such experience, together with its best judgment based on all relevant experience, in nuclear plants and elsewhere, in estimating the likelihood of significant accidents and in determining whether they should be within the design basis envelope or, alternately, are sufficiently remote in likelihood so as to be an acceptable risk.

In establishing the boundary between accident sequences that are to be within the design basis envelope, and hence for which engineered safety features are provided, and accidents that reasonably may be assigned to that small residuum for which no further protective features are considered necessary, the regulatory staff uses the safety objective that the risk to the public from all reactor accidents should be very small compared to other risks of life such as disease or natural catastrophes. The staff believes this safety objective is met by requiring a design basis accident envelope that extends to very unlikely postulated accidents, and by establishing the further objective that accidents not included in the design basis envelope should have an average recurrence interval of at least a thousand years for all nuclear plants combined.

For an anticipated population of about one thousand nuclear plants in the United States by the end of the century, the safety objective will require that there be no greater than one chance in one million per year for an individual plant of an accident with potential consequences greater than the Part 100 guidelines. Since plants now being designed and constructed are expected to have service lives approaching 40 years, and may thus be part of the century-end population, the staff believes it appropriate to consider their designs in the light of this future requirement. In view of the difficulty of determining such a low probability, the staff regards this number as an 'aiming point,' or design objective, rather than as a fixed number that must be demonstrated for a given plant design.

As noted above, the licensing framework is implemented through the AEC's safety review process. Section 50.34 of 10 CFR 50 of the AEC regulations requires that each application for a construction permit or an operating license include a safety analysis report and specifies, in general terms, the information to be supplied. Taking cognizance of the differences in plant design and technology between light-water reactors and LMFBRs, the AEC regulatory staff has recently prepared an LMFBR Edition of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants.¹²⁶ As noted in that document, the safety analysis report (SAR) is the principal vehicle for conveying information necessary to permit review of an applicant's conclusion that a given facility can be built and

operated safely. Accordingly, the SAR format and content document reflects the overall requirements for presenting information that objectively supports the applicants conclusion. In this regard, the LMFBFR standard format and content document outlines in considerable detail the principal information that is required by the regulatory staff in its evaluation of an application. The licensing procedures do not identify what specific design features are required but, rather, identify performance requirements for systems selected by the applicant. For example, Section 4.0 reads "...the applicant should provide an evaluation and supporting information to establish the capability of the reactor to perform its safety functions throughout its design lifetime under all normal operational modes, including both transient and steady state, and accident conditions." (ref. 126, 4.0-1) While continued evolution of specific regulatory requirements for LMFBFRs can be expected, the general safety requirements as outlined above already exist and provide a reasonable basis for projections to commercial LMFBFRs. Requirements specific to LMFBFRs will be developed as appropriate during the licensing process. Interim criteria for the CRBRP were transmitted to PMC on July 19, 1974 by AEC/REG.

Consistent with the approach summarized in Section 3, "LMFBFR Program," the goal of the AEC's LMFBFR safety program is to assure that information is made available that will permit development of commercial plant designs that will fully comply with applicable licensing requirements. In the following sections, key safety issues are discussed in the context of an assessment of the design options available and likely to be used to accommodate postulated accidents or to prevent their occurrence. To the extent practicable, results from the aforementioned 1000-MWe studies¹⁰⁸⁻¹¹⁴ are cited as the base point, modified to account for subsequent research and development efforts and detailed design studies and operating experience of recent fast reactors such as SEFOR and the FFTF. These 1000-MWe designs were reviewed in 1968 in an International Conference on Sodium Technology and Large Fast Reactor Design.¹²⁷ Principal features of these designs and of a version of the LMFBFR demonstration plant are summarized in Annexes A and B to this section.

As noted in ref. 125, postulated accidents are broadly categorized into three major groups: (1) anticipated operating occurrences and other events leading to no significant releases of radioactivity, (2) events with the potential for small releases of radioactivity from the plant, and (3) events that are postulated to establish performance requirements for safety systems for the plant. This same grouping is utilized in subsequent sections for the purpose of discussing various LMFBFR plant accidents and their significance.

4.2.7.4 General Safety-Related Characteristics of an LMFBR

In the LMFBR, relatively high (typically 15 to 20%) plutonium fuels, a compact core with a high-energy neutron spectrum, and a coolant that efficiently removes heat are important features in attaining the breeding objective. As a direct consequence of selecting plant features that will attain the desired objective (be it to develop an LWR, HTGR, or LMFBR), certain unique or inherent safety-related characteristics are evident. A given characteristic is seldom wholly positive or wholly negative. For example, light-water reactors have negative void coefficients but they use high-pressure coolants. LMFBRs have positive void coefficients but they use low-pressure coolants. In both cases, the overall power coefficient is negative, assuring an inherent resistance to reactivity transients. Similarly both reactor systems are designed to withstand substantially greater than normal pressures without serious plant damage.

A number of comments were received which argued that LMFBRs were inherently dangerous.* For example, Ilene Younghein** stated:

Because of the high heat of a fast breeder reactor and the great toxicity of plutonium, fast breeders are to regular reactors as lightening [sic] is to lightening bugs.

NRDC,[†] quoting from the report of the Pugwash Conference, stated:

None of the problems described in the foregoing sections (which included releases from accidents, natural disasters, etc.) are significantly diminished in the case of the breeder reactors, most especially the liquid metal cooled fast breeder reactor now favored by all countries, and some are significantly aggravated...Additionally, some aspects of the reactor safety problem would appear to be compounded in the case of the breeder.

As noted above, safety-related characteristics of reactors are seldom wholly positive or negative. As was stated in Section 4.2.7.1, safety issues associated with LMFBRs have been extensively investigated, and there are means of accommodating potential risks so as to assure the LMFBRs are designed and operated safely. This matter is treated in more detail throughout Section 4.2.7. To aid in understanding how the general safety requirements (summarized in the preceding section) are

* (1) Comment Letter 7, pp. 13-15; 8 enc. 1, p. 6, enc. 2, p. 6, enc. 9, p. 30; 16 enc., p. 37; 58, p. 2; (2) Hearings Transcript, pp. 48, 107, 111-112.

** Ilene and Gaylord Younghein Comment Letter 8, enc. 1, p. 6.

† Hearings Transcript. p. 48.

implemented in LMFBRS so as to assure that LMFBRS will be safe and not inherently dangerous, this section provides a discussion of general safety-related characteristics of LMFBRS.

The EPA, in its comment letter,* suggested that the safety characteristics of LMFBRS and LWRs be compared. In the following sections, typical characteristics of LMFBRS are compared and contrasted with analogous features of the more-widely-known light water reactors, where such comparisons can usefully serve as an aid in developing an understanding of the LMFBRS. Because of the overall interrelationships of various safety features and characteristics, a relative comparison of any one safety feature among reactor types should not be considered a quantitative comparison of the overall comparative safety of these types; such comparisons, if found desirable, would require an extensive evaluation of numerous design details which are not yet available for LMFBRS power plants. Notwithstanding the lack of specific design details and failure analyses for future LMFBRS, a system of nationally recognized codes and standards, including AEC regulatory criteria and guides, has been developed and is being expanded. Application of these codes and standards to LMFBRS (as discussed in Sections 4.2.7.2 and 4.2.3.2) will assure that a minimum level of quality and performance is provided. Simply put, adherence to these rules of practice will assure that licensed LMFBRS will be safe and that risks of potential accidents will be within the general requirements set forth in Section 4.2.7.3.

4.2.7.4.1 Inherent Features

The principal characteristics of an LMFBRS, important to safety, center on the use of sodium as the coolant and on a compact reactor core with a large plutonium content. A general summary of these characteristics, and examples of resultant design implications, is provided in Table 4.2-11.

Reactor Coolant

The high boiling point of sodium (1619°F at 1 atm) permits operation at high reactor coolant outlet temperatures (1000 to 1100°F) in low pressure systems. As a consequence, LMFBRS achieve a high thermal efficiency (about 40% compared to about 33% in LWRs), which is desirable from an economy and waste heat point of view, at system pressures determined only by the pumping requirements (i.e., at most a few hundred pounds per square inch compared with 1000 to 2000 psi in LWRs). Consequently major "blowdown" forces and large scale coolant vaporization on depressurization (assuming a pipe failure) are not important factors as they are in LWRs.

*Comment Letter 53, p. 14.

TABLE 4.2-11
 LMFBR CHARACTERISTICS
 (as compared with LWR)

I. LMFBR Advantageous Characteristics	Design Consequences
. Low Coolant Vapor and Operating Pressure . Good Coolant Heat Transfer--All Flows	No "Blowdown" Accident (Jet Forces, Mechanical Damage, etc.) Building Design Pressure Set by Sodium Fire Excellent Convective Cooling No Required Additional Pump Inertia Large Margin to Fuel Damage
. Coolant Pressure/Temperature Insensitivity	Large Temperature Variations Allowable Without Excess Coolant Pressure
. Good Power Shape Stability	Negligible Power Shifts with Burnup Rod Position Xenon Temperature Load Following Transients No Partial Rods
. Lower Excess Reactivity	Fewer Control Rods for Cold Shutdown
. Austenitic Stainless Steel Reactor Vessel	No Low Temperature Brittle Behavior

Table 4.2-11 (continued)

II. LMFBR Disadvantageous Characteristics	Design Consequences
. Smaller Prompt Neutron Lifetime and Smaller Delayed Neutron Fraction	. More Sensitive to Reactivity Variations (However Not a Dominant Effect Over All Anticipated Design Conditions)
. Positive Coolant Void and Core Compaction Coefficient	. Requirement for Greater Attention to Coolant Boiling/Fuel Melting . Core Clamp--Close Control of Bowing
. Sodium Chemical Reactions (with Water and Air)	. Secondary Loop . Steam Generator Rupture Relief Systems . Inert Gas Filled Vaults . Building Designed for Sodium Fire . Mobiltherm and N ₂ Auxiliary Cooling (vs Water)
. High Coolant Radioactivity	. Design for Effects of Sodium Fire . Shielded Primary Vaults . Delay for Coolant Decay Prior to Maintenance (~10 Days)
. No Simple Soluble Poison System	. More Dependence on Mechanical Devices for Alternate Shutdown Systems

The high thermal conductivity of sodium relative to water (about 100:1) leads to fuel-to-coolant heat transfer coefficients that are largely insensitive to coolant velocity. This characteristic together with the large subcooling during normal operation (500 to 600°F at outlet) essentially makes LMFBR reactor cores insensitive to short-term flow reductions (inherent resistance to fuel damage from short-term transients). On the other hand, the rapid response of the coolant to power level or temperature changes in the core (due to sodium's high thermal conductivity) can give rise to relatively rapid thermal transients in the primary coolant system, which must be accommodated in the design.

These same features of sodium permit decay heat removal by natural circulation, independent of the need for any pumping power. Taking full advantage of this potential places certain demands upon the design (careful layout of the heat transport system elevations, for example). But if those demands are met, then under a wide range of postulated accident conditions, the reactor can be cooled by natural convection in the absence of any electrical supply and without requiring the active function of any component following a reactor shutdown.

The high chemical affinity of sodium for water and air and the relatively high radioactivity induced in the sodium during passage through the reactor core require that LMFBR reactor coolant circuits be isolated from the environment. LMFBRs typically employ intermediate sodium circuits (nonradioactive sodium) between the reactor coolant circuits and the steam circuits. Current designs also include placing the radioactive sodium circuits in an inert atmosphere to preclude contact of radioactive sodium with air, although this feature may not be necessary (some European LMFBRs do not employ the concept, for example). Because of the radioactivity of the coolant (principally 15-hr half-life Na-24) several days of decay are required before maintenance of primary coolant systems can be accomplished. A postulated large sodium fire is typically a design basis event for an LMFBR.

Reactor Fuel

LMFBRs employ higher fissile-to-fertile fuel ratios (by factors of 4 to 5) and operate at higher power densities (factors of 4 to 5) than LWRs. LMFBRs contain two to three times more plutonium than the amount that accumulates in LWRs of comparable power rating. The higher power densities in LMFBRs also lead to larger concentrations of short-lived fission products; however, total fission products rather than concentrations are of primary importance and these are comparable in LMFBRs and LWRs of the same power level. An estimate of relative inventories is provided in Table 4.2-12.

TABLE 4.2-12

TYPICAL FISSION-PRODUCT INVENTORIES OF LWR AND LMFBR REACTORS (CURIES)

Isotope	1000-MWe LWR	1000-MWe LMFBR
Kr-85	1.1×10^5	5.7×10^5
Sr-90	8.0×10^5	3.2×10^5
Nb-95	1.3×10^8	1.0×10^8
I-129	0.5	0.25
I-131	8.0×10^7	7.0×10^7
Xe-133	1.6×10^8	1.2×10^8
Cs-137	1.1×10^6	8.5×10^5
Ba-140	1.5×10^8	1.1×10^8

Neutronic Characteristics

Under normal conditions, the time scale for action that controls the power of a reactor is governed by delayed neutrons, even though the delayed neutrons make up a small fraction of the total number of the neutrons produced in the fission process. Delayed neutrons are emitted from decaying fission products, and the typical decay times are from a few tenths of a second to several tens of seconds and are essentially independent of reactor type. Under any anticipated reactor design condition, delayed neutrons are necessary to sustain the repeating fission process, and correspondingly the reactor time response to transients is of the order of seconds. Reactor scram systems are designed to give a very large margin (an order of magnitude) of protection in terms of time response as compared with the time scale of off-normal anticipated events.

If the fission process can be sustained by prompt neutrons alone, then the reactor is termed to be prompt critical." If all delayed neutrons are needed, then the reactor is delayed critical. In any case, the excess (or deficiency) of neutrons is termed reactivity. The reactivity is zero if the reactor is delayed critical and is given an arbitrary unit of \$1 if the reactor is prompt critical.

The fraction of neutrons delayed is small and is primarily a function of the fuel or fissile isotope used rather than the reactor type. With plutonium as the dominant fissionable material as in a typical LMFBR, the fraction of delayed neutrons is about 0.0035. For U-235, the effective fraction is about 0.0065, or nearly twice as large, and therefore, the value of a dollar in a plutonium-fueled reactor is smaller. While designs must be adjusted depending on the use of U-235 or plutonium (such as

by differences in control worths), there is no significant difference, as will be explained, as regards the kinetic behavior of a reactor using either fuel. The general neutronic behavior of fast and thermal (uranium-fueled) reactors is very similar throughout the range of normal operating conditions.* The rate of loss of reactivity with fuel burnup is much smaller in LMFBRs than in LWRs because of breeding and the smaller neutron poisoning effect of the fission products. This slower loss leads to smaller requirements on maximum excess reactivity at the beginning of an operating cycle in the LMFBR. For comparable operating cycle durations, the excess reactivity is typically 1 to 3% in LMFBRs as compared with 8 to 12% in LWRs. The excess reactivity is compensated by control rods in LMFBRs, and by control rods, burnable poisons, and/or chemical neutron poisons in the water reactors. In all cases, suitable techniques exist for adjusting the neutron poisons to compensate for burnup effects. Stable operation and power level control in both is maintained with the delayed neutrons controlling the effective neutron generation time, and these neutrons have a relatively long average lifetime of about 0.1 sec in both systems.

One of the commonly noted differences between fast and thermal reactors is that prompt neutron lifetime in fast reactors is lower by a factor of about 10^3 . The prompt neutron lifetime is simply the average time a fission-produced neutron survives in the reactor before it is absorbed in either fissile or nonfissile nuclei or before it leaks out of the core. This lifetime is indeed short in relation to the lifetime of delayed neutrons and to reactor control-system time response in any reactor system (Figure 4.2-21), being about 5×10^{-7} sec in a fast reactor and of the order of 5×10^{-4} sec in a thermal reactor (depending on reactor type).

There has been a tendency to interpret this neutronic characteristic as indicating that thermal reactors are inherently safer than fast reactors and, specifically, that an accident can begin and end before a reactor scram can even take place. In SIPI's prepared statement,** included in the record of the public hearing of April 25 and 26, 1974, the following was stated:

*One notable exception is that in thermal systems, the fission product xenon-135 is a very strong neutron absorber and complicates the control process. The control system not only must compensate for reactivity changes due to buildup and depletion (by both radioactive decay and neutron absorption), but also must avoid the possibility of xenon spatial oscillations that could cause spatial power variations with time. In the LMFBR, although Xe-135 is produced, it is not a strong neutron absorber. Consequently, control compensation is not required and related spatial power oscillations are not a consideration.

**Scientist Institute for Public Information, Testimony on the Draft LMFBR Program Environmental Impact Statement, April 25, 1974, Section II of written comments, pp. 5-6.

EFFECT OF PROMPT NEUTRON LIFETIME ON RESPONSE TO REACTIVITY INSERTIONS

Figure 4.2-21

Among the other points neglected by the AEC is one imbedded in the phrase, 'neutron lifetime' in the passage quoted from Cochran above.* The time between steps in the nuclear chain reaction in an LMFBR is very short, because the neutrons which carry on the reaction are not appreciably slowed in their passage from one uranium atom to the next. This, of course, is the reason for calling these reactors 'fast.' The very short 'neutron lifetime,' only one thousandth of the length of time between steps in the chain reaction in water-cooled reactors, means that things can happen very quickly in fast reactors if control is lost. The 'period' of a fast reactor, the length of time on the average between links in the chain reaction, would under uncontrolled circumstances be only one ten-thousandth or one hundred thousandth of a second, compared to one-tenth of a second in light water reactors. This means that in a fast reactor an accident can run its course in a small fraction of a second. This point is not mentioned by the AEC, but is essential for an understanding of safety problems in fast reactors. As Cochran points out, 'if sufficient reactivity has been added to reach or exceed prompt criticality [the level of reactivity at which the nuclear reaction is sustained without "delayed neutrons" which create a safety margin up to several seconds during normal operation of all reactors], the power level can rise so rapidly that the ordinary safety controls would not have time to operate before damage has occurred to the reactor.... One of the principal fast reactor safety problems remaining is to make failure to shut down the reactor 'incredible'.

SIPI's statement is of relevance only in the most severe (prompt critical) situations, as shown in Figure 4.2-21. For any prompt critical situation, the analysis must take into account the prompt neutron lifetime. In virtually all such cases intrinsic reactivity feedback mechanisms will dampen or self-limit reactivity perturbation; they can arise from changes in temperature of core materials, from coolant boiling, or, in extreme accident conditions, from fuel melting and relocation. These mechanisms tend to be quite geometry dependent for fast and thermal reactor types. In the LWR, the most important mechanism is the temperature coefficient, which by design is nearly always negative in the range of interest, which means that an increase in reactor temperature results in a decrease in reactor power, an important and effective self-controlling mechanism. If temperatures are extended into the coolant voiding regions, this mechanism tends to become stronger and even more effective. For LMFBRs the initial temperature coefficients tend to be negative, due to structural material expansion and Doppler feedback, although not as strong as in LWRs. If coolant voiding is reached (not a likely event because of the large subcooling--i.e., the coolant temperature is a large number of degrees below the boiling temperature) in some locations positive voiding coefficients would occur.

In addition to the reactor control and scram systems that regulate the reactor power in the delayed critical regime, any practical reactor must have an inherent negative power coefficient of reactivity. That is, if the reactivity goes up, due

*["Moreover, the small prompt neutron lifetime of the fast reactor emphasizes the effects of the rate of reactivity insertion."]. Additional remarks by Dr. Cochran on this matter are provided in Comment Letter 38.8.

to removal of a control rod (a neutron absorber for example, and the reactor power rises, then the increased reactor temperatures will result in a decrease in reactivity. This effect then tends to control the power. In an LMFBR, as in LWRs and HTGRs, the Doppler effect helps provide an effective and prompt negative power coefficient. This effect gives rise to an increase in neutron absorption thus decreasing reactivity as fuel temperatures rise. That this effect is real and is strongly negative has been demonstrated dramatically by transient experiments in the SEFOR reactor. In some of the experiments in SEFOR, the reactivity was intentionally increased well beyond prompt critical, and the rapid transient that resulted was controlled by the negative Doppler reactivity effect.¹¹⁹ The Doppler effect has also been demonstrated in critical test assemblies and in small, fast reactors in the U.S. and abroad.

In general, the Doppler effect increases with reactor size. For different fuel types (e.g., mixed carbide), the Doppler effect should depend more on the reactor design characteristics (pin size, reactor size, etc.) used to take advantage of the fuel characteristics than directly on the fuel properties. If sodium is removed from an LMFBR core, the Doppler effect decreases to about 60% of its value with sodium in the core. However, even with sodium removed, the Doppler effect remains large enough to be effective in controlling rapid transients.

Because the Doppler effect is proportional to fuel temperatures, the amount of reactivity for which the Doppler effect can compensate without exceeding fuel melting is limited. In the SEFOR reactor experiments, this amount was considerably more than \$1 because the fuel operated at relatively low temperatures prior to the induced transients. In a large LMFBR, the amount would be less than \$1 in a sustained steady-state condition; however, under transient conditions initiated at full power, an LMFBR could withstand a near or slightly above \$1 instantaneous insertion of reactivity without reaching fuel melting, provided the reactor protective system was operative. In this situation, the Doppler effect would compensate for enough of the inserted reactivity that the initially rapid reactor power rise would peak, then be turned over by the Doppler effect, and begin a more gradual rise characteristic of a delayed neutron time response. This sequence would provide the time required for the protective system to act. This example is hypothetical, of course, but it indicates that prompt criticality is not a "barrier" and that reactor systems have the capacity to survive transients in excess of \$1 without damage.

As noted above, the existence of a positive sodium void coefficient can reduce the effectiveness of the Doppler feedback. Reactivity change can accompany the

introduction of voids into coolant sodium (by boiling for example). If voids are large enough and occur very rapidly, the concern arises as to whether other inherent, negative reactivity effects would override that from voiding. For incidents where failures occur in regions where the void coefficient is positive, the design must be such that the resultant local reactivity perturbation does not propagate into a whole core reactivity accident. Depending on core size, geometry, and the particular mix of materials in the core, the net overall sodium void coefficient can be negative, zero, or positive. In small cores such as EBR-II and Fermi, the sodium coefficient is negative. With increasing core size, the sodium void coefficient can become positive (see Annex A to Section 4.2.7 for typical values in 1000-MWe designs). For concepts where the overall void coefficient is positive, very rapid and large-scale voiding could result in a significant reactivity addition and possibly extensive damage. As described in subsequent sections, means are provided to avoid such situations.

Some early LMFBR conceptual designs have included core configurations as a means of inherently minimizing the void effects, through "spoiled" geometry (pancake, annular ring, modular). The outcome of one of these studies (ref. 113, p. 49) was:

The results of the analysis indicated that the response to the accident conditions was very similar for the advanced and conservative cores. Thus, no very important differences were found in the reactivity insertions that would cause damage, the amount of flow reduction that could be tolerated, or the required response of the control system.

In large measure, this conclusion resulted because of the general neutronic considerations summarized earlier, which permit other means of accommodating the existence of positive sodium void effects.

This argument does not mean that positive reactivity effects from sodium voiding do not introduce another consideration in LMFBR design as compared with LWRs; it does mean that this factor is not a compelling disadvantage when weighed against other very favorable characteristics of the LMFBR. In subsequent sections (for example, Sections 4.2.7.6.4 and 4.2.7.8) the implications of sodium voiding on accident analyses are discussed further.

Another feature in the LMFBR is that the fuel is not in its most reactive configuration. Core geometry control, including compaction, is an important consideration in the analysis of a wide range of accidents and has a direct influence on design.

Comparatively small variations in core geometry and fuel distribution can have significant reactivity effects in an LMFBFR. The reactor core must therefore be designed with an adequate margin to ensure freedom from unacceptable fuel redistributions in any anticipated operational transient or accident. A typical value of the core radial compaction coefficient for a fast reactor of demonstration plant size is 0.010\$/0.001 in. This value requires that the core geometry be extremely stable. The design of the core must be such that there will be no significant fuel redistribution during normal operation or accidents. The designs provide for support of the core system in such a way as to assure geometric stability at all times. Of particular importance are the bow of fuel assemblies caused by temperature gradients and radiation-induced swelling. The reactivity effects caused by thermal bowing are either positive or negative depending on how the fuel assemblies are supported. Coolant temperatures are generally coolest on the side of the fuel assembly away from the core center. Thus, assuming fuel assemblies were only retained at the bottom, they would bow away from the core center. If, however, a simple restraint above the core were also employed, then the fuel assemblies would bow inward and have a positive temperature coefficient. For situations where some form of lateral restraint above the fuel column is desirable to assure core location, a more complicated restraint system consisting of two load points above is necessary to assure a negative bowing coefficient. Designs of both free-standing and restrained cores with negative bowing characteristics have been developed. Substantial efforts are under way on the development and testing of core restraint systems.

However, the simple fact that the core is not in its most reactive configuration affords the potential that the consequences of a severe accident might be accentuated if compaction could take place. This factor was cited by one commenter as an inherently dangerous feature of LMFBFRs. SIPI,* citing Cochran, stated:

In order to make LMFBFR fuel undergo a nuclear explosion, it is necessary only to bring it together into a more compact form fairly quickly. As Thomas Cochran points out, 'It takes only a slight compaction (about 2 percent volume reduction of a core) to trigger an explosive nuclear runaway.' (See below) Small fractions of the core can by themselves sustain nuclear runaway reactions if they are compressed by a proportionally greater amount.

It is true that compaction of the fuel will add reactivity. However, the implication that this feat can be realistically accomplished is not correct. The fuel material is high density (greater than 80% of theoretical), the structural materials

*SIPI, Testimony on the Draft EIS, April 25, 1974, Section II of the written comments, p. 1.

(pin cladding, assembly duct walls, core supports) are steel. These materials are not easily deformed or compacted. By design, LMFBR fuel configurations are precluded from any significant compaction so long as the fuel elements remain intact. Because of the built-in features, including plant protection systems, there is no real likelihood of a "nuclear runaway reaction," which could lead to loss of fuel element integrity. Only when very extreme assumptions are made, leading to large-scale failure of fuel elements, does there exist a possibility for significant fuel motion and recriticality. Since the reactor core is a tightly packed array containing fuel of high density and highenergy content, any freeing of fuel from its constraint is likely to lead to removal of fuel from the core. However, there remains the possibility that such failures could lead to a more compact arrangement of the fuel. Such a possibility is discussed in Section 4.2.7.8. Even when extreme assumptions are made, the intrinsic reactivity feedback mechanisms discussed earlier act to limit the rate and progression of the accident. (Section 4.2.7.8 treats this subject more fully.) Such factors as a short prompt neutron lifetime appear to have little practical significance throughout the range of accidents for which protection of the reactor core is required.

In summary, while there are important inherent differences between thermal reactors and LMFBRs, there appear to be no factors that would render the LMFBR fundamentally less safe than thermal reactors.

4.2.7.4.2 General Design Features

The model plant features, summarized in Section 4.2.3, provide a representative means of addressing the various factors noted above. Annexes A and B to Section 4.2.7 provide more specific discussions of safety-related design features.

The overall design of an LMFBR, or any nuclear plant, involves the integration of individual system designs in a manner that assures optimum performance and overall reliability. Interfaces between the various systems are recognized as areas of particular significance, and a sound design assures that the interaction of one system with another has been properly accommodated.

Codes and standards (e.g., ASME Section III, IEEE-279) have been developed which are specifically designed to ensure that the probability, and therefore the frequency, of component failures will be appropriately reduced as compared with conventional power plant component failures. Yet it is important to recognize that, in addition to the steps taken to prevent accidents, a nuclear power plant

is explicitly designed to safely accommodate a wide range of possible events. (See discussion of general safety requirements in Section 4.2.7.3.)

In implementing these codes, the designer estimates the frequency for each occurrence, based on industrial and nuclear experience with similar systems and components. The plant design is expected to assure that no occurrences within a given frequency class will cause consequences that exceed the specified limits for that class. The designer can provide this assurance by taking design measures to reduce either the frequency of an occurrence to a lower class or the consequences to be within the class limit. Reduced component-failure rates can be obtained through design measures such as improved component design margins and/or improved quality control to assure performance capability. Reduced component-failure consequences can be obtained through design measures such as provision of diverse, separate systems with functional redundancy or features to mitigate the consequences of failures.

A number of comments were received concerning the general subject of reactor safety, citing malfunctions and incidents that have occurred in power and test reactors.* Opinions on the relevance of this history differ widely; some persons conclude that severe accidents will happen and that previous experience bears this out. Others claim that the existing base fully supports the safety of LMFBRs:

For example, Inglis** in referring to the Fermi accident stated:

...it was not expected to happen. And yet it happened. It happened in a larger program of commercial reactors which has by now accumulated the not very impressive sum total of about 150 reactor years of operating experience with other types of reactors. Compared with the experience that is forecast for the entire program for the rest of the century, this amounts to only about 2 percent of the expected experience. It is not enough to give us any basis in experience for estimating what is the probability of an accident more severe than the Fermi accident, serious enough, indeed, to cause a rather complete meltdown and burst, or penetration of containment.

*Comment Letters 8, p. 5 of encl. 2; 8, pp. 6-8; 16, pp. 32-33 of encl; 22, p. 2; 25 pp. 2-4; 26, p. 5; 27, p. 2; 38.1, p. 11; 38.8, pp. 4, 8-9; 40, p. 2; 42, pp. 3-4. Testimony on the Draft Statement, April 25 and 26, 1974; SIPI's written comments, Section II, pp. 2, d; Dr. Inglis (SIPI), pp. 112-118; Dr. Stryker (SIPI), pp. 127-128; Commonwealth Edison (Behnke), pp. 152, 154; Westinghouse Electric Corp. (Simpson), pp. 167-169; Hoffman, p. 186; Atomics International (Balent), p. 215; Environmental Impact Assessment Project (Dr. Martell), pp. 235-236; Electric Power Research Institute (Levenson), pp. 263-266; Eggen, pp. 291-294; written statement of Jack B. Moore (Southern California Edison Co.), pp. 13, 17; Dr. Kepford, p. 313; Project Management Corp. (Van Nort), pp. 334-336.

**Hearing Testimony, p. 114.

And Younghein* said:

Failures have frequently occurred in the conceptualization, design, construction, operation or maintenance phase. However, those conventional machines and systems [referring to complex submarines, ships, aircraft, or spacecraft] have an almost childlike innocence and simplicity compared to a plutonium-fueled liquid metal fast breeder reactor.

On the other hand Levenson** stated:

The results of such professional in-depth technical analyses all indicate that the probability of a major nuclear power plant failure which would create a public risk is estimated to be so small--ranging from one in a million reactor years of operation to one in a million million reactor years--that in a familiar non-nuclear context the public risk would be considered truly negligible ...

We believe that all nuclear power plants and their associated fuel cycles including the LMFBR can now be provided with the sufficient redundant and multiple safety features to meet any agreed upon level of public risk acceptance.

This approach to nuclear power safety is unusual in the history of new and significant technical systems. For example, in the development of our motor vehicle and air transport systems no such average long-range assessments projecting environmental and public health impacts were made.

And Eggen pointed out:

The safety record of fast reactors speaks well for the care in design and the concern and understanding of safety questions.

There have been accidents, but they were not catastrophic. They were contained within the design capabilities, and the plants were returned to useful operation.

In the case of Fermi, an unexpected but not an unplanned-for occurrence led to a partial melting of a fuel assembly and some damage to adjacent assemblies....

I believe that fast reactors can be designed to operate safely and are needed to meet our energy needs.

The examination of potential failure modes, failure rates, and consequences of failures in mechanical systems is a long-standing aspect of the design of nuclear

*Comment Letter 8, p. 5 of encl. 2.

**Hearing Testimony, p. 263.

†Hearing Testimony, p. 293.

facilities. As long as mechanical equipment is used, malfunctions and failures will occur. This precept, as noted above, is an inherent assumption in reactor safety design because, from the standpoint of design, the assumption is that accidents will occur. Plant systems and the plant protection system are designed to provide margins of safety against just such contingencies. The direct benefit is that should an accident occur, the detailed advance planning will assure that the consequences pose no threat to the health and safety of the public. In the AEC's opinion, the existing base supports an argument for the defense-in-depth approach used in the licensing framework. Subsequent parts of Section 4.2.7 address in more detail steps taken to minimize and control potential accidents in LMFBRs.

4.2.7.5 Anticipated Operating Occurrences and Other Events Leading to No Significant Release of Radioactivity

In Section 4.2.3, releases associated with normal operation were described. Through the process summarized in Section 4.2.7.3, extensive precautions are taken to assure that accidents that individually may be expected to occur one or more times during the plant lifetime do not result in a significant release of radioactivity. For example, experience in conventional power-generating stations has shown that loss of offsite power may be expected to occur one or more times during the plant lifetime. In such an event, controlled reactor shutdown or scram is accomplished; the plant systems and components are designed to accommodate a number of these events with no significant loss in their capability--and no release of radioactivity.

Within the general class of events, there are situations with the potential for releases to the environment. For example, small spills or leaks may occur, outside of containment areas, where chemical analysis of small samples of sodium is performed. This analysis involves handling of microcurie levels of material, typically in confined and filtered work areas; these spills would result in trivial releases.

An important consideration in any reactor is the ability to properly match the reactor power to coolant flow to keep operations well within known, safe limits. To this end, a number of steps are taken to decrease the possibility of fuel failures, which might in turn result in some release to the coolant and cover gas. As examples:

- (1) Conservative data from development programs is used to assure that basic information on materials properties is factored into design, e.g., creep, stainless-steel swelling, and compatibility data on materials.
- (2) A basically high integrity fuel form and high integrity material are

used, e.g., austenitic stainless steel for fuel cladding and as the primary structural material throughout.

- (3) Tested or proven components are used throughout. When necessary, testing of prototype units before installation and operation of the units in a reactor is performed.
- (4) Analytical programs are used extensively in the design process, such as the high temperature stress analysis methods for heat transport and reactor system components.
- (5) Prototypic fuel and control rod materials are tested in irradiation facilities.

With attention given to assure that ventilation flow will route potentially radioactive atmospheres to the atmosphere processing and cleanup systems, the overall risk of a release is reduced through the first-line efforts to prevent fuel failure and through the second-line efforts to control and safely terminate incidents if they should occur.

Normal operation is maintained by the reactor control systems; a separate plant protection system (pps) automatically shuts down the reactor should a malfunction or accident occur.

The PPS terminates the results of all postulated faults within defined limits to prevent plant damage or release of radioactivity. It provides protection for the fuel cladding as the first barrier and the primary coolant system boundary as the second barrier and prevents release by isolating the containment as the third barrier. The PPS is designed to stringent national and industry standards that assure the quality and reliability of the system and its equipment.

When necessitated by off-normal conditions, the PPS initiates and completes a scram (reactor, sodium coolant pump trip) to assure that:

- (1) For anticipated faults (off-normal conditions that individually may be expected to occur once or more during the plant lifetime), there is no impairment of the fuel cladding or primary coolant system boundary to function (the first and second barriers, respectively, to activity release), and no loss of effective lifetime of either the fuel cladding or primary system boundary.
- (2) For unlikely faults (off-normal conditions that individually are not expected to occur but might occur in the plant lifetime), there

is no impairment of the fuel cladding or primary coolant system boundary to function as the first and second barrier, respectively, to activity release.

- (3) For extremely unlikely faults (off-normal conditions that represent an extreme failure and are postulated to test the PPS capability), there is no impairment of the primary coolant system boundary to function as the second barrier to activity release (unless failure of the boundary is postulated as the initiating fault) and no gross core geometry distortion sufficient to render the core uncoolable.

When necessitated by off-normal conditions, the PPS initiates and completes the appropriate containment isolation function to assure that releases are controlled and will be safely mitigated.

Basically, a PPS has two major divisions that perform different functions: the reactor shutdown system and the containment isolation system. The reactor shutdown system initiates and completes scram for the postulated events. The containment isolation system initiates and completes closure of valves in certain lines that penetrate containment.

For reactor shutdown, two independent shutdown systems are provided. Each system is individually capable of terminating the results of all postulated abnormal events without exceeding stated damage limits, and, either system will prevent an event from resulting in significant plant damage. To maximize the reliability of the protection provided, each system typically includes the following internal redundancy and coincidence to provide the shutdown:

- (1) Multiple instrument channels for each plant parameter to prevent single-channel failures from preventing protective action;
- (2) Redundant logic trains and power interruption devices;
- (3) Multiple control rods.

Functional diversity is provided by using different physical parameters to initiate action for the postulated event. In the FFTF, for example, loss of power to the primary coolant pumps is sensed by undervoltage, speed, flow, and pressure sensors. One shutdown system uses one type of sensor to initiate action; the second shutdown system uses a different type. Where response requirements dictate the use of the same physical parameters, differences in the equipment used to sense the parameter

provide additional protection against postulated common mode failures of the two systems.

In addition, each shutdown system typically includes the following testing and monitoring features:

- (1) Test circuitry to provide periodic functional and performance testing of redundant components.
- (2) Monitoring and annunciation to ensure operator cognizance of the state of the system.
- (3) Manual scram buttons in the control room and elsewhere in the plant.

To assure that the two shutdown systems are independent, each is mechanically and electrically isolated from the other. Further, the redundant components in each system are mechanically and electrically isolated from each other to maintain independence.

To provide added assurance of reliable scram performance, functional and equipment diversity is included in the design.

The reactor shutdown system and containment isolation system employ highly reliable components that have been proven in previous service or qualification tests. All components are designed to meet the stringent requirements imposed by the designated AEC, military, commercial, and other standards (cf. 10 CFR 50).

In addition to the extensive analysis that is performed in the process of the shutdown system design, programs are undertaken to test the various components and subsystems that make up the reactor shutdown system.

In development of a specific plant protection system, efforts are made to demonstrate not only the random failure reliability of the shutdown systems but also the freedom from common mode failures.

Protection Against Failures

As noted earlier, review of faults or accidents is conducted for the purpose of assuring that the protection system and other systems and components are capable of accommodating reasonably possible faults or accidents without significant fuel damage. Accidents and incidents have occurred in earlier sodium-cooled reactors. In addition to steps such as those described above to terminate safely a wide

range of postulated reactivity accidents, the lessons of earlier experience are used to avoid similar reoccurrences. Considerable efforts have been made to preclude the possibility of a Fermi-like fuel assembly meltdown. (See Annex C for a summary of this accident.)

As noted in Section 4.2.7.2 (from ref. 125), "Further, the lessons of operating experience as applied to the design, construction, and operation of each new plant help assure an even greater degree of safety for new units." Even though the Fermi accident involved a metallic fuel, which may be more susceptible to the sort of melting incident that took place than the oxide or carbide fuels, the probability for this sort of accident in future LMFBRs has been greatly reduced. For example, in ref. 128 (p. 2.1-19), it was noted:

The low-temperature sodium used to cool components within the Reactor Vessel enters through three inlet nozzles (Figure 2.1-1) and mixes thoroughly within the inlet plenum. From the inlet plenum, the coolant flows through twelve holes, 16 inches in diameter, in the lower surface of the Core Support Structure (CSS) and into 12 interconnected chambers. The largest fraction of the coolant entering these chambers flows radially inward through a strainer comprised of 19,000 holes of 0.25 inches diameter, then into the core basket. The core basket serves as a source of coolant for in-core assemblies as well as for the vessel liner and the spent fuel storage annulus. Coolant supplied to each of the in-core assemblies flows from the core basket through six slots of width 0.7 inches and height 2.25 inches into the assembly nozzle. The coolant then flows through a number (from four to ten) of axial shield-orifice plates, each having at least six holes, and thence into the sub-assembly. These features assure that the likelihood of plugging of a coolant passage by some foreign object is extremely low.

In spite of these measures, a nonmechanistic, limiting-core analysis was performed to assess the margin in the design of the reactor shutdown system of the FFTF. The entire fuel region of the assembly was assumed to be voided of coolant, with fuel melting occurring only at the top. Despite the incoherent behavior of the fuel slumping that would occur, the top one-third of the fuel column was assumed to slump coherently to the middle one-third of the fuel region under the force of gravity without retarding force. The resulting reactivity insertion was a total of 60¢ added at a rate of \$2.4 per sec. For this situation, reactor trip would be initiated by the primary wide-range nuclear high setting protective function. The insertion of 60¢ at \$2.4 per sec would result in hot channel cladding, and fuel temperature rises of about 50°F and 210°F. This insertion would not result in any additional fuel failure. Lesser reactivity accidents, such as might result from control system malfunction, would lead to an increase in reactivity of only a few cents per second and no fuel failure.

A number of comments* were received regarding the Fermi accident in 1966.

Daifuku** stated:

The potential for a serious reactor accident resulting in widespread dispersion of radioactive wastes is higher in some types of breeder reactors than in the light water reactors used today: an explosive accident is possible if a cooling duct becomes plugged. The latter is precisely what happened to Detroit's Enrico Fermi Power Plant in 1966. The reactor was an advanced LMFBFR. A piece of metal blocked the liquid sodium coolant, causing partial melting of a few fuel assemblies. Some radioactive gas escaped within the plant, although none was detected outside the building. While they analyzed the possibilities, the operating staff hesitated for a month to probe for the source of trouble and scope of damage fearing that an explosion might be triggered because of the distorted fuel configuration that resulted from the meltdown (Fabricant and Hallman, 1971).

In an article included with the comments of Younghein[†]:

The accident potential of fast reactors offers unprecedented scope for the sort of catastrophes averted only by good luck in such accidents as the Detroit fast reactor meltdown in 1966.

and SIPI^{††} stated:

The Fermi accident is misleadingly treated in the Statement as though it were a routine and almost expected occurrence.⁵ Actually, it was a completely unexpected event arising from an engineering fault made in spite of supposedly careful regulatory procedures and at first it caused serious concern about the possibility of fast reactor explosion characteristics. It was a close approximation to the maximum credible accident, defined for that plant as 'partial or complete meltdown of one fuel assembly' whereas two fuel assemblies partially melted. No one was hurt, but it may be lucky that the consequences were not much worse: it was a small plant by future standards, it was operating at about 1/6 power and the excursion was terminated not by design trip, but by operator intervention.

The Fermi accident was indeed a major accident in terms of loss of the facility from use--although there was no significant release of radioactivity and no hazard to the public. The SIPI claim that "it was a completely unexpected event" is not

*Comment Letters: 7, p. 15; 8, p. 5 of encl. 1; 25, p. 3; 26, pp. 5-7; 38.8, p. 35; 38.9, p. 13; 42, p. 4; 59, p. 3. Testimony on the Draft Statement: Written comments of SIPI, pp. 16-17 of Addendum; Dr. Inglis (SIPI), pp. 113-114; Eggen, pp. 293-294.

**Comment Letter 7, p. 15.

†Comment Letter 8, encl. 11, pp. 2-3.

††SIPI, Testimony on the Draft Statement, Written Comments-Addendum, pp. 16-17.

borne out by their own remarks that it was "a close approximation to the maximum credible accident." Although it was unexpected, it was not unanticipated. An event of this type was evaluated during the design of the plant to assure the plant's capability to safely accommodate it. As noted herein, postulated severe accidents are considered in the design specifically for the purpose of assuring that if such an accident occurs, it will be safely accommodated. The operator of the plant, the Power Reactor Development Company, who reported on this incident noted:

PRDC believes that the October 5 incident demonstrated the basic adequacy of the Fermi plant operating procedures and safety systems. Despite the occurrence of 95 percent flow blockage and subsequent fuel melting in two adjacent subassemblies, the malfunction was detected and the reactor was shutdown in an orderly manner without any hazard to the public health and safety.

Information from the PRDC report from which this quote was taken (APDA-233, page 3) is included in the more detailed summary of the Fermi accident in Annex C to Section 4.2.7.

In summary, in meeting the requirements of applicable codes, standards, and regulatory criteria, substantial capabilities already exist to assure that the consequences of postulated accidents are safely accommodated. In the case of reasonably anticipated occurrences, the AEC judges that there would be no significant releases of radioactivity. While design features can and will be employed to preclude a Fermi-type incident, capabilities exist to accommodate such an event. Continued research and development on protection system reliability and on possible initiating events (Section 4.2.7.6) for which PPS action may be required is expected to provide proven systems for use in commercial LMFBRs.

4.2.7.6 Unlikely Events with the Potential for Small Releases

4.2.7.6.1 Accidents Involving Radioactive Waste Treatment Systems

The bulk of the solid radioactive wastes will be miscellaneous low activity items. The principal sources of solid waste activity will be from the sodium purification system and the concentrates resulting from the processing of liquid wastes. These wastes will be packaged and shipped offsite to a licensed disposal facility. A typical accident involving solid radioactive material could be a fire in the solid waste storage area. Combustible materials would not normally be permitted in the area; however, if a fire should occur, the fire suppression system would act quickly, and the heating and ventilating system filters would prevent a significant release of radioactive materials.

The liquid radioactive system contains primarily low level wastes. Liquid waste cells will be equipped with sumps to enable any releases to be routed to another tank.

The only mechanism for release that can be identified is the release to the heating and ventilation system of small quantities of tritiated water vapor that have evaporated from the liquid pool during the time the spill fluid is on the floor. Neither the liquid nor the solid radioactive waste systems in the LMFBF contain a sufficient source of activity that accidents involving them could have a significant impact on public safety (see Section 4.2.3.2.4).

Two types of serious but very unlikely accidents are postulated for the gaseous radioactive waste system: (1) release of the primary system cover gas and (2) release of the contents of a gas storage or surge tank. The first type of accident could occur as a result of multiple failures of reactor vessel head seals, failure of a pipe connecting the reactor vessel/primary heat exchanger cover gas to the purifier, or a rupture of the purifier unit. A failure of the head seals would not result in a release directly to the environment, but rather to the containment. This failure could involve on the order of 1000 Ci of noble gases (mostly short-lived isotopes of krypton and xenon). The cover gas is at low pressure (a few inches of water); therefore, the release would probably be at a slow rate. Prompt isolation of the containment would follow with no significant offsite exposures.

An accident of the second type (release of contents of cover gas storage or surge tanks) could result in a release to the low leakage cell housing the tank. Assuming, nominally, that 0.1% or less of the fuel is failed, and that the surge tank is at maximum tank pressure, of the order of a half-million curies of krypton and xenon would be involved. Except for cell leakage, these gases can be diverted through the cell atmosphere processing system which will effectively prevent all but the Kr-85 and trace quantities of Ar-39 from being released (10 Ci); this capability will limit releases to concentrations below the levels specified in 10 CFR 20.

These two events are extremely unlikely failures; anticipated events such as leaking seals or temporary loss of cooling to some components would have lesser consequences.

The above discussion assumes the use of sealed fuel. Vented fuel concepts would routinely discharge radioactive gases or volatile materials into the cover gas and/or coolant, probably in excess of the levels conservatively assumed in the

design of gas purification systems for a 1% failure in sealed fuel. However, the system for effective cleanup and waste treatment would be well within existing design capability and would probably be similar in either case. Cleanup system inventories could be larger with vented fuel, but such changes would not materially alter the preceding discussion. In both systems, with proper consideration of possible faults in the design, the potential releases will not constitute a hazard to the public.

4.2.7.6.2 Fuel Handling Accidents

There are a number of activities within an LMFB reactor complex that are required in support of power generation. Several of these involve the transfer and temporary storage of fuel. For commercial-sized LMFBs, there are a number of possible configurations and options for fuel storage and transfer activities (see Section 4.2.3). The transparency of water makes refueling in LWRs relatively straightforward. The opacity of sodium and its chemical reactivity with air and water introduce some complexity to the LMFB refueling process. One of the outcomes of design options usually employed is use of what are basically double containment barriers between fuel being handled and the outer containment atmosphere. A number of possible failures have been analyzed. These include complete loss of cooling of fuel transfer machines, complete loss of cooling in interim fuel assembly examination cells, and loss of cooling capability in interim decay storage tanks. These events would be unlikely.

For example, an accident was postulated within the examination cell of the FFTF which involved complete loss of gas cooling. This event is considered to be extremely unlikely because there are redundant coolers, both of which are provided with a backup power supply. Fuel transfer activities are performed in sodium-filled pots that greatly increase heat removal capability (and in turn provide time for remedial action to restore effective cooling). For the maximum decay heat assembly (7 kWt), pin rupture would occur approximately 25 min after loss of gas cooling, with clad melting occurring about 45 min after the event. The elements present in the cell would have decayed on the average about 20 days after shutdown. The elements in the cell would be approximately:

- 6.6 x 10⁴ Ci of noble gases (Kr-85, Xe-133)
- 8.6 x 10⁴ Ci of halogens (I-131, I-132)
- 4.7 x 10⁴ Ci of volatile solids
- 4.6 x 10⁶ Ci of non-volatile fission products

As the cell atmosphere heats up, the internal pressure (originally negative with respect to surrounding air spaces) would rise and become positive; the examination cell design leak rate of 0.01 cfm would provide a leakage of approximately 0.002 Ci/sec into the containment work area. Even for this situation, containment isolation would afford ample mitigation and control of the release. The cell atmosphere is constantly circulated to the cell atmosphere processing system; virtually no release of radioactivity would take place. While the inventories and specific process flow activities might vary significantly in a commercial LMFBR, as compared with the various analyses performed for the FFTF, the AEC concludes that design provisions related to fuel storage or transfer can and will be initiated to minimize the potential for serious accidents and, further, if such accidents should occur, to render any resultant releases relatively inconsequential.

4.2.7.6.3 Small Sodium Fires

Two types of sodium circuits are used in LMFBRs: primary coolant circuits that circulate sodium through the reactor core and accumulate significant amounts of radioactivity in the process and intermediate coolant circuits that are entirely outside the core and contain little or no radioactive materials.

The principal source of activity in the primary sodium is from neutron activation of the sodium itself. Neutron capture in Na-23 forms Na-24 (15-hr half-life). Smaller concentrations of Na-22 (2.6-year half-life) are formed by (n,2n) reactions with Na-23. Depending on the number of fuel failures that may have occurred during prior plant operations and on the amount of corrosion of irradiated core structural members, other radioactive species may also be present.¹²⁹ Because of their relatively high activity, the primary coolant circuits are located within cells that contain an inert gas atmosphere. Consequently any leaks in the primary system will be into a confined region that, in turn, is housed in a low-leakage containment. The sodium in the secondary coolant circuit is not normally radioactive except for tritium that has diffused through the intermediate heat exchanger tubing. No potential radiation hazard of any consequence would result from leaks of secondary sodium. However, because of the potential for extensive facility damage in the event of a large sodium leak and because of the danger to personnel from caustic fumes, equipment is provided to detect any leaks that might occur in either system. There is an ongoing development program to examine the capability of various fire-fighting systems. Because of the caustic nature of sodium combustion products, specific systems would be provided to assure that fires involving nonradioactive sodium would be controlled and would not, of themselves, create significant risk to personnel and the public.

Experience to date has shown that one or more leaks, generally small in extent, may occur during a plant lifetime.¹³⁰ In general, the failures can be classified as nuisance, minor, or major leaks. The nuisance leaks have been very small with no attendant fire or damage to adjacent equipment. Minor leaks have been moderate in magnitude and accompanied by small but controllable fires and light-to-moderate damage to adjacent equipment. Major leaks were spills of several hundred pounds which resulted in large intense fires and heavy equipment damage. Of the 80 leaks reported, 69 were nuisance types, 9 were minor, and 2 were major. Of the two major leaks, one caused extensive damage and the other light-to-moderate damage.

The smallest leaks observed were approximately 1 lb in magnitude and accounted for 79% of the entire leak population. The major portion (82%) of these leaks resulted from valve failures. The largest leak was approximately 600 lb. It occurred in a pipe section and was caused by human error. The second largest leak was about 400 lb and was the result of tube failure in a gas-fired sodium heater.

The construction of most sodium systems gives rise to the possibility that small leaks may develop which may not be immediately detected.¹³¹ Small leaks behind pipe insulation, within valve bonnets, and from systems within inert enclosures are some examples of this type of leak. If undetected, such leaks have the potential for progressing and causing significant damage. Leaks in secondary coolant systems can produce hot sodium/air reaction products (oxide, peroxide, hydroxide, etc.) that in turn may cause rapid deterioration of austenitic steels and other metals or alloys used for sodium systems. Thus, in the presence of air, sodium from an initially small leak will create a very serious corrosion condition on the exterior surface of a metallic system. Similarly, flanges and flange bolts may also be seriously corroded by sodium from a small leak. Small leaks can develop into large leaks, the rate of growth of the leak depending on the original leak size, details of the container (such as thickness), system pressure, temperature, and the availability of oxygen and probably moisture. The early detection of small leaks is, therefore, very important in minimizing the consequences of the leak in both primary and secondary coolant systems. If a leak can be detected while it is still small, system pressure can be reduced and the sodium can be drained from the vicinity of the leak or it can be frozen; therefore, damage to the facility can be held to a minimum. As noted in Section 4.2.3, this area receives special attention in plant design. Further research and development efforts are directed at devising new or improved methods for sensing and controlling sodium leaks. Even in the event of very large leaks, however, releases of radioactivity do not pose a serious risk to public health and safety. This question is addressed in Section 4.2.7.2.

4.2.7.6.4 Local Fuel Assembly Faults

As noted in Section 4.2.7.5, features are included in the design of reactor core assemblies to prevent total flow blockage or mislocation of assemblies into positions that could cause overheating or unacceptable reactivity effects.

In addition to the multiple openings in the core support area, the lower assembly nozzles also provide a mechanical key system to prevent placement of assemblies into adverse unplanned core locations. Typical positioning errors that are precluded are the location of a highly orificed assembly in a higher power zone and the location of a fuel assembly in a control rod location. The keying arrangement assures the location of assemblies in positions of acceptable cooling and reactivity.

In addition to the Fermi-type blockage accident, the possibility of fuel element failure propagation has been a technical issue in LMFBR safety. Fuel failure propagation studies have considered the possibility of local core faults causing more widespread damage by some form of failure propagation. Conceivable modes of fault initiation include: (1) manufacturing defects in fuel cladding, (2) inadvertent use of fuel having an abnormal content of fissile material, or (3) development of partial flow blockage. Conceivable modes of fault propagation include: (1) thermal attack by molten fuel, (2) mechanical attack by pressure pulses generated by thermal interactions between coolant and molten fuel or cladding, (3) debris transport and accumulation, and (4) cooling degradation by release of fission gas.

Random fuel pin failures are expected to occur, and if circumstances exist where propagation could occur, such circumstances must be detected or avoided. Sub-assembly-to-subassembly propagation could be an initiator for an accident with whole-core consequences. The potential consequences of such an occurrence dictate that it be thoroughly investigated.

Pin-to-Pin Failure Propagation

Since spontaneous failure of some fuel pins is probable during the reactor lifetime, in spite of the efforts to design and fabricate a high quality fuel assembly, a demonstration that extensive propagation of such failures will not occur is necessary. Extensive analysis and experimentation have shown that pin-to-pin failure propagation is very unlikely.¹³² Three general categories of faults that are considered include cladding defects, in-core blockage during operation, and fuel pin loading errors.

Cladding defects are expected to occur over the reactor lifetime. The initial consequence is a release of fission gas contained in the fuel pin. Thermal effects on the adjacent fuel pins have been shown to be much too small to cause their failure.¹³³ In the event of a rapid gas release, say, from a failure in the fission gas plenum, the pressure pulse seems very unlikely to be sufficient to fail adjacent pins. If, however, several pins should fail simultaneously, the gas release could affect the coolant flow in the subassembly. Even for this unlikely event, the flow would be reestablished before failure of other pins in the subassembly occurred. Based on the evidence at hand, the AEC judges that gas release from cladding defects will not cause failure of adjacent pins.

Long Term Effects

Long term effects resulting from full-power operation with failed fuel pins may include (1) intermittent escape of fuel particles into the coolant stream, resulting in possible small scale blockage propagation and (2) chemical interaction between fuel and sodium within the fuel pin, resulting in pin swelling. In this regard, available information on the development of failures including both naturally occurring defects and artificially defected pins^{134,135} is extremely encouraging. No fuel was lost from the pins, and most of the fission products were entirely retained within the clad. In both low and high burnup oxide pins, diametral increases were minor; i.e., the observed swelling was not sufficient to alter the flow significantly. Much additional irradiation experience with failed fuel pins under full-power operation is required and will be obtained both with currently operating reactors (EBR-II and sodium-cooled reactors abroad) and forthcoming demonstration plants.

Local Blockage

From available irradiation experience with failed fuel pins, the probability of blockage formation due to fuel release appears very small. Even if blockage propagation due to fuel release should occur, the probability of detection by delayed neutron would appear very high prior to any significant growth. Also any long-term effects from a localized undetected small blockage which may suddenly lead to failure of a large number of pins (~20 to 30) would not be a problem in view of our understanding of fission gas release rates and their effects on subassembly flow stability. Furthermore, analysis and experiments have demonstrated that local boiling due to a hypothetical solid blockage of detectable size (<50% of the subassembly cross-sectional area) appears unlikely to lead to excessive overheating or flow instability.¹³⁶ For a summary of the extensive work carried out on local blockage effects, see refs. 137 and 194.

Molten Fuel Release

Release of small amounts of molten fuel could be postulated due to (1) enrichment errors, affecting a limited number of fuel pins, and (2) long-term effects from failed fuel pins (sodium-fuel chemical interaction leading to low thermal conductivity products that result in overheating). However, analysis,^{137,133} in-pile experiments,¹³⁵ and long-term reactor tests with failed oxide fuel pins^{134,135} have demonstrated the low probability of molten fuel release in both the above cases. However, even if such a release is assumed, the local void growth can be demonstrated to be insufficient to cause subassembly voiding.¹³⁶ Since molten UO_2 readily fragments when introduced into subcooled sodium and since the fuel can be carried along as small particles by the coolant sodium, blockage propagation appears unlikely. The effectiveness of fuel sweepout, however, needs to be demonstrated. Such tests are included in the out-of-pile experiments in the LMFBF safety program. Finally, formation of shock waves due to released fuel is considered very unlikely in view of current understanding of the thermal interactions between LMFBF materials,¹³⁸ and therefore rapid pin-to-pin propagation is also ruled out.

Subassembly-to-subassembly Propagation

While the probability of sustained subassembly voiding and subsequent meltdown can be demonstrated to be very small, assessments of the consequences of such a situation are useful in evaluations of (1) the capabilities of instrumentation, such as individual subassembly instrumentation (thermocouples, etc.), to detect and initiate protective actions and (2) the resultant damage if reactor scram occurs following subassembly voiding and/or meltdown.

For subassembly melting, studies to date have not yet developed convincing arguments supporting termination of the accident within the affected subassembly; i.e., further propagation may take place. The continued integrity of the adjacent duct walls may not be maintained. Recent analysis¹³⁹ of this problem indicates that, if no action is taken to shut down the reactor, the subassembly duct wall could breach (for specific configurations of molten fuel) due to thermal effects alone, i.e., as a result of thermal stress concentrations rather than melt-through.

Instrumentation

Due to the very low probability of pin-to-pin failure propagation, it does not appear to be necessary to require that diagnostic instrumentation, which is already installed, also be included in the protective system. Even if a failure were to propagate to a number of other pins, there would be no significant release of

radioactivity to the cover gas unless a large fraction of the pins in the subassembly were affected. Even in this event, no significant release to the environment would take place because of the capabilities of the cover gas cleanup systems.

Whole-subassembly accidents with potential for inter-subassembly failure propagation must be shown to be of sufficiently low risk (including considerations of probability and consequences) or else the plant must have protection against such accidents. There is protective instrumentation in LMFBRs that would provide alarms and/or scrams due to the effects of a whole subassembly accident or subsequent failure propagation. For example, in the FFTF there are cover gas and delayed neutron monitors that would respond to large increases in cover gas and coolant activity by sounding alarms in the operating room. A potential mechanism by which a subassembly-scale accident could affect the whole core is through fuel slumping and reactivity insertion. This event provides an immediate high-flux signal to the plant protection system which responds automatically by shutting the reactor down. Thus, even if the operator does not take corrective action when the alarms sound, the public safety is protected by the automatic plant protection system.

A need for additional instrumentation has not been specifically identified, although it is presently still being evaluated. There is a potential for using individual-subassembly instrumentation to guard against subassembly-scale accidents. Types of instruments that might be used for local surveillance include thermocouples, flow-meters, and acoustic noise detection devices. Development is continuing on instrumentation and detection techniques for surveillance of conditions in early LMFBR cores. Although such instrumentation does not appear necessary at the present time, capabilities exist and development activities are continuing, so that if a judgment is made that the risk of fuel element failure propagation is not sufficiently low, means of timely detection and reactor scram can be provided.

In SIPI's prepared testimony, submitted for the record in the public hearings of April 25 and 26, 1974, fuel element failure propagation was cited* as one initiator of a "nuclear explosion":

Suppose a fuel pin ruptured, through overheating, and caused a slight blockage and local boiling of the sodium coolant. This would create a void in the coolant which in turn, in much of the fuel assembly, might lead to a local increase in the rate of nuclear reaction. In theory, this might lead to further overheating; further boiling, further increases in reactivity, and so on. The failure would 'propagate' until a large

*SIPI, Testimony on the Draft Statement, written comments, Section II, pp. 3-4.

portion of the fuel assembly was affected, increasing the rate of the nuclear reaction in a large portion of the fuel. At this point, some disturbing consequences might follow. One of these, as we shall see, is a nuclear explosion.

A great many possibilities leading to nuclear explosions can be imagined, other than the one just described, if it is possible for minor faults in the fuel to propagate into major faults. The possibilities for a propagation, the significance of which is not explained or discussed, are mentioned by the AEC in section 4.2.7.5, 'Unlikely Events with the Potential for Small Releases.' The 'conceivable modes of fault propagation' are listed as three, but the effect of sodium voids is inexplicably omitted (the positive void coefficient is mentioned in another section of the report, but without indicating its significance for safety). Past research is referred to, without description, and the conclusion is drawn that 'Evidence from experience, tests and analyses to date indicate the propagation of local faults into adjacent subassemblies will not occur.' Only a faint degree of uncertainty in this conclusion is expressed in the paragraph that follows in which the AEC says, 'Since it remains conceivable that all mechanisms of fault propagation have not been identified,' measures to detect its occurrence will be installed in LMFBRS. Whether detection would be of use in time to prevent an explosion is not discussed.

The Draft Statement stated, "Fuel failure propagation studies have considered the possibility of local core faults causing more widespread damage by some form of failure propagation." It is true that specific step-by-step accident sequences were not discussed (although the cited material addressed this point in detail and the text is clear as to the significance of failure propagation). A substantial effort of the development program is related to this area; the subject is treated as a major subtask in the LMFBRS Program Plan, Element 10, Safety.¹²⁴ Our current assessment is that fuel failures will not propagate into core disruptive accidents and that means are available to detect and safely control propagating events should any occur. However, fuel element failure propagation is discussed as a potential accident in Section 4.2.7.8.

4.2.7.6.5 Steam Generator Failures

Energetic chemical reactions can occur when sodium and water are brought into contact. Consequently, the potential for and consequences of water-to-sodium leaks in the steam generators must be minimized in LMFBRS design. Studies are under way to develop and prove out design concepts for commercial-sized reactors.

Additional preventive measures can be provided such as suitable selection of materials, use of modular construction, and stringent quality assurance procedures. Consequences of leaks, if they should occur, will be minimized by provision of adequate detection means for the operator and by incorporation of specific protective features. Detection means are available, using hydrogen and oxygen

concentrations in the sodium as the principal monitor. It is an objective of the development program to provide a detection system of sufficient sensitivity to assure that appropriate corrective action can be taken to prevent a minor defect from developing into a major leak.

In spite of care taken in design and fabrication, occasional unplanned leakage of these tubes may occur. This leakage would permit water/steam injection into the sodium. The reaction is exothermic and liberates hot hydrogen and other reaction products. Since the net specific volume of the reaction products is greater than that of the reactants, a pressure source is developed.

A variety of schemes have been used, or are being considered for future use, to provide early detection and control of such a leak; early plants (Fermi, EBR-II, HNPf) typically utilized hydrogen detectors. In order to safely control any resultant pressure source, some sort of relief system is incorporated in the design. The system components will be designed to prevent any damage from propagating into the primary coolant system.¹⁴⁰ Pressure relief systems will be used to protect the steam generators, secondary coolant circuits, and intermediate heat exchangers.*

These systems consist basically of rupture diaphragms that vent the reaction products through relief lines to retention tanks. The retention tanks are basically reaction product tanks to collect and retain solid and liquid products. Typically, these would be conventional tanks with internal spiral guide vanes--with most of the liquids and solids separated by centrifugal force. Downstream a separator is usually installed to remove any entrained liquid and solids from the gaseous discharge (hydrogen and steam) to ensure no significant sodium or sodium reaction products are discharged to the atmosphere. Because only secondary-system sodium would be involved, no significant potential for release of radioactivity (except for a small amount of tritium) would attend steam generator tube failure in an LMFBR.

Operating experience has been gained from sodium-heated steam-generation systems in the U.S. (EBR-I, EBR-II, SRE, HNPf, Fermi) and in foreign reactors. Some of these steam-generators have operated reliably over extended periods of time. The EBR-II

*In their comments (Comment Letter 38.7, p. 27), NRDC claimed that "A steam leak into the primary sodium loop could result in severe pressure transients or loss of coolant flow in or loss of coolant from the reactor core." Given an intermediate loop, this does not appear to be possible. Even if an intermediate loop were not provided, experience related to reactors with secondary or intermediate sodium systems does not indicate that the consequences would be as severe as implied. Of course, before any system is utilized which does not employ an intermediate loop, this matter would have to be fully explored.

units have been successfully operated for over ten years. However, some failures have occurred; these failures have not resulted in significant releases of radioactive material, sodium, or sodium reaction products.¹⁴¹ Most recently, the USSR BN-350 encountered significant problems in the operation of three of its twelve steam generators. Water inleakage into sodium occurred in these units as a result of suspected faulty welds. In two of these units, the vigorous reaction between the water and sodium resulted in bursting their rupture discs. A USSR representative stated that there was no explosion* and that the plant is in routine operation during 1974 at 20 to 30% of its rated output. As with other steam-generator failures, this incident apparently did not result in any plant damage beyond the affected loops and did not result in any significant release to the environment.

4.2.7.7 Extremely Unlikely Events

4.2.7.7.1 Severe Natural Phenomena

A requirement for any nuclear power plant, including the LMFBFR, is that severe natural phenomena (tornadoes, earthquakes, etc.) do not result in damage of such an extent as to endanger public health and safety. Important safety-related features of the plant, including the components of the reactor coolant boundary, can withstand the most severe of each of these type of events appropriate to the site and retain the capability to prevent or mitigate the consequences of accidents within the applicable criteria of 10 CFR 100. Detailed analyses of the impacts of such events are usually design specific; based on the results of analyses on facilities such as the FFTF, no fundamental problems appear to exist in this regard.

Tornado protection is provided in the form of structural strength in, or shielding of, all components important to safety, such as the containment building, in the same manner as for a water-cooled reactor. With this protection, the worst consequences of a tornado would be a reactor reactor scram with no release of fission products from the fuel and possibly some structural damage to unprotected components that are not important to maintaining a safe shutdown.

Seismic events have a number of safety implications because they could affect structural integrity, cooling capability (by failure of offsite power), reactivity effects (by motion of assemblies within the core), and also the speed of response of the shutdown system (since seismic motion could retard control rod insertion).

*Contrary to the allegation made by NRDC (Comment Letter 38.8, p. 27).

Thus, a proper assessment of a seismic event requires a thorough study of all component failures that might result from it, in combination with the potential effect on the shutdown system. As in LWRs, LMFBR plant systems and components are designed so that the plant can be shutdown and maintained safely following a severe seismic event. The plants are typically designed for continued safe operation subsequent to seismic events that would be anticipated during the plant life.

Protection against maximum floods is provided either by selecting a site that is immune from floods or by providing flood protection to assure that components and systems important to safety retain their ability to perform their safety function during and following the maximum conservatively assessed flood.

4.2.7.7.2 Large Sodium Fires

As described in Section 4.2.7.6.3, massive sodium leaks or spills (many thousands of pounds) are not expected to occur. However, the consequences of such an event are commonly evaluated as a test or measure of the capability of containment systems to accommodate a sodium fire.

The sequence of events typically postulated in defining the consequences of sodium fire is the following. The reactor has been shut down for several days to permit decay of Na-24 activity, preparatory to opening a primary cell(s) for maintenance operations. The cell is opened, and its inert atmosphere replaced with normal air. Some failure, such as operator error, is then assumed to drain the loop sodium onto the cell floor where it commences to burn. The cells are typically lined with steel as a means of avoiding possible sodium-concrete reactions.

Calculations based on such a sequence of events indicate peak pressures of a few psig and temperatures of about 250°F.^{110,128} These values are well within typical containment building specifications.

Spills may be postulated to take place in closed cells; in this instance the low oxygen content (1 to 2%) will prevent rapid or complete combustion of the sodium. If the spill does not involve a large spray component, the resultant pressure-temperature transient does not impose a significant load on the cell (10 psig, gas temperature of a few hundred degrees). If, however, the sodium is ejected under pump pressure in a manner to cause splashing or fine dispersal of much of the sodium, somewhat higher pressures and gas temperatures could result. The exact conditions would probably be much less than the theoretical value (~30 psig,

1050°F); the specific value would be design specific and depend on a number of values such as cell geometry, break location, duration of spray volume coverage, and gas-liquid heat transfer. In addition to the option of providing cell designs of suitably high pressure and temperature capability, there are a number of design options that could be imposed to limit the consequences of such spills. Such options might involve, for example, deflectors or curtains to partially isolate and contain the discharged sodium.¹⁴² Other options include limited spill areas, sloped floors with sumps or gratings, and pressure relief to other volumes. Work is under way to develop an improved experimental and analytical basis for assessing the consequences of potential sodium fires and design features to mitigate such events.

Assessments of the response of such containment systems (including consideration of the pressure transient from the fire, leakage that may occur prior to containment isolation, action of filtration systems, and long-term leakage following the accident) indicate radiation doses (due to radioactive sodium and contaminants that become airborne as a result of the fire) at and beyond the site boundary which are well within applicable guideline values.^{110,128,143} A 2-hr, 2000-ft whole-body dose of 12 millirems and a 30-day, 2-mile whole-body dose of 4.5 millirems were estimated for a 1000-MWe LMFBR.¹¹⁰

4.2.7.7.3 Refueling Accidents

There are several variations in LMFBR refueling methods under active consideration. A typical current scheme involves so-called "under-the plug" refueling in which the reactor head closure (plug) remains in place on the reactor vessel throughout the refueling operation. Double containment normally is maintained at all times. The potential for environmental release of radioactive material therefore is exceedingly low. In the case of direct removal with no interim in-vessel storage, the total inventory of short-lived fission products being handled would be similar in LMFBRs and LWRs (less fuel with higher fission-product concentrations in LMFBRs). With in-vessel storage of a few months or more, short-lived activities are down by an order of magnitude or more in LMFBR fuels. Total inventories of long-lived fission products being handled are similar in LWR and LMFBR spent fuel.

An example of an analysis of a postulated major fuel-handling accident can be found in the Preliminary Safety Analysis Report for the FFTF.¹⁴³ Normal cooling in the handling machine outside the reactor vessel keeps the fuel pin clad temperatures at less than 1000°F. Should normal power to the machine be lost, emergency power

would be supplied to maintain normal cooling levels. In the event all normal cooling were lost, natural circulation of the air through the machine would be sufficient to keep fuel from melting.

For the purpose of demonstrating the capability of the containment to provide protection against these low probability events, calculations were performed, assuming that the contents of the fuel handling machine, including an entire high-power fuel subassembly plus the pot of sodium, are spilled onto the reactor building floor. The sodium was assumed to burn resulting in a peak pressure of 2.8 psi along with release of 100% of the noble gases, 50% of the halogens, and 50% of the volatile solids. Resultant doses were calculated to be well below 10 CFR 100 guideline values.

For larger plants, the power density of the assemblies is not anticipated to be significantly different than an FFTF assembly. The fuel dwell time in the reactor is expected to be greater by a factor of 2 to 3 with a corresponding increase in fission-product inventory. The offsite dose from such an event for a commercial plant, therefore, would be expected to be no more than about a factor of 3 greater than that calculated for FFTF and would still be well within the guideline values of 10 CFR 100. This example illustrates the capability of the reactor containment building to limit the release from even extremely unlikely fuel handling accidents.

4.2.7.7.4 Heat Transport System Failures

LMFBR heat transport systems are made up of multiple, parallel circuits. Each circuit is comprised of three subsystems connected in series: (1) a radioactive primary system, (2) a secondary nonradioactive sodium, system (and 3) the steam system (see Section 4.2.3). Potential faults that might be common to the multiple circuits as well as faults in each of the subsystems are included in LMFBR design-safety considerations. Reliable systems are provided to sense heat removal faults and to provide for reactor shutdown and continued removal of fission-product decay heat.

Potential heat transport system faults include: (1) loss of offsite power and consequent loss of full-power pumping capability, (2) mechanical difficulty with one or more of the main circuit pumps, (3) loss of main heat sinks through faults in the steam generators, the turbine generator, or the main condenser, and (4) rupture of a main line coolant pipe. Various design features can be provided to assure that an LMFBR can tolerate any of these faults without serious damage or significant release of radioactive material to the environment. Pony motors, powered by standby diesel generators, typically are used on the main coolant pumps for decay

heat removal in the event of loss of offsite power. Further, elevated circuits typically are employed to ensure flow in the total absence of pumping power or for any other fault that might incapacitate the main pumps. Loss of the heat sink is made unlikely through the use of multiple loops, steam dump capability, and, in some designs, auxiliary systems that have some measure of independence from the main heat transport systems. Consideration of rupture of a main coolant line has led to several standard LMFBF design features as described below.

Because of the low pressure, subcooled character of LMFBFs, coolant blowdown characteristic of a postulated LWR pipe rupture cannot occur. Rather, the potential problem is one of a draindown or pump-out condition. Elevation and/or containment of the system by guard tanks can, therefore, be employed, in combination with tripping the pumps, to limit the loss of coolant inventory. By restricting the loss of sodium inventory such that submergence of the reactor vessel main coolant nozzles is maintained, the other unaffected primary loops and associated primary sodium system components remain operable as redundant emergency and long-term decay heat removal systems. The flow provided by the coastdown of the pumps, following the reactor and pump trips, cools the core in the period immediately following a system rupture.

Because the reactor vessel outlet nozzles must remain covered after an accident, the elevation or containment of the remainder of the system is considered in relationship to those nozzles. All major components and directly attached piping are contained in guard vessels that provide double containment to an elevation above the reactor vessel outlet nozzles so that the pumps, when operating at pony motor speed, cannot pump sodium down to a level below the top of the reactor vessel outlet nozzles. The primary sodium piping, except for the connections to the components within the guard vessels, is installed above this minimum elevation without additional containment. The guard vessels are sized to prevent the sodium level in the vessel from dropping below the top of the outlet nozzle elevation.

Analyses¹⁴⁴ of large pipe ruptures indicate that an LMFBF can respond to a large range of ruptures without impairing fuel capability or leading to significant core damage. Rupture downstream of a primary pump raises the possibility of pump-out of the sodium inventory in the reactor vessel over the reactor outlet nozzles and thereby a loss of suction in the unfaulted loops. This situation is prevented by prompt tripping of the pumps. Rupture at or near the reactor inlet nozzles would appear to lead to the greatest core-bypass flow and consequent core flow reduction. Additional protection against core damage can be provided by appropriate resistance

to backflow at this location, if necessary, or through development of improved capability for rapid detection of leaks and reactor shutdown.

Potential pipe ruptures will continue to be assessed for each LMFBFR. If the probability of occurrence is sufficiently high and the system cannot adequately accommodate the transient with the main heat transport system, the consequences will be mitigated by design techniques similar to those discussed above.

4.2.7.8 Hypothetical Core Disruptive Accidents (HCDA)

4.2.7.8.1 Introduction

Beyond the range of potential accidents discussed previously, there can be postulated a spectrum of very low probability events and accident sequences that could involve substantial damage to the core. This area has received widespread attention in LMFBFRs. With a core that is not in its most reactive configuration and with a positive sodium void coefficient, a substantial nuclear excursion has been considered as at least a theoretical possibility deserving study and by some a clear requirement to be accommodated in design. Early studies of fast reactor accidents emphasized the analysis of very energetic nuclear excursions,¹⁴⁵⁻¹⁴⁷ primarily in recognition of the theoretical potential for significant reactivity addition by fuel compaction or sodium voiding. Somewhat arbitrary, but intended to be conservative, fuel compaction models were postulated in most early studies. For example, the EBR-II Hazards Summary¹⁴⁵ considered a model in which the upper half of the core fell coherently (uniformly) toward the somewhat densified lower half of the core. Since there was no attempt to analyze the detailed sequence of events from some identifiable fault condition up to the point of massive core melting, energetic excursions of the above type became known as "hypothetical" core disruptive accidents (HCDAs). The analysis of the consequences of HCDAs has become a traditional consideration in fast-reactor safety assessments.

Over the last decade, increasing emphasis has been devoted, in both plant design studies and in the safety development program, toward evaluations of the events (identification of a logically progressing accident sequence) that conceivably could initiate core disruption and of the subsequent sequences of events leading to core damage and ultimately to releases of radioactive materials. Emphasis on initiating events has a very direct benefit in that extensive efforts can be made in design to assure that reasonably possible HCDA initiators are identified and steps taken to eliminate them or reduce the risk of their occurrence. As an example, given the existence of a significant positive void coefficient, a large gas bubble

in the core region could introduce a large, rapid reactivity insertion. Means to control gas entrainment and to assure that large gas bubbles do not enter the core is provided in design.¹⁴⁸ Similarly, another possible initiator, rapid control rod movement, is prevented through selection of appropriate design features such as positive mechanical stops and electronic interlocks. (See Section 4.2.7.4.2 for additional discussion on safety-related design features.)

The earlier 1000-MWe design studies considered and evaluated a number of possible faults that could result in core damage. In all cases, design features were suggested which appeared to have the potential for accommodating even such very low probability events. Alternative approaches were also suggested whereby sufficient steps could be taken (such as improved detection of the onset of such events) to clearly eliminate the need for a requirement that the consequences of HCDAs be within the design basis envelope. In recent years, three general classes of events have been under study. The first is referred to as an unprotected transient overpower accident. This accident is assumed to begin by a reactivity insertion with complete failure of the plant protection system. The insertion rates considered can be related to specific phenomena or faults that could potentially add reactivity or can be selected arbitrarily. The second accident is an unprotected loss-of-flow accident. This accident is usually assumed to occur when the coolant pumps lose power and coast down and when the plant protective system fails to respond when needed. The third type of accident involves the propagation of local fault conditions not terminated by any control or protection system action.

As will be described in subsequent parts of this section, events of this type necessarily result in fuel motion or core damage sufficient to shutdown the reactor (since scram action is assumed not to be available). Processes such as coolant boiling and voiding, clad melting and relocation, and fuel melting and relocation must all be considered in assessing the ultimate consequences of the HCDA. If a core disruptive accident is assumed to occur, the resultant damage to the core could in time lead to releases of fuel, fission products, and sodium to containment spaces. Current assessments indicate that the likely consequences of HCDAs would be contained, although there are technical uncertainties that require resolution or improved understanding if the resultant consequences are to be well characterized.

A substantial research and development program that specifically investigates such accidents is under way, and these HCDA-related investigations constitute a major segment of the LMFBR safety program (see refs. 121-123).

In addition to studies that relate to specific initiating events, parametric evaluations or arbitrarily assumed conditions are commonly used to gain a better understanding of the significance or sensitivity of various phenomena or of the capabilities of various plant systems to accommodate the consequences of a spectrum of accidents involving core damage. This program, together with other efforts on preventing or providing early detection and control of possible events that could lead to core disruption, should provide a sound base for decisions on treatment of core disruptive accidents in commercial LMFBRs.

The application of such evaluations in designs produces considerably divergent views. (A spectrum of these views was offered in the recent Conference on Fast Reactor Safety--particularly in the panel discussions on Selection of Safety Design Bases for Fast Power Reactors.¹⁴⁹ A summary of the panel discussions is provided as Annex D to this section.) In general, the views concern: (1) the propriety of postulating shutdown failure as a means of determining the HCDA consequences and (2) the magnitude of the energy release that must be contained.

Assumptions of scram failure historically have not been made in establishing design bases for commercial reactors constructed in the United States. However, as noted in ref. 125, analyses of the consequences of the postulated anticipated transients without scram (ATWS) have been performed by water-cooled power reactor designers, to improve assessments of LWR safety.

As noted elsewhere, efforts are being made to provide LMFBR protection systems designs that are fully as dependable as those given in ref. 125 for anticipated transients without scram. However, there exist opposing views regarding the likelihood of achieving this goal or of obviating the need to consider scram failure in LMFBR design.

NRDC* stated:

There is general agreement that failure to shutdown the LMFBR should be made 'incredible' by providing sufficient redundancy and diversity in the control and safety rod systems and designing against common mode failures. At this time, there is no general consensus on criteria for establishing required reliability of fast reactor shutdown systems.... Furthermore, it is impossible to demonstrate experimentally that any number of independent shutdown systems would have the required degree of reliability. It is impossible to demonstrate they will shutdown at all, much less shut down with the more rapid speed required to prevent an LMFBR prompt critical excursion.

*Comment Letter 38.8, p. 7.

As noted elsewhere (cf. Sections 4.2.7.5 and 4.2.7.8.2), efforts are being made to provide LMFBR protection system designs that meet the objectives set forth in ref. 125, the ATWS document (the general objectives are quoted in their entirety in Section 4.2.7.3). If these efforts are fully successful, it is possible that they will establish a vanishingly low probability of occurrence of an HCDA.

Based on the information to date, protection systems that conform to existing criteria (such as IEEE-279) can be designed, constructed, and operated. Implementation of these criteria does provide high confidence that shutdown systems will respond in a timely fashion when needed. Should the ongoing efforts indicate potential failure modes, such as common mode failures, additional design modifications could be accomplished to eliminate or minimize their possibility of occurrence. To the extent that commercial LMFBRs may be required to consider protection system failure in design, these reactor designs will be expected to demonstrate that means are incorporated to accommodate the consequences of such failures without excessive risk to the public.

What follows in this section is an assessment of current understanding of HCDAs. Available design options that potentially could be used to accommodate the consequences of core disruptive accidents are also described.

4.2.7.8.2 Assessment of General Analysis Techniques*

HCDA Initiation Modes

A first-priority intent of the design, review, and regulation functions of reactor construction and operation is to preclude accidents. Extensive efforts to identify potential accident initiation and development modes and to prevent significant damage or release are standard practice in the nuclear industry. Consideration of severe accidents, therefore, starts with an assumption that faults nonetheless will occur. Conceivable modes of initiation include: (1) control rod runout (mechanical malfunction or operator error) and (2) loss of primary cooling capability (due to mechanical faults in main circuits, loss of offsite power, or loss of primary heat sink, that is, steam generator, main turbine, or condenser). A typical list of faults considered is included in the Table 15-1 of ref. 126. Protection for each of these faults is provided in reactor designs through such means as control rod stops and interlocks, plant protective systems, and alternative heat dump methods. Operation of these redundant protective systems results in safe termination of the

*A more detailed survey is presented in ref. 150.

initial fault. Additional failures of these protective systems, therefore, must be postulated in order to identify situations (HCDAs) in which severe core damage may be envisioned.

Accident Initiation

This phase assumes, regardless of probability, that some initial set of failures has occurred and involves a calculation of the resultant core neutronic and thermal behavior up to the point of loss of subassembly geometry. Because most LMFBR designs have used rigid subassembly ducts, intact subassemblies are only coupled through inlet and outlet thermal-hydraulic conditions and reactor power. Subassemblies with similar power, flow and irradiation conditions are generally lumped together into groups for the purpose of analysis. All the subassemblies within a given group, or "channel," are assumed to behave identically. Phenomena such as thermal-hydraulic transient, sodium boiling, fuel-pin mechanics and failure, cladding and fuel motion, and fuel-coolant interactions are treated with one-dimensional models. Multichannel codes of this type which were developed under USAEC sponsorship are the FREADM,¹⁵¹ MELT,¹⁵² and SAS¹⁵³ systems. The original purpose of these codes was simply to give a reasonable and conservative accounting of the reactivity effects in order to provide initial conditions for the core disassembly codes.

In the past few years, the analytical models used for evaluating initial accident phase sequences of HCDAs in LMFBRs have progressed significantly. This progress is due both to the knowledge gained in research into the phenomena, such as coolant boiling, fuel failure, fuel-coolant interaction, etc., which can occur in a disruptive accident, and to increased sophistication of the computer codes. Also, the models provided in the large code systems such as SAS and MELT are beginning to be verified by in-pile experiments (such as those in the TREAT reactor). Because of these developments, the ability to assess the behavior of an LMFBR system prior to significant loss of geometry is reasonably good, although the predictive capabilities of these codes are not complete. Extensive code development and experimental programs are continuing to refine the ability to predict accident phenomena to the point of loss of subassembly geometry. In addition to an in-reactor experimental program, extensive laboratory investigations are under way in areas such as sodium boiling, fuel-coolant interactions, cladding motion, and fuel slumping and dispersal. From the results of research and development to date (cf. p. 24 of ref. 154), improvements in understanding of HCDA-related phenomena should result in less conservative assessments of the consequences of such accidents.

Disassembly Analysis

Given the assumption that some situation has arisen which by design would be detected and safely terminated but nonetheless proceeds without any interference from the protection system, the only possible consequence is motion of materials in such a way as to be self-terminating.

For example, a reactivity excursion will cause continued heating of fuel, cladding, and coolant. At some point, sufficient heat (and pressure) will be generated to disassemble the core and thereby end the power burst--if it is assumed that no protective action is taken. Because of the variety of possible initiating sequences and possible outcomes of such sequences that can be postulated, a completely deterministic or mechanistic analysis of the disassembly process is not feasible.

The efforts in this area have concentrated on identifying likely or most probable processes together with depictions of generic and reasonably bounding situations.

The disassembly process has often been referred to as an "explosion." Technically, the term "explosion" could be used in that one of the forces contributing to the disassembly can be fuel vapor. However, in virtually every instance, other earlier acting forces are available; for example, sodium flow, sodium vapor, and retained volatile materials, which would cause early disassembly. Only in the most conservative analyses does fuel vapor appear to be the driving force but, for conservatism, this is frequently assumed. Even for such "explosions" the energies involved correspond to those of modest amounts, up to a few tens of pounds, of normal chemical explosives. It is absolutely impossible for any nuclear incident to lead to explosions of the magnitude associated with nuclear weapons, that is, many thousands of tons of explosive equivalent.

A conservative treatment of the disassembly process was first described by a model developed by Bethe and Tait.¹⁵⁵ In their approach, the core was treated as a homogeneous fluid so that the material motion during disassembly could be calculated using a hydrodynamic approach. The calculations were done in spherical geometry. The reactor power was calculated using point kinetics with first-order perturbation theory to estimate the reactivity feedback associated with the material motion.

A number of variations and improvements have subsequently been made to this basic approach to improve the correspondence of the models to real reactor geometry and conditions.¹⁵⁶ Doppler feedback was included, improvements were made to the

equation-of-state used to estimate the pressures, more accurate neutronics were implemented in some cases, and the capability for doing calculations in two-dimensional (r,z) geometry was added.

A number of disassembly computer codes have been developed in the United States as successive improvements were made.¹⁵⁷⁻¹⁶⁰ One of the most recent of these is the VENUS-II¹⁶¹ code developed at Argonne National Laboratory (ANL).

VENUS-II calculates the disassembly motion by a direct numerical solution of the two-dimensional (r,z geometry) hydrodynamics equations. Because the density changes associated with material motion are explicitly calculated, a more accurate density-dependent equation of state can be employed. A number of refinements have been developed to increase the applicability of the code to the calculation of mild excursions.

An area of consideration in disassembly analysis is the possibility of autocatalytic effects. In the broadest sense, autocatalysis could embrace any source of positive reactivity feedback. A particular type of autocatalysis that can potentially increase the severity of disassembly excursions occurs when the transient-induced material motion temporarily increases the reactivity instead of decreasing it. Such an effect could occur during an initial excursion or as part of a recriticality.*

Earlier studies^{162,163} of autocatalytic effects failed to uncover any physically reasonable situations that would greatly increase the disassembly energy release. These studies focused on implosive effects at the enrichment zone boundaries. Although a small amount of net positive feedback can be produced under certain idealized conditions, the general outward expansion of the core produces enough negative feedback to quickly dominate any implosive motion.¹⁶³

These autocatalytic effects are associated with the early phases of material motion. The possibility of an autocatalytic recriticality that results from gross fuel motion has also been considered.¹⁶⁴ This mode of recriticality occurs if excursion-induced pressure gradients can drive the fuel into a critical mass at some position in the reactor. An example of this would be a situation where a disassembly excursion ejects molten material out against the surrounding structure where it temporarily collects and goes critical. This mode of recriticality appears

*A state of criticality which can theoretically be achieved after a core meltdown.

at least conceptually possible in large reactors;¹⁶⁴ planning has begun to evaluate effects of this mode of recriticality.

In the earlier analyses, including those supportive of the 1000-MWe studies, a number of conservative assumptions were made, principal among which were assumptions of widespread coherence of coolant expulsion and fuel collapse. These assumptions led to estimates of large rates of increase of reactivity (hundreds of \$/sec) and correspondingly energetic prompt critical power bursts (thermal energy equivalent of a few seconds of full power operation). Such bursts were estimated to be sufficiently energetic as to disperse large fractions of the fuel and assure that the core was permanently shutdown. Subsequent work has achieved a more detailed accounting of the space-time history of coolant and fuel motion during a severe accident. This advance, together with inclusions of the retarding effect of negative Doppler feedback and the dispersive forces from fuel vapor pressures, has led to estimates of substantially less severe power bursts (characterized by peak pressures of a few tens of atmospheres). However, such power bursts often are estimated to be incapable of dispersing sufficient fuel to assure permanent nuclear shutdown; therefore, the possibility of recriticality and repeated power bursts is created. Recriticality, including the possibility of self-aggravating, autocatalytic processes, has been investigated extensively for the FFTF reactor.^{150,163,165} This work has resulted in the conclusion that a series of small power bursts are possible but are likely to generate no greater total energy than would be contained in a single, large, permanently dispersive burst.

With the greater sophistication of analytical tools used to aid the understanding of the disassembly process, the possibilities of an initially mild prompt critical burst have increased, at least for HCDA events arising from loss of flow and failure to scram. Secondary criticalities have correspondingly been given increasing emphasis in the research and development program.

In their comments, NRDC* cited preliminary work of Jackson and Boudreau on this subject, noting:

Jackson and Boudreau estimated reactivity insertion rates of 40 \$/sec to 200 \$/sec and for cases where the subcritical fuel was driven together by 10 atm pressure from molten-fuel coolant interaction the ramp rates were 140 \$/sec to 900 \$/sec.

*Comment Letter 38.8, p. 16.

NRDC* further cited results of analyses by Lee and Pigford on the effect of fuel zoning relative to thermal energy for reactivity rates in the range analyzed by Jackson and Boudreau and stated:

These results further demonstrate the tremendous uncertainties with respect to estimating the explosive potential of these reactors.

It is true that significant uncertainties exist in the analyses of recriticality situations. In large measure, these uncertainties stem from the as yet incomplete research and development program; analytical methods do not permit detailed assessment of all the contributing factors that would affect the outcome of such situations, if indeed such situations are possible at all. However, the tools currently being used are known to contain substantial conservatism. In the recent paper by Jackson and Boudreau,¹⁶⁵ they state (p. 1): "The results also indicate that a number of effects are expected to be present in realistic situations and can strongly mitigate the reactivity insertion rates." As research and development continues, further refinements in the analytical methods should permit accounting for constraints in a reactor system. Though not well characterized, those processes that would lead to large thermal energy yields appear to demand a great deal of coherence and are possible in only a limited range of circumstances. If such processes are physically realizable, they are certainly associated with a low probability set of possible consequences of HCDAs.

In the following section, evaluations of the damage potential (conversion of the thermal energy to mechanical work) are described. However, before proceeding to that section, note that the above NRDC comments and others on this area in the Draft Statement constitute an argument against the use of a so-called mechanistic approach to HCDA evaluations.

On p. 10 of their comments, the NRDC** states,

....one should not conclude...that the 'mechanistic' approach...provides or can ever provide a defensible upper bound on the explosive potential of this reactor.

This assertion regarding the value of the mechanistic approach is supported by three points summarized on p. 11 of the NRDC comments and expanded upon in pp. 11 through 19. The three points are as follows:

*Comment Letter 38.8, p. 16.
**Comment Letter 38.8, p. 17.

1. The mechanistic approach is limited to the examination of a few accident scenarios. And there is uncertainty as to whether the scenarios analyzed represent worse case (or conservative) estimates of the explosive potential of the reactor.
2. There are innumerable uncertainties with respect to modeling assumptions and input data with respect to critical mathematical models used to analyze accident scenarios.
3. These models, and therefore the mechanistic approach, break down for certain sequences of events where significant loss of the reactor core geometry has occurred.

A brief discussion of each of these points follows.

The thrust of the first point appears to be that since every possible accident scenario is not analyzed, scenarios might be overlooked that would produce significantly worse consequences than those that are selected for detailed examination. However, the disruption of a fast reactor core is the result of a gross power-flow mismatch, and such a mismatch can potentially arise for only two reasons: an abnormal reactivity insertion and/or an impairment in the normal cooling of the core. Although there are essentially an infinite number of specific initiators that can be hypothesized to produce either of these conditions, the net results will be very similar. The transient overpower and loss-of-flow accidents are extensively analyzed because they cover these two generic causes for power-flow mismatch. These two accidents can effectively serve as umbrellas to cover a very broad spectrum of possible accident initiators. Thus, consideration of other accident types would not be expected to lead to strongly different results.

NRDC's second point appears to be that there are innumerable uncertainties in the models, assumptions, and parameters used in the mechanistic analysis of accidents. While uncertainties do exist, a number of the models have been verified by comparison with experiments.¹⁶⁶⁻¹⁶⁸ The uncertainties that remain are well recognized, and the ongoing experimental and analytical programs are designed to produce an increased understanding in the areas of key uncertainty. In the interim, the influence of these uncertainties on the potential accident consequences is determined by scoping calculations and sensitivity studies.

The third point is that the mechanistic approach breaks down for accident sequences where a significant loss of reactor geometry has occurred. Although the so-called accident initiation models (such as the SAS series of codes at ANL) begin to break down as gross core disruption ensues, the mechanistic approach does not break down.

The break down of one code simply requires use of a different modeling approach when this stage of an accident is reached. This point is extensively discussed in ref. 150.

The mechanistic approach has traditionally been broken into a number of calculational phases. These phases are characterized by the different modeling techniques required to describe the different classes of phenomena that occur as an accident sequence unfolds. The accident initiation models, such as the SAS codes, are only intended to analyze accidents up to the point of subassembly disruption. Once subassembly deterioration is encountered, the state of the system is passed on to the next phase of the analysis.

A major effort is under way to develop, improve, and verify analytical descriptions of accidents following the loss of core geometry. Although the precision with which the detailed accident sequence can be analyzed generally decreases as the degree of core disruption increases, the mechanistic approach still provides information that is essential in determining the consequences of disruptive accidents. It provides the framework for a probabilistic analysis that can ultimately define the consequence distribution for HCDAs.

Damage Evaluation

In the event that a whole-core nuclear accident is not terminated by protective system action or by inherent nuclear feedback mechanisms, the accident may result in disassembly of the reactor core due to internal pressure generated from rapid thermal energy buildup. During and following this period of rapid thermal energy buildup, some percentage (conversion efficiency) of the thermal energy in the reactor materials will be converted to mechanical work. This mechanical work results from various high-pressure vapor-expansion processes, such as the expansion of vaporized fuel and steel and/or the expansion of vaporized sodium resulting from thermal interactions between sodium and molten fuel or molten steel. Mechanical work determines the magnitude of the displacement of reactor materials and components from their original position and is therefore a measure of damage to the reactor. The rate and quantity of thermal energy converted to work must be determined in order to establish the magnitude of a pressure source term. Once the pressure source term is established, it is used as input to computer codes from which the structural response of the reactor system can be analyzed. The REXCO series of computer codes,¹⁶⁹ which couple a hydrodynamic calculation of pressure propagation with an analysis of the motion and deformation of important reactor components, is

widely used for this purpose. Determination of the pressure requires knowledge of the particular accident sequence under study, the high-temperature physical properties of core materials, and the effect of molten fuel-coolant and/or molten steel-coolant thermal interactions. The importance of being able to predict the effect of thermal interactions with liquid sodium is that significantly more mechanical work is produced if the thermal energy in the fuel and cladding goes into vaporizing sodium than if mechanical work is performed by fuel and steel vapor pressures alone. This difference is a result of the physical properties of sodium which make it, like water, an efficient working fluid.

The current best evidence is that a severe nuclear accident, whether it be a single energetic burst or a series of less energetic ones, may be characterized by a total energy release equivalent to a few seconds of normal power operation. The amount of energy associated with a nuclear accident is normally expressed in megawatt-seconds, both when referring to total energy and work potential. The actual damage potential is still a different quantity, in that the work potential is obtained by integrating a pressure/volume curve to some assigned equilibrium pressure, frequently 1 atm. The pressure/volume or pressure/time history appropriate to any given accident sequence has substantial uncertainties associated with the working fluid and the heat transfer mechanisms. For both of these reasons, characterizing an accident in terms of megawatt-seconds work potential should be thought of as a short-hand notation, with use of the more explicit pressure/time histories being restricted to references. Thus, for a 1000-MWe plant, operating at about 2500 MWt, less than 15,000 MW-sec thermal energy generation is estimated for a severe accident. The mechanical work can now be determined if this thermal energy generation is multiplied by the conversion efficiency for the thermal processes in the accident. A theoretical limit to the conversion efficiency in a LMFBR accident has been determined to be about 30%.¹⁶² This determination was based on the thermodynamically optimal conditions for heat transfer to liquid sodium in a molten fuel-coolant thermal interaction, and many of the assumptions made to simplify the analytical effort were physically unrealistic. These assumptions, involving interaction processes that are impossible under any real circumstances, greatly simplified the calculational effort. As understanding has increased and as the analytical models for thermal interaction processes have become more realistic, the maximum possible conversion efficiency has more accurately been estimated as less than 10%,¹⁷⁰ the precise number depending on the circumstances for the interaction. However, these analytical descriptions inherently assume the occurrence of highly energetic fuel-coolant thermal interactions from large-scale, rapid vaporization of sodium by hot fuel.

All experimental evidence to date illustrates that these energetic interactions will not occur in an LMFBF environment.^{171,172} Mechanical work is done only by expansion of fuel and steel vapors, with no significant vaporization of sodium, and subsequently low conversion efficiencies have been observed. In-reactor experiments in the TREAT facility under prototypic LMFBF accident conditions have produced conversion efficiencies of no more than 0.2%.

The experimental and analytical evidence¹⁷¹⁻¹⁷⁷ for low conversion efficiency in a LMFBF accident is as follows:

- (1) No energetic thermal interactions have been observed in a variety of in-reactor tests with prototype LMFBF accident conditions involving the meltdown of oxide fuel pins in the presence of sodium.
- (2) Energetic thermal interactions have been produced only under carefully controlled laboratory conditions that can not be duplicated in a LMFBF.
- (3) The results of all experiments are consistent with theories that predict that energetic thermal interactions will not occur under LMFBF conditions.

If a large-scale energetic interaction is to occur, a number of conditions must be met. One of these conditions is that sodium must be heated to a temperature that exceeds its stability limit, at which point the overheated liquid sodium begins to vaporize violently by explosive boiling. This stability limit may be reached when molten fuel and liquid sodium come into contact if there are no nucleation sites for sodium vaporization to develop. This situation implies a very clean system, with no materials except pure fuel and sodium. A nucleation site, which can be produced by dissolved gases or solid particulates, is any foreign material or discontinuity that can act as a site for formation of vapor bubbles and thus initiate nucleate boiling. Nucleate boiling will prevent overheating of the liquid sodium and will lead to vapor generation rates many orders of magnitude smaller than those associated with explosive boiling. The abundant nucleation sites available in an LMFBF (gas bubbles, solids, fill gases, fission gases, and fragments) will therefore act to prohibit large-scale energetic thermal interactions between molten fuel and sodium.

Energetic thermal interactions between steel and sodium are theoretically possible when steel is near its boiling point. However, the high vapor pressure of steel at these temperatures prevents good contact with sodium and results in low sodium-vaporization rates. An energetic steel-sodium thermal interaction has never been observed for any in-reactor experiment.

If continuing research and development efforts are successful in demonstrating that large-scale energetic thermal interactions are not possible in an LMFBF environment, current conservative estimates of potential damage following a severe accident can be significantly reduced.

Considerable work to assess the capability of LMFBF primary coolant systems for accommodating the mechanical consequences of HCDAs has been performed over the past two decades.¹⁷⁸⁻¹⁸⁰

These assessments have been made on a variety of different designs. Principal structural members that are important in determining any mechanical damage are the core support structure, the reactor vessel and attached piping, the reactor vessel support system, the intermediate heat exchangers, pumps, the reactor vessel head and head-mounted equipment and the means of sealing these components to the reactor vessel and the plant structural features (e.g., interior walls in the containment).

Experiments have been carried out to verify the calculational methods and computer codes used for analytical predictions of mechanical consequences and to determine damage to scale-model equipment. Such experiments include scale-model testing in support of the Fermi¹⁷⁸ and Prototype Fast Reactor (PFR) and, more recently, the FFTF reactor.¹⁷⁹ Whereas most previous work has employed high explosive sources, the FFTF tests used a buffered source that more nearly simulates the characteristics of a postulated nuclear accident.¹⁷⁹ Based mostly upon the results of the tests of a 1/10-scale model of FFTF (reactor vessel, vessel head, holddown bolts, key internal structures, and a typical heat transport circuit) and associated analysis, the AEC has concluded that LMFBF coolant systems are designed to accommodate several hundred megawatt-seconds of work energy. This capability appears to encompass the likely consequences of HCDAs, although some persons have suggested that larger design capability should be assured. Some conceptual designs including general 1000-MWe designs have been developed which would provide additional capability, such as provision of crushable material below the reactor vessel head or energy-absorbing columns above the reactor. These systems, of course, create additional system complexity; for example, use of a sealed machinery dome with crush columns would require longer times for refueling and would pose difficulties for routinely maintaining control rod drive mechanisms.

A number of comments were received regarding the damage potential of an HCDA* either expressing the view cited earlier by NRDC that there are tremendous uncertainties in this area or citing such views as those of Webb.

Webb¹⁸¹ has asserted that the explosion potential must be established before development of the LMFBR should proceed. He defines the explosion potential as the maximum conceivable accident (presumably an explosion) that cannot be ruled out as physically impossible. (An AEC staff report on Webb's treatise and a letter from Webb in response is attached as part of the NRDC comments on Section 4.2.7.7 of the draft.) In discussing HCDA damage potential, NRDC** stated:

The foregoing results are of utmost concern to us because they raise serious questions with respect to whether even a small LMFBR (the size of the FFTF) can contain a severe HCDA, much less whether the 1000 MW and 2000 MW LMFBRs being projected by the AEC will be able to do so.

and, citing Webb:[†]

Dr. Richard Webb, a critic of the LMFBR safety program, appears to share a similar concern over the ability to establish the explosive potential of the LMFBR...

For the reasons cited earlier under the topic of "Disassembly Analysis," recriticality situations are being extensively studied in the ongoing research and development program. Necessarily, parametric analyses and assumptions are made to better understand areas of uncertainty and interest. However, assumption of artificial or illogical reconfiguration of materials for the purpose of arbitrarily increasing the potential consequences of recriticality is not considered either a realistic or reasonable basis for responsible inquiry. The maximum damage potential consistent with regulatory approaches is necessarily assessed as part of the safety analysis effort. The theoretically conceivable maximum could be evaluated by assuming arbitrary and extreme rearrangements of material, but this analysis would generally be of academic interest only. Such extreme limit determinations are useful in some circumstances (for example, when scoping studies are desired or when the possibility of ultimate capabilities on consequence thresholds is being examined).

*Comment Letters: 7, p. 15; 8, enc. 2, p. 6, enc. 6, p. 3, enc. 11, pp. 2,3; 16, p. 6; 20, Appendix 3, p. 2; 38.8, pp. 4, 17, 20, 21, 24-30; 50, pp. 1, 2; 53, pp. 14, 16; 57, p. 2; 58, p. 2; 59, p. 4; 62, p. 5; Hearing Transcript, pp. 44, 47-48, 122, 127; Hearing Record, Testimony of SIPI, pp. 405, 7-8, 10-12.

**Comment Letter 38.8, p. 17.

†Comment Letter 38.8, p. 20.

Application of such results to a conclusion that there is a significant risk of a very damaging HCDA is neither warranted nor appropriate. A more useful approach involves development of an understanding of the functional relationship between possible HCDA consequences and the probability of occurrence of event sequences leading to those consequences. The mechanistic analysis of accident sequences is an essential element in this development. As noted earlier in this section, the occurrence of any type of HCDA must necessarily be made extremely remote. Extensive programs are under way to develop and confirm designs of effective and reliable plant protective systems. These programs, coupled with operating experience of existing and planned smaller plants, are expected to confirm that the possibility of occurrence of any major accident in an LMFBR is very remote. As noted above, capabilities exist to accommodate a spectrum of HCDA events. This matter is treated again in the subsequent section on "Plant Specific HCDA Analyses" (Section 4.2.7.8.3). This matter will also be treated as necessary for individual plant safety analyses as plant designs are developed.

Post-Accident Heat Removal

A major technical issue involves the ability to contain core debris following disassembly. The underlying problem is that of removing residual decay heat in order to maintain the integrity of the primary coolant system boundary. Associated with this problem is the possibility of recriticality, caused by relocation of fuel debris.

Certain classes of HCDAs may involve very little fuel movement and core distortion. For situations of this type, long-term decay heat removal can be evaluated with the analytical tools generally used to calculate decay heat removal following reactor scram.

On the other hand, substantial failure of fuel rods and distortion or gross loss of core geometry could result from an HCDA.

The final phase in the analysis then is an evaluation of the post-accident heat removal, that is, assessment of how the fuel comes to rest in various parts of the system where it can be permanently cooled. This analysis has mainly taken the form of predicting the spectrum of post-accident dispositions that the fuel could have and analyzing the subsequent coolability. The disposition of core materials and the required measures to assure cooling of the core debris are, of course, strongly dependent on the reactor design under consideration and the HCDA being investigated.

In recognition of this dependency, research upon heat removal from distorted fuel geometries has been under way for a number of years.^{182, 183} This work has included

the limiting configurations of beds of finely fragmented fuel debris and pools of molten fuel. Extensive testing has been done with simulant materials and with UO_2 and sodium. Decay heating has been simulated electrically in these tests.

The debris bed tests have employed distributions of finely divided UO_2 particles characteristic of those formed upon contact between molten fuel and sodium. These tests indicate that substantial depths of such beds can be cooled by one-directional heat transfer to overlying sodium with no cooling on the undersides of the beds. The principal mechanism for heat removal is a counter-current natural convection flow of liquid sodium into the bed with vapor from the bed. In addition to providing cooling, the vapor flow may act as an effective self-leveling mechanism, thus improving heat removal from initially irregular beds. These tests indicate that an entire FFTF core, if it were fragmented and mixed with steel and spread over a flat cross section of a reactor vessel, would be cooled by these intrinsic natural convection processes.

The work with molten pools has utilized electrical resistance heating in simulant fluids. Further work using electrically heated UO_2 and in-pile fission heating is planned. The work to date has identified the main ways in which heat is transported to the surfaces of such pools. The tendency of molten pools to form stagnant, temperature-inversion layers near the bottom is expected to provide a large impedance to downward heat flow in low-conductivity mixed-oxide fuel. This behavior can significantly reduce the cooling required to prevent melt-through of containing structures. Current results indicate that maximum heat fluxes of the order of $100,000 \text{ Btu/ft}^2\text{-hr}$ are possible from the underside of molten UO_2 pools. Heat fluxes of this magnitude can be removed readily by natural convection sodium flow across slightly curved surfaces.

Concern has been expressed regarding the ability to demonstrate cooling effectiveness within the primary coolant system. The fundamental heat transfer results described above would form the basis for designing out-of-vessel devices to catch and retain any fuel debris that might eventually melt through the reactor vessel. Such devices have been under active consideration in certain reactor designs^{184,185} and have been considered to be a reserve option for the FFTF. A recent report¹⁸⁶ reviews the state-of-the-art and discusses several areas of uncertainty. Various possible design approaches are reviewed including in-reactor-vessel sodium-cooled trays, out-of-vessel sacrificial beds, and special designs of reactor cavity liners. Considerable progress has been made in some of the areas important to the design and evaluation of post-accident heat removal (PAHR) systems for an LMFBR. Because

of the strong dependence of such evaluations on preceding analyses (the d sassembly phase), some of the more recent developments in the HCDA analytical tools have yet to be carefully integrated with PAHR evaluations. This area is receiving considerable emphasis in the safety program, and efforts continue both in the areas of in-vessel and out-of-vessel post-accident heat removal.

Radioactive Material

During the stage in which such mechanical deformations would be occurring (vessel expansion, head lift, etc.), expulsion of sodium may take place. This expulsion will be largely governed by the expansion pressures, the amount of deformation, and specific seal or component designs. Estimates of sodium releases therefore will be determined through the output of calculations of earlier phases of the accident. The subsequent heatup of the reactor cavity or containment spaces is generally analyzed using tools developed to assess the consequences of sodium fires. As described in Section 4.2.7.7.2, the radiological consequences of a sodium release, of itself, are very modest.

There does exist the possibility of a subsequent release of fission products and fuel materials from the reactor vessel if any permanent deformations or failed seals should result and if a vapor bubble should break through the sodium pool surface. At present, analytical methods are not available to fully characterize this process.

Because the vapor bubble (containing fuel, fission products, and sodium) would be expanding and rising, any radioactive material (source) therein would tend to move from the core region to the top of the sodium pool. Qualitatively, a number of removal or source-reduction processes must exist. Removal mechanisms such as condensation of fuel vapor and sodium, entrainment of suspended materials in the sodium pool, plateout on structural members, and local fallout would act to reduce or eliminate any release of radioactive products. Additional removal mechanisms such as fallout and deposition in leak paths would act to further reduce the source. With increasing knowledge of the earlier phases of the HCDA, the ability to mechanistically model the release of radioactive materials can be expected to improve. There are active programs aimed at this area. For example, a recently prepared paper¹⁸⁷ suggested that the core-gas bubble may collapse into small numerous bubbles under the intense pressure waves produced by the slug impact on the reactor head. This event would promote rapid cooling and retention of core debris. If this action can be confirmed, then a fairly conclusive argument will exist that, even in the event of an HCDA, little or no plutonium or fission products

other than noble gases would be released from the reactor vessel. (With very small bubbles, essentially complete scrubbing out of any suspended fuel would occur.)

In any event, a release of sodium appears to be a required precursor to a release of materials originally in the core gas bubble, and substantial mitigation of any plutonium release will take place due to aerosol removal and fallout within the containment envelope. This process (and the process of joint agglomeration of plutonium with sodium particles) has been well demonstrated through a variety of experiments and analytical studies. Some efforts continue in this area, to assure that all situations of interest are examined and that proven models are available for use in commercial LMFBRs.

4.2.7.8.3 Plant-Specific HCDA Analyses

The foregoing briefly summarizes the status of the analytical and experimental base for assessments of the consequences of HCDAs. As noted, the continued extensive research and development program in this area¹²¹ has permitted, with time, increasingly more sophisticated assessments of the consequences of HCDAs. In almost all respects, this increased sophistication has meant, as a principal impact, that estimates of the likely consequences of such events have diminished.¹⁸⁸ In a similar fashion, assessments of probabilities of accidents, of scram reliability, and of risk have become increasingly sophisticated.

Contemporary concept-related HCDA analyses in the U.S. are currently centered on the FFTF and LMFBR Demonstration Plants. In both instances, detailed analyses in support of plant design and safety analysis reports are in progress and utilize best available experimental and analytical capabilities. Complete safety analysis reports for these two plants, based on today's technology, are scheduled for issuance within approximately a year. To the extent practicable, preliminary results from these efforts have been considered in this section.

Basically, three types of approaches have been considered relative to accommodating HCDAs: (1) those concepts in which the HCDA is basically a design-basis accident for structures and components, (2) those concepts in which no explicit provisions are made to accommodate HCDAs, and (3) those concepts in which the HCDAs are rendered as truly hypothetical or Class 9 events but in which some features are still provided as a means of further reducing the overall risk of such events. In all cases, a screening process is employed to assist in the judgments of appropriate HCDA initiators (see, for example, ref. 110, p. 14).

Examples of the first category are the SEFOR,¹⁸⁹ AI 1000-MWe,¹¹⁰ B&W 1000-MWe,¹¹¹ and SNR-300 reactors.¹⁸⁵ Concepts in this category have uniformly selected a double containment barrier approach with special provisions for accommodating the mechanical consequences of what were considered very conservatively analyzed HCDA events. The AI 1000-MWe concept (Figure 2 from ref. 110, p. 17; shown in Annex A to this section) includes a number of features employed to accommodate the HCDA event. The HCDA consisted of the following postulated sequence of events:

- (1) power is lost to the pumps
- (2) control system action fails
- (3) both the primary and secondary protective systems fail to respond on demand
- (4) the control rods fail to drop
- (5) no effective operator action is taken.

Once this scenario is assumed, voiding of the core was estimated to begin in 7 to 10 sec as the flow decreased to about 25% of its initial value. Due to the positive sodium void coefficient (about \$6.5 maximum), reactivity rose. The resultant ramp rate reached about 66 \$/second when \$1.50 had been inserted through voiding. This transient was terminated through disassembly; the heat transferred to sodium was estimated to have a work potential of approximately 2200 MW-sec.

As discussed in Section 4.2.7.8.2, this analysis represents what today is regarded as a very conservative appraisal of the accident consequences. Calculations would indicate that substantial leak paths would develop in the primary coolant system boundary subjected to this energy release.

In the AI concept study, the primary release was to the inner containment, with the inerted machinery dome, missile barrier, and cavity liner cooling features provided to safely accommodate the mechanical and thermal effects.

The disassembly analysis gave an upper limit estimate of 73% of the core fuel melting and 3% of the core fuel vaporizing. Drawing from these results, Atomics International assumed that 100% of the noble gases and halogens and 10% of other fission products and plutonium were released from the reactor vessel to the inner containment. No allowance was made for the probable trapping of a considerable portion of the non-gaseous (fission product and plutonium) radioactive material in the sodium. These materials were available for leakage to an outer containment. Because of the high mass concentrations of airborne material, substantial removal would occur by fall-out. Further removal by fallout and by the containment air cleanup system would take place in the outer containment.

For this case, the following whole-body dose and 50-year integrated internal doses were estimated at 1000 meters (2-hr exposure period):

whole body	0.025 rem
lung	0.5 rem
thyroid	3 rems
bone	7 rems

Plutonium contributed approximately 95% of the dose to bone. Of the 250,000 g assumed to be released from the core (10% of core inventory), roughly 0.1 g was released to the environment; this ratio is equivalent to a source-reduction factor of 4×10^{-7} .

Similar analyses have been made for the other concepts. One point of interest is the impact that aerosol fallout has on the relationship of the amount released to the amount leaked to the environment. Aerosol removal by coagulation and fallout is increasingly effective as the mass concentration is increased.¹⁹⁰ Therefore, as the amount of plutonium released increases, the potential offsite consequences will not necessarily increase. For example, in a case where a 100-g release was assumed in a double barrier system,¹⁹¹ roughly 0.03 g was released to the environment. That is, roughly comparable amounts were released even though the amount released from the core varied from 100 g of plutonium to 250 kg.

The view expressed in the B&W 1000-MWe study¹¹¹ appears valid: "In summation we believe that with incorporation of certain safety features the integrity of the secondary containment can be assured." If one is willing to accept the economic penalties of this approach, capabilities exist to satisfy an objective of near-zero release of plutonium to the environment even for conservatively analyzed HCDAs. A conclusion was reached that a conservatively designed LMFBR, from an HCDA standpoint, could be economically viable. The AI study (ref. 110, p. 11) noted: "The approach, in this study, is one of providing containment of a core disassembly accident. This approach was selected to demonstrate LMFBR plant safety capability because it was based on precedent and is believed to be conservative, with respect to showing that a 1000-MWe LMFBR is economically viable." An alternate approach was considered but not pursued at that time. "The alternate approach of providing additional protective feature and safeguards to eliminate the core excursion as the DBA, which was mentioned earlier, is expected to have the net effect of increasing the overall plant safety and reducing the capital investment" [emphasis added].

The GE 1000-MWe concept studies did in fact include a treatment of the alternate approach.¹¹³ One important conclusion (ref. 113, p. 49), was that "the response to the accident conditions was very similar for the advanced and conservative cores. Thus no very important differences were found in the amount of reactivity insertions that could cause damage, the amount of flow reduction that could be tolerated, or the required response time of the control system." While these results were based on a limited conceptual study, they do indicate that considerable optimization of core performance characteristics could be accomplished and the consequences of HCDAs would not markedly worsen. The "advanced" concept did not include blast shields or an outer containment, although an inerted inner barrier was provided (refueling hot cell).

As evidenced by the structural analyses of the FFTF primary vessel and reactor head, considerable capability for accommodating energetic HCDAs would exist even in the absence of specific design provisions to accommodate HCDA loads. This capability, together with an inner barrier (the refueling hot cell of the GE design), would provide for substantial mitigation of any release. For example, if the hot cell had a leakage rate of 400% per day under the conditions cited in ref. 190 (for a 100-g release), 99% of the released material would be retained and not released.

The third approach, to be used in the LMFBR demonstration plant, is currently under development with REG. One of the proposed statements of approach, prepared as a beginning of a dialogue with REG, has been summarized in a recent paper¹⁹² (included as Annex B to this section). In this instance, explicit features such as head hold-down bolts and a low leakage outer containment building are provided as a means of extending the range of capabilities to accommodate HCDAs well beyond those that would be inherently available. In the case of the FFTF, this capability has been extended to 150 MW-sec,* and analyses indicate that this capability is sufficient to accommodate the estimated consequences of HCDAs without exceeding the guideline values of 10 CFR 100.¹²⁸ Analyses are in progress for the Final Safety Analysis Report (FSAR) using the most recently developed models and experimental data. The analyses of ref. 191 (specifically the case cited on pp. 117-121) utilized the basic FFTF case in the Preliminary Safety Analysis Report (PSAR) with more realistic parameters being selected, and the calculated plutonium bone dose was about 90 milli-rem (again for an assumed 100-g release). The FSAR analyses in support of the FFTF are expected to confirm the PSAR estimates. Some efforts are under way to evaluate the possible margins in the FFTF design, over and above the 150-MW-sec

*See discussion on work-energy equivalents in Section 4.2.7.8.2 (subsection titled "Damage Evaluation").

capability. When completed, these efforts will permit improved risk assessments to be made for a broad spectrum of possible core disruptive events. For example, one consequence of a postulated HCDA greater than 150 MW-sec in the FFTF would obviously be increased deformations and movement of the reactor head. For a postulated HCDA significantly beyond 150 MW-sec (as described earlier, the current best estimates indicate that HCDA consequences will be less, and probably substantially less, than 150 MW-sec in the FFTF), increased motion of the head and further deformation of seals could occur--in turn leading to increased sodium expulsion. While the shield deck within the head compartment would be expected to prevent significant sodium combustion and outer containment pressurization, a potential leak path to the outer containment could result. Presumably, one of the outcomes of an increased energy release would be an increased quantity of fuel in the core gas bubble. If condensation and other removal factors, such as those mentioned in Section 4.2.7.8.3, did not take place, then a significant release of plutonium to the outer containment could take place.

The consequences of such a scenario might approach those cited in the Directorate of Licensing report¹⁴² for the FFTF--in which the calculated doses were approximately those of the guideline values of 10 CFR 100. At the upper end of the spectrum, one could qualitatively argue that the event was so large that the basic structural integrity of the primary system was lost. In this case, massive sodium and fuel expulsion could take place with consequences correspondingly more severe. Because of the effectiveness of aerosol removal processes with increasing releases, the assumption that significant retention of the fuel would take place is reasonable. Further, likely leak paths would be through penetrations--which lead into adjacent support facilities where further removal through the ventilation and filtration systems of the support buildings would occur.

The above discussion is intended to reflect a general assessment of the maximum realistic consequences of HCDA events well beyond the capability of the FFTF system and particularly to note, qualitatively, that even at the end of the spectrum of possible accidents significant mitigating factors exist to preclude massive releases of radioactivity to the environment.

As noted earlier, detailed probabilistic risk assessments have not yet been performed, although some preliminary studies have begun. In recent testimony,¹⁹³ Dr. N. C. Rasmussen noted for LWRs that:

The amount of radioactivity that gets released from the nuclear plant into the environment depends upon how much is released from the core during the melting process and how much of this is trapped inside the containment prior to its escape. All plants have provisions to trap radioactivity within the containment. In addition there are natural processes that lead to deposition of many of the radioactive species on the walls and other surfaces in the containment building. In most core melt accidents these processes would be expected to be quite effective in reducing the amount of radioactivity released. However if an analyst were asked what the worst possible release could be, he could imagine a series of unlikely circumstances where the processes for removing radioactivity would not be very effective and a much larger release would result. Our analysis of core melt accidents shows just this effect, namely, that the most likely course of events following core melt results in rather modest releases and larger releases are even less likely to occur. This means, of course, that the largest release is considerably less likely than the expected or typical release in such an accident.

and

I hope our study will help people understand that the most likely consequence of a core melt accident, which itself is unlikely, would be quite modest, in fact, no worse than many other kinds of accidents such as fires and airplane crashes that society has experienced. Just as it is possible to imagine an airplane crash producing 10 or 100 times more serious consequence than the average under a very unlikely set of circumstances, it is also possible to identify an unlikely set of circumstances in which reactor accidents could produce much more serious consequences.

The qualitative arguments cited earlier for LMFBRs provide evidence that this same finding will be obtained when risk assessments are completed for large LMFBRs. A very preliminary attempt to quantify such risk assessments of LMFBRs¹¹⁵ tends to bear this argument out. The summary portion of the report¹⁹⁵ of the recent Reactor Safety Study headed by Dr. Rasmussen is reproduced as Annex E for the convenience of the reader.

4.2.7 ANNEX A: SUMMARY DESCRIPTION OF 1000-MWe PLANT
FOLLOW-ON STUDY CONCEPTS

A.1 Introduction

Each of the five contractors in the 1000-MWe LMFBR Follow-on Study was responsible for selecting a design of the entire plant, except for Combustion Engineering, whose scope was limited to the heat generation system of the nuclear steam supply system. The first task was to select a preliminary reference concept, which was altered, depending upon the results of tradeoff studies and engineering analyses, to form the final reference design.

For their preliminary reference configurations, Combustion Engineering (CE), Atomics International (AI), and Westinghouse used the loop-type system design, and Babcock and Wilcox (B&W) and General Electric (GE) used the pool-type system design.

Conventional containment structures were used by all contractors except GE, who stated that adequate safeguards could be incorporated in the design to ensure against any major accident. Based on this premise, the GE design has a confinement-type structure enclosing the reactor complex.

All contractors used reheat in their steam-cycle design, except Westinghouse, who used a nonreheat cycle.

The Westinghouse design has a four-module core; all other contractors' designs have a single cylindrical core. For the fuel, Westinghouse and CE used carbides, and the other contractors use oxides. A hexagonal fuel subassembly design was used by AI, B&W, and GE; CE used a circular design and Westinghouse used a square design.

Summary descriptions of the contractors' designs follow; the main plant parameters are given in Table 4.2-13.

A.2 Atomics International

A.2.1 General

The AI 1000-MWe LMFBR reference plant uses a loop-type primary coolant system and a tandem-compound reheat turbine. Thermal power of the reactor is 2400 MW, and net power output from the plant is 1002 MWe. The main characteristics of this plant are given in Table 4.2-13.

Table 4.2-13

1000-MWe PLANT PARAMETERS FOR FOLLOW-ON STUDY

	Atomics International	Babcock & Wilcox	Combustion Engineering	General Electric	Westinghouse
Overall Plant					
Reactor Thermal Power (MWt)	2400	2450	2465	2417	2600
Gross Electrical Power (MWe)	1040	1034		1038	1074
Net Electrical Power (MWe)	1002	1009	1000	1008	1044
Plant Efficiency (%)	41.7	41.2	40.6	41.7	40.2
Net Heat Rate (Btu/kWhr)	8175	8281		8192	8500
Reactor Outlet Temperature (°F)	1140	1100	1095	1150	1000
Primary Loop ΔT (°F)	360	300	298	340	230
Steam Generator sodium inlet temperature (°F)	1070	100		1071	967
Secondary Loop ΔT (°F)	400	300		425	309
Steam Conditions					
Superheat/reheat temperature (°F)	1000/1000	950/950	900/900	950/950	900/(no reheat)
Pressure (psig)	2400	2400	2400	2400	2400
Condenser Pressure (in. Hg abs)	1.5	1.5		1.5	1.5
Refueling Cycle (years)	1.00	1.11	0.94	1.00	1.10
Reactor Containment	Double Barrier	Single Barrier		Single Barrier	Double Barrier
Fuel Material	PuO ₂ -UO ₂	PuO ₂ -UO ₂	PuC-UC	PuO ₂ -UO ₂	PuC-UC
Reactor Coolant	Sodium	Sodium	Sodium	Sodium	Sodium
Reactor Geometry	Regular	Cylindrical	Cylindrical	Cylindrical	Modular
Heat Transfer System Geometry	Loop	Pool or Primary Tank	Loop	Pool or Primary Tank	Loop

Table 4.2-13 (Continued)

	Atomics International	Babcock & Wilcox	Combustion Engineering	General Electric	Westinghouse
Reactor					
Reactor Geometry					
Core height (in.)	42.8	34.7	24.0	30.0	39.3
Core length-to-diameter ratio	0.42	0.29	0.225	0.308	0.842
Reactor Subassemblies (total)	505	451	369	463	593
Fuel	274	288	219	234	244
Control Rod	15	25	22	31	60
Shim-Safety	9	18	13	12	44
Safety	6	7	9	19	16
Radial Blanket	138	138	60	126	261
Inner	66	66		60	120
Outer	72	72		66	141
Reflector	78		66	72	88
Fission Gas Plenum Height (in.)	38	1	20	21	25
Tank or Vessel OD (ft-in.)	29-0	52-0	16-6	48-0	17-8
Vessel Height (ft-in.)	48-0	44-6	40-4	40-4	45-0
Reactor Cavity ID (ft-in.)	35-0		16-10	51-4	18-8
Top Shield Type	Double Rotatable Plug	Three Rotatable Plugs	Double Plug	Removable Plug	Removable Plug
Overall Thickness (ft-in.)	11-3	10-0	3-9	4-0	4-3
Steel Plates (in.)	24	14	20		16
Shielding (in.)	60	72	24		36
	(Serpentine)	(Serpentine)			(Graphite)

Table 4.2-13 (Continued)

	Atomics International	Babcock & Wilcox	Combustion Engineering	General Electric	Westinghouse
Steel Top Plate (in.)	15	6			
Seal Type	Double Lip, He Purge	Dip Ring with Mechanical Backup	Single		O-ring
Containment Dome or Refueling Cell (ft-in.)	35-0 ID	100-0 ID		20.0 x 78.0	100-0 ID x 38.7 high
Gas Atmospheres					
Reactor Cover Gas	He	Ar	Ar	Ar	Ar
Containment Dome or Refueling Cell	N ₂	Air	Air	Ar	Ar
Reactor Cavity	N ₂	N ₂		N ₂	N ₂
Thermal-Hydraulics					
Thermal Power (MWt)	2400	2450	2465	2416	2600
Sodium Outlet Temperature (°F)	1140	1100	1095	1150	1000
Reactor ΔT (°F)	360 (average)	300	398	340	230
Core ΔT (°F)	371 (average)	324	328	360	44
Flow Control	Fixed Orifices	Fixed Orifices	Fixed Orifices	Fixed Orifices	None
Peak Fuel Pin Power (kWt/ft)	16.0	15.7	43.5	16.3	39.2
Maximum Nominal Cladding ID temperature (°F)	1221	1240		1225	1400
Hot Spot Cladding ID Temperature (°F)	1282	1300	1223	1300	1378
Maximum Heat Flux (Btu/hr-ft ²)	690,000	656,000	1,360,000	930,000	1,691,000
Average Core Heat Flux (Btu/hr-ft ²)	437,000		892,000	502,000	798,000

Table 4.2-13 (Continued)

	Atomics International	Babcock & Wilcox	Combustion Engineering	General Electric	Westinghouse
Cladding Hot Channel Factor	1.14	1.04			
Radial Peaking Factor	1.22	1.28	1.18	1.22	1.32
Axial Peaking Factor	1.23	1.25	1.18	1.20	1.30
Reactor ΔP (psi)	87.0	35.0	75.0	64.3	88.0
Coolant Velocity (ft/sec)					
Average (maximum element)	27.5	13.6	32.7	25.4	27.0
Reactor flowrate (lb/hr)	75.6×10^6	92.4×10^6	94.0×10^6	80.6×10^6	128.5×10^6
Nuclear (midequilibrium) Characteristics					
Reactor Power Fractions					
Core	0.917	0.923	0.834	0.876	0.954
Axial Blankets	0.031	0.055	0.108	0.076	
Radial Blankets	0.052	0.022	0.058	0.058	0.046
Volume Fractions ^a					
Core and Axial Blanket					
Fuel	0.439	0.442	0.306	0.467	0.276
Sodium	0.326	0.381	0.400	0.380	0.590
Metal	0.235	0.177	0.214	0.153	0.134
Radial Blanket, Inner					
Fuel	0.556	0.644	0.527	0.546	0.426
Sodium	0.281	0.195	0.323	0.177	0.309
Metal	0.163	0.161	0.15	0.277	0.234
Core Fissile Enrichments (at. %)					
Inner Zone	10.5	10.4	9.78	10.73	

^aIncludes all metal, control materials, and moderator materials.

Table 4.2-13 (Continued)

	Atomics International	Babcock & Wilcox	Combustion Engineering	General Electric	Westinghouse
Intermediate Zone	13.1	11.2	11.79	13.45	
Outer Zone		13.0			
Fissile Masses (kg of Pu + U ²³⁵)	2744	3439	2633	2235	
Core	2191	2814	1880	1624	
Axial Blanket	176	390	429	299	
Radial Blanket	377	235	324	312	
Initial Loading (kg of Pu + U)					
Core	19,090	23,818	17,315	13,452	14,120
Axial Blanket	13,132	19,064	25,972	15,035	11,000
Radial Blanket	15,688	16,071	5,693	16,683	35,573
Breeding ratio					
Total	1.30	1.36	1.441	1.42	1.325
Core	0.94	0.93	0.84		0.563
Average Fuel Burnup (MW-day/metric ton)	67,000	100,000	100,000	100,000	100,000
Doubling Time (year)	13.0	10.4	6.99	7.75	9.9
Reactivity Coefficients					
Doppler Constant (T dk/dT)	-0.0080	-0.0051	-0.0064	-0.006	-0.00395
Maximum Radial Sodium Void (%)	8.37	8.07		2.24	
Total Sodium Void, core only (%)	8.07	7.5	3.82		
Fuel and Blanket Assemblies					
Geometry	Hexagonal	Hexagonal	Circular (fuel), Hexagonal (blanket)	Hexagonal	Square
Lattice Spacing (in.)	5.667	6.500	6.527	5.696	

Table 4.2-13 (Continued)

	Atomics International	Babcock & Wilcox	Combustion Engineering	General Electric	Westinghouse
Distance Across Flats (external) (in.)	5.617	6.450	6.492	5.666	5.253
Pins per Subassembly/OD (in.)					
Fuel and Axial Blanket	217/0.30	331/0.28	163/0.40	331/0.25	168/0.302
Inner Radial Blanket	91/0.50	91/0.608	91/0.42	127/0.42	60/0.552
Outer Radial Blanket	61/0.62		127/0.42	127/0.42	60/0.522
Fuel and Cladding Material	Mixed PuO ₂ -UO ₂ , Type 316 SS	Mixed PuO ₂ -UO ₂ , Type 305 SS	Mixed PuC-UC, Type 316 SS	Mixed PuO ₂ -UO ₂ , Type 316 SS	Mixed PuC-UC, Type 316 SS
Average Clad thickness (in.)	0.0175	0.010	0.011	0.010	0.012
Fuel-clad Bond Material	He	He	sodium	He	sodium
Blanket and Cladding Material	Depleted UO ₂ , Type 316 SS	Depleted UO ₂ , Type 304 SS	Depleted UC, Type 316 SS	Depleted UO ₂ , Type 316 SS	Depleted UC in Axial Blanket, Depleted UO ₂ in Radial Blanket, Type 316 SS
Smeared Fuel Density (% theoretical)					
Fuel and Axial Blanket	85	85		85	70.5
Radial Blanket	93	90			93.6
Control and Safety Rods					
Reactivity Swing-over Equilibrium Cycle (\$)	3.5	3.0		4.4	8.11
Absorber Material	Tantalum Pins	Tantalum Pins	Tantalum (shim), B ₄ C (safety)	Tantalum (shim), B ₄ C (safety)	Tantalum Cruciform
Cladding	Type 316 SS	Type 300 SS	SS	Type 316 SS	None
Pins in Array					
Shim-Safety		91	10	19	

Table 4.2-13 (continued)

	Atoms International	Babcock & Wilcox	Combustion Engineering	General Electric	Westinghouse
Safety		91	18	19	
Average Rod Worth (\$)					
Shim-Safety	1.2	0.71	0.82	0.48	0.42
Safety	2.4	0.60	1.06	0.85	0.80
Type of Cooling	Thermal Siphon	Bypass Flow			
Type of Drive	Electromechanical	Electromechanical	Electro- mechanical	Electro- mechanical (shim), Pneumatic (safety)	Magnetic Jack
Thermal Bond	NaK	Helium	Na (shim)		
Heat Transfer					
Number of Loops	3 primary, 3 secondary	6 primary, 3 secondary	6 primary	3 primary, 3 secondary	2 primary, 2 secondary
Total Reactor Flow (lb/hr)	75.6×10^6	92.4×10^6	94.0×10^6	80.6×10^6	128.5×10^5
Reactor Inlet Temperature (°F)	780	800	797	810	770
Reactor Outlet Temperature (°F)	1140	1100	1095	1150	1000
Pressure at Top of Reactor Vessel (psig)	1				35
Primary System Sodium Volume ^b (ft ³)	40,000	41,000		52,000	
Primary Sodium Pumps ^b					
Type	Free-surface	Free-surface	NA ^c	Free-surface	Free-surface
Flow (gpm)	62,000	37,000	NA	63,800	150,200

^bFor 100% power condition.^cNA = not available.

Table 4.2-13 (Continued)

	Atomics International	Babcock & Wilcox	Combustion Engineering	General Electric	Westinghouse
Head (ft)	432	119	NA	237	314
Brake Horsepower	7000	1175 (5-hp pony motor)	NA ^c	4000 (30-hp pony motor)	14,500 (100 hp pony motor)
NPSH Available (ft)	30		NA	51	89
Intermediate Heat Exchanger	3 (one per loop)	6 (two per loop)	NA	3 (one per loop)	2 (one per loop)
Type	Shell-and-tube Unit (hockey stick)	Shell-and-tube Unit	NA	Shell-and-tube Unit	Shell-and-tube Unit
Duty per IHX (Btu/hr)	2735×10^6	1396×10^6	NA	2760×10^6	4337×10^6
Surface Area per IHX (ft ²)	29,900	10,509	NA	25,800	49,770
Secondary Sodium Pumps ^b					
Type	Free-surface Centrifugal	Free-surface Centrifugal	NA	Free-surface Centrifugal	Free-surface Centrifugal
Flow (gpm)	55,600	71,000	NA	49,076	109,200
Head (ft)	256	205	NA	243	128
Brake Horsepower (bhp)	4000	4100	NA	3000	10,900
Steam Generator Banks	3 (one per loop)	3 (one per loop)	NA	3 (one per loop)	2 (one per loop)
Evaporator Duty per Bank (Btu/hr)	1610×10^6		NA	1912×10^6	2920×10^6
Evaporator Surface Area per Bank (ft ²)	21,800		NA	35,760	35,000
Superheater Duty per Bank (Btu/hr)	670×10^6	2348×10^6	NA	403×10^6	1520×10^6
Superheater Surface Area per Bank (ft ²)	11,200	53,820	NA	403×10^6	1520×10^6
Reheater Duty per Bank (Btu/hr)	450×10^6	445×10^6	NA	433×10^6	5120×10^6

^bFor 100% power condition.^cNA = not available.

Table 4.2-13 (continued)

	Atomics International	Babcock & Wilcox	Combustion Engineering	General Electric	Westinghouse
Reheater Surface Area per Bank (ft ²)	12,200	17,400	NA	9440	
Sodium Inlet Temperature(°F)	1070	1000	NA	1071	967
Sodium Outlet Temperature (°F)	670	562	NA	645	658
Steam Generator					
Superheater Module					
Type Construction	Shell-and-tube (hockey stick)	Shell-and-tube (once-thru)	NA	Shell-and-tube (once-thru)	Shell-and-tube (once-thru)
Total Number in Plant	3 banks	3 banks	NA	3 banks	2 banks
Number of Modules per Bank	4	1	NA	1	1
Duty per Module (Btu/hr)	167 x 10 ⁶	2348 x 10 ⁶	NA	403 x 10 ⁶	1520 x 10 ⁶
Surface Area per Module (ft ²)	2800	53,820	NA	8040	21,000
Material	Type 321 SS	Croloy 2-1/4	NA	Type 316/Type 304 SS	Incoloy 800/ Type 304 SS
Evaporator Module					
Type Construction	Shell-and-tube Unit	Included with Superheater Data	NA	Shell-and-tube (once-thru)	Included with Superheater Data
Total Number in Plant	3 banks	Included with Superheater Data	NA	3 banks	Included with Superheater Data
Number of Modules per Bank	6	Included with Superheater Data	NA	2	Included with Superheater Data
Duty per Module (Btu/hr)	268 x 10 ⁶	Included with Superheater Data	NA	956 x 10 ⁶	Included with Superheater Data

Table 4.2-13 (continued)

	Atomics International	Babcock & Wilcox	Combustion Engineering	General Electric	Westinghouse
Reheater Module					
Type Construction	Shell-and-tube	Shell-and-tube	NA	Shell-and-tube	No Reheat
Total Number in Plant	3 banks	3 banks	NA	Shell-and-tube	
Number of Modules per Bank	3	1	NA	1	
Duty per Module (Btu/hr)	149×10^6	445×10^6	NA	432×10^6	
Surface Area per Module (ft ²)	4060	17,400	NA	9440	
Turbine Generator					
Type of Machine	Tandem-compound 6-flow, 33.5 LSB	Tandem-compound 6-flow, 30.0 LSB	NA	Tandem-compound 6-flow	Tandem-compound 4-flow, 44.0 LSB
Speed (rpm)	3600	3600	NA	3600	1800
Main Steam Flow (lb/hr)	6.85×10^6	7.35×10^6	NA	7.625×10^6	9.813×10^6
Main Steam Pressure (psig)	2400	2400	NA	2400	2400
Main Steam Temperature (°F)	1000	950	NA	950	900
Reheat Steam Temperature (°F)	1000	950	NA	950	
Reheat Steam Pressure (psia)	515	540	NA	540	
Reheat Steam Flow (lb/hr)	6.4×10^6	6.84×10^6	NA	6.56×10^6	
Condenser Pressure (in. Hg abs)	1.5	1.5	NA	1.5	1.5
Number of Feedwater Heater Stages	7	6	NA	7	7
Final Feedwater Temperature (°F)	478	484	NA	480.7	500
Gross Electrical Output (kW)	1,040,000	1,034,000	NA	1,038,000	1,074,000
Net Electrical Output (kW)	1,002,000	1,009,000	NA	1,011,000	1,044,000
Net Plant Heat Rate (Btu/Whr)	8,175	8281	NA	8192	
Net Plant Efficiency (%)	41.7	41.2	NA		40.2

Table 4.2-13 (continued)

	Atomics International	Babcock & Wilcox	Combustion Engineering	General Electric	Westinghouse
Support Equipment Segment					
Refueling Cycle (months)	12.0	13.3	NA	12.0	13.1
Assemblies Handled per Refueling (avg)					
Fuel	137	96	73	124	84
Inner Blanket	13	22	7	20	85
Outer Blanket	10	12		16	
Control Rod	1				
Sodium Temperature (°F)	500	800	350-400	400	350-400
Refueling Concept	Under the Plug	Under the Plug	Under the Plug	Hot Cell	Hot Cell
Refueling Fractions					
Core and Axial Blanket (Zone 1/Zone B)	0.500	0.333	0.333	0.50/0.33	0.500
Inner Radial Blanket	0.200	0.333	0.111	0.333	0.250
Refueling Time (hr)	266	229	306	168	
Fuel Transfer Machine					
Coolant System Capacity (kW)	50	20	>300	>50	
Shielding Thickness (in)	13	6-23	12		
Shielding Material	Lead	Cast Iron	Lead		
Overall Height (ft.)	-45	40	26	-37.5	35
Design Temperature (°F)		1100	1000	-1100	600
Fuel Storage					
Facilities	1 tank	2 drums	1 tank	2 tanks	1 tank
In-Vessel	No	Yes	No	No	No
Total Storage Capacity (units)		480	300	348	765

Table 4.2-13 (continued)

	Atomics International	Babcock & Wilcox	Combustion Engineering	General Electric	Westinghouse
Coolant for Irradiated Fuel	Sodium	Sodium	Sodium	Sodium	Sodium
Fuel Transfer Tubes					
ID (in.)	8	18	11-1/4	-10	
Length (ft)		6-16	12-5/12	8-10/12	
Site Structures and Services					
Reactor Building					
Shape, dimensions (ft.)	Cylindrical, 140 ID	Cylindrical, 100 ID	NA	Rectangular, 204 x 105	Cylindrical, 100 ID
Inside Height (ft)	121	129	NA	105	
Hemisphere Dome Radius (ft)	70	50	NA		
Overall Height (ft)	191	183	NA	105	
Type of Containment	Double Barrier	Double Barrier	NA		Double Barrier
Containment Leak Rates					
Inner Barrier (% per day)	10.0 (at 20 psi)		NA	1.0 (at 10 psi)	0.1
Outer Barrier at 10 psi (% per day)	0.5	0.1	NA		
Atmospheres					
High Bay	Air	Air	NA	Air	Air
Heat Transfer Vaults	N ₂	Air	NA		

The reactor building contains the reactor, fuel-handling system, radioactive sodium systems, and the potentially most-radioactive portions of the liquid and gaseous waste systems. The building is a conventional reinforced-concrete right-circular cylinder with a flat base and a hemispherical head; the cylindrical section has a 140-ft ID, 121-ft inside height, and 191-ft overall inside height. Radioactive materials are contained by the double containment barriers.

The 200-ft long, 114-ft wide steam generator buildings extends 120 ft above grade and, at its deepest part, 67 ft below grade; it adjoins the turbine generator building.

A.2.2 Core Data and Performance

The reactor core consists of 274 fuel and 15 control subassemblies, surrounded by two rows of blanket and a row of stainless-steel reflector assemblies. The core fuel material is $\text{PuO}_2\text{-UO}_2$, and the blanket material is depleted UO_2 . The fuel and blanket pins are nonvented.

The fuel subassembly consists of 217 wire-wrapped fuel pins, lower- and upper-end fitting assemblies, fuel-pin support bars, and a wrapper tube. The fuel-pin cladding is free-standing, Type 316 SS seamless tubing, 109 in. long, with 0.017-in.-thick walls. The fuel pin is composed of 15 in. of lower axial blanket, 42.84 in. of core fuel pellets, and 12 in. of upper axial blanket; the remaining 38 in. of tube above the upper axial blanket serves as a plenum for fission gases. The hexagonal fuel assembly housing has a 0.140-in.-thick wall and is 5.62 in. across the flats, with rounded corners to reduce bending stress concentrations.

Outside the core are two rows of radial blanket assemblies of two different designs, although their external dimensions are the same as those of the core elements. The 66 inner-row blanket assemblies each have 91 0.500-in.-OD fuel pins; the 72 outer row assemblies each have 61 0.617-in.-OD fuel pins.

The two-enrichment-zone reactor core consists of a 10.5%-enriched inner-core zone that includes 133 fuel assemblies and six safety rods and a 13.5%-enriched outer zone that contains 141 fuel assemblies and nine shim-safety rods. Average discharge fuel exposure is 67,000 MW-day/metric ton. If the core and upper blanket are progressively voided, the radial sodium void worth is $\sim \$8.07$ at maximum, when 80 to 90% of the core is voided, and about $\$7.90$ when the total core is voided. The Doppler coefficient ($T dk/dT$) is -0.0080.

Reactivity is controlled by 15 tantalum control rods. A bank of six rods of high individual worth is above the upper axial blanket during normal power operation and enters the core only during scram or shutdown. The remaining nine control rods, identified as shim safeties, have a low individual worth and are in the active core and upper axial blanket during normal operation.

The core contributes 92% of the 2400-Mwt total reactor output. Maximum hot-spot temperature of the cladding inner surface is 1282° F. Coolant pressure drop through the core is 87 psi; the average coolant temperature rise across the reactor is 360° F, and the bulk outlet coolant temperature is 1140° F. Chromel-Constantan thermocouples are located in the sodium stream discharging from the fuel assemblies.

A.2.3 Plant Data and Performance

The reactor vessel, which is Type 304 SS, has a 1-in.-thick wall, is 29 ft in diameter and extends 48 ft below the floor level. The vessel has three 38-in. inlets and three 44-in. outlets for the main-coolant-loop pipe penetrations and one small inlet penetration for coolant supply to the low-pressure region below the reactor inlet plenum. Sodium enters the vessel at the same elevation as the outlet and flows downward to the high-pressure inlet plenum and upward past the fuel and blanket elements. A double-rotatable shield plug forms the upper closure of the reactor vessel and is an integral part of the fuel-handling system. The reactor cover gas is helium.

The primary system consists of three identical piping loops, connected in parallel at the reactor vessel. Each loop includes piping, reactor inlet and outlet block valves, pump, shell-side of an intermediate heat exchanger, and butterfly-type throttling valve and a check valve--both located in the reactor inlet piping. At full power, the primary sodium in each piping loop flows at 62,800 gpm from the reactor upper plenum, through the primary circulating pump, through the shell side of the intermediate heat exchanger, and back to the reactor lower plenum. Flow through the loops is regulated by pump speed control in order to program sodium temperatures properly throughout the power range. This control method allows electrical load changes without large temperature changes, which could cause thermal shocks and stress problems.

The intermediate heat exchanger is a fixed-tube sheet, tube-and-shell type of the hockey-stick configuration. It is mounted vertically, with a 90° bend at the lower end to provide a short horizontal leg. This design accommodates the differential thermal expansion between the tube and the shell.

The secondary heat-transfer system consists of three identical, independent, separate piping loops, each of which includes the tube-side of an intermediate heat exchanger, a pump, the shell-side of a steam generator, two butterfly-type throttling valves, and two blocking gate valves. The sodium in each secondary loop flows at 55,600 gpm from the upper plenum of the intermediate heat exchanger through the circulating pump and surge tank, downward through the shells of the superheater, reheater, and evaporator modules, and back to the intermediate heat exchanger. Flow through the loop is regulated by pump speed control.

The steam generator bank in each loop consists of an evaporator section and a superheater section that is in parallel with a reheater section. The three sections utilize the modular heat-exchanger concepts, in which enough small units are connected in parallel to provide the required heat-transfer area. Each module is a vertical shell-and-tube unit, with hockey-stick expansion compensation. This modular approach localizes the effects of tube leaks and ruptures. Each loop has one bank of heat-exchange units that consist of four superheater, six evaporator, and three reheater modules. The evaporator material is 2-1/4 Cr-1 Mo, and the superheater and reheater material is Type 321 SS.

The turbine generator is a 3600-rpm, tandem-compound, six-flow, reheat unit. Steam conditions are 2400 psig and 1000° F at the throttle, with 1000° F reheat. The boiler feedwater goes through seven stages in heating the water at 478° F at the evaporator inlet.

The main heat-transfer loops are also used for auxiliary cooling and emergency cooling by forced or natural convection. The steam generated in the evaporator is routed through a pressure-reducing station to the flash tank, then to the superheater, and then through the reheater modules. The exit steam from the superheater and reheater is routed through desuperheaters to the main condenser. This steam dump system is capable of rejecting up to 30% of rated thermal power.

A.2.4 Containment

Radioactive materials are contained by the double-containment barriers indicated in Figure 4.2-22. The inner containment barrier encloses the reactor, the primary heat transfer vaults (including sodium pumps and intermediate heat exchangers), and the primary drain vault. The barrier is a welded 1-1/2-in.-thick steel liner backed up by concrete and is designed for a maximum leak rate of 10% per day of its contained volume at 20 psig. Rupture discs in common walls between vaults permit the entire

PLANT ENGINEERING SAFETY FEATURES (A1)

Figure 4.2-22

enclosed volume to be used for containment of any energy released during a design-basis accident.

During an HCDA, if the fuel melts through the vessel and collects on the bottom of the cavity liner, the reactor cavity liner cooling system will provide a final safeguard against damage of the inner containment barrier.

The reactor containment dome, which covers the reactor top shield and the structure and the equipment installed on it, also forms part of the inner barrier. An energy-absorbing steel structure covers the top shield within the dome and provides shield holddown and missile protection. The outer containment barrier is the reactor building structure; it consists of a welded 3/8-in.-thick steel plate lining the inside of the reinforced-concrete building envelope and is designed for a maximum leak rate of 0.5% per day at 10 psig.

A.2.5 Refueling

Fuel-handling-system operations that require the reactor to be shut down are scheduled concurrently with an annual turbine maintenance period of two weeks. One-half of the fuel assemblies, one-fifth of the inner blanket assemblies, and one-seventh of the outer blanket assemblies in the reactor are to be replaced during an average equilibrium refueling period. In addition, one-fifth of the inner blanket assemblies and one-seventh of the outer blanket assemblies are to be rotated 180° to provide a more uniform plutonium buildup in the blanket assemblies. Spent fuel from the reactor region is lifted vertically into the sodium pool by the in-vessel fuel-handling machine and placed in the in-reactor vessel storage rack. The reactor top shield plugs are rotated in a sodium-filled canister. The spent fuel is removed from the in-vessel transfer position to the out-of-vessel storage tank by the out-of-vessel fuel-handling machine. From this tank, it is removed for shipment. Blanket assemblies and other core components are removed from the reactor in a similar manner. New fuel and blanket assemblies are transferred to the core in the reverse manner.

The out-of-vessel fuel-handling machine and the auxiliary handling machine are mounted side by side on a common gantry. The out-of-vessel fuel-handling machine utilizes a circulating-NaK-cooled inner sleeve, or "cold wall," to remove heat from irradiated core components in canisters. Redundant NaK systems are used to assure continuity of cooling capability. Heaters in the NaK systems are used to maintain preheat of new core components during transfer. During the cooling mode, the cold wall is maintained at 300° F; during the preheat mode, it is maintained at 500° F. The atmosphere in the fuel-handling machine is helium. A maximum heat load of 50 kWt

can be handled with stagnant helium in the 1/4-in. annulus between the canister and the cold wall.

The fuel assemblies have an in-core residence time of two years; the inner radial blanket assemblies, five years; the outer radial blanket assemblies, seven years; and the shim-safety rods, eight years. After each 12-month equilibrium cycle, an average of 137 fuel assemblies, 13 inner blanket assemblies, 10 outer blanket assemblies, and 1 control rod absorber column (a total of 161 assemblies) are exchanged. In addition, 23 blanket assemblies are rotated 180°.

The refueling time is less than the two weeks scheduled for turbine maintenance. The time is estimated at 312 hr (with 85% efficiency) and includes 44 hr for preparation, transition, and termination steps and 268 hr for fuel and blanket handling.

A.3 Babcock and Wilcox

A.3.1 General

The Babcock & Wilcox 1000-MWe LMFBR reference plant uses a pool-type primary coolant system and a tandem-compound reheat turbine. Thermal power of the reactor is 2450 MW and net power output from the plant is 1009 MWe.

The main characteristics of this plant are shown in Table 4.2-13. A cross section of the plant is shown in Figure 4.2-23.

The primary coolant system is a pool type with an under-the-plug refueling system. In this design, all primary system components--including the core, core pressure vessel, radial neutron shielding, primary pump, intermediate heat exchanger, irradiated fuel decay storage, and the cold traps--are housed inside the large primary sodium tank.

The reactor building contains the primary sodium tank with its internal equipment, the argon coolers, reactor-building air conditioning coolers, three argon tanks, a sodium tank, and various equipment-handling and cooling equipment.

The steam-generator building and the turbine-generator building substructures are of reinforced concrete. The superstructures are conventional structural steel frame with insulated metal siding and a precast concrete slab roof.

SECTIONAL VIEW OF REACTOR BUILDING (B&W)

Figure 4.2-23

A.3.2 Core Data and Performance

The reactor core is contained in the core vessel, which has a diameter of 161 in. and is located within the primary sodium tank. The core consists of hexagonal fuel assemblies, blanket assemblies, and control rods arranged in a modified hexagonal configuration with an effective diameter of 120.735 in. and an equivalent length-to-diameter ratio of 0.288.

Each fuel and radial blanket assembly is vertically positioned, supported at the bottom by the grid plate and at the top by the core hold-down structure. The hexagonal fuel assemblies are 6.450 in. across the outside flats and contain 331 0.280-in.-OD fuel pins that are located on a 0.337-in. triangular pitch by helical-ribbon spring spacers. The fuel pins are clad with 0.010-in.-thick, Type 304 SS and are fueled with vibratorily compacted mixed $\text{PuO}_2\text{-UO}_2$ at 85% of theoretical density. The fuel pin has an active fuel length of 34.7 in. with a 14-in. axial blanket of depleted UO_2 at each end and is vented to the coolant through a porous plug. A pre-formed, hexagonal, solid-wall, 0.128-in.-thick fuel assembly can support the fuel pins and spacers.

The radial blanket assemblies are very similar to the fuel assemblies, except that the number and OD of the blanket pins are different; each radial blanket assembly contains 91 0.608-in.-OD pins.

The control-rod system comprises 18 shim-regulating rods and 7 safety rods. Each control-rod drive unit consists of a control rod that contains tantalum as a neutron absorber, an extension rod, a drive mechanism, and the associated electronic circuitry.

The core has three radial fissile-plutonium enrichment zones of 10.4, 11.2, and 13%, respectively. Average discharge fuel burnup is 100,000 MW-day/metric ton. The radial and axial power peaking factors are 1.30 and 1.25, respectively, at the end of cycle. The Doppler coefficient ($\text{d}k/\text{d}T$) for the core is -0.0047. The sodium void reactivity effect of the core and axial blanket is \$7.50. The breeding ratio is 1.36, and the doubling time is 10.4 years.

The reactor produces 2450 MWt and is cooled by upward-flowing sodium entering the reactor at 800° F and exiting at a mixed mean temperature of 1100° F; the reactor pressure drop is 35 psi. The maximum temperature of the cladding surface at 103% power is 1209° F. The maximum fuel center-line temperature at 111% power and beginning of life is 4447° F.

The core instrumentation system includes four subsystems that monitor core performance. The instrumentation subsystems monitor core neutron flux, fuel assembly outlet temperature, fuel assembly flow, and core differential pressure. The core neutron instrumentation continuously measures neutron flux at 22 points within the core. A thermocouple placed in the coolant outlet flow path of each of the 288 fuel assemblies is used as a sensing device for a fuel assembly outlet temperature monitor. Flowmeters are located at eight fuel assemblies; each flowmeter is welded to the lower support grid structure and measures the coolant flow through one fuel assembly. Three identical core differential-pressure detectors are provided for protection of the core; each detector consists of a differential-pressure transducer connected by metal pipe to the core outlet and outlet plenums.

A.3.3 Plant Data and Performance

The primary coolant pumps, intermediate heat exchangers, and other primary equipment are suspended from the cover structure and penetrate into the primary sodium tank.

The primary sodium tank, which is located off-center within the reactor building, is 52-ft. in diameter and 44.5-ft. high and has a 1.5-in.-thick wall. Design pressure and temperature of the tank are 30 psig and 925° F.

Each of six primary coolant system loops consists of a variable-speed centrifugal pump, a shell-and-tube heat exchanger, and 28-in.-OD coolant piping to route sodium from the pump discharge to the reactor inlet plenum. The primary pumps draw 222,000 gpm of 800° F sodium from the tank and force it through the coolant piping, the core inlet plenum, and the reactor; sodium from the reactor outlet plenum flows by gravity to the intermediate heat exchangers at 1100° F. For loads between 20% and 100% of rated plant output, the core inlet temperature remains constant, but the core outlet temperature is permitted to vary to maintain constant steam conditions. Varying the flow rate of primary and secondary coolant is the method used for handling plant load changes.

Primary sodium is purified by two Fermi-type modular cold traps and a small centrifugal pump located in the cover gas space of the reactor vessel. The sodium flow rate through each trap is 100 gpm, and each trap can retain approximately 250 lb of oxide. Heat is removed from the trap via a closed-loop NaK system.

The intermediate coolant system comprises three independent loops, each containing two intermediate heat exchangers, one steam generator, one reheater, and one centrifugal pump that is located in the cold leg of the loop. Piping in each loop is

arranged so that the intermediate coolant pump discharges into a single pipe which then splits before entering the reactor building and connecting to the intermediate heat exchangers. Return piping from the intermediate heat exchangers joins outside the reactor building before connecting to the steam generator and reheater. Each loop circulates 30.70×10^6 lb/hr of sodium at 100% load, with 22.0×10^6 lb/hr for the steam generator and 8.50×10^6 lb/hr for the reheater. The sodium enters the steam generator and the reheater at 1000° F and leaves them at 652° F and 825° F, respectively. The exit streams are mixed and enter the intermediate heat exchangers at 700° F. The steam generator is of a vertically mounted and bottom-supported once-through design. The heat transfer surface is a helical-coil, single-bundle design with counterflow of water and sodium. The sodium reheater is similar in design to the steam generator.

The turbine unit is a 3600-rpm, tandem-compound, six-flow reheat machine; it consists of a single-flow high-pressure turbine, a double-flow intermediate-reheat turbine, and three identical double-flow low-pressure turbines. Six stages of the boiler feedwater heating are used to heat the feedwater to 484° F before it enters the steam generator.

A decay-heat removal system is used to remove fission-product decay heat from the core after a reactor shutdown. The system consists of two independent closed loops, each with a decay-heat removal capability of 50.2 MWt. Each loop consists of one Na-NaK heat exchanger, one NaK-air heat exchanger, three electromagnetic pumps, a circumferential biological-shield cooling coil, and other equipment. The Na-NaK heat exchanger is submerged in the primary coolant, and the NaK-air heat exchanger is on the roof of the steam-generator building. If both Na-NaK heat exchangers are damaged after a design basis accident, heat is removed by the biological-shield cooling coils.

A.3.4 Containment

The containment building has an air atmosphere and is accessible to personnel during all normal and reactor on-load operations. In the event of an HCDA, the primary sodium tank and its cover structure need not necessarily be leakproof, but any possibility of a major sodium spray ejection followed by a fire must be eliminated. The reactor building is the containment barrier. No damaging effects (such as shock waves, sodium hammer, internal pressure, or temperature effects capable of causing damage to structural components) are expected to propagate beyond the primary sodium tank as the first barrier. This first barrier is intended to prevent both major leakage of radioactive material and sodium fires. The containment barrier completely

envelops the primary sodium tank and is intended to act as a final barrier for any radioactive material. A backup emergency cooling system is provided for use if systems immersed in the primary sodium are inoperative. This system is to be activated no later than 3 hr after an accident, at which time the cooling requirement would be 1% of normal reactor power.

The reactor building is a 4-ft-thick reinforced-concrete cylinder, 183 ft high above grade level and with a 100-ft ID. It has a flat foundation slab and a hemispherical head which is also 4 ft thick. Within this concrete shield structure is a carbon steel liner (3/8 in. thick for the dome and wall and 1/4 in. thick for the base). This liner provides an airtight seal to prevent leakage and is thus the barrier for radioactive materials. The design pressure is 10 psig at the design temperature of 200° F.

A.3.5 Refueling

For fuel handling within the reactor plant, an under-the-plug refueling mechanism transfers fuel and blanket assemblies from core positions to decay storage positions during reactor shutdown periods. The transfer is made while the subassemblies are submerged in sodium coolant. Subsequent transfer from decay storage during full-power operation is accomplished with a fuel-transfer machine.

The refueling mechanism is mounted on the smallest of the three rotating plugs in the primary tank cover. The mechanism weighs approximately 2500 lb and consists of a housing, a guide tube, a grapple tool, and a grapple-tool drive system. The fuel-transfer machine is ~40 ft high and 8 ft in diameter and weighs ~200 tons. A circular stainless steel vessel ~29 ft high by 3.33 ft in diameter is the basic component of the machine and is designed to retain its argon gas atmosphere at 100 psig and 1100° F. This vessel is surrounded by cast iron shielding and thermal insulation. The radial shield is 23 in. thick on the lower 19 ft, tapering to about 6 in. thick at the top. A telescoping section of shielding is attached to the bottom end of the machine and is lowered into place for personnel shielding each time the machine is seated on an access port.

During handling by the fuel-transfer machine, a forced-flow argon gas heat-exchanger system fitted directly to the machine removes decay heat from spent fuel assemblies and preheats the new fuel assemblies. The heat exchanger heats new fuel assemblies to ~800°F and cools spent fuel assemblies being removed to ~500°F. After a 70-day decay storage period, the fuel cladding temperature can be limited to a maximum of 500°F with a cooling gas inlet temperature of 100°F and a flow rate of 1341 lb/hr.

The rotating fuel storage drums, which provide storage for 100% of the core fuel and blanket assemblies, are installed in the primary sodium tank. They are located 180 degrees apart in the annulus formed by the core and tank. The central support shaft of the drum extends upward, penetrating the primary tank cover structure, and is connected to a drive and indexing mechanism at the operating floor level.

The refueling time, estimated at 229 hr, is based on 3 hr for reactor cooling to 800°F, 7.9 hr for the refueling preparation, 60.8 hr for refueling of one-third of the core and blanket assemblies, 7.9 hr for preparation for startup, 148 hr for physics testing performed three times during the refueling period, and 1.5 hr for startup procedure to 80% of full power.

A.4 Combustion Engineering

A.4.1 General

The Combustion Engineering work was performed on the heat generation system only; their design does not include a heat-transfer system, an energy conversion system, plant layout, and buildings. The heat generation system is composed of reactor core, reactivity controls, reactor-vessel, reactor support systems, and safety system.

The CE 1000-MWe LMFBR reference reactor has a flat cylindrical core with a thermal power of 2465 MW. The reactor design assumes that the plant has a loop-type primary coolant system, an intermediate heat-transfer system, and a turbine generator with steam conditions of 2400 psig and 900°F with reheat of 900°F. The main characteristics of this reactor are given in Table 4.2-13. The cross section of the reactor is shown in Figure 4.2-24.

A.4.2 Core Data and Performance

The reactor core has an equivalent diameter of 8.87 ft in the fuel region and a length-to-diameter ratio of 0.225. The reactor is composed of 219 fuel, 60 radial blanket, and 22 control subassemblies.

Each fuel subassembly contains 163 fuel pins on a triangular pitch, pin spacers, a shroud can, and upper and lower shroud fittings. The subassembly is 12 ft long and has a 6.527-in. OD (hot). Fuel-pin lateral support is provided by helical-wrap spacers tentatively placed at 12-in. axial intervals. The 0.050-in.-thick cylindrical shroud acts as a pressure-containing flow channel and overall stiffener, and six

REACTOR VESSEL ARRANGEMENT (CE)

Figure 4.2-24

0.30-in.-thick semicircular elements containing BeO moderator are attached to the outside of the subassembly shroud.

The fuel pin is clad with Type 316 stainless steel and contains mixed uranium-plutonium carbide fuel with depleted uranium carbide axial-blanket material at each end. The pins are sodium-bonded and vent fission gas directly to the sodium coolant. The outside diameter of the fuel pin is 0.400 in. and the cladding thickness is 0.011 in. Each fuel pin contains 18 in. of upper blanket material, 8 in. of vent system, 10 in. of expansion space, 24 in. of active fuel, 18 in. of lower blanket material, and plugs in each end.

Each hexagonal radial-blanket subassembly has a 0.100-in.-thick shroud and contains 91 pins, each with a 0.60-in. OD. The radial-blanket pins are nonvented, and their cladding is 0.020 in. thick. Depleted uranium carbide fuel slugs are 0.52 in. in diam, and the blanket fuel column is 3.0 ft high.

The control rod system consists of 13 shim-regulating rods containing tantalum, 9 safety rods containing B_4C , and electromechanical drives and auxiliary equipment. The stainless steel shroud of a control assembly has the same external dimension as a core fuel assembly.

The core is divided into two zones of 9.78% and 11.78% fissile plutonium enrichments. Average discharge fuel exposure is 100,000 MW-day/metric tons. The radial power peaking factor is 1.214, and the axial value is 1.191. The isothermal Doppler coefficient with sodium ($T dk/dT$), is -0.0064. The sodium void reactivity effect of the removal of sodium from the core and axial blanket is 1.36%. The breeding ratio of the reactor is 1.441, and the doubling time is 7.99 years.

The reactor generates 2465 MWt and is cooled by upward-flowing sodium. The coolant enters the reactor at 797°F and exits at a mixed mean temperature of 1095°F. The peak velocity and pressure drop of the coolant in the reactor are 32.7 ft/sec and 75.0 psi. The maximum cladding inside-surface temperature of the hottest pin at full power is 1223°F; the peak temperature of fuel at the hottest pin is 2310°F at full power. (These temperatures do not include operating allowances and statistical uncertainties.)

The cylindrical reactor vessel is stainless steel and is approximately 16.5 ft in diameter and 36-ft long, flanged at the top end and provided with an elliptical shell at the lower end. A nonrotating removable inner plug and fixed ring plug form the

top closure to the vessel. The vessel is equipped with six inlet and six outlet nozzles that are above the core level. In addition, two emergency cooling loops are provided to furnish coolant flow if the main coolant loops are inoperable.

A.4.3 Plant Data and Performance

(Not included in CE work.)

A.4.4 Containment

(Not included in CE work.)

A.4.5 Refueling

The refueling procedure is based on an under-the-plug refueling concept that employs a plug-exchange scheme and a "wet" transfer of fuel subassemblies from the core to the decay storage tank.

The subassembly handling machine and its integral shield plug constitute a refueling island that is inserted into the center of a fixed shield plug in the top of the reactor vessel. The subassembly handling island and the control-rod drive island have identically-shaped plugs so that they are interchangeable. All the control-rod drives and the subassembly monitoring system are attached to the removable center of the shield plug. The control-rod drives and extensions, and the assembly monitoring system are also attached to the removable center of the shield plug. The control-rod drives and extensions and the assembly monitoring system may be kept at the storage tank during reactor refueling.

The subassembly handling machine extracts the assembly from the core position and deposits it into a pot at the transfer location. This operation is carried out under the sodium, which is kept at 350 to 400°F. The subassembly transfer machine transfers the pot with the spent fuel subassembly to the decay storage tank and lowers the pot into a transfer location within the tank. The operating temperature range of the transfer machine is 350 to 1000° F; system pressure range is 0.2 to 0.5 psig. A three-loop intermediate NaK heat-transfer system on the transfer machine enables the decay heat in the subassembly to be rejected to the containment building air during the transfer operations. Each loop can transfer 103 kWt. The reverse procedure is used for moving the new fuel subassemblies into the core. During refueling, the fuel-pin cladding of the hottest subassembly can be maintained at 800° F, assuming transfer starts 10 hr after shutdown. The barrel of the transfer machine is surrounded by a 12-in. lead shield for personnel protection.

The assembly handling machine has three degrees of freedom: 360° rotation about the core vertical centerline, vertical movement of the radius arm, and horizontal movement of the gripper assembly. The complete unit rotates on a 4-ft-diam bearing located between the drive mounting plate and the removable shield plug. Refueling is scheduled approximately once a year, and 90 subassemblies are transferred. The refueling time is estimated at 106 hr, and it consists of 11.5 hr for the preparation, 82 hr for the fuel movement, and 12.5 hr for the preparation of the reactor operation.

A.5 General Electric

A.5.1 General

The reference LMFBR plant has an oxide-fueled reactor, a pool-type primary coolant system, and a tandem-compound, six-flow reheat turbine. The net power output from the plant is 1008 MWe. Steam conditions for the plant are 2400 psia and 950°F with reheat of 950°F; the overall station efficiency is 41.7%. The basic plant parameters are given in Table 4.2-13. A vertical cross section of the facility is shown in Figure 4.2-25.

The basic design philosophy of the GE concept is to reduce the probability of large accidents and to confine small ones. In its safety assumptions, the advanced concept considers that engineered safeguards will prevent or mitigate any accident. There are no blast shields within the primary vessel to absorb large quantities of energy during a hypothetical fuel meltdown or large sodium voiding accident.

A.5.2 Core Data and Performance

The reactor is a right circular cylinder with a 128.7-in. mean-OD reflector region; it is 111 in. high and is composed of 234 fuel subassemblies, 31 control-rod fuel subassemblies, 126 blanket subassemblies, and 72 reflector subassemblies. The reactor is contained in the 14.5-ft-diam reactor vessel, which is located in the primary sodium tank. The core has a mean diameter of 97.4 in. and a height of 30 in. The height-to-diameter ratio of the core is 0.308. The axial blanket is 15 in. thick at both ends of the fuel pin. The reactor has a 10.5-in.-thick radial blanket zone.

Each fuel subassembly contains 331 fuel pins of mixed UO_2 - PuO_2 , and the fuel subassemblies modified for control rods contain 204 fuel pins. Each radial-blanket subassembly contains 127 pins. The cross section of the subassemblies is hexagonal, and the pins are arranged in a triangular array. Each vented fuel and nonvented

4.2-194

VIEW THROUGH CROSS SECTION (GE)

Figure 4.2-25

radial-blanket pin is supported at its lower end and is subjected to an upward flow of sodium.

The upper 21 in. of the fuel pin forms a space for venting fission-product gas to the coolant stream, thus preventing a buildup of internal gas pressure. The entire fuel pin is enclosed in a Type 316 stainless steel cladding tube with 0.248-in. OD and 0.010-in.-thick wall. The outside diameter of the radial blanket pins is 0.422 in.; the wall is 0.015 in. thick. Helical wire spacers are wrapped around the fuel pins to control their spacing; the wires have a pitch of 6 in. per turn.

The core is divided into two radial fuel enrichment zones for power flattening. The axial power peaking factor is 1.22 at beginning of burnup cycle and 1.17 at end of cycle.

The maximum total neutron flux in the reactor is 1.1×10^{16} neutrons/cm²-sec, and the neutron flux above 825 keV, which is an important indicator of possible neutron irradiation damage, is about 13% of the total flux. The average discharge-fuel exposure is 100,000 MW-day/metric tons. The Doppler coefficient ($T dk/dT$) for the two-zone, power-flattened core is calculated to be -0.0060 with sodium in and -0.0037 with sodium removed. The sodium void reactivity effect of the removal of sodium from the core and blanket is \$2.24. There are 12 tantalum rods for shim and burnup control and 19 B₄C rods for scram.

Thermal output of the reactor core is 2118 MWt out of a total output of 2417 MWt. Maximum fuel design temperature is 5200° F at 10% overpower condition, and maximum cladding temperature (mean radius) is 1300°F at full power.

The overall core pressure drop is 64.3 psi, based on a total coolant flow of 80.6×10^6 lb/hr, with a net temperature rise of 340°F. The maximum flow rate of coolant in the core is 25.4 ft/sec. The sodium is forced upward through the reactor core. The flow and outlet temperature of the coolant for each fuel subassembly are measured. Sodium temperature is also measured at the outlet of each primary pump and at the inlet of each intermediate heat exchanger.

A.5.3 Plant Data and Performance

The GE plant concept employs a three-loop primary system and a three-loop secondary system that feeds steam to a tandem-compound, six-flow reheat turbine. The main characteristics of the plant are given in Table 4.2-13.

In the pool-type primary system, the heat-transfer system components and the reactor core are located within a 48-ft-diam, 40-ft-high tank filled with sodium. The components of the primary heat-transfer system are submerged in the sodium and are accessible through removal of a shielding plug.

The major components of the primary heat-transfer system are the primary sodium tank, containment vessel, reactor vessel, intermediate heat exchangers, and primary pumps. The sodium from the primary tanks flows to the suction side of the three primary centrifugal pumps and is discharged into the reactor vessel plenum below the core. At the reactor core outlet, the sodium mixes and then flows through three parallel pipes to the intermediate heat exchangers. The sodium is cooled as it passes through the tubes of the intermediate heat exchangers and then is discharged within the primary tank. Flow through the tubes of the intermediate heat exchanger may be stopped by gradually increasing the pressure of the cover gas. When this pressure slightly exceeds that of the reactor-vessel cover gas, flow to the unit will stop.

The secondary heat-transfer system consists of three independent heat-transfer circuits, each with independent pumps and steam generators. The steam generators consist of two evaporators, a superheater, and a reheater for each circuit. The heated sodium from the intermediate heat exchanger is split into two streams that flow through the shell side of the steam superheater and reheater in parallel. After leaving the superheater and reheater units, the two sodium streams are mixed in a header. From the header the sodium is again split into two streams that flow in parallel through the shell sides of two water evaporators to the sodium pump.

The plant utilizes a regenerative reheat steam cycle, with a tandem-compound, six-flow reheat turbine. Steam at 2400 psia and 950°F is fed to high-pressure turbines; the high-pressure turbine exhaust steam is reheated to 950°F and enters the intermediate-pressure turbine at 540 psia. Seven stages of boiler feedwater heating heat the water to 480°F before it enters the evaporators.

Load change is automatically controlled from 20% of full load to full load. To minimize thermal shock of the fuel cladding and reactor structural components during load changes, the reactor coolant outlet and inlet temperatures are maintained approximately constant as the load is varied. This action is accomplished by varying the primary coolant flow rate by pump speed control. Flow rate of the secondary sodium coolant is also varied with load.

The main heat-transfer loops are also used as an auxiliary cooling system for normal reactor shutdown cooling and emergency cooling. In normal reactor shutdown cooling, the primary sodium pumps are rotated at 20% of their full speed by pony motors. The secondary sodium pump of one secondary sodium loop is operated at 50% of its full speed by its main drive motor while the remaining secondary sodium pumps are operated at 20% of full speed by pony motors. Steam flows from the two evaporators of the selected loop through the startup steam separator and the desuperheater to the main condenser. If the plant site suffers a power loss, the reactor is scrammed from full power and heat is rejected from the steam through a special high-pressure condenser instead of the main condenser. The primary and secondary sodium pumps coast down to 20% of full speed and are maintained at that speed by power from the emergency power system to the pony pump motors. The high-pressure condenser and its condensate pump are designed for a maximum operating pressure of 800 psia.

A.5.4 Buildings and Containment

Figure 4.2-25 shows a vertical cross section of the power plant. The reactor, refueling cell, and support systems (sodium storage, cold traps, gas-handling systems) are located in a reinforced-concrete structure. A concrete wall rises 28 ft above grade from this structure to protect primary system components from tornado-generated missiles.

The reactor building is 105 ft wide by 204 ft long and extends 105 ft above grade and 50 ft below grade. The reactor cell, below-grade structures, and foundations are reinforced concrete, while the above-grade structure is steel framing and insulated metal siding. For sodium systems, the cells in the reactor building are filled with nitrogen to minimize the possibility of a sodium fire. If radioactive material leaks into the area, the building effluent is discharged through a filtering-scrubbing system and up the stack. Adjacent to the reactor building are the steam-generating building and the turbine building. The turbine building is the largest structure in the plant; it is 300 ft long, 110 ft wide, and 107 ft tall.

By definition, the primary containment boundary contains all the primary coolant and cover gas. Portions of the sodium treatment system and cover gas system are excluded from the primary containment when they can be isolated from the reactor by a quick-operating valve. Primary containment is provided by a carbon steel outer tank surrounding the primary sodium tank on the bottom, sides, and top. Outside both tanks is concrete shielding with a steel liner. These various tanks and liner maintain coolant integrity in the event of a vessel failure.

The inside of the primary containment is insulated, and the outer surface of the primary containment is cooled by forced circulation of cold nitrogen gas. In this manner, the primary containment is maintained essentially at room temperature, thus permitting the use of carbon steel and eliminating problems of differential thermal expansion in relation to the various fixed penetrations and shield structure. The nitrogen gas must be circulated at all times to prevent overstress of the carbon steel containment vessel.

The primary containment extends upward around the supports for the reactor, intermediate heat exchangers, and primary sodium pumps. During refueling, the reactor is shut down, and the shield plug over the reactor is raised. At this time, the walls of the refueling cell act as the primary containment barrier. The primary containment, refueling cell, and reactor building are not designed to withstand damage caused by pressure rise.

The rectangular reactor building does not constitute containment; however, the building is designed for low leakage and is operated at a slight negative pressure.

A.5.5 Refueling

The reactor is refueled by raising the shield plug inside a hot cell which has an inert atmosphere of argon. The reactor is refueled once a year. The operations are preplanned, with computer control of all fuel accounting, including location of positions in the core. All operations are monitored by direct viewing by the operators. The refueling operations can be manually controlled in the event of unforeseen difficulties, such as a sticking fuel assembly.

After reactor shutdown and total insertion of all control rods, the entire primary sodium tank is cooled gradually to 400°F. During this operation, the control-rod drives are all disconnected. The operations are performed by the telescoping-tube overhead manipulator. Also during cooldown, the reactor cover gas is purged by flowing in fresh argon and exhausting the old gas to the radioactive-gas-waste-handling system.

The overhead manipulator disconnects all gas and electrical connections to the shield plug, including the multiple connector plugs for thermocouple wiring and the wiring and pipe connections for the control-rod drive system. The shield plug is raised slowly for the first 3 ft as a final precaution against the possibility that some control rods are still connected to their drives. Lifting is achieved by two

ball-nuts on vertical lead screws driven simultaneously by a motor outside the refueling cell.

The fuel-shuffling machine transfers fuel and blanket assemblies under sodium between the reactor core and the transfer thimbles. The transfer thimbles are in the 24 temporary locations surrounding the reactor.

The fuel-transfer machine moves the assembly transfer thimbles, which contain the fuel, between the temporary storage positions around the core and either the fuel decay tank or the new-fuel storage tank. When the transfer is from the reactor vessel, the transfer thimbles remain filled with sodium to ensure adequate heat transfer from the fuel bundles that have high heat generation--as much as 50 kWt each. As additional heat removal capacity, a heat exchanger that is filled with NaK and sealed is an integral part of the fuel transfer-machine grapple. The number of assemblies replaced per year is 160, including 36 radial blanket units.

The total refueling time, estimated at 168 hr, includes 22 hr from reactor shutdown to raising shield plug, 112 hr for fuel-assembly handling, and 34 hr for reactor startup.

A.6 Westinghouse

A.6.1 General

The Westinghouse 1000-MWe LMFBR reference plant utilizes carbide fuel, a loop-type primary coolant system, and a nonreheat turbine. The main characteristics of the plant are given in Table 4.2-13. Figure 4.2-26 shows a cross section through the main part of the power plant. The containment building contains all the sodium primary components, including the reactor, intermediate heat exchangers, pumps, hot cell, and support systems. Cells in this building contain inert gas to prevent any possibility of sodium fire.

The reactor hot cell provides a containment above the reactor. The inner barrier of the hot cell is a cylindrical carbon steel shell, 100 ft in diameter and 1-1/8 in. thick, with a hemispherical dome. The intermediate heat exchanger vault is located below grade in the containment building. It contains intermediate heat exchangers, sodium pumps, piping and valves, and two primary loops.

4.2-200

REACTOR PRIMARY AND SECONDARY SYSTEM ELEVATION VIEW (WESTINGHOUSE)

Figure 4.2-26

A.6.2 Core Data and Performance

The core is composed of four modules, each with 61 fuel subassemblies separated by three rows of blanket subassemblies. The equivalent modular core diameter is 3.888 ft. The length-to-diameter ratio of one module is 0.842.

Each of the fuel subassemblies contains 168 fuel pins in a square array, with 28 pin positions used by support rods or control elements. The fuel pin cladding is Type 316 stainless steel and contains mixed uranium-plutonium carbide fuel with depleted uranium carbide axial-blanket material at each end. The pins are sodium-bonded and vented directly to the sodium coolant. The pin OD is 0.302 in., and the cladding is 0.012-in. thick. The fuel pin consists of 25 in. of sodium expansion and fission gas vent space, 15 in. of upper blanket material, 39.3 in. of active fuel, 15 in. of lower blanket material, and 6 in. of end plug shielding. The fuel subassemblies are open lattice and are not enclosed in a structural can. Grid-type fuel-pin spacers are employed.

The radial blanket subassemblies each contain 60 pins plus four structural members. The blanket pins also have Type 316 stainless steel cladding, but they contain depleted uranium dioxide over their entire length. These pins are gas bonded and nonvented.

The core of each of the four modules is divided into two enrichment zones. The fresh feed for the inner zone is 0.145 and the outer zone is 0.174 fissile plutonium concentration. Average discharge fuel exposure of the core is 100,000 MW-day/metric tons. The isothermal Doppler coefficient for the reactor core is -0.00395. The reactor has a breeding ratio of 1.325 and a doubling time of 9.9 years.

The control system consists of control rods with the associated drive mechanisms and electronic circuitry. The 60 control rods specified for this reactor are driven by actuating mechanisms located on top of the reactor vessel. The control rods are cruciform in cross section and are inserted into cruciform-shaped guide channels, formed at the juncture of four fuel assemblies. There are four safety, eight shim, and three regulating rods for each module. Tantalum is the absorber material in the rods.

The thermal output of the reactor core plus axial blanket is 2480 MW and of the radial blanket is 120 MW, at the beginning of core life. Maximum temperatures of fuel center and cladding are 2622° F and 1400° F. The coolant pressure drop in the fuel assembly

is 56 psi. The average core temperature rise of the reactor is 230°F, and the core outlet temperature is 1000°F. The average coolant velocity in the core is 27 ft/sec.

The nuclear instrumentation system, thermocouples, and other instruments make up the in-vessel instrumentation. The neutron flux levels are measured by the low-, intermediate-, and high-power flux instruments. The instrument measurements allow monitoring of the core power as well as of the control-rod positions. Core outlet temperature is monitored by thermocouples positioned on the basis of one thermocouple for four fuel assemblies. Other in-vessel instruments are NaK pressure transducers to give core pressure drop, an ion chamber in the cover gas to measure the activity, and sodium level indicators.

A.6.3 Plant Data and Performance

The reference plant utilizes a loop-type primary system for removing the 2600 MWt of heat from the reactor. This system includes a reactor vessel, primary pumps, intermediate heat exchangers, and interconnecting piping.

The reactor vessel is a 17.2-ft-ID single-wall cylindrical shell with a hemispherical bottom head and a flat shield plug closure. All nozzle penetration elevations are located above the top of the core and include two 48-in. inlet and two 48-in. main coolant outlet nozzles, two 22-in. auxiliary cooling system nozzles, a sodium overflow nozzle, an argon outlet nozzle, instrumentation ports, and a 24-in. fuel-transfer port. Sodium enters the vessel through the two divergent inlet nozzles and flows down an annulus formed by the core support barrel and vessel into a lower plenum, up through the core and blanket, and out through transition spacers between the vessel and core support barrel to the outlet nozzles. The vessel is surrounded by a minimum-volume guard tank to prevent the sodium level from dropping below the bottom of the main coolant nozzles in the event of a vessel rupture.

The primary coolant system consists of two main parallel heat removal loops. The plant also features an independent auxiliary cooling system. A single pump and intermediate heat exchanger are used in each primary loop, and these components are connected by single-walled piping. Neither isolation nor throttle valves are used in the main primary loops. Check valves are installed within the pump casing to prevent reverse flow. Flow control for startup and partial-load operation is provided by varying the pump speed. To increase or decrease reactor power to the desired operating level, from 25 to 100% of full power, pump flows are adjusted automatically or manually, as necessary, to maintain the desired temperature rise across the reactor and secondary side of the intermediate heat exchangers. In

general, the variation in primary system flow rate is approximately linear with core power.

The intermediate heat exchangers are vertical, counter-flow, tube-in-shell units with a maximum capacity of 1300 MWt each at full power. Primary sodium is confined to the shell side and secondary sodium to the tube side of the unit. The entire primary system is located within the containment building and is contained in steel-lined vaults below the reactor operating floor.

The secondary coolant system is composed of two parallel independent heat-transfer loops. Each loop consists of one intermediate heat exchanger, one steam generator, one pump, and connecting piping. At full power operation, a total sodium flow of 47.6×10^6 lb/hr per loop maintains a system hot-leg temperature of 967°F and a cold-leg temperature of 658°F, while transporting 1300 MWt of heat per loop.

The design of the steam generator is based on the once-through, single-walled tube concept, in which heat is added continuously to the water/steam as it flows through the tubes and finally emerges as high-pressure, superheated steam. The unit is of the tube-in-shell design, vertically mounted, with secondary sodium on the shell in six serpentine tube banks or modules enclosed in a vertical cylindrical shell. Water/steam flows in counter-cross flow to the sodium. Under normal operating conditions, the feedwater, which enters at the bottom of the steam generator, exits from the top as high-pressure superheated steam at 903°F and 2430 psia.

The energy conversion system employs a nonreheat moisture-separation steam cycle and develops a net station output of 1044 MWe. This output corresponds to a net plant heat rate of 8500 Btu/kWhr. The turbine is a direct-coupled, 1800-rpm, tandem-compound, four-flow unit with 44-in. low-pressure exhaust blading. Seven stages of the boiler feedwater heating heat the feedwater up to the steam generator inlet temperature of 500°F.

The auxiliary cooling system consists of a single heat-transport system connected directly to the reactor vessel and independent of the primary heat removal systems. The auxiliary loop transports sodium through the reactor vessel to remove core decay heat and through the shell side of an auxiliary heat exchanger to reject heat. System flow is provided by a 12,800-gpm electromagnetic pump located in the hot leg. Heat is transported by NaK from the auxiliary heat exchangers to three forced-air heaters. The system has a maximum heat removal capability of 108.5 MWt at core inlet and outlet temperatures of 770°F and 1000°F, respectively.

A.6.4 Containment

The reactor containment is of the double-barrier type, consisting of a gas-tight, free-standing, 1-1/8-in. steel shell inside a 4-ft-thick concrete shield. The 3-ft air-atmosphere annular space between these two barriers is maintained at a slightly negative pressure by blowers that exhaust through filters to the exhaust stack.

The containment building is composed of the reactor hot cell (located above grade) and the reactor vault and a series of equipment vaults (located below grade). The building is designed to contain all inadvertent releases of radioactive material caused by conditions up to and including those in the design basis accidents: the worst sodium fire accident and the core meltdown and compaction accident. It is also designed to protect the reactor and safeguard all systems from environmental disturbances.

A number of hypothetical sodium-fire accidents were studied, in which there was postulated either a fire in an intermediate heat exchanger vault with sodium covering the floor or a fire in the exposed vessel and fuel storage units. These studies resulted in the choice of a maximum containment pressure of 30 psig and a temperature of 500°F. The sodium-fire design basis accident is the limiting case, because preliminary meltdown accident calculations resulted in containment pressures of less than 30 psig.

Radial blast shielding is provided to protect the containment structure from shock wave damage in the event of a meltdown-and-reassembly accident. The shield will absorb the shock resulting from a maximum predicted explosion equivalent to 3800 lb of TNT.

A.6.5 Refueling

The reactor is refueled once a year by raising the shield plug inside a hot cell which has an inert atmosphere of argon. The hot cell is a 100-ft-dia concrete-shielded steel containment building that contains the reactor refueling system components. The upper closures of the reactor vessel and the fuel storage tank project through the floor of the hot cell, permitting access to them from within the hot cell. Laydown space on the cell floor is provided for the reactor shield plug, upper internals assembly, and fuel storage tank cover. Operations within the hot cell are observed from an air-filled, shielded, refueling-control tunnel.

The main fuel-transporting assemblies in the hot cell are two vertical fuel-handling machines. One of the vertical fuel-handling machines transfers core assemblies between the reactor and the lateral fuel-handling machine; it can also rotate to provide angular orientation when placing core assemblies into the reactor and moving existing core assemblies from one position to another within the reactor. In addition, the carriage is capable of accepting special tools for performing maintenance within the reactor vessel. The other vertical fuel-handling machine moves core assemblies between the fuel-storage tank and the lateral fuel-handling machine. It also has a can-handling mechanism that can transport a sodium-filled transfer can containing fuel between the spent-fuel pit and the fuel-service area. The lateral fuel-handling machine is used for moving core assemblies between the reactor and the fuel storage tank without removing them from the primary sodium. The machine accepts the assemblies from the first vertical fuel-handling machine, rotates them into a horizontal position, moves them through a horizontal run of pipe that connects the reactor vessel and the spent-fuel storage tank, and then returns the assemblies to the vertical position for subsequent handling by the other vertical fuel-handling machine.

Control consoles for both vertical fuel-handling machines and the lateral fuel-handling machine are located in the refueling-control tunnel. A semiautomatic control system is provided for these machines. In this system, individual fuel-handling steps are initiated by an operator and then carried to completion automatically.

Direct viewing of refueling operations is provided through five shield windows in the refueling-control tunnel. Plugged ports are provided at each window to permit installation of periscopes if necessary. Hot-cell surveillance is provided by five television cameras.

To prepare for refueling, the reactor is shut down, the control rods are disconnected from their drive-rod extension shafts by personnel wearing environmental suits, and the primary coolant system temperature is reduced to the refueling temperature of 350 to 400°F. The reactor and spent-fuel tank cover gas volumes are purged to remove all residual gaseous fission products.

4.2.7 ANNEX B: SAFETY RELATED CRITERIA AND DESIGN
FEATURES IN THE CLINCH RIVER BREEDER REACTOR PLANT*

B.1 Summary Design Description

The Clinch River Breeder Reactor Plant (CRBRP) is being designed to be located on a TVA site in Tennessee adjacent to the Oak Ridge National Laboratory. The plant will demonstrate an alternate power option which could greatly extend our fuel resources. The plant net electrical output of about 350 megawatts will be utilized in the TVA system.

The principal plant parameters are indicated in Table 4.2-14 and the reactor parameters in Table 4.2-15. The reactor core plan is shown in Figure 4.2-27 and the reactor elevation in Figure 4.2-28. The core, consisting of hexagonal fuel assemblies in stainless steel cans, is divided into two enrichment zones, each fueled with mixed uranium-plutonium dioxide. The fuel assemblies include upper and lower blankets of depleted uranium dioxide. Radial blanket assemblies surround the core, followed by stainless steel reflector assemblies. The breeding ratio is approximately 1.2.

Sodium enters the reactor vessel at its lower end and flows upward through the orificed assemblies and is discharged from the vessel above the core. The fuel assemblies are hydraulically held down; and a mechanical holddown plate above the core serves as a backup.

Three parallel but well separated sets of loops are used to remove the heat generated by the reactor. The heat in the primary sodium system is transferred to non-radioactive sodium in the intermediate heat exchanger. This heat is then transferred from the intermediate sodium system to the feedwater to generate steam to supply the turbine-generator. The steam conditions are 1450 psig and 900°F.

The primary sodium system components are housed in concrete compartments that are steel lined and inerted. A low leakage containment provides the final barrier between the potential sources of radioactivity and the environment.

*This annex is a paper by L. E. Strawbridge, Westinghouse Electric Corporation, Advanced Reactors Division, from the proceedings of the Fast Reactor Safety Meeting, CONF-740401-P1, Beverly Hills, Calif., April 2-4, 1974. The CRBRP design and safety approach are still under development, and are expected to differ in specifics from that presented here.

Table 4.2-14

PRINCIPAL PLANT CHARACTERISTICS

Reactor power, MWt	975
Gross electrical power, MWe	380
Number of primary heat transport loops	3
Leg location for sodium pumps, primary/intermediate	Hot/Cold
Primary plant materials	304 SS
Reactor vessel outlet temperature, °F	995
Total core sodium flow rate, 10 ⁶ lb/hr	41.45
Total intermediate sodium flow rate, 10 ⁶ lb/hr	38.34
Feedwater temperature, °F	452
Steam pressure at turbine throttle, psig	1450
Steam temperature at turbine throttle, °F	900
Total steam flow to turbine, 10 ⁶ lb/hr	3.34
Turbine generator plant gross efficiency, %	39.0

Table 4.2-15
PRINCIPAL REACTOR PARAMETERS

Core Fuel Assemblies	
Core fuel material	PuO ₂ /UO ₂
Fuel cladding and assembly duct material	316 SS
Fuel rod outer diameter, in	0.23
Rod pitch to diameter ratio	1.25
Core height, in	36
Axial blanket height at both ends, in	14
Fuel rods per assembly	217
Number of core assemblies	198
Peak fuel burnup goal, MWd/t	150,000
Maximum linear power, kW/ft	15.5
Average linear power, kW/ft	7
Radial Blanket Assemblies	
Blanket fuel material	Depleted UO ₂
Rod outer diameter, in	0.52
Cladding thickness, mils	15
Fuel rods per assembly	61
Number of radial blanket assemblies	150
Maximum linear power, kW/ft	17.5
Control Rod Assemblies	
Poison material	B ₄ C
Number of control rods	19
Refueling	
Frequency, mo	12
Average number of core assemblies replaced	72
Average number of radial blanket assemblies replaced	30

Plan View of Initial Core Arrangement

Figure 4.2-27

Reactor and Closure Head Schematic
Figure 4.2-28

B.2 Design Guidelines

The CRBRP preliminary design is based on the guidelines specified by Project Management Corporation (PMC) in its March 15, 1972 request for proposals from the three participating reactor manufacturers. These guidelines focus attention on the following objectives for the plant:

- . Reliability, safety, and ready licensing capability - achieved through the use of a proven technology base, and design simplicity and practicality
- . High plant availability and maintainability, with a high degree of in-place component inspectability
- . Minimal environmental effect and impact and good public acceptance, to the extent that these are not already covered under "licensability"
- . Component prototypicality and extrapolability to commercial plants
- . Demonstration of a reasonable breeding capability, not less than a 1.2 breeding ratio
- . Demonstrating the prospect for economic competitiveness of commercial LMFBRs.¹

The over-riding emphasis on the guideline criteria for reliability, safety, and licensing capability in turn dictates that the demonstration plant design proceed along the following path: (1) maximum use of proven experience as derived from design and operating experience of existing and planned fast reactors, with a substantial incorporation of applicable FFTF technology;^{2,3,4} (2) use of the "three levels of design approach" which provides defense in depth through accident prevention, arrest of off-normal conditions and margins to prevent accidental releases to the environment; (3) capability to achieve the desired plant and component design life of 30 years; and (4) operational and design characteristics providing visible and demonstrable margins to assure reliable and safe operation.

B.3 Overall Design Approach to Assure Safe Operation

The overall approach to design recognizes that good design is the controlling factor in attaining the desired high level of reliability and safety. Consequently, good design provides defense in depth through accident prevention, and

through protection of the public and plant personnel for all foreseeable occurrences. In addition, capabilities for providing public protection for extremely unlikely circumstances are provided. This approach is expressed in terms of three levels of design.

The first level of design addresses the reliability of operation and the prevention of accidents through the intrinsic features of the design of the plant and the quality, redundancy, testability, inspectability, and fail-safe features of the components of the reactor and plant. The design must be such that the plant will be safe in all phases of operation and will have a maximum tolerance for errors, off-normal operation, and components malfunction. Analyses will be made and test programs conducted to determine those types of malfunctions or faults that could affect reliability of operation so that they can be guarded against by design, quality assurance, or inherent fail-safe characteristics, as appropriate. As a basic part of the LMFBF development program, a large number of large scale engineering proof tests are being conducted to verify each design concept. The testing process in the first level is to provide predictability of performance and, hence, safety through assurance of the use of proven materials and technology.

Fabrication and construction of the plant will be performed using applicable codes and standards, and under rigorous, documented quality assurance measures. Where adequate codes and standards do not exist, where appropriate, new ones will be developed and applied to the design.

Extensive preoperational test programs will be conducted in the plant to assure conformance of components and systems to the established performance requirements. Key parameters will be monitored continuously or routinely and a well-defined surveillance, in-service inspection and preventive maintenance program will be carried out by a trained operating and maintenance staff to provide assurance that the as-built high quality is retained throughout the life of the plant.

The second level of design provides protection against Anticipated Faults and Unlikely Faults (such events as partial loss of flow, reactivity insertions, failure of parts of the control system, or fuel handling errors) which might occur in spite of the care taken in design, construction and operation of the plant. This additional level of defense for the public and the operating staff is provided by reliable protection devices and systems and decay heat removal systems, designed to assure that such events will be prevented, arrested, or accommodated. The requirements for these protection systems are based on a spectrum of occurrences

which could lead to off-normal operation which the plant design must safely accommodate. Conservative design practices, including redundant detecting and actuating equipment, will be incorporated in the protection systems to assure both the effectiveness and reliability of this second level of defense. These systems will be designed to be routinely monitored and tested to provide full assurance that when they are required to operate, they will do so reliably.

The third level of design supplements the first two levels by providing protection against Extremely Unlikely Faults (such as large pipe ruptures, large sodium fires, large sodium-water reactions, or a loss of cooling to a single fuel assembly). The faults are of exceedingly low probability and no events in this category are expected to occur during the plant lifetime. They represent extreme cases of failures and will be analyzed using nominal calculations and sensitivity studies to establish conservative design bases. Conservative assumptions are made to assess the radiological effects of such accidents on the public.

Embodied in these three levels of design is the philosophy that any accident that could lead to loss of in-place coolable geometry must be prevented. Accordingly, it must be demonstrated that failure to scram and other potential initiators of accidents that could lead or contribute to loss of in-place coolable geometry are sufficiently remote that they do not set design bases for the plant. To support this position, an extensive research and development program is planned to assure the necessary reliabilities.

B.4 Design Bases

To implement the overall design approach described above, design bases are developed for the overall plant and for each system or major component. These design bases consider both natural phenomena (seismic, climatic, tornadic and flood conditions) and operational conditions (normal, off-normal, and potential accidents) and assure that the plant objectives noted in Section B.2 can be met. The following discussion emphasizes the design bases related to the safety and licensing objectives.

B.4.1 Natural Phenomena Design Bases

Conditions associated with extremely unlikely natural phenomena, which bound the most severe that have been historically reported for the site and the surroundings, are used as design bases for the plant.

The CRBRP preliminary design is based on a Safe Shutdown Earthquake (SSE) maximum horizontal ground acceleration of 0.18 g. This is consistent with the design basis

for other TVA nuclear plants in the vicinity and with preliminary evaluations for the Clinch River Site. Additional site investigations are in progress to confirm the seismic characteristics.

The tornado design bases are consistent with those being applied to other nuclear plants east of the Great Divide. The tornado wind force is based on a 360 mph wind, consisting of a combined 300 mph rotational velocity and 60 mph translational velocity, applied over the full height of the structures. The bases also include a tornado associated pressure variation of 3 psi in 3 seconds and typical tornado generated missiles.

The CRBRP is designed so that the Probable Maximum Flood (PMF) does not exceed the plant grade elevation. A conservative calculation of the PMF including the effects of dam failure and wave runup results in an elevation of about 810 feet, while the plant grade will be at approximately 815 feet.

The plant is designed so that it can be shut down safely and maintained in a safe condition in the event of any of these extremely unlikely natural phenomena. For less severe natural phenomena which would have a significant probability of occurring during the plant life (expected frequency approximately once in one hundred years), the plant is designed to remain operable during the event.

B.4.2 Operational Design Bases

The full spectrum of plant conditions that can reasonably be postulated is divided into four categories in accordance with the expected frequency of occurrence of each condition. The four categories of design basis events are:

- a) Normal Operation
- b) Anticipated Faults
- c) Unlikely Faults
- d) Extremely Unlikely Faults

Events and conditions associated with each of these categories are evaluated and the design must accommodate the consequences of them. The consideration of this wide range of events in the design bases provides protection for all conditions that can reasonably be postulated.

The terminology and definitions selected for the CRBRP are adopted from those in RDT C16-1T, "Supplementary Criteria and Requirements for RDT Reactor Plant

Protection Systems." Similarity exists between the four frequency classifications and the component operating conditions defined in "Rules for Construction of Nuclear Plant Components," ASME Code, Section III. The classifications were developed jointly by ANS and ASME committees. The ANS sponsored committee developed the ANSI N18.2, which provides guidance for the reactor designer who must select the plant events considered within the different component operating conditions used in ASME Code application. However, a one-to-one correlation between plant events defined here and the ASME component "operating conditions" is not intended. For example, the fault that would initiate operation of a standby component may be an ASME normally expected design condition for that component. These faults might be in any of the three fault frequency categories described above. For example, a Primary Heat Transport System guard vessel is only required to perform its normal function (containment of primary sodium) following an Unlikely or Extremely Unlikely Fault which results in a sodium leak from the enclosed vessel or pipe into the guard vessel.

B.4.2.1 Normal Operation

Definition

Normal operation includes steady power operation and those departures from steady operation which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant.

Examples

- a. Steady operation
- b. Startup
- c. Normal shutdown
- d. Stand-by
- e. All refueling operations
- f. Load following
- g. Random fuel rod leakage
- h. Operation with specific equipment out of service as permitted by Technical Specifications
- i. Routine inspection, testing and maintenance of components and systems during any of the above operations (within Technical Specifications requirements).

B.4.2.2 Anticipated Faults

Definition

An off-normal condition which individually may be expected to occur one or more times during the plant lifetime.

Examples

- a. Uncontrolled withdrawal of a control rod from a subcritical condition
- b. Uncontrolled withdrawal of a control rod at power
- c. Control rod drop with pump trip
- d. Spurious trip of primary pumps
- e. Turbine trip
- f. Loss of normal feedwater
- g. Loss of off-site power
- h. Small steam generator tube leaks which do not require immediate automatic pressure relief
- i. Spurious trip of intermediate pumps
- j. Control system malfunction
- k. Generator load loss
- l. Single error of an operator
- m. Spurious reactor trip

B.4.2.3 Unlikely Faults

Definition

An off-normal condition which individually is not expected to occur during the plant lifetime; however, when integrated over all events in this category, some may be expected to occur during the plant lifetime.

Examples

- a. Primary or intermediate sodium pump locked rotor
- b. Gaseous waste storage tank rupture
- c. Loss of both on-site diesel A-C power units
- d. Cover gas purification system failure releasing radioactivity
- e. Turbine bypass valves failure to open at turbine trip
- f. Enrichment error resulting in an overenriched fuel rod
- g. Small sodium leak in primary or intermediate system, with leak rate less than normal makeup flow

B.4.2.4 Extremely Unlikely Faults

Definition

An off-normal condition of such extremely low probability that no events in this category are expected to occur during the plant lifetime, but which nevertheless represents extreme or limiting cases of failures which are identified as design bases.

The consideration of Extremely Unlikely Faults in the design bases provides the third level of design. For some events in this category, protection will be provided through action of the plant protection system. In other cases design features in the plant will mitigate the event or accommodate the consequences.

Examples

- a. Large primary, intermediate or steam system pipe rupture
- b. Sodium fire above the operating floor
- c. Large sodium spills inside or outside containment
- d. Loss of cooling to a single fuel assembly
- e. Large sodium-water reactions

It should be noted that plant conditions may need to be reclassified as the design is developed. Further detailed failure evaluations or design changes may modify the estimated frequency of occurrence.

B.4.3 Damage Severity Limits

Damage severity limits are used in combination with the frequency classification to establish design bases for the plant design. Each damage severity limit defines (1) fuel damage; (2) plant equipment damage; and (3) a radioactivity release limit. The following incident definitions provide bases for evaluations of damage severity limits.

No Damage

No damage is defined as (1) no reduction of effective fuel lifetime, or fuel power capability, below the design value; (2) accommodation of conditions within the fuel and plant operating margins without requiring automatic or manual protective action; and (3) no planned release of radioactivity.

Operational Incident

An operational incident is defined as an occurrence which results in (1) no reduction of effective fuel lifetime, or fuel power capability, below the design value; (2) accommodation with, at most, a reactor trip that assures the plant will be capable of returning to operation after corrective action to clear cause of the trip; and/or (3) plant radioactivity releases that may approach the 10 CFR 20 guidelines.

Minor Incident

A minor incident is defined as an occurrence which results in (1) a general reduction in the fuel burnup capability and, at most, a small fraction of fuel rod cladding failures; (2) sufficient plant or fuel rod damage that could preclude resumption of operation for a considerable time and/or (3) plant radioactivity releases that may exceed 10 CFR 20 guidelines, but do not result in interruption or restriction of public use of areas beyond the exclusion boundary.

Major Incident

A major incident is defined as an occurrence which results in (1) substantial fuel cladding failures or distortion in individual fuel rods, but the configuration remains coolable; (2) plant damage that may preclude resumption of plant operation, but no loss of safety function necessary to cope with the occurrence; and/or (3) radioactivity releases that may exceed the 10 CFR 20 guidelines but are within the 10 CFR 100 guidelines.

B.4.4 Application of the Frequency and Consequence Classifications to the Plant Design and Evaluation

The consequences of plant conditions of a given frequency are restricted by damage severity limits established as design bases. The plant design is required to control the consequences of these plant conditions within the damage severity limits indicated in Table 4.2-16.

Table 4.2-16
DESIGN REQUIREMENTS

<u>Plant Conditions</u>	<u>Damage Limits</u>
Normal Operation	No Damage
Anticipated Fault	Operational Incident
Unlikely Fault	Minor Incident
Extremely Unlikely Fault	Major Incident

Specific Design Requirements for the Reactor Shutdown Systems

Two independent reactor shutdown systems will be provided. The design requirements for those systems are indicated in Table 4.2-17. The failure of either system is highly unlikely. Failure of one of these systems coincident with an Anticipated Fault is considered extremely unlikely. Nevertheless, the design basis as given in Table 4.2-17 conservatively restricts the consequences of such a failure coincident with an Anticipated or with an Unlikely Fault to within the limits of a Minor or Major Incident, respectively. The probability of an Extremely Unlikely Fault coincident with a failure of one of the reactor shutdown systems is exceedingly low and, therefore, not appropriate as a design basis.

In addition, the plant design must also meet the requirement that, for any initiating plant condition (except the Extremely Unlikely Faults) coincident with any single active component* failure, the resulting plant damage will be within the next higher damage limit.

B.5 Safety Related Design Provisions

The preliminary design that has resulted from the application of the Design Guidelines and the more detailed design bases provides defense in depth. The principal safety related design features are described below.

B.5.1 Reactor Features

Reactivity Characteristics

The reactor design provides reactivity coefficients that assure a large stability margin and facilitate the control of the reactor.

The Doppler reactivity coefficient, which is negative in all core regions, is the most important feedback in assuring stability. The core average Doppler constant is -0.006 T dk/dT . For a totally voided core (a limiting case for certain accident analyses), the average value is approximately -0.004 T dk/dT .

The overall sodium temperature coefficient including expansion of sodium and structures is negative and small. The coefficient is position dependent, being positive in the central region and negative in the peripheral regions. Sodium voiding can produce a positive reactivity feedback, depending on the assembly

*An active component is one in which mechanical movement must be initiated or electrical power must be provided to accomplish a function of the component. An active component failure means failure of an active component to operate or stop as intended on demand or a change when no change is intended.

Table 4.2-17

SPECIFIC DESIGN REQUIREMENTS FOR THE REACTOR SHUTDOWN SYSTEMS

Plant Condition	Damage Severity Limits			
	Primary System		Secondary System ^a	
	Without Stuck Rod	With Stuck Rod	Without Stuck Rod	With Stuck Rod
Normal Operation	No Damage	No Damage	No Damage	Operational Incident
Anticipated Faults	Operational Incident	Operational Incident	Minor Incident	Minor Incident
Unlikely Faults	Minor Incident	Minor Incident	Major Incident	Major Incident
Extremely Unlikely Faults	Major Incident	Major Incident	None ^b	None ^b

^aAssumes primary system has failed to operate.

^bCombined probability of extremely unlikely fault and complete failure of primary control rod system is exceedingly low and not suitable as a design basis.

location, but the effect is less than 10¢ for any assembly. Voiding the core and blankets (but not the control channels) would result in a reactivity increase of only about 60¢. The additional voiding of the control channels would result in an overall negative reactivity.

Bowing of the fuel assemblies is controlled by the core restraint system design so that a negative power coefficient results at all power levels, under transient and steady-state operation. The fuel and structure reactivity expansion coefficients are small. The core is radially restrained from motion, resulting in a small radial coefficient during a full power transient. The fuel axial expansion reactivity coefficient is also small and negative, but because of uncertainties in fuel-cladding interface forces, no credit for the effect is assumed in the safety analyses. The cladding axial expansion reactivity coefficient is also negative, but is not as prompt as the fuel component alone.

The temperature coefficients combine to give a prompt negative power coefficient for all conditions and a large stability margin exists in the plant.

Fuel Assembly Features

Features are included in the design of the reactor core assemblies to prevent total flow blockage or mislocation of assemblies into positions which could cause overheating or unacceptable reactivity effects.

Each assembly contains multiple openings in the inlet nozzle, so that no foreign object could conceivably result in total flow blockage. Blockage of an opening would not significantly affect the flow rate or the temperatures.

The lower nozzles also provide a mechanical key system to prevent placement of assemblies into adverse unplanned core locations. Typical positioning errors which are precluded include the location of a highly enriched assembly in a higher power zone and the location of a fuel assembly in a control rod location. The keying arrangement guarantees the location of assemblies in positions of acceptable cooling and reactivity.

Holddown Features

The fuel assembly and core support structure designs provide hydraulic holddown during all conditions of flowing sodium. The design provides a low pressure plenum in the core support structure, which is in communication with the core

outlet pressure. This low pressure sodium below the fuel assembly largely offsets the normal upward force associated with the pressure drop of the flowing sodium. The assembly is then held down by the net force, which is essentially that due to gravity.

A mechanical holddown grid is included in the upper reactor internals. This grid serves as a backup to the hydraulic holddown and would limit the upward motion of any assembly to less than two inches in the event of failure of the hydraulic holddown. The reactivity effect of such limited motion for one or several assemblies is very small.

B.5.2 Heat Transport System Features

Provisions for Reliable Cooling

The multiple sodium system loops provide cooling during normal operation, shutdown, and emergency conditions. A malfunction in one loop must not prevent the other loops from functioning. This is accomplished by the following system design features: The primary system operates at a low pressure (less than 250 psia at any point) with the pressurization supplied by the main circulating pumps. Tripping these pumps permits the system to be rapidly depressurized as the flow coasts down to the levels provided by the pump pony motors. Since the pumps are tripped when the reactor is tripped to minimize thermal shock to the components, a relaxation of system dynamic pressure occurs as a standard operating procedure.

Because a loss-of-coolant accident in a sodium system is essentially a draindown or pump-out condition, elevation and/or containment of the system by guard tanks can be employed, in combination with tripping the pumps, to limit the loss of coolant inventory (See Figure 4.2-29). By restricting the loss of sodium inventory such that the submergence of the reactor vessel main coolant nozzles is maintained, the other unaffected primary loops and associated primary sodium system components remain operable as redundant emergency, and long-term decay heat removal systems. The flow provided by the coastdown of the pumps, following the reactor and pump trips, cools the core in the period immediately following a system rupture.

Since the reactor vessel outlet nozzles must remain covered after an accident, the elevation or containment of the remainder of the system is considered in relationship to those nozzles. As shown in Figure 4.2-29, all major components and directly attached piping are contained in guard vessels which provide double containment to an elevation above the reactor vessel outlet nozzles so that the

4.2-223

Primary Sodium Loop Schematic

Figure 4.2-29

pumps, when operating at pony motor speed, cannot pump sodium above the guard vessels. The primary sodium piping, except for the connections to the components within the guard vessels, is installed above this minimum elevation without additional containment. The guard vessels are sized to prevent the sodium level in the vessel from dropping below the outlet nozzle elevation.

In the event of a pipe leak, the remaining intact primary sodium loops and their associated heat transport systems are available to provide a reliable method of long-term core cooling. Even without electrical power to drive the primary and intermediate sodium system pumps, the system is capable of removing decay heat by natural circulation as long as water is supplied to the evaporators.

Prevention of Gas Entrainment

The primary sodium system is designed to avoid gas entrainment since large coherent bubbles could result in undesirable heat transfer and reactivity effects. The design includes a number of provisions to assure that large gas bubbles will not enter the core:

1. A vortex suppressor plate in the upper vessel plenum provides a quiescent pool and inhibits gas entrainment. The vessel outlet piping and the pump impeller are sufficiently submerged to avoid entrainment.
2. The pump lubricant is isolated from hot sodium by a multiplicity of seals, gas barriers, and leak sumps.
3. No part of the primary sodium system is below atmospheric pressure during normal operation, thereby avoiding in-leakage of external gases.
4. Gas accumulation points are limited to locations which are vented.

These design measures should prevent gas bubbles in the primary system. However, even if a bubble entered the primary system, turbulence in the inlet plenum and the design of the core support structure and fuel assembly inlet nozzles will break up the bubble before it enters the core (as has been shown in tests for FFTF). Consequently, coherent gas bubbles cannot enter the reactor core.

B.5.3 Plant Protection System

Reactor Shutdown System

A highly reliable reactor shutdown system is an essential feature for the plant. A design requirement for the shutdown system is that the reliability is sufficiently high so that successful shutdown is assured in the event of off-normal conditions. Based on guidance from the Regulatory assessment of Anticipated Transients Without Scram in water reactors,⁵ the need to assume scram failure in accident analyses is negated if the failure rate is less than about 10^{-7} per reactor year. The design of the shutdown system and the associated development programs are aimed at achieving the required high reliability.

The design includes two independent fast acting shutdown systems, each capable of terminating anticipated transients without action of the other system. The systems are actuated by functionally diverse signals wherever possible. Equipment diversity is being included in the design to minimize the possibility of degradation of the shutdown system due to postulated common mode failures. Both shutdown systems use mechanical poison rods inserted from above the reactor. Differences in the absorber assembly design, in the control rod drive mechanism design and in the disconnect and insertion features are planned. Extensive testing and evaluation programs are planned to assure reliable performance of all components required for shutdown.

Although no driving force which could eject a control rod has been identified, the drive mechanisms contain an anti-ejection device that positively prevents any rapid upward motion of the rod which would result in more than a few cents of reactivity.

Control rod withdrawal speeds are limited so that any malfunction resulting in inadvertent withdrawal is readily accommodated by the protection system.

Sodium-Water Reaction Protection

The protection system includes provisions to limit the consequences of potential sodium-water reactions that could occur as a result of a failure in the steam generator. The protection system detects leaks through hydrogen or oxygen detection or by an overpressure in the intermediate sodium system. Upon detection of a large sodium-water reaction, the protection system vents the sodium system of the affected loop to a sodium-water reaction products tank and shuts off the affected loop to a sodium-water reaction products tank and shuts off and

drains the water to minimize the extent of the reaction. The plant is designed so that any pressure loadings resulting from sodium-water reactions will not fail the intermediate heat exchanger (so that no radioactive sodium is involved) and will not result in failures in any of the other loops (so that heat removal capability is retained).

Other Plant Protection System Functions

The plant protection system also provides for containment isolation in the event of excessive radiation levels in the plant, startup of emergency diesel generators if offsite power is lost, and actuation of any devices required to remove decay heat upon reactor shutdown.

B.5.4 Power Supply Features

In addition to ties with offsite electrical transmission systems, an emergency power system is provided to supply important auxiliaries including engineered safety equipment in the event of loss of the preferred power supplies. The emergency power system supplies the loads necessary to shut down the reactor, remove residual heat for extended periods and indicate the status of plant parameters. Two diesel generators, each capable of carrying the emergency load, are automatically started upon loss of offsite power. The diesel generators are hardened and are designed to be up to speed and capable of accepting load in ten seconds. Batteries are provided to maintain continuity of supply to vital instruments. In the event of a delayed startup of the diesels, the batteries can supply vital loads. This requirement is minimized by the natural circulation capability provided in the design. A system of inverters brings these batteries into operation automatically after failure of incoming electrical and diesel supplies.

The redundancy of supplies, transformer, and buses and the division of critical loads among the buses results in a system of high reliability. The physical separation of buses, switchgear, and components is intended to limit or localize the consequences of electrical faults or mechanical failures occurring anywhere in the system.

B.5.5 Containment Features

The fuel rod cladding and the primary sodium system boundary provide the first two barriers to prevent the escape of radioactive fuel and fission products to the environment. The plant design provides additional barriers. These include the

compartments which house the reactor vessel and the other primary sodium system components. These compartments are heavy concrete structures lined with steel plate. An inert atmosphere (nitrogen) is used in those compartments of the containment where there are large volumes of sodium such as the primary sodium system. The inert atmosphere prevents sodium fires and minimizes the consequences of any sodium spills or leaks.

The outer barrier is a low leakage containment designed for a pressure of 10 psig and a leak rate of 0.1%/day at the design pressure. This pressure capability provides a large margin since no accidents are calculated to pressurize the containment to more than a few psig. The containment is designed to withstand the extremely unlikely natural phenomena including tornado generated missiles.

Plutonium handling and storage will be limited to specifically designated and controlled areas within the containment. Strict security measures and accountability procedures will be enforced to safeguard all plutonium at the plant.

B.5.6 Capabilities Beyond Design Level Three

The design features described above provide capability to accommodate the four categories of plant conditions identified in Section B.4; namely Normal Operation, Anticipated Faults, Unlikely Faults, and Extremely Unlikely Faults. Many of these same design features provide capability to accommodate even more severe occurrences than are considered in the four categories of plant conditions. For example, the reactor containment capability is not challenged by any of these events and thus provides a large margin for unforeseen events.

Since there is some chance that the project may not be able to demonstrate adequate scram reliability and sufficiently low probability of other potential initiators of accidents involving loss of in-place coolable geometry in time to impact the early design decisions, margins over and above levels 2 and 3 will be provided in the following manner:

- a. Head holddown bolts
- b. Primary system features as required to accommodate substantial dynamic loadings
- c. Reactor core internals designed to enhance post accident cooling capability and reduce the potential for secondary criticality

d. A low leakage containment building

The detailed analyses of the consequences of low probability accidents leading to loss of in-place coolable geometry (such as anticipated transients with failure to scram) will be continued in order to gain a complete understanding of their consequences.

The project will maintain contact with and awareness of national research and development programs undertaken with the objective of developing design features capable of mitigating the consequences of accidents leading to loss of in-place coolable geometry.

In order to be in a position to take advantage of any practical and definitive results of these R and D programs, should the need arise, the following flexibilities are maintained in the design:

- a. Space below the reactor cavity
- b. Head access area which could be sealed.

The consideration of capabilities beyond design level three must not overshadow the areas in which the greatest safety assurance can be developed; i.e., in design level one, which provides reliable operation to prevent accidents; and in design levels two and three which assure that all foreseeable accidents can be accommodated safely.

REFERENCES FOR ANNEX B

1. "Liquid-Metal Fast Breeder Reactor (LMFBR) Program Plan," USAEC Reports WASH-1101 through WASH-1110, Argonne National Laboratory, August 1968.
2. "Fast Flux Test Facility, Preliminary Safety Analysis Report," Volumes 1 and 2, September 1970.
3. "Fast Flux Test Facility Design Safety Assessment," USAEC Report HEDL-TME-72-92, Hanford Engineering Development Laboratory, July 1972.
4. Safety-Assessment Philosophy of the Fast Flux Test Facility, Nuclear Safety, 14(2):79 (March-April 1973).
5. "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," USAEC Report WASH-1270, prepared by the Regulatory Staff, September 1973.

4.2.7 ANNEX C: THE FUEL MELTING INCIDENT IN THE
ENRICO FERMI ATOMIC POWER PLANT ON OCTOBER 5, 1966

The Enrico Fermi reactor, located at Newport, Michigan, is a liquid metal cooled fast reactor rated at 200 Mwt (65 MWe) and operated by Power Reactor Development Company (PRDC).

On October 5, 1966, during a controlled, slow increase in power, a fuel melting incident occurred. The reactor was shut down in an orderly manner from a power level of 31 Mwt after several subassemblies were observed to have abnormally high outlet temperatures and after radiation alarms sounded due to the leakage of a small amount of fission products into the reactor building. Monitoring the on-site radiation levels showed that the incident had not caused any public safety hazard.¹ To quote from a paper presented on the incident:²

During operation of the Fermi reactor at a power level of 34 Mwt on Wednesday, October 5, 1966, the radioactivity level of the argon gas which blankets the primary system sodium was observed to rise substantially due to the presence of gaseous fission products. Some of this radioactivity leaked into the reactor building due to normal leakage from the gas system to the building. Following the sealing of the building ventilation system, the reactor power was reduced and ultimately the reactor was shut down manually. At the time of building isolation due to high radiation levels, an observation was made, and later confirmed in detail, that an abnormal amount of reactivity (approximately 22 cents) had been lost and that two outlet core thermocouples were reading abnormally high.

Seven subassemblies were removed from the reactor after the incident and subjected to destructive examination; melting had occurred in two. Determinations of the weight and change in the center of gravity showed that the change in weight was less than the errors associated with the measurements and that the center of gravity had moved downward by about 3/4 and 1-7/8 in. for the two subassemblies. In addition, two other subassemblies had suffered some overheating.

During inspection of the inlet plenum with remote viewing devices, two objects were found loose in the inlet plenum and were identified as two of six zirconium segments originally installed on the conical flow guide. These two segments were removed from the reactor for inspection and analysis.

From the analyses and inspections made, the fuel melting is believed to have been caused by partial blocking of the inlet nozzles of four adjacent subassemblies in the southeast quadrant of the core by one of these detached segments. This blockage caused the fuel in two of those subassemblies to melt and the fuel in the other two to become overheated. Thermal-hydraulic calculations showed that the flow must have been reduced to about 3% of normal in the two subassemblies in which melting had occurred. Modeling experiments using water in a geometric mock-up of the plenum indicated that either zirconium segment could have reduced the flow rate in two adjacent subassemblies to about 3%. Also, the other detached segment is suspected to have partially blocked, to a lesser degree, the inlet nozzles of one or more subassemblies in the northeast quadrant of the core prior to and on October 5, 1966, causing the observed anomalous thermocouple readings.

Much has been made of this accident, and at least one author has noted that the accident exceeded the "maximum credible accident" described in the hazard report for the Enrico Fermi reactor and therefore represented an accident greater than anything the designers had anticipated. In his book, The Careless Atom, Sheldon Novick stated,

Whether a disastrous accident like that described in the 1957 report [referring to WASH-740] could have happened remains a matter for speculation. Despite the confidence of its designers, the Fermi reactor is surrounded with a steel and concrete containment which is designed to withstand the force of an explosion of 600 pounds of TNT. The question of whether a core meltdown could produce an explosion greater than this and whether the containment would in fact completely contain such an explosion, is the subject of a great deal of detailed calculation and educated guesswork. Since an experimental test of these calculations would require the destruction of at least one extremely expensive reactor, they are likely to remain unconfirmed unless and until a still more 'incredible' accident happens.

To consider the actual accident in the context of the maximum credible accident as envisaged by the designers, the remarks in the Hazards Summary Report³ describing the maximum credible accident must be considered:

The maximum credible accident in the Fermi Reactor is the melting of some or all of the fuel in one core subassembly due to either complete or partial plugging of the nozzle of that subassembly, or to a flow restriction within the subassembly due to radiation damage which resulted because the subassembly in question was left in the reactor too long as the result of an administrative error. The results of this maximum credible accident are as follows:

- a) The reactor would probably be shut down automatically as a result of the reactivity loss due to the melting of the fuel and the subsequent escape of that fuel from the core. If the reactivity change were not

great enough to cause shutdown, then only a relatively small amount of the fuel could have melted, and the fact that more did not melt would indicate that the thermal stability had been restored and that the remaining fuel was adequately cooled even though failure had occurred. Operating instructions will require shutdown on any unscheduled change in reactivity.

- b) A large fraction of the fission products initially contained in the fuel would have been released to the primary coolant system and it is likely that some of the noble gases would eventually reach the inert gas system. Since these systems are normally radioactive and are sealed, there would be no additional outward effect caused by the additional fission products from melted fuel.
- c) No in-plant or public hazard would result in the maximum credible accident.
- d) A cleanup problem would be presented to the operators which would vary with the amount of and location of the melting but which would undoubtedly entail the removal of much of the contents of the reactor vessel in order to ascertain the extent of this dispersion of the melted fuel and to allow removal of the melted fuel. The interior components of the reactor vessel have been designed to be removable after operation and this design provision would allow extensive renewal of components if this action were required.

In the actual accident, all of the fuel in one channel was not melted but part of the fuel in two channels was melted; in this respect the accident varied from the definition of the maximum credible accident given by the Fermi designers. However, the consequences of that partial melting were very similar to the consequences calculated by the designers. The reactivity change was found to be 22¢ negative as compared with the possibility of a 30¢ negative reactivity change calculated by the designers. In this respect, the accident was less severe than the designers' analysis. Since the reactivity change was less than analyzed and was inserted very slowly, the reactor did not scram on this signal. The operator did note that some reactivity had been lost and initiated manual shutdown.

The maximum credible accident and the incident that occurred in Fermi bear no relation to hypothetical accidents in which large quantities of energy are released, such as the 600 lb of TNT referred to by Novick. In the actual accident case, there was no energy release beyond that normally produced in the reactor. In fact, the melting of the fuel caused the reactivity to decrease, and this decrease led to a reduction in the energy produced by the reactor.

As described in Section 4.2.7.5, recent designs contain a number of provisions (beyond the use of oxide rather than the metallic fuel of Fermi) to preclude the possibility of a Fermi-type accident. The Fermi design itself had capabilities

for safely accommodating far more severe accidents. The Fermi structure was designed to contain the consequences of an accident involving widespread core damage.

Following their detailed investigations, the operators reported the following:¹

PRDC believes that the October 5 incident demonstrated the basic adequacy of the Fermi plant operating procedures and safety systems. Despite the occurrence of 97 per cent flow blockage and subsequent fuel melting in two adjacent subassemblies, the malfunction was detected and the reactor was shut down in an orderly manner without any hazards to the public health and safety.

REFERENCES FOR ANNEX C

1. Atomic Power Development Associates, Inc., "Report on the Fuel Melting Incident in the Enrico Fermi Atomic Power Plant on October 5, 1966," Report APDA-233, December, 1968.
2. J. G. Duffy et al., "Investigation of the Fuel Melting Incident at the Enrico Fermi Atomic Power Plant," Report ANS-101, April, 1967.
3. Enrico Fermi Power Plant, Power Reactor Development Company, "Technical Information and Hazards Summary Report," Section VI. Evaluation of Hazards, vol. 7, June, 1961.

4.2.7 ANNEX D: A REVIEW OF PANEL DISCUSSION--SESSION 16*

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Discussion following the presentations both indicated some agreements and emphasized a number of controversial areas in which decided disagreements are evident. The fundamental question reviewed by the panelists is what should be the design basis of fast reactors to assure the safety of the public. It is axiomatic that the risk to the public should be low, and all panelists agreed that an acceptable low risk is presently attainable. However, the level of risk that is acceptable and the most effective approach to limit the risk is not at all clear.

There was a general consensus that all design approaches should assure positive termination of any nuclear chain reaction. This should be achieved by use of independent and diverse shutdown systems, and should include both diverse and independent trip signals, information channels, and neutron absorbers.

It was also agreed that provision for containment should be included in the design basis of fast reactors. However, there was no clear agreement on the magnitude of the energy release which the design should contain, or whether the containment should be based on a specific energy release at all.

According to one approach, some engineered safety features which are currently incorporated to protect against hypothetical accidents, in fact, compromise plant safety. The emphasis should be on reliable operation and control, rather than on "hypothetical safety." Under this approach, the fast reactor containment design would incorporate standard margins. Design safety would then be verified by establishing the reliability of systems and components.

An alternative approach was based on the position that this level of reliability cannot be assured, and therefore, the plant must be designed for specified hypothetical accidents. Under this view, if reactors were designed to contain hypothetical accidents, it would be easier to establish low risk levels for the public, since the level of reliability required for plant shutdown and other systems need not be as great.

*This annex is from the proceedings of the Fast Reactor Safety Meeting, CONF-740401-P1, Beverly Hills, Calif., April 2-4, 1974.

A third approach recommends the study of hypothetical accidents, but with the analysis aimed at understanding the phenomena in the accidents, and using this understanding to form judgments and provide guidance for developing containment designs.

The panelists appeared to agree on the need to include provisions for containment of explosive energy releases and molten fuel. They also agreed that the major emphasis should be on reliability of systems and instrumentation, rather than on engineered safety features. They seemed to disagree on what standards should be applied in deciding the level of containment and the provisions necessary to assure molten fuel cooling following some hypothetical accident. They also disagreed on the emphasis that should be placed on hypothetical accidents. As one panelist pointed out, we are not free to perform unlimited analysis and testing in all areas, because we have limited resources to work with. Therefore, it is necessary to establish priorities in deciding how to allocate the resources.

This divergence in the approaches to the safety design basis for fast reactors affects the question of what is an acceptable risk, how should it be decided, and who should make the decision. While these questions are not all directly relevant to the safety design basis, they are an integral part of the area of public acceptance of nuclear reactors, and hence are a natural outgrowth of any discussion of safety designs and evaluations. Again, no clear consensus answers developed in the discussion. The nearest approach to consensus was that a major accident probability of 10^{-7} per year would be an acceptable one for fast power reactors, and it was generally believed that an accident rate at least this small would be attainable for commercial fast reactors. In the discussion, some took the view that public risk is not going to be defined quantitatively, in the mathematical sense; rather, the risk will evolve as a result of the comparison of the risk of obtaining electrical power from nuclear sources with the risks from other sources of power (e.g., fossil fuels). In particular, this comparison must include an overall "social risk," which involves not only cost of power and hazards from the various fuels, but also the costs of not having power, or the deleterious effect of prohibitively expensive power.

The question of how the decision should be made and who should make it presented a significant controversy which ranges from the approach that, since this is a public risk, it should involve public debate and perhaps be decided by public vote to the approach that the issue is a highly technical one and hence should be decided by those sufficiently technically informed to make the judgments. There

was universal agreement that the bases for all such decisions should be made completely available to the public.

In summary, the discussion appeared to reflect complete confidence that fast power reactors can be designed and built with satisfactory safety and acceptable risk to the public; however, the definition of acceptable risk has not yet evolved, in a generally agreed quantitative form, so that the criteria for evaluation of acceptable risk have also not yet evolved in an agreed format. Though there is a general consensus on some significant aspects of the approach to the safety design of the fast power reactors, the extent of the accident analysis and the applications of the results of the analysis to specific designs leave considerable license to the individual design groups.

It is becoming clear that diverse views exist on the approach which should be followed in establishing design basis for the safety of fast power reactors. The conference provided a forum for airing these views and attempting to understand the fundamental issues which are at stake. The discussions provided the opportunity to air views, define philosophies, and begin to crystalize the issues which must ultimately be resolved.

4.2.7 ANNEX E: REACTOR SAFETY STUDY, AN
ASSESSMENT OF ACCIDENT RISK IN U.S. COMMERCIAL
NUCLEAR POWER PLANTS, SUMMARY REPORT*

E.1. Introduction and Results

The Reactor Safety Study was sponsored by the U. S. Atomic Energy Commission to estimate the public risks that could be involved in potential accidents in commercial nuclear power plants of the type now in use. It was performed under the independent direction of Professor Normal C. Rasmussen of the Massachusetts Institute of Technology. The risks had to be estimated, rather than measured, because although there are about 50 such plants now operating, there have been no nuclear accidents to date. The methods used to develop these estimates are based on those developed by the Department of Defense and the National Aeronautics and Space Administration in the last 10 years.

The objective of the study was to make a realistic estimate of these risks and to compare them with non-nuclear risks to which our society and its individuals are already exposed. This information will be of help in determining the future use of nuclear power as a source of electricity.

The basic conclusion of this study is that the risks to the public from potential accidents in nuclear power plants are very small. This is based on the following considerations:

- a) The consequences of potential reactor accidents are no larger, and in many cases, are much smaller than those of non-nuclear accidents. These consequences are smaller than people have been led to believe by previous studies which deliberately maximized risk estimates.
- b) The likelihood of reactor accidents is much smaller than many non-nuclear accidents having similar consequences. All non-nuclear accidents examined in this study, including fires, explosions, toxic chemical releases, dam failures, airplane crashes, earthquakes, hurricanes and tornadoes, are much more likely to occur and can have consequences comparable to or larger than nuclear accidents.

Figures 4.2-30 to 4.2-32 compare the nuclear reactor accident risks for the 100 plants expected to be operated by about 1980 with risks from other man-made and

*Because of the preliminary nature of this study, this report (WASH-1400, August 1974), reprinted here for the convenience of the reader, was issued in draft form.

natural phenomena. These figures indicate the following:

- a) Figures 4.2-30 and 4.2-31 show the likelihood and number of fatalities from both nuclear and a variety of non-nuclear accidents. These figures indicate that non-nuclear events are about 10,000 times more likely to produce large accidents than nuclear plants.
- b) Figure 4.2-32 shows the likelihood and dollar value of property damage associated with nuclear and non-nuclear accidents. Nuclear plants are about 100 to 1000 times less likely to cause comparable large dollar value accidents than other sources. Property damage is associated with three effects, 1) the cost of temporarily moving people away from contaminated areas, 2) the denial of use of real property during the few weeks to a few months during which the radioactivity is cleaned up, and 3) the cost of assuring that people are not exposed to potential sources of radioactivity in food and water supplies. This latter cost reflects the efforts required to survey agricultural products, plus the loss of products which might be contaminated.

In addition to the overall risk information in Figures 4.2-30 to 4.2-32, it is useful to consider the risk to individuals of being fatally injured by various types of accidents. The bulk of the information shown in Table 4.2-18 is taken from the 1973 U.S. Statistical Abstract and applies to the year 1969, the latest year for which this data has been tabulated. The nuclear risks are very small compared to other possible causes of fatal injuries.

Table 4.2-18
RISK OF FATALITY BY VARIOUS CAUSES

Accident Type	Total Number	Individual Chance Per Year
Motor Vehicle	55,791	1 in 4,000
Falls	17,827	1 in 10,000
Fires and Hot Substances	7,451	1 in 25,000
Drowning	6,181	1 in 30,000
Firearms	2,309	1 in 100,000
Air Travel	1,778	1 in 100,000
Falling Objects	1,271	1 in 160,000
Electrocution	1,148	1 in 160,000
Lightning	160	1 in 2,000,000
Tornadoes	91	1 in 2,500,000
Hurricanes	93	1 in 2,500,000
All Accidents	111,992	1 in 1,600
Nuclear Reactor Accidents (100 plants)	0	1 in 300,000,000

*An example of the numerical meaning of Figures 11.2-30 to 11.2-32 can be seen by selecting a vertical consequence line and reading the likelihood that various types of accidents would cause that consequence. For instance, in Figure 11.2-30, 100 plants would cause this consequence with a likelihood of one in 10,000 per year. Chlorine releases are about 100 times more likely, or about one in 100; fires are about 1,000 times more likely, or about one in 10 per year; air crashes are about 5,000 times more likely, or about one per 2 years.

FREQUENCY OF FATALITIES DUE TO MAN-CAUSED EVENTS*

Figure 11.2-30

FREQUENCY OF FATALITIES DUE TO NATURAL EVENTS

Figure 11.2-31

FREQUENCY OF PROPERTY DAMAGES DUE TO
NATURAL AND MAN-CAUSED EVENTS

Figure 11.2-32

In addition to fatalities and property damage, a number of other health effects can be caused by nuclear accidents. These include injuries and long-term health effects such as cancers, genetic effects, and thyroid gland illness. The injuries expected in potential accidents would be about twice as large as the fatalities shown in Figures 4.2-30 and 4.2-31; however, such injuries would be insignificant compared to the 8 million injuries caused annually by other accidents.

The number of cases of genetic effects and long-term cancers are predicted to be much smaller than the normal incidence rate of these diseases. Even for a large, very unlikely accident, the small increases in these diseases would not be detected.

Thyroid illnesses that might result from a large accident are the formation of nodules on the thyroid gland that can be treated by medical procedures and rarely lead to serious consequences. For most accidents, the number of nodules caused would be small compared to their normal incidence rate. The number that might be produced in very unlikely accidents would be comparable to their normal rate of occurrence. These would be observed during a period of 10 to 20 years following the accident and would be about equal to their normal incidence in the people exposed.

While the study has presented the estimated risks from nuclear power plant accidents and compared them with other risks that exist in our society, it has made no judgment on the acceptability of nuclear risks. Although the study believes nuclear accident risks are very small, the judgment as to what level of risk society should accept is a broader one than can be made here.

E.2 Questions and answers about the study

This section of the summary presents more information about the details of the study than was covered in the introduction. It is presented in question and answer format for ease of reference.

2.1 Who did this study and how much effort was involved?

The study was done principally at the Atomic Energy Commission headquarters by a group of scientists and engineers who had the skills needed to carry out the study's tasks. They came from a variety of organizations including the AEC, the national laboratories, private laboratories, and universities. About 10 people were AEC employees. The Director of the study was Professor Norman C. Rasmussen of the Department of Nuclear Engineering of the

Massachusetts Institute of Technology, who served as an AEC consultant during the course of the study. The Staff Director who had day-to-day responsibility for the project was Mr. Saul Levine of the AEC. The study was started in the summer of 1972 and took two years to complete. A total of 60 people, various consultants, 50 man-years of effort, and three million dollars were involved.

2.2 What kind of nuclear power plants are covered by the study?

The study considered large power reactors of the pressurized water and boiling water type being used in the U.S. today. This present generation of reactors are all water cooled and therefore, the study limited itself to this type. Although high temperature gas cooled and liquid metal fast breeder reactor designs are now under development, no large reactors of this type are expected to operate in this decade; thus they were not considered.

Nuclear power plants produce electricity by the fissioning (or splitting) of uranium atoms. The nuclear reactor fuel in which the uranium atoms fission is in a large steel vessel. The reactor fuel consists of about 100 tons of uranium. The uranium is inside metal rods about 1/2 inch in diameter and about 12 feet long. These rods are formed into fuel bundles of about 50-200 rods each. Each reactor contains several hundred bundles. The vessel is filled with water which is needed both to cool the fuel and to maintain the fission chain reaction.

The heat released in the uranium by the fission process heats the water and forms steam; the steam turns a turbine to generate electricity. Similarly, coal and oil plants generate electricity using fossil fuel to boil water.

Today's nuclear power plants are very large. A typical plant has an electrical capacity of 1,000,000 kilowatts, of 1,000 megawatts. This is enough electricity for a city of about five hundred thousand people.

2.3 Can a nuclear power plant explode like an atom bomb?

No. It is impossible for nuclear power plants to explode like a nuclear weapon. The laws of physics do not permit this because the fuel contains only a small fraction (3-5%) of the special type of uranium (called uranium-235) that is used in weapons.

2.4 How is risk defined?

The idea of risk involves both the likelihood and consequences of an event. Thus, to estimate the risk involved in driving an automobile, one would need to know the likelihood of an accident in which, for example, an individual could be 1) injured or 2) killed. Thus, there are two different consequences, injury or fatality, each with its own likelihood. For injury, an individual's chance per year is one in 130 and for fatality, it is one in 4000. This type of data concerns the risk to individuals and can affect attitudes and habits that individuals have toward driving.

However, from an overall societal viewpoint, different types of data are of interest. Here, 1.5 million injuries per year and 55,000 fatalities per year due to automobile accidents represent the kind of information that might be of use in making decisions on highway and automobile safety.

The same type of logic applies to reactors. From the viewpoint of a person living in the general vicinity of a reactor, the likelihood of being killed in any one year in a reactor accident is one chance in 300,000,000 and the likelihood of being injured in any one year in a reactor accident is one chance in 150,000,000.

From a broader societal viewpoint, one individual of the 15 million people living in the vicinity of 100 reactors might be killed and 2 individuals might be injured every 25 years. This type of information might be of some use to the Congress or other decision makers in thinking about the overall risk to society from reactor accidents.

2.5 What causes the risks associated with nuclear power plant accidents?

The risks from nuclear power plants are due to the radioactivity formed by the fission process. In normal operation nuclear power plants release only minute amounts of this radioactivity under controlled conditions. In the event of highly unlikely accidents, larger amounts of radioactivity could be released that could cause significant risks.

The fragments of the uranium atom that remain after its fissions are radioactive. These radioactive atoms are called fission products. They disintegrate further with the release of nuclear radiations. Many of them decay away quickly, in a matter of minutes or hours, to non-radioactive forms.

Others decay away more slowly and require months, and in a few cases, many years to decay. The fission products accumulating in the fuel rods include both gases and solids. Included are iodine, gases like krypton and xenon, and solids like cesium and strontium.

2.6 How can radioactivity be released?

The only way that potentially large amounts of radioactivity can be released is by melting the fuel in the reactor core. The fuel that is removed from a reactor after use and stored at the plant site contains considerable amounts of radioactivity. However, accidental releases from such fuel were found to be very small compared to potential releases of radioactivity from the full reactor core.

The safety design of reactors includes a series of systems to prevent the overheating of fuel and to control potential releases of radioactivity from the fuel. Thus, to get an accidental release of radioactivity to the environment there must be a series of sequential failures that cause the fuel to overheat and release its radioactivity. There must also be failures in the systems designed to remove and contain the radioactivity.

The study has examined thousands of potential paths by which radioactive releases could occur and has identified those that determine the risks. This involved defining the ways in which the fuel in the core could melt and the ways in which systems to control the release of radioactivity could fail.

2.7 How might a core melt accident occur?

It is significant that not once in some 200 reactor years of commercial operation of reactors of the type considered in the report has there ever been fuel melting. To melt the fuel requires that a failure in the cooling system occur that allows the fuel to heat up to its melting point, about 5,000°F.

To those unfamiliar with the characteristics of reactors, it might seem that all that is required to prevent fuel from overheating is a system to promptly stop, or shut down, the fission process at the first sign of trouble. Although reactors have such systems, they alone are not enough since the radioactive decay of the fuel continues to generate heat (called decay heat) that must be removed even after the fission process stops. Thus,

redundant decay heat removal systems are also provided in reactors. In addition, emergency core cooling systems (ECCS) are provided to cope with a series of potential but unlikely accidents.

The Reactor Safety Study has defined two broad types of situations that might potentially lead to a melting of the reactor core: the loss of coolant accident (LOCA) and transients. In the event of a loss of coolant, the normal cooling water is lost from the cooling systems and core melting would be prevented by the use of the emergency core cooling system (ECCS). However, melting could potentially occur in a loss of coolant if the ECCS were to fail to operate.

The term transient refers to any one of a number of conditions which can occur in a plant that require the reactor to be shut down. Following shutdown, the decay heat removal systems operate to keep the core from overheating. Certain failures in either the shutdown or the decay heat removal systems have the potential to cause melting of the core.

2.8 What features are provided in reactors to cope with a core melt accident?

Nuclear power plants have numerous systems to prevent core melting. Furthermore, there are inherent physical processes and additional features that remove and contain the radioactivity released from the molten fuel should core melting occur. Although there are features provided to keep the containment building from being damaged for some time after the core melts, the containment will ultimately fail, causing a release of radioactivity.

An essentially leak tight containment building is provided to prevent the initial dispersion of the airborne radioactivity into the environment. Although the containment will fail a number of hours after the core melts, until that time, the radioactivity released from the fuel will be deposited by natural processes on the surfaces inside the containment. In addition, plants are provided with systems to contain and trap the radioactivity released within the containment building. These systems include such things as water sprays and pools to wash radioactivity out of the building atmosphere and filters to trap radioactive particles prior to their release. Since the containment buildings are made essentially leak tight, the radioactivity is contained as long as the building remains intact. Even if the building were to have sizable leaks, large amounts of the radioactivity

would be removed by the systems provided for that purpose or would be deposited on interior surfaces of the building by natural processes.

Even though the containment building would be expected to remain intact for some time following a core melt, eventually the molten mass would be expected to eat its way through the concrete floor into the ground below. Following this, most of the radioactive gases will be trapped in the soil; however, a small amount would escape to the surface and be released. Almost all of the nongaseous radioactivity would be trapped in the soil.

It is possible to postulate highly unlikely core melt accidents in which the containment building fails by overpressurization or by missiles created by the accident. Such accidents could release a large amount of airborne radioactivity and have more serious consequences. The consequences of these less likely accidents have been included in the study's results shown in Figures 4.2-30 to 4.2-32.

2.9 How might the loss of coolant accident lead to a core melt?

Loss of coolant accidents are postulated to result from failures in the normal reactor cooling water system and plants are designed to cope with such failures. The water in the reactor cooling systems is at a very high pressure (between 50 to 100 times the pressure in a car tire) and if a rupture were to occur in the pipes, pumps, valves, or vessels that contain it, then a "blow out" would happen. In this case the water would flash to steam and blow out of the hole. This could be serious since the fuel could melt if additional cooling were not supplied in a rather short time.

The loss of normal cooling in the event of a LOCA would stop the chain reaction so that the amount of heat produced would drop almost instantly to a few percent of its operating level. However, after this sudden drop the amount of heat being produced would decrease much more slowly and would be controlled by the decay of the radioactivity in the fuel. Although this decrease in heat generation is helpful, it would not be enough to prevent the fuel from melting unless additional cooling were supplied. To deal with this situation, reactors have emergency core cooling systems (ECCS) whose function is to provide cooling for just such events. These systems have pumps, pipes, valves, and water supplies

which are capable of dealing with breaks of various sizes. They are also designed to be redundant so that if some components fail to operate, the core can still be cooled.

The study has reviewed a large number of potential sequences of events following LOCAs of various sizes. In almost all of the cases, the LOCA must be followed by multiple failures in the emergency core cooling system for the core to melt. The principal exception to this is the massive failure of the large pressure vessel that contains the core. However, the accumulated experience with pressure vessels indicates that the chance of such a failure is indeed very small. In fact, the study found that the likelihood of pressure vessel failure is so small that it does not contribute to the overall risk from reactor accidents.

2.10 How might a reactor transient lead to a core melt?

The term reactor transient refers to a number of events that require the reactor to be shut down. These range from normal shutdown for such things as refueling to such unplanned but expected events as loss of power to the plant from the utility transmission lines. The reactor is designed to cope with unplanned transients by automatically shutting down. Following shutdown, cooling systems would be operated to remove the heat produced by the radioactivity in the fuel. There are several different cooling systems capable of removing this heat, but if they all should fail, the heat being produced would be sufficient to eventually boil away all the cooling water and melt the core.

In addition to the above pathway to core melt, it is also possible to postulate core melt resulting from the failure of the reactor shutdown systems following a transient event. In this case it would be possible for the pressure to increase enough so that the normal reactor cooling system might rupture. This would create a loss of coolant accident and could lead to core melting.

2.11 How likely is a core melt accident?

The Reactor Safety Study carefully examined the various paths leading to core melt. Using methods developed in recent years for estimating the likelihood of such accidents, a probability of occurrence was determined for each core melt accident identified. These probabilities were combined

to obtain the total probability of melting the core. The value obtained was one in 17,000 per reactor per year. With 100 reactors operating, as is anticipated for the U.S. by about 1980, this means that one such accident would occur, on the average, every one and three quarters centuries.

It is important to note that a melting of the core in a nuclear power plant does not necessarily involve an accident with serious public consequences. One of the major findings of the study is that only about one in 10 potential core melt accidents, occurring on the average of once every 17 centuries, might produce measurable health effects.

2.12 What is the nature of the health effects that a core melt accident might produce?

It is possible for a core melt accident to release enough radioactivity so that some fatalities might occur within a short time (a few weeks) after the accident. Other people may be exposed to radiation levels which would produce observable effects which would require medical attention but from which they would recover completely. In addition, some people may receive even lower exposures which produce no noticeable effects but may increase the incidence of certain diseases over a period of many years. The observable effects which occur shortly after the accident are called short term or acute effects.

The delayed or latent effects of radiation exposure can cause some increase in the incidence of diseases such as cancer, genetic effects and thyroid gland illnesses in the exposed population. These effects would appear as an increase in these diseases over a 10 to 20 year period following the exposure. Such effects would be difficult to notice because the increase is usually small compared to the normal incidence rate of these diseases.

The study has conservatively estimated the increased incidence of potentially fatal cancers over the 20 years following an accident. This has been done following a procedure which estimates the number of extrapolating data from high dose rates to low dose rates. It is generally believed that this procedure probably overestimates the effect considerably, but it is not possible to do experiments with large enough populations to determine these very small effects. The number of latent cancers are predicted to be very small compared to the normal incidence of cancer.

Thyroid illness refers to small lumps on the thyroid gland that can be felt by an examining physician; they are treated by medical procedures that sometime involve simple surgery and rarely lead to serious consequences. For very large potential reactor accidents, the increase in nodules would be about equal to their normal incidence rate.

Radiation is recognized as one of the factors that can produce genetic effects which appear as defects in a subsequent generation. From the total population exposure caused by the accident, the expected increase in congenital defects in subsequent generations can be estimated. These effects are predicted to be very small compared to their normal incidence rate.

2.13 What are the most likely consequences of a core melt accident?

The most likely core melt accident would occur on the average of one every 17,000 years per plant. The size of the consequences of such an accident are given below.

CONSEQUENCES OF THE MOST LIKELY CORE MELT ACCIDENT

	<u>Consequences</u>
Fatalities	<1
Injuries	<1
Latent Fatalities	<1
Thyroid Nodules	~4
Genetic Defects	<1
Property Damage*	\$100,000

*This does not include damage that might occur to the plant.

2.14 How does the annual risk from nuclear accidents compare to other common risks?

Considering the 15 million people who live within 20 miles of current or planned U.S. reactor sites, and based on current accident rates in the U.S., the annual number of fatalities and injuries expected from various sources are shown in the table below.

ANNUAL FATALITIES AND INJURIES EXPECTED AMONG THE 15 MILLION
PEOPLE LIVING WITHIN 20 MILES OF U.S. REACTOR SITES

<u>Accident Type</u>	<u>Fatalities</u>	<u>Injuries</u>
Automobile	4,200	375,000
Falls	1,500	75,000
Fire	560	22,000
Electrocution	90	--
Lightning	8	--
Reactors (100 plants)	0.3	6

2.15 What is the number of fatalities and injuries expected as a result of a core melt accident?

A core melt accident is similar to many other types of major accidents such as fires, explosions, dam failures, etc., in that a wide range of consequences is possible depending on the exact conditions under which the accident occurs. In the case of a core melt, the consequences depend mainly on three factors; the amount of radioactivity released, the way it is dispersed by the prevailing weather conditions, and the number of people exposed to the radiation. With these three factors known it is possible to make a reasonable estimate of the consequences.

The study calculated the health effects and the probability of occurrence for 4800 possible combinations of radioactive release magnitude, weather type, and population exposed. The probability of a given release was determined from a careful examination of the likelihood of various reactor system failures. The probability of various weather conditions was obtained from weather data collected at many reactor sites. The probability of various numbers of people being exposed was obtained from U.S. census data for current and planned U.S. reactor sites. These thousands of calculations were carried out with the aid of a large digital computer.

These calculations showed that the probability of accidents having 10 or more fatalities is predicted to be about 1 in 250,000 per plant per year. The probability of 100 or more fatalities is predicted to be about 1 in 1,000,000 and for 1000 or more, 1 in 100,000,000. The largest calculated value was 2300 fatalities with a probability of about one in a billion.

The estimates given above are based on the assumption that evacuation procedures would be used to move most of the people out of the path of the airborne radioactivity. Experience has shown that evacuations have been successfully carried out in a large number of non-nuclear accident situations. Since nuclear power plants have evacuation plans prepared and since there is warning time before radioactivity would be released to the environment, it seems highly likely that evacuation would be effective in the case of nuclear accidents.

If we consider a group of 100 similar plants then the chance of an accident causing 10 or more fatalities is 1 in 2500 per year or, on the average, one such accident every 25 centuries. For accidents involving 1000 or more fatalities the number is 1 in 1,000,000 or once in a million years. Interestingly, this is just the probability that a meteor would strike a U.S. population center and cause fatalities.

The table shown below can be used to compare the likelihood of a nuclear accident to non-nuclear accidents that could cause the same consequences. These include man-caused as well as natural events. Many of these probabilities are obtained from historical records but others are so small that no such event has ever been observed. In the latter cases the probability has been calculated using techniques similar to those used for the nuclear plant.

PROBABILITY OF MAJOR MAN-CAUSED AND NATURAL EVENTS

<u>Type of Events</u>	<u>Probability of 100 or more Fatalities</u>	<u>Probability of 1000 or more Fatalities</u>
<u>Man-Caused</u>		
Airplane Crash	1 in 2 years	1 in 2000 years
Fire	1 in 7 years	1 in 200 years
Explosion	1 in 16 years	1 in 120 years
Toxic Gas	1 in 100 years	1 in 1000 years
<u>Natural</u>		
Tornado	1 in 5 years	very small
Hurricanes	1 in 5 years	1 in 25 years
Earthquake	1 in 20 years	1 in 50 years
Meteorite Impact	1 in 100,000 years	1 in 1,000,000 years
<u>Reactors</u>		
100 plants	1 in 10,000 years	1 in 1,000,000 years

In regard to injuries from potential nuclear power plant accidents, the number of injuries that could require medical attention shortly after an accident is about two times larger than the number of fatalities predicted.

2.16 What is the magnitude of the latent or long term health effects?

As with the short term effects the magnitude of latent cancers, treatable latent thyroid illness, and genetic effects vary with the exact accident conditions. The table below illustrates the potential size of such events. The first column shows the consequences that would be produced by core melt accidents, the most likely of which has one change in 17,000 per plant per year of occurring. The second column shows the consequences for an accident that has a chance of 1 in million of occurring. The third column shows the normal incidence rate.

MAGNITUDE OF LATENT HEALTH EFFECTS EXPECTED IN A 20 YEAR PERIOD
FOR AN ACCIDENT THAT PRODUCES 100 FATALITIES

Effect	Chance Per Plant Per Year		Normal* Incidence Rate
	One in 17,000	One in 1,000,000	
Latent Cancers	< 1	450	64,000
Thyroid Illness	4	12,000	20,000
Genetic Effects	< 1	450	100,000

*This is the normal incidence that would be expected for people in the vicinity of any one reactor.

In these accidents, only the production of thyroid nodules would be observed and this only in the case of an exceedingly unlikely accident. These nodules are easily diagnosed and treatable by medical or surgical procedures. The other effects are too small to be discernable above the high normal incidence of these two diseases.

2.17 What type of property damage might a core melt accident produce?

A serious nuclear accident would cause no physical damage to property beyond the plant site but may contaminate it with radioactivity. At high levels of contamination, people would have to be moved temporarily from their homes until the radioactivity either decayed away or was removed. At levels lower

than this but involving a larger area, people might take simple actions to reduce possible contamination, but would continue being able to live in the area. The principal concern in this larger area would be to monitor farm produce to keep the amount of radioactivity ingested through the food chain small. Farms in this area would have to have their produce monitored and any produce above a safe level could not be used.

The most likely core melt accident, having a likelihood of one in 17,000 per plant per year, would result in little or no contamination. The probability of an accident that requires temporary evacuation of 20 square miles is one in 170,000 per reactor per year. Ninety per cent of all core melt accidents would be expected to be less severe than this. The largest accident might require temporary evacuation from 400 square miles. In an accident such as this, agricultural products, particularly milk, would have to be monitored for a month or two over an area about 100 times as large until the iodine decayed away. After that, the area requiring monitoring would be very much less.

2.18 What would be the cost of a core melt accident?

As with the other consequences, the cost will depend upon the exact circumstances of the accident. The cost calculated by the Reactor Safety Study included the cost of moving and housing the people that were evacuated, the cost caused by denial of land use and the cost associated with the denial of use of reproducible assets such as dwellings and factories. The most likely core melt accident, having a likelihood of one in 17,000 per plant per year, would cause property damage of about \$100,000. The chance of an accident causing \$100,000 damage would be about one in 50,000 per plant per year. Such an accident would be expected on the average to occur once every 5 centuries for 100 operating reactors. The probability would be about one in 1,000,000 per plant per year of causing damage of about 2-3 billion dollars. The maximum value would be predicted to be about 4-6 billion dollars with a probability of about one in 1,000,000,000 per plant per year.

This property damage risk from nuclear accidents can be compared to other risks in several ways. The largest man-caused events that have occurred are fires. In recent years there have been an average of three fires with damage in excess of 10 million dollars every year. About once every two years there is a fire with damage in the 50 to 100 million dollar range. There have been four hurricanes in the last 10 years which caused damage in the range of .5

to 5 billion dollars. Recent earthquake estimates suggest a one billion dollar earthquake can be expected in the U.S. about once every 50 years.

A comparison of the preceding costs shows that, although a severe reactor accident would be very costly, it would not be significantly larger than a number of serious accidents which our society deals with quite often, and the probability of such a nuclear accident is of course estimated to be much smaller than the other events.

2.19 What will be the chance of a reactor melt down in the year 2000 if we have 1000 reactors operating?

One might be tempted to take the per plant probability of a particular reactor accident and multiply it by 1000 to estimate the chance of an accident in the year 2000. This is not a valid calculation however because it assumes that the reactors to be built during the next 25 years will be the same as those being built today. Experience with other technologies such as automobiles and aircraft show that as more units are built and more experience is gained the overall safety record in terms of the probability of accidents per unit decreases. There are already changes in plants now being constructed that appear to be improvements over the plants analyzed in the study.

2.20 How do we know that the study has included all accidents in the analysis?

The study devoted a large amount of its effort to ensuring that it covered all potential accidents important in determining the public risk. It relied heavily on over 20 years of experience that exists in the identification and analysis of potential reactor accidents. It also went considerably beyond earlier analyses that have been performed by considering a large number of potential failures that had never before been analyzed. For example, failure of reactor systems that can lead to core melt and the failure of systems that affect the consequences of core melt have been analyzed. The consequences of the failure of the massive steel reactor vessel were considered for the first time. The likelihood that various external forces such as earthquakes, floods and tornadoes could cause accidents were also analyzed.

In addition there are further factors that give a high degree of confidence that all significant accidents have been included. These are: 1) the identification of all significant sources of radioactivity located at nuclear power plants, 2) the fact that a large release of radioactivity can occur only if reactor fuel melts, and 3) knowledge of the factors that can cause fuel to

melt. This type of approach led to the screening of thousands of potential accident paths to identify those that would determine the public risk.

While there is no way of proving that all possible accident sequences which contribute to public risk have been considered in the Study, the systematic approach used in identifying possible accident sequences make it very unlikely that an accident which would contribute to the overall risk was overlooked.

2.21 How do your calculations of reactor accidents compare with those of earlier studies that predicted much larger consequences?

The principal earlier study of reactor accidents (WASH-740) was published by the AEC in 1957, before any commercial nuclear power plants were operating. Thus, this study was necessarily vague about the engineering details of reactor accidents. The purpose of that study was to essentially maximize the consequences that could occur in an accident. This was done because it was to serve as a basis for the Congress to use in establishing adequate indemnification of the public in the event that an accident occurred. Thus, WASH-740 served as the basis for the Price-Anderson Act which provides such indemnification.

The reactor used for the WASH-740 study was one that generated 500 million watts (mgeawatts) of thermal energy as opposed to today's reactor of about 3200 megawatts. To compare the earlier estimates with the more realistic approach used in this study, calculations were made for a 500 megawatt reactor using the Reactor Safety Study model. The results are presented in the table below.

COMPARISON OF CONSEQUENCES FROM ACCIDENTS IN A 500 MWt
REACTOR AS CALCULATED IN WASH-740 AND AS PREDICTED BY WASH-1400

Parameter	WASH-740	WASH-1400	
	Peak	Peak	Average
Acute Deaths	3,400	92	0.05
Acute Illness	43,000	200	0.01
Total Dollar Damage (billions)	7 ¹	1.7 ²	0.51 ²
Approximate Chance per Reactor Year		One in a billion	One in ten thousand

¹This is the value predicted in 1957 dollars.

²The values shown are in 1973 dollars. In 1957 dollars, these values should be about two-thirds of that shown.

The differences between these two sets of results can in large part be explained as follows:

1. This study used actual population data from the census bureau for the areas in the vicinity of actual reactor sites. WASH-740 used an estimated population that was much higher.
2. WASH-740 assumed that 50% of all the core radioactivity would be released to the environment. This study, using available experimental data, finds it physically impossible to attain total core releases as large as those used in WASH-740.
3. The WASH-740 calculation made no provisions for the evacuation of people. Experience shows that evacuation is highly likely and would significantly reduce the consequences of an accident should it occur.
4. The radioactivity released in a potential reactor accident would be in the form of a plume such as can be seen from smoke stacks. The radioactivity has sufficient heat associated with it to cause the plume to rise, thus reducing the concentration of radioactivity near the ground. This has some effect in reducing consequences. WASH-740's calculations did not include this effect.

2.22 What techniques were used in performing the study?

The latest methodologies, developed over the past 10 years by Department of Defense and National Aeronautics and Space Administration, were used in the Study. These techniques are called event trees and fault trees and help to define potential accident paths and their likelihood of occurrence.

An event tree defines an initial failure within the plant. It then examines the course of events which follow as determined by the operation or failure of various systems that are provided to prevent the core from melting and to prevent the release of radioactivity to the environment. Event trees were used in this Study to define thousands of potential accident paths which were examined to determine their likelihood of occurrence and the amount of radioactivity that they might release.

Fault trees were used to determine the likelihood of failure of the various systems identified in the event tree accident paths. A fault tree starts with the definition of an undesired event, such as failure of a system to operate, and then determines, using engineering and mathematical logic, the ways in which the system can fail. Using data covering 1) the failure of components such as pumps, pipes and valves, 2) the likelihood of operator errors, and

3) the likelihood of maintenance errors, it is possible to estimate the likelihood of system failure, even where no data on total system failure exists.

The likelihood and the size of radioactive releases from potential accident paths were used in combination with the likelihood of various weather conditions and population distributions in the vicinity of the reactor to calculate the consequences of the various potential accidents.

2.23 How will the results of the study affect safety decision making?

This study, using an overall methodology directed toward risk assessment, has developed new insights that contribute to a better understanding of reactor safety. However, many of the techniques used were developed and used only for the purpose of overall risk assessment and are not directly applicable for optimizing safety designs or evaluating the acceptability of specific designs or reactor site locations. Although the techniques developed in the Study may someday be useful for such purposes, considerable additional development is needed before they can assist effectively in safety decision making.

Decision making processes in many fields, and especially in safety, are quite complex and should not lightly be changed. This is especially true for nuclear power plants. The use of quantitative techniques in decision making associated with risk is still in its early stages and is highly formative. It appears that for the near future considerable additional development is needed in quantitative techniques before they can be used effectively in safety decision making processes.

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4.3 FUEL-FABRICATION PLANTS

4.3.1 Introduction

4.3.1.1 Purpose and Need

Fuel fabrication is a major part of the fuel cycle for LMFBRs. It involves all the operations required to consolidate the uranium- and plutonium-bearing materials into dense materials suitable for loading into metal tubes and to assemble welded fuel pins into fuel elements for the reactor. Fuel must be fabricated not only to load the reactor initially but also to reload the reactor when fuel is discharged at the end of its exposure period. Several fuel forms have been considered for use in LMFBRs. These include pellets or microspheres of uranium and plutonium in the form of oxides, carbides, nitrides, or carbonitrides in sealed metal cladding.

Stainless-steel-clad $\text{UO}_2\text{-PuO}_2$ (mixed-oxide) pellets were selected as the reference fuel for this Statement since this is expected to be the principal fuel form in the near-term LMFBRs.¹ Two types of rods are used in the model LMFBR--the core fuel rods and the radial-blanket rods. The core fuel rods would contain not only mixed-oxide fuel pellets but also depleted UO_2 axial-blanket pellets, both above and below the mixed-oxide fuel pellets. The radial-blanket rods would contain only depleted UO_2 pellets.

4.3.1.2 Status of Fuel-Fabrication Technology

Nine commercial plants licensed by the AEC can now perform part or all of the plutonium fuel-fabrication operation. Table 4.3-1 lists these plants and the operations they perform. Two of these (Kerr-McGee at Crescent, Oklahoma, and NUMEC at Parks Township, Pennsylvania) are currently involved in production-scale operations in the fabrication of mixed-oxide fuel for the FFTF reactor. The other plants have operations, generally on a smaller scale, to fabricate fuel for tests of plutonium-recycle fuel in LWRs.

No mixed-oxide fabrication plants of the 1500-metric-ton/year size described in Section 4.3.4 exist today; however, a plant designed to produce about 200 metric tons of mixed-oxide (3 to 4% plutonium) per year for plutonium recycle in LWRs is scheduled for operation in 1979.²

Two methods, coprecipitation and mechanical blending of individual oxides, are used commercially to prepare mixed oxides. When well controlled, these methods provide a high degree of microhomogeneity. Other preparation methods, such as the sol-gel process, have been developed on a laboratory scale and could be scaled up for commercial use if a significant economic incentive

Table 4.3-1

COMMERCIAL PLANTS LICENSED FOR PLUTONIUM FUEL FABRICATION

Licensee	Plant Location	Feed Material	Plant Product
Atomics International ^a	Canoga Park, Calif.		
Babcock & Wilcox ^b	Lynchburg, Va.		
Exxon Nuclear	Richland, Wash.	UO ₂ and PuO ₂ powders	UO ₂ -PuO ₂ fuel assemblies
General Electric	Pleasanton, Calif.	Nitrate solutions (U and Pu)	UO ₂ -PuO ₂ fuel pins
Gulf United Nuclear ^c	Elmsford-Pawling, New York	UO ₂ and PuO ₂ powders	UO ₂ -PuO ₂ fuel pins
Kerr-McGee	Crescent, Okla.	Nitrate solutions (U and Pu)	UO ₂ -PuO ₂ fuel pins
Nuclear Fuel Services ^d	Erwin, Tenn.	Nitrate solutions (U and Pu)	UO ₂ -PuO ₂ fuel pins
NUMEC	Parks Township, Pa.	UO ₂ and PuO ₂ powders	UO ₂ -PuO ₂ fuel pins
Westinghouse	Cheswick, Pa.	UO ₂ and PuO ₂ powders	UO ₂ -PuO ₂ fuel pins

^aNo UO₂-PuO₂ fabrication for about 3 years.

^bAll work performed by NUMEC.

^cOperations terminated late 1973.

^dPlutonium recovery operations only at present time.

were shown to result either from improved operational behavior of the fuel or in simplification of reprocessing.

Chemical and microstructural characterization methods have been developed to measure such things as porosity, grain size, compositional inhomogeneity, and inclusion condition. Fuel-fabrication parameters adequate to provide several significant chemical and structural variations in test fuels have been developed. However, fuel-fabrication development and microstructure-evaluation methods adequate to allow assessment of the effects of these variables on performance require additional development. These efforts should allow identification of fuel-microstructure and fabrication processes for fuels having improved performance.

Fuel-element fabrication experience for the FFTF has provided proof-of-principle information for commercial fabrication. Procedures and quality-control measures to ensure high-quality irradiation specimens have been standardized for these fuel elements. Fabrication of fuel elements and the development of quality-control measures leading to proof-of-principle for improved fuels must continue for demonstration and commercial plants.³

Industry is gaining experience in the fabrication of LMFB-type fuel by preparing fuel pins for FFTF. Recently, Kerr-McGee and Babcock and Wilcox each were awarded a contract for a core load of FFTF fuel pins.... Operating experience with a large number of fuel elements under well-characterized conditions (in FFTF) will permit a systematic assessment of the impact of fabrication specifications and the QA program on both vendor and fuel performance.⁴

4.3.1.3 Status of Technology for Confinement of Radioactivity

The technology for confinement of radioactivity has been developed through many years of handling radioactive materials at national laboratories, at AEC contractor sites, and at industrial plants. Specific technology for confinement of plutonium during large-scale fabrication operations was demonstrated at Hanford during the Plutonium Utilization Program and at Rocky Flats. For the past few years, Hanford Engineering Development Laboratory has operated a pilot plant to demonstrate processes for fabrication of UO_2 - PuO_2 fuel for the FFTF reactor. Plutonium-fabrication operations have also been conducted at Argonne National Laboratory, Battelle Memorial Institute, Los Alamos Scientific Laboratory, Mound Laboratory, Oak Ridge National Laboratory, and the industrial sites listed in Table 4.3-1.

Confinement of plutonium has been provided through the use of glove-box facilities, which are described in Section 4.3.4.1.3. Glove boxes that contain plutonium are operated at negative pressure relative to that of the surrounding area so that any leakage is always inward.

Laboratories or buildings used for plutonium operations are also operated at negative pressure relative to that of surrounding areas so that the flow of air is inward. All air exhausted from glove boxes and the rooms or buildings that house

the glove boxes is filtered through high-efficiency particulate air (HEPA) filters prior to release to the atmosphere. HEPA filters are used in series to ensure adequate confinement. The efficiency of HEPA filters in removing airborne particulates is discussed in Section 4.3.5.3.

DISCUSSION 11

Facilities where plutonium is handled are designed and operated in such a manner that potentially contaminated liquids are not discharged to the environment until they have been monitored for radioactivity.

All solid waste that leaves plutonium-handling facilities is monitored for radioactivity. Contaminated solid waste is disposed of in AEC-licensed waste-management facilities, which are described in Section 4.6.

AEC-funded programs to measure the effectiveness of present-day plutonium containment techniques⁵ and to design systems for confining plutonium during large-scale operations are continuing.⁶

4.3.2 The Sites--Requirements and Considerations

Considerations relative to the selection of a site for a future LMFBR fuel fabrication plant are very similar to those discussed in Section 4.2.2 for an LMFBR power plant. The principal differences between the site requirements for a fuel fabrication plant and a power plant are:

- (1) The cooling water requirements of a fuel fabrication plant are small relative to those of an LMFBR power plant. However, as discussed in Section 4.3.6.2, a river with a flow rate of 10^4 to 10^5 cfs would be required to dilute the chemical effluents from the ADU process (described in Section 4.3.4.2.1) for making UO_2 pellets. This requirement would severely limit the number of acceptable sites, unless processes to reduce chemical effluents are developed (as discussed in Section 4.3.6.2).
- (2) Electrical transmission lines from the site are not required.
- (3) There is little incentive for location of the fabrication plant near electrical load centers.

Approval for construction and operation of an LMFBR fuel fabrication plant must be obtained from the AEC and other governmental bodies, both state and Federal. Requirements for approval by such agencies will be similar to those for power stations. Although 10 CFR 100 applies only to reactor sites, similar criteria are

expected to be used in evaluating the acceptability of proposed sites for fuel fabrication plants.

The selection of a site for a fuel-fabrication plant will be based primarily on economics. Factors that will influence the selection are: availability of large tracts of undeveloped land, proximity to other elements of the nuclear-fuel cycle (nuclear reactors and fuel-reprocessing plants), and accessibility to major highway systems. Recent nuclear industry practices indicate that a typical site would be about 10 miles from a small town (population 10,000) and about 20 to 50 miles from a larger town (population >40,000).

4.3.3 Environmental Impacts of Construction

Material on the environmental impact of construction is presented in Section 4.2.4. Some major differences between the impact of fuel-fabrication (and fuel-reprocessing) plant construction and that for the reactor are related to the considerably smaller physical dimensions of excavations, site structures, and cooling systems as compared with those required for the reactor.

4.3.4 Plant Description

An LMFBR fuel-fabrication plant was assumed to be divided into two sections. One section would manufacture depleted uranium dioxide powder, uranium-dioxide pellets for both the axial and radial blankets, and radial-blanket rods. The other section would manufacture mixed-oxide pellets and core fuel rods, using axial-blanket pellets obtained from the uranium dioxide section of the plant.

Although the two sections of the future fuel-fabrication plant could be at different locations, in this study they are assumed to be located on the same site. A further assumption is that the two sections of the future fuel-fabrication plant would be operated as completely separate plants because the containment requirements for plutonium are more stringent than those for depleted uranium.

The assumption is made that the capacity of the fuel-fabrication plant is 5 metric tons of uranium plus plutonium per day to the reactor. This includes the mixed-oxide fuel pellets and the uranium-dioxide blanket pellets. Figure 4.3-1 is a material-balance flow sheet for fabrication plants which have this capacity. The initial fuel fabrication plants will probably be smaller than this; however, plants with this capacity can reasonably be expected to be built once the LMFBR fuel cycle reaches equilibrium. Also, for the purpose of this environmental statement, selection of a plant capacity of 5 metric tons/day makes the fabrication plant the

4.3-6

EQUILIBRIUM MATERIAL BALANCE FLOW CHART FOR A
5-METRIC TON OF HEAVY METAL PER DAY LMFBR FUEL-FABRICATION PLANT

Figure 4.3-1

same relative size as the reprocessing plant. The isotopic compositions of the uranium and plutonium assumed to be used in the fuel fabrication plant are given in Table 4.3-2.

These plants, operating 300 days/year (1500 metric tons/year), would produce the annual fuel requirement for about eighty 1000-MWe LMFBRs operating at 80% load factor.⁷

A future LMFBR fuel-fabrication plant would be expected to consist of one large manufacturing building each for the UO_2 section and the mixed-oxide section, a combined administration and personnel services building, and several smaller utility buildings, all located on 100 to 150 acres near the center of a 1000-acre site. Each manufacturing building would be a two- or three-story building of concrete-and-steel construction and would occupy 100,000 to 150,000 ft². The administration and personnel-services building (50,000 to 75,000 ft²) would be close to the manufacturing buildings. Good architectural design concepts are expected to be used to yield an integrated complex that is aesthetically pleasing and functionally arranged.

The mixed-oxide section of the plant would be enclosed by a security fence. A parking lot for employees (500 to 600 spaces) would be located outside the security fence. Areas disturbed by construction would be landscaped with grass, shrubs, and trees as appropriate.

Figure 4.3-2 is an artist's concept of Westinghouse's proposed Recycle Fuels Plant² for fabrication of 200 metric tons/year of mixed-oxide fuel for LWRs to be located near Anderson, South Carolina. The mixed-oxide section of the fuel-fabrication plant described herein could be similar to this plant.

4.3.4.1 Mixed-Oxide Section of Fuel-Fabrication Plant

In the mixed-oxide section of a future LMFBR fuel-fabrication plant, fuel rods that would contain UO_2 - PuO_2 fuel pellets and depleted UO_2 blanket pellets would be manufactured. Also, completed fuel rods would be assembled into fuel elements.

4.3.4.1.1 Process Description

The assumption is made that mixed-oxide pellets would be fabricated from mechanically mixed UO_2 and PuO_2 powders. Figure 4.3-3 is a flow sheet that shows the preparation of mixed-oxide fuels. The PuO_2 powder used as feed material would be received from a fuel-reprocessing plant in approved shipping containers⁸ and would be transferred into

Table 4.3-2
ISOTOPIC COMPOSITION OF URANIUM AND PLUTONIUM
ASSUMED TO BE USED IN THE FUEL FABRICATION PLANT

Element	Isotope	Content (wt %)
Uranium	U-234	0.0006
	U-235	0.21
	U-236	0.003
	U-238	99.79
Plutonium	Pu-236	4.2×10^{-6}
	Pu-238	1.2
	Pu-239	64.7
	Pu-240	24.5
	Pu-241	5.8
	Pu-242	3.8
	Pu-244	1.8×10^{-5}

4.3-9

ARTIST'S CONCEPT OF THE WESTINGHOUSE RECYCLE FUELS PLANT

Figure 4.3-2

FLOW CHART SHOWING PREPARATION OF MIXED-OXIDE FUELS

Figure 4.3-3

storage vessels. Depleted UO_2 powder would be obtained from the UO_2 section of the plant described in Section 4.3.4.2 and would be transferred into storage.

A third component used as feed to the first step of the pellet-fabrication process would be UO_2 - PuO_2 powder made by recycling off-specification material such as chipped or broken pellets and pellets that do not meet density or dimensional standards. Clean scrap powders and pellets would be collected, analyzed, and then subjected to an oxidation-reduction treatment that yields mixed-oxide powder suitable for blending with the other feed materials. The clean recycle scrap material would be stored in scrap-storage vessels pending its introduction into the blending operation.

The PuO_2 powder would be pulverized prior to being blended with the UO_2 powder. The blending operation would be done by weighing individually each of the three components (PuO_2 , UO_2 , and recycle mixed-oxide powder) in accordance with process requirements. Each component would then be charged into the blender. The blended materials would be discharged to a powder compactor and granulator to produce a granulated, free-flowing powder for subsequent use in the pellet-pressing operation.⁹

The powder would be pressed into pellets of the desired shape and then would be sintered to the required density in a reducing atmosphere to yield substoichiometric mixed-oxide pellets. The sintered pellets would be ground to the required diameter, inspected, and transferred to the rod-loading area. After the fuel was shown to meet the specifications, it would be loaded into fuel-rod tubes along with UO_2 axial-blanket pellets. End plugs would be welded on the loaded rods. After a final inspection to ensure the integrity of the welds, the fuel rods would be assembled into fuel elements, which would be stored for shipment to the reactor. Figure 4.3-4 describes the operations to be performed in the mixed-oxide section of a fuel-fabrication plant.

4.3.4.1.2 Process Alternatives

An alternative method for making powder for mixed-oxide pellets is the coprecipitation process. In this process, solutions of uranyl nitrate, $\text{UO}_2(\text{NO}_3)_2$, and plutonium nitrate, $\text{Pu}(\text{NO}_3)_4$, are blended to obtain the required ratio of plutonium to uranium. Ammonium hydroxide, NH_4OH , is used to coprecipitate ammonium diuranate, $(\text{NH}_4)_2\text{U}_2\text{O}_7$, and plutonium hydroxide, $\text{Pu}(\text{OH})_4$; subsequently, the coprecipitate is dried and is calcined in a reducing atmosphere to yield UO_2 - PuO_2 powder. After granulation, this powder is pelletized and is sintered in the same manner as are the mechanically mixed powders.

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FLOW CHART DESCRIBING OPERATIONS IN
MIXED-OXIDE SECTION OF FUEL FABRICATION PLANT

Figure 4.3-4

The coprecipitation process was not chosen as the reference for this Statement because it requires the use of nitrate solutions of uranium and of plutonium as feed materials. Although shipping plutonium nitrate solutions was permissible under previous regulations,¹⁰ a change¹¹ to these regulations limits the shipment of plutonium to the solid form. If coprecipitated powder were required for mixed-oxide fuel fabrication, probably the powder would be prepared at the fuel-reprocessing plant.

4.3.4.1.3 Gaseous-Containment and -Discharge Systems

A double-containment system would be used to prevent the release of radioactive materials to the atmosphere. The primary containment would consist of the glove boxes or shielded facilities in which all process operations would be conducted. Figure 4.3-5 is a photograph of an unshielded glove box used for handling plutonium in a small fabrication facility. Typical glove boxes have steel frames that support glass or acrylic plastic windows. The windows have glove ports to allow the operator access to the process equipment. These glove boxes are airtight in order to contain radioactive material within the boxes. Material and equipment are introduced into glove boxes through sealed ports in such a manner that containment of radioactive material within the glove box is ensured. Glove boxes of the type shown in Figure 4.3-5 are adequate to prevent exposure of operating personnel to alpha radiation; however, additional shielding would be required to protect against the more-penetrating beta, gamma, and neutron radiation from recycled plutonium.

The building that houses the glove boxes or shielded facilities would provide the secondary containment. Both the primary- and secondary-containment areas would be maintained at reduced pressure relative to that of the surrounding areas so that the flow of air would always be inward, thus decreasing the potential for release of radioactive material. Gases discharged from either the primary- or secondary-containment areas would be filtered prior to being released to the environment.

Primary Containment

The primary containment¹² would be either partially shielded glove boxes or a completely shielded hot-cell facility. The neutron flux from recycled plutonium that contains significant amounts of the heavier isotopes of plutonium would require the use of shielding to prevent excessive exposure to operating personnel. Whether a shielded glove-box facility could be used or whether a hot-cell facility would be required has not been determined. An objective of the LMFBR-development program¹³ is to determine the shielding requirements for fabrication of recycled

4.3-14

AN UNSHIELDED GLOVE-BOX USED FOR HANDLING PLUTONIUM

Figure 4.3-5

plutonium into a usable fuel form. The assumption is made that the primary containment in the mixed-oxide section of a fuel-fabrication plant would consist of shielded glove boxes. If a hot-cell facility were used instead, the releases to the environment would be expected to be no greater than those from the glove-box facility assumed here.

Glove-box air would be received through a HEPA filter and would be exhausted through another HEPA filter to the glove-box exhaust system.^{12,14} Filters of this type have been used for many years in the nuclear industry to effectively remove radioactive particulates from air streams. A modular HEPA filter has a cross section of 2 ft by 2 ft, a depth of 1 ft, and a capacity of about 1000 cfm at a pressure differential of 1 in. H₂O. The modules are formed into banks to achieve the required capacity for filtering air. The filter medium is a pleated mat of woven Fiberglas. By definition, a HEPA filter is an expendable (single-use), extended-medium, dry filter that has: (1) a minimum particle-removal efficiency of no less than 99.97% for 0.3 μ m-size particles, (2) a maximal flow resistance of 1.0 in. H₂O when clean and when operated at the rated air-flow capacity, and (3) a rigid casing that extends the full depth of the medium.¹⁵ In addition to the HEPA filter located at the point of discharge from each glove box, the glove-box exhaust system would have two HEPA filters, in series, through which glove-box air would pass before being released to the atmosphere. These three filters in series are the minimum requirements for ventilation systems in plutonium fuel fabrication plants as outlined in Regulatory Guide 3.12.¹² The calculated aggregate transmission factor (for 0.7- μ m particles) of these three HEPA filters would be 1.25×10^{-10} . (See additional discussion of filter efficiency in Section 4.3.5.3.) All filters will be tested in place^{12,15} regularly with dioctyl phthalate (DOP) to ensure that their efficiency is not less than 99.95%. The glove-box exhaust system would be designed to maintain the air pressure in the glove box below that of the room in which it would be located.^{12,16} The gloves would present the principal weakness in the primary-confinement barrier since they would be subject to attack by chemical agents used in the glove box and to physical damage from abrasion, cutting, or puncture and since they would likely fail in case of an explosion or pressurization within the glove box. Gloves periodically will develop small leaks; however, normally, only trace contamination will be observed outside a glove box.¹⁴ Prudent operating procedures designed to ensure early detection and correction of such leaks would prevent the spread of contamination within the secondary-containment area.¹⁶

Secondary Containment

The secondary containment¹² would consist of the building in which the glove boxes would be located and its associated ventilation system. The ventilation system would be designed to maintain the pressure within the secondary-containment area below that of surrounding areas. All air from the secondary-containment area would be exhausted through two HEPA filters, arranged in series, which would be DOP-tested in place. These two filters would have a calculated aggregate transmission factor of 2.5×10^{-7} (see additional discussion of filter efficiency in Section 4.3.5.3). The final filters would be protected by a water-deluge system so that they would maintain their integrity even in the event of a general facility fire.¹²

The twice-filtered air from the building ventilation system and the thrice-filtered air from the glove-box exhaust system of the mixed-oxide section of a fuel-fabrication plant would be released to the atmosphere through a common rooftop stack, about 60 ft above ground level. The release point would be about 1000 m from the site boundary.

4.3.4.1.4 Liquid-Waste-Containment and -Disposal Systems

The mixed-oxide section of the future fuel-fabrication plant is assumed to have three separate liquid-waste systems. One system would handle all liquids known to be contaminated. Another would handle potentially contaminated liquids, such as mop water, water from laboratory sinks, and water from the dehumidifier and duct drains. The third system would handle sanitary wastes and water from cooling-tower blowdown. Figure 4.3-6 shows a schematic of these three systems.

Liquid waste contaminated with plutonium would be minimal because of the dry process used for mixed-oxide fabrication. The liquid wastes known to be contaminated would be mixed with concrete or with some other solidification agent and would be transferred to an AEC-licensed waste-management facility where they would be treated as "other than high level waste (plutonium contaminated)." Procedures for disposing of such wastes are described in Section 4.6.

Potentially contaminated liquid wastes would be assumed to be collected in a retention tank, analyzed, and released to join the sanitary wastes only if the concentration of radionuclides were below the limits set forth in 10 CFR 20 for liquid effluents in unrestricted areas. If the concentration of radionuclides in any tank should exceed those limits, the liquid in that tank would be treated by ion exchange or distillation to reduce the concentration to as low as practicable below the prescribed limits. If such treatment were not feasible,

DIAGRAM OF THE LIQUID-WASTE-HANDLING SYSTEMS FOR THE MIXED-OXIDE SECTION OF AN LMFBR FUEL-FABRICATION PLANT

Figure 4.3-6

the liquid would be transferred to the contaminated-waste-disposal system for conversion to a solid form.

The sanitary wastes and the water from the cooling-tower blowdown would be treated in a waste-treatment facility. Industrial plants of this type would be expected to have waste-treatment plants comparable with that proposed by Westinghouse Electric Corporation for the Recycle Fuels Plant, that is:

This plant will be 'tertiary', which will be capable of removing approximately 25% of the suspended solids from the normal effluent of a secondary treatment extended aeration system. Based on actual performance of some operating units, the total organic BOD suspended solids will be expected to be reduced to about 3.5 ppm and the effluent BOD to less than 5 ppm.¹⁷

4.3.4.2 Uranium-Dioxide Section of Fuel-Fabrication Plant

In the uranium-dioxide section of a future LMFBR fuel-fabrication plant, uranium hexafluoride (UF_6) would be converted to uranium dioxide (UO_2) powder. A portion of the UO_2 powder would be transferred to the mixed-oxide section of the plant for use in fabricating mixed-oxide pellets. The remainder would be used to make UO_2 pellets for use in both the axial- and radial-blanket regions of the core.

Some of the UO_2 pellets would be used to make radial-blanket rods in the UO_2 section of the plant. The UO_2 pellets to be used in the axial blanket would be transferred to the mixed-oxide section of the plant, where they would be loaded into fuel rods above and below the mixed-oxide fuel pellets. Figure 4.3-7 is a flow sheet that describes the operations which would take place in this section of the plant.

4.3.4.2.1 Process Description

The UO_2 powder would be assumed to be formed from depleted UF_6 by the ammonium diuranate (ADU) process.¹⁸ Depleted UF_6 probably would be received from an enrichment facility in 14-ton cylinders,¹⁹ although smaller cylinders (2-1/2 ton or 10 ton) could be used. Depleted UF_6 is a by-product of the enrichment of natural uranium to the isotopic composition required for use in LWRs.

In the ADU process, described in Figure 4.3-8, UF_6 is first reacted with water to form uranyl fluoride, UO_2F_2 , by the reaction

FLWSHEET DESCRIBING OPERATIONS IN URANIUM
DIOXIDE SECTION OF FUEL FABRICATION PLANT

Figure 4.3-7

FLWSHEET FOR PREPARATION OF
URANIUM DIOXIDE BY THE ADU
PROCESS

Figure 4.3-8

The UO_2F_2 is then reacted with ammonia, NH_3 , to precipitate ammonium diuranate, $(\text{NH}_4)_2\text{U}_2\text{O}_7$, commonly referred to as ADU. Excess NH_3 is added to neutralize the hydrofluoric acid, HF . The ADU is subsequently calcined to form UO_2 powder. The principal liquid effluent from the ADU process is water that contains dissolved ammonium fluoride, NH_4F , and small amounts of UO_2F_2 . This liquid effluent is discharged to a lined lagoon where lime, $\text{Ca}(\text{OH})_2$, is added to precipitate most of the fluorine as insoluble calcium fluoride, CaF_2 . This precipitation also reduces the uranium content of the effluent from the lagoon. The addition of lime produces a decontamination factor of 4 for uranium.

Ammonia gas and HF are released in the gaseous effluents during the calcination of the ADU to yield UO_2 powder; however, most of these chemicals would be removed from the gaseous effluents by water scrubbers. The aqueous discharge from these scrubbers would be assumed to join the remainder of the process liquid wastes prior to discharge to the lined lagoon.

The UO_2 powder would be pressed into pellets and sintered to the required density. Typically, sintering is done in an atmosphere of nitrogen and hydrogen at temperatures as high as 1700°C . The sintered pellets would be ground to the required diameter, inspected, and transferred to the rod-loading area. After the pellets were shown to meet specifications, they would be loaded into radial-blanket rods. End plugs would be welded onto the loaded rods. After a final inspection to ensure the integrity of the welds, the radial-blanket rods would be assembled into elements, which would be stored for shipment to the reactor.

4.3.4.2.2 Process Alternatives

An alternative way to make UO_2 powder is the direct conversion process, in which UF_6 is reacted with water vapor and hydrogen in a fluidized bed to form UO_2 . The gaseous effluent from the fluidized-bed converter vessel is passed through a sintered-nickel filter in the top of each vessel before going to the treatment process for removal of HF and uranium. The HF is removed in a wet scrubber by means of a solution of potassium hydroxide, KOH .²⁰ The potassium fluoride, KF , that is formed is discharged as a solution, whose KF concentration is about 3 molar.²⁰ The KF solution can be converted in an evaporator to a solid waste for disposal at a commercial waste-management facility. Therefore, no liquid effluent results from this process.

The UO_2 powder from the direct conversion process is used to prepare UO_2 pellets in the same manner as that described for the ADU process in Section 4.3.4.2.1.

The direct conversion process was not chosen as the reference for this Statement because it is not yet the established process for making UO_2 powder in the U.S. nuclear industry. However, this process is expected to be used in fuel-fabrication plants in the near future. General Atomic Company²¹ has described the use of this process to convert UF_6 to UO_2 for use in making fuel for the high-temperature gas-cooled reactor.

4.3.4.2.3 Containment and Discharge Systems for Gases

The radioactivity of gaseous-effluent streams would arise from the entrainment of fine particles that are generated from either the drying of entrained droplets of process liquids or the entrainment of small particles of UO_2 in the fabrication operations. The particles would be compounds of uranium and its decay products. Because the fuel contains no gaseous radioactive constituents, the gaseous-effluent-treatment systems would be designed to remove solid particles.

The hazard associated with handling uranium is much less than that of handling plutonium; therefore, operations with uranium would be performed in open-faced hoods rather than in glove-box facilities. The hood exhaust would be such that air would flow from the areas of low contamination into areas of high contamination. The concentration of radionuclides in building air thus would be kept sufficiently low that the health of employees would not be endangered. Roughing filters would be used to remove the bulk of the entrained UO_2 particulates from the hood exhaust. All the process off-gas streams from the ADU process for preparing UO_2 powder would be assumed to have water scrubbers.

The ventilation systems of UO_2 -fabrication plants would be designed to maintain UO_2 operating areas at reduced pressures relative to those of surrounding areas. All ventilation air, as well as that exhausted from process hoods, would be filtered through HEPA filters before being released to the atmosphere from rooftop stacks. Particulate matter can be removed from air streams with an efficiency of 99.97% by HEPA filters, but this efficiency is not always realized because of leaks that allow bypass of the filters. Since the filter banks in the UO_2 section of the plant would not be tested with DOP aerosol after installation, they would be credited with an efficiency of only 95%. This type of installation is generally monitored by pressure drop to detect leakage in excess of that required to give an overall efficiency of 95%.

4.3.4.2.4 Containment and Disposal Systems for Liquids

The radioactivity of the liquid effluent would arise from uranium and from the decay products of uranium. The early stages of liquid-waste treatment would be directed at returning the valuable uranium component to the process line with a minimum of intervening operations. As treatment of liquid-effluent streams diminished uranium to the point where it could no longer be economically recovered or where such recovery would return an intolerable amount of impurities to the process line, the objective would become the prevention of the escape of these trace amounts of radioactive materials in the liquid effluents. In addition to liquid effluents from the main ADU process line, contaminated effluents would result from scrap-recovery operations and from miscellaneous liquid wastes produced from showers, floor drains, laboratory sinks, and laundry water.

Figure 4.3-9 is a flow sheet that shows the waste-treatment system for the ADU process waste. The liquid waste is allowed to age for 16 to 20 hr before the residual $(\text{NH}_4)_2\text{U}_2\text{O}_7$ is separated by a combination of filtration and centrifuging. The fluoride is then precipitated by adding lime to form insoluble CaF_2 . The waste is pumped to a lagoon, where the CaF_2 settles out and the supernate overflows into a stream. The lagoon is lined to prevent seepage of radioactive and nonradioactive materials into the groundwater. The liquid effluents from the miscellaneous liquid waste-treatment systems are also pumped to the lined lagoon.

The scrap-recovery liquid wastes are pumped to a lined nitrate-storage lagoon, where they are retained. Accumulation of the nitrate wastes requires that a new lagoon be constructed periodically for this purpose.* Various nonradioactive liquid effluents, such as sanitary wastes and cooling-tower blowdown, would be treated in a waste-treatment facility similar to that described in Section 4.3.4.1.4. Figure 4.3-10 is a schematic diagram that shows all of the liquid-waste-treatment systems for the UO_2 section of the fuel-fabrication plant.

4.3.4.3 Plant Water Use

The mixed-oxide section of an LMFBR fuel-fabrication plant would use water at the rate of about 128,500 gal/day (gpd). Of this total, 99,500 gpd would be used for

*About 72 g of uranium per day would be discharged to these lined nitrate-storage lagoons. This discharge would result in the accumulation of the following amounts of radioactivity in the lagoon: ^{234}U , 780 $\mu\text{Ci}/\text{year}$; ^{235}U , 90 $\mu\text{Ci}/\text{year}$; ^{236}U , 39 $\mu\text{Ci}/\text{year}$; ^{238}U , 7140 $\mu\text{Ci}/\text{year}$. Techniques for the ultimate disposal of the nitrate wastes and contained uranium are under development. Source: T. Tamura et al., "Waste Management Studies - Biological Denitrification of High Nitrate Wastes," Environmental Sciences Division Annual Progress Report Period Ending September 30, 1974, Report ORNL-4935, Oak Ridge National Laboratory, March 1974, pp. 13-14.

ADU LIQUID PROCESS WASTE-TREATMENT SYSTEMS

Figure 4.3-9

SCHEMATIC SHOWING DILUTION OF LIQUID RADIOACTIVE WASTES FROM
URANIUM DIOXIDE SECTION OF FUEL FABRICATION PLANT WITH
SANITARY WASTES AND COOLING TOWER BLOWDOWN

Figure 4.3-10

cooling-tower makeup and 28,000 gpd for potable water (including sinks, washrooms, showers, and cafeteria), and 1000 gpd would be used in the manufacturing process (including analytical laboratory, personnel decontamination, and floor-mop water). Of the 99,500 gpd supplied to the cooling tower, 75,000 gpd would be needed because of evaporative loss, 1500 gpd because of drift, and 23,000 gpd for blowdown.

The UO_2 section of the plant would use water at the rate of about 338,000 gpd. Of this total, 198,000 gpd would be used for cooling-tower makeup, 56,000 gpd for potable water, and 84,000 gpd in the manufacturing process. Of the water that would be supplied to the cooling tower at the rate of 198,000 gpd, 150,000 gpd would be needed because of evaporative loss, 3000 gpd because of drift, and 45,000 gpd for blowdown.

Table 4.3-3 gives the estimated daily water requirement for the combined plants.

4.3.4.4 Heat-Dissipation System

Heat that would result from heating the facility during winter, from air-conditioning during the summer, and from process heat requirements would be dissipated to the environment from the fuel-fabrication plant. Most of this heat would be dissipated through a cooling tower. Based on the cooling tower described by Westinghouse for the Recycle Fuels Plant,²² the estimate is made that the mixed-oxide section of a fuel-fabrication plant would require a cooling tower that has a capacity of about 2.5×10^7 Btu/hr. The UO_2 section of the plant is estimated to require a cooling capacity of about 5×10^7 Btu/hr. The size of this cooling tower is about one-hundredth of that required for a 1000-MWe LMFBR. Inhibitors would probably be added to the recirculating water to prohibit corrosion and algae buildup.²²

4.3.5 Effluents from Normal Operation

In this assessment, the thermal, chemical, and radioactive effluents that would be released during normal operation were determined separately for each section of the plant. However, the effluents from the two sections of the fuel-fabrication plant were combined for the purpose of assessing their effects on the environment.

The mixed-oxide and UO_2 sections of the plant each would discharge gaseous effluents from separate rooftop stacks at individual rates of 100,000 scfm. The chemical and radioactive gaseous effluents discussed in Sections 4.3.5.2 and 4.3.5.3 would be contained in these total gaseous effluents.

Table 4.3-3

PLANT WATER USE (gpd)

Use	Mixed-oxide Section	Uranium Dioxide Section	Combined Plant Use
Process	1,000	84,000	85,000 ^a
Potable	28,000	56,000	84,000 ^a
Cooling Tower	99,500	198,000	297,500
Evaporative Loss	75,000	150,000	225,000
Drift	1,500	3,000	4,500
Blowdown	23,000	45,000	68,000 ^a
Total	128,500	338,000	466,500

^aLiquid effluents total 237,000 gpd.

The liquid effluents from the mixed-oxide and the UO_2 sections of a combined fuel-fabrication plant would be combined, prior to being released to the environment, as shown in Figure 4.3-11. The chemical and radioactive liquid effluents discussed in Sections 4.3.5.2 and 4.3.5.3 would be contained in the estimated total liquid effluent of 237,000 gpd from the combined plant.

The liquid effluents of the combined LMFBR fuel-fabrication plant are assumed to be discharged into a river that would have a flow rate of 600 cfs. Dilution of the liquid radioactive effluents by a factor of 2000 would occur within a short distance downstream. Complete mixing would be assumed to occur before any interaction with a human-exposure pathway took place.

4.3.5.1 Thermal

Heat would be released to the environment in both the liquid and gaseous effluents from a future LMFBR fuel-fabrication plant. Of the liquid effluents that would be released to the waste-treatment plant from the combined fuel-fabrication plant at a rate of 153,000 gpd (see Figure 4.3-11), all except about 27,000 gpd would be expected²³ to be at a temperature about 80°F. The remainder (27,000 gpd) would be shower water at a temperature less than 120°F. The resulting overall temperature of the liquid effluents at discharge would be expected to be less than 90°F. The process liquid waste discharged from the UO_2 section of the plant at a rate of 84,000 gpd would be retained in a lagoon for 70 days prior to being released to the environment; therefore, it is expected to be at ambient temperature.

The major portion of the heat that would result from heating the facility during the winter, from air-conditioning during the summer, and from process heat requirements would be dissipated to the atmosphere by a cooling tower. A cooling-capacity requirement of about 7.5×10^7 Btu/hr would be assumed for the fuel-fabrication plant.

Evaporative loss from the cooling tower would result in the addition of about 216,000 gpd of water to the atmosphere. The large volume of air that would circulate through the tower would result in broad dispersion and good mixing with ambient air.²⁴

4.3-29

SCHEMATIC DIAGRAM FOR DILUTION OF LIQUID RADIOACTIVE
WASTES FROM FUEL-FABRICATION PLANT

Figure 4.3-11

4.3.5.2 Chemical Effluents

4.3.5.2.1 From the Mixed-Oxide Section of the Fuel-Fabrication Plant

Gaseous Effluents

Small amounts of nitrogen oxides would be evolved from laboratory hoods; however, most would be removed with caustic scrubbers.²⁵ After the scrubbing, NO_x (assumed to be NO_2) would be released at a rate of 0.003 g/sec. This effluent would be combined with the total gaseous effluent from the mixed-oxide section of the plant, which would be discharged from a rooftop vent at a rate of 100,000 scfm.

Liquid Effluents

The only chemicals that would be discharged in the liquid effluents from the mixed-oxide section of the fabrication plant would be small amounts obtained from laboratory-sink drains, detergents from floor scrubbing, and solids from the cooling-tower blowdown. These effluents would be combined with the sanitary wastes prior to being released to the environment. The chemical effluents are summarized in Table 4.3-4. The quantities were derived by multiplying the chemical effluents described by Westinghouse²⁵ for the Recycle Fuels Plant by a scale factor of 2.5 to account for differences in plant size and material throughput. The Westinghouse plant has a capacity of 200 metric tons of mixed oxide per year compared with 855 metric tons/year in the mixed-oxide section of the future LMFBR fabrication plant described here. The ratio of these capacities (855/200) is 4.3; however, it is not expected that the size of the plant or the effluents would increase linearly with capacity. Therefore, the smaller scale factor of 2.5 was used. The chemicals would be contained in the liquid effluents from the mixed-oxide section of the plant and would be discharged at a rate of 52,000 gpd (see Figure 4.3-6).

4.3.5.2.2 From the UO_2 Section of the Fuel-Fabrication Plant

Gaseous Effluents

The gaseous effluents would contain about 2000 g of HF and about 6000 g of NH_3 per day. These gases would be mixed with the total gaseous effluent from the UO_2 section of the plant and would be discharged from a rooftop vent at a rate of 100,000 scfm.

Liquid Effluents

Table 4.3-4 summarizes the chemicals in the liquid effluents from the UO_2 section of the fuel-fabrication plant. The chemicals would be contained in liquid effluents

Table 4.3-4

CHEMICALS IN THE LIQUID EFFLUENTS FROM AN LMFBR FUEL-FABRICATION PLANT

Chemical	Release Rate (grams/day)		
	Mixed-oxide Section	Uranium Dioxide Section	Combined Plant
H ₂ SO ₄	147		147
HNO ₃	68		68
HCl	45		45
NaNO ₃	9640		9,640
NaOH	1390		1,390
NH ₄ OH		7.80×10^6	7.80×10^6
Ca(OH) ₂		4.56×10^5	4.56×10^5
CaF ₂		202	202
PO ₄ ³⁻	2040	4,080	6,120
PO ₄ ³⁻ (after degrading)	204	408	612
Total solids	7710	15,400	23,110
PO ₄ ³⁻ (in cooling-tower drift)	110	230	340

discharged from the UO_2 section of the plant at a rate of 185,000 gpd (see Figure 4.3-10).

4.3.5.3 Radioactive Effluents

4.3.5.3.1 From the Mixed-Oxide Section of the Fuel-Fabrication Plant

A small amount of radioactive material will be released in the gaseous and liquid effluents from the mixed-oxide fabrication plant. The amount released will depend on the characteristics of the material being processed, for example, powder particle size, the detailed design of the containment system, and the operating procedures in effect. Assumptions about the design of a plant are required to calculate the amount of radioactive material expected to be released from a particular plant at this time. Where assumptions were necessary, they were made such as to overestimate the expected release.

Gaseous Effluents

The radionuclides in the gaseous effluents from the mixed-oxide fuel-fabrication plant were identified as follows. The isotopic content of uranium and plutonium reprocessed 365 days after reactor discharge was determined by an ORIGEN²⁶ calculation. Fuel of this composition was assumed to decay for 365 days from completion of reprocessing to completion of fabrication in the mixed oxide fabrication plant.* An ORIGEN calculation was performed to determine the amounts and activities of radionuclides being processed in the mixed-oxide section of the fuel-fabrication plant. Assuming a plant confinement factor of 10^{12} and an atmospheric dispersion factor (χ/Q) of 10^{-5} sec/m³, those radionuclides whose concentrations at the site boundary were estimated to be less than $1/10^{12}$ of the 10 CFR 20 limits for the soluble form of the isotope in air in unrestricted areas were excluded from further consideration.

The estimate of the radioactivity of the remaining radionuclides that would be expected to be released in the gaseous effluents from the mixed-oxide section of the fuel-fabrication plant is based on the following assumptions:

- (1) 0.1% (suspension factor = 10^{-3}) of the material processed becomes airborne, passes through the roughing filter in the glove box, and reaches the inlet face of the first HEPA filter, yielding a confinement factor of 10^3 .

*No difference in environmental effects would result if a 30-day time were used here.

- (2) Each of three stages of HEPA filters has an efficiency of 99.9%, providing an overall filter confinement factor of 10^9 and an overall plant confinement factor of 10^{12} . (Confinement factor is defined as the ratio of the material processed to the material released. Transmission factor = $1/\text{confinement factor}$.)

The first assumption, which concerns the amount of airborne material that would reach the first HEPA filter, is based on the following analysis. The fuel-fabrication step most likely to create significant amounts of airborne materials is the grinding of the PuO_2 powder, which is part of the UO_2 - PuO_2 powder-blending operation. The concentration of PuO_2 particles in the glove-box atmosphere was assumed to reach 100 mg/m^3 each time the ball mill is unloaded and reloaded and to remain at that level for the 20 min estimated for the unloading-reloading operation. A concentration of 100 mg/m^3 appears to be the upper limit for suspension of heavy particles in air.²⁷⁻²⁹ (Fog has a concentration of about 10 mg/m^3 .) Of the airborne particles, 90% were assumed to be removed by a roughing filter inside the glove box. The remaining 10% is exhausted through the roughing filter during the 3 to 4 hr required for grinding the powder. (No powder would be released from the ball mill during the grinding.) Assuming 11.3 kg of plutonium per batch of PuO_2 powder and a glove-box volume of 200 ft^3 with a purge rate of 13 cfm, the fraction of the processed material which would become airborne in the glove box would be 10^{-4} . The fraction of the processed material that would pass the roughing filter and reach the inlet side of the first HEPA filter would be 10^{-5} .

Two other operations in the fuel-fabrication process could create airborne-particle concentrations comparable with that described for the PuO_2 -grinding operation, namely, blending the UO_2 and PuO_2 powders and granulating the blended UO_2 - PuO_2 powder to prepare feed material for the automatic pellet press.

The airborne material produced by all of the other steps in the fuel-fabrication process combined (pellet pressing, sintering, centerless grinding, inspection, and fuel-rod loading) should not exceed that produced in the PuO_2 -powder-grinding step.

The upper limit for the fraction of the processed material which might become airborne, pass through the roughing filter, and reach the inlet face of the first HEPA filter should be:

PuO_2 grinding	1×10^{-5}
UO_2 - PuO_2 blending	1×10^{-5}

UO ₂ -PuO ₂ granulating	1×10^{-5}
All other operations	1×10^{-5}
Total	4×10^{-5}

The analysis of the fraction of material processed which becomes airborne, passes the roughing filter, and reaches the first HEPA filter is not exact. It does, however, illustrate the upper limit that might be expected to reach the first filter. It further indicates that the use of a suspension factor of 1×10^{-3} overestimates the amount of material expected to reach the first filter, thus providing conservative estimates of the amount of radioactivity released.

The second assumption is based on the following considerations. In laboratory tests to determine the effectiveness of HEPA filters in removing airborne PuO₂ particles, Ettinger et al.³⁰ found efficiencies of 99.999, 99.998, and 99.86% for the first, second, and third stages, respectively, of HEPA filters in series. The apparent lower efficiency of the third filter was believed to be due to contamination by radon and to poor counting statistics because of the low concentration of radioactivity in the effluent from the third filter.

HEPA filter efficiencies as a function of aerosol aerodynamic diameter are given in Table 4.3-5.³⁰ The minimum efficiency of the first HEPA filter occurs against particles in the 0.43- to 0.65- μ m size range. The minimum efficiency of the second HEPA filter in series occurred against particles smaller than 0.43 μ m.³⁰

The overall transmission factor observed by Ettinger et al.³⁰ for three HEPA filters in series was 2.8×10^{-13} . Therefore, the use of an overall transmission factor of 1×10^{-9} for three stages of HEPA filters in series in the mixed-oxide section of a fuel-fabrication plant appears to be conservative. This conclusion is further substantiated by a recent statement by Ettinger³¹ that a removal of 99.9% for particles of submicron size can be achieved for each filtration stage in a reasonably well designed and tested filter system. Downgrading the efficiency rating of each stage of filters from the experimentally observed efficiencies³⁰ to 99.9% allows operation of a system that has some leaks.

The annual release of plutonium in the gaseous effluent from the mixed-oxide section of a fuel-fabrication plant, determined by the above manner, would be 149 μ g. Table 4.3-6 indicates the radioactive material released annually for each radionuclide; the estimated ground-level concentrations in air at the site boundary are also listed in this table.

Table 4.3-5
HEPA FILTER EFFICIENCY
AS A FUNCTION OF AEROSOL SIZE^a

Aerodynamic Diameter Range (μm)	Mean Efficiency (%)	
	HEPA Filter No. 1	HEPA Filter No. 2
>11	99.99976	99.99974
7.0 to 11	99.99981	99.99688
4.7 to 7.0	99.99988	99.99912
3.3 to 4.7	99.99985	99.99819
2.1 to 3.3	99.99971	99.99970
1.1 to 2.1	99.99901	99.99978
0.65 to 1.1	99.99758	99.99851
0.43 to 0.65	99.99702	99.99812
<0.43	99.99827	99.99728

^aSource: H. J. Ettinger, J. C. Elder, and M. Gonzales, "Performance of Multiple HEPA Filters Against Plutonium Aerosols for Period January 1 through June 30, 1973," Report LA-5349-PR, June 1973.

Table 4.3-6
RADIONUCLIDES RELEASED IN THE GASEOUS EFFLUENTS
FROM AN LMFBR FUEL-FABRICATION PLANT^a

Radionuclide	Annual Release ($\mu\text{Ci}/\text{year}$)			Concentration ^b ($\mu\text{Ci}/\text{m}^3$)	
	Mixed-Oxide Section	Uranium-Dioxide Section	Combined Plant	Wind Frequency = 0.0533 ^c	Wind Frequency = 0.127 ^d
Pu-236	2.58 E-3 ^e		2.58 E-3	1.06 E-16	2.50 E-16
Pu-238	2.88 E1		2.88 E1	1.18 E-12	2.79 E-12
Pu-239	5.91		5.91	2.42 E-13	5.73 E-13
Pu-240	8.07		8.07	3.30 E-13	7.83 E-13
Pu-241	8.34 E2		8.34 E2	3.41 E-11	8.09 E-11
Pu-242	2.20 E-2		2.20 E-2	9.00 E-16	2.13 E-15
U-232	9.12 E-5		9.12 E-5	3.73 E-18	8.85 E-18
U-234	2.17 E-4	1.01 E2	1.01 E2	4.13 E-12	9.80 E-12
U-235		1.25 E1	1.25 E1	5.11 E-13	1.21 E-12
U-236		1.58 E1	1.58 E1	6.46 E-13	1.53 E-12
U-238	2.35 E-4	9.19 E2	9.19 E2	3.76 E-11	8.92 E-11
Th-228	2.40 E-5		2.40 E-5	9.82 E-19	2.33 E-18
Th-231		1.25 E1	1.25 E1	5.11 E-13	1.21 E-12
Th-234		9.19 E2	9.19 E2	3.76 E-11	8.92 E-11
Am-241	1.37		1.37	5.60 E-14	1.33 E-13
Np-237	2.30 E-7		2.30 E-7	9.41 E-21	2.23 E-20
Pa-234		9.19 E2	9.19 E2	3.76 E-11	8.92 E-11

^aThe radionuclides listed for the mixed-oxide section and the UO₂ section would be released from individual rooftop stacks (60 ft above ground level) about 1000 m from the site boundary.

^bEstimated ground-level concentrations in air at the site boundary (1000 m) of an LMFBR fuel-fabrication plant.

^cFor a sector not opposite the prevailing wind direction.

^dFor sectors opposite the prevailing wind direction.

^e2.58 E-3 = 2.58×10^{-3} .

For comparison, the plutonium that would be released in the gaseous effluents of the mixed-oxide section would be about 2.5 times that which would be released in the gaseous effluents of the mixed-oxide plant described by Selby et al.,³² with a plutonium throughput equal to that of the plant described here (450 kg Pu/day instead of 40 kg Pu/day). On the other hand, the radioactive material released annually in the gaseous effluents of the mixed-oxide section of the LMFBR fuel-fabrication plant described here would be about 1.25% of that which has been assumed for the Westinghouse Recycle Fuel Plant³³ at Anderson, South Carolina, when calculated for comparable plutonium throughput.

These radionuclides would be in the form of particulates of PuO_2 , UO_2 , and $(\text{U-Pu})\text{O}_2$. The mass median particle diameter of the PuO_2 powder used in the mixed-oxide plant would be primarily in the range 0.2 to 30 μm ; about 10% would have a size less than 1 μm .³⁴ The UO_2 powder would have a mass median particle diameter less than 1 μm .³⁵ The majority of the particulates released would be expected to have a mass median diameter of about 0.3 μm since this particle size has the greatest fractional penetration through a HEPA filter.³⁶ Actual measurements³⁷ at plutonium-fabrication plants showed radioactive particles downstream from glove box HEPA filters (and just upstream from the main building stack and its HEPA filters) to have an activity median aerodynamic diameter of 1.0 to 4.0 μm .* Westinghouse³⁸ reports that, of the particles downstream from the HEPA filters in their recycle-fuels plant, 74% will have a mass median diameter in the range 0.05 to 0.1 μm and 25% in the range 0.1 to 0.3 μm .

The radionuclides listed in Table 4.3-6 would be contained in the gaseous effluents from the mixed-oxide section of the plant and would be discharged from a rooftop vent in an air flow of 100,000 scfm.

Applicable Experience

None of the routine stack effluent measurements that have been recorded over the years from facilities engaged in plutonium fabrication operations can be translated to precise values for the plutonium confinement factors achieved by those facilities. This statement is true of AEC facilities, such as the Rocky Flats plant, in which the bulk of plutonium experience to date has been accumulated, and it is true of the several commercial plutonium fuel-fabrication facilities that have been placed in operation in recent years.

Routine confinement factors for plutonium alpha particulates have been at least 10^{10} , and unquestionably larger, in facilities with air ventilation systems comparable with

*These numbers may not be representative, because these filters may have been damaged by acid.

the minimum requirements assumed for the model LMFBR fuel fabrication plant¹³ (see Primary Containment in Section 4.3.4.1.3). The reason that the question of "how much larger?" cannot be answered from routine stack effluent monitoring is that these results have been generally below detectable limits and have reached "readable" values only during abnormal occurrences, such as the 1969 fire at the Rocky Flats plant (see discussion in Appendix II.G.1.2).

Facilities of the type under consideration generally sample both for routine and abnormal releases. For abnormal releases, continuous samplers connected to alarms are used in the plant to control occupational exposures. For routine (daily or less frequent) effluent sampling, filter paper is used to collect the sample from the plant stack. Between 1 and 5 cfm of air are diverted from the stack flow through the sample collector. Stack flows typically run between 15,000 and 50,000 cfm or higher.

The concentration of plutonium and other alpha activities in discharged air has been typically so small that verification of day-to-day plant containment performance has been gained from the usually negative finding of "below detectable limits." The sensitivity of equipment available for such routine plant monitoring has increased over the years, but even with the less sensitive equipment of prior years, a negative finding has given assurance that air concentrations at the plant site boundary and beyond have been well below applicable limits. Additional assurance comes from air monitoring on the site and its perimeters.

The following tabulation is typical of both the kind of routine effluent release data available and the impossibility of reconstructing plant confinement factors from it. These are the 1970 "measurements" of alpha activity effluents from the stack of the 308 Fuels Laboratory at the Hanford Engineering Development Laboratory. It is currently a pilot plant for assembly of LMFBR-type fuel (FFTF) and has been in use for over a decade for plutonium fuels development.

	<u>Alpha</u> $\mu\text{Ci/week}$		<u>Alpha</u> $\mu\text{Ci/week}$
January 1970	<26	July 1970	<26
February	<26	August	<26
March	<26	September	<26
April	<26	October ^a	<0.04
May	<26	November	<0.04
June	<26	December	<0.04

^aBegan counting on more sensitive counter in October

The intelligence offered by such data prior to October 1970 was that something less than 1350 μCi ($<1.4 \text{ mCi}$) of alpha activity per year was being released from this facility. From October 1970, data became apparent that considerably less than 1350 $\mu\text{Ci}/\text{year}$ had been released, but the data (to date) still only allows that something less than 2 $\mu\text{Ci}/\text{year}$ escapes from the pilot plant. It has been estimated that the alpha particulate confinement factor of this pilot plant is something greater than 6×10^{10} .

Note that developmental and commercial plutonium fuel-fabrication experience to date has been very small in comparison with projected commercial activities in terms of plant capacities and throughputs. Moreover, the bulk of plutonium processed to date in AEC or commercial facilities has not been characteristic of plutonium to be processed in the future in terms of alpha activity per unit weight (specific activity). Larger proportions of shorter-lived Pu-238, -240, and -242 and Am-241 (from decay of Pu-241) will be present in plutonium generated and recycled for fuel fabrication in the LMFBR fuel cycle than has been the case thus far. Thus, both quantity and specific activity of plutonium particulates to be expected in the model LMFBR fuel-fabrication plant will be larger than those encountered to date.

In summary, alpha particulate effluent data available to date do not confirm the confinement factor of 10^{12} assumed in this Statement but are not inconsistent with that assumption. Containment factors of 10^{10} or greater have been achieved and, in combination with relatively small quantities of plutonium of relatively low specific activity handled in facilities comparable with the model LMFBR fuel-fabrication plant, have maintained effluent concentrations below detectable limits on the routine samples taken. In the interim, the experimental data of Ettinger et al.³⁰ and the considerations discussed in the preceding section provide a reasonable basis for assuming that confinement factors of the order of 10^{12} will, in fact, be achieved.

Liquid Effluents

The radionuclides in the liquid effluents from the mixed-oxide section of the fuel-fabrication plant (Table 4.3-7) were assumed to be the same as those given in Table 4.3-6 for the gaseous effluents. The quantity of each radionuclide that would be released was estimated as follows.

All potentially contaminated liquids were assumed to be collected in retention tanks. These wastes would include cold nonradioactive liquid wastes from the analytical laboratory, water from personnel decontamination, mop water, and water

from the dehumidifier and duct drains. The liquids in the retention tanks would be monitored and released to join the remainder of the plant effluent (sanitary waste and cooling-tower blowdown) only if the level of the radioactivity were within the limits set forth in Table II, column 2, of Appendix B of 10 CFR 20.

Table 4.3-7
RADIONUCLIDES IN THE LIQUID EFFLUENTS
FROM AN LMFBR FUEL-FABRICATION PLANT

Radionuclide	Annual Release ($\mu\text{Ci}/\text{year}$)		
	Mixed-Oxide Section	Uranium Dioxide Section	Combined Plant ^a
Pu-236	2.8 E-1 ^b		2.8 E-1
Pu-238	2.4 E3		2.4 E3
Pu-239	5.0 E2		5.0 E2
Pu-240	6.7 E2		6.7 E2
Pu-241	7.3 E4		7.3 E4
Pu-242	1.8		1.8
U-232	7.6 E-3		7.6 E-3
U-234	1.8 E-2	2.6 E4	2.6 E4 ⁻
U-235		3.2 E3	3.2 E3
U-236		4.1 E3	4.1 E3
U-238	2.0 E-2	2.4 E5	2.4 E5
Th-228	2.0 E-3		2.0 E-3
Th-231		3.2 E3	3.2 E3
Th-234		2.4 E5	2.4 E5
Am-241	1.2 E2		1.2 E2
Np-237	1.9 E-5		1.9 E-5
Pa-234		2.4 E5	2.4 E5

^aContained in liquid effluent from combined plant and discharged at a rate of 237,000 gpd.

^b2.8 E-1 = 2.8×10^{-1} .

Based on an assumed discharge rate of 1000 gpd for potentially contaminated liquids, the permissible concentration of plutonium to yield 1 MPC (MPC = maximum permissible concentration from Table II, column 2, Appendix B of 10 CFR 20) in this effluent was determined as follows:

$$A_1/P_1 + A_2/P_2 + \dots + A_i/P_i = 1,$$

where A_i = activity of the i th radionuclide and P_i = MPC for the i th radionuclide.

The annual release of plutonium in the liquid effluent from the mixed-oxide section determined by the above manner is 1.25×10^4 μ g. Table 4.3-7 indicates the radioactivities that would be released annually for each radionuclide.

Before leaving the mixed-oxide section, the effluent from the retention tanks, discharged at a rate of 1000 gpd, would be diluted with sanitary waste and cooling-tower blowdown, discharged at a rate of about 51,000 gpd, as shown in Figure 4.3-6.

Compared with anticipated industrial practice in fabricating mixed-oxide fuel, the technique described above for handling the potentially contaminated liquid effluents of the mixed-oxide section is the same as that described by Westinghouse³⁹ for their Recycle Fuels Plant at Anderson, South Carolina. The liquid-discharge rate has been adjusted to accommodate the difference between the mixed-oxide throughputs of the two plants.

Mixed-oxide fabrication plants of the future will be designed and operated in accordance with the "as low as practicable" concept. Therefore, the assumption that the sum of the concentrations of radionuclides in the liquid-effluent retention tanks would be equal to 1 MPC should lead to a conservative (maximized) estimate of the environmental effects of these releases.

Since the volume of potentially contaminated liquid effluents from the mixed-oxide section of a future fuel-fabrication plant would be relatively small (a flow rate of only about 1000 gpd), an attractive alternative approach to treatment of these effluents would be to reduce their volume by evaporation and to incorporate the residue in concrete. This procedure would virtually eliminate the radioactive releases in the liquid effluents from the mixed-oxide section of the plant.

Solid Wastes

Solid wastes consisting of a wide assortment of alpha-contaminated materials would be generated in a fuel fabrication plant. These wastes would include such items as paper, cloth, wood, plastic, rubber, glass, ceramic, metal, filters from the ventilation system, and used alpha-contaminated equipment. Contaminated or potentially contaminated solid waste would be processed in a waste-treatment facility within the fuel-fabrication plant. The waste would first be monitored to determine whether or not it contained sufficient quantity of fuel material to be practicably recovered. If so, the uranium and plutonium would be recovered by

incinerating, shredding, washing, and/or leaching. Wastes that contained smaller amounts of fuel material than would be practicable to recover would be incinerated to minimize their volumes and packaged for offsite disposal.⁴⁰

In addition to those described above, solid wastes would result from the solidification (for example, with cement) of liquid wastes.

The shipping and disposal of these solid wastes are described in Sections 4.5.4.1 and 4.6.2.3, respectively. The quantities of these wastes expected to be generated are also discussed there.

4.3.5.3.2 From the UO₂ Section of the Fuel-Fabrication Plant

Gaseous Effluents

The amounts of radioactive material expected in the gaseous effluents from the UO₂ section of a fuel-fabrication plant were derived by multiplying the radioactive releases from operating LWR fuel-fabrication plants by the appropriate factor to account for differences in plant throughput and isotopic composition. Table 4.3-6 gives the annual releases of radionuclides in the gaseous effluents. They would be contained in the gaseous effluents discharged from a rooftop stack at a rate of 100,000 scfm.

Liquid Effluents

The radioactive materials in the liquid effluents from the UO₂ section of a fuel-fabrication plant were derived by multiplying the radioactive releases from operating LWR fuel-fabrication plants by the appropriate factor to account for differences between plant throughputs and between isotopic compositions. Table 4.3-7 gives the annual releases of radionuclides in the liquid effluents.

The radionuclides listed in Table 4.3-7 for the UO₂ section of a fuel-fabrication plant would be contained in the contaminated process liquid waste (Figure 4.3-10), that would be discharged from a lined lagoon at the rate of 84,000 gpd. Before leaving the UO₂ plant at a rate of 84,000 gpd, the contaminated waste would be diluted with the 101,000-gpd flow of liquid from the waste-treatment plant (see Figure 4.3-10).

4.3.6 Environmental Effects of Normal Plant Operation

4.3.6.1 Effects of Thermal Effluents

4.3.6.1.1 On the Aquatic Environment

If a ΔT of 15°F is assumed, at 0.29- and 0.08-cfs flows for the UO_2 - and mixed-oxide-fabrication facilities, respectively, thermal effects would be negligible under reasonable design conditions. For example, dilution of the heat by means of a diffuser pipe in a stream or lake to a ΔT less than 1°F should result in no significant effects of discharge heat on aquatic biota. Combination of the two discharges would require a stream with a minimum flow of 22 cfs to meet Federal guidelines for mixing zones.⁴¹

4.3.6.1.2 On the Terrestrial Environment

There is no expected effect on the terrestrial environment from the thermal effluents of the fuel-fabrication plant.

4.3.6.2 Effects of Chemical Effluents

4.3.6.2.1 Gaseous Effluents

The principal nonradioactive gaseous effluents expected to be released to the atmosphere from routine operations of the fuel-fabrication facility are hydrogen fluoride (HF) and ammonia (NH_3). Hydrogen fluoride is highly toxic to green plants and, in foodstuff, concentrations of >50 to 100 ppm may be toxic to animals, including humans. Cattle and sheep develop fluorosis from these fluoride concentrations in forage. Ammonia generally is regarded as a nutrient for plant growth, but it may exert toxic effects at extremely high concentrations.

Both substances are highly reactive; thus, they would persist for relatively short intervals in their original chemical form. Residence time and secondary reaction products in the atmosphere for HF and NH_3 are not well known.

Atmospheric Dilution of HF and NH_3

Release rate and atmospheric dilution influence initial concentrations of HF and NH_3 in the atmosphere. For routine releases, the source strength is calculated to be 0.023 and 0.093 g/sec for HF and NH_3 , respectively (Table 4.3-8). Atmospheric dilution factors (χ/Q) are calculated from the same formula used to calculate annual average concentration of radioactivity in the atmosphere. The product of dilution factor and source strength is the concentration ($\mu g/m^3$) in air at specified distances from the point of release (Table 4.3-8).

Table 4.3-8

ANNUAL AVERAGE AMBIENT FLUORIDE AND AMMONIA CONCENTRATIONS
FOR ROUTINE RELEASE FROM A FUEL FABRICATION PLANT

Distance (m)	x/Q (sec/m ³)	Fluoride		Ammonia	
		Source Strength ^a (g/sec)	Concentration in Air ($\mu\text{g}/\text{m}^3$)	Source Strength ^b (g/sec)	Concentration in Air ($\mu\text{g}/\text{m}^3$)
1000	1.6×10^{-6}	2.31×10^{-2}	0.037 (0.09) ^c	9.26×10^{-2}	0.05
1609 (1 mile)	7.8×10^{-7}	2.31×10^{-2}	0.018 (0.04)	9.26×10^{-2}	0.07
3218 (2 miles)	2.7×10^{-7}	2.31×10^{-2}	0.006 (0.014)	9.26×10^{-2}	0.02
8045 (5 miles)	6.5×10^{-8}	2.31×10^{-2}	0.002 (0.004)	9.26×10^{-2}	0.006

^aCalculation of source strength:

$$\text{HF} = \frac{2000 \text{ g/day}}{8.64 \times 10^4 \text{ sec/day}} = 2.31 \times 10^{-2} \text{ g/sec.}$$

^bCalculation of source strength:

$$\text{NH}_3 = \frac{8000 \text{ g/day}}{8.64 \times 10^4 \text{ sec/day}} = 9.26 \times 10^{-2} \text{ g/sec.}$$

^cValues in parentheses are maxima in two adjacent sectors in which the wind blows 25% of the time. Concentrations are 2.4 times the annual average.

Because of the extreme toxicity of HF to green plants, and because effects may be manifested at short exposures (several hours to several days) at either continuous or intermittent concentrations ranging from 1 to 50 $\mu\text{g}/\text{m}^3$, potential impact must be considered during severe, but short-term meteorological conditions. Severe conditions that would limit HF dispersion, that is, cause an increase in air concentration, may be related to any one or a combination of the following: atmospheric calms, moderate to extreme atmospheric stability (Pasquill types F and G), and nighttime inversions. An extended period of air stagnation would be the most severe manifestation of these conditions. The frequency of atmospheric calms is usually low ($\sim 5\%$), the frequency of Pasquill type-F stability is 39%, and the frequency of inversion is 25 to 45% of total hours for the continental and coastal United States.⁴² To consider dispersion for all variations of unusual meteorology is impractical, but atmospheric concentration of HF was calculated for a severe condition defined strictly by Pasquill type-F stability and a 5 mph (2.26 m/sec) wind speed. The severe condition, however, does not fully represent possibly poorer circumstances of dispersion related to air entrapment by short-term inversions, fumigation (type-G stability), and persistent stagnation. Figure 4.3-12 illustrates χ/Q values for the severe condition. The χ/Q values at given distances were used to calculate the air concentration of HF and NH_3 (Table 4.3-9) from an average of the centerline and sector-averaged values for the severe condition.

Both NH_3 and HF are highly reactive; thus, they would persist for relatively short intervals in their original chemical form. Residence time and secondary reaction products in the atmosphere for HF and NH_3 are not well known.

Effects of Ammonia on Biota

The calculated air concentrations of NH_3 at 1000 m from the fuel fabrication plant ranged from 0.15 (annual average meteorology, Table 4.3-10) to 12.1 $\mu\text{g}/\text{m}^3$ (severe meteorological condition, Table 4.3-9). Lower concentrations would occur at distances greater than 1000 m. No detrimental effect on vegetation is expected because these concentrations are more than three orders of magnitude below observed concentrations (15 ppm) which caused 50% injury of leaf tissue of tomato plants.⁴³ The effects were manifested after 16 hr of exposure but were not permanent because plant recovery was observed following the exposure. Higher concentrations of NH_3 (250 to 1000 ppm) caused pH changes in the cell sap after 5 min of exposure, and the tomato plants recovered following the treatment. These concentrations are five orders

X/Q FOR SEVERE METEOROLOGY

Figure 4.3-12

Table 4.3-9

HYDROGEN FLUORIDE AND AMMONIA CONCENTRATIONS
FOR SEVERE METEOROLOGICAL CONDITIONS^a

Distance (meters)	x/Q^b (sec/m ³)	Fluoride Concentration in Air ($\mu\text{g}/\text{m}^3$)	Ammonia Concentration in Air ($\mu\text{g}/\text{m}^3$)
1000	1.3 E-4	3.0	12.1
1609 (1 mile)	6.8 E-5	1.6	6.3
3218 (2 miles)	2.5 E-5	0.6	2.3
8045	7.2 E-6	0.2	0.7

^aRoutine release rate, same as in Table 4.3-8.^bBased on an average of centerline and sector averaged values, Figure 4.3-12.

Table 4.3-10

ESTIMATED CONCENTRATION OF FLUORIDE IN VEGETATION

Distance from Source (meters)	Meteorological Condition	Concentration in Air (mg/cm^3) ^a	Concentration in Vegetation (ppm) ^b	
			5-day Exposure	18-day Exposure
1000	Annual Average	0.037	0.27	0.65
	Severe	3.0	22	52
1609 (1 mile)	Annual Average	0.018	0.1	0.3
	Severe	1.6	12	28
8045 (5 miles)	Annual Average	0.002	0.01	0.03
	Severe	0.2	1.5	3.5

^aValues taken from Tables 4.3-8 and 4.3-9.^bCalculated from procedure described in Appendix II.L.

of magnitude greater than those expected from routine releases from the fuel-fabrication plant.

Accumulation of NH_3 in vegetation would not be hazardous because NH_3 is an essential element for plant nutrition and it is regularly supplied to green plants as a fertilizer. Green plants would convert the assimilated NH_3 to organic nitrogen, ammonia acids, etc., effectively utilizing atmospheric NH_3 at concentrations <1 ppm.

Effect of HF on Vegetation

At the expected release rate (0.023 g/sec) from routine operations, and for average meteorological conditions, no detrimental effect on plants is anticipated. Annual average air concentrations of HF (Table 4.3-8) are well below threshold values that will induce effects on green plants (see Figure II.J.1, Appendix II.J). The calculated concentration ($0.037 \mu\text{g}/\text{m}^3$) in air at 1000 m is at least a factor of 5 less than the concentration ($<0.2 \mu\text{g}/\text{m}^3$) at which effects have been observed (Table II.J-2, Appendix II.J). Annual average concentrations of HF in air at distances of 1 to 5 miles from the source decrease by at least another factor of 10. For the annual average maxima, where the prevailing wind blows towards two adjacent sectors for 25% of the time (values in parentheses, Table 4.3-8), the calculated HF concentration ($0.09 \mu\text{g}/\text{m}^3$) at 1000 m is still 50% less than the threshold values at which effects are observed. In fact, concentrations of HF from a routine release occurring under annual average meteorological conditions are at the limits of detectability for most analytical techniques.⁴⁴

Because the type-F stability may occur 39% of the time, and because inversions may occur for 25 to 45% of total hours, it is estimated that the severe meteorology (as defined above) can be expected for 35% of the time. Thus, the total of intermittent type exposure (see Appendix II.J for a discussion on effects from intermittent vs continuous exposure) under the severe meteorology would be 142 days/year. The fraction of total days that any two adjacent sectors will be subjected to the severe meteorology is (142 days $\times$ 0.127 =) 18 days based on the average frequency of prevailing winds. Figure II.J.1 (Appendix II.J) shows that concentrations of 1 to $10 \mu\text{g}/\text{m}^3$ will cause foliar damage for a 10-day exposure and that similar effects will be manifested at concentrations of 0.5 to $5.0 \mu\text{g}/\text{m}^3$ for exposure as long as 100 days. These concentrations and durations of exposures would represent the range of exposure regimes which will induce effects on sensitive plants.

For severe meteorological conditions, the HF concentration in air within one mile of the source approaches or exceeds threshold values at which effects on green plants have been observed. At the 1000-m distance, intermittent exposure to $3.0 \mu\text{g}/\text{m}^3$ for several days could approach the threshold for inducing effects in conifers and fruit trees (Figure II.J-1, Appendix II.J). For longer exposures under type-F conditions, damage to conifers can be expected. Intermittent but additive exposure for as long as five days would approach the threshold for effects on corn and sorghum. At the calculated HF concentrations of $1.6 \mu\text{g}/\text{m}^3$ at one mile, the threshold for effects on conifers is approached for a two-day exposure. No damage to plants would be expected for two-day exposures under severe meteorological conditions at distances greater than one mile.

Figure II.J-1 (Appendix II.J) shows the HF concentration-time dependency for exposures that cause markings on leaves or fruits (chlorotic, necrotic lesions) for particularly sensitive species. Other susceptible species, for which concentration and length of exposure data are insufficient to determine the threshold for effect, are box elder, sweet potato, blueberry, young barley plants, crabgrass, tulip, and grape.

Accumulation of Fluoride by Plants

Toxic concentrations in grass are not expected to occur from short-term exposure under annual average meteorological conditions. A comparison of calculated ambient values from Tables 4.3-8 and 4.3-9 with accumulation curves (Figure II.J-2, Appendix II.J) shows that the threshold concentration in grass (30 ppm) for incipient fluorosis effects in livestock is approached only after 10 days' exposure to severe conditions ($3.0 \mu\text{g}/\text{m}^3$).

The accumulation curves in Figure II.J-2, Appendix II.J, do not permit the estimation of accumulation of fluoride in forage for concentrations <1 to $2 \mu\text{g}/\text{m}^3$ and for exposure periods >10 days. By using the procedure described in Appendix II.L, the concentration of fluoride in forage at 1000 m, one mile, and five miles was calculated for 5- and 18-day intervals of exposure (Table 4.3-10). For exposures under the annual average meteorology, the calculated fluoride concentrations for a five-day exposure are 0.27 and 0.01 ppm at 1000 m and 5 miles, respectively.

Under severe meteorological conditions and for 18 days of exposure, the maximum fluoride accumulation in forage in two adjacent sectors is 52 ppm at 1000 m from the source. This concentration exceeds the fluoride level in forage (30 to 50 ppm) above which the forage is considered unsafe for continuous consumption by livestock.

A concentration of 52 ppm would rarely, if ever, be encountered at 1000 m because 18 days represents the total annual exposure time in any two adjacent sectors. Intermittent exposure followed by losses due to weathering would normally occur; thus the loss rate function would be greater than 0.562, the value calculated according to $e^{-(0.032)(18 \text{ days})}$ for an 18-day period (Appendix II.L).

The maximum calculated concentration of fluoride in forage one mile from the source is 28 ppm. This concentration does not exceed the level considered unsafe in livestock forage. At five miles from the source, the maximum calculated concentration of HF in forage is 3.5 ppm, a concentration equivalent to background values for forage (Appendix II.J).

At a given distance from the source, for example, 1000 m, fluoride accumulation in forage is expected to range from a low of 0.6 ppm to a high of 52 ppm depending on conditions of exposure. Low accumulation for annual average conditions would pose no hazard to livestock. The maximum concentration may exceed the level considered safe for consumption by livestock. No hazard would be expected at the calculated accumulation (28 ppm) beyond one mile from the source.

Summary--Effects from Routine Releases of HF

Damage to vegetation is not expected for routine releases (~ 2.0 kg/day) under average meteorological conditions. However, for severe meteorological conditions, represented by type-F stability, the ambient concentration ($1.6 \mu\text{g}/\text{m}^3$) at one mile approaches the threshold for effect on conifers if the duration of intermittent exposure exceeds five days. The higher concentration at 1000 m ($3.0 \mu\text{g}/\text{m}^3$) approaches the threshold for effects on sorghum and corn. Cumulative damage from intermittent exposures to concentrations $> 1 \mu\text{g}/\text{m}^3$ is anticipated, but reference data are not available for adequately estimating this effect. Forage plants will not accumulate HF to levels that are toxic to livestock for exposures under average meteorological conditions.

For severe meteorological conditions, fluoride accumulation in forage at 1000 m may exceed the 30 to 50 ppm considered the safe level for continuous consumption by livestock.

4.3.6.2.2 Liquid Effluents

Uranium Dioxide Plant

The concentrations of chemical constituents of liquid wastes from the fuel fabrication plant are NH_3 (5410 ppm), Ca^{2+} (351 ppm), PO_4^{3-} (0.58 ppm), and F^- (0.14 ppm). The concentration of F^- is comparable with that in most natural waters of the United States;^{45,46} thus, F^- is expected not to be toxic. Because of the high pH (approximately 9) produced by the high concentrations of Ca^{2+} and NH_3 in the effluents, most of the PO_4^{3-} should precipitate from solution.⁴⁷ Thus, orthophosphate levels in the discharge should be reduced to less than 10 ppb, a value that should meet Federal criteria for phosphate discharges into natural waters⁴⁸ in most localities. Unpolluted waters have relatively low concentrations of NH_3 , usually <0.2 ppm. Ammonia is soluble in water and reacts to form NH_4OH . The toxicity of ammonia to aquatic biota is related to the amount of un-ionized NH_4OH and is a function of pH. A high concentration of NH_4^+ ions at pH 5 to 6 may not be toxic, but the toxicity will increase if the pH increases. Free ammonia at 0.3-ppm concentration is sufficient to kill many freshwater fishes within a matter of days.⁴⁹

If dilution were an acceptable means of NH_3 reduction, a factor of 10^4 to 10^5 would be required to reduce the concentration to satisfactory levels. This high dilution requirement would severely limit the number of acceptable sites. At most sites, discharges of NH_3 would result in unacceptable environmental impacts, both as a result of direct toxicity and as a result of potential eutrophication caused by increased aquatic nitrogen levels. A more satisfactory approach would involve the recycling of NH_3 in the facility or another means of removal before discharge.* Calcium levels should be reduced such that the incremental addition is <5 ppm in areas of naturally occurring soft waters. In hardwater areas, increments of 20 to 40 ppm above natural levels should produce no significant effects.

*Bennett⁵⁰ reported the results of a pilot plant to demonstrate the feasibility of a process to recover uranium and ammonia for recycle and recovery of fluoride as a by-product. The process yielded an effluent containing <1 mg U/liter, <100 mg NH_3 /liter, and <5 mg F/liter. If such a process were used in the UO_2 section of the fuel-fabrication plant, the concentration of NH_3 in the liquid wastes would be reduced by a factor of 100 from the concentrations reported above. Dilution by a river with a minimum flow rate of 600 cfs would reduce the concentration of NH_3 to about 0.2 ppm. If this were the case, no harmful effect on the aquatic biota would be expected. Although such a process is feasible, no major manufacturer of UO_2 fuel uses it to recover NH_3 from the main process effluents.

Mixed-Oxide Facility

The high concentrations of orthophosphate (1.02 ppm) and nitrate (35.5 ppm) will require a combination of treatment and dilution to reduce the concentrations of these substances to acceptable levels. The combined reduction factor for dilution and treatment should be two to three orders of magnitude. An estimated dependable low flow of 0.8 to 8 cfs would be required for the dilution stream using available nutrient removal technology.⁵¹ Other chemical wastes (SO_4^{2-} , Cl^- , Na^+) will be released in small amounts, and no significant impacts on biota are expected. Residual biocide levels are not expected to exceed those previously described for the reactor effluents. Figure 4.3-13 summarizes the available information on chlorine toxicity to aquatic life, mostly freshwater fish. A total residual chlorine level of 0.5 ppm would be lethal to most organisms after exposures of several minutes. Brungs, after a comprehensive review of the available information,⁵² has recommended the following criteria for residual-chlorine releases: (1) in areas receiving wastes treated continuously with chlorine, not to exceed 0.01 mg/liter for the protection of more resistant organisms only or not to exceed 0.002 mg/liter for the protection of most aquatic organisms; and (2) in areas receiving intermittently chlorinated wastes, not to exceed 0.2 mg/liter for a period of 2 hr a day for more resistant species of fish or not to exceed 0.04 mg/liter for a period of 2 hr a day for trout and salmon. If free chlorine persists, more restrictive criteria are warranted. Alternative procedures or substitutes for chlorination should be investigated.

The conclusion is that minimal environmental impact would result if these recommendations are followed.

4.3.6.3 Effects of Radioactive Effluents on Biota Other than Man

Two recent comprehensive reports have been concerned with radioactivity in the environment and with the pathways by which the radioactive materials can reach the biota.^{53,54} No guidelines have been established for desirable limits for radiation exposure to species other than man. However, the general agreement is that the limits established for humans are also conservative for other species.⁵⁵

4.3.6.3.1 Terrestrial Biota

External doses to terrestrial animals other than man are estimated on the basis of gaseous-effluent concentrations and of direct-radiation contributions at the locations where such animals may actually be present. Because the concentrations are

SUMMARY OF RESIDUAL-CHLORINE TOXICITY DATA

Figure 4.3-13

KEY TO FIGURE 4.3-13

Species	Point No.	Effect end point ^a	Reference
Protozoa	1	Lethal	Hale, 1930
Cladoceran	2	Lethal (4 days)	Biesinger, 1971
Scud	3	Safe concentration	Arthur, 1971
	4	Safe concentration	Arthur and Eaton, 1972
Trout fry	5	Lethal (2 days)	Coventry et al., 1935
	6	Lethal (instantly)	Coventry et al., 1935
Brook trout	7	Median mortality (90 min)	Pyle, 1960
	8	Mean survival time 8.7 hr	Dandy, 1967
	9	Mean survival time 14.1 hr	Dandy, 1967
	10	Mean survival time 20.9 hr	Dandy, 1967
	11	Mean survival time 24 hr	Dandy, 1967
	12	67% lethality (4 days)	Dandy, 1967
	13	Depressed activity	Dandy, 1967
	14	7-day TL50	Arthur, 1971
	44	Not found in streams	Tsai, 1971
Brown trout	45	Not found in streams	Tsai, 1971
Fingerling rainbow trout	17	Lethal (4 to 5 hr)	Taylor and James, 1928
Rainbow trout	15	Slight avoidance (10 min)	Sprague and Drury, 1969
	16	Lethal (2 hr)	Taylor and James 1928
	18	96-hr TL50	Basch, 1971
	19	7-day TL50	Merkens, 1958
	20	Lethal (12 days)	Sprague and Drury, 1969
Chinook salmon	21	First death 2.2 hr	Holland et al., 1960
Coho salmon	22	7-day TL50	Arthur, 1971
	23	100% kill (1-2 days)	Holland et al., 1960
	24	Maximum nonlethal	Holland et al., 1960
Pink salmon	25	100% kill (1-2 days)	Holland et al., 1960
	26	Maximum nonlethal	Holland et al., 1960
Fathead minnow	27	TL50 (1 hr)	Arthur, 1972
	28	TL50 (12 hr)	Arthur, 1972
	29	96-hr TL50	Zillich, 1969
	30	7-day TL50	Arthur, 1971
	31	Safe concentration	Arthur and Eaton, 1972
White sucker	32	Lethal (30-60 min)	Fobes, 1971
	33	7-day TL50	Arthur, 1971
Black bullhead	34	96-hr TL50	Arthur, 1971
Largemouth bass	35	7-day TL50	Arthur, 1971
	37	TL50 (1 hr)	Arthur, 1972
	38	TL50 (12 hr)	Arthur, 1972
Smallmouth bass	36	Not found in streams	Tsai, 1971
	39	Median mortality (15 hr)	Pyle, 1960
Yellow perch	40	TL50 (1 hr)	Arthur, 1972
	41	TL50 (12 hr)	Arthur, 1972
	42	7-day TL50	Arthur, 1971
Walleye	43	7-day TL50	Arthur, 1971
Miscellaneous fish	46	Initial kill 15 min	Truchan, 1971
Rainbow trout	47	100% lethal in plant effluent	Michigan Water Resources Commission, 1971
<i>Daphnia magna</i>	48	0 recovery	National Water Quality Lab, 1971

^aTL50: median tolerance limit.

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- Zillich, J. A. 1969b. The toxic effects of the Grandville Wastewater Treatment Plant effluent to the fathead minnow, *Pimephales promelas*, November 17-21, 1969. Michigan Water Resources Commission, Michigan Department of Natural Resources. 9 pp.
- Zillich, J. A. 1969c. The toxicity of the Wyoming Wastewater Treatment Plant effluent to the fathead minnow, December 8-12, 1969. Michigan Water Resources Commission, Michigan Department of Natural Resources. 12 pp.

primarily used in the calculations for dose to man, they are discussed in that section. The external doses to animals at the same locations approximate those to man.

A conservative estimate can be made for the ingestion dose to a terrestrial animal such as a muskrat by maximizing the conditions that might contribute to radionuclide intake. Such a situation might exist if a muskrat derived all of its diet from aquatic vegetation growing in the plant discharge water before release to the river. The estimated internal dose is 1.2×10^3 millirads/year of intake. Animals with a strictly terrestrial diet would receive a lower foodchain dose from atmospheric releases because the bioaccumulation factors for terrestrial radionuclide transfers are lower.

4.3.6.3.2 Aquatic Biota

Doses to aquatic plants, to invertebrates, and to fish that live in the river below the discharge region were calculated to be 4.4×10^2 , 4.7×10^1 , and 4.5 millirads/year, respectively (Table 4.3-11). These doses are due to water intake and ingestion by organisms that live in a river (600 cfs) which receives the liquid effluent of a fuel-fabrication plant. The discharge-region concentrations were those listed in Tables 4.3-11 and 4.3-12, and the organisms were assumed to spend all year in water having maximum concentrations of discharged radionuclides. All calculations were completed by use of adaptations of standard models and procedures for estimation of radiation doses.⁵⁶ However, the doses are very conservative, since the possibility is highly unlikely that any of the mobile life forms will spend a significant portion of their life-span in the discharge region that has maximum concentration of radioactivity. Both radioactive decay and additional dilution would reduce the dose at other points in the river.

Most of the estimated dose (78%) to fish is due to U-238. The uranium radionuclides are also the principal contributors of the dose to plants (92%) and invertebrates (86%).

The literature that relates to radiation effects on organisms is extensive, but very few studies have been conducted to determine the effects of continuous low-level exposure to radiation from ingested radionuclides on natural aquatic or terrestrial populations. The most recent and pertinent studies point out that, whereas the existence of extremely radiosensitive biota is possible and increased radiosensitivity in organisms may result from environmental interactions, no biota have yet been discovered which show a sensitivity to radiation exposures as low as those anticipated in the area that will surround an LMFBR fuel-fabrication plant. The BEIR

Table 4.3-11
PREDICTED INTERNAL DOSES TO AQUATIC BIOTA^a FROM THE
LIQUID EFFLUENTS OF A FUEL-FABRICATION PLANT

Radionuclide	Concentration ($\mu\text{Ci}/\text{ml}$)	Annual Dose (millirads)		
		Aquatic Plants	Invertebrates	Fish
U-232	$1.4 \text{ E-}17^b$	$7.3 \text{ E-}05$	$7.3 \text{ E-}06$	$7.3 \text{ E-}07$
U-234	$4.9 \text{ E-}11$	$4.5 \text{ E+}01$	$4.5 \text{ E+}00$	$4.5 \text{ E-}01$
U-235	$6.0 \text{ E-}12$	$5.2 \text{ E+}00$	$5.2 \text{ E-}01$	$5.2 \text{ E-}02$
U-236	$7.6 \text{ E-}12$	$6.7 \text{ E+}00$	$6.7 \text{ E-}01$	$6.7 \text{ E-}02$
U-238	$4.4 \text{ E-}10$	$3.5 \text{ E+}02$	$3.5 \text{ E+}01$	$3.5 \text{ E+}00$
Th-228	$3.6 \text{ E-}18$	$2.3 \text{ E-}05$	$7.7 \text{ E-}06$	$4.6 \text{ E-}07$
Th-231	$6.0 \text{ E-}12$	$3.0 \text{ E-}02$	$1.0 \text{ E-}02$	$6.1 \text{ E-}04$
Th-234	$4.4 \text{ E-}10$	$1.1 \text{ E+}01$	$3.7 \text{ E+}00$	$2.2 \text{ E-}01$
Pa-234	$4.4 \text{ E-}10$	$1.0 \text{ E+}01$	$1.0 \text{ E+}00$	$1.0 \text{ E-}01$
Pu-236	$5.1 \text{ E-}16$	$1.1 \text{ E-}03$	$3.2 \text{ E-}04$	$1.1 \text{ E-}05$
Pu-238	$4.4 \text{ E-}12$	$1.6 \text{ E+}00$	$4.7 \text{ E-}01$	$1.6 \text{ E-}02$
Pu-239	$9.3 \text{ E-}13$	$3.2 \text{ E-}01$	$9.2 \text{ E-}02$	$3.2 \text{ E-}03$
Pu-240	$1.3 \text{ E-}12$	$4.5 \text{ E-}01$	$1.3 \text{ E-}01$	$4.5 \text{ E-}03$
Pu-241	$1.4 \text{ E-}10$	$2.1 \text{ E+}00$	$6.0 \text{ E-}01$	$2.1 \text{ E-}02$
Pu-242	$3.4 \text{ E-}15$	$1.1 \text{ E-}03$	$3.2 \text{ E-}04$	$1.1 \text{ E-}05$
Am-241	$2.2 \text{ E-}13$	$1.2 \text{ E+}00$	$2.3 \text{ E-}01$	$5.9 \text{ E-}03$
Np-237	$3.5 \text{ E-}20$	$3.2 \text{ E-}08$	$1.3 \text{ E-}08$	$3.2 \text{ E-}10$
Total Dose		$4.4 \text{ E+}02$	$4.7 \text{ E+}01$	$4.5 \text{ E+}00$

^aBiota living in a river with an average annual flow of 600 cfs (5.4×10^{14} ml/year).

^b $1.4 \text{ E-}17 = 1.4 \times 10^{-17}$.

Table 4.3-12
CALCULATED RELEASES OF RADIOACTIVE MATERIALS IN LIQUID
EFFLUENTS FROM AN LMFBR FUEL-FABRICATION PLANT

Radionuclide	Average Annual Release ($\mu\text{Ci}/\text{year}$)		Combined Uranium and Mixed Oxide	
	Uranium Oxide Fuel	Mixed Oxide Fuel	Average Annual Concentration at Point of Discharge ($\mu\text{Ci}/\text{ml}$) ^a	Average Annual Concentration in River ($\mu\text{Ci}/\text{ml}$) ^b
U-232	c	7.6 E-3	2.8 E-14	1.4 E-17
U-234	2.6 E4 ^d	1.8 E-2	9.7 E-8	4.9 E-11
U-235	3.2 E3	c	1.2 E-8	6.0 E-12
U-236	4.1 E3	c	1.5 E-8	7.6 E-12
U-238	2.4 E5	2.0 E-2	8.9 E-7	4.4 E-10
Th-228	c	2.0 E-3	7.4 E-15	3.6 E-18
Th-231	3.2 E3	c	1.2 E-8	6.0 E-12
Th-234	2.4 E5	c	8.9 E-7	4.4 E-10
Pa-234	2.4 E5	c	8.9 E-7	4.4 E-10
Pu-236	c	2.8 E-1	1.0 E-12	5.1 E-16
Pu-238	c	2.4 E3	8.9 E-9	4.4 E-12
Pu-239	c	5.0 E2	1.8 E-9	9.3 E-13
Pu-240	c	6.7 E2	2.5 E-9	1.3 E-12
Pu-241	c	7.3 E4	2.7 E-7	1.4 E-10
Pu-242	c	1.8	6.7 E-12	3.4 E-15
Am-241	c	1.2 E2	4.5 E-10	2.2 E-13
Np-237	c	1.9 E-5	7.1 E-17	3.5 E-20

^aThe average annual concentration is computed from dilution of the combined plant release with 2.69×10^{11} ml/year.

^bThe liquid discharges are completely mixed in a river with a flow rate of 5.4×10^{14} ml/year.

^cNo entry in the table indicates that this radionuclide is not released.

^d2.6 E4 = 2.6×10^4 .

report⁵⁷ states, in summary, that evidence to date indicates that no other living organisms are very much more radiosensitive than man. Therefore, no detectable radiological impact is expected in the aquatic biota or in terrestrial mammals as a result of the quantity of radionuclides to be released into aquatic habitats or into the air by the fuel-fabrication plant.

4.3.6.4 Effects of Radioactive Effluents on Man

The estimated total-body and critical-organ (reference organ receiving the greatest dose) doses to adult individuals at locations of maximum exposure to direct radiation and to the gaseous and liquid effluents from the LMFBR fuel-fabrication plant are summarized in Table 4.3-13. Table 4.3-14 gives the estimated population doses, for a 50-mile radius, from direct radiation and from exposure to the gaseous and liquid effluents released by the plant.

4.3.6.4.1 Gaseous Effluents

The gaseous effluents will be discharged through rooftop stacks located about 20 m above ground level. Plumes released at a rooftop level can be caught in the turbulent wake of the building or other nearby buildings and can be mixed downward (aerodynamic downwash). For this reason, the effective stack height used in calculations of the atmospheric dispersion of radioactive effluents was selected to be 10 m instead of 20 m, which is the approximate roof height of the building. A deposition velocity of 1 cm/sec was assumed in the atmospheric dispersion calculations for each of the nuclides.

Estimated Concentrations in Air at Ground Level and Rates of Deposition of Particulates on the Ground

Estimates of the ratio of ground-level air concentration of a nuclide to its release rate for typical meteorological conditions are listed in Table 4.3-15. These values (χ/Q) were estimated for a release height of 10 m and deposition velocity of 1 cm/sec. The second column lists χ/Q values from the site boundary (1000 m) out to 80,000 m in a 22.5° sector not opposite the prevailing wind direction (i.e., a frequency prevalence of 0.0533, see Appendix II.I). Values of χ/Q for the sectors opposite the prevailing wind directions are listed in the third column (i.e., a frequency prevalence of 0.127).

Immersion

The estimated maximum annual total-body dose from continuous immersion in the gaseous effluents, 1.3×10^{-7} millirem, occurs at the site boundary; the estimated

Table 4.3-13
SUMMARY OF THE ESTIMATED DOSES TO ADULTS
AT LOCATIONS OF MAXIMUM EXPOSURE

Exposure Pathway	Location	Dose (millirem)		
		Total Body	Bone	Lungs
<u>Gaseous Effluents</u>				
Immersion	Boundary ^a	1.3 E-7 ^b	1.3 E-7	1.3 E-7
Contaminated Ground Surface	Boundary	2.5 E-3	2.5 E-3	2.5 E-3
Inhalation	Boundary	6.9 E-3	2.7 E-1	4.1 E-2
Ingestion (terrestrial food chain)	Boundary	3.3 E-2	5.4 E-1	3.3 E-2
<u>Liquid Effluents</u>				
Drinking Water	Along River	1.0 E-2	1.7 E-1	1.0 E-2
Swimming and Shore Use	Along River	3.6 E-3	3.6 E-3	3.6 E-3
Ingestion (aquatic food chain)	Along River	1.7 E-3	2.8 E-2	1.7 E-3
Ingestion (irrigation)	Along River	8.9 E-4	1.5 E-2	8.9 E-4
<u>Direct Irradiation</u>				
From the Plant		Negligible	Negligible	Negligible
Total		5.9 E-2 ^c	1.0	9.3 E-2

^aThe plant boundary would be located 1000 m from the stack.

^b1.3 E-7 = 1.3×10^{-7} .

^cThe estimated total-body dose to a person involved in crop production utilizing irrigation would be increased from 0.059 to 0.098 millirem/year as a result of exposure to the contaminated ground surface.

Table 4.3-14

SUMMARY OF THE ESTIMATED POPULATION DOSES
FROM ALL PATHWAYS^a

Exposure Pathway	Population Dose (man-rem)		
	Total Body	Bone	Lungs
<u>Gaseous Effluents</u>			
Immersion	2.3 E-8 ^b	2.3 E-8	2.3 E-8
Contaminated Ground Surface	2.5 E-3	2.5 E-3	2.5 E-3
Inhalation	6.7 E-3	2.6 E-1	3.9 E-2
Ingestion (terrestrial food chain)	3.2 E-2	5.2 E-1	3.2 E-2
<u>Liquid Effluents</u>			
Drinking Water	1.0	1.7 E+1	1.0
Swimming and Shore Use	3.6 E-2	3.6 E-2	3.6 E-2
Ingestion (aquatic food chain)	2.8 E-2	4.6 E-1	2.8 E-2
Ingestion (irrigation)	6.3 E-2	1.0 E-2	6.3 E-2
<u>Direct Irradiation</u>			
From the Plant	Negligible	Negligible	Negligible
Total	1.2	19	1.2

^aPer year of operation of an LMFBF fuel fabrication plant.

^b2.3 E-8 = 2.3×10^{-8} .

Table 4.3-15

ESTIMATED GROUND-LEVEL χ/Q VALUES FOR TYPICAL METEOROLOGY
(FOR A RELEASE HEIGHT OF 10 m AND A DEPOSITION VELOCITY OF 1 cm/sec)

Distance from Stack (meters)	χ/Q (sec/m ³)	
	Wind Frequency = 0.0533 ^a	Wind Frequency = 0.127 ^b
1,000	1.29 E-6 ^c	3.06 E-6
1,500	6.17 E-7	1.47 E-6
2,000	3.47 E-7	8.26 E-7
3,000	1.63 E-7	3.87 E-7
4,000	9.35 E-8	2.22 E-7
5,000	6.05 E-8	1.44 E-7
10,000	1.58 E-8	3.76 E-8
15,000	7.18 E-9	1.71 E-8
20,000	4.05 E-9	9.65 E-9
40,000	1.01 E-9	2.40 E-9
60,000	4.22 E-10	1.00 E-9
80,000	2.23 E-10	5.30 E-10

^aFor a sector not opposite the prevailing wind direction.

^bFor sectors opposite the prevailing wind direction.

^c1.29 E-6 = 1.29×10^{-6} .

average annual total-body dose from this pathway is 2.3×10^{-11} millirem and occurs at a distance of 16,000 m (about 10 miles) from the plant. Radionuclides contributing to the immersion dose include Pa-234 (93%), Th-234 (5%), U-235 (1%), and Th-231 (1%). The estimated annual population dose for this exposure pathway is 2.3×10^{-8} man-rem.

Ground Contamination

For direct exposure to the contaminated ground surface, the estimated maximum annual individual dose of 2.5×10^{-3} millirem occurs at the site boundary; the estimated average annual individual dose is 2.5×10^{-6} millirem at 30,000 m (approximately 19 miles). The radionuclides that contribute to this exposure pathway include U-238 (69%), U-235 (16%), U-234 (9%), Pu-238 (2%), Pu-240 (1%), and Am-241 (1%). The estimated annual population dose for this exposure pathway is 2.5×10^{-3} man-rem.

Inhalation

The estimated maximum annual individual total-body dose of 6.9×10^{-3} millirem due to inhalation of contaminated air occurs at the boundary. The estimated average individual total-body dose from this pathway is 6.7×10^{-6} millirem; this occurs at 30,000 m from the plant. Table 4.3-16 summarizes the maximum and average individual total-body and organ doses and the percentage contributions from the principal radionuclides. The estimated population dose for this exposure pathway is 6.7×10^{-3} man-rem/year of plant operation.

Ingestion (Terrestrial Food Chain)

For the ingestion pathway, the same assumptions are used as discussed in Appendix II.I.

For most but not all situations, the model used to determine the dose from the terrestrial food-chain pathway is probably conservative.⁵⁸ Each individual's entire food supply is assumed to be derived from the land that immediately surrounds his location. Radioactivity transfer from other areas was neglected. The entire food needs of cattle were derived from the pasture, and the only loss mechanisms included were radioactive decay and movement to below the root zone. Some nonconservative simplifying assumptions were made. For example, cattle excretions were not returned to the pasture and radioactivity uptake by cattle from inhalation and from water consumption was neglected. The only feedback loop modeled explicitly is from pasture soil to pasture grass. The nonconservative

Table 4.3-16

SUMMARY OF THE ESTIMATED CONTRIBUTIONS TO THE INDIVIDUAL
DOSES RESULTING FROM INHALATION

Organ of Reference	Annual Dose (millirems)		Contribution to Dose (%) ^c										
	Maximum Individual ^a	Average Individual ^b	Th-234	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Am-241	
Total Body	6.9 E-3 ^d	6.7 E-6		1				11	42	9	12	21	2
Bone	2.7 E-1	2.6 E-4		1				5	43	10	14	27	
Thyroid	6.9 E-3	6.7 E-6		1				11	42	9	12	21	2
Lung	4.1 E-2	3.9 E-5		9	1	1	74	9	2	2			
Liver	3.0 E-2	2.9 E-5						2	54	12	17	12	2
Kidney	3.0 E-2	2.9 E-5		1				9	41	9	13	23	3
GI Tract	5.4 E-5	5.4 E-8	61	3	1	1	31	1				1	
Muscle	6.9 E-3	6.7 E-6		1				11	42	9	12	21	2
Spleen	6.9 E-3	6.7 E-6		1				11	42	9	12	21	2
Testes	6.9 E-3	6.7 E-6		1				11	42	9	12	21	2
Ovaries	6.9 E-3	6.7 E-6		1				11	42	9	12	21	2

^aThe maximum individual would be located at the plant boundary, 1000 m from the stack.^bThe average individual would be located 30,000 m (19 miles) from the stack.^cContributions to annual dose not less than 1%.^d6.9 E-3 = 6.9×10^{-3} .

assumptions were made either because detailed studies on selected radionuclides demonstrated a pathway was of second-order importance or because inclusion of the pathway would require more elaborate modeling, for which transfer coefficients are unavailable for some radionuclides.

The estimated maximum individual annual total-body dose of 3.3×10^{-2} millirem occurs at the site boundary. The estimated average dose for this pathway is 3.2×10^{-5} millirem and occurs at 30,000 m. Table 4.3-17 summarizes the maximum and average individual total-body and organ doses and the percentage contributions from the principal radionuclides. For this exposure pathway the estimated population dose is 3.2×10^{-2} man-rem/year of plant operation.

4.3.6.4.2 Liquid Effluents

Drinking Water

Drinking water is assumed to have the concentration of radionuclides given in Table 4.3-11, which are the liquid effluents diluted by a factor of 2000 (see Section 4.3.5). No further reduction in concentration is credited, although normal water-purification processes would cause a reduction. For an average consumption of 1.2 liters/day, the estimated total-body and organ doses and the percentage contributions from the principal radionuclides are presented in Table 4.3-18. If 10% of the population obtains drinking water from the river, the estimated annual dose of 1.0 man-rem could be received by the population.

Swimming and Shore Use

For the case of swimming, an individual is assumed to spend 1% of the year swimming submerged in the diluted effluent waters. The annual total-body dose from swimming is 2.6×10^{-4} millirem. The radionuclide contributing most (over 99%) to the exposure is Pa-234.

If an individual were to spend 1% of the year engaged in shore activities, an estimated total-body dose of 3.3×10^{-3} millirem could be received annually. Radionuclides that contribute to this dose include U-238 (72%), U-235 (16%), U-234 (10%), and U-236 (2%). If 1% of the population within 50 miles is assumed to utilize the river for recreational purposes, the annual population doses are estimated to be 0.0026 man-rem from swimming and 0.033 man-rem from shore use.

Table 4.3-17

SUMMARY OF THE ESTIMATED CONTRIBUTIONS TO THE INDIVIDUAL
DOSES RESULTING FROM INGESTION OF FOOD

Organ of Reference	Annual Dose (millirems)		Contribution to Dose (%) ^c			
	Maximum Individual ^a	Average Individual ^b	Th-234	U-234	U-238	Am-241
Total body	0.033	3.2 E-5 ^d		8	62	30
Bone	0.54	5.2 E-4		8	64	28
Thyroid	0.033	3.2 E-5		8	62	30
Lung	0.033	3.2 E-5		8	62	30
Liver	0.075	7.3 E-5		3	27	69
Kidney	0.16	1.6 E-4		6	48	45
GI Tract	0.25	2.4 E-4	85	1	8	5
Muscle	0.033	3.2 E-5		8	62	30
Spleen	0.033	3.2 E-5		8	62	30
Testes	0.033	3.2 E-5		8	62	30
Ovaries	0.033	3.2 E-5		8	62	30

^aThe maximum individual would be located at the plant boundary, 1000 m from the stack.

^bThe average individual would be located 30,000 m (19 miles) from the stack.

^cContributions to annual dose not less than 1%.

^d3.2 E-5 = 3.2×10^{-5} .

Table 4.3-18

SUMMARY OF THE ESTIMATED CONTRIBUTIONS TO THE INDIVIDUAL
DOSES RESULTING FROM INGESTION OF DRINKING WATER

Organ of Reference	Annual Dose to Individual (millirems)	Contribution to Dose (%) ^a						
		Th-234	Pa-234	U-234	U-235	U-236	U-238	Pu-238 Pu-241
Total body	0.010			11	1	2	86	
Bone	0.17			10	1	2	85	1 1
Thyroid	0.010			11	1	2	86	
Lung	0.010			11	1	2	86	
Liver	0.010			11	1	2	83	2
Kidney	0.039			11	1	2	85	
GI Tract	0.031	59	3	5	1	1	31	
Muscle	0.010			11	1	2	86	
Spleen	0.010			11	1	2	86	
Testes	0.010			11	1	2	86	
Ovaries	0.010			11	1	2	86	

^aContributions to annual dose not less than 1%.Ingestion (Aquatic Food Chain)

All food obtained from the aquatic system are assumed to be freshwater species. Whether any edible species will be harvested for human use is not known; however, the assumption is made that an individual consumes 20 g of fish per day (about 1 lb per month). The total-body dose from ingestion of fish is 1.7×10^{-3} millirem/year of intake; Table 4.3-19 summarizes the total-body and organ doses and the principal contributing radionuclides for this pathway. An estimate of the population dose resulting from fish ingestion can be obtained in the following manner. A conservative value for fish production in a natural river was estimated to be 55 g/m² annually as reported by Chapman.⁵⁹ If the 600-cfs river is assumed to be 6 ft deep and 100 ft wide and have a harvest length of 50 miles, a yield of 1.3×10^8 g of fish per year is potentially available for human consumption. Ingestion of the entire yield of fish from the river will result in an annual population dose to the total body of 2.8×10^{-2} man-rem.

Table 4.3-19

SUMMARY OF THE ESTIMATED CONTRIBUTIONS TO THE INDIVIDUAL
DOSES RESULTING FROM INGESTION OF FISH

Organ of Reference	Annual Dose to Individual (millirems)	Contribution to Dose (%) ^a						
		Th-234	Pa-234	U-234	U-235	U-236	U-238	Pu-238 Am-241
Total Body	0.0017			11	1	2	86	
Bone	0.028			10	1	2	86	
Thyroid	0.0017			11	1	2	86	
Lung	0.0017			11	1	2	86	
Liver	0.0017			11	1	2	84	1 1
Kidney	0.0065			11	1	2	86	
GI Tract	0.011	82	2	2			14	
Muscle	0.0017			11	1	2	86	
Spleen	0.0017			11	1	2	86	
Testes	0.0017			11	1	2	86	
Ovaries	0.0017			11	1	2	86	

^aContributions to annual dose not less than 1%.

Irrigation

Two exposure pathways occur for irrigation: direct exposure and the terrestrial food chain.

The assumptions are made that 76 cm (30 in.) of water is applied to the land surface annually by use of the contaminated river as the sole source of irrigation waters and that the area irrigated is small enough that the river flow is not altered appreciably (10% of the river flow). For direct exposure 25% of the time to the irrigated land surface, the estimated annual total-body dose is 0.039 millirem. The radionuclide contributions to this dose are Pa-234 (98%), Th-234 (1%), and U-238 (1%).

For the food-chain pathway, the assumption is made that all of an individual's food crop intakes are produced on the irrigated land. The exposure resulting from ingestion of irrigated crops considers only the radionuclides added from the irrigation water (radionuclides deposited on crops from atmospheric releases are considered in Section 4.3.6.4.1). The only loss mechanisms included are radioactive decay and movement to below the root zone. The annual total-body dose from the irrigation-food-chain pathway is 8.9×10^{-4} millirem. The radionuclides that contribute to this dose include U-238 (76%), Am-241 (12%), U-234 (10%), and U-236 (1%). The organ that receives the greatest dose from this pathway is bone with an annual

dose of 1.5×10^{-2} millirem resulting from U-238 (78%), Am-241 (11%), U-234 (10%), and U-236 (1%). The land area assumed to be required for crop production per person per year is 1000 m².⁵⁸ The river carrying liquid effluents of the fabrication plant could potentially irrigate a land area large enough to support 70,000 persons (assuming that 10% of the river flow is used). Under these conditions, the calculated annual population dose to the total body is 0.063 man-rem.

4.3.6.4.3 Direct Radiation

From the Plant

The LMFBF fuel-fabrication plant is designed so that direct radiation at the boundary which results from storage or operation will be much less than 0.01 millirem/year; therefore, this pathway was not investigated further.

4.3.6.4.4 Assessment of Estimated Doses

Radionuclide releases from normal operation of the LMFBF fuel-fabrication plant will result in low radiation exposures to individuals and to the population. The estimated maximum individual adult total-body dose resulting from all exposure pathways is 0.059 millirem/year and the estimated population total-body dose is 1.2 man-rem/year. Likewise, organ doses are low; the maximum adult organ (bone) dose is 1.0 millirem/year. A summary of the radionuclide contributions to the maximum total-body dose of an individual is presented in Table 4.3-20.

The estimated total-body dose to a person involved in crop production utilizing irrigation will increase from 0.059 to 0.098 millirem/year as a result of exposure to the contaminated ground surface. Since the number of such individuals is small compared with the total population, the population dose will remain essentially unchanged.

The maximum individual total-body dose resulting from all exposure pathways for the effluents of the fuel-fabrication plant (0.059 millirem/year) is approximately the same as the average dose (0.1 millirem/year) resulting from television viewing.⁶⁰ The annual population dose from the fabrication plant of 1.2 man-rem can be compared to the expected annual population dose of 100 man-rem from television viewing. No discernible adverse radiological effects are expected to occur to an individual or to the population as a result of normal operation of the LMFBF fuel-fabrication plant.

Table 4.3-20

CALCULATED PERCENT OF THE INDIVIDUAL TOTAL-BODY DOSE
(FROM THE EFFLUENTS FROM NORMAL OPERATION OF AN LMFBR FUEL-FABRICATION PLANT)

Pathway	Contribution of Radionuclide (%) ^a								Total
	U-234	U-235	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Am-241	
Immersion									
Ground		1	3						4
Inhalation			1	5	1	1	2		10
Ingestion (terrestrial)	4		35					17	56
Drinking Water	2		15						17
Swimming and Shore Use	1	1	4						6
Ingestion (aquatic)			2						2
Ingestion (irrigation)			1						1
Total	7	2	61	5	1	1	2	17	96

^aContributions to annual dose not less than 1%.

4.3.7 Effluents and Environmental Measurement and Monitoring Programs

The requirements for environmental monitoring and surveys at a nuclear facility can be divided into four stages: (1) a preoperational survey, (2) routine monitoring during operations, (3) design and organization of procedures to be used for emergency surveys in case of accidents, and (4) postoperational surveys.

4.3.7.1 Aquatic Monitoring

A discussion on biological studies and ecological monitoring has been provided in Section 4.2.6.1. Since the needs and objectives of such programs are the same for the fuel-fabrication (and fuel-reprocessing) facilities as for the reactor (although the scope may be considerably smaller than for the reactor), the reader is referred to the reactor discussion for information on this subject.

4.3.7.2 Terrestrial Monitoring

(See comments under Section 4.3.7.1.)

4.3.7.3 Radiological Monitoring

4.3.7.3.1 Preoperational Survey

A preoperational radiological survey should be undertaken at least one year prior to operation of the facility. Table 4.3-21 outlines the type of information to be obtained in the preoperational radiological survey.

Table 4.3-21
REQUIREMENTS FOR ROUTINE RADIOLOGICAL MONITORING AT LMFBR FUEL-FABRICATION PLANTS

Sample type	Approximate location and number of sampling points	Collection frequency	Type and frequency of analysis ^a
Sources of release	NOTE: It is assumed that this is covered by the in-plant system. However, each source of release should be monitored to provide information on gross alpha, beta and gamma activity, with periodic analyses made of radionuclide composition.		
Air particulates	Approximately 10 sampling points within a 10-mile radius. Sampling locations should be concentrated in the directions estimated to have the highest off-site concentrations and in the direction estimated to give rise to the greatest contribution to population dose. One to three sampling station should be located in the least prevalent wind direction at a distance of more than 20 miles	Continuous collection (filter)	Gross α , β , and γ activity weekly; recount samples after one week for long-lived α activity. After measurement the filters should be combined and analyzed each four weeks for TRU's.
Gamma radiation dose rate	Same as for air particulates	Continuous	Collect TLD's monthly.
Precipitation	Same as for air particulates	Continuous proportional sampling	Gross α , β , and γ activity monthly
Surface water	One upstream and one far enough downstream to ensure that complete mixing has occurred	Monthly	Gross α , β , and γ activity monthly
Groundwater	1 or 2 samples from zones most likely to be affected	Annual	Gross α , β , and γ activity
Drinking water			
a. Individual wells	All wells within a 10-mile radius	Quarterly	Same as groundwater
b. Municipal wells	All wells within a 50-mile radius	Continuous proportional sampling	Gross α , β , and γ activity (quarterly)
c. Municipal surface water supplies	All supplies within 10 miles and all supplies within 100 miles downstream from the point of liquid releases	Continuous proportional sampling	Gross α , β , and γ activity (quarterly)
Sediment, benthic organisms, and aquatic plants in surface waters	One sampling point upstream and one immediately downstream from the point of discharge; if the stream is impounded a further sampling location should be in the impounded region	Semiannually	Gross α , β , and γ activity
Fish and shellfish	One sampling point upstream and one in the immediate vicinity of the discharge	Semiannually	Gross α , β , and γ activity of edible portions

Table 4.3-21 (continued)

Sample type	Approximate location and number of sampling points	Collection frequency	Type and frequency of analysis
Milk	One sampling point at the nearest off-site farm having a dairy cow in the prevailing downwind direction and one from a local dairy representative of the milkshed for the area	Monthly	Gross α , β , and γ activity
Fruits and vegetables	One sample of each principal foodstuff grown near the point of estimated maximum annual ground concentration from stack releases; one sample from any area irrigated with water into which liquid wastes from the plant have been discharged; one sample of each of the same foodstuffs grown at a distance of more than 20 miles in the least prevalent wind direction	Annually (at harvest)	Gross α , β , and γ activity
Soil	Same as for fruits and vegetables		Same as for fruits and vegetables plus plutonium analysis
Meat and poultry	Meat, poultry, and eggs from animals fed on crops within 10 miles of the plant in the direction of prevailing winds or where drinking water for the animals is supplied from a downstream source	Annually during the grazing season	Same as for fruits and vegetables

Discharges of radioactive effluents from an LMFBF fuel fabrication plant are expected to be in the form of gaseous effluents to the atmosphere and liquid effluents to surface waters. Thus, information on the groundwater system would likely be of secondary importance.

Because the information gathered in the preoperational survey will be graded according to contribution of specific pathways to the overall impact, much more information is required in the review than will be accumulated during subsequent operation. Data must be obtained also on background levels of radiation dose.

4.3.7.3.2 Operational Program

Environmental monitoring of radioactivity in the vicinity of nuclear facilities operating under license issued by the AEC is required by the Code of Federal Regulations.⁶¹ The objectives and principles of a radiological monitoring program are outlined by the International Commission on Radiological Protection.⁶² Guidance for the design of acceptable monitoring programs is provided by the USAEC⁶³ and the U.S. Environmental Protection Agency.⁶⁴ Environmental monitoring programs in operation at major USAEC contractor sites are described in a report compiled by the USAEC.⁶⁵

The major objective of routine environmental radiological monitoring during operations is to provide information that allows an estimate to be made of the radiation doses received by the general public as a result of plant operation. The monitoring programs should be designed to obtain information to estimate the radiation doses for both the group within the population receiving the highest dose and the collective dose to the population within the 50-mile vicinity. An outline of the requirements for routine radiological monitoring at fuel-fabrication plants in the LMFBF program is given in Table 4.3-21.

The monitoring program should provide information on plant operations with respect to the adequacy of effluent control. The information can also be used to inform the public of the adequacy of the control program.

Periodic reviews should be made of the routine monitoring program to take into account changes in operations, in the environment, or in demography which may alter exposure pathways and exposed population groups.

The rates of release of radionuclides from the fuel-fabrication plant are anticipated to be so low that detection in environmental samples will be difficult. If the levels of radioactivity in the environmental samples are below detection limits, measurements of radionuclides made in the effluent at the point of release should be combined with meteorological and hydraulic information to estimate exposure rates at various points in the environment.

Because uranium and transuranic elements are the most significant radionuclides expected to be released from an LMFBR fuel-fabrication plant, primary emphasis should be devoted to their analysis in the environmental surveillance program. Normally their anticipated concentration will be low so that gross α -activity measurements will suffice. If an occasional sample shows an unusually high level of activity, a radiochemical analysis should be made.

4.3.7.3.3 Emergency Surveys

The major objective of an emergency radiological survey is to establish, as soon as possible, the level of radiation doses received by members of the public so that, if warranted, protective measures can be taken. If an accidental release has occurred, the geographic area most likely to be affected and the approximate levels of contamination should be estimated on the basis of the most reliable release data and of the environmental conditions prevailing at the time of release. When continuous samples are being collected in the routine monitoring system, they should be analyzed as soon as possible. Samples of drinking water, biota, and foodstuffs should be collected in the area most likely to be affected and in the areas where the greatest contribution to population dose is expected. The intensity of sampling depends on the magnitude of the accidental release.

4.3.7.3.4 Postoperational Surveys

If the operation of the facility has led to a buildup of long-lived radionuclides in the vicinity of the plant, periodic surveys may be advisable following decommissioning of the facility. The need for such surveys can be determined only by reviewing information obtained during operation of the plant.

The movement of radionuclides from their point of release through various pathways to man and other receptors is both site dependent and time dependent. As technological changes are made in effluent control, the spectrum of critical radionuclides and their environmental behavior is subject to change. Environmental pathways may

change due to changing land-use patterns and cultural habits of the exposed population. Thus a continuing effort is required to assure that the monitoring and evaluation program remains consistent with its objectives.

Information is particularly important on any changes with time of the behavior of long-lived radionuclides for the environment in which they accumulate. A research program should not be required at each site, but information should be available for comparable situations. Priority should be given to long-term studies of the behavior of Pu-239 and other transuranics in the environment, including such processes as uptake in the food web, resuspension, and other environmental transfer processes.

4.3.8 Postulated Accidents

Abnormal occurrences in the fuel-fabrication plant are discussed in the following sections, and the bases for analyses are given for use in determining their effects on the environment.

Reports issued by both industrial and AEC-sponsored organizations were examined for information useful in evaluating effects of various accidents. The most current and comprehensive AEC report on the impact of plutonium fuels technology was issued by Pacific Northwest Laboratory (PNL).¹⁵ Also, Westinghouse Electric Corporation has issued a three-volume set² on the environmental effects of its proposed plant in Anderson, South Carolina, which describes the physical plant necessary to fabricate fuels that contain plutonium for LWRs. Primary containment is provided by metallic glove boxes, that, in the Westinghouse case, have safety-plate glass windows.² Both studies deal with plants having a daily capacity of 1 metric ton of mixed-oxide fuel; that contains as much as 4% PuO₂; some portions of the PNL report discuss the characteristics of fuels containing up to 30% PuO₂. Emissions from the fuel-fabrication plants occur as both gases and particles and are released either continuously (at very low levels) due to normal operations or as a puff following an accident. Particles are removed from the exhaust gases as they pass through high-efficiency particulate air (HEPA) filters. The purge atmosphere from the primary containment is filtered first by a single HEPA filter, then by a pair of HEPA filters in series at the secondary containment barrier. This arrangement provides an aggregate filter transmission factor⁶⁶ of 1×10^{-9} . If primary containment is breached, the exhaust gases pass through the pair in series that have combined transmission factor⁶⁸ of 1×10^{-6} .

The only gaseous radioactive material handled in either the UO_2 or the mixed-oxide section of the fuel-fabrication plant is UF_6 , which readily reacts with atmospheric moisture to form particles of UO_2F_2 and hydrofluoric acid. Virtually all of the radioactive material found in the exhaust system, therefore, is in the form of a very fine powder or aerosol. The UO_2 powders will have an average particle size of less than $1\text{ }\mu\text{m}$, and the PuO_2 will range from 0.2 to $30\text{ }\mu\text{m}$ in diameter. The amount of powder suspended in the atmosphere of a box or room is dependent on the degree of turbulence. Very fine powders agglomerate rapidly, resulting in more rapid settling. Although an explosion could cause a high initial concentration of dust, agglomeration and settling, especially of dense materials such as UO_2 and PuO_2 , which as agglomerates probably have an average density near 6 g/cm^3 , would rapidly reduce the concentration. Information reported in the literature^{15,30,67} points to an upper value of 100 mg/m^3 for the concentration of particles in the respirable range.

Both the normal access areas and the limited or restricted access areas of the manufacturing buildings will be equipped with either a water-sprinkler system or a dry extinguisher system, whichever is compatible with operations in the area.

Glove boxes contain fire-suppression equipment,⁶⁸ and the exhaust system is equipped with heat detectors, a deluge chamber, and a water-mist-removal chamber to protect the final two stages of HEPA filters from fire and water damage.⁶⁹ Noxious or corrosive gases, such as compounds of fluorine, are removed with a caustic scrubber.

4.3.8.1 Natural-Origin Accidents

The plants will be designed to adequately accommodate the forces of an earthquake or a tornado and will be located above floodplains. Each plant will be thoroughly reviewed as part of the licensing process. Criteria for siting and design of plants for manufacture of plutonium-bearing fuels are currently being developed by the Regulatory Staff of the USAEC and by the American National Standards Institute (ANSI). AEC Division of Construction design criteria for new AEC plutonium facilities require that "critical items of the structure and its critical equipment (ventilation, electrical, fire protection, and utility systems) shall be designed to provide confinement of radioactive materials under normal operations and design basis accident conditions."⁷⁰ Design requirements to withstand natural-origin accidents, such as tornadoes and earthquakes, are also specified.

Unusual weather conditions could cause the loss of either the primary electric supply by damaging transmission lines or failure of the water supply as a result of

washout damage to a water main. Critical electrical requirements, such as blowers for the glove-box exhaust, health physics monitoring instruments, and emergency lighting will be supplied from a standby diesel generator or batteries. Total failure of primary and secondary power supplies would not result in the release of radioactive material to the environment, but work in glove boxes would be curtailed until power was restored. Similarly, water sprinklers for fire control and cooling water for furnaces would have a secondary source.

4.3.8.2 Man-Originated Accidents

A number of accidents could result from mechanical or human failure, which would curtail operations but would not cause a release of radioactivity nor have significant environmental impact. For example, a fire in the electrical switch-gear room, could cause a general power outage, but its consequences would only be to stop operations in the glove boxes until repairs were made.

Accidents representative of those which have environmental consequences are described in the following sections for both the UO_2 and the mixed-oxide plants. The data developed as the basis for analysis for these accidents are summarized in Table 4.3-22.

4.3.8.2.1 Uranium Dioxide Section of the Fuel Fabrication Plant

Those accidents that have occurred or are conjectured as possible in a plant for the fabrication of UO_2 powder and pellets for use in LWRs from UF_6 feed have been reviewed.⁷¹

Because the method and materials for fabricating UO_2 pellets for the blanket of an LMFBFR are identical to those for fabricating UO_2 fuel for LWRs, except for an increase in plant capacity and for use of depleted instead of enriched uranium, the accidents projected are similar. However, the absence of enriched uranium precludes criticality accidents. The referenced study identified a number of credible accidents that have various degrees of severity and involve either nonradioactive process chemicals or the UF_6 feed material.

Minor Accidents

Typically, occurrences such as loss of electrical power, pump-valve-piping leaks, inadequate ventilation of enclosures, and spills of UO_2 powder have trivial consequences. These incidents and others such as filter failures, minor fires, UF_6 releases in the plant, and inadvertent release of liquid wastes have occurred or might occur in the lifetime of the plant. These can result in consequences which are undesirable, such as increased

Table 4.3-22
SUMMARY OF DATA FOR ACCIDENTS IN
FUEL FABRICATION PLANTS

Nature of accident	Location of accident	Material released ^d				μCi of activity and source						% of normal gaseous effluent release
		Type	Amount (g)	Effluent volume (ft ³ of air $\times 10^6$)	Duration (min)							
UO ₂ Plant												
						U-234	U-235	U-236	U-238			
Rupture of UF ₆ cylinder	Volatilize UF ₆	UF ₆ , UO ₂ F ₂ , HF	784000 U ^b	As seepage at ground level	30	28700	3500	1450	26200			28500
Removal of UF ₆ cylinder valve	Volatilize UF ₆	UF ₆ , UO ₂ F ₂ , HF	58000 U ^c	As seepage at ground level	30	2130	259	107	19400			2100
Explosion in reducing furnace	Dry and calcine ADU	UO ₂	3×10^{-4} U	3.5	35	1.10×10^{-5}	1.34×10^{-6}	5.52×10^{-7}	9.98×10^{-5}			1×10^{-5}
UO ₂ -PuO ₂ Plant												
						Pu-236	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	
Failure of dissolution tube	Analytical laboratory	PuCl ₄ , UCl ₄ ^d	2×10^{-10} Pu	4.5	45	5×10^{-9}	4×10^{-5}	8×10^{-6}	1×10^{-5}	1×10^{-3}	3×10^{-8}	1×10^{-4}
Glove pulled from box	UO ₂ -PuO ₂ blending	PuO ₂	2×10^{-7} Pu	3	30	3×10^{-6}	4×10^{-2}	8×10^{-3}	1×10^{-2}	1	3×10^{-5}	1.2×10^{-1}
Explosion in metallography box	Metallography laboratory	UO ₂ -PuO ₂ ^e	5×10^{-6} Pu	3	30	1.1×10^{-4}	1	2×10^{-1}	2.7×10^{-1}	29	7.5×10^{-4}	3.5
Fire in waste compaction box	Solid waste handling	PuO ₂ , UO ₂ -PuO ₂ ^e	5×10^{-6} Pu	3	30	1.1×10^{-4}	1	2×10^{-1}	2.7×10^{-1}	29	7.5×10^{-4}	3.5
Explosion in oxidation-reduction furnace	Recycle scrap	UO ₂ -PuO ₂ ^e	7×10^{-4} Pu	3	30	2×10^{-3}	20	4	5.3	580	2×10^{-2}	70
Criticality	UO ₂ -PuO ₂ blending	PuO ₂	6×10^{-4} Pu	3	30	1.3×10^{-2}	120	24	32	3500	9×10^{-2}	420
General facility fire	UO ₂ -PuO ₂ fuel fabrication;	UO ₂ -PuO ₂ ^e	1.3×10^{-2} Pu	5	240	3×10^{-1}	2500	520	700	76000	2	9100

^aParticles 100% >0.03 <0.3 μm in average diameter.

^b296 kg of HF.

^c20 kg of HF.

^dContributes negligible radioactivity.

^e(0.83U-0.17Pu)O₂.

exposure of employees, damage to equipment and loss of operating time, but which have negligible impact on the environment.⁷²

Small releases of UF_6 vapor within the building due to defective valves or tubing will occur. In 13 such instances at the AEC gaseous diffusion plants over the past 20 years, an average of about 44 kg of UF_6 was released within the building. These small releases can be accommodated by the caustic scrubber systems normally used in UO_2 fuel-fabrication plants; therefore, they will not result in abnormal releases to the environment.

Severe Accidents

Three accidents of a potentially greater severity but having a very low probability of occurrence have been postulated and evaluated for the model plant. Two of these concern accidental release of UF_6 from a cylinder--rupture of a hot UF_6 cylinder and removal of a valve from a cylinder containing molten UF_6 , either of which releases hydrogen fluoride and soluble uranium. The third severe postulated accident is an explosion in a reduction furnace, which releases uranium in an insoluble form.⁷³

The depleted UO_2 feed is prepared from UF_6 which is supplied in steel cylinders with a capacity of 14 tons.²⁰ The UF_6 is a solid at room temperature but is melted by steam heat and removed as a gas by heating to about 200°F. If a leak develops in the cylinder, a portion of the liquid will evaporate to replace the escaping gas. Evaporation will continue, even though the steam heater is turned off, until enough heat is removed by the escaping gas or by other means, such as a water spray, to cause the remaining UF_6 to solidify.

In analyzing the results of a release from a UF_6 cylinder, a concurrent failure of the protective water-spray system is presumed. The principal cooling effect would be, therefore, from evaporation of liquid UF_6 . The UF_6 is released into a room with a volume of 100,000 ft³ at 90°F and 100% relative humidity.

Removal of Valve from Hot UF_6 Cylinder. An accident of this type occurred in 1966 when an operator inadvertently backed a valve out of a 10-ton-capacity cylinder while attempting to open the valve preparatory to supplying UF_6 vapor to a UF_6 -to- UF_4 reduction process.⁷⁴

The cylinder was enclosed in a steam-heated vaporizer chest, and the valve, which the operator could not see directly, was operated by a wrench that extended through

the vaporizer enclosure. The cylinder was oriented with the valve positioned to release vapor from a space over the liquid UF_6 .

The analysis of this type of accident assumed that the UF_6 was at 200°F and that the vapor would behave essentially as a perfect gas. Also, the flow of gas through the coupling on the cylinder is described by the equation for an orifice such as that given in Perry's Chemical Engineer's Handbook.⁷⁵ For a 1-in. valve fitting (1-1/4-in.-diam hole), the equation reduces to

$$W = 13.02 \rho_1 P_1,$$

where

W = UF_6 vapor released, lb/min

ρ_1 = UF_6 vapor density, lb/ft³

P_1 = UF_6 vapor pressure, psia.

Since the vapor pressure and density of the UF_6 vapor decrease as the remaining liquid cools, release rates were calculated for 10°F increments of temperature drop. The amount of gas formed for each 10°F drop in temperature was also determined by equating the heat lost by the liquid to that gained by the vapor.

$$\begin{aligned} &(\text{Specific heat of liquid})(\text{temperature change})(\text{weight of liquid}) \\ &= (\text{heat of vaporization})(\text{weight of vapor}) \end{aligned}$$

$$(0.143 \text{ Btu/lb}^\circ\text{F})(10^\circ\text{F})(x + \frac{y}{2}) = (35.1 \text{ Btu/lb}) y$$

where

x = fraction of liquid remaining,

y = fraction of liquid evaporated = 0.04/10°F.

A summary of the temperature, time, and weight loss relations is given in Table 4.3-23.

Short of permitting the UF_6 to vent until the vapor pressure approaches atmospheric pressure, and thus releasing about 53% of the UF_6 contained in the cylinder, one or more operators in completely protective clothing must plug the valve coupling. Experiments simulating accidents and using steam as a substitute for UF_6 have shown that a suitably clothed and trained operator can drive a wooden pin into a 1-in. valve coupling against 50 psig of pressure. The operators should be able to dress up in protective clothing and find their way through the resulting fog to the

cylinder in about 15 min. A time lapse of 30 min to stop the leak has been used in this discussion. At this time the operator must overcome a resisting force of about 23 psig.

Table 4.3-23

VAPOR RELEASED THROUGH 1-1/4-IN.-DIAM VALVE FITTING HOLE IN 14-TON CYLINDER OF UF_6 (WHOLE COOLING FROM 200 TO 150°F)

Incremental Release Conditions				UF_6 Vapor Weight Loss			Release Time (min)	
Temperature		UF_6						
Step (°F)	Avg (°F)	Pressure (psia)	Density (lb/ft ³)	ΔW (lb)	Rate (lb/min)	Total (lb)	Δ Time	Total
200-190	195	45	2.1	1102	127	1102	8.7	8.7
190-180	185	40	1.8	1058	111	2160	9.5	18.2
180-170	175	34	1.5	1016	93	3176	10.9	29.1
170-160	165	29	1.2	975	77	4151	12.7	41.8
160-150	155	25	1.1	936	68	5088	13.8	55.6

The UF_6 released in 30 min is estimated from the values in Table 4.3-23 at 3245 lb. The release of 3245 lb of UF_6 calculated here shows reasonable agreement with those of 1500 and 4550 lb calculated elsewhere^{76,77} and 3800 lb actually lost when a valve was inadvertently removed from a hot cylinder.⁷⁴ However, in these instances, the cylinder capacity was 10 tons.

The gaseous UF_6 will react with the moisture in the air to form HF and UO_2F_2 until no atmospheric moisture remains. Thus 2082 lb of UF_6 will react to form 473 lb of HF and 1821 lb of UO_2F_2 ; and 1163 lb of UF_6 will remain. This reaction causes an increase in the volume of gases in the 100,000-ft³ room of 6284 ft³ and an accompanying rise in pressure. The reactants and products were assumed to mix completely with the air in the room, and the mixture escapes until the room pressure returns to atmospheric. This adjustment results in 6% of the HF, UO_2F_2 , and unreacted UF_6 being released from the building.

The heat released from hydrolysis of UF_6 has not been taken into account, but its effect is compensated by the initial release of essentially pure air from the building and the reaction of HF with the interior of the building and its contents. The UF_6 , HF, and UO_2F_2 remaining in the building are removed by water sprayed into

room atmosphere using a high-pressure fog nozzle. The resulting solution is routed to a suitable collection basin.

Basis for analysis:

Materials - soluble HF, UO_2F_2 , and UF_6

Total amount - HF 28 lb (13 kg)

UO_2F_2 108 lb (49 kg)

UF_6 69 lb (30 kg)

[HF equivalent of UF_6 = 15 lb (7 kg)]

[UO_2F_2 equivalent of UF_6 = 59 lb (27 kg)]

Concentration - in 6284 ft³ of mixture (air, UF_6 , and HF) during 30 min at ground level

Physical properties - HF and UF_6 as gases and UO_2F_2 as powders ranging from about 10 μm to submicron particles

Isotopic content (U depleted in U-235 to 0.2%)

U-234 2.13 E-3 Ci

U-236 1.07 E-4 Ci

U-235 2.59 E-4 Ci

U-238 1.94 E-2 Ci

Rupture of Hot UF_6 Cylinder. The most severe potential accident would be a failure near the bottom of a 14-ton-capacity cylinder containing molten UF_6 . The entire contents would flow out of the cylinder into the steam chest and onto the floor; a portion would volatilize and react with steam condensate and the moisture content of the air in the room.

Failure at a seam would result from a severe overpressurization of the steel cylinder. This could occur only if a cylinder was overfilled, leaving insufficient expansion volume on reheating, or if air leaked into the cylinder during storage and was not pumped out prior to heating. Several mechanical and/or administrative controls would have to fail concurrently to permit either of these conditions to progress to a release, and in practice, a catastrophic failure of a UF_6 cylinder has never occurred. Nevertheless, the subject failure is presumed to occur to a cylinder located in a room with a volume of 100,000 ft³ containing air at 90°F and 100% relative humidity, leading to the following sequence of events:

- (1) Molten UF_6 at 200°F is forced from the cylinder during a 15-min interval. The liquid then flows from the cylinder into the surrounding steam chest through a drainage line into a condensate catch basin below or within the steam chest.

- (2) Initial reaction is with 20 gal of water condensate in the catch basin, but high vapor pressure of the hot UF_6 liquid rapidly releases gaseous UF_6 , which then combines with atmospheric moisture. The heat contained in the molten UF_6 is assumed to be lost only by vaporizing UF_6 . This process would volatilize approximately 53% of the original 14 tons of UF_6 while the remainder cooled to 100°F and solidified. Evaporation and reaction of UF_6 with water would result in a net increase in gas volume and the release of heat.
- (3) The increasing gas volume forces a mixture of air, UF_6 , HF, and entrained UO_2F_2 powder from the building until the original pressure is restored in the building. The period of release from the building lasts for 30 min, the maximum time considered necessary for an emergency crew to respond and apply water fog. The amount and composition of gas forced from the building is calculated by assuming no loss of reactants or reaction products until vaporization and reaction is complete, at which time an increment of volume at the starting temperature of 90°F is released to restore the starting pressure.

Composition and volumes of gases at 90°F before release:

<u>Gas</u>	<u>Volume (ft³)</u>
Dry air	95,243
UF_6	12,347
HF	<u>17,024</u>
Total	124,614

24,610 ft³ or about 20% of 124,614 ft³ leaks from building, having the following content:

<u>Gas</u>	<u>Volume (ft³)</u>	<u>Weight</u>	
		<u>lb</u>	<u>kg</u>
Dry air	18,814	1,347	612
UF_6	2,438	2,134	970
HF	<u>3,362</u>	167	76
	24,614		

A proportional fraction of the UO_2F_2 formed from UF_6 reacting with atmospheric moisture remains suspended and is swept from the building. This release would amount to 364 lb of UO_2F_2 .

- (4) The ventilation system is turned off when the spill begins, and near the end of the release period, the emergency crew begins the application of a fine water fog from high-pressure hoses to knock down and remove the UF_6 , UO_2F_2 , and HF residual in the room and thus stop further leakage from the building. The resulting aqueous solution of UO_2F_2 is drained to a collection basin.

The effect of heat generated by the reaction of UF_6 with moisture, which would cause a higher release from the building, has not been taken into account but is compensated by the initial release of essentially pure air from the building and the absorption of HF on the interior surfaces of the building.

Basis for analysis:

Material - UF_6 , UO_2F_2 , and HF

Total Amount - UF_6 2134 lb (970 kg)

UO_2F_2 364 lb (165 kg)

[UO_2F_2 equivalent of UF_6 = 1867 lb (849 kg)]

HF 167 lb (76 kg)

[HF equivalent of UF_6 = 485 lb (220 kg)]

Concentration - In 26,000 ft³ of mixture (air, UF_6 , UO_2F_2 and HF) during 30 min at ground level

Physical properties - soluble UF_6 ; UO_2F_2 and HF as gases or respirable powder

Isotopic content (U depleted in U-235 to 0.2%)

U-234	2.87 E-2 Ci	U-236	1.45 E-3 Ci
U-235	3.50 E-3 Ci	U-238	2.62 E-1 Ci

Prevention and Containment of Accidental UF_6 Releases

The chemical and isotopic content of a cylinder must be measured before it is shipped from the uranium enrichment plant. This measurement involves a heating step to melt and homogenize the UF_6 followed by removal of a sample for analysis. The operators of the Oak Ridge Gaseous Diffusion Plant have altered their equipment to reduce the chance of an accidental release and to provide containment in the event of a release.

In the past the cylinder was heated by placing it in a steam chest that was not designed for high-pressure operation and, therefore, would not contain a massive release. The steam chest has been replaced by an autoclave with the capacity to contain the UF_6 if the shipping cylinder or a sampling attachment fails. An electric cell continuously monitors the conductivity of the steam condensate and causes the automatic closure of the drain line and cutoff of electric power if any UF_6 is released. The possibility of accidentally backing a valve out of the

cylinder has also been eliminated by changing the operating procedure to require opening the cylinder valve prior to closing the autoclave and heating the cylinder. Further, the valve is equipped with an actuator that penetrates the autoclave wall and has a ratchet mechanism that will operate only in the direction to close the valve. These two provisions should virtually eliminate the possibility of the releases described here.

Explosion in Reduction Furnace

The calcination of ADU to UO_2 powder is performed in furnaces with a reducing atmosphere of hydrogen and nitrogen. The hydrogen concentration is carefully controlled to prevent the development of an explosive atmosphere. It is hypothesized that these controls fail and an explosion occurs in one of the furnaces. The explosive force would not be sufficient to destroy the furnace, but uranium could be blown out of the ends. It is assumed that the furnace contains 30 kg of uranium, all in the form of UO_2 pellets and that all of the uranium is blown out. (An explosion of this type conceivably could happen in either the powder-reduction furnace or the pellet-sintering furnace. The material released would be in the form of UO_2 powder in the former case but primarily UO_2 pellets in the latter.) It is unlikely that any of the uranium would be released from the plant, but it is hypothesized that 0.3 kg of the uranium is drawn into the air exhaust system...⁷⁸

The uranium is removed by HEPA filters that have a transmission factor of 1×10^{-6} .² The UO_2 is swept from the plant in 35 min at an exponentially reducing rate. Exhaust leaves the building at the rate of 100,000 cfm.

Basis for analysis:

Material - insoluble UO_2

Total amount - 3×10^{-4} g (as uranium)

Concentration - in 3,500,000 ft³ of air at 60 ft elevation during 35 min

Physical properties - 0.03 - 0.3 μ m diam

Isotopic content -

U-234	1.095 E-11 Ci	U-236	5.52 E-13 Ci
U-235	1.334 E-12 Ci	U-238	9.973 E-11 Ci

4.3.8.2.2 Mixed-Oxide Section of the Fuel-Fabrication Plant

The mixed-oxide section of the fuel-fabrication plant does not require the use of toxic chemicals such as derivatives of fluorine; therefore, the principal factors that have environmental impact are those which could cause the accidental release of plutonium.

The detailed effect of the relatively high neutron and gamma radioactivity of high-exposure plutonium on plant design is not now known but could have two consequences.

If shielded glove boxes were used, the maximum batch size would be limited by allowable exposure to operating personnel, and the amount of plutonium released in an accident would be reduced proportionately. If more massive shielding, such as concrete, were used, the chance of primary containment failure would be reduced, but the batch size, then being limited only by criticality considerations, would be larger; therefore, more plutonium could potentially be released in the event of an accident.⁷⁹

However, if organic material is used for neutron shielding, the combustible inventory in the plant would be increased.

Minor Accidents

There are a large number of accidents that might occur which would result in either no abnormal release of radioactive material or release of an amount so small as to be of no consequence to the environment.

Failure of Quartz High-Pressure Dissolution Tube. Complete dissolution preparatory to chemical analysis of $(U,Pu)O_2$ that has been sintered at high temperature often requires exposure to hydrochloric acid at 200 to 300°C. This treatment is done in a sealed quartz tube which in turn is contained in a steel capsule; failure of the quartz tube due to internal pressure on heating is prevented with 1500 psi of CO_2 in the steel capsule. The quartz tube is assumed to have a pinhole in the end seal which allows high-pressure CO_2 to leak into the tube during overnight dissolution. On removal from the steel container, the quartz tube fails, releasing approximately 0.2 g of soluble plutonium to the glove box as a fine aerosol. The HEPA filter on the box remains intact, providing, with the filters at the point of secondary containment, a transmission factor of 10^{-9} . The aerosol is purged from the box during a 45-min period (three box changes).

Basis for analysis:

Material - soluble $PuCl_4$ and UCl_4

Total amount - 2×10^{-10} g

Concentration - in 4,500,000 ft³ of air at 60 ft elevation during 45 min

Physical properties - 0.03-0.3 μ m diam

Isotopic content -

Pu-236	5 E-15 Ci	Pu-239	8 E-12 Ci	Pu-241	1 E-9 Ci
Pu-238	4 E-11 Ci	Pu-240	1 E-11 Ci	Pu-242	3 E-14 Ci

Intermediate Accidents

This class of accident will occur with low frequency but will cause an increase in radioactivity in the plant effluent above normal levels; however, the increase will be of short duration.

Glove Pulled from Glove Box. While changing the glove on a box, the operator inadvertently pulls it off. The glove box contains the mill used to condition PuO_2 , which is a dusty operation. Approximately a gram of oxide powder is on the surface of the glove previously exposed to the box interior; 20% of this dust becomes suspended in the room atmosphere. The suspended dust is removed in 30 min in three changes of air and passes through a pair of HEPA filters in series having a combined transmission factor of 10^{-6} .

Basis for analysis:

Material - insoluble PuO_2 powder

Total amount - 2×10^{-7} g

Concentration - in 3,000,000 ft^3 of air at 60 ft elevation during 30 min

Physical properties - 0.03-0.3 μm diam

Isotopic content -

Pu-236	3 E-12 Ci	Pu-239	8 E-9 Ci	Pu-241	1 E-6 Ci
Pu-238	4 E-8 Ci	Pu-240	1 E-8 Ci	Pu-242	3 E-11 Ci

Explosion in Metallographic Glove Box. Sintered pellets are examined as part of the quality control of the fuel. Polished samples are flushed with absolute alcohol after polishing preparatory to examination. The volume of alcohol is kept to a minimum and would be less than 500 ml. Because of an assumed buildup of dust on the input filters, the purge rate on the box is substantially reduced. An excessive use of alcohol on a sample leads to an explosive concentration of solvent, which is ignited by a static electric spark. The resulting explosion blows the windows from the glove box and suspends 10% of the accumulated inventory of about 50 g of $(\text{U,Pu})\text{O}_2$ powder from the box. The heat detector triggers the fire extinguisher system, which extinguishes the fire. Dust is removed from the room in 30 min and is filtered through a pair of HEPA filters in series having a transmission factor of 10^{-6} .

Basis for analysis:

Material - (0.83 U, 0.17 Pu) O_2 powder

Total amount - 5×10^{-6} g

Concentration - in 3,000,000 ft^3 of air at 60 ft elevation during 30 min

Physical properties - 0.03-0.3 μm diam

Isotopic content -

Pu-236	1.1 E-10 Ci	Pu-239	2.0 E-7 Ci	Pu-241	2.9 E-5 Ci
Pu-238	1.0 E-6 Ci	Pu-240	2.7 E-7 Ci	Pu-242	7.5 E-10 Ci

Fire in Waste-Compaction Glove Box. Burnable wastes are collected at a compaction station until enough (~ 100 lb) is collected to fill a 55-gal drum. Waste is composed of polyethylene bagging material, rubber gloves, and cellulose wipes. The waste is contaminated to the extent of 0.01% by weight or the equivalent of approximately 4.5 g of plutonium. An equal amount of general contamination is present in the glove box. A solvent-damp wipe in the waste ignites due to the discharge of static electricity. Concurrently, the heat detector in the box fails, so the fire extinguisher system does not operate. The fire burns the gloves from the box and subsequently causes the glass panels in the box to collapse. During a 30-min period necessary to bring the fire under control, combustion products containing approximately 5 g of plutonium are released to the room. The air, containing plutonium dust, passes through a pair of fire-protected HEPA filters with a transmission factor of 10^{-6} before being released from the building. The air exhausted from the room will also contain products of incomplete combustion (i.e., CO_2 , CO, H_2O , and soot).

Basis for analysis:

Material - PuO_2 and (0.83 U, 0.17 Pu) O_2 powder

Total amount - 5×10^{-6} g of plutonium content

Concentration - in 3,000,000 ft^3 of air at 60 ft elevation during 30 min

Physical properties - 0.03-0.3 μm diam

Isotopic content -

Pu-236	1.1 E-10 Ci	Pu-239	2.0 E-7 Ci	Pu-241	2.9 E-5 Ci
Pu-238	1.0 E-6 Ci	Pu-240	2.7 E-7 Ci	Pu-242	7.5 E-10 Ci

Severe Accidents

Accidents of this class would release significant radioactive material to the environment. Because of the impact of this kind of release, every precaution is taken to prevent its occurrence. The additional radioactivity released for the accidents considered here ranges from a fraction of the annual activity released from the mixed-oxide plant to 100 times the annual release. The reader should note, however, that in each event several controls, administrative and/or mechanical, are presumed to fail simultaneously, and pessimistic assumptions have been made about the fraction of the fissile inventory suspended in the air leaving the various work areas.

Explosion in Oxidation-Reduction Furnace. Scrap (U,Pu)O₂ pellets are conditioned for recycle by heating successively in air and a 4% H₂-N₂ atmosphere. A multiple failure of controls on the H₂-N₂ mixture allows pure H₂ to reach the hot airfilled furnace after an oxidation step. The resulting explosion, following autoignition, blows the door off the furnace, knocks the window out of the glove box, and releases mixed-oxide pellets (150 kg) and dust (1.5 kg) to the room, measuring 15 x 50 x 225 ft (169,000 ft³). The suspended dust, at the limiting concentration of 100 mg/m³, (refs. 30,66, and 67), is purged from the room in 30 min to the secondary-containment filters, which have a transmission factor of 10⁻⁶.

Basis for analysis:

Material - insoluble (0.83 U, 0.17 Pu)O₂ powder

Total amount - 7×10^{-4} g of oxide containing 1×10^{-4} g of plutonium

Concentration - in 3,000,000 ft³ of air at 60 ft elevation during 30 min

Physical properties - 0.03-0.3 μ m diam

Isotopic content -

Pu-236	2 E-9 Ci	Pu-239	4.0 E-6 Ci	Pu-241	5.8 E-4 Ci
Pu-238	2.0 E-5 Ci	Pu-240	5.3 E-6 Ci	Pu-242	2 E-8 Ci

Criticality. Although a criticality accident has never occurred with powdered materials, the assembly of a critical mass in equipment for conditioning the PuO₂ is assumed. Triple batching due to operator error results in the assembly of a critical mass of about 25 kg⁶⁷ of PuO₂ powder. In the resulting excursion, 10¹⁸ fissions occur in 1 sec and release 30,000 Btu of heat.

The heat content equation⁸⁰ and the heat of fusion⁸¹ for PuO₂ can be used to show that the 30,000 Btu of heat will momentarily heat the PuO₂ to about 3000°C, which is above the melting point (2390°C). The rapid heating and expansion of air blows the oxide from the mill, and the resulting rise in pressure ruptures the glove box and disperses the 25 kg of PuO₂ into the atmosphere of the 15 x 50 x 225 ft room that contains the fabrication line. The increase in pressure in the secondary containment area is small due to its large volume; therefore, the HEPA filters in the secondary containment exhaust system are not damaged by the accident. Powder is suspended at the limiting concentration of 100 mg/m³. Fractional releases of the fission products from the PuO₂ occur. The fractional release values selected are based on melting experiments by Parker⁸² with irradiated UO₂, thermodynamic calculations by Leitnaker⁸³ to determine their chemical form, and the vapor pressure of metallic elements⁸⁴ and selected oxides.⁸⁵ The PuO₂ powder and fission products are swept from the plant at an exponentially reducing rate that results in

about 90% removal in about three room-volume changes of air. The assumption is made that the halogens and noble gases pass unaffected through the filter at the secondary container, but a transmission factor of 1×10^{-6} was assigned to PuO_2 and solid elements. The computer program ORIGEN²⁶ was used to analyze the level of fission-product release by use of the above parameters. Those fission-product isotopes that account for 99% of the radioactivity contributing to inhalation, ingestion, and gamma-exposure hazard were selected to determine the fission products used as the basis for analysis. Although 30 min is required for three room-volume changes of air, a reduced time of 15 min is assumed for removal of fission products. This condition exaggerates the environmental effects of the release.

Basis for analysis:

Material - insoluble PuO_2 powder

Total amount - 7×10^{-4} g containing 6×10^{-4} g of plutonium

Concentration - in 3,000,000 ft³ of air at 60 ft elevation during 30 min period.

Physical properties - 0.03-0.3 μm diam

Isotopic content -

Pu-236	1.3 E-8 Ci	Pu-239	2.4 E-5 Ci	Pu-241	3.5 E-3 Ci
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Pu-238	1.2 E-4 Ci	Pu-240	3.2 E-5 Ci	Pu-242	9 E-8 Ci
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Basis for analysis:

Material - halogens and noble gases

Total amount at stack after 15 min - see Table 4.3-24

Concentration at the stack -

Tabulated isotopes released in 1,500,000 ft³ of air at 60 ft elevation in 15 min

General Facility Fire. A fire of this scope is not expected in a properly designed and operated plant for the fabrication of mixed oxide pellets. The fact that such a fire did occur at Rocky Flats in May 1969 makes it prudent to consider how to prevent a similar fire and to anticipate the consequences if one should occur.

An investigation⁸⁶ into the cause of the Rocky Flats fire established, among other things, that:

- (1) Metallic plutonium briquettes were the probable source of ignition.
- (2) Heat detectors close to the origin of the fire were located exterior to and below the glove boxes and did not sense the fire; other internal

Table 4.3-24
 PORTION OF THE FISSION YIELD FROM 10^{18} FISSIONS (FAST)
 IN PuO_2 RELEASED TO THE ENVIRONMENT

Nuclide	Half-life	Release (Ci)
Kr-85m	4.4 hr	6.96 E0
Kr-87	1.3 hr	4.17 E0
Kr-88	2.8 hr	2.51 E0
I-131	8.1 day	1.22 E-1
I-132	2.3 hr	5.14 E0
I-133	20.8 hr	7.00 E0
I-134	52.0 min	1.43 E2
I-135	6.7 hr	4.46 E1
Rb-89	15.0 min	2.16 E-4
Rb-90	2.7 min	7.23 E-5
Cs-138	32.0 min	2.64 E-4
Sr-89	54.0 day	2.89 E-8
Sr-90	28.0 yr	4.23 E-10
Sr-91	9.7 hr	1.33 E-5

- detectors located elsewhere in the glove-box line gave an alarm after the fire had progressed considerably.
- (3) The plastic windows of the glove boxes contributed heavily to the spread of the fire.
 - (4) The long interconnected conveyor system without physical barriers provided a path for the fire to spread.

The investigators concluded that:

- (1) Plutonium operations involving interconnecting glove boxes should have: internal heat detectors, automatic fire-suppression equipment in the boxes, and means of isolating a fire to a small portion of a system.
- (2) Automatic sprinklers should be provided where compatible with criticality considerations in a given operating area.

The mixed-oxide plant will obviously not have the complication of handling metallic plutonium. The use of glass windows in the glove boxes and of fire-suppression equipment both interior and exterior to the glove boxes is expected. The glove-box lines shall also have suitable barriers to isolate various portions of the lines. The combustible inventory of the plant shall be kept to a minimum, and approved containers shall be provided for combustible supplies and wastes. Nonetheless, a general facility fire is postulated and an estimate of amounts developed below.

A lift truck crushes a 5-gal can of ethyl alcohol, which ignites and spreads to involve a large fraction of glove boxes in several parallel fabrication lines. One percent⁸⁷ of the $(U,Pu)O_2$ in powder and pellet treatment operations is suspended and swept out through the final two filters of the exhaust system, having a transmission factor of 10^{-6} . Total $(U,Pu)O_2$ inventory in powder and pellet treatment steps is 7800 kg. Supply fans shut down due to the fire, but the exhaust fans continue to operate, resulting in an 80% reduction in air drawn through the plant. Duration of the fire is 4 hr.

Basis for analysis:

Material - $(0.83 \text{ U}, 0.17 \text{ Pu})O_2$

Total amount - $7.8 \times 10^{-2} \text{ g}$ of $(U,Pu)O_2$ containing $1.3 \times 10^{-2} \text{ g}$ of plutonium

Concentration - in $5 \times 10^6 \text{ ft}^3$ of air and combustion products at 60 ft elevation during 4 hr

Physical properties - $0.03\text{-}0.3 \text{ }\mu\text{m}$ diam

Isotopic content -

Pu-236	3 E-7 Ci	Pu-239	5.2 E-4 Ci	Pu-241	7.6 E-2 Ci
Pu-238	2.5 E-3 Ci	Pu-240	7.0 E-4 Ci	Pu-242	2 E-6 Ci

4.3.9 Environmental Effects of Postulated Accidents

4.3.9.1 Effects of Chemical Releases

4.3.9.1.1 Atmospheric Dilution of HF Released in a Postulated Accident

Minor accidents, described in Section 4.3.8.2.1, may cause short-term increases in the ambient HF concentrations. Environmental effects are not considered to be serious because an order of magnitude increase in source strength (from 0.02 g/sec routine to 0.2 g/sec) would result in air concentrations $< 20 \mu\text{g}/\text{m}^3$, a level that would be tolerable for short-term exposure (< 30 min). Two more consequential accidents (cases 1 and 2) were studied. In case 1, the accidental release of UF_6 from a valve malfunction, 13 kg of HF is produced by the immediate reaction of UF_6 with H_2O in the building atmosphere. An additional 7 kg of HF forms from the instantaneous reaction of UF_6 with atmospheric moisture outside the building. A total of 20 kg of HF is produced in this case-1 UF_6 accident over a 30-min interval. The release rate is calculated as 11 g/sec.

For the accidental release of UF_6 from a tank rupture (case 2), 76 kg of HF will be produced over the 0.5-hr interval of release. An additional 970 kg of UF_6 will escape the building, and at outside air conditions of 50% relative humidity and 75°F at a mixing height of 100 m, 970 kg of UF_6 will be reacted to 220 kg equivalent of HF within a distance of 20 m of the point of release. At a lower specific humidity, the lateral distance required for total reaction of UF_6 to HF will increase proportionately. The release rate for this case-2 accident is 164 g/sec.

Air concentrations of HF released from both accidents (cases 1 and 2) were calculated for various distances from the source using the Gaussian plume model. The calculated concentrations are highly dependent on joint frequencies of wind speed and stability categories, and in the absence of site-specific joint frequency data, several generic conditions of meteorology were used. Air concentrations of HF were calculated for the meteorological conditions used for the radiological accident, approximately the "50% probability condition." A second generic condition is based on meteorological circumstances (described in Section 4.3.6.2.1) which tend to restrict dispersion. The second condition is considered the severe meteorology. Depending on duration of release and uniformity of meteorological parameters, either the centerline concentration or the sector-averaged values for these conditions may

be considered in the assessment of HF effect from accidents. For the condition represented by severe meteorology, air concentrations of HF were calculated from an average of centerline and sector-averaged χ/Q values.

Hydrogen fluoride concentrations in air from the accidental release were calculated at successive distances for the two different meteorological conditions (Table 4.3-25). In a single sector for the 50% probability condition, the χ/Q values are 2.8×10^{-5} and 8.4×10^{-7} at 1000 and 8045 meters (5 miles) respectively. At the same distances, the respective χ/Q values are 1.3×10^{-4} and 7.2×10^{-6} for the severe meteorology.

The impact from accidental releases of HF is highly dependent on the duration of exposure. Many meteorological factors will affect the exposure conditions, for example, atmospheric stability, wind velocity, washout, and dry deposition. Massive air movements (passages of fronts) will enhance mixing and dispersion, thereby minimizing the duration of exposure. Stagnation and inversions will extend the period of exposure. Exposure times could range from the duration of release (0.5 hr) to as long as two days for severe stagnation conditions.⁸⁸ Because consideration of the influence of all these meteorological conditions on the duration of exposure is impractical, the calculation of atmospheric dispersion of HF is based on an exposure time equivalent to the length of release (0.5 hr). Diffusion of HF into the plant canopy will enhance the duration of exposure. Surface roughness features may also exert a drag on the HF plume, thereby increasing the exposure time.

4.3.9.1.2 Effect of Accidentally Released HF on Vegetation

Both air concentration and duration of exposure determine the magnitude of HF effects on plants. Unlike routine releases where plants are exposed continuously, exposures in relation to accidents are short-term governed by duration of release, atmospheric dispersion conditions and plume depletion through deposition. The effect of HF on green plants is examined for an exposure period of 0.5 hr. Both longer and shorter exposure periods may occur, but there is insufficient information to evaluate them quantitatively.

Because green plants are affected by extended exposure to HF concentrations as low as $0.2 \mu\text{g}/\text{m}^3$, the experimental work has focused on threshold concentrations and exposures that induce effects at very low concentrations. To assign definite effects to a short-term exposure of 30 min is difficult because very limited experimental data are available for such exposure regimes. Only four observations

Table 4.3-25
CALCULATED CONCENTRATIONS OF ACCIDENTALLY RELEASED HF
FOR ACCIDENT CASES 1 AND 2

Meteorological Condition	Distance from Fuel Fabrication Plant (meters)	χ/Q (sec/m ³)	Concentration in Air (μg/m ³)
<u>Case 1^a</u>			
50% Probability ^b	1000	2.8 E-5 ^c	3.1 E2
	1609 (1 mile)	1.3 E-5	1.4 E2
	3218 (2 miles)	4.0 E-6	4.4 E1
	8045 (5 miles)	8.4 E-7	9.3 E0
Severe ^d	1000	1.3 E-4	1.4 E3
	1609 (1 mile)	6.8 E-5	7.6 E2
	3218 (2 miles)	2.5 E-5	2.8 E2
	8045 (5 miles)	7.2 E-6	8.0 E1
<u>Case 2^e</u>			
50% Probability ^b	1000	2.8 E-5	4.6 E3
	1609 (1 mile)	1.3 E-5	2.1 E3
	3218 (2 miles)	4.8 E-6	6.6 E2
	8045 (5 miles)	8.4 E-7	1.4 E2
Severe ^d	1000	1.3 E-4	2.1 E4
	1609 (1 mile)	6.8 E-5	1.1 E4
	3218 (2 miles)	2.5 E-5	4.1 E3
	8045 (5 miles)	7.2 E-6	1.2 E3

^aSource strength = 11.11 g/sec for 30 min.

^bSee Appendix II.1 for definition; value includes plume depletion based on a deposition velocity of 0.003 m/sec.

^c2.8 E-5 = 2.8×10^{-5} .

^dDescribed in Sect. 4.3.6.2.1; χ/Q values based on type-F stability, 2.26-m/sec wind speed blowing in direction of a single sector, and deposition velocity of 0.003 m/sec. χ/Q values are the same as in Table 4.3-9; condition is assumed to represent an approximation of atmospheric dilution for a severe meteorological condition.

^eSource strength = 164.4 g/sec for 30 min.

at durations of 30 minutes or less are reported in McCune's treatise⁴⁴ on establishing air quality criteria, and for these tests, effects were observed at air concentrations ranging from 800 to 6000 $\mu\text{g}/\text{m}^3$ of HF. Recent data⁸⁹ concerning effects of HF on barley, a moderately sensitive plant, and alfalfa, a relatively resistant plant, show a 50% inhibition of photosynthesis at an HF concentration of 200 ppb for a 2-hr exposure. The effect was reversible; that is, normal levels of CO_2 uptake were observed the following day. Foliar necrosis (permanent damage) was formed at a 2-hr exposure to 150 ppb of HF. A 5 to 10% reduction in photosynthesis was reported for exposure to ~100 ppb HF for 30 min. McCune's data⁴⁴ show that tomato, corn, fruit trees (noncitrus), and conifers exhibit leaf markings at HF concentrations ranging from 500 to 1000 $\mu\text{g}/\text{m}^3$ for 1-hr exposures. By extrapolating McCune's graphs (observations for 1- to 10-hr exposure periods) to estimate a concentration for a 0.5-hr exposure, leaf markings would be manifested at air concentrations of HF approximately 1000 $\mu\text{g}/\text{m}^3$ or greater. Bennett and Hill's data⁸⁹ indicate a reversible inhibition of photosynthesis at an HF concentration of 200 ppb.

Effects on Vegetation Caused by the Case 1 Accident

The HF concentration in a single sector at 1000 m approaches the threshold of effect on green plants at an exposure time of 0.5 hr (Table 4.3-26) for the 50% probability meteorological condition. A reversible effect on photosynthesis (about 50% inhibition) will occur at the estimated HF concentration (310 $\mu\text{g}/\text{m}^3$) for the 50% meteorology. At a similar exposure regime, this effect could be expected out to a distance of two miles under severe meteorology. For the severe meteorology, the maximum estimated concentration (1400 $\mu\text{g}/\text{m}^3$; Table 4.3-26) exceeds the threshold for foliar damage to tomato, corn, fruit trees, and conifers for a 30-min exposure. The area affected within the one-mile radius at concentrations of 310 $\mu\text{g}/\text{m}^3$ (50% probability) and 1400 $\mu\text{g}/\text{m}^3$ (severe meteorological conditions) would be 0.2 square miles. The area affected within a five-mile radius at concentrations of 9.3 and 80 $\mu\text{g}/\text{m}^3$ (50% probability and severe meteorology, respectively) would be 4.9 square miles. Within these areas, effects on vegetation would range from temporarily diminished photosynthesis to foliar damage to reduction in yield. The latter effect would be manifested if exposures occurred during the seedling and floral initiation stages of growth.

Effects on Vegetation Caused by the Case-2 Accident

Damage to sensitive vegetation would be expected to a distance of one mile for the 50% probability condition; the calculated HF concentration is 2100 $\mu\text{g}/\text{m}^3$ (Table 4.3-26). From two to five miles, the HF concentrations (660 to 140 $\mu\text{g}/\text{m}^3$) exceed the threshold at which temporary inhibition of photosynthesis has been observed.⁸⁹

Table 4.3-26

ESTIMATED CONCENTRATION OF FLUORIDE IN VEGETATION

AFTER A 0.5-HR EXPOSURE (ACCIDENT CASES 1 AND 2)

Distance from Source (meters)	Meteorological Condition	χ/Q (sec/m ³) ^a	Deposition Rate ($\mu\text{g}/\text{m}^2$ sec)	Fluoride Concentration		
				Air ($\mu\text{g}/\text{m}^3$)	Ground ($\mu\text{g}/\text{m}^2$)	Vegetation ^b (ppm)
Case 1						
1000	50% Probability	2.8 E-5 ^c	0.94	310	1,700	11.3
	Severe	1.3 E-4	4.2	1,400	7,560	50
1609 (1 mile)	50% Probability	1.3 E-5	0.43	140	770	5.2
	Severe	6.8 E-5	2.3	760	4,140	28
8045 (5 miles)	50% Probability	8.4 E-7	0.03	9.3	54	0.4
	Severe	7.2 E-6	0.24	80	432	2.9
Case 2						
1000	50% Probability	2.8 E-5	14	4,600	25,000	168
	Severe	1.3 E-4	63	21,000	110,000	756
1609 (1 mile)	50% Probability	1.3 E-5	6.3	2,100	11,000	76
	Severe	6.8 E-5	33	11,000	59,000	396
8045 (5 miles)	50% Probability	8.4 E-7	0.4	140	720	4.8
	Severe	7.2 E-6	3.6	1,200	6,500	43

^a χ/Q and air concentration data were taken from Table 4.3-25.^bStanding crop of 150 g/m² is assumed.^c2.8 E-5 = 2.8×10^{-5} .

Under severe meteorology, plants would be exposed to 4100 and 1200 $\mu\text{g}/\text{m}^3$ at two and five miles respectively. Damage to vegetation would be expected at these distances because the HF concentrations would exceed the threshold to a distance of five miles. Calculated air concentrations at five miles (140 and 1200 $\mu\text{g}/\text{m}^3$) are also above those (100 to 200 ppb or 80 and 160 $\mu\text{g}/\text{m}^3$)⁴⁴ at which temporary inhibition of photosynthesis has been observed.

The magnitude of damage to vegetation from these accidental releases of HF would be affected by site specific parameters: meteorology, vegetation, and terrain. Duration of exposure is also very important. To the extent possible in a generic evaluation, the estimates of damage are based on circumstances that are not conservative; thus, an order of magnitude overestimate of HF damage should not be assumed for this assessment. On the contrary, credible circumstances have been identified in this analysis which could increase the potential for damage from HF, but there is insufficient information to evaluate the effects on vegetation.

4.3.9.1.3 Accumulation of Fluoride by Forage

Fluoride accumulation in forage was evaluated because this pathway of intake by livestock would constitute a potential hazard associated with accidental releases. Accumulation was assumed over the same interval--0.5 hr. The procedure for calculating fluoride accumulation in forage is based on a deposition velocity of 0.003 m/sec and is described in Appendix II.K. Ground concentrations of fluoride are calculated simply from air concentrations for the residence time of the plume, 1800 sec (0.5 hr) for these accident cases. Fluoride accumulations in forage were estimated at distances of 1000 m, 1 mile, and 5 miles for the 0.5-hr exposure associated with case-1 and -2 accidents, as shown in Table 4.3-26.

Fluoride Accumulation in Forage--Case-1 Accident

For the 50% probability condition and in a single sector, the calculated HF accumulation in forage is 11 and 0.4 ppm for 1000-m and five-mile distances, respectively. These values are well below the concentrations considered un-safe for consumption by cattle. For the severe meteorological condition, the estimated fluoride concentrations are 50 and 28 ppm at 1000-m and one-mile distances, respectively. Continuous consumption of forage having these concentrations of HF should be avoided because bone and dental effects may be manifested in cattle grazing forage possessing fluoride >30 to 50 ppm. But, following a single acute exposure to HF, weathering phenomena reduce the fluoride content of forage.

This decrease in forage, based on first order kinetics, has been described by the function $e^{-0.032D}$, where D is days (see Appendix II.L). This function is used to calculate the postcontamination fluoride content of forage at 1000 m for a weathering period of 35 days--the assumed time interval between hay harvest or the time for pasture regrowth. For maximum initial concentrations of 50 and 28 ppm, from 16 to 8 ppm of fluoride would remain in forage after 35 days. Lower concentrations would be expected for less severe meteorology and at distances greater than 1000 m. After 35 days, the concentration would decrease to less than 30 ppm, a level considered safe for consumption by livestock.

Fluoride Accumulation in Forage--Case-2 Accident

At the 50% probability meteorological condition and to a distance of approximately one mile, a 0.5-hr exposure would result in fluoride accumulation in forage (76 ppm) that exceeds the level considered safe for continuous consumption by cattle. Beyond two miles, the concentration (24 ppm) would be considered safe for consumption by cattle. Although these concentrations in forage may temporarily make grazing on pasture or the harvesting of hay inadvisable, fluorosis effects in cattle can be avoided by implementing countermeasures while weathering decreases the fluoride content.

Substantial accumulation of fluoride in forage is predicted for exposures under severe meteorology. From 756 to 396 ppm of fluoride are estimated in forage at 1000 m and two-mile distances for a 0.5-hr exposure. These concentrations might constitute acute doses if the forage were consumed by cattle. Substantial repression of food intake was observed after 18 days of feeding pregnant beef heifers forage with 600 to 1200 ppm of fluoride.⁹⁰ After 60 days, those receiving 1200 ppm had to be slaughtered because they were too weak to rise. Throughout 18- and 60-day weathering periods (the noted duration causing acute effects), the respective fluoride concentrations of forage--initially possessing 756 ppm fluoride--would range from 756 to 246 ppm (18 days) and from 756 to 110 ppm (60 days). Repression of food intake and corresponding weight loss may be expected if cattle grazed daily on forage possessing these fluoride levels. Livestock would be weakened, but death or slaughter of weak animals would not be expected.

Fluoride Accumulation in Forage from Different Sources

Calculated accumulation of fluoride in forage from an accidental release represents an additional increment above that which normally is taken up by the plants from soil. Natural concentrations of fluoride in vegetation range from 2 to 20 ppm and, for areas not contaminated by industrial sources, the mean concentration was 3.6

ppm.⁹⁰ High soil additions (>200 ppm) will increase the concentration in forage, but the results of numerous surveys⁹⁰ indicate that fluoride added to soil from industrial sources will not significantly increase the soil-derived fluoride in crops. Facilities that process UF_6 were not included in these surveys.

For accidental inputs to an ecosystem, maximum fluoride accumulation by the foliar pathway would exceed that which is taken up by roots by factors of 10 to 100. The foliar route is the critical pathway of accumulation.

Fluoride accumulation in forage in relation to the case-1 accident would be in excess of the chronic values estimated for the routine release (0.3 ppm for annual average meteorology). At 1000 m, total fluoride concentration in forage following an accident would approach 20 ppm at 35 days after contamination, based on accumulation from natural sources (4 ppm) plus residual fluoride from the accident (16 ppm). The total estimated concentration would not exceed 30 to 50 ppm, which is considered unsafe for continuous consumption by cattle. Fluoride accumulation in relation to the case-2 accident substantially exceeds the 30- to 50-ppm threshold, and contributions from natural sources and chronic exposure would not constitute a significant addition.

4.3.9.1.4 Effects of HF on Humans

In contrast to the approach in radiological assessments in which avoidance of effects on man will also protect the environment, the converse is true for exposure to HF; man will be protected by avoiding an effect on the environment. Obviously, an accidental release of HF will damage vegetation and possibly will affect livestock; therefore, the prospects of its effect on man were also considered.

Inhalation of atmospheric HF is one mode of exposure for man. For the case-1 accident, direct effects on man would not be expected for exposures occurring under the 50% probability meteorological condition. Under severe meteorology, the maximum HF concentration of 1.4 mg/m^3 (at 1000 m) is 56% of the limit for a daily 8-hr occupational exposure (2.5 mg/m^3).

For the case-2 accident, the air concentrations approach the threshold that would be hazardous to human health. For the 50% probability meteorological condition, the HF concentration (2.1 mg/m^3) at one mile will approach the limit for daily 8-hr occupational exposure (2.5 mg/m^3). At 1000 m, the maximum air concentration (21 mg/m^3) for severe meteorological conditions would be about 80% of the tolerable exposure for several minutes (25 mg/m^3). Human response would be conjunctival respiratory

discomfort. This condition (21 mg/m^3) would be expected to persist for 30 min in the case-2 accident.

Man is also detrimentally affected by high concentrations of fluoride when consumed as food or water. Effects usually are related to long-term (months to years) intake of food and water containing >50 to 100 ppm HF. If vegetables accumulate HF to the same extent as forage crops, then foodstuffs from gardens in sectors exposed to 50% probability and severe meteorological conditions (case-2 accident) would contain >50 ppm HF for at least one month after contamination. For these circumstances of dose and duration of exposure, toxic effects are not anticipated from ingestion by man, although there are limited case studies on which to base an evaluation for such exposures.

4.3.9.1.5 Limitations Inherent in Environmental Assessment of Impact from HF

Although valid in principle, there may be special considerations for applying the atmospheric transport model to the dispersion of HF. Because it is more buoyant than air, HF may be dispersed more than the calculations predict, and this would tend to reduce concentrations below those estimated in Tables 4.3-25 and 4.3-26. For air concentrations close to threshold levels for effects, the buoyancy effect would decrease the potential for damage. However, for air concentrations considerably above threshold values, the buoyancy factor may only cause larger areas to be damaged.

In the atmospheric transport of accidentally released HF, an exposure time of 0.5 hr (the duration of the release) is assumed for exposures in a single sector. In reality, the exposure time may be either shorter or longer, depending on wind speeds during the period. Increased wind speed would increase transport and dispersion of HF, thereby decreasing exposure times and effects. Conversely, decreased wind speed would extend exposure times and increase the potential for effects. Surface-roughness features of the landscape would exert a drag on the plume and increase its residence time as it moves through an area. Exposure times >30 min may occur, but more information is needed on the dynamics of HF in the boundary layer and plant canopy before longer intervals of exposure can be evaluated for a 30-min accidental release.

The extreme reactivity of HF could cause higher concentration in forage closer to the source. Resultant damage in the close-in environment would be greater, but air concentrations and potential for damage beyond 1000 m would be decreased. Offsite hazards to humans would be decreased accordingly.

Exposure of vegetation to high concentrations of HF may impose practical limits on accumulation and toxicity to livestock and humans. The mechanism of accumulation is not clearly understood, and a saturation of uptake sites may occur when plants are exposed to extremely high air concentrations, such as the $2100 \mu\text{g}/\text{m}^3$ calculated at 1000 m. Whether or not vegetation severely damaged by acute exposure has accumulated or will continue to accumulate HF to levels toxic to livestock has not been determined.

4.3.9.1.6 Summary of Effects of Chemical Releases

The postulated accidental release of HF would cause a localized environmental impact for accident cases 1 and 2. Effects on vegetation (foliar necrosis) and possibly on livestock (dental mottling, repression of food intake) would be expected at 1000 m. Sensitive plant species would be damaged, but there are insufficient data to predict the response of resistant perennial species. The effects on biota would be manifested in a single sector to a distance of possibly five miles if exposure conditions are severe. Vegetation may accumulate HF to levels that will cause fluorosis in livestock if contaminated forage is consumed for periods longer than 18 days. Under extreme circumstances, human health could be affected by HF in the air. Air concentrations of HF at 1000 m would be sufficiently high to cause respiratory irritation.

With the exception of direct effects on plants, there are uncertainties in this assessment because neither experimental observation nor documented experiences are sufficient to confirm estimates of effects at air concentrations and exposure intervals expected from accidents. Use of site-specific meteorology and local vegetation characteristics would enable a more accurate assessment of effects. Research is still needed on complex relationships and effects involving HF at the ecosystem level.

4.3.9.2 Effects of Radioactive Release

Of the nine potential accidents postulated in Section 4.3.8, only four--the rupture of the UF_6 cylinder, the explosion in an oxidation-reduction furnace, inadvertent criticality, and a general facility fire--release significant quantities of toxic material. Of these, the first is a chemical hazard and is considered in Section 4.3.9.1. The explosion and the general facility fire release material in essentially the same isotopic ratio. The doses calculated for the general facility fire can be easily converted to doses due to other postulated accidents in which plutonium in the same isotopic composition is released. This conversion is accomplished by first dividing the tabulated doses by the grams of plutonium released ($1.3 \times 10^{-2} \text{ g}$) for the general facility fire to obtain dose per gram released. Multiplying the dose

per gram released by the number of grams released in a particular accident yields the dose corresponding to that accident. Postulated accidents for which this procedure is valid are: (1) explosion in oxidation-reduction furnace, (2) fire in waste compaction glove box, (3) explosion in metallographic glove box, and (4) failure of quartz high-pressure dissolution tube. Hence, only the general facility fire--which, while less probable, releases about 21 times as much activity as the explosion--must be considered. The criticality accident releases fission products as well as heavy elements and is considered separately.

Potential radiation doses to individuals and to the population were calculated for these two accidents. The methodology and dose conversion factors were the same as used for normal releases (Appendix II.E). The population distribution and the meteorological conditions are those described in Appendix II.I. The releases were assumed to occur from a roof vent at an elevation of 10 m, and no credit was taken for plume rise during the general facility fire, although this would likely mitigate the consequences.

All ground deposition was assumed to occur at the time of cloud passage. Deposition velocities were those used in the dose calculations associated with normal plant operation.

The doses so computed, shown in Tables 4.3-27 and 4.3-28, are well below the guidelines for doses due to accidents set forth by the AEC in 10 CFR 100, that is, 25-rem whole body dose and 300-rem thyroid dose, and are below the usual annual man-rem doses associated with normal releases from water moderated reactors, that is, 10 to 50 man-rems per 1000 MWe.

Table 4.3-27

POTENTIAL DOSES RESULTING FROM ACCIDENTAL RADIONUCLIDE
RELEASES FOR A FUEL FABRICATION PLANT (DOSE TO AN
INDIVIDUAL 1000 m FROM THE STACK)^a

Accident Description	Inhalation and Submersion in the Passing Cloud						
	Total-Body Dose (rems)			Highest Reference Organ Dose (rems)			
	Dose	Critical Radio- nuclides ^b	Percentage Dose from Critical Radio- nuclides	Dose	Reference Organ	Critical Radio- nuclides	Percentage Dose from Critical Radio- nuclides
Criticality Accident	3.4E-3 ^c	Kr-88	11	6.5E-2	Thyroid	I-133	24
		I-134	60			I-134	26
		I-135	15			I-135	46
General Facility Fire	4.2E-3	Pu-238	50	2 E-1	Bone	Pu-238	46
		Pu-239	11			Pu-239	10
		Pu-240	14			Pu-240	14
		Pu-241	25			Pu-241	29

^aStack height = 10 m.

^bA critical radionuclide is one that contributes the largest dose to a reference organ.

^c3.4E-3 = 3.4×10^{-3} .

Table 4.3-28

POTENTIAL POPULATION DOSES RESULTING FROM ACCIDENTAL
RADIONUCLIDE RELEASES FOR A FUEL FABRICATION PLANT

Accident Description	Inhalation and Submersion in the Passing Cloud						
	Total-Body dose (man-rem)			Highest Reference Organ Dose (man-rem)			
	Dose	Critical Radio- nuclides ^a	Percentage Dose from Critical Radio- nuclides	Dose	Reference Organ	Critical Radio- nuclides	Percentage Dose from Critical Radio- nuclides
Criticality Accident	3.3E-1 ^b	Kr-87	10	6.5E0	Thyroid or Bone	I-133	28
		Kr-88	47			I-134	5
		I-134	10			I-135	53
General Facility Fire	5.1E-1	Pu-238	50	22 E0	Bone	Pu-238	46
		Pu-239	11			Pu-239	10
		Pu-240	14			Pu-240	14
		Pu-241	25			Pu-241	29

^aA critical radionuclide is one that contributes the largest dose to a reference organ.

^b3.3E-1 = 3.3×10^{-1} .

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4.4 FUEL-REPROCESSING PLANTS

4.4.1 Introduction

4.4.1.1 Purpose and Need

The fuel in an LMFBR must be removed and replaced periodically. The major reasons for removal are: (1) damage to cladding and other structural materials by fast neutrons; (2) a depletion of plutonium in the core region and a buildup of plutonium in the blanket regions, leading to a reduction in overall breeding gain; (3) a buildup of fission products in the fuel rods, causing a reduction in breeding gain and increased stressing of the cladding; and (4) a need to remove excess plutonium for use in other reactors.

Maximum use of resources requires recovery of valuable materials that remain in spent fuel, regardless of reactor type. Recovery of these materials is accomplished in the reprocessing step. Because a fast reactor requires a large investment of fissile material, recovery of fissile material (plutonium) from LMFBR spent fuels is mandatory.

The model plant basis for this Environmental Statement is a plant having a processing rate of five metric tons of heavy metal (uranium plus plutonium) per day. At a processing rate of 1500 metric tons of fuel per year, about 80,000 MWe of installed LMFBR electrical generating capacity could be serviced. The recovered fuel would contain about 150 tons of plutonium, most of which would be recycled to the original power-producing reactors. The recovered uranium could be recycled to power-producing reactors to be used as fertile material or be stored to satisfy future needs for fertile material.

Although fuel decayed for substantially less than one year can be processed using essentially current technology fission product retention processes and procedures, a preprocessing decay time of one year has been selected as the basis for this Environmental Impact Statement. An on-going development program is directed toward advancing the technology such that fuels decayed for much shorter periods, perhaps as short as 30 days, may be safely processed. The problems involved in short-decay processing, the current state of development, and the program to advance the technology sufficiently to establish the feasibility of short preprocessing decay times will be discussed in Section 4.4.4.4. The objective of this development program is to advance fuel-reprocessing technology so that fuel decayed for as short a time as only 30 days may be reprocessed without undue hazard from routine

or accidental releases. As indicated in Section 11, short-decay reprocessing, although not essential, is economically desirable.

In Appendix II.P, a plant for reprocessing short-decayed fuel is described, estimates of normal and accidental radioactive emissions are given, and estimates are made of the population exposure due to these emissions. This appendix is provided in response to comments from the Mid-America Coalition for Energy Alternatives and the Environmental Protection Agency.*

4.4.1.2 Status of Reprocessing Technology

The Purex solvent-extraction process,¹ which is contemplated for use in separating uranium and plutonium from fission products, has been used for two decades in this country and is in use in other countries where spent fuels are processed. Its use for recovery of LMFBR spent fuels will require only relatively minor modification of some of the process variables to accommodate the relatively high plutonium content of LMFBR fuel. The various steps and equipment that will require optimization for use with LMFBR feed materials include such items as means to reduce the valence of plutonium, choice of solvent-extraction contactor, and determination of optimum solvent-extraction reagent concentration.

The process to be used to prepare the spent breeder fuel for treatment by the Purex process is an adaptation of the chop-leach process² used for preparing light water reactor (LWR) fuels for recovery by solvent extraction. The new problems expected to be encountered in the reprocessing of LMFBR spent-fuel assemblies are related to the unusually rugged construction of the fuel assemblies, the reduced ductility of the metals after neutron irradiation, the potential presence of metallic sodium in the fuel rods, the refractory nature of the UO_2 - PuO_2 , the increased amount of fission product decay heat, and the presence of a substantial quantity of insoluble noble-metal fission products. Nuclear criticality control is a potentially more restrictive problem because of the high plutonium content of the dissolver feed and because of preferential solubility of uranium that occurs with most UO_2 - PuO_2 fuels.

The means to satisfactorily and safely prepare solvent-extraction process feed from LMFBR fuel assemblies is the objective of a continuing development program.³ The various problems have been analyzed, and alternatives have been identified and evaluated. From the analyses of the various problems, from the current state of

*Comment Letters 22, pp. 3-5; 53, p. 82.

information, and from the similarity of this development task to other tasks that have been successfully completed, the conclusion is that means for adequately transforming LMFBR spent fuel into solvent-extraction process feed can be developed during the next decade.

4.4.1.3 Status of Radioactivity Confinement Technology

The technology for confinement of radioactivity in reprocessing plants has been well developed for many years,⁴ with the exception of processes for retention of iodine, noble gas, and tritium. These processes have been the subject of recent intensive development. For the model plant, new processes to recover iodine and noble gases were assumed to be employed. These processes are in an advanced development stage, although they have not been applied in a highly radioactive system.

The general concept for radioactivity confinement used in the model plant analysis has been used for many years and has performed reliably in plants processing LWR fuels decayed for one-half year or more. Short-decay processing of LMFBR fuels will require a marked change in the radioactivity confinement concept involving a reduction in plant effluent volumes by a factor of about 1000. Such reductions in effluent volumes are deemed essential for attainment of very high confinement factors, defined as plant input/plant release, which ultimately will be needed for short-decay reprocessing.

4.4.2 The Sites--Requirements and Considerations

The general requirements for a site for a reprocessing plant are similar to those described for an LMFBR power plant (see Section 4.2); however, some factors relating to siting are decidedly different from power plants. For example, a reprocessing plant draws its major material (spent fuel) from a multiplicity of power plants that are spread over a wide area. Without a significant economic penalty, a reprocessing plant can be located in an area that is relatively remote from centers of power usage, which usually coincides with centers of population and allows greater site selectivity as pertains to prevailing meteorology.

Availability of public transportation facilities for heavy loads by rail, truck, and perhaps barge is a necessity for a reprocessing facility, whereas access by only one of the heavy transportation modes might be acceptable at a power plant site.

Because the quantity of heat to be dissipated from a 5 tons/day fuel-reprocessing plant is much less than from a 1000-MWe power reactor, there is no need for location near a large body of water.

4.4.3 Environmental Impacts of Construction

Activities related to construction of an LMFBR fuel-reprocessing plant will be similar to those required to construct the model LMFBR fuel-fabrication plant and the model LMFBR nuclear power plant. The environmental impacts resulting from the construction activities are expected to be no greater than those resulting from construction of the LMFBR fuel-fabrication plant and the LMFBR power plant if the characteristics of the facility sites are similar. The potential environmental impacts related to construction of LMFBR power plants and fuel-fabrication facilities, respectively, are discussed in Sections 4.2.4 and 4.3.6.

4.4.4 Plant Description

The following assumptions are made regarding the model reprocessing plant. It would be centrally located on a tract of about 1000 acres. It would obtain its water supply (1.4 million gpd) from wells.

The major facilities to be constructed on the site would be: (1) fuel receiving and storage facility, (2) main process building for housing both the reprocessing and waste solidification equipment, (3) radioactive area ventilation air filtration and discharge system, (4) high-level radioactive liquid waste storage, (5) offices, (6) warehouse and shops, (7) steam-generating plant, (8) cooling towers, and (9) a retention basin. These facilities will be described below. (The assumption is that no radioactive solid wastes will be buried on the site; should onsite burial be used, the facilities would be similar to those described in Section 4.6.)

The fuel receiving and storage facility will serve to transfer the spent fuel from the shipping casks to the water-filled fuel storage pool. This transfer would involve withdrawal of the fuel assemblies from the cask into an inert gas-filled shielded cell, transfer of assemblies into a reaction chamber, where residual external sodium would be reacted and removed, and then to the storage pool. The receiving portion of the facility might be about 50 ft square by 100 ft high; the storage pool would be below grade but would have an enclosure over the top that would be about 30 ft wide, 100 ft long, and 12 ft high.

The main process building may be about 200 ft wide, 150 ft high above the foundations, and 300 ft long. All process and analytical functions will be accomplished in this building. It will house a number of concrete-shielded cells having walls of thickness between 3 and 5.5 ft. Maintenance operations in those cells in which undissolved fuel is handled, and in which maximum radioactivity levels are encountered, will be accomplished entirely by remote means through use of electro-mechanical devices and remote, as well as direct viewing systems. In those cells that house lower activity-level operations, the method of maintenance is optional. Personnel entry into these cells for maintenance can be permitted after equipment decontamination, provided suitable precautions are taken. All process operations will be accomplished remotely, with the possible exception of those that relate to the treatment and handling of the product. In those cases, personnel will be isolated from the material by a contamination barrier. In all cases, the process cells are ventilated by drawing in air from surrounding areas.

The radioactive area ventilation-air filtration and discharge system will consist of about 200 HEPA filters in two banks of equal number installed in series, a pair of blowers, and a 100-m-tall stack. In order to provide resistance to damage by tornadoes and other major disruptive forces, the filters and blowers will probably be installed below grade. The space occupied by the filters will probably be about 40 by 100 ft. The fans will occupy a space of about 15 by 30 ft.

The high-level liquid waste storage system will consist of at least three underground storage tanks. If the high-level waste is retained as a liquid for the five years prior to solidification, the active storage capacity will be about 1×10^7 liters; a suitable amount of spare tankage will be available for use in the event of failure of one of the active tanks. These tanks will be water-cooled and made of stainless steel. They will be housed inside stainless steel lined concrete vaults. The liquid level in the tanks will be continuously monitored and sumps located in the bottom will be monitored for the presence of liquid to provide rapid indication of any leakage of liquid from the waste storage tank. In addition, the areas underneath and around the vault will be monitored for radioactivity. [The method described is comparable to the storage system that is being constructed for the Barnwell Nuclear Fuel Plant as described in Figures 4.4-1 and 4.4-2. The Hanford plant tanks that leaked and contaminated the soil underneath those tanks did not use the double encapsulation system described here (see Appendix II.F, Figure II.F-2). In that case, the nature of the soil, the great distance to the water table, and the slow movement of the water beneath the tanks to the environment served to protect against damage to the environment.]

4.4-6

HIGH-LEVEL LIQUID WASTE STORAGE SYSTEM

Figure 4.4-1

4.4-7

MONITORING SYSTEM FOR LIQUID WASTE STORAGE SYSTEM
SHOWN IN FIGURE 4.4-1
Figure 4.4-2

Offices will be required for about 50 people in a building that will probably be separate from the other buildings. Its design will be consistent with those in general use in the area.

A warehouse and shop will be required. Estimates reveal that this facility will be about 60 ft wide, 20 ft high, and 125 ft long. Its construction will be consistent with that generally employed for such buildings in the area involved.

A steam generation plant, presumably using oil or gas as the source of heat, will be required to supply heat for process fluid evaporation, to make steam to accomplish process fluid transfers, and to control temperature during periods of cold weather. This facility will produce about 150,000 lb/hr of steam. It will have one or more relatively short stacks (perhaps 20 to 30 m tall).

A cooling tower, either wet or dry, will be required to dissipate waste heat. So that the estimate will be conservative, use of a wet cooling tower is assumed, because it produces the more severe environmental impact. It will be about 50 ft by 200 ft by 35 ft high. A dry tower could be used for most, if not all, heat dissipation problems in most localities, although there probably would be some economic penalty.

A retention basin is assumed to provide cooling for warm to hot fluids and to provide time for bacterial action on chemical discharges from the steam generation plant and from the wet cooling towers. The area of the basin is about 15 acres, and its average depth is about 10 ft.

The specific design and arrangement of buildings and other structures are largely dependent on the particular nature of the site. A photograph of a scale model of the Barnwell Nuclear Fuel Reprocessing Plant is provided for reference in Figure 4.4-3.

4.4.4.1 Process Description

The overall process function is to recover the contained plutonium from the spent-fuel assemblies as PuO_2 and the contained uranium as UO_3 . Both recovered streams are to be essentially free of fission products. The PuO_2 product is to be suitable for incorporation into LMFBR fuel pellet preparations and will be shipped to a fuel-fabrication facility. The UO_3 may be either stored or shipped to a fuel-fabrication facility.

4.4-9

SCALE MODEL OF BARNWELL NUCLEAR FUEL REPROCESSING PLANT

Figure 4.4-3

To accomplish the overall function, several processes must be used; they include:

- (1) removal of external sodium from incoming spent-fuel assemblies;
- (2) disintegration of the fuel cladding to permit dissolution of the core and blanket material in nitric acid;
- (3) dissolution of the core and blanket material and adjustment of concentrations to the appropriate values;
- (4) separation of uranium and plutonium from fission products and from each other;
- (5) conversion of product streams to desired form; and
- (6) confinement of radioactivity in appropriate liquid, gaseous, and solid waste streams.

The processes described below for the treatment of spent LMFBR fuels are subject to modification and to substitution by other processes in order to more expeditiously or effectively accomplish the desired objective. In particular, those steps that precede solvent extraction, as well as those relating to fission product retention, are the subject of a significant developmental effort; hence, the selection of preferred process alternatives may well be modified. These modifications are likely to be relatively minor and to be concerned primarily with equipment and with details of operation rather than with changes in the basic process.

4.4.4.1.1 Receiving and Storage

In the model plant under consideration, the long-decayed fuel will be received in a shielded shipping cask cooled with helium. (Short-decayed fuel would probably be shipped in sodium.) The fuel assemblies will have residual sodium (from use in the reactor) on their external surfaces. These assemblies will be removed from the cask and moved to a shielded cell that has an inert atmosphere. Later, the residual sodium will be removed from the exteriors of the fuel assemblies in a sodium-reaction chamber through which the fuel passes before it is stored in a water-filled fuel storage basin. In the chamber, the sodium will be reacted with water vapor entrained in an inert gas, and the solid reaction products will be removed by rinsing with water. Hydrogen resulting from the reaction will be combined with oxygen in a low-temperature catalytic converter. Figure 4.4-4 is a block flow diagram for those operations.

Internal sodium in failed pins that entered the pin through the cladding leak is not likely to be removed, because the reaction products tend to plug the leak and

4.4-11

FLOW DIAGRAM OF FUEL RECEIVING SYSTEM

Figure 4.4-4

inhibit further reaction. Also, the quantity of sodium is small. Failed fuel can be separately canned for storage pending processing.

4.4.4.1.2 Fuel Cladding Disintegration

Each fuel assembly will be transferred from the storage basin to a mechanical processing cell where it will be moved into an enclosure having a controlled atmosphere. In this enclosure, that portion of the assembly composed of the gas plenum section and the top end fitting will be severed from the remainder of the assembly, probably through use of a plasma-arc cutting torch fitted with appropriate devices to render it remotely operable and maintainable. The removal of the plenum will remove bulk internal sodium that may have entered the plenum via a cladding failure.

The remainder of the fuel assembly will be transferred into a shear where the portions of the assembly that contain the axial blankets and the core material will be cut and broken into short lengths (probably about 1 in. long). The bottom portion of the assembly will be withdrawn from the shear-feeding system and, along with the gas plenum section, will be transferred to a metal waste consolidation station. Figure 4.4-5 is a block flow diagram for the above operations and those related to internal sodium reactions.

4.4.4.1.3 Internal Sodium Reaction

The sheared fuel will fall through a chute and will pass through a gas lock that consists of a pair of swing gate valves. The gas lock will function by allowing only one of the two valves to be open at any given time. Water (or perhaps very dilute nitric acid) will be sprayed as a fine mist onto the cut fuel as it rests on the gates. The water will react with the small quantity of sodium that might be present and accessible. Only failed fuel rods will contain sodium, and prior removal of the gas plenums should remove the potential for relatively large amounts of sodium being in with the shear product. The water used for this purpose will be at or near the boiling point. Residual or excess water and reaction products will pass with the fuel through the gate valves and will enter the dissolver where it will be mixed with high acid content dissolver product.

4.4.4.1.4 Fuel Dissolution

On leaving the gas lock, the sheared fuel will pass on into the primary dissolver. Here, it will be contacted with a counter current flow of nitric acid in an

FLOW DIAGRAM FOR FUEL DISINTEGRATION
AND INTERNAL SODIUM REACTION

Figure 4.4-5

essentially continuous operation. The dissolver might be any one of three, or possibly more, different styles of dissolvers but all maintain the fumes, spent cladding, and dissolver product in an enclosure that is sealed from the cell atmosphere. Special dissolver systems to accomplish this step are under development. Regardless of dissolver type, a soluble nuclear poison will be used to preclude attainment of nuclear criticality with the quantity of fissile matter in the dissolver. A small fraction, perhaps as much as 2%, of the PuO_2 contained in the initial feed material may remain undissolved in the primary dissolver and will leave the dissolver as a suspension of finely divided particles in the product stream. (If the plutonium-containing solids are not separate from the cladding hulls, nitric acid containing fluoride can be used on the hulls, if necessary, in order to achieve essentially complete dissolution of all heavy metal.) This stream will be centrifuged to remove the particulate matter. The particulate material will then be subjected to boiling nitric acid that contains fluoride ion, which rapidly dissolves PuO_2 particles. This will be a batchwise operation and will be carried out in a small secondary dissolver. The secondary dissolver product may be centrifuged again to prevent entry of particulate matter into the solvent extraction system; on the other hand, it could be mixed with the centrifuged primary dissolver product if the solvent extraction system can tolerate particles. Dissolution will release noble gas fission products and a fraction of the fission product iodine. The oxidation of uranium from the 4^+ to the 6^+ valence will cause the production of some nitrogen oxides from the nitric acid dissolvent. The production of these oxides can be reduced by use of an oxygen atmosphere in the dissolver. See Figure 4.4-6 for a block flow diagram of these operations.

Before the feed enters the solvent extraction system, its composition will be adjusted, and the quantity of fissile material will be accurately determined.

4.4.4.1.5 Purification of Uranium and Plutonium

The solvent extraction system removes fission products from the uranium and plutonium and separates the uranium and plutonium from each other. The solvent extraction process to be used will be a variant of the Purex process,⁵ which has been used for about 20 years for this purpose. In the Purex process, a water-immiscible solvent (tributyl phosphate in a normal paraffin hydrocarbon diluent) is vigorously mixed with the aqueous dissolver product that contains uranium, plutonium, and fission products. The mixing is accomplished by either pulse columns or centrifugal contactors. Because uranium and plutonium nitrates are

FLOW DIAGRAM OF FUEL DISSOLUTION

Figure 4.4-6

extracted from nitrate solutions more strongly than the various fission products, separation factors in the range of 10^6 to 10^8 can be achieved between the heavy metals and the fission products through use of several solvent extraction cycles. A similar solvent extraction step is used to separate plutonium from uranium. The products will be uranyl nitrate and plutonium nitrate in aqueous solution. A liquid waste stream will be produced which contains most of the fission products and various chemicals used in the process. The gaseous waste from the Purex process will be made up primarily of air that is used in the operation of equipment and in the mixing of solutions plus that which leaks into the equipment. It will be contaminated with small amounts of volatile and entrained particulate radioactive material.

Incidental to the functioning of the Purex process are numbers of evaporation steps and a solvent treatment step for each solvent extraction cycle. See Figure 4.4-7 for a block flow diagram of the major peripheral streams of the solvent extraction system.

4.4.4.1.6 Conversion of $\text{Pu}(\text{NO}_3)_4$ to PuO_2

The model plant is based on use of oxalate precipitation followed by drying and calcination of the precipitate to convert the plutonium to the desired form for shipment and for use in the fuel-fabrication facility. Alternative processes, which could conceivably be used for oxide preparation are hydroxide precipitation, thermal denitration, and any one of several sol-gel processes. Figure 4.4-8 indicates the steps involved in oxalate precipitation and the quantities to be processed per day in a 5 metric-ton/day plant.

Criticality control is achieved by using batch-size limits and by controlling the dimensions of vessels used in the processing. The conversion facility will use local filters on off-gas streams to ensure retention of all except trace quantities of plutonium in the conversion facility. Plutonium collected on local filters will be recycled.

4.4.4.1.7 Conversion of Uranyl Nitrate to Uranium Trioxide

The uranium product from the solvent extraction system is a solution of uranyl nitrate. In order to reduce the difficulty of storage and the eventual return of the uranium to the fuel-fabrication plant, the uranyl nitrate solution is converted to uranium trioxide. Most of the water is removed and the uranium is evaporated

SOLVENT EXTRACTION SYSTEM INFLUENT AND EFFLUENT STREAMS
Figure 4.4-7

CONVERSION OF $\text{Pu}(\text{NO}_3)_4$ TO PuO_2
Figure 4.4-8

to give hydrated uranyl nitrate. The dilute nitric acid from this evaporation is condensed and returned to the system.

The residual uranyl nitrate molten salt is then calcined to uranium trioxide at 320°C:

With current technology this operation is conducted in a bed of uranium trioxide fluidized with superheated steam. The off-gas from the calciner is condensed, and the remaining oxides of nitrogen are used to produce nitric acid by reaction with oxygen and water as shown in Figure 4.4-9.

Thermal denitration is a simple process requiring no added chemicals, but it is not the method currently used to prepare the uranium dioxide for reactor fuels. Present day plants prepare the uranium dioxide by the ammonium diuranate method. Conversion of the uranium product from the fuel recovery plant to uranium trioxide therefore provides a convenient form for safe storage. When the uranium is needed to fuel future reactors, it can be converted to the dioxide or dissolved and converted to a fuel form by other processes.

If the uranium were to be converted to uranium dioxide by the ammonium diuranate method, the ammonium nitrate effluent from the ammonium hydroxide precipitation could be recovered as follows:

- (1) Magnesium hydroxide would be added to the ammonium nitrate solution, and ammonium hydroxide would be removed by distillation:

- (2) The magnesium nitrate could then be calcined to give magnesium oxide and nitric acid, both of which would be recycled in the system. The overall reaction is as follows:

Releases of chemicals from the plant could thereby be minimized.

FLOW DIAGRAM OF URANYL NITRATE
TO URANIUM TRIOXIDE CONVERSION

Figure 4.4-9

4.4.4.1.8 High-Level Waste Solidification

Prior to transfer of high activity level liquid wastes to a Federal repository for permanent disposal, they must be converted to inert, stable solids. Several processes are available that could be employed for this purpose.⁶ Of the available processes, spray calcination followed by in-pot melting to form a borosilicate glass has been assumed for the model plant. Solidification of the liquid waste might occur immediately after the reprocessing operation, or it might be delayed for a number of years to allow decay of the shorter lived fission products prior to solidification.

The spray calciner in-pot melting process converts aqueous high-level waste to a rock-like microcrystalline product.⁷ The equipment required includes a radiant heat spray calciner with integral off-gas filtration, a furnace heated stainless steel pot, and a means for introducing metered quantities of glass frit into the pot along with the waste calcine. The pot is the final container for the solidified waste. New pots are put in place and filled pots are removed as the operation proceeds. The filled pots are transferred into a water-filled pool where they might be stored for several years prior to shipment to the Federal repository, or they might be shipped with little delay. The periods of storage at the reprocessing site will depend upon the particular waste management scheme that is selected as being most advantageous.

In this process, the water and nitrates are removed from the fission products and other metal ions by the high-temperature evaporation and decomposition that occurs in the spray calciner. The product of this operation, a finely divided oxide powder, must be separated from the water vapor and nitrogen oxides. This process is accomplished through use of sintered metal filters, which are periodically cleaned by short pulses of preheated air directed backwards through the filters. Only a small fraction of the filters are cleaned at a time to minimize changes in system pressure during the blowback pulse.

The gas leaving the waste solidification system will contain water vapor, nitrogen oxides, ruthenium oxide, and other volatile materials. These gases will pass through an evaporator system, an acid fractionator, and a water scrubber. The scrubber will probably operate at a pressure somewhat greater than atmospheric. This treatment will reduce the nitric acid, nitrogen oxide, and fission product content of the solidifier off-gas. Following this treatment, the off-gas will be subjected to a caustic scrub, primarily for trace ruthenium removal. It will then pass to the environment via the process off-gas cleanup system.

Details concerning equipment and process for spray solidification and other solidification processes can be found in reference 7. In this reference are flow-sheets for commercial-scale waste solidification plants, observations of various design considerations, and a discussion of waste management economics.

4.4.4.1.9 Preparation of Miscellaneous Solid Wastes for Disposal

Various solid wastes would be produced in the course of operation of a fuel reprocessing facility. These wastes include metal scrap from fuel assembly structures, metallic objects such as pumps and vessels which have been removed from radioactive service, alpha-emitting solid wastes, beta-gamma wastes, and alpha-beta-gamma wastes. These wastes are described in Section 4.5.5. The model reprocessing plant includes facilities for volume reduction and encapsulation of these wastes prior to shipment to a repository.

Waste metals from fuel assembly structures (end pieces, gas plenums, and leached cladding) and from equipment items that have been decontaminated and retired from radioactive service can be reduced in volume either by compaction or melting and casting. Large equipment items will require disassembly or size reduction (probably via cutting) prior to compaction or melting. For purposes of this Statement, compaction to 70% of theoretical density has been assumed. The compacted waste will be placed in steel containers and sealed prior to shipment.

The combustible alpha, beta-gamma, and alpha-beta-gamma emitting solid wastes will be incinerated, the ash leached to remove plutonium, and the residue canned prior to removal from the plant.

4.4.4.2 Gaseous Confinement and Discharge Systems

Radioactive materials, when present in substantial quantity, are usually confined inside a process vessel, which is housed in a shielded cell that is in turn surrounded by a building. The spaces inside all barriers are maintained at controlled subatmospheric pressures. The building usually forms the outermost envelope, and it is operated at a pressure approximately 0.3 in. H₂O lower than atmospheric pressure. This pressure is a higher vacuum than a 30-mph wind could be expected to produce on the lee side of a rectangular building. All openings in the building which communicate either with uncontaminated personnel areas or with the outside are equipped with two doors in series, only one of which can be open at any time.

at bottom v

The vestibule formed by the space between the doors is maintained as an uncontaminated area. The building as the outermost envelope is sometimes dispensed with for those shielded cell surfaces that are free of penetrations. The shielded process cells form the second envelope of confinement and are operated with a vacuum of about 0.7 in. H₂O with respect to the building. The cell exhaust system has a rated capacity of approximately 0.1 cell vol/min (to accommodate explosions or fires without pressurizing the cell), and the cell inleakage rate is limited to approximately 10⁻³ cell vol/min at a 2-in. H₂O differential pressure. Seals that are used to close cracks and crevices are designed to withstand a minimum pressure of 10 in. H₂O. The cell structure and its closures are designed to withstand the pressure that could be generated by any credible accident. Finally, the third envelope is provided by the process equipment in the cell, which is operated at a negative pressure with respect to the cell, of about 10 in. H₂O.

Gases of two different categories that are routinely discharged to the atmosphere are the ventilation air from process cells and other contaminated areas and the off-gas from process vessels. The ventilation air normally has a relatively low radioactivity content and is of large volume, perhaps 100,000 cfm. The process vessel off-gas is usually highly contaminated with aerosols, iodine, and noble gases. The volume of this gas stream would be on the order of 1000 cfm. Processes and equipment to reduce the quantities of the various objectionable constituents that are found in these two gas streams are discussed below.

4.4.4.2.1 Iodine Removal

A total of about 1 kg of iodine must be removed from the off-gas stream per ton of fuel (core plus blankets) processed. The iodine is present in two forms--elemental and combined. The elemental iodine constitutes the bulk of the iodine and is relatively easily removed. The combined iodine is, for the most part, present as organic iodides. Two iodine recovery steps are included in the model plant in order to recover the bulk of the iodine in an easily cooled and stored form, to improve recovery of iodine, and to reduce operating costs. The first recovery process, called Iodox, is effective for removal of both elemental and organic iodides and is used to recover essentially all the iodine. The second recovery process is intended as a "backup" for the Iodox process to remove trace amounts of either elemental or combined iodine.

The Iodox process⁸ is a new method that is now being developed for removing elemental iodine and organic iodides, such as methyl iodide, from gaseous

effluents. The process is in the later stages of development and, on the basis of current test results, appears to be a highly effective means for primary removal of the bulk of the iodine from the dissolver off-gas stream and from the solvent extraction system off-gas stream. The iodine can be isolated as a solid that is probably suitable for storage and for encapsulation in stainless steel cans for shipment to a waste repository.

The steps in the Iodox system are: (1) oxidation of the iodine species to the soluble, nonvolatile iodate form by use of nitric acid (over 18 M) in a bubble-cap column; (2) concentration of the iodine-bearing nitric acid scrub solution in an evaporator; (3) recycle of essentially iodine-free nitric acid condensate from the evaporator to the plant nitric acid system; and (4) transfer of the iodine-nitric acid concentrate from the bottom of the evaporator to a second evaporator, where it is evaporated to dryness to form HI_3O_8 . The condensate from the second evaporator is recycled to the first evaporator. The final evaporation to dryness could be carried out in the waste-storage shipping can.

The Iodox process employs hyperazeotropic concentrations of nitric acid which probably will require equipment constructed of zirconium, but the process has the following advantages over other alternative processes that are possibly applicable for the primary and secondary iodine removal, such as caustic scrubbers, mercuric nitrate scrubbers, or solid sorbents.

- (1) No reagent addition is required. The Iodox system uses nitric acid as a scrubbing medium which can be simply purified and recycled.
- (2) The process is capable of continuously handling kilogram quantities of iodine. It avoids loading and/or regeneration problems associated with any type of solid sorbents.
- (3) It delivers concentrated iodine waste. The stable HI_3O_8 can be temporarily stored as a solution or concentrated slurry for I-131 decay and then solidified by simple evaporation to solid HI_3O_8 , a stable form that has potential advantages for permanent storage of the residual I-129.
- (4) The system is not affected by contaminants in feed gas; the normal contaminants in the feed gas do not limit the total decontamination factors (DF) attainable as contrasted to mercuric nitrate and caustic scrubbers. Caustic scrubbing is ineffective for removal of organic iodides. Mercuric nitrate is effective for removal of short chain organic iodides, but its effectiveness is impaired somewhat in the

presence of long chain aliphatics and is destroyed in the presence of cyclic organics. The presence of organics causes "bleed-off" of previously captured iodine from the mercuric nitrate solution.

The Iodex process has been demonstrated successfully on a laboratory and on a small engineering scale by use of simulated off-gas that contains small amounts of radioactive iodine. Decontamination factors greater than 10^4 have been obtained for removal of both elemental iodine and methyl iodide.

Solid sorbers are useful for removing trace quantities of radioiodine from gas streams. These sorbers are normally used in conjunction with a gas-liquid scrubber system, such as Iodex.

A number of metal-exchanged zeolites have been investigated on a laboratory scale for possible use for iodine removal from gaseous effluents. The adsorbent is prepared by a partial replacement of sodium ions with silver ions in the zeolite. When the adsorbent is maintained at about 200°C, the adsorption efficiency of 2 in. deep beds of the silver zeolites for both elemental iodine and methyl iodide is over 99.9% (DF = 10^3 to 10^4).^{9,10} Other cations, such as those of cadmium, copper, mercury, lead, titanium, and mixed rare-earth elements have also been studied; they are much less effective than silver.

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

- (1) Stability. Iodine is not released at temperatures up to 910°C. Iodine and methyl iodide are removed from air streams at temperatures up to 400°C.
- (2) Nonflammable. Zeolites are inorganic aluminum silicates and are not flammable.
- (3) Low explosion hazard. Zeolites do not react explosively with the oxides of nitrogen.
- (4) Efficient in humid conditions. The efficiency of iodine removal is not materially decreased by 90 to 100% relative humidity.
- (5) Resistant to poisoning (i.e., gross loss in effectiveness due to contamination by trace materials). Zeolites are resistant to the poisons normally contained in air. However, they are readily poisoned by halides such as AsOCl_2 .

- (6) Solid waste product. The adsorbed radioiodine species form a solid insoluble product with silver zeolite, which is desirable from a waste-management standpoint.

Although it has been demonstrated only on a relatively small scale, the efficiency of the silver zeolites appears to be one or two orders of magnitude higher than that of the older type silver reactors used at the Hanford and Savannah River plants. The high cost of silver will probably preclude the use of silver zeolites for removal of the bulk of the iodine, and their use will be confined to that of polishing adsorbers after the bulk-removal treatment steps.

The model plant uses both Iodex and silver zeolite sorption for removal of iodine from the off-gas that leaves all process equipment, including the waste-storage and waste-solidification system. The DF that can be obtained by this equipment, when it is designed and operated properly, is probably of the order of 10^9 or greater; however, there are opportunities for iodine to escape to the environment without passing through the Iodex-silver zeolite treatment system. The major routes for such escape are: (1) through leakage of iodine-containing liquids from the process equipment to the cell atmosphere and then to the environment, and (2) via a process waste water stream that is evaporated and injected into the main air stream. Thus, the overall plant confinement factor for iodine normally will be controlled by the amount that bypasses the Iodex-silver zeolite system.

4.4.4.2.2 Nitrogen Oxides Removal

Nitrogen oxides (NO_x) present in the gases discharged from the Iodex scrubbers will be removed by water scrubbing in the presence of a surplus of oxygen.

4.4.4.2.3 Separation of Noble Gases

Selective absorption in a fluorocarbon solvent¹¹ is an effective method for removing the noble gases krypton and xenon from gaseous effluents; it is not subject to explosions or fire and can be operated for sustained periods at high efficiency with low maintenance requirements. The retained gases are recovered in a form suitable for bottling in gas cylinders. The absorption process takes advantage of the relative solubilities of gases in a fluorocarbon solvent (chlorofluoromethane). Water is removed from the gas in an initial step to prevent ice formation in subsequent steps. The gas is then compressed to about 500 pounds per square inch absolute (psia), cooled to about -4°F , and contacted countercurrently with the liquid solvent in an absorber column. The less soluble gases (including nitrogen

and oxygen) exit from the top of the column, exchange heat with the feed gas, and are discharged to the atmosphere. The solvent from this column, loaded with krypton, xenon, and other soluble gases, is routed into a packed fractionating column operated at 30°F and 45 psia. At the lower pressure and higher temperature, solvent vapor is recycled in the column between the reboiler and condenser, driving the remaining slightly soluble gases in a recycle back to the feed stream and concentrating the more soluble gases, including krypton and xenon, in the liquid solvent flowing down the column. The enriched solvent is routed from the reboiler of the fractionator to a stripper column operated at a temperature of about 12°F and a pressure of about 30 psig. Krypton, xenon, and other soluble gases are vaporized in the stripper and may be collected as a moderately concentrated product. Final concentration and purification will be accomplished through use of molecular sieves (for removal of fluorocarbons) and hot reactive metal (perhaps titanium sponge and copper) traps (for removal of nitrogen and oxygen). Essentially pure solvent, suitable for recirculation to the absorber, is collected in the reboiler of the stripper. The product gases can be collected in pressurized cylinders in a form suitable for storage or shipment to a waste repository.

In pilot plant tests with various concentrations of natural krypton and xenon in air, more than 99.9% of these noble gases were retained and transferred to a product stream that had less than 0.001 of the flow rate of the feed stream.¹¹

Selective absorption is included in the model plant; the krypton-recovery effectiveness is taken as 99%. The process is used only on the off-gas stream from the dissolver and related head-end process equipment.

4.4.4.2.4 Air Filtration

High-efficiency particulate air (HEPA) filters have been used for many years in the nuclear industry to effectively remove radioactive particulates from air streams. A modular HEPA filter has a cross section of 2 by 2 ft, a depth of 1 ft, and a capacity of about 1000 cfm. The modules are formed into banks to achieve the required capacity for filtering air. The filter medium is a pleated mat of woven Fiberglas. By definition, an HEPA filter is an expendable (single-use), extended-medium, dry filter that has: (1) a minimum particle-removal efficiency of no less than 99.97% for 0.3- μ m size particles (the most penetrating particle sizes fall in the 0.1- to 0.3- μ m range), (2) a maximum flow resistance of 1.0 in. H₂O when clean and operated at the rated air-flow capacity, and (3) a rigid casing that extends the full depth of the medium.¹² Apart from electrostatic forces, the four

mechanisms active in the deposition of aerosols in fibrous filters are inertia, interception, sedimentation, and diffusion. Filter efficiency must tend to unity for very large and very small particle radii, and filter efficiency also tends to unity for very large and very small flow velocities.¹³ With the fibrous media-type filter, theory and tests have indicated that total efficiency tends to unity for particles of smaller size than 0.03-um and larger than 0.5-um, and that the most penetrating particle is in the 0.1- to 0.3-um size range. This range appeared to be very important in plutonium work.¹⁴ Tests made by Ettinger^{15,16} under controlled conditions have demonstrated removal factors of over 10^{12} for three stages of filtration, indicating that the addition of filters in series improves the removal efficiency. For field conditions where imperfections in gasketing and frame construction will exist, he recommends that the design value for submicron particle-removal efficiency be 99.9% per stage.¹⁷

The direct exposure of unprotected HEPA filters to hot gases, fires, smoke, and moisture can degrade the integrity of the system and in severe cases cause a loss of containment. A sand filter, last in the series, (see Figure 4.4-10) provides protection from such loss of containment owing to its relative immunity to heat, moisture, and chemicals.

Sand filters¹⁸ are constructed of graded layers of aggregate and sand. The aggregate is in layers of relatively coarse (2-1/2 by 1-1/2 in.) to fine particles (No. 4 to No. 16 mesh). The sand layer is generally about 30 in. thick and is made up of No. 20 to No. 50 mesh sand. The flow through the filter is upward. Sand filters have good resistance to heat, shock, and chemical attack, and the disadvantages are higher cost, higher pressure drop, and lower aerosol-collection efficiency compared with other aerosol filters, such as HEPA filters.

Large fixed-bed sand filters at Hanford have been operated for about 20 years of continuous service without maintenance or replacement and at Savannah River for about 13 years before being shut down because of water inleakage. Based on actual measurements at both installations, the collection efficiencies of the filters are greater than 99%.¹⁸ Penetration-rate tests have demonstrated that a 30-in.-deep sand bed is roughly equivalent to a single HEPA filter. The major advantages of the sand filter are its resistance to damage and its large particulate-holding capacity.

The filtration system assumed for the model plant for treatment of process vessel off-gas consists of two HEPA filters in series followed by a 30-in.-deep sand

FLOW DIAGRAM OF GASEOUS CONFINEMENT SYSTEM

Figure 4.4-10

filter. The sand filter is included in this system because of its potential for providing additional reliability. The filtration system for the ventilation air will consist of two banks of HEPA filters in series. These filtering systems will be equipped with suitable means for determining efficiency; such determinations will be made at intervals sufficiently frequent as to essentially ensure their satisfactory operation. The transmittance of submicron particles through the vessel off-gas and ventilation-air filters under normal conditions is conservatively estimated at 10^{-8} and 10^{-6} , respectively. These transmittance factors take into account a degree of degradation and by-passing. HEPA filters can be, and regularly are, installed and operated to provide a transmittance for 0.3-um particles of 5×10^{-4} per stage, or 2.5×10^{-7} for two filtration stages. Although the filters will be tested after installation and at frequent intervals to establish the adequacy of particle removal effectiveness, a minor probability exists that the filters might be degraded before the next test period or by some unusual circumstance occurring during an accident. To determine the sensitivity of the system to such degradation, 10% of the filters (or filter area) were assumed to be operating at 95% removal efficiency for 0.3-um particles, while the remaining 90% were operating at 99.9% efficiency. This amount of degradation would yield an efficiency of 99.4% per stage, or a transmittance factor for two stages of 3.6×10^{-5} . In subsequent sections of this Statement dealing with the fuel-reprocessing plant, the calculation of the amount of radioactivity released through the cell off-gas system due to an accident was based on both "normal" and "degraded" filters. The value calculated to be released from "normal" filters already allows for a substantial amount of imperfections, and the increased release indicated for degraded filters is a low probability occurrence. Figure 4.4-10 is a block flow diagram of the gaseous confinement system assumed for the model plant.

4.4.4.3 Liquid Confinement Systems and Disposal System

Highly radioactive liquids are normally contained within the process vessels and piping that are located within the shielded cells. These cells are lined with stainless steel and are provided with sumps and return pumps and piping to convey any leakage back into the process.

Steam is used in steam-jet eductors for liquid pumping and in various reboilers for heating and evaporation of liquids. The steam that is used for these purposes will be produced by a steam-driven steam generator. The condensate from the steam that is used for process heat will be returned to the steam-driven steam generator.

(The source of the steam to drive the generator is presumed to be a fossil-fueled

boiler.) Use of the steam-driven generator and recycling of condensate will provide assurance that a leaking heat exchanger will not cause contamination of external water streams.

The cooling water supply system for radioactive streams and vessels, including the waste storage tanks, will be designed similar to the steam supply system; that is, an intermediate heat exchanger will be provided such that the water used for cooling radioactive equipment will be in an enclosed captive circuit. The water that cools the intermediate heat exchanger will be supplied from an evaporative cooling tower, or could be supplied by a dry cooling tower, which would provide an additional confinement barrier. The recirculating water streams, including steam condensate, will be monitored for the presence of radioactivity.

Some water will be added to the process in various chemical makeup streams. Inasmuch as all products and wastes will leave the facility as essentially dry solids, a surplus of water that must be discharged will accumulate. In the model plant, the accumulated liquid is to be first purified to remove chemical constituents and essentially all the radioactive components, with the exception of tritium. (This latter nuclide will be present in the effluent water and is not removable by practical means.) Following purification, the water will be evaporated and routed to the stack where it will mix with the other gas streams being discharged to the atmosphere. Evaporation is effective for removal of all the fission products that appear in the liquid streams with the exception of iodine and tritium. The evaporation process and the iodine removal processes will be described below. The liquid confinement system that is assumed for the model plant is depicted in block flow diagram form in Figure 4.4-11.

4.4.4.3.1 Evaporation

Evaporation is very effective in separating dissolved radioactive solids from waste water; essentially all sizes and types of evaporators have been used in the nuclear industry.¹⁹ However, materials that have vapor pressures appreciably higher than that of water or that combine with water to form high-vapor pressure materials are difficult to separate from water by evaporation. Because of these factors, iodine, ruthenium, and tritium are among the few radioactive materials that are poorly separated from waste water by evaporation. System decontamination factors (DFs), defined as the ratio of feed to condensate concentration, of 10^3 to 10^4 can be expected for nonvolatile radioactive contaminants treated in single-stage evaporators. Similar DFs can be expected for ruthenium under alkaline,

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FLOW DIAGRAM OF LIQUID CONFINEMENT SYSTEM

Figure 4.4-11

but not acidic and oxidizing conditions. The DFs for iodine under alkaline, but not moderate oxidizing or acidic conditions, can be expected to be a factor of 10 to 100 lower than those expected for nonvolatile species. These values assume that the evaporator is well designed, adequately sized, and operated with reasonable skill. An overall decontamination (separation) factor of more than 10,000 between condensate (distillate) and thick liquor (concentrate) is generally expected for nonvolatile radioactive contaminants treated in single stage evaporators with a small volume of bottoms being withdrawn or recycled.

4.4.4.3.2 Iodine Removal

Iodine control is most readily accomplished by limiting its entry into the various process streams. Ideally, all the iodine would be removed from the dissolver product prior to its entry into the solvent extraction system. However, in practice there is no means available to accomplish total removal of iodine from the dissolver product. Lacking the ideal capability for iodine removal from dissolver product, iodine must be removed from liquid streams prior to the discharge of the liquids to the environment. Although in the model plant no liquid is discharged, water is vaporized and is discharged to the environment via the process stack. The quantity of iodine to enter the main process streams will be limited by evolving the bulk of the iodine from the dissolver product and trapping the evolved iodine in an Iodex system. The water to be discharged to the environment as vapor will be treated prior to evaporation to further reduce its iodine content. The processes for accomplishing these two removal steps are described in the following paragraphs.

About 95% of the iodine in the dissolver solution will be evolved by steam-stripping or partial evaporation in the presence of excess N_2O_3 or a mixture of nitrogen oxides (NO_x).²⁰ Additional iodine is evolved by oxidizing the solution with ozone and by a second partial evaporation in the presence of excess N_2O_3 . More than 99% of the iodine has been evolved from fuel solutions by this method. The iodine-containing water-vapor streams are to be routed to the Iodex column, where the iodine will be converted to a nonvolatile form.

Solid sorbents have been used in laboratory studies to remove radioactive iodine from water and from acid solutions with high efficiency. Macroreticular resins absorb iodine efficiently from water or from acid solutions.^{21,22} These resins are hard, insoluble beads of porous polymer. They have discrete pores whose diameters range from 50 to 200,000 Å and are available in a wide range of surface

polarities. Macroreticular resin XAD-12 (Rohm and Haas Company) adsorbed about 99.4% of the iodine from water. The sorbed iodine could be removed from the resin with thiosulfate solution, and the resin could be reused while still maintaining high-iodine-removal efficiency. Macroreticular resin XAD-4 is also effective for removing iodine from nitric acid solutions. Removal efficiencies of about 99% were obtained when 3 M HNO_3 solution that contained 5×10^{-4} M I_2 was treated with the XAD-4 resin. An overall confinement factor for iodine of 10^4 was assumed for the model plant.

4.4.4.4 Process Alternatives

The LMFBR fuel reprocessing development program is committed to the examination of alternative approaches that may result in improved operating economy, safety, and/or reduction in harmful emissions. The greatest potential improvement in economy is contingent upon the development of processes and systems for reducing fractional fission-product releases (principally iodine) to a point that will permit reprocessing of fuel after short decay times. Hence, in addition to seeking optimum solutions to those reprocessing problems not related to decay time, an objective of the development program is an advancement of the total reprocessing technology such that fuels decayed for as little as 30 days can be treated without undue hazard from normal or accidental emissions.

4.4.4.4.1 Product Recovery

No alternative process for oxide fuels has been brought to an advanced stage of development. Pyrometallurgical techniques have been studied, but the work in this area was suspended in favor of concentrating development effort on aqueous processes, which have been applied in government facilities for over three decades and are in wide use in fuel-reprocessing facilities throughout the world.

Alternatives exist within the general shear-leach-Purex purification process which bear primarily on methods of accomplishing a specific objective. They do not significantly affect the quantity or the nature of effluents or the safety of the operation. All the process steps described earlier have not been fully developed. Those that require additional development will be fully investigated by laboratory tests, by nonradioactive pilot-plant operation, and by use in a radioactive pilot plant whose feed material is spent LMFBR fuel assemblies from a test or demonstration reactor.

4.4.4.4.2 Emissions Control

Several existing and potential alternatives might be used to control emissions. Table 4.4-1 below lists alternative emissions-control processes. Generally, these processes are considered to be less desirable (usually because of lesser effectiveness or reliability) than the emissions-control processes previously described.

One of the alternative processes under development is intended to collect the bulk of the contained tritium found in irradiated fuels prior to fuel dissolution. Thus, dilution of the tritium in the total water inventory of the facility would be avoided. The elements of this tritium-collection process, which is called voloxidation, are oxidation of the chopped fuel in an oxygen-bearing atmosphere at a temperature of about 500°C and dehydration of the effluent gases. The high-temperature oxidation step enhances release of gases that are trapped in the fuel matrices. The tritium will appear in the off-gas as tritiated water that can be removed from the off-gas by conventional gas-drying techniques and then collected as a small volume of tritium-containing water.

A drastic reduction in effluent volume is proposed as the most practical route to attainment of substantially larger confinement factors. If such reductions in effluent volume can be achieved and if the processes already discussed are used to treat the effluents, confinement factors that are orders of magnitude higher than those predicted for the model plant can be achieved. In the model plant, the use of available processes to treat all streams is not practical; hence, some contaminants bypass the retention equipment. However, in a facility having very small volumes of effluents, it is practical to either treat the stream exhaustively or to isolate it from the environment.

Reductions in effluent volumes are thought to be attainable for the two radioactivity-bearing streams, one liquid and one gaseous, that are released to the environment. Under a design concept now being considered, no deliberate inflow of air into the process cells will occur, and, insofar as possible, all liquids will be recycled. The nitrogen atmosphere in the cells will be monitored and oxygen removed as needed to provide a low-oxygen-content nitrogen atmosphere. Gas used in process operations for liquid-level bubblers and for liquid mixing by sparging will be purified and recycled. The total volume of air that inadvertently enters the process cells by leakage will be maintained at a very low level through use of appropriate construction features. All of the gas that leaves the process cells will pass through

Table 4.4-1

ALTERNATIVE EMISSIONS-CONTROL PROCESSES

Function	Process Identification	Reference
Iodine Removal from Gases	Caustic Scrubbing	1
	Silver Nitrate Reactors	2
	Mercuric Nitrate Scrubbing	3
	Charcoal Sorption	4
Iodine Removal from Liquids	Charcoal Sorption	5
Noble-Gas Collection	Cryogenic Distillation	6
	Charcoal Sorption	6
Tritium Collection and Retention	Voloxidation	7

References:

1. "Aqueous Processing of LMFBR Fuels -- Technical Assessment and Experimental Program Definition," ORNL-4436, June 1970, pp. 173-186.
2. W. C. Schmidt, "Removal of Radioiodine by Use of Silver Reactors," Symposium on the Reprocessing of Irradiated Fuels, Brussels, Belgium, May 20-25, 1957, USAEC Report TID-7534, Bk. 1, pp. 371-376.
3. D. E. Ferguson et al., "Chemical Technology Division Annual Progress Report for Period Ending March 31, 1972," ORNL-4794, August 1972, p. 22.
4. R. E. Adams and R. D. Ackley, "Removal of Radioiodine from Gases," Nucl. Safety 9(5): 373-382 (1968).
5. W. E. Unger et al., "LMFBR Fuel Cycle Studies Progress Report for December 1970, No. 22," ORNL-TM-3281, p. 27.
6. J. P. Nichols and F. T. Binford, "Status of Noble Gas Recovery and Disposal," ORNL-TM-3515, August 1971.
7. J. H. Goode, "Voloxidation -- Removal of Volatile Fission Products from Spent LMFBR Fuels," ORNL-TM-3723, January 1973.

high-efficiency systems for removal of noble gases, iodine, ruthenium, nitrogen oxides, and particulates. (The currently available activity-retention processes, plus those currently under development for retention of iodine and noble gases when their development is satisfactorily completed, are thought to be adequate, provided the effluent volume is drastically reduced.) Exhaustive treatment of this type for all gaseous effluents from the process cell is practical for this case, because the total volume of off-gas from the process cell is estimated to be of the order of 100 cfm, instead of the 100,000 cfm used in the model plant design. Use of a system of this nature would essentially negate the external effect of all accidents that occur within the process cells because spills, explosions, and miscellaneous mishaps would only cause contamination of the interiors of the process cell and atmosphere. The low volume of water added to the reprocessing plant and the low gas throughput render retention and isolation of all liquid effluents from the environment an apparently viable means to prevent contamination of the environment from this source. Such a system of liquid retention and isolation would yield a confinement factor of about 10^3 for tritium.

The concept described above has not been used in any aqueous-reprocessing facility, regardless of size or purpose, whether developmental or production. Therefore, the difficulties and costs to be encountered in the implementation of such a plan are not completely known. However, the advantages to be gained are sufficient that the successful application of such a system could reduce the fractional release of entering radioactivity to levels that would permit processing of fuels decayed for relatively short periods, such as 30 days.

4.4.4.4.3 Short Preprocessing Decay Time

Extending the duration of the preprocessing decay period has an effect on fuel cycle cost and on the time required to double the fissile inventory necessary to sustain a reactor (see Section 11, Cost-Benefit Analysis). Hence, the variation of preprocessing decay time is an important process alternative. The difficulty of reprocessing spent fuel increases as the preprocessing decay time decreases because of the presence of short-lived nuclides that contribute to both fission product heat dissipation and confinement problems. The necessary confinement requirements are related to the relative biological hazard of the various fission products. Increases in amounts of volatile fission products significantly aggravate the confinement problem, whereas increases in the nonvolatile fission product content of fuel are of concern primarily for reasons of fission product decay heat. The

gaseous or volatile fission products present in vastly greater quantity at short decay times are I-131, Xe-133, and Ru-103.

The solvent extraction system and its feed preparation complex are affected to a degree by reductions in the preprocessing decay time as a result of increased fission product heat generation, increased radiation damage to electric motors, and viewing devices, and increased exposure of the process solvent to ionizing radiation. Of these items, the fission product heat generation in the fuel prior to shearing is considered the most significant. However, if the designer of the fuel handling equipment gives due consideration to normal and emergency cooling of the fuel assembly, the removal of the decay heat can be readily accomplished through use of established engineering methods. The increased exposure of the process solvent has been shown to create no unacceptable conditions.

The quantities of radioiodine and xenon increase by seven and nineteen orders of magnitude, respectively, as the decay time is reduced from 365 days to 30 days. The curies of radoruthenium, on the other hand, increase by only a factor of about 7 with a similar decrease in decay time. Of these nuclides, the principal increase in biological hazard results from the increase in I-131. The required iodine confinement factor for 30-day-decayed fuel is of the order of 10^9 . Overall iodine decontamination factors for gas streams that are treated by a combination of Iodox scrubbing and silver-zeolite sorbers can apparently be 10^{10} or more. However, there must be no bypassing of the treatment system, which requires, in a practical sense, that all effluent gas be fully treated. Practical considerations, therefore, call for recycle of essentially all gas, discharging only that represented by the cell inleakage, and for low cell inleakage rates. When such recycle techniques are employed for the gases, the retention of krypton will improve, and the increased quantity of xenon in the feed is not expected to cause significant changes in the discharge rate for noble fission gases. The anticipated confinement factor for xenon with recycle of the gases is expected to be at least 1×10^4 in normal operation.

An ongoing program exists for advancement of the technology to the level required for short preprocessing decay times (such as 30 days). This program is made up of laboratory-scale investigations using both nonradioactive and radioactive materials; individual engineering-scale tests using nonradioactive materials adapted to simulate, insofar as possible, the characteristics of irradiated fuels; nonradioactive engineering-scale tests of most of the various process systems operated in concert as in a production facility; and tests of an engineering-scale facility

using as feed spent fuel from a test and/or demonstration LMFBR power plant. At this time, most of the laboratory-scale investigations have been accomplished and engineering-scale tests of many of the process steps are in progress.

Unresolved problems associated with the preparation of LMFBR fuel for purification by solvent extraction include: (1) those pertaining to transport of the fuel assemblies from the receiving and storage areas to the shear, (2) shearing, (3) dissolution of inhomogeneous mixtures of UO_2 - PuO_2 , (4) nuclear safety, (5) controlled reaction of sodium that may be present in failed fuel rods, and (6) separation of solids from the dissolver products. The generation of heat by fission products at reduced decay periods requires fuel assembly cooling during preshearing transport operations and increases cooling needs in the centrifuge that is used for separating fission product solids from the dissolver product. At 180 days decay, the fission product heat generation in the core section of a fuel assembly is about 1.75 times as great as it is at 365 days decay. This will require that cooling systems be incorporated in the handling equipment to assure that the temperature of the fuel rods does not reach a level that might cause their rupture and consequent release of fission products to the cell atmosphere. The fission product confinement requirements for 180-day-cooled LMFBR fuel are comparable to those of 150-day-cooled LWR fuel, which can be confidently anticipated for plants of present day design. The effects of the various accidents in the chemical processing plant previously discussed will be increased by factors ranging up to 2 by decreasing the preprocessing decay time from 365 to 180 days decay. The development program now in progress is expected to demonstrate solutions to the various problems mentioned above shortly after irradiated demonstration reactor fuel becomes available. Hence, the technology to accommodate a preprocessing decay time of 180 days is expected to be available in the near term. The achievement of the total aims of the current development program will permit the reprocessing of fuel decayed much less than 180 days.

4.4.4.5 Plant Water Use

The plant will require makeup water for its steam generating facility, wet cooling towers, sanitary system, and the chemical makeup to the process equipment.

The total quantity of water required for this plant is 1,400,000 gpd. If dry cooling towers were used, this total might be reduced to 260,000 gpd.

Purity of water as supplied to the plant is not crucial, because the option for use of dry cooling towers would probably be available at most reprocessing sites.

Also, chemical treatment systems are available, at an acceptable cost, to remove objectionable materials from most water supplies.

4.4.4.6 Heat-Dissipation System

Process heat will be dissipated by a cooling tower and a retention basin. The cooling tower probably will be of the wet (evaporative) type, although the dry type might be used if there are circumstances at the site that are contraindicated for evaporative cooling. For control of algae in the wet cooling tower, acrolein might be used; orthophosphate would probably be employed for scale control.

4.4.5 Effluents from Normal Operation

4.4.5.1 Thermal

Approximately 2×10^8 Btu/hr are to be dissipated in a wet cooling tower. Approximately 2×10^7 Btu/hr will be released to the atmosphere via the fossil-fueled steam generators. A negligible amount of heat (less than 1,000,000 Btu/hr) will be dissipated in a retention pond. Water from the retention pond will be discharged to a local stream at essentially ambient temperature.

4.4.5.2 Chemical

4.4.5.2.1 Gaseous

The nonradioactive chemical pollutants in the stack gases result from incomplete recovery of nitrogen oxides released in the course of dissolution and calcination and from the nitrogen oxides formed during combustion of fossil fuels in the steam generators. Table 4.4-2 gives the quantities of the off-gas streams and their chemical constituents.

4.4.5.2.2 Liquid

All liquid chemical waste is discharged to a retention pond. The composition of this stream is shown in Table 4.4-3.

4.4.5.3 Radioactive

Radioactive materials will be released by the model plant to the environment under normal conditions only via a 100-m-high stack. The estimated ground level concentrations in air at the site boundary (1000 m) are listed in Table 4.4-4. The

Table 4.4-2

PARAMETERS FOR NONRADIOACTIVE GASEOUS EFFLUENTS

	Process Stack	Boiler Stack
Standard Air Flow, scfh ^a	8,500,000	1,600,000
Stack Height, m	100	25
H ₂ O Content, lb/hr	16,000	1,500
NO ₂ Content, mg/sec	13,500	1,360
SO ₂ Content, mg/sec	Nil	12
CO Content, mg/sec	Nil	550

^aStandard cubic feet per hour.

Table 4.4-3

COMPOSITION OF WATER DISCHARGED INTO
THE RETENTION POND (~400 gpm)

Component	Concentration, ppm
Acrolein	0.2 to 1.0
Calcium	1.1
CO ₂	39.6
Chloride	5.2
Dissolved Oxygen	4.8
Dissolved Solids, total	80.0
Hardness (CaCO ₃)	5.9
Iron	0.16
Magnesium	0.52
Nitrogen (nitrate)	0.65
Phosphorus (ortho)	0.39
pH	6.5
Potassium	15.2
Sulfate	2.8
Sulfite	0.05
Chemical Oxygen Demand	0.02
Total Organic Carbon	60.0 (max)

Table 4.4-4

ESTIMATED GROUND LEVEL CONCENTRATIONS
OF RADIONUCLIDES IN AIR AT THE SITE BOUNDARY (1000 m)
OF AN LMFBR FUEL REPROCESSING PLANT

Radionuclide	Concentration ($\mu\text{Ci}/\text{m}^3$)	
	Wind Frequency	Wind Frequency
	=0.0533 ^a	=0.127 ^b
H-3	2.25 E-4 ^c	5.34 E-4
Kr-85	1.81 E-4	4.29 E-4
I-129	8.71 E-12	2.07 E-11
I-131	1.79 E-17	4.25 E-17
Ru-103	9.18 E-12	2.18 E-11
Ru-106	1.53 E-9	3.62 E-9
U-234	7.37 E-16	1.75 E-15
U-235	1.17 E-17	2.78 E-17
U-236	3.66 E-17	8.68 E-17
U-238	1.28 E-15	3.02 E-15
Pu-238	2.10 E-11	4.99 E-11
Pu-239	4.25 E-12	1.01 E-11
Pu-240	5.82 E-12	1.38 E-11
Pu-241	6.37 E-10	1.51 E-9
Sr-89	2.90 E-12	6.69 E-12
Sr-90	2.07 E-11	4.92 E-11
Y-90	2.07 E-11	4.92 E-11
Y-91	7.29 E-11	1.73 E-10
Zr-95	2.03 E-11	4.81 E-11
Nb-95	4.31 E-11	1.02 E-10
Te-127m	1.57 E-12	3.72 E-12
Te-129m	1.87 E-14	4.43 E-14
Ag-110m	1.75 E-13	4.14 E-13
Sb-125	9.02 E-12	2.14 E-11
Cs-134	9.37 E-12	2.22 E-11
Cs-137	5.51 E-11	1.31 E-10
Ce-141	3.32 E-13	7.87 E-13
Ce-144	2.22 E-10	5.27 E-10
Pm-147	1.20 E-10	2.86 E-10
Eu-154	6.40 E-13	1.52 E-12
Eu-155	1.67 E-11	3.97 E-11
Am-241	1.31 E-12	3.11 E-12
Am-243	2.40 E-14	5.69 E-14
Cm-242	7.69 E-11	1.82 E-10
Cm-244	6.40 E-13	1.52 E-12

^aFor a sector not opposite the prevailing wind direction.

^bFor sectors opposite the prevailing wind direction.

^c2.25 E-4 is equal to 2.25×10^{-4} .

various fission products that are separated from the heavy metal and not dissipated to the atmosphere are collected and eventually will be shipped to a repository for permanent storage. All the retained fission products except the noble gases will be in solid form when shipped from the plant.

The radionuclides considered in this study were arrived at through use of the ORIGEN²³ computer code, which calculates the relative inhalation hazard for the amounts of all the nuclides present in the fuel. This calculation is obtained by determining the volume of air required to dilute the quantities of the various nuclides found in one ton of fuel to the limit specified in the Radiation Concentration Guide. The radionuclides selected for detailed consideration were:

- (1) the gaseous nuclides tritium, krypton, I-129, and I-131;
- (2) the actinide nuclides whose contribution to the total actinide inhalation hazard is 0.02% or more;
- (3) the fission-product nuclides whose contribution to the total fission-product inhalation hazard is 0.02% or more; and
- (4) the isotopes of uranium (included because of their importance in the fuel cycle).

The amount of radioactive materials contained in the spent fuel charged to the plant was calculated with the ORIGEN computer code²³ for core and blanket material containing, on the average, 9.7% plutonium, having been irradiated to an average of 37,000 MWd/metric ton at a specific power of 49 kW/kg, and having decayed 365 days. The activity per ton is shown in column 2 of Table 4.4-5. The tritium content indicated in this table corresponds to 10% of the calculated yield; the remaining 90% is expected to diffuse through the fuel cladding and to be removed at the reactor as a coolant impurity.²⁴

In column 3 of Table 4.4-5, the total quantities of the various radionuclides to be discharged annually to the environment from a 1500-ton/year plant, as a result of normal operation, are indicated. The plant confinement factors assumed in calculating the column 3 values are shown in the last column. The confinement factors indicated for the volatile materials are based on conservative estimates of the

Table 4.4-5

RADIOACTIVITY CONTENT OF FEED AND GASEOUS EFFLUENT^a
STREAMS FOR FUEL DECAYED ONE YEAR

Radionuclide	Input Radioactivity, Ci/ton	Radioactivity Release, Ci/year	Confinement Factor ^d
<u>Volatile</u>			
H-3 ^c	1.01 E2 ^b	1.52 E5	1
Kr-85	8.12 E3	1.22 E5	1 E2
I-129	3.92 E-2	5.88 E-3	1 E4
I-131	3.1 E-8	4.65 E-9	1 E4
<u>Semivolatile</u>			
Ru-103	4.16 E3	6.2 E-3	1 E9
Ru-106	6.88 E5	1.03	
<u>Uranium</u>			
U-234	1.66 E-1	4.98 E-7	
U-235	2.64 E-3	7.92 E-9	5 E8
U-236	8.24 E-3	2.47 E-8	
U-238	2.87 E-1	8.61 E-7	
<u>Plutonium</u>			
Pu-238	1.89 E4	1.42 E-2	
Pu-239	3.85 E3	2.87 E-3	2 E9
Pu-240	5.24 E3	3.93 E-3	
Pu-241	5.71 E5	4.3 E-1	
<u>Particulates</u>			
Sr-89	6.52 E3	1.96 E-3	
Sr-90	4.67 E4	1.40 E-2	
Y-90	4.67 E4	1.40 E-2	
Y-91	1.64 E4	4.92 E-3	
Zr-95	4.57 E4	1.37 E-2	
Nb-95	9.71 E4	2.91 E-2	
Te-127m	3.54 E3	1.06 E-3	
Te-129m	4.20 E1	1.26 E-5	
Ag-110m	3.93 E2	1.18 E-4	
Sb-125	2.03 E4	6.09 E-3	
Cs-134	2.11 E4	6.33 E-3	5 E9
Cs-137	1.24 E5	3.72 E-2	
Ce-141	7.46 E2	2.24 E-4	
Ce-144	5.18 E5	1.5 E-1	
Pm-147	2.71 E5	8.13 E-2	
Eu-154	1.44 E3	4.32 E-4	
Eu-155	3.78 E4	1.13 E-2	
Am-241	2.95 E3	8.85 E-4	
Am-243	5.41 E1	1.62 E-5	
Cm-242	1.73 E4	5.19 E-3	
Cm-244	1.44 E3	4.32 E-4	

^aPlant processes 1500 metric tons/year at 37,100 MWd/metric ton and average specific power of 49 MW/metric ton.

^b1.01 E2 means 1.01×10^2 .

^cOf the H-3 produced, 10% is assumed to be associated with fuel delivered to the reprocessing plant.

^dConfinement factor = $\frac{\text{Ci processed}}{\text{Ci released}} = \frac{\text{Column 2} \times 1500}{\text{Column 3}}$

recovery that developmental data indicates can be obtained by use of near current design practice. This practice involves no attempt to retain or recover tritium.*

The assumed fractional release of other nuclides is consistent with the experience of the first commercial fuel-reprocessing plant, which is briefly described here-with.

4.4.5.3.1 Brief Discussion of Airborne Effluent Experience at the Nuclear Fuel Services Plant

The Nuclear Fuel Services (NFS) plant at West Valley, New York, is the first commercial fuel-reprocessing plant in the United States, having started operations early in 1966. To date, it is the only U.S. commercial fuel-reprocessing facility from which operating experience can be drawn. The NFS experience with airborne (stack) effluents during active operations extends from the beginning of operations in 1966 to early 1972, when reprocessing operations were suspended in preparation for modifications to increase the capacity of the plant.

During the first half of 1968, there was a spate of adverse experience with the process vessel (including dissolver) air ventilation system.²⁵ As a result of these difficulties, the measured beta particulate releases for 1968 amounted to 1.14 Ci, largely due to short-term releases following mechanical failures in filters used at that time. This is about a factor of 10 higher than 1966 and 1967 experience²⁶ and corresponds to a plant annual average confinement factor for beta particulates of roughly 5×10^7 , the lowest in NFS experience.

Both immediate and longer range steps were taken at once to correct the 1968 troubles.²⁵ The immediate fixes maintained the plant confinement capability, as reflected by measured beta particulate releases of 0.12 Ci for 1969 and 0.18 Ci for 1970, corresponding to annual average confinement factors of 5.6×10^8 and

*The magnitude of the need to control tritium is open to argument. The estimate of the quantity of tritium in the feed is thought to be high by a factor of 2 to 3. In a plant of the design described here, use of the voloxidation process (see Table 4.4-1) when it is developed is expected to reduce the tritium emission rate by a factor of 10 to 100. As indicated in Section 7.3.3, use of water recycle and retention methods in a plant designed for minimal effluent volume is expected to reduce the tritium emission rate by a factor of 1000. The approach taken in this evaluation is to assume that all tritium delivered to the reprocessing plant is released to the environment. If sufficient need exists to reduce the tritium emission rate, the technology is expected to be available by the time the first commercial-scale reprocessing facility is designed for LMFBR spent fuel.

2.6×10^8 , respectively.²⁷ The longer range fix consisted of the design (starting 1968) and installation of a new process vessel air-ventilation system, which was brought into service during October 1970. Among other things, the new system included three banks of HEPA filters in series instead of the previous two. The measured beta particulate release for 1971 was 0.01 Ci^{27} corresponding to an annual average confinement factor of 5×10^9 (or 2×10^{-10} release factor).

The regulatory Technical Specifications^{26,28} governing the operation of the NFS plant require the monitoring and reporting of routine beta and alpha activities in the stack air releases but do not require a separate identification of alpha activities. As a consequence, confinement factors for plutonium and other transuranic nuclides cannot be inferred from routinely reported NFS stack effluents. Confinement factors for airborne plutonium releases can, however, be inferred from direct measurements made during a survey of the NFS plant made by the Public Health Service (PHS) in 1969.²⁹

The 1969 PHS survey included stack air samples from which Pu-238 and Pu-239 activities could be identified. From these measurements, an approximate calculation can be made which places the plant confinement factor for airborne plutonium at about 2×10^9 during a 28-day period of continuous fuel dissolving and reprocessing. From this base point, additional inferences may be made, as follows:

- (1) Assuming that plant performance throughout 1969 remained consistent with the PHS measurements, the plutonium confinement factor was about 3.6 times greater than that for beta particulates (2×10^9 vs 5.6×10^8).
- (2) The improved beta particulate confinement in 1971 (5×10^9), achieved after installation and operation of an improved air ventilation system, implies a plutonium confinement factor of the order of 10^{10} , assuming that the ratio inferred above was maintained.

From these considerations, the confinement factor assumption used in this Statement of 5×10^9 for beta particulates and transuranics, other than plutonium, seems reasonable. The assumption of 2×10^9 for plutonium nuclide confinement is consistent with early NFS experience but is probably conservative in the light of more recent experience.

The rationale for using a somewhat more conservative approach to plutonium retention is that plutonium appears in several portions of the model plant system, giving rise to more opportunities for particulates to be generated in off-gas

streams, than is the case for fission products and other transuranics that are quickly dispatched to high-level waste tanks.

4.4.5.3.2 Prospects for Higher Confinement Factors

In principle, the achievable confinement factors for particulate materials in fuel-reprocessing plants could be much higher than those assumed in this Statement or thus far achieved in practice. There is, therefore, apparent room for improvement. If, for example, the gas reaching the process off-gas filters initially has a liquid content of 40 mg/m^3 and that liquid is composed of 30% fuel solids (both of which are conservative assumptions), the fraction of the plant feed material reaching the process off-gas filters would be 0.01%. In actuality, the concentration of fuel solids reaching the process off-gas filters can be expected to be less than this because of the liquid scrubbing of the off-gas for iodine removal, which will also serve to reduce the quantity of solids in the process off-gas. However, if the conservative value of 0.01% of the feed material reaching the process off-gas filters is used in combination with a transmittance of 1×10^{-8} for the process off-gas filters, the calculated release of activity via the process off-gas will be $\leq 1 \times 10^{-12}$. The amount lost via the cell ventilation filters should be equally small, because the particulates reaching these filters will probably be less than 10^{-6} of the feed quantity, and the cell ventilation filters exhibit a normal transmittance of $\leq 1 \times 10^{-6}$. Thus, the total calculated loss of nonvolatiles is of the order of 10^{-12} and confinement factors ought to be of the order of 10^{12} .

Sufficient knowledge is not currently available to provide an adequate explanation for the discrepancy between calculated expectations and the confinements achieved in practice. A known fact is that relatively large amounts of moisture and corrosive chemicals in the off-gas streams can degrade filter performance, particularly under conditions wherein condensation could take place in the filters or wherein bonding between filter materials and filter frame could be attacked. Such factors may have contributed to the difficulties experienced with the first NFS air ventilation system. An obvious direction of inquiry would address the question of whether or not the residual moisture and reactive chemical content of off-gas streams must be made vanishingly small in order to achieve optimum filter performance.

Whatever answers may emerge from future research and from additional practical experience, today's assumptions clearly must be guided by the only available experience that is applicable to the model LMFBR fuel-reprocessing plant. This experience indicates that confinement factors for beta particulates and transuranic particulates fall in the 10^9 to 10^{10} range, and not in the 10^{11} to 10^{12} range that

could be expected of a three bank filter system.^{15,16} On the other hand, confinement factors in plutonium (ceramic mixed oxide) fabrication plants will evidently more closely approach the highest expected values (see discussion in Section 4.3.5.3.1).

Various analyses discussed in the following sections show that a fuel-reprocessing plant confinement factor in the 10^9 to 10^{10} range is quite satisfactory and will adequately protect the public from harm, even if severe accidents should occur.

4.4.6 Environmental Effects of Normal Plant Operation

4.4.6.1 Effects of Thermal Effluents

The flow of effluent will be 2.1 cfs at a maximum ΔT of $\sim 13.7^\circ\text{F}$. The effluent will enter a holding reservoir with a storage time of 30 days. If the effluent discharge from the holding reservoir is at a ΔT that is less than 1°F above ambient (as might be expected), no significant effects on aquatic biota would be anticipated.

4.4.6.2 Effects of Chemical Effluents

4.4.6.2.1 Effects of Gaseous Effluents -- NO_2 , SO_2 , CO -- on Biota

Negligible quantities of SO_2 and CO (0.6 and 0.02 g/sec, respectively) will be released due to fuel-reprocessing operations. After dilution by factors of 10^6 to 10^8 , ambient concentrations will be less than 0.1 ppb, and there should be no environmental impact for such low concentrations of SO_2 and CO.

At release rates of 1.4 to 13.5 g/sec of NO_2 (Table 4.4-2), the concentrations in air at ground level will range from $2.8 \mu\text{g}/\text{m}^3$ at 1000 m to $0.2 \mu\text{g}/\text{m}^3$ at five miles (Table 4.4-6). The maximum concentration for annual average meteorological conditions would not exceed 1.5 ppb at 1000 m. Based on dilution factors that are 100 times less for dispersion under severe meteorological conditions (see x/Qs in Tables 4.3-8 and 4.3-9), the maximum NO_2 concentration at 1000 m would be 140 ppb (0.14 ppm) for intermittent short-term exposures.

Concentrations of $\text{NO}_2 > 1$ ppm are required to induce leaf damage.³⁰ Hill and Bennett³¹ report a threshold of 0.5 to 0.7 ppm NO_2 for inhibition of photosynthesis.

The plants recovered and no leaf damage occurred at exposures of 45 to 90 minutes. Thompson et al.³² attributed increased leaf and fruit damage to chronic NO_2 exposures at concentrations of 0.25 and 0.5 ppm. They suggested that the threshold

Table 4.4-6

GROUND LEVEL CONCENTRATIONS OF NO₂ IN AIR

Distance (m)	From Process Stacks ^a		From Boiler Stacks ^a		Total NO ₂ Concentration in Air (μg/m ³)
	X/Q ^b (sec/m ³)	NO ₂ Concentration in Air (μg/m ³)	X/Q ^b (sec/m ³)	NO ₂ Concentration in Air (μg/m ³)	
1000	4.7 E-8	0.6	1.6 E-6	2.2	2.8
1609 (1 mile)	3.1 E-8	0.4	7.8 E-7	1.0	1.4
8045 (5 miles)	1.0 E-8	0.1	6.5 E-8	0.1	0.2

^aSource strengths are 13.5 and 1.4 g/sec for process and boiler stacks, respectively.

^bFrom Table II.I-2 of Appendix II.I; dilution factors for a 10 m release are used to calculate air concentration for boiler-stack effluents.

for these effects was below 0.25 ppm. Existing data indicate that effects on plants may be manifested when the concentration of NO_2 in air exceeds 0.25 to 0.5 ppm.

Ambient NO_2 concentrations for exposures related to routine releases and annual average meteorological conditions are two to three orders of magnitude less than those at which effects have been observed experimentally. For exposures under severe meteorological conditions, the maximum expected NO_2 concentration is 0.14 ppm, a concentration that is a factor of 2 to 5 lower than probable threshold concentrations. No detrimental environmental impact on vegetation from direct exposure to NO_2 is expected in conjunction with NO_2 releases from the fuel-reprocessing facility.

Direct effects on animals and humans resulting from exposure to the calculated concentration of NO_2 are not expected. Acute effects in man are manifested at NO_2 concentrations above 10 ppm.³³ Animals appear to be affected at concentrations higher than those that affect man. The Federal Ambient Air Quality Standard³⁴ for NO_2 is 0.05 ppm ($100 \mu\text{g}/\text{m}^3$). For annual average meteorological conditions, the calculated maximum concentration ($2.8 \mu\text{g}/\text{m}^3$) is well below the standard.

An indirect effect of NO_2 is related to its role in the formation of photochemical smog. Low concentrations of NO_2 can produce large amounts of atomic oxygen which in turn form O_3 and complex compounds which cause eye irritation and crop damage.³⁵ Thus, NO_2 from the fuel-reprocessing facility will contribute to the formation of secondary products in the atmosphere, but the magnitude and environmental impact will depend on site specific meteorology and atmospheric chemistry.

4.4.6.2.2 Liquid

Sanitary wastes are to be discharged over land through a spray-irrigation network. Thus, the only significant chemical discharge will involve orthophosphates from cooling tower blowdown; PO_4 concentrations will be 0.25 ppm in a flow of 2.1 cfs. This effluent will result in the discharge of 1.69 kg of PO_4 per day to the holding reservoir. The holding reservoir might be about 3 m deep with a surface area of approximately five hectares. If the phosphate removal capacity of such a reservoir can be maintained at about $30 \text{ mg of } \text{PO}_4/\text{m}^2/\text{day}$, the phosphate concentration in water discharged from the holding reservoir would meet Federal criteria for phosphate discharges in streams. If the phosphate removal capacity of the reservoir is inadequate, alternate procedures should be followed.

Alternatives might involve either cooling tower treatments that do not use phosphates (or other unacceptable compounds) or the installation of phosphate removal equipment in the system prior to discharge into the holding reservoir.

Residual acrolein concentrations of 0.2 to 1.0 ppm are expected in water discharged to the holding reservoir. Acrolein in such concentrations would be lethal to most organisms after several hours of exposure.³¹ However, reduction of the residual acrolein level during a 30-day holdup time in the storage reservoir could result in negligible levels in the discharge from the reservoir. If the concentrations are of the order of a few parts per billion, or less, the impact on aquatic organisms should be minimal.

4.4.6.3 Effects of Radioactive Effluents on Biota Other than Man

The effects of radioactive effluents on biota other than man have been discussed under Section 4.3.6.3.

4.4.6.3.1 Terrestrial Biota

The dose to terrestrial biota from operation of the LMFBR fuel-reprocessing plant will be from atmospheric releases only, because no radioactive liquid effluents will be released to the environment during normal operations. These doses will be similar in magnitude to the doses to man from atmospheric releases, which are discussed under the impact to man from gaseous effluents (4.4.6.4.1). The conclusions of the BEIR report³⁶ are that no other living organisms are very much more radiosensitive than man. Therefore, no detectable radiological impact is expected in the terrestrial biota from operation of the reprocessing plant.

4.4.6.3.2 Aquatic Biota

Since the LMFBR fuel-reprocessing plant will not release any wastes to natural receiving waters, no adverse radiological effects are expected in aquatic biota.

4.4.6.4 Effects of Radioactive Effluents on Man

Radiation doses to individuals (in millirems) and to the population (man-rems) from the pathways discussed in Appendix II.I were estimated per year of release of radioactive gaseous effluents from normal operation of the LMFBR fuel-reprocessing plant. No significant radioactive liquid effluents will be released from the reprocessing plant. Table 4.4-7 summarizes the estimated total-body doses (millirems/year) to adult individuals at locations where the various exposure pathways result in maximum doses. Table 4.4-8 is a summary of the estimated population doses (man-rems) for a 50-mile radius from exposure to the gaseous effluents released by the plant.

Table 4.4-7

SUMMARY OF THE ESTIMATED DOSES TO ADULTS
AT LOCATIONS OF MAXIMUM EXPOSURE

Exposure Pathway	Location	Dose (millirems)		
		Total Body	GI Tract	Bone
<u>Gaseous Effluents</u>				
Immersion	Boundary ^a	0.0077	0.0077	0.0077
Contaminated Ground Surface	Boundary	0.098	0.098	0.098
Inhalation	Boundary	0.53	0.14	5.0
Ingestion (terrestrial food chain)	Boundary	1.4	16.0	6.4
<u>Liquid Effluents</u>				
(No radioactive liquid effluents)		None	None	None
<u>Direct Irradiation</u>				
From the Plant		Negligible	Negligible	Negligible
Total		2.0	16.0	12.0

^aThe plant boundary is located 1000 m from the stack.

Table 4.4-8

SUMMARY OF THE ESTIMATED POPULATION DOSES

Exposure Pathway	Population Dose (man-rems)		
	Total Body	GI Tract	Bone
<u>Gaseous Effluents</u>			
Immersion	0.17	0.17	0.17
Contaminated Ground Surface	1.6	1.6	1.6
Inhalation	11.0	3.3	83.0
Ingestion (terrestrial food chain)	22.0	260.0	100.0
<u>Liquid Effluents</u>			
(No radioactive liquid effluents)	None	None	None
<u>Direct Irradiation</u>			
From the Plant	Negligible	Negligible	Negligible
Total	35	270	190

4.4.6.4.1 Gaseous Effluents

The fuel-reprocessing plant will release a total of 274,000 Ci/year. Release rates are based on the assumption that the fuel is allowed to decay for one year before it is reprocessed. The two nuclides released as gases, tritium (H-3) and Kr-85, account for essentially all of the total number of curies released.

The effluents will be discharged through a 100-m stack. Plume rise was computed from Holland's equation,³⁷ in which it is a function of wind speed. Plume rise values varied from 79 m for a wind speed of 0.8 m/sec to 4 m for a wind speed of 15.6 m/sec. (See Table II.I-2 for dilution factors.)

The deposition velocity used for Kr-85 was 10^{-6} cm/sec. A deposition velocity of 1 cm/sec was used for each of the 33 radionuclides, which are assumed to be in particulate form or, in the case of the iodine nuclides, as molecules that can react chemically with surface vegetation.

The quantities of each of the radionuclides released from the plant are listed in Table 4.4-4 and resulting doses are given in Table 4.4-9.

Immersion

For the immersion pathway, the maximum total-body dose of 0.0077 millirem/year occurs at the site boundary (1000 m); Kr-85 will contribute over 99% of that dose that is calculated at the skin surface from continuous immersion in the gaseous effluents. The dose to the skin at a depth of 0.04 mm from the Kr-85 is approximately 100-times this whole-body dose.³⁸ The average dose for immersion to individual members of the population within 50 miles of the site is estimated to be 1.7×10^{-4} millirem/year; Kr-85 contributes more than 99% of this dose.

A population dose of 0.17 man-rem is estimated to result annually from the immersion pathway.

Ground Contamination

For exposure to the contaminated ground surface, the maximum individual total-body dose of 0.098 millirem/year (assuming exposure 100% of the time) occurs at the site boundary. Radionuclides that contribute to this dose include Ru-106 (49%), Cs-137 (35%), Cs-134 (5%), and I-129 (3%). The average individual dose from this pathway is 0.0016 millirem/year at 40,000 m (~25 miles) with the radionuclide contributions the same as for the maximum case. Ground contamination is estimated to contribute an annual population total-body dose of 1.6 man-rems.

Table 4.4-9

SUMMARY OF THE ESTIMATED CONTRIBUTIONS TO THE INDIVIDUAL DOSES
RESULTING FROM INHALATION

Organ of Reference	Annual Dose (millirems)		Contribution to Dose (%) ^a								
	Maximum ^b	Average ^c	H-3	Ru-106	Pu-238	Pu-239	Pu-240	Pu-241	Am-241	Cm-242	Cm-244
	Individual	Individual									
Total Body	0.53	0.011	79		10	2	3	5	1		
Bone	5.0	0.083	8		41	10	13	27	1		
Thyroid	0.53	0.011	79		10	2	3	5	1		
Lung	0.56	0.012	75	5	12	2	3		1		
Liver	0.96	0.018	43		30	7	9	7	2	1	1
Kidney	0.92	0.017	45		24	6	7	15	2		1
GI Tract	0.42	0.0087	99								
Muscle	0.53	0.011	79		10	2	3	5	1		
Spleen	0.53	0.011	79		10	2	3	5	1		
Testes	0.53	0.011	79		10	2	3	5	1		
Ovaries	0.53	0.011	79		10	2	3	5	1		

^aContributions to annual dose greater than 1%.

^bThe maximum individual is located at the plant boundary, 1000 m from the stack.

^cThe average individual is located 40,000 m (25 miles) from the stack.

Inhalation

The estimated annual dose of 0.53 millirem to the total body (Table 4.4-7) of an individual at the site boundary is based on an inhalation rate for an average adult of 2×10^4 liters/day.³⁹ Table 4.4-9 summarizes the maximum and average individual total-body and organ doses and percentage contributions from the principal nuclides. The population dose from the inhalation pathway is estimated to be 11 man-rem/year

Ingestion in the Terrestrial Food Chain

For the ingestion pathway, the same assumptions are used as those discussed in Appendix II.I, and Section 4.3.6.4.1. Dose due to tritium ingestion, as estimated in Appendix II.0, includes a contribution via drinking water following rainout on the terrestrial environment. A summary of the maximum (0.37 millirem/year) and average (0.022 millirem/year) individual total-body and organ doses and the principal contributing radionuclides is presented in Table 4.4-10. This pathway contributes an annual total-body population dose of 22 man-rem.

4.4.6.4.2 Liquid Effluents

During the normal operation of the fuel-reprocessing plant, no liquids that contain significant quantities of radioactive materials will be released.

4.4.6.4.3 Direct Irradiation

The fuel-reprocessing plant is designed so that direct radiation at the site boundary that results from plant components and stored radioactivity will be much less than 0.01 millirem/year; therefore, this pathway will not be investigated further.

4.4.6.4.4 Assessment of Estimated Doses

Some perspective on the estimated radiation dose that results from the normal operation of facilities within the LMFBR fuel cycle may be obtained by a review of the ionizing radiation to which an average individual in the United States is exposed. Table 4.4-11 presents the estimated annual average total-body doses received from natural radiation sources⁴⁰ and from medical exposures.⁴¹ From one location to another, the average natural exposure may vary by a factor of 2 or more; from one individual to another, the medical exposure may vary by an order of magnitude or more.

Table 4.4-10

SUMMARY OF THE ESTIMATED CONTRIBUTIONS TO THE
INDIVIDUAL DOSES RESULTING FROM INGESTION OF FOOD

Organ of Reference	Annual Dose (millirems)		Contribution to Dose (%) ^a								
	Maximum ^b	Average ^c	H-3	I-129	Ru-106	Cs-134	Cs-137	Sr-90	Cm-244	Cm-242	Am-241
	Individual	Individual									
Total Body	1.4	0.022	74		2		2	1	4		17
Bone	6.4	0.10	16		2		1	12	13	1	54
Thyroid	2.2	0.036	45	39	1		1	1	2		11
Lung	1.3	0.022	75		2				4	1	17
Liver	2.4	0.039	42	1	1	1	2	1	1	1	50
Kidney	3.4	0.055	30		11		1	1	7		51
GI Tract	16.0	0.26	6		89				1	1	2
Muscle	1.4	0.022	72		2	1	3	1	4		16
Spleen	1.4	0.023	72		2	1	4	1	4		17
Testes	1.4	0.022	72		2		2	1	4		17
Ovaries	1.4	0.022	72		2		2	1	4		17

^aContributions to annual dose greater than 1%.

^bThe maximum individual is located at the plant boundary, 1000 m from the stack.

^cThe average individual is located 40,000 m (25 miles) from the stack.

Table 4.4-11

ESTIMATED ANNUAL AVERAGE TOTAL-BODY DOSES FROM NATURAL
RADIATION AND FROM MEDICAL EXPOSURES IN THE UNITED STATES

Source	Annual Dose (millirems)
Cosmic Rays (natural)	44
Terrestrial Radiation (natural)	
External	40
Internal	18
Medical (man-made)	
Diagnostic	<u>72</u>
Total	174

The maximum individual total dose that results from exposure to the effluents of the reprocessing plant under assumed typical meteorological conditions and population distribution is 1.0 millirem/year to the total body. For the average individual within a 50-mile radius, this dose is 0.019 millirem/year. The calculated maximum individual total dose is less than 0.6% of the average total-body dose received by individuals from natural radiation and medical exposures. Table 4.4-12 summarizes the radionuclide-pathway percentage contributions to the average total-body dose from the fuel-reprocessing plant. The annual population total-body dose from all sources of exposure to the effluents of the LMFBR reprocessing plant is 19 man-rems (Table 4.4-13). This total-body population dose of 19 man-rems can be compared to an annual population dose of 202,000 man-rems from natural radiation and medical exposures (i.e., the annual dose of 202 millirems from Table 4.4-11 multiplied by a population of one million people). The reference organs having the highest estimated dose are the GI tract and bone. A GI tract dose of 16 millirems/year and a bone dose of 12 millirems/year will be received by an individual residing continuously at the site boundary of an LMFBR fuel-reprocessing facility. The calculations involving deposition of radioactive materials within the body are 50-year dose commitments and all of the food ingested is assumed to have been produced at the point of radionuclide deposition. Therefore, these estimates of maximum individual dose are very conservative (i.e., the dose is overestimated).

Table 4.4-12

CALCULATED PERCENT OF AVERAGE INDIVIDUAL TOTAL-BODY DOSE^a

Pathway	Contribution of Radionuclide (%) ^b											Total
	H-3	Kr-85	Sr-90	Ru-106	Cs-137	Pu-238	Pu-239	Pu-240	Pu-241	Am-241	Cm-244	
Immersion		1										1
Ground				2	2							4
Inhalation	25					3	1	1	2			32
Ingestion	<u>47</u>	—	<u>1</u>	<u>1</u>	<u>1</u>	—	—	—	—	<u>11</u>	<u>2</u>	<u>63</u>
Total	72	1	1	3	3	3	1	1	2	11	2	100

^aThe average individual is located 40,000 m (25 miles) from the stack.

^bContributions to annual dose greater than 1%.

Table 4.4-13

CALCULATED POPULATION ORGAN DOSES

Reference Organ	Population Organ Dose (man-rems/year)				
	Immersion	Ground	Ingestion	Inhalation	Total
Total Body	0.17	1.6	22	11	35
Bone	0.17	1.6	100	84	190
Thyroid	0.17	1.6	36	11	49
Lung	0.17	1.6	22	11	35
Liver	0.17	1.6	39	19	60
Kidney	0.17	1.6	55	17	74
GI Tract	0.17	1.6	260	3.3	270
Muscle	0.17	1.6	22	11	35
Spleen	0.17	1.6	23	11	36
Testes	0.17	1.6	22	11	35
Ovaries	0.17	1.6	22	11	35

In the event that the population and food sources are nonuniformly distributed, the average individual dose and hence the population dose may change significantly for the ingestion pathway. An upper limit for total-body dose from the ingestion pathway would prevail if the population's food sources were located near the reprocessing plant site. In this highly unlikely case, the average individual total-body dose from ingestion would approach 0.37 millirem/year to the total body (Table 4.4-10) and the population dose for this pathway would be 370 man-rems.

The preceding assessment of the environmental effect of a fuel-reprocessing plant is based on a site having average meteorology, as defined in Appendix II.I.1. Obviously, all potential sites would not have average meteorological conditions; therefore, an additional analysis was made to assess the effect of a less uniform distribution of the frequency of wind direction. From the wind rose data of Appendix II.I.1, the upper quartile of the wind direction frequency distribution was computed to be 16.2% in each of two adjacent 22.5° sectors. For this situation, the maximum individual doses to the total body and to the organs would be increased by 30%, suggesting that careful consideration should be given to meteorological conditions in the vicinity of LMFBR fuel-reprocessing plants. Alternatively, future improvements in the technology of effluent control systems of the plant, as described in Section 4.4.4, could reduce the importance of this site selection criterion.

4.4.7 Effluents and Environmental Measurement and Monitoring Programs

4.4.7.1 Aquatic Monitoring

Aquatic monitoring has been discussed in Section 4.3.7.1.

4.4.7.2 Terrestrial Monitoring

See Section 4.3.7.1 for discussion of terrestrial monitoring.

4.4.7.3 Radiological Environmental Monitoring Program

At a plant that reprocesses spent reactor fuel in an LMFB program, no liquid releases of radioactive wastes are expected. All releases are expected to be in the form of gaseous effluents and to contain fission products at levels similar to those released in the processing of spent LWR and, in addition, these releases will contain higher concentrations of plutonium and other transuranics.

The routine monitoring program should include all of the items outlined in Table 4.3-21 for fuel-fabrication plants. Measurements should be made on all samples for tritium as well as for gross alpha, beta, and gamma activity. Whenever sufficient levels of gross alpha, beta, or gamma activity are detected, composited samples should be subjected to periodic radiochemical analyses.

In addition to the sampling outlined in Table 4.3-21, provisions should be made at each point where air particulates are sampled to test for tritium (using silica gel absorbers), for iodine (using charcoal cartridges), and for krypton.

4.4.8 Postulated Accidents

4.4.8.1 Natural Origin Accidents

The model plant will be protected from floods by siting and by engineering safeguards such that the risk from this eventuality will be exceedingly low. The vital structures will be designed and constructed to be resistant to tornadoes and earthquakes of greater intensity than can be reasonably predicted for the area.

4.4.8.2 Man-Originated Accidents

All abnormal occurrences not covered by Section 4.4.8.1 (acts of God) are included in man-originated accidents. The plant design will be intended to severely restrict the occurrence and the consequences of the accidents indicated below. These abnormal events have been divided into three categories according to the probability of measurable effect beyond the site boundary.

4.4.8.2.1 Accidents Having no Measurable Effect beyond Site Boundary

The loss of normal electrical supply is expected to have no effect on plant operation, safety, or release of radioactive pollutants, because the facility will have an emergency electrical supply (probably diesel generators) that can provide all the power required for operation for as long as the normal power supply may be interrupted.

Loss of ventilation zone control differential might result in some minor contamination spread within the confines of the facility, but no activity is expected to be released outside the process building(s).

Leakage in the primary shell of a high-level liquid waste storage tank should be detected by a buildup of liquid in the sump of the stainless steel lined vault that houses the storage tank. The contents of the storage tank (see Figure 4.4-2) in question, plus the liquid content of the vault, would then be transferred to a spare storage tank. No activity should escape to the environment (see Appendix II.F for discussion of the Hanford waste tank leak).

A leak in the fuel-storage-pool cooler (heat exchanger) would most likely result in a loss of some of the captive cooling water into the fuel storage pool. Alternatively, storage-pool water might enter the captive-cooling-water circuit, causing an increase in liquid level in the captive-cooling-water surge tank and in the radioactivity of this stream. Changes in liquid level or the presence of radioactivity in the captive-cooling-water circuit would result in an alarm and the need for investigation and corrective action.

Spillage of nitric acid as it is unloaded from rail cars or tank trucks, or as it is handled within the plant, would be locally inconvenient, but it is almost inconceivable that either the nitric acid liquid or the vapors would reach the site boundary in measurable amounts.

4.4.8.2.2 Accidents Involving Release of Small Amounts of Radioactivity

An accident typical of this category would be leakage of a solution from a process vessel or pipe in the acid recovery system that could occur as a result of corrosion or inadequate sealing of a mechanical joint in a pipe. The consequences of a leak in a recovered acid pipe have been examined. The results were as follows:

<u>Element</u>	<u>Feed Equivalent (tons)</u>
Iodine	1 E-5
Ruthenium	1 E-13
Uranium	1 E-18
Plutonium	1 E-17
Other solids	1 E-15

A release of this magnitude is a small fraction of the normal daily emission rate.

Failure of one or both of the Iodex scrubbers would cause an increased amount of iodine to reach the silver zeolite sorption beds. These beds are expected to perform with a removal factor of 10^4 . Under normal operating conditions, with both the Iodex and the silver zeolite systems in operation, the fraction of iodine in the plant feed escaping via these treatment systems will be approximately 10^{-8} ; however, an additional amount of approximately 10^{-4} will escape via water that is evaporated and via miscellaneous losses of radioiodine to the cell atmosphere. Thus, in the event of failure of the Iodex scrubbers, the emission rate for iodine would be expected to increase by about a factor of 2 over the normal emission rate.

4.4.8.2.3 Accidents in which Greater Amounts of Radioactivity Would be Released

The cell ventilation exhaust system will be equipped with two banks of HEPA filters operated in series. The indicated efficiency for this system is 99.9999% for aerosols (see Section 4.4.4.2). Although the filter system will be tested periodically to assure satisfactory performance, there is a possibility that the HEPA filters might degrade from a variety of causes. Because a deterioration in filter effectiveness could conceivably exist at the time of an accident, the amount of each release of radioactivity to the environment resulting from upper-limit accidents is calculated on the basis of a transmittance for submicron particles of 3.6×10^{-5} , as well as the normal transmittance of 10^{-6} .

A fuel assembly might conceivably be ruptured as a result of improper handling procedures and thereby release gaseous fission products to the cell atmosphere. The most probable cause of a massive rupture of fuel rods in the cell atmosphere appears to be the dropping of the assembly in the course of moving it from the cell entry point to the shear feed envelope. The metal fuel rods and the wrapper have little ductility after irradiation and might break if seriously abused. We have assumed that fuel rods containing 100 kg of heavy metal will be ruptured; this quantity of ruptured fuel constitutes about 85% of what may be the total quantity of fuel in an assembly. An upper limit estimate was that 20% of the cesium, 85% of the noble gases, and 100% of the iodine present in the 100 kg of fuel would escape

Table 4.4-14
RADIONUCLIDE RELEASE RESULTING FROM FUEL
ASSEMBLY RUPTURE IN CELL ATMOSPHERE

Nuclide	Quantity Released (Ci)	
	Degraded Filters	Normal Filters
Kr-85	6.9 E2	6.9 E2
I-129	3.9 E-3	3.9 E-3
I-131	3.1 E-9	3.1 E-9
Cs-134	1.5 E-2	4.2 E-4
Cs-137	3.9 E-2	2.5 E-3

to the cell atmosphere. These are considered to be conservative estimates of the amount of fission products available for removal from a fuel rod when ruptured under the temperature conditions that will prevail in a fuel assembly that has decayed for one year. The cesium would be converted to Cs_2O and would be partially removed from the cell ventilation air by the ventilation gas filters. Iodine and the noble gases would not be trapped from the cell off-gas. The calculated quantities of the fission products that would be released to the environment as a result of fuel assembly rupture through degraded and through normal filters are shown in Table 4.4-14.

The leakage of a vessel or a pipe usually is not a source of a substantial discharge of fission products to the atmosphere. Except under very unusual circumstances, the fluid will simply find its way to a sump, and all except a very small fraction of the contained activity will be returned to the process. An extreme case has been considered in which one of the most active streams in the plant leaks out onto a hot surface to produce a powder of all the contained nonvolatile materials and to volatilize the iodine and tritium. All materials are assumed to be airborne and to pass to the cell ventilation exhaust system. (Note: This fraction is probably high by a factor of 10^2 to 10^4 on the basis of work with burning plutonium performed by Stewart.⁴²) The stream was assumed to be the dissolver product, or its equivalent, and we assumed that all the materials associated with 1 kg of heavy metal would be dispersed to the cell atmosphere. In this case, noble gases would not be present, and only 10% of the iodine would be present. (The remaining 90% is assumed to have been previously evolved from this solution.) The calculated quantities of the various significant fission products that would be released to the environment through degraded and through normal filters are shown in Table 4.4-15.

Table 4.4-15

RADIONUCLIDE RELEASE RESULTING FROM LEAK IN VESSEL OR
PIPE CONTAINING HIGHLY RADIOACTIVE MATERIAL^a

Nuclide	Quantity Released (Ci)	
	Degraded Filters	Normal Filters
H-3	1.01 E-1	1.01 E-1
Sr-89	2.35 E-4	6.52 E-6
Sr-90	1.68 E-3	4.67 E-5
Y-90	1.68 E-3	4.67 E-5
Y-91	5.90 E-4	1.64 E-5
Zr-95	1.64 E-3	4.57 E-5
Nb-95	3.50 E-3	9.71 E-5
Ru-103	1.50 E-4	4.16 E-6
Ru-106	2.48 E-2	6.88 E-4
Ag-110	1.41 E-5	3.93 E-7
Sb-125	7.31 E-4	2.03 E-5
Te-127	1.27 E-4	3.54 E-6
Te-129	1.51 E-6	4.20 E-8
I-129	3.92 E-6	3.92 E-6
I-131	3.10 E-12	3.10 E-12
Cs-134	7.49 E-4	2.11 E-5
Cs-137	4.46 E-3	1.24 E-4
Ce-141	2.68 E-5	7.46 E-7
Ce-144	1.86 E-2	5.18 E-4
Pm-147	9.75 E-3	2.71 E-4
Eu-154	5.18 E-5	1.44 E-6
Eu-155	1.36 E-3	3.78 E-5
U-234	5.97 E-9	1.66 E-10
U-235	9.50 E-11	2.64 E-12
U-236	2.96 E-10	8.24 E-12
U-238	1.03 E-8	2.87 E-10
Pu-238	6.80 E-4	1.89 E-5
Pu-239	1.39 E-4	3.85 E-6
Pu-240	1.89 E-4	5.24 E-6
Pu-241	2.05 E-2	5.71 E-4
Am-241	1.06 E-4	2.95 E-6
Am-243	1.95 E-6	5.41 E-8
Cm-242	6.23 E-4	1.73 E-5
Cm-244	5.18 E-5	1.44 E-6

^aBased on fission products associated with 1 kg heavy metal being made airborne as submicron particles.

The probability of an occurrence of a fire in a cell is very small owing to the high flash point of such hydrocarbons as are employed in the process and to the relative absence of ignition sources. However, to examine the consequences of an upper-limit accident involving a fire, the assumption made was that 250 liters of the most active solvent stream in the plant, namely, the first cycle extraction column product stream (which is known as the HAP stream), leaked out into a pool and then was ignited. This stream was assumed to contain about 80 g/liter of heavy metal. Because the single-cycle separation factor for fission products from heavy metal is of the order of 10^5 , except for ruthenium, zirconium-niobium, and iodine, only the actinides plus the just-mentioned fission products were considered. The feed ratio between these fission products and the heavy metal was reduced by the following factors to account for separations occurring in the solvent extraction steps: iodine, 100; ruthenium, 100; and zirconium-niobium, 1000. Because the cell ventilation exhaust system is equipped only with filters, a conservative assumption was that all the iodine would be released. The fire may oxidize some of the ruthenium to volatile RuO_4 ; however, experience in high-temperature oxidation systems has indicated that the volatile material readily reverts to a nonvolatile form at lower temperatures. On this basis, 1% of the contained ruthenium was assumed to remain volatile and to pass through the filters. The zirconium-niobium activity was treated as particulate. On the basis of 50% consumption of oxygen in the cell atmosphere and a maximum loading of submicron particles of about 75 mg/m^3 , the estimate was that no more than about 1.7% of the contained solids will be carried to the filters as submicron particles. These assumptions are considered to be conservative. The calculated quantities of the various isotopes released to the atmosphere through degraded and normal filters are shown in Table 4.4-16. Nuclear criticality is to be guarded against by a multiplicity of methods that will require failure of at least two independent restraints before a situation can occur in which criticality is possible. In most of the nuclear criticality incidents that have occurred to date, the number of fissions occurring before the reaction stops, owing to a change in neutron leakage or capture,⁴³ is about 1×10^{18} . Also, the energy release has been sufficiently gradual that vessels have not been ruptured. A series of experiments with aqueous solutions of U-235 have shown that the first-pulse energy varies as the rate of addition of reactivity and is reduced significantly by the presence of a neutron source.⁴⁴ The mixture of plutonium isotopes found in LMFBR fuels will provide a significant neutron source. Calculations show that a cylindrical vessel made of stainless steel could withstand repeated nuclear bursts in a water solution without being ruptured. Hence, the assumption made was that a nuclear excursion would not result in a rupture of a process vessel. It was further assumed that attainment of criticality is credible only within process vessels, all of which

Table 4.4-16

RADIONUCLIDE RELEASE RESULTING FROM A SOLVENT FIRE

Nuclide	Quantity Released (Ci)	
	Degraded Filters	Normal Filters
Zr-95	5.55 E-7	1.54 E-8
Nb-95	1.18 E-6	3.26 E-8
Ru-103	8.23 E-3	8.23 E-3
Ru-106	1.37 E0	1.37 E0
I-129	7.84 E-6	7.84 E-6
I-131	6.20 E-12	6.20 E-12
U-234	2.02 E-9	5.58 E-11
U-235	3.20 E-11	8.87 E-13
U-236	1.00 E-10	2.77 E-12
U-238	3.48 E-9	9.64 E-11
Pu-238	2.29 E-8	6.35 E-10
Pu-239	4.67 E-5	1.29 E-6
Pu-240	6.37 E-5	1.76 E-6
Pu-241	6.92 E-3	1.92 E-4

Table 4.4-17

RADIONUCLIDE RELEASE RESULTING FROM A NUCLEAR CRITICALITY INCIDENT

Nuclide	Quantity Released (Ci)	
	Dissolver	Other
Br-83	1.25 E-3	1.25 E-3
Kr-83	1.63 E-1	1.63 E1
Br-84m	1.04 E-3	1.04 E-3
Br-84	9.03 E-3	9.03 E-3
Br-85	4.28 E-2	4.28 E-2
Kr-85	2.95 E-6	2.95 E-4
Kr-87	3.78 E-1	3.78 E1
Kr-88	5.95 E-2	5.95 E0
Kr-89	1.65 E1	1.65 E3
I-129	6.70 E-14	6.70 E-14
I-130	2.10 E-6	2.10 E-6
I-131	9.77 E-5	9.77 E-5
Xe-131m	7.10 E-5	7.10 E-3
I-132	1.15 E-2	1.12 E-2
I-133	1.20 E-3	1.20 E-3
Xe-133	2.73 E-2	2.73 E0
I-134	4.27 E-2	4.27 E-2
I-135	5.08 E-3	5.08 E-3
Xe-135	3.95 E-1	3.95 E1
I-136	4.56 E-1	4.56 E-1
Xe-137	5.10 E1	5.10 E3
Xe-138	1.05 E1	1.05 E3
Xe-139	1.89 E2	1.89 E4

are vented to the iodine removal equipment, which is assumed to have equal effectiveness for removal of bromine (i.e., $DF = 10^4$). If the incident were to occur in the dissolver, the off-gas would also pass through a system that would remove at least 99% of the noble gas fission products. The assumption made was that 100% of the noble gases and halogens would be evolved as a result of the nuclear excursion. The amounts of these materials released to the environment for a criticality incident in a dissolver and in other vessels are shown in Table 4.4-17.

[Note: The values presented in Table 4.4-17 do not correspond exactly with the values presented in Table 4.3-24 (fuel-fabrication plant), because the values in the first table pertain to an aqueous system and the values in the second table pertain to a ceramic powder. In the first case, the conservative assumption is made that all noble gases and halogens will be released from the solution, and no credit was taken for decay of any nuclides prior to release from the plant stack. Some nuclides with half-lives <5 min were ignored. Because the system is aqueous, no temperatures significantly above 100°C would be reached; therefore, no evolution of cesium, rubidium, strontium, or other relatively nonvolatile nuclides is expected. The first table also reflects removal of 99.99% of the halogens and, in the case of a criticality incident in a dissolver, 99% of the noble gases. In the second case (fuel-fabrication plant), there is evidence that the solids will retain a significant fraction of certain fission product gases; in particular, all those fission product gases that are formed as a decay product of a nonvolatile nuclide will be retained by the solid. By contrast with the fuel-reprocessing plant, no facilities are available at the fuel-fabrication plant for partial retention of noble gases or halogens.]

The consequences of explosions in three different systems (i.e., the high-level liquid waste concentrator, a plutonium product concentrator, and an iodine scrubber system) were addressed. The probability of such an occurrence in any of these systems will be made small by operational procedures and by process equipment design. The principal sources for explosion in a fuel-reprocessing plant are nitrated organics, mixtures of hydrocarbon vapors and air, and mixtures of radiolytically produced hydrogen in air. Nitration of organics could occur in the first two of the above systems and result in the formation of a material known as "red oil," which can be formed from uranyl nitrate and/or nitric acid solutions mixed with tributyl phosphate solvent⁴⁵ at temperatures exceeding 130°C. The temperature in these systems will be kept below 130°C. The energy per pound of "red oil" is 400 Btus. Explosive mixtures of hydrocarbon vapors and air will be avoided by the use of relatively low vapor pressure hydrocarbon diluents, such as dodecane, and by

maintaining the temperature of these hydrocarbons well below the flash point. Hydrogen-oxygen explosions will be avoided by constantly purging the gas space of each vessel with a sufficient flow of air to assure that the concentration of hydrogen does not approach the minimum explosive concentration in air of 4%. The absence of organics and the continual gross air purging of an Iodex iodine scrubber would seem to leave little opportunity for an explosion; however, determination of the consequences of an explosion is being made on a contingency basis. In all cases, the assumption was that after the explosion a limited volume of air will be laden with droplets of the material in question. Observations made by Walsh and Schlea⁴⁶ indicate that a single right-angle impingement will reduce a liquid aerosol concentration to 10 mg/m³ or less. The volumes of gas laden with 10 mg/m³ of aerosol droplets were taken to be 1000 m³ for the two concentrators and 100 m³ for the iodine scrubber system. (Inasmuch as the system is not designed, the volume can only be estimated.) Ruthenium was treated as a particulate for the calculation of the release resulting from the evaporator explosion, because no condition that would cause oxidation of any significant fraction to RuO₄ is visualized. The iodine collected in the scrubber system will be converted to HI₃O₈, which operational experience indicates has negligible vapor pressure. (Tests in which nitric acid containing iodine⁴⁷ was introduced into the top of an Iodex scrub column exhibited good recovery of iodine from the feed gas.) Therefore, the collected iodine is expected to behave as a nonvolatile particulate. Tables 4.4-18 to 4.4-20 give the calculated quantities of the various isotopes that will be released to the environment by the three explosions.

A catastrophic failure of a Kr-85 storage vessel is an improbable event, because this material is presumed to be contained by steel cylinders of the type commonly used for storing nitrogen, oxygen, and other such gases under pressure. A catastrophic failure could result in the event of gross violation of reasonable practice in the handling of the cylinders or of the presence of an undetected materials flaw of gross proportions. Administrative procedures will be used to reduce the probability of either of these occurrences to a negligible value. On the assumption that krypton will not be separated from xenon, each cylinder will contain about 150,000 Ci of Kr-85; no radioactive xenon will be present at a one-year decay time.

4.4.9 Environmental Effects of Postulated Accidents

Potential doses to individuals and to a population were calculated for postulated accidents in the reprocessing plant in which radioactivity was released through

Table 4.4-18

RADIONUCLIDE RELEASE RESULTING FROM AN EXPLOSION
IN HIGH-ACTIVITY-LEVEL WASTE CONCENTRATOR^a

Radionuclide	Quantity Released (Ci)	
	Degraded Filters	Normal Filters
Ru-103	1.15 E-6	3.20 E-8
Ru-106	1.90 E-4	5.29 E-6
Te-127m	9.79 E-7	2.72 E-8
Te-129m	1.16 E-8	3.22 E-10
Cs-134	5.83 E-6	1.62 E-7
Cs-137	3.43 E-5	9.53 E-7
U-234	4.61 E-14	1.28 E-15
U-235	7.31 E-16	2.03 E-17
U-236	2.28 E-15	6.33 E-17
U-238	7.96 E-14	2.21 E-15
Pu-238	5.12 E-8	1.45 E-9
Pu-239	1.06 E-8	2.96 E-10
Pu-240	1.45 E-8	4.03 E-10
Pu-241	1.58 E-6	4.39 E-8
Sr-89	1.80 E-6	5.01 E-8
Sr-90	1.29 E-5	3.59 E-7
Y-90	1.29 E-5	3.59 E-7
Y-91	4.54 E-6	1.26 E-7
Zr-95	1.26 E-5	3.51 E-7
Nb-95	2.69 E-5	7.47 E-7
Ag-110m	1.09 E-7	3.02 E-9
Sb-125	5.61 E-6	1.56 E-7
Ce-141	2.07 E-7	5.74 E-9
Ce-144	1.43 E-4	3.98 E-6
Pm-147	7.49 E-5	2.01 E-6
Eu-154	4.00 E-7	1.10 E-8
Eu-155	1.04 E-5	2.90 E-7
Am-241	8.18 E-7	2.27 E-8
Am-243	1.50 E-8	4.16 E-10
Cm-242	4.79 E-6	1.33 E-7
Cm-244	4.00 E-7	1.11 E-8

^aBasis: Volume of concentrate per ton of fuel = 1000 liters
 Loss of uranium to waste = 0.1%
 Loss of plutonium to waste = 1.0%.

Table 4.4-19

RADIONUCLIDE RELEASE RESULTING FROM EXPLOSION
IN $\text{Pu}(\text{NO}_3)_4$ CONCENTRATOR^a

Radionuclide	Quantity Released (Ci)	
	Degraded Filters	Normal Filters
Pu-238	1.00 E-5	2.8 E-7
Pu-239	2.09 E-6	5.8 E-8
Pu-240	2.84 E-6	7.9 E-8
Pu-241	3.13 E-4	8.7 E-6

^aBasis: plutonium concentration $\approx$ 150 g/liter.

Table 4.4-20

RADIONUCLIDE RELEASE RESULTING FROM
EXPLOSION IN IODOX SCRUBBER SYSTEM^a

Radionuclide	Quantity Released (Ci)	
	Degraded Filters	Normal Filters
I-129	2.82 E-11	7.84 E-13
I-131	2.23 E-17	6.20 E-19

^aBasis: Volume of nitric acid per ton fuel = 50 liters.

degraded filters (see Tables 4.4-21 and 4.4-22). The methodology and dose factors (Appendix II.E and II.I) used in these calculations were the same as those used to calculate doses due to radionuclide releases from plants during normal operation.

The meteorological conditions assumed to exist during and for a short time after the accident are discussed in Appendix II.I. The assumed population distribution is tabulated in Appendix II.I. These releases were assumed to be made through the 100 m stack.

These meteorological conditions and population distribution were also used in calculating doses due to radionuclide releases from normal plant operation. However, the accidental release was assumed to occur when the wind was blowing in a single direction. Thus, the radioactivity stayed within a single 22.5° sector during the entire diffusion and translation of the radioactivity.

All ground deposition was assumed to occur at the time of cloud passage. Deposition velocities were the same as those used in dose calculations associated with normal plant operation.

The doses calculated here are within the following AEC accident guidelines of 10 CFR 100: 25 rems or less whole-body dose to an individual and 300 rems or less dose to the thyroid of an individual and below the usual annual man-rem dose associated with water-moderated power reactors (i.e., 10 to 50 man-rems/1000 MWe).

Table 4.4-21

POTENTIAL INDIVIDUAL DOSES RESULTING FROM ACCIDENTAL RADIONUCLIDE RELEASES^a

Accident Description	Inhalation and Submersion in the Passing Cloud						
	Total-Body Dose			Highest Reference Organ Dose			
	Dose (rems)	Critical Radio-Nuclides	Percentage Dose from Critical Radio-nuclides	Dose (rems)	Reference Organ	Critical Radio-nuclides	Percentage Dose from Critical Radio-nuclides
Fuel Element Rupture ^b	1.4 E-6 ^b	Kr-85 Cs-134 Cs-137	50 20 27	8.8 E-6	Lung	Cs-134 Cs-137	32 67
Leak of Vessel or Pipe ^c	8 E-5	Pu-238 Pu-240	43 13	3.1 E-3	Bone	Pu-238 Pu-240 Pu-241	43 14 28
Solvent Fire ^c	1.4 E-5	Pu-239 Pu-240 Pu-241	19 26 45	8.9 E-4	Bone	Pu-239 Pu-240 Pu-241	19 26 53
Nuclear Criticality (other vessels) ^d	1.1 E-3	Xe-138 Kr-89 Xe-137	34 55 6	1.1 E-4	GI	Xe-138 Kr-89	65 33
Explosion in HLW ^b	<1 E-6	Am-241 Cm-244	55 15	1.2 E-6	Bone	Am-241 Cm-242	48 14
Explosion (Pu) ^c	1.0 E-6	Pu-238 Pu-240 Pu-241	47 15 26	4.5 E-5	Bone	Pu-238 Pu-240 Pu-241	45 14 30
Explosion (I) ^c	<1 E-6	I-129	100	<1 E-6	Thyroid	I-129	100
Kr-85 Storage	1.3 E-4	Kr-85	100				

^aStack height = 100 m. The highest potential doses to an individual occurred at ≈ 1000 m from the stack.^b1.4 E-6 means 1.4×10^{-6} .^cCalculations are for the degraded-filter case.^dThe criticality accident indicated by "dissolver" in Table 4.4-17 would result in doses ≈ 0.01 times those listed for "other vessels."

Table 4.4-22
POTENTIAL POPULATION DOSES RESULTING FROM ACCIDENTAL RADIONUCLIDE RELEASES^a

Accident Description	Inhalation and Submersion in the Passing Cloud						
	Total-Body Dose			Highest Reference Organ Dose			
	Dose (man-rem)	Critical Radio-nuclides	Percentage Dose from Critical Radio-nuclides	Dose (man-rem)	Reference Organ	Critical Radio-nuclides	Percentage Dose from Critical Radio-nuclides
Fuel Element Rupture ^b	2.9 E-3	Kr-85 Cs-134 Cs-137	50 20 27	1.4 E-2	Lung	Cs-134 Cs-137	32 67
Leak of Vessel or Pipe ^b	1.2 E-1	Pu-238 Pu-240 Pu-241	43 13 24	4.8	Bone	Pu-238 Pu-240 Pu-241	43 14 28
Solvent Fire	2.1 E-2	Pu-239 Pu-240 Pu-241	19 26 45	8.6 E-1	Bone	Pu-239 Pu-240 Pu-241	19 26 53
Nuclear Criticality (other vessels) ^c	2.2 E-1	Xe-138 Kr-89	45 55	1.1 E-1	GI	Xe-138 Kr-89	75 25
Explosion in HLW ^b	1.0 E-4	Am-241 Cm-244	55 15	1.9 E-3	Bone	Am-241 Cm-242	48 14
Explosion (Pu) ^b	1.6 E-3	Pu-238 Pu-240 Pu-241	47 15 26	6.8 E-2	Bone	Pu-238 Pu-240 Pu-241	45 14 30
Explosion (I) ^b	<1 E-6	I-129	100	<1 E-6	Thyroid	I-129	100
Kr-85 Storage	0.38	Kr-85	100				

^aStack height = 100 m.

^bCalculations are for the degraded-filter case.

^cThe criticality accident indicated by "dissolver" in Table 4.4-17 would result in doses $\approx$ 0.01 times those listed for "other vessels."

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4.5 TRANSPORTATION OF RADIOACTIVE MATERIALS

4.5.1 Introduction

This section presents a general analysis of the predicted environmental impact from the shipment of radioactive materials in the LMFBR fuel cycle. The analysis is based on quantities of materials that would be required to maintain one 1000-MWe LMFBR in operation for one year. The materials considered in this analysis are divided into three categories: unirradiated fuel materials and assemblies, irradiated fuel, and waste.

The basis for the quantities shipped,¹ as developed in this analysis, is given in the equilibrium material balance flow chart for the Atomics International (AI) follow-on reactor concept (Figure 4.5-1).

Table 4.5-1 summarizes the material quantities that would be generated and the types of shipments made.

4.5.1.1 Principles of Safety in Transport

Most shipments of radioactive materials move in routine commerce via conventional transportation equipment. Therefore, shipments are subject to the same transportation environment, including accidents, as nonradioactive cargo. Although a shipper may impose some conditions on his shipment, such as speed limitations or providing an escort, most of the conditions to which his shipment is subjected and the probability of his shipment being involved in an accident are not subject to his control. The public and transport workers are protected from radiation during the shipment of radioactive materials by the container designs and limitations on the contents. The limitations are set according to the quantities and types of radioactivity and to the standards and criteria for package design and control.

Primary reliance for safety in transport of radioactive material is placed on the packaging. The packaging must meet applicable Federal and state regulatory standards (see Appendix II.M). The standards require that the packaging shall prevent the loss or dispersal of the radioactive contents, retain shielding efficiency, ensure nuclear-criticality safety, and provide adequate heat dissipation under normal conditions of transport and under specified accident-damage test conditions (i.e., the design-basis accident). The contents of packages not designed to withstand accidents are limited; therefore, the risks from releases that could occur in an accident involving these packages are limited.

EQUILIBRIUM MATERIAL BALANCE FLOWSHEET FOR 1000-MWe LMFBR
 (AI FOLLOW-ON DESIGN)

Figure 4.5-1

Table 4.5-1
A SUMMARY OF MATERIALS AND QUANTITIES SHIPPED FOR A 1000-MWe LMFBR REACTOR

Type of shipment	Probable mode of transport	Quantity shipped per year ^a (kg)	Quantity shipped per package ^a (kg)	Est. gross package weight (kg)	No. of packages per vehicle	Heat generated per package (W)	Est. activity per package (Ci)	Average No. of shipments per year	Est. average shipping distance ^{h,i} (miles)	Shipment destination ^j
Unirradiated material										
UF ₆	Truck	17,335	8,604	15,090	1		3.21	2.01	750	FP
UO ₂	Truck	16,048	97	245	64	2.6×10^{-3}	1.60	2.58	750	FP
PuO ₂	Truck	1,679	9	105	64	81	1.04×10^5	2.91	750	FP
Fresh fuel — core and axial blanket	Truck	16,123	235	1,820	4	218	2.81×10^5	17.15	750	PS
Fresh fuel — radial blanket	Truck	2,702	229	1,820	4	6.3×10^{-3}	3.78	2.95	750	PS
Irradiated material										
Spent fuel — core and axial blanket	Rail	15,462	667	114,000	1	1.33×10^4	3.26×10^6	22.83	750	RP
Spent fuel — radial blanket	Rail	2,669	681	114,000	1	1.83×10^3	4.54×10^5	3.92	750	RP
Type of shipment	Probable mode of transport	Quantity shipped per year (ft ³)	Quantity shipped per package (ft ³)	No. of packages per year	No. of packages per vehicle	Heat generated per shipment (W)	Est. activity per shipment (Ci)	Average No. of shipments per year	Est. average shipping distance ^{h,i} (miles)	Shipment destination ^j
Waste from fuel preparation and fabrication plants										
Alpha waste	Rail	5,000 ^b	^c	^c	^c	22.5	2.79×10^4	5	1,000	FR/BG
Low-level beta-gamma waste	Truck	21,000 ^b	7.4	2,800	64	2×10^{-3}	0.475	45	500	BG
Solid waste from power stations										
Low-level beta-gamma waste	Truck	2,000	7.4	270	64	2.85	475	4.2	500	BG
Tritium	Truck	6.12 ^d	0.16	38	64	1.13	3.2×10^4	0.60	1,000	FR/BG
Solid wastes from reprocessing plants										
Alpha waste	Rail	770 ^b	^c	^c	^c	22.5	2.79×10^4	0.77	1,000	FR/BG
Alpha-beta-gamma waste	Rail	2,000 ^b	25	80	3	1.12	500	27	1,000	FR/BG
Low-level beta-gamma waste	Truck	3,000 ^b	7.4	405	64	2×10^{-3}	0.475	6.3	500	BG
Cladding hulls	Rail	165 ^e	3.5	47	36	10.3×10^3	1.46×10^6	1.3	1,000	FR/BG
High-level waste	Rail	55	6.28	9	12	2.5×10^4	7.8×10^6	0.75	1,500	RSSF
Noble gases	Truck	1.7 ^f	1.7 ^g	1.0	6	1.47×10^3	9.0×10^5	0.164	1,500	NGSF
Iodine	Truck	0.078	0.16	0.5	64	1.0×10^{-3}	1.46	0.0076	1,000	FR/BG

^aAll quantities of materials shipped are given in kilograms of heavy metal.

^bCompacted a factor of 10 from original volume generated.

^cAlpha waste is packaged in 55-gal (7.4-ft³) drums and large boxes; each rail car contains 1000 ft³ of waste.

^dShipped as Ca(O³H)₂.

^eHulls compacted to 8.8 ft³ per ton of fuel.

^fCompressed gas at 2,200 psi.

^gStandard gas cylinder.

^hEstimated distance to one-of-a-kind repository, 1500 mi; between facilities, 750 mi; to multiple burial ground sites, 500 mi.

ⁱDistance of 1000 mi is a compromise between 1500 mi to one-of-a-kind repository and 500 mi to multiple burial ground sites.

^jFP: fabrication plant; PS: power station; RP: reprocessing plant; FR: Federal repository; BG: burial ground, RSSF: retrievable surface storage facility; NSGF: noble gas storage facility.

The Natural Resources Defense Council (NRDC) in its comments in the Draft Statement suggested controlling the package's environment* which would get away from placing total reliance on the shipping container for safety. This could be accomplished, they suggest, by restricting material shipments to a single train (the unit train concept) or controlling vehicle speed. The AEC feels such control, while it may help mitigate accident severity in some cases, should not be relied upon as the only method of ensuring safety to the public. The environment, in fact, cannot be fully controlled since other forms of transportation (such as gasoline trucks) could interact with the transport system carrying radioactive materials. However, future regulations of the type suggested by NRDC may be implemented to provide additional safety regarding the transportation of radioactive materials. Restrictions on vehicle speeds and special routings may be developed.

Protection against 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 are specified for loading and storing packages of radioactive material to keep the exposure of persons to a minimum.

4.5.1.2 Types of Transportation

The four main modes of commodity transport in the United States are truck, rail, aircraft, and barge. Each can be used to transport radioactive material in the fuel cycle, but truck is used most frequently. Many of the packages, particularly those that would carry unirradiated material, would be of convenient size and weight for shipment via truck.

Rail is frequently used to transport containers whose weight is too high to be transported by truck. Such packages include spent-fuel casks, cladding-hull casks, etc. However, alpha wastes may also be shipped by rail. Rail cars used to ship this waste have the advantage of large volume per car, and they provide a high degree of safety because of the design (see Section 4.5.4.1.1). Barges are advantageous primarily in the shipment of massive weights such as spent-fuel casks.

*Comment Letter 38c, p. 32.

Aircraft, while used frequently in the shipment of radiopharmaceuticals (particularly when short half-life material is involved), are not expected to be used for the shipment of radioactive materials in the fuel cycle; therefore, air shipment has not been assessed. Shipping costs are expected to remain higher than for rail or truck, and the high speeds available in air shipment would be of little advantage. Also, there are obvious weight limitations for air shipments.

4.5.2 Unirradiated Fuel Materials and Assemblies

4.5.2.1 Materials and Packaging

Feed materials for the LMFBR fuel cycle could come from one or both of two sources. First, sufficient depleted UF_6 exists to fuel all projected LMFBR reactors between now and the year 2000 with no recycle of product uranium. Product uranium could then remain stored at the reprocessing plant site. Since this is the most likely situation, it was evaluated as the base case. An alternative scheme would be to recycle all product uranium, adding only sufficient UF_6 to make up the uranium burned in the cycle. Also considered is product plutonium that would be stripped of essentially all its fission products and would be shipped to the fuel-fabrication plant.

Following conversion of all or makeup UF_6 to UO_2 , powder materials would be blended, pressed into pellets, and fabricated into fuel and blanket assemblies. These completed assemblies would then be shipped to a reactor.

4.5.2.1.1 Uranium Hexafluoride

Uranium hexafluoride, UF_6 , is a white solid at room temperature. At higher pressures and temperatures, it melts to form a clear, colorless, high-density liquid. Large quantities of depleted UF_6 , remnants of the enrichment programs, are stored at the three U.S. gaseous diffusion plants. The material meets the specifications of low-specific-activity (LSA) material and consequently is shipped as a material whose hazard stems from its chemical, rather than nuclear, properties.

Typically, UF_6 is packaged in either a 10-ton cylinder, about 4 ft in diameter by 10 ft long, with a capacity of 21,000 lb of UF_6 and a nominal gross weight of 11.6 metric tons, or a 14-ton cylinder, about 4 ft in diameter by 12.5 ft long, with a capacity of 28,000 lb of UF_6 and a nominal gross weight of 15 metric tons.

Currently, almost all shipments of UF_6 are made by truck, but, as numbers of shipments increase, rail may be used more frequently. Normally, one 14-ton cylinder or two 10-ton cylinders are shipped per single truck. Four 14-ton cylinders may be shipped by rail car. All shipments are expected to be under exclusive use (full load) arrangements.

In this study, the assumption is made that only 14-ton cylinders are used. Table 4.5-2 gives shipping information. A shipment of UF_6 contained in both 10- and 14-ton cylinders is shown in Figure 4.5-2.

4.5.2.1.2 Uranium Dioxide

Packages exist that are currently approved for shipping low-enriched UO_2 powder in bulk form.² The package shown in Figure 4.5-3 can transport 110 kg of UO_2 powder in six high-strength, water-resistant cardboard drums. These drums are placed in a removable insert that is, in turn, placed into the inner container of the shipping package. Thermal and mechanical protection is provided by a solid fire-resistant foamed material that separates the inner cylinder from the outer drum container. Table 4.5-3 gives shipping information.

Packages of this type which use a phenolic foamed insulating material have been extensively tested^{3,4} and have been shown to meet all requirements of the hypothetical accident conditions specified in the regulations.

It is expected that all product uranium will be stored at the reprocessing plant in the oxide form and that UO_2 product shipments will not be made until all existing quantities of UF_6 have been used up.

4.5.2.1.3 Plutonium Dioxide

Packages currently approved for the shipment of PuO_2 (Figure 4.5-4) have been tested extensively under both hypothetical accident conditions⁵ and extremely severe accident conditions.⁶ Future PuO_2 packages may have the same external appearance but probably will have different interior construction. A hydrogenous, nonflammable insulating material such as gypsum, concrete, or water-extended polyester (WEP) would be used to provide shielding, spacing (for criticality control), and fire protection. The inner cans would meet special form require-

Table 4.5-2**SHIPPING INFORMATION FOR URANIUM HEXAFLUORIDE**

Container	Model 48 F
Outer Diameter	4.0 ft
Length	12.5 ft
Wall Thickness	5/8 in
Material	Carbon steel
Tare Weight	5,200 lb
Net Weight of UF ₆	28,000 lb

Shipping Requirements ^a		
Annual Requirements, uranium	17,335 kg	
Uranium Shipped per Package	8,604 kg	
Packages Shipped per Year	2.01	
Packages per Motor Vehicle	1	
Radioactivity per Package	3.21 Ci	
Vehicle Shipments per Year	2.01	
Average Shipping Distance, one way		750 miles
Shipping Time, one way	1.5 days	

^aFor a single 1000-MWe LMFBR.

4.5-8

SHIPMENTS OF 10- AND 14-TON UF_6 CYLINDERS

Figure 4.5-2

FOAM-LINED UO_2 SHIPPING CONTAINER
 Figure 4.5-3

Table 4.5-3

SHIPPING INFORMATION FOR URANIUM DIOXIDE

Depleted UO_2 Package Specifications	
Type	Double-stacked 55-gal drum
Height	74 in.
Outer Diameter	24 in.
Inner Container Diameter	11.5 in.
Inner Container Height	63.5 in.
Inner Drum Diameter	9.5 in.
Inner Drum Height	9.75 in.
Number of Drums per Package	6
Tare Weight of Package	135 kg
Net Weight UO_2 per Package	110 kg
Shipping Requirements ^a	
Annual Production, uranium	16,048 kg
Net Weight U per Package	97 kg
Powder Density, g U/cm^3	2.0
Packages per Year	165.44
Packages per Vehicle	64
Shipments per Year	2.58
Radioactivity per Package	1.60 Ci
Thermal Power per Package	0.0026 W
Shipping Distance, one way	750 miles
Shipping Time, one way	3 days

^aFor a single 1000-MWe LMFBFR.

Table 4.5-4

SHIPPING INFORMATION FOR PLUTONIUM DIOXIDE

PuO_2 Package Specifications	
Type	55-gal drum
Height	35 in.
Diameter	22.5 in.
Inner Container	4.81 in.
Inner Container Height	18 in.
Usable Volume	0.16 ft ³
Tare Weight of Package	90.9 kg
Net Weight PuO_2	10.24 kg
Shipping Requirements ^a	
Annual Production, plutonium	1679 kg
Pu Shipped per Package	9 kg
Packages per Year	186.55
Packages per Vehicle	64
Shipments per Year	2.91
Radioactivity per Package	1.04×10^5 Ci
Thermal Power per Package	81 W
Shipping Distance, one way	750 miles
Shipping Time, one way	1.5 days

^aFor a single 1000-MWe LMFBFR.

INSULATED SHIPPING CONTAINER
FOR SHIPPING SOLID FISSILE MATERIAL

Figure 4.5-4

ments in accordance with currently proposed regulations,⁷ and two containment barriers probably would be provided. Table 4.5-4 gives shipping information.

4.5.2.1.4 Unirradiated Fuel Assemblies

The assumed LMFBR design specifies a fuel assembly that consists of 217 fuel pins arranged hexagonally in a steel sheath. The assemblies are assumed to be shipped in packages similar to those used in shipping LWR fuel. Both the core plus axial blanket and radial blanket assemblies would be sealed in a plastic bag and placed in a metal container. This package would support the assembly along its entire length and would protect it from those vibrations normally incident to transport. Two of these containers would be placed in a protective overpack. Four overpacks would be placed on a trailer. Shipment would be by motor freight.

Table 4.5-5 describes the package. The fuel assemblies would contain some gamma ray and neutron sources. The amount of shielding required during shipment, if any, has not yet been determined. A hydrogenous, nonflammable insulating material such as gypsum, concrete, or WEP might be used as part of the overpack to provide the shielding necessary to meet the Department of Transportation (DOT) requirements.

The AI "follow-on" fuel-assembly design was used as a model for this analysis (Figure 4.5-5). The core assemblies would include upper and lower depleted-uranium blankets. The radial-blanket assemblies would also contain depleted uranium. Table 4.5-5 gives shipping specifications.

The overpack for the LMFBR fuel, described above, is expected to look something like the cylindrical package for shipping unirradiated LWR fuel shown in Figure 4.5-6.

4.5.2.2 Transport Conditions

Feed materials, including UF_6 , fresh fuel, and product material, are expected to be shipped by common or contract carriers under special provisions that would permit no intermediate off-loading, storage, or intervehicular transfers en route. In some instances, private carriage might be used; in which case the facility from which the shipment was being made might use its own vehicles.

4.5-13

FUEL SUBASSEMBLY (ATOMICS INTERNATIONAL "FOLLOW-ON" DESIGN)

Figure 4.5-5

Table 4.5-5

SHIPPING INFORMATION FOR UNIRRADIATED
CORE AND BLANKET ASSEMBLIES

Unirradiated Fuel Package Specifications	
Type	Cylindrical overpack
Length	17 ft
Diameter	4 ft
Material	Stainless steel
Tare Weight	2500 lb
Net Weight	1500 lb
Fuel Assemblies per Package	2
Activity per Package	2.81×10^5 Ci
Thermal Power per Package	218 W

Shipping Requirements ^a	
Core and Axial Blanket Assemblies Shipped per Year	137
Heavy Metal Shipped Annually (U + Pu)	16.123 tons
Radial Blanket Assemblies Shipped per Year	23.5
Heavy Metal Shipped Annually, U	2.702 tons
Shipping Packages per Vehicle	4
Assemblies per Vehicle	8
Shipments per Year	20.06
Shipping Distance, one way	750 miles
Shipping Time, one way	1.5 days

^aFor a single 1000-MWe LMFBR.

4.5-15

A SHIPPING CONTAINER FOR SHIPPING
UNIRRADIATED LWR FUEL ASSEMBLIES

Figure 4.5-6

4.5.2.3 Effects on the Environment under Normal Conditions

4.5.2.3.1 Heat Load

The heat load per shipping container for all unirradiated LMFBR materials (Table 4.5-1) is expected to have essentially no impact on the environment. The temperature of the outer surfaces of these packages would be no higher than 5°F above the average ambient air temperature.

4.5.2.3.2 Traffic Density

Table 4.5-1 gives the projected number of annual shipments of each type of package. All shipments would be made by truck and would remain within the gross vehicle weight (GVW) limitations imposed by all states through which the shipment would travel. Most states impose a 73,280-lb GVW limitation on tractor-trailer combinations.

The largest number of shipments would involve the core and blanket assemblies that would be shipped from the fuel-fabrication plants to the reactor plants. About 20 shipments would be made annually; they would require about 60 days, including round-trip time.

The traffic would be over public roads and interstate systems. Since these vehicles would all be within legal weight restrictions, there would not be excessive loads on roadbeds or bridges. The total number of shipments would be small compared with normally expected traffic density.

4.5.2.3.3 Radiation Exposure

The radioactive material that would be in each package is indicated in Table 4.5-1. Only in the case of PuO_2 and the unirradiated fuel assemblies (which contain the recycled plutonium) could there be sufficient penetrating radiation to require some shielding. In any case, the external dose rates would be negligible, as in the case of UF_6 shipments, or would meet the dose-rate limits imposed by DOT (see Appendix II.M).

Table 4.5.6 gives the estimated doses that would accrue to transport workers and to the general public from the shipment of these materials. The estimates are based on the exposures incurred per 1000 miles and the total distance traveled (see Appendix II.B).

Table 4.5-6

ESTIMATED DOSES TO TRANSPORT WORKERS AND THE GENERAL PUBLIC
FROM SHIPPING UNIRRADIATED MATERIALS

Material	Total Miles per Year	Man-Rems Received per Year	
		Transport Workers	General Population
<u>Low Radiation Level</u>			
UF ₆	1,500	0.027	0.0062
UO ₂	1,900	0.033	0.0080
Fresh Fuel			
Radial Blanket	2,200	0.038	0.0093
<u>High Radiation Level^a</u>			
PuO ₂	2,200	0.40	0.095
Fresh Fuel			
Core and Axial Blanket	13,000	<u>2.4</u>	<u>0.56</u>
Total		2.9	0.68

^aThese packages meet all DOT limits on external dose rates.

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4.5.3 Irradiated Fuel

Each year between a third and a half of the LMFBR core assemblies¹ would be removed from a reactor and replaced with fresh fuel. For this evaluation, it was assumed that 137 fuel assemblies and an average of 23.5 radial-blanket assemblies would be discharged from an LMFBR. Properties of the fuel assembly are given in Table 4.5-7.

The core sections of these assemblies would be irradiated to burnups in the range of 67,000 to 100,000 MWd/metric ton of fuel, depending on the reactor design. Both the core and blanket assemblies would have larger quantities of plutonium in them than when they were charged to a reactor. The assemblies would be producing large amounts of heat and penetrating radiation after removal and would be stored at the reactor site for a period of time to permit these to subside to lower levels.

The assumption is made that, initially, the radial blankets would be shipped in the same cask as the core assemblies, although a similar cask with less shielding may be used later since the radiation problem is not as severe as with the core assemblies. Possibly some radial-blanket assemblies would be shipped in the same cask with core assemblies. Table 4.5-8 gives information on core and blanket shipments.

4.5.3.1 Cask Description

The LMFBR cask would be designed to meet the prevailing DOT and AEC regulatory requirements for fissile-material packages and for large-quantity packages.

The LMFBR spent-fuel cask would look somewhat like an LWR spent-fuel cask on the outside (Figure 4.5-7). It must be capable of dissipating tens of kilowatts of heat, depending on the number of fuel assemblies carried and on the time between discharge from the reactor and shipment. It would probably weigh 50 to 100 tons and, as a consequence, be shipped by rail or barge.

Currently, no design of an LMFBR spent-fuel cask has been completed, although a preliminary design has been completed for a cask to ship spent FFTF fuel and LMFBR demonstration plant fuel. The reference primary coolant for this design is phenyldiphenyl eutectic, but helium is a possible alternative. Because of the high burnup of the fuel, a significant number of neutrons, like the gammas, would have to be shielded against. The shield would consist of either water or WEP and be situated on the exterior surface of the gamma shield.

Table 4.5-7
FUEL-ASSEMBLY^a PROPERTIES

Property	Core	Blanket			
		Axial		Radial	
		Lower	Upper	Inner	Outer
Uranium + Plutonium, kg	69.7	26.6	21.3	127	102
Power, ^b MWt	8.0	0.14	0.3	1.3	0.51
Burnup, MWd/kg	67.4	3.1	3.6	14	10

^aAtomics International follow-on design (Report AI-AEC-12791, Vol. 1).

^bAverage thermal power generation during reactor cycle

Table 4.5-8
CHARACTERISTICS OF LMFBR CORE AND BLANKET SHIPMENTS

Characteristic	Spent Core Plus Axial Blanket	Radial Blanket
Assemblies per Year ^a	137	23.5
Assemblies per Cask	6	6
Shipments per Year	22.83	3.92
Casks per Vehicle	1	1
Shipping Distance, one way, miles	750	750
Shipping Time, one way, days	3	3
Radioactivity per Cask, ^b Ci	3.26×10^6	4.54×10^5
Thermal Power per Cask, kW	13.32	1.83

^aFor a single 1000-MWe LMFBR.

^bAssumes 360 days of cooling.

Table 4.5-9
**ESTIMATED DOSES TO TRANSPORT WORKERS AND THE
GENERAL PUBLIC FROM SHIPPING IRRADIATED MATERIALS**

Material	Total (miles/year)	Man-Rems Received per Year	
		Transport Workers ^a	General Population
Spent Fuel			
Core and Axial Blanket	17,000	5.1	0.73
Radial Blanket	3,000	<u>0.92</u>	<u>0.13</u>
Total		6.0	0.86

^aIncludes doses acquired by transport workers while transferring casks from truck to rail car.

4.5-20

AN LWR SPENT-FUEL SHIPPING-CASK

Figure 4.5-7

For this study, certain assumptions were made. The LMFBR spent-fuel cask was designed to carry six assemblies cooled 360 days and will be built to current standards and with current technology.

A cooling time of 360 days has been evaluated because it appears that the consequences of extremely severe accidents, assuming current cask technology, would be no worse than has been postulated for current LWR fuel shipments if they were involved in a similar accident. It is economically desirable, however, to ship shorter-cooled fuel. Because short-cooled fuel has associated with it a substantial radioactive iodine inventory, any significant leakage could not be tolerated. Thus, short-cooled fuel shipments should not be made unless the cask in which it is shipped, and the procedures by which it is shipped, are adequate to ensure that essentially no radioactivity could be lost even if the cask were subject to extreme accident conditions.

Research and development programs are currently being funded by the AEC to develop information to enable predictions of behavior under severe accident conditions. Significant additional funding is planned for future years to determine whether complete invulnerability can be achieved. Evaluation of such a cask would be premature at this time. It would be a cylindrical cask with a massive gamma shield and a 6-in.-thick water neutron shield and weigh about 75 tons. The fuel cask would be about 18 ft long by 5-1/4 ft outside diameter (OD), with a cavity 16 ft long by 2-1/2 ft inside diameter (ID). Helium would be used in the cask cavity as the primary coolant to transfer the decay heat from the fuel elements to the body of the cask. Second-generation casks may use sodium as a primary coolant if current studies continue to support the belief that the cask cannot be breached by any extremely severe accident which can be reasonably postulated and if the design incorporates procedures that prevent improper assembly of the cask prior to its shipment. Future cask designs are discussed in Section 7.3.4.

4.5.3.2 Transport Conditions

Currently, shipments of irradiated fuel are made by truck or rail on an exclusive-use basis. Some barge shipments may be made in the future. Such shipments will probably not be shipped in general freight as less-than-truckload or less-than-carload lots.

The estimated average distance from the nuclear power plant site to the fuel-reprocessing plant over which the irradiated fuel would be transported is 750 miles. This journey would require an average one-way trip time of 8 to 10 days by rail. Barge shipments might require somewhat longer time, depending on the route.

Shipments of spent LMFBR fuel might have to make part of the journey to the reprocessing site by motor freight. However, because of the LMFBR cask weight and the probable distance between the reactor and the reprocessing plant, shipment by truck over the entire route is not considered to be practicable. At present, many LWR nuclear power plants have no rail service directly to the plant site. Should this be the pattern for LMFBR plant sites, then the cask would have to be trucked to the nearest railhead.

As an alternative, the cask could be reduced in size to handle one assembly instead of the six assumed in this study. The weight of such a cask might be reduced such that the cask could be transported by truck and would not require overweight permits for shipment. However, the number of shipments per year would increase by a factor of 6, and the radiation workers would receive additional exposure since they would be required to work with six single-element casks rather than one six-element cask. Transport workers and the general public would also receive increased doses because of the large number of shipments. More injuries and deaths would occur from common causes.

If barge transport of casks were to be used, construction of docking facilities might be required, at a cost of from \$25,000 to \$1,000,000.⁸ Because of the cost, docking facilities would be unlikely to be built only for the purpose of shipping irradiated fuel elements. If docks were required for other purposes, they might be used for the transportation of irradiated fuel. Barge shipments of spent fuel appear to have some inherent safety advantages. Barge speeds are generally below 10 mph, thereby reducing potential impact velocities, and large quantities of water are readily available at all times for fighting any fires that might occur. In addition, dose rates received by the general public under normal operating conditions would be reduced.

4.5.3.3 Effects on the Environment under Normal Conditions

4.5.3.3.1 Heat

For casks designed to transport six 360-day-cooled core assemblies, the rate of release of heat to the air from each cask would be about 13 kW or about 45,000 Btu/hr. This rate can be compared with the rate at which waste heat is released from a 100-hp truck engine operating at full power, which is about 50 kW or 180,000 Btu/hr. Most of the heat would be produced in the core section of the fuel assemblies which, in the case of the AI follow-on fuel, is 43 in. long. Because of the concentrated nature of the heat source, the secondary cooling system would be very desirable. This coolant system would not be required for safety, since failure of the system would not cause release of any radioactive materials, but would be used to ease operational problems at the fuel-receiving site.

With the coolant system operating normally, the temperature of the cask surface would be less than 50°F above ambient temperature; in any case, the temperature of accessible cask surface would, in accordance with the regulations,⁹ not exceed 180°F. Because the amount of heat would be small and would be released over the entire transportation route, no appreciable effect on the environment would result.

4.5.3.3.2 Package Weight and Traffic Density

Shipping the irradiated core and blanket assemblies from the reactor to the fuel recovery plant would require an average of about 27 railroad car shipments per year. The empty casks would be returned to the reactor. The weight of the spent fuel in a loaded cask would constitute only about 2% of the total weight of the loaded cask. Because the cask being returned empty weighs almost as much as the cask loaded with irradiated fuel, the weight and number of shipments of empty casks must be considered in assessing the impact on the environment of the shipment of irradiated fuel. Therefore, ^{including} ^{return} shipments of the cask, shipping the irradiated fuel would involve about 54 rail-car movements.

The total number of such shipments would be too small to have any measurable effect on the environment as a result of increase in traffic density.

The rail cask itself would weigh about 75 tons. It might require the design of a special car for handling and tie-down purposes, but heavy loads are common in rail traffic. The cask would be expected to have no adverse effect on rail facilities.

Should any part of the trip be made by truck (e.g., to a railhead), then the specific route and distance would have to be considered. Proper axial loadings and potential damage to bridges and roadbeds would be evaluated before overweight permits were granted by the state affected.

4.5.3.3.3 Radiation Exposure

The radioactive material contained within the irradiated fuel elements is given in Table 4.5-8. The spent fuel elements would be transhipped from truck to railhead; dose rates associated with the shipment would be within the limits set by DOT.

Table 4.5-9 gives the estimated doses to transport workers and the general public from the shipments of these materials. The estimates are based on the exposures incurred per 1000 miles as specified in Appendix II.B and the total distance traveled.

4.5.4 Radioactive Wastes

Radioactive wastes would be generated in fuel fabrication, reactor, and fuel-reprocessing plants. Some wastes would be in a combustible form when generated, but some would be converted to a noncombustible form for shipment. Some wastes, other than high level, would not require shielding and would contain radioactivity in amounts from traces to several curies per cubic foot. In other cases waste might be reduced in volume to concentrate the alpha and beta-gamma activity to higher levels per cubic foot. Some shielding would be required.

4.5.4.1 Material Quantities and Packaging

4.5.4.1.1 Alpha-emitting Solid Waste from Fuel-Fabrication Plants

The so-called "alpha" wastes are defined as those solid materials that contain plutonium or other long-lived alpha emitters in concentrations greater than 10 $\mu\text{Ci/kg}$ and yet have sufficiently low external radiation levels that they can be handled directly without supplementary shielding (surface dose rates of less than 10 millirems/hr). The criterion of 10 $\mu\text{Ci/kg}$ is used as a lower radioactivity density for alpha wastes because this value corresponds to the upper range of alpha-emitting isotopes in naturally occurring deposits. Wastes that contain low alpha radioactivity would be regarded as disposable in carefully selected burial grounds. Alpha wastes with concentrations greater than 10 $\mu\text{Ci/kg}$ would be stored in special repositories that offer maximal assurance of permanent containment.

Alpha wastes would consist of a wide assortment of solid materials including items such as paper, cloth, wood, plastic, rubber, glass, ceramic, and metal, as well as salts and sludges that would arise in the treatment of liquid-waste streams and filters from cleanup of off-gases and contaminated equipment. Transuranium elements in these wastes would range from trace amounts to several grams per cubic foot and would average about 0.25 g/ft^3 . The densities of the uncompacted wastes would vary from about 2 lb/ft^3 to as much as 200 lb/ft^3 . About one-half to two-thirds of these wastes (by volume) would be combustible; much of this could be reduced via incineration by factors of about 50 and about 20 in volume and weight respectively. About one-half to three-fourths of the wastes (by volume) could be reduced in volume by factors of 2 to 10 through compaction.

For fuel-preparation operations, it is estimated that alpha wastes would be generated at a rate of about $10,000 \text{ ft}^3$ per ton of plutonium processed. Fuel-fabrication operations would generate waste at about $20,000 \text{ ft}^3$ per ton of plutonium processed. Waste from both operations was assumed to be volume-reduced by a factor of 10, rendered noncombustible, and combined prior to shipment.

About 60% of the alpha waste would be placed in 55-gal drums, and the balance in steel boxes whose approximate dimensions would be 4 by 6 by 6 ft. This material can be shipped by any number of methods that would suitably protect the waste during shipment. For this analysis, the assumption is made that these boxes and drums would be placed inside a steel cargo container, 8 by 8 by 20 ft, and two cargo containers would be put inside a rail car, similar to an ATMX-500 or -600 rail car that would have a total waste capacity of 1000 ft^3 . These cars are specially modified to handle radioactive materials.^{10,11} Shipments would be to a Federal repository for permanent storage. Figure 4.5-8 shows an external view of an ATMX-500 rail car. Table 4.5-10 gives shipping information.

4.5-26

EXTERNAL VIEW OF THE ATMX-500 SERIES RAIL CAR

Figure 4.5-8

Table 4.5-10

PACKAGE AND SHIPPING INFORMATION FOR LOW-LEVEL SOLID
WASTES GENERATED AT A FUEL-FABRICATION PLANT

	Alpha	Beta-Gamma
Annual Production ^a	50,000 ft ³	210,000 ft ³
Annual Shipping Load ^b	5,000 ft ³	21,000 ft ³
Containers per Year	400 drums, ^c 14 boxes	2,800 drums ^d
Containers per Vehicle		64
Waste Volume per Rail Car ^e	1,000 ft ³	
Shipments per Year	5	45
Radioactivity per Shipment	27,900 ^f Ci	0.475 ^g Ci
Plutonium Content	2.5 kg	
Thermal Output per Shipment	22.5 W	0.002 W
Shipping Distance, One Way	1000 miles	500 miles
Shipping Time, One Way	4 days (rail)	1.0 days (truck)

^aFor a single 1000-MWe LMFBR.

^bVolume reduced by a factor of 10.

^cAssumes 60% waste shipped in 55-gal drums (7.4 ft³) and 40% shipped in 4 x 6 x 6 ft steel boxes.

^d55-gal (7.4 ft³) nonreturnable drums.

^eRail cars will have two cargo containers, each housing five burial units that average 100 ft³ each.

^fPlutonium has a specific activity of 11.6 Ci/g (0.263 alpha Ci/g Pu).
90.001 Ci of beta-gamma activity per ft³.

4.5.4.1.2 Low-Level Beta-Gamma-Emitting Waste from Fuel-Fabrication Plants

Diverse low level beta-gamma-emitting solid wastes are common to all facilities that handle radioactive materials. They range from concentrates of the radio-nuclides generated in the decontamination of plant-effluent streams to almost every conceivable type of contaminated solid refuse from plant operations. Since they contain less than 10 μCi of long-lived alpha radioactivity per kilogram, they are disposable in surface burial grounds. From operating experiences at AEC sites, the estimate is made that 40,000 and 80,000 ft^3 of waste would be generated in LMFBF fuel-preparation and -fabrication operations, respectively, for each ton of plutonium processed.

Because of the large volume generated in this projection, efforts are expected to be made to reduce the volume shipped, both by better housekeeping and via compaction or incineration. The volumes are expected to be reduced by a factor of 10 with little difficulty. The activity level would average 0.001 Ci/ft^3 .

The waste would be transported by truck in 55-gal disposable drums to a licensed burial facility. Table 4.5-10 gives shipping specifications. This waste material would be about 50% combustible and could be shipped as LSA material.¹²

4.5.4.1.3 Low-Level Beta-Gamma-Emitting Waste from Power Stations

This waste would consist primarily of spent resins, sludges, filters, etc. contaminated with activation products to about 1.0 Ci/ft^3 of waste. The material would be sent to a commercial, licensed burial ground and would be shipped by truckload in 55-gal drums. Table 4.5-11 gives shipping information.

4.5.4.1.4 Other Low-Level Wastes from Power Stations

Most of the tritium that would be produced in a power reactor would be converted to sodium tritide (Na^3H) in the primary reactor coolant where it would appear as a sludge in the primary-coolant cold traps. This material could be converted to a stable, solid form (e.g., tritiated calcium hydroxide, $\text{Ca}(\text{O}^3\text{H})_2$) and shipped to a burial ground or to an as yet undesignated storage site should regulations be changed to require such storage.

For this study, the assumption is made that the tritium would be in the form of $\text{Ca}(\text{O}^3\text{H})_2$, which would be placed in 0.16- ft^3 disposable steel cans that could fit inside the insulated shipping container (Figure 4.5-4). Table 4.5-11 gives shipping specifications.

Table 4.5-11

PACKAGE AND SHIPPING INFORMATION FOR LOW-LEVEL
SOLID WASTES GENERATED AT A POWER STATION

	Beta-Gamma	Tritium
Annual Production ^a	2000 ft ³	19,135 Ci
Annual Shipping Load	Same as production	12.24 ft ³ of Ca(O ³ H) ₂ ^b
Containers per Year	270 drums	38 cylinders
Containers or Units per Vehicle	64	64
Shipments per Year	4.2	0.60
Radioactivity per Shipment	475 Ci	32,000 Ci
Thermal Output per Shipment	2.85 W	1.13 W
Shipping Distance, One Way	500 miles	1000 miles
Shipping Time, One Way	1 day (truck)	2 days (truck)

^aFor a single 1000-MWe LMFBF.^bAssumes tritium is converted from Na³H to Ca(O³H)₂ at a rate corresponding to 640 ft³/MCi of tritium.

Power stations will produce some small quantities of sodium and noble gas wastes during normal operations. Sodium purged from an operating reactor will be converted to an inert solid compound (e.g., sodium carbonate) and temporarily stored on site or shipped out with other waste shipment. Radioactive noble gas will be removed from the reactor cover gas system. Should as much as 1% of the noble gases leak to the cover gas, approximately 1500 Ci of Kr-85, the most significant isotope, will be removed per year. This quantity would amount to about 1% of the capacity of a standard 2200 psi gas cylinder should it be filled to capacity as is done at the reprocessing plant (probably a partially filled cylinder will be shipped with other waste every few years). Neither of these wastes was significant enough to report in Table 4.5-11.

4.5.4.1.5 Alpha-Emitting Solid Waste from a Fuel-Reprocessing Plant

The alpha-emitting waste that would be generated in a fuel-reprocessing plant could be shipped in the manner discussed in Section 4.5.4.1.1. It is assumed to be placed in rail cars and transported to a Federal repository. The volume of alpha waste produced in a reprocessing plant would be about 2000 ft³ per ton of plutonium processed. Table 4.5-12 gives shipping information.

4.5.4.1.6 Low-Level Beta-Gamma Solid Waste from a Reprocessing Plant

The waste is similar to that generated at the fabrication plant and power station except that it is produced at a rate of approximately 16,000 ft³ per ton of plutonium processed. It contains 0.001 Ci of beta-gamma activity per cubic foot. The material is placed in 55-gal disposable drums and shipped to a licensed burial site. Shipping information is given in Table 4.5-12.

4.5.4.1.7 Alpha-, Beta-, and Gamma-Emitting Solid Waste from a Fuel-Reprocessing Plant

Alpha-beta-gamma-emitting wastes are defined as those solid materials, other than high-level and cladding wastes, that contain long-lived alpha radioactivity in concentrations greater than 10 μ Ci/kg and that have gamma-radiation levels sufficient to require biological shielding and remote-handling techniques (typical surface dose rates lie between 10 and 1000 millirems/hr). These wastes would consist of an assortment of materials similar to alpha wastes, including contaminated equipment. About 10,000 ft³ of this waste would be generated per ton of plutonium processed and would contain an average of 0.025 g of plutonium or uranium per cubic foot. Its volume

Table 4.5-12

SHIPPING INFORMATION FOR WASTE FROM A REPROCESSING PLANT

	Alpha Waste	Low-Level Beta-Gamma Waste	Alpha, Beta- Gamma Waste	Cladding Hulls	High-Level Waste	Noble Gases	Iodine
Annual Production ^a	7700 ft ³	30,000 ft ³	20,000 ft ³	580 ft ³	55 ft ³	680 g atoms	0.078 ft ³
Annual Shipping Load	770 ft ³	3,000 ft ³	2000 ft ³	165 ft ³	Same as production	Same as production	Same as production
Containers per Year	104 drums	4405 drums	80 cylinders	47	9.0	1.0 cylinders	0.5 canisters
Containers per Unit, if Combined			3 per cask	36 per cask	12 per cask	6 per cask	1 per package
Containers or Units per Vehicle		64	1 cask	1 cask	1 cask	1 cask	64
Waste volume per Rail Car	1000 ft ³						
Shipments per Year	0.77	6.3	27	1.3	0.75	0.164	0.0076
Radioactivity per Shipment, Ci	27,900	0.475	500	1.46 x 10 ⁶	7.8 x 10 ⁶	9 x 10 ⁵	1.46
Plutonium Content per Shipment, ^b g	2500		18.75	700			
Thermal Output per Shipment, W	22.5	0.002	1.12	10.3 x 10 ³	2.5 x 10 ⁴	1470	0.001
Shipping Distance, One Way, miles	1000	500	1000	1000	1500 ^c	1500	1000
Shipping Time, One Way, days	4 (rail)	1 (truck)	2 (truck)	4 (rail)	6 (rail)	3 (truck)	2 (truck)

^aFor a single 1000-MWe LMFBR.

^bPlutonium has an isotopic composition of 1% Pu-238, 60% Pu-239, 24% Pu-240, 11% Pu-241, and 4% Pu-242; specific activity is 11.6 Ci/g (0.263 alpha Ci/g).

^cEPA (Comment Letter 53, p. 18) suggested that an average shipping distance of 2500 miles would appear to be more reasonable since repositories may be located in the western United States. However, the 1500 mile distance in this Table appears to be more reasonable. Even though such repositories may be located in the western United States, any average shipping distance developed should include the fact that some of the LMFBR fuel reprocessing plants and LMFBR power plants would also be located in the western United States. If the distance were 2500 miles, the dose due to transportation would be increased by only about 1%.

would be reduced by a factor of 10, and it would be packed in disposable steel canisters.

These canisters would be assumed to be 10 ft long, have a 24-in. OD, and contain 25 ft³ of waste. They would be handled remotely and seal-welded prior to shipment.

Three of these canisters would be placed in a cask that would have about a 4-in. equivalent of steel shielding and a gross weight of 55,000 lb. The wastes would be assumed to be shipped to a burial site or a repository for permanent storage. Table 4.5-12 gives shipping information.

4.5.4.1.8 Cladding Hulls from a Fuel-Reprocessing Plant

Cladding wastes would consist of stainless steel cladding hulls and structural components of the fuel assemblies that would remain after the fuel cores had been dissolved. Although their radioactivity would arise mainly from neutron-induced isotopes (100,000 Ci per ton of fuel after one year of cooling), the hulls would be similar in some respects to high-level waste in that they might contain as much as 0.05% of the plutonium originally in the spent fuel and 0.05% of the fission products and they might have heat-generation rates of 50 to 100 W/ft³. These cladding hulls and associated hardware would be assumed to be compacted to an average of 70% theoretical density (8.8 ft³ per ton of fuel) and placed in disposable steel canisters that would have a volume of 3.5 ft³ each. The waste would have a heat-generation rate of 50 to 100 W/ft³. Thirty-six disposable steel canisters would be placed in a shipping cask that would have a shield equivalent to 5 in. of lead. The cask would have a length of 12.5 ft and a diameter of 7.5 ft, would weigh 188,000 lb, and would be shipped to a Federal repository. Table 4.5-12 summarizes shipping information.

4.5.4.1.9 Solidified High-Level Waste from a Fuel-Reprocessing Plant

High-level wastes are defined in Federal regulations 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." These wastes contain virtually all of the nonvolatile fission products, about 1% of the uranium and plutonium originally in the spent fuels, and nearly all the other actinides formed by transmutation of the uranium and plutonium in the reactors. They can be generally characterized by their very intense, penetrating radiation

and their high heat-generation rates. Regulations call for these wastes to be solidified within five years after they are generated and for the resultant stable solids to be shipped to a Federal repository within ten years after the liquids are generated.¹³ These solidified wastes would be thermally and radiologically stable and could be handled remotely, shipped, and stored for long periods of time.

High-level solidified wastes are expected to be produced at a rate of about 3 ft³ per ton of fuel and would have the following characteristics:

	<u>Ci per ton of fuel</u>	<u>W per ton of fuel</u>
Fission products	306,000	865
Actinides	<u>6,000</u>	<u>130</u>
Total	312,000	995

The solidified waste would be contained in a disposable steel canister with a welded closure. If the canister were 12 in. in ID by 10 ft long, it would contain about 6.28 ft³ of waste.

The assumption was made that as many as 12 disposable cans would be shipped in one cask. The cask would require both gamma and neutron shielding, weigh about 229,000 lb, and would probably be shipped to a retrievable surface storage facility operated by the Federal Government. Table 4.5-12 summarizes shipping information.

4.5.4.1.10 Noble-Gas Fission Products from a Fuel-Reprocessing Plant

The noble-gas fission products would consist principally of stable and radioactive isotopes of krypton and xenon. After irradiation of the fuel in a reactor is stopped, the short-lived isotopes decay quickly, and after 150 days, the only significant noble-gas isotope is Kr-85 (10.8 year half-life). Currently, over 99% of this gas is discharged to the atmosphere following dissolution of the fuel at a fuel-reprocessing plant. Several processes are available or are under development for recovery of these gases from plant off-gas streams.¹⁴ Initially, the mixed gases would most likely be collected under high pressure in gas cylinders, shipped to a remote site, and stored under conditions that would promote long-term integrity of the cylinders.

After one year of decay, about 37.5 g-atoms of noble gases exist per ton of fuel. The noble gas is expected to be collected and shipped in standard 50-liter cylinders, 9 in. in diameter by 50 in. high. Cylinders that would conform with

ICC specification 3AA have a wall thickness of about 1/4 in., weigh 135 lb, and normally be used for 2200 psig nitrogen service might be used. Each cylinder could contain 693 g-atoms of krypton and xenon at rated pressure.

The shipping cask for these cylinders, basically a tank filled with water¹⁵ (Figure 4.5-9), would be a modification of one that has been shown to meet the impact-, puncture-, and fire-resistance specifications of the 10 CFR 71 and that has been licensed for shipping capsules of curium oxide.¹⁶ The cask is 5 ft in diameter, made of 1-in.-thick type 304 stainless steel, and equipped with external fins to enhance heat dissipation. The water provides shielding, serves as a heat-transfer medium, and provides the heat capacity needed to withstand a 1475°F fire for 30 min. A 200-psia rupture disk is provided as a safety measure in addition to 16 fusible plugs, which are designed to allow steam to escape in case of a fire. In a cask of the dimensions shown, the cylinder temperature would be about 20°F above the ambient.

Six cylinders would be shipped per cask; the cask would contain 900,000 Ci of Kr-85 and have a gross weight of 21,300 lb. Shipment would be by truck to a noble Gas Storage Facility (NGSF). Table 4.5-12 summarizes shipping information.

4.5.4.1.11 Iodine from a Fuel-Reprocessing Plant

Iodine is a semivolatile fission product that, because of its complex physical and chemical properties and its high biological significance, has always required special attention to ensure adequate safety in its management. The isotope of greatest concern during reactor operation is I-131 (8.05 year half-life). However, following the 360-day decay period for spent reactor fuel, the species of consequence is I-129 (1.6×10^7 year half-life).

Research and development work aimed at reducing iodine releases from fuel-reprocessing plants to "near zero" levels show promise of removing at least 99.9% from the other fuel constituents by volatilization at the head-end of the process.¹⁷ Once separated, it would subsequently be trapped and retained separately from the other waste streams. The final form into which the iodine might be processed for packaging, shipment, and disposal has not been completely defined but probably will be some stable form like potassium iodide, KI. This material would be produced at a rate of 0.0043 ft³ per ton of fuel, packaged in a 0.16-ft³ cylinder, and shipped in a protective package similar to the one

4.5-35

CONCEPTUAL DESIGN OF SHIPPING CASK
FOR CYLINDERS OF COMPRESSED FISSION-PRODUCT GASES

Figure 4.5-9

shown in Figure 4.5-4. The package would be shipped to a burial ground or repository for permanent storage. Table 4.5-12 gives shipping information.

4.5.4.2 Transport Conditions

All shipments are expected to be made under exclusive-use full-load conditions and to be transferred either to a Federal repository, noble gas storage facility, or licensed burial site, depending on the amount and type of radioactivity in the waste. Alpha-emitting wastes would be packaged in single "burial units" and placed in cargo containers which would, in turn, be placed in rail cars.

4.5.4.3 Effects on the Environment under Normal Conditions

4.5.4.3.1 Heat

The radioactivity of most waste material would be dilute; consequently, very little heat would be generated in the packages. One notable exception would be the high-level waste produced in fuel-reprocessing plants. High-level waste would be concentrated and solidified to a thermally and radiolytically stable solid contained in a stainless steel can with a welded closure. The heat release from one cask would be about 25 kW, which is less than that generated by an automobile engine in normal turnpike driving. Precautions would be taken to prevent the general public from coming in contact with the hot cask surface. Both the cladding hulls and noble-gas shipments would generate moderate quantities of heat but less than that which would be generated by the high-level waste on a per package basis.

4.5.4.3.2 Package Weight and Traffic Density

All trucks that would carry waste material would be expected to be within the weight limitations imposed by the states through which the shipments would pass. In most cases the GVW must be less than 73,280 lb.

The heavy casks and the alpha-emitting waste would be expected to be shipped by rail. The high-level-waste cask would be the heaviest at about 115 tons. Such weights would be expected to pose no problem, with the possible exception of infrequently used rail spurs, built for lighter traffic, which could lead them around the plant site to the main rail line. Improvements in the carrying capacity of such road beds, if they should exist, would have to be made before these casks would be permitted on the rail line.

The most frequent shipments would be of low-level beta-gamma waste from the fuel-fabrication plants. The quantity of waste that would be generated would result in about 45 shipments/year. These shipments would be over normally used state and Federal highway systems and would produce a negligible increase in traffic density.

4.5.4.3.3 Radiation Exposure

The predicted radioactive inventory for the waste that would be generated at the fuel-fabrication plants is given in Table 4.5-10. Table 4.5-11 describes the wastes that would accumulate at the power station. The radioactive wastes that would be generated at the fuel-reprocessing facilities are listed in Table 4.5-12. None of these waste materials would likely be transhipped. The low-level and high-level materials would be always shipped separately.

Table 4.5-13 gives the estimated doses to transport workers and the general public from the shipments of these materials. The estimates are based on the exposures incurred per 1000 miles (as specified in Appendix II.B) and on the total distance traveled.

4.5.5 Postulated Accidents

Each package shipped in the fuel cycle will be exposed to possible accidents. Estimates are made of the releases of radioactive material which might occur within the framework of accidents and conditions described in this section and Appendix II.N. Results provide radioactivity release estimates from which population exposures are calculated (see Section 4.5.6.2).

Five accident-severity categories have been defined to include the type of accidents described in Table II.A-2 of Appendix II.A. The severity is a function of vehicle-impact speed and of duration of the fire that might result. The definition of the duration of the fires are as follows: 0 to 0.5 hr = 0.25 hr; 0.5 to 1 hr = 0.75 hr, and over 1 hr = 2 hr. The definition of the vehicle speed at impact is as follows: 0 to 30 mph = 20 mph; 30 to 40 mph = 40 mph; 50 to 70 mph = 60 mph; over 70 mph = 75 mph; the average impact speed and duration of fire for each category can be determined for rail and truck by weighting the speed-fire combination by the probability of occurrence given in Table II.A-2. For example, about six out of every eight extra-severe truck accidents occur at speeds between 50 and 70 mph, and two of eight accidents occur above 70 mph (see Table II.A-2). The average speed for the category is, therefore, $(0.75) \times (60 \text{ mph})$ plus $(0.25) \times (75 \text{ mph})$, or 63.75 mph. The approximate length of fire in that accident is $(0.75) \times (2 \text{ hr})$ plus $(0.25) \times (0.75 \text{ hr})$, or 1.69 hr. Table 4.5-14 presents the results.

Table 4.5-13

ESTIMATED DOSES TO TRANSPORT WORKERS AND THE
GENERAL PUBLIC FROM SHIPPING WASTE MATERIALS GENERATED
WITHIN THE LMFBR FUEL CYCLE

Material (Shipment Mode)	Total (miles per year)	Man-Rems Received per Year Transport Workers	General Population
<u>Waste from Fuel Preparation and Fabrication Plants--Low Level</u>			
Alpha (rail)	5,000	0.037	0.021
Beta-Gamma (truck)	22,000	0.4	0.096
<u>Solid Waste from Power Stations--Low Level</u>			
Beta-Gamma (truck)	2,100	0.038	0.0091
Tritium (truck)	600	0.011	0.0026
<u>Wastes from Reprocessing Plants</u>			
Low level			
Alpha (rail)	700	0.0056	0.0032
Beta-Gamma (truck)	3,200	0.059	0.014
Iodine (truck)	8	0.0002	0.00005
High level			
Alpha-Beta-Gamma (rail)	27,000	2.1	1.1
Cladding Hulls (rail)	1,300	0.10	0.054
High-Level Waste (rail)	1,100	0.084	0.048
Noble Gases (truck)	250	0.045	0.011
Total		2.9	1.35

Table 4.5-14

AVERAGE IMPACT SPEED AND DURATION OF FIRE
BY VEHICLE TYPE AND ACCIDENT SEVERITY CATEGORY

Severity Category	Rail		Truck		Average	
	Speed (mph)	Fire (hr)	Speed (mph)	Fire (hr)	Speed (mph)	Fire (hr)
(1) Minor	28	0.002	35	0.0011	30	0.0015
(2) Moderate	59	0.022	59	0.008	59	0.015
(3) Severe	59	0.47	74	0.016	66	0.4
(4) Extra Severe	62	1.8	64	1.7	62	1.7
(5) Extreme	75	2.0	75	2.0	75	2.0

The length of the fire in the first two accident categories would be so short (about 1 min or less for category 2) that it is considered zero with regard to the effect on the packages that would be shipped.

These average impact speeds and durations of fires have been applied to each shipment of radioactive material that would be made in the fuel-cycle, and the possible loss of material has been assessed. Factors that influence the results include knowledge about results of impact and fire tests applied to the particular package design being considered or applied to packages that are likely to behave in similar fashion to the package in question.

All Type-B packages are assumed to be capable of withstanding all accidents in categories 1 through 3, whereas Type-A packages are expected to withstand only accidents in categories 1 and 2. Most people involved in setting regulations judge the accident requirements specified in the regulations to be equivalent to a transportation accident at about 60 mph and involving about a 1-hr fire.¹⁸ Accident categories 4 and 5 involve both high-speed-vehicle impacts and long-duration fires. Extended fire is the most likely agent to cause release of radioactive materials since the vehicle speeds are not greatly different from those encountered in accident categories 2 and 3.

For each given set of accident conditions, a spectrum of results will be produced that could range from the release of nothing to the release of large quantities of radioactive material. As the accident conditions become more severe, the average releases are likely to be greater.

The fire condition was assumed to transfer a net heat flux to the package of $17,000 \text{ Btu}/(\text{hr}\cdot\text{ft}^2)$ if the surface temperature of the package remained below about 400°F for the duration of the fire. This flux corresponds to a flame temperature of 1475°F having an emissivity of 0.9 and the receiving surface having an absorptivity of 0.8.¹⁹ In several cases where the package would heat up rapidly, lower average heat fluxes were assumed. If the average surface temperature of the package were expected to be about 1050°F , the average net heat flux to the package would be reduced to about $10,000 \text{ Btu}/(\text{hr}\cdot\text{ft}^2)$.

4.5.5.1 Packages of UF_6

Depleted uranium for preparation of UO_2 and mixed-oxide pellets for the LMFB core will be shipped in steel cylinders to the fuel-fabrication plant as solid UF_6 . By current practice such cylinders are shipped without a protective shipping package. Protective shipping packages are required for cylinders as large as the 48-Y type containing recycle uranium returned from the reprocessing plant, however, because of the amount of radioactive material they contain. Single cylinders of the 48-Y type, having a nominal capacity of 14 tons (maximum net 27,560 lb) and a tare weight of 5200 lb, are shipped by truck, but railroad cars have been equipped to accommodate four of these cylinders on one car. The characteristics of UF_6 are such that exposure to ordinary atmospheric conditions following an accident in which a hole is made in the steel cylinder will not result in the release of an appreciable amount of UF_6 .

By virtue of the large amount of UF_6 in a type 48-Y shipping cylinder and especially in transfer by rail, where four cylinders can be carried on one car, the potential exists for the release of a very large amount of UF_6 in an accident. The available statistics on accident frequency for carriers show that the chance of such accidents is very small.

Three accidents (see Appendix II.N) were analyzed. The analyses showed the effect of cylinder damage (or its absence), of fire, and of water submersion on the release of UF_6 .

Case 1, involving a single 14-ton cylinder and a 1.7-hr fire, led to the release of 10 metric tons of UF_6 (and 2.28 metric tons HF equivalent) with about 1.1 Ci of uranium. Case 2 examines the behavior of an undamaged cylinder on exposure to fire. The cylinder may survive a fire of 0.5-hr duration but will fail due to internal pressure in fires lasting 1.7 hr or longer. The release would be the

same as that in Case 3. Case 3, with a leak to water instead of a fire, led to the release of 12.5 metric tons of UF_6 (and 2.85 metric tons HF equivalent) containing about 1.5 Ci of uranium.

4.5.5.2 Packages of UO_2

All UO_2 is expected to be stored at the fuel-reprocessing plant. However, packages for UO_2 shipping are available and accidents involving such packages were analyzed. Packages used to ship UO_2 , which have a phenolic foamed insulating material, have been extensively tested^{6,20,21} and have been shown to meet all requirements of the hypothetical-accident conditions specified in the regulations (see Appendix II.M).

Tests on other packages have indicated that under very severe impacts, lids with ring closures might be lost, exposing the inner container to a fire.⁶ As an extra margin of security against possible failure under severe-accident conditions, the flanges of this package are secured with bolts (see Figure 4.5-3). It was assumed that in accident of severity category 5, only the lid and top insulation of one package in the shipment are lost, thus exposing the inner bolted lid to the radiant source of energy.

The temperature of the inner steel container, surrounded by insulation on the sides would reach 1475°F in a 2-hr fire, and 2.1 mg (3.4×10^{-8} Ci) of uranium would be released.

4.5.5.3 Packages of PuO_2

The packages for shipping PuO_2 will probably be similar to the insulated shipping container²² shown in Figure 4.5-4. This type of package has been extensively tested and is believed to be capable of surviving categories-1 through -4 accidents.

In the category-5 accident, the outer drum lid was assumed to be lost, thus releasing the insulating cover, as in the case of the UO_2 package. The PuO_2 was assumed to be packaged in special-form containers and placed inside the inner package. This packaging requirement is suggested in currently proposed regulations.²³ The temperature of the inner PuO_2 cans will reach 1475°F in the 2-hr fire.

The special-form cans held in the inner steel vessel must be capable of surviving a 1475°F temperature for 10 min to meet regulatory requirements. It is assumed

that if the cans could meet the requirements for 10 min, they could remain at that temperature indefinitely since the resulting maximum internal pressures of about 37 psig would remain constant and the cans would be capable of sustaining that internal pressure at that temperature. Therefore, even if exposed to a category-5 accident, containers would not rupture, and no PuO_2 would be lost.

4.5.5.4 Alpha-Waste Shipments

Alpha-contaminated waste is assumed to be packaged in 55-gal drums and 4 by 4 by 6-ft boxes that would be placed in an 8 by 8 by 20-ft cargo container. Two cargo containers would be placed in a rail car for shipment to a Federal repository. The assumption has been made that, after reduction of this combustible material to a noncombustible form, there might be as much as 0.1% of the material still capable of producing some gases should the package reach a high enough temperature ($\sim 400^\circ\text{F}$). Any drums or boxes that would contain alpha waste not immobilized in a form such as concrete would have CaO added to them to act as a getter for any water, carbon dioxide, or other acidic gases that might be formed. The rail cars, which would carry 1000 ft^3 of waste, are designed to be considerably stronger than the average rail car and, consequently, will withstand very severe accidents with little damage to the car.^{10,11}

Only the most severe impact and longest fire is assumed to be capable of providing a mechanism such that some activity might be lost. Accident category-5 would most likely involve a derailment and overturn of the car. Because of the strength of the vehicle underframe and internal bracing, the accident would probably create only rips or holes in the rail car and the cargo containers would not be lost. If the vehicle were immersed in a 2-hr fire, 0.005 g of plutonium (0.058 Ci) could be released.

4.5.5.5 Low-Level Beta-Gamma Waste

Low-level beta-gamma waste would be transported in 55-gal drums and would contain waste that is contaminated with fission products. The shipments would be made by truck. This waste material is about 50% combustible and will burn at about 400°F . It could be shipped as LSA material.¹²

Calculations indicated that radioactive material might be released in accidents in categories 3, 4, and 5 (see Table 4.5-15 and Appendix II.N.5). The drums are assumed to stay with the vehicle in category-3 and -4 accidents; 2% of the activity on the combustible waste might be released with ash and smoke. In category-5

accidents, 25% of the containers are assumed to be thrown free of the vehicle and ruptured. In this case, 4% of the activity in the combustible waste might become airborne. For categories-3 and -4, a 2% loss involving 50% combustible waste would yield a 1% release. For category-5, a 1% release from three-fourths of the drums plus a 2% release from the remaining drums would yield a 1.25% release.

Assuming that each shipment contains 0.475 Ci of activity, predominantly Sr-90, the release of radioactive material is 0.00475 Ci for categories-3 and -4 and 0.0059 Ci for category-5.

4.5.5.6 Low-Level Beta-Gamma Solid Waste from Power Stations

Low-level beta-gamma waste from a power station is contaminated primarily with activation products to a level of about 1.0 Ci per cubic foot of waste. The material is considered to be LSA and would be shipped by the truckload in 55-gal drums.

The accident analysis is the same as for the beta-gamma waste, and the fractional activity release per category is predicted to be the same. The activity, however, is predominantly manganese, iron, and cobalt. Each shipment is assumed to contain as much as 475 Ci divided among several isotopes in the following proportion: 43% Mn-54, 45% Fe-55, 11% Co-58, and 1% Co-60. The releases of radioactive material are therefore 4.75 Ci for categories-3 and -4 and 5.93 Ci for category-5.

The isotopes released are listed in Table 4.5-15.

Table 4.5-15

RADIOACTIVITY RELEASED IN ACCIDENTS INVOLVING LOW-LEVEL BETA-GAMMA SOLID WASTE FROM POWER STATIONS

Isotope	Amount released (Ci)		
	Category 3	Category 4	Category 5
Mn-54	2.04	2.04	2.55
Fe-55	2.14	2.14	2.67
Co-58	0.52	0.52	0.65
Co-60	0.05	0.05	0.06

4.5.5.7 Tritium

Tritium is assumed to be converted to tritiated calcium hydroxide, $\text{Ca}(\text{O}^3\text{H})_2$, which would have a volume of 640 ft^3 . This material must be protected from exposure to high temperatures since, at 1076°F , a water molecule would be driven off, this producing CaO and tritiated water vapor. The material is, therefore, assumed to be packaged in steel cans which would, in turn, be placed in an insulated shipping container similar to the package shown in Figure 4.5-4. Because physical testing of this type of package has shown that it will withstand more than the hypothetical-accident conditions, it is assumed that category-1 through -4 accidents will not damage the package beyond its capability to maintain containment.

A category-5 accident, however, might be sufficiently severe to throw 25% of the packages free of the transporting vehicle. If 25% (i.e., four drums) of those released lose their lids and top insulation, exposure to fire might release the tritium (2000 Ci).

4.5.5.8 Alpha Wastes from Reprocessing Plant

The alpha waste packaging from the reprocessing plant would be the same as for alpha wastes from the fuel-fabrication plant. The accident analysis is identical.

4.5.5.9 Low-Level Solid Beta-Gamma Wastes from a Reprocessing Plant

Low-level beta-gamma waste from a reprocessing plant is similar to the beta-gamma waste generated at the power station and is handled, packaged, and shipped in the same way. The accident analysis and consequences are identical.

4.5.5.10 Alpha-Beta-Gamma Solid Waste from a Reprocessing Plant

Alpha-beta-gamma waste from a reprocessing plant would be volume-reduced, packed in disposable steel cylinders, and shipped in a shielded cask. All material would be handled remotely, and the steel cylinders would be remotely seal-welded.

Shielded casks have been tested under accident conditions and have been shown to be capable of absorbing large amounts of energy.^{24,25} These casks are predicted not to lose their lid plugs in severe accidents. If the end plugs are large, the seals might be damaged, thus permitting some leakage.

A small fraction of the waste carried in the welded steel cylinders might be combustible; if so, it could begin to gasify at 400°F . However, pressure buildup would be slight and no leakage is expected.

4.5.5.11 Cladding Hulls

Stainless steel cladding hulls will be produced in the head-end facilities of the reprocessing plant. The hulls would be placed inside a disposable steel cylinder for shipping. The accident analysis indicated that no radioactive material would be released, even in a category-5 accident.

4.5.5.12 Solidified High-Level Waste from a Reprocessing Plant

Processed high-level waste is expected to be in the form of a thermally stable solid placed inside a welded stainless steel canister. The waste would be doubly contained by the canister and the shielded shipping cask.²⁶ The accident analyses indicated that no release of radioactive material would occur in any accident category.

4.5.5.13 Noble Gases

Noble gases, predominantly Kr-85, are assumed to be compressed into standard 2200-psi cylinders. For shipment, six of these could be placed into a water-filled shipping cask.²⁷ The cask is assumed to be built of 3/4-in.-thick steel plate and equipped with fusible plugs²⁸ designed to relieve excessive steam pressure if the package is involved in a fire. No releases of radioactive material due to accidents are expected, except for category-5 accidents, where a complete release is anticipated (900,000 Ci).

4.5.5.14 Iodine

Iodine is assumed to be converted to potassium iodide (KI) for shipment to a repository. The material would be placed into canisters which would, in turn, be placed into insulated shipping packages described in Figure 4.5-4. Since so few packages are produced per year, they could be shipped to the repository along with other waste. No releases due to accidents are anticipated, except in category-5 accidents. In such accidents the maximum release would be 1.83 Ci of I-129.

4.5.5.15 Unirradiated Fuel

According to the design, an LMFBR fuel assembly would consist of 217 fuel pins arranged hexagonally and encased in a steel sheath. The assemblies are assumed to be shipped in packages similar to those used in shipping LWR fuel. Both the core and radial-blanket assemblies would be sealed in plastic bags and placed in metal containers. These packages would support the assemblies along their entire lengths and would protect them from vibration normally incident to transport. Two of these

containers would be placed in a protective overpack, and four overpacks would be placed on a trailer. Shipment would be by motor freight.

Packages with overpacks would be capable of withstanding accidents in categories 1, 2, and 3 since these accidents produce forces and conditions less severe than those specified in the hypothetical accident conditions of the regulations. Accidents of category-4 would expose the packages to longer fires but to no more severe forces. Pressure buildup due to expansion of gas in the fuel rods would not be significant, and no rods would fail.

A category-5 accident might generate forces somewhat in excess of those generated by the regulatory-accident conditions; however, packages would be expected to be built with enough safety factors to keep the assemblies retained within the overpack and adequately cushioned. The fuels would be designed to withstand significant internal gas pressure, and the closely packed fuel tubes would be surrounded with a stainless steel sheath that would make the assembly quite rigid.

The fuel would be a ceramic pelletized to 85-90% theoretical density and cleaned of all fines (material which can be wiped or blown from the surface of the pellet) at time of fabrication. Should a category-5 accident cause any pellets to crack, it is unlikely that any small pieces could be lost from the tubes. Therefore, no release would be likely to occur.

The possibility that fresh fuel assemblies might be arranged in a critical array has been considered. Eight assemblies would be transported on each vehicle in cans and protective overpacks. While the fuel assemblies would be in these packages, they would be less reactive by virtue of spacing and material such as steel interspersed between assemblies. To become more reactive, the assemblies would have to be separated from their overpacks and inner packages and would have to be arranged in such a way as to have close nuclear interaction.

The possibility is most unlikely that the fuel assemblies in one shipment could, by virtue of the forces in a severe accident, become separated from their containers and become arranged in such a fashion as to have sufficient nuclear interaction to sustain a chain reaction. However, if it is assumed that such an accident occurred, one possible reactive arrangement of fuel assemblies is a planar array in which the fuel assemblies are arranged side-by-side.

The effective multiplication factor of an infinite planar array of unirradiated AI follow-on fuel assemblies on 12-in. centers was calculated to be 0.835 ± 0.006 in air. One reviewer* stated that this was not enough information on this subject. The following table (Table 4.5-16) was added in response to that comment. Calculations were also made for other arrays of fuel assemblies (see Table 4.5-16). The close-packed array exhibited the largest neutron multiplication factor and was subcritical. These calculations are very conservative in that such geometries are quite unlikely following an accident.

Table 4.5-16
CALCULATED EFFECTIVE MULTIPLICATION FACTORS
FOR LMFBR FUEL ASSEMBLIES

3 x 3 Square Lattice Arrangements of LMFBR Fuel Elements Represented
as Circular Cylinders, Reflected by 40.6-cm-thick Concrete, and Flooded

Surface Separation of Elements (cm)	Calculated k_{eff}
0	0.965 ± 0.003
0.64	0.955 ± 0.004
1.27	0.915 ± 0.005
1.91	0.856 ± 0.006

Eight LMFBR Fuel Elements in Hexagonal Geometry,
Reflected by 40.6-cm-thick Concrete,
and Various Conditions of Moderation

Surface Separation of Elements (cm)	Calculated k_{eff}
0	0.721 ± 0.006^a
0	0.780 ± 0.008^a
0	0.848 ± 0.006
1.27	0.904 ± 0.007
3.18	0.903 ± 0.007
3.81	0.890 ± 0.008

^aNo water between elements and concrete reflector.

*Comment Letter 47, p. 1, encl. 2.

4.5.5.16 Spent Fuel

A description of the cask upon which the following analysis is based is given in Section 4.5.3.1. Primary cask coolant, assumed to be helium at atmosphere pressure, is contained entirely within the cavity and not circulated outside the cask. The cask is cooled by a forced coolant system that maintains the cask wall temperature between 150 and 200°F under normal operating conditions. The cask is about 18 ft long and 4 ft in diameter and weighs 75 tons loaded. The fuel carried in the cask will have been irradiated to the design burnup (67,000 Mwd/metric ton). The cladding is expected to have very little ductility owing to radiation damage, particularly in the region of the core. The temperature distribution throughout the fuel assemblies and cask was calculated on the basis of a two-dimensional heat transfer model of the cask and the use of data from heat transfer experiments with an electrically heated fuel assembly. Temperatures calculated inside the cavity are shown in Table 4.5-17.

Table 4.5-17
AVERAGE TEMPERATURES OF THE FUEL CLADDING
AND CASK WALL UNDER NORMAL OPERATING CONDITIONS^a

	Temperature (°F)
Fuel Cladding	
Plenum	600
Peak	1040
Cask Wall	175

^aFor fuel assemblies cooled 360 days prior to shipment.

In a category-5 accident, the cask is assumed to impact into some highly resistant object at 75 mph in such a way as to cause one of the seals to fail. The actual mechanism required to cause the seal to fail is difficult to postulate considering the array of "soft" objects which are likely to be hit (the ground, etc.) and the energy absorption capability of the transporting vehicle.

The sequence of events assumed in the accident follows. The rail car carrying the cask derails and the cask hits a very rigid object end on, causing it to decelerate almost instantaneously. The fuel assemblies inside would be jammed against the end of the cask, and the deceleration forces are assumed to be sufficient to cause all fuel rods in each of the six assemblies to crack or in some way fail such that

whatever gaseous fission products are in the fuel rod will be immediately released to the cask cavity. Gas pressure in these fuel rods at the time of shipment will be about 400 psig at the existing temperature in the cask.

This pressure would be released to the cask cavity causing it to be pressurized to several atmospheres and create the driving force that could expel fission product activity from the cask should a leak path be available. The quantity of radioactive material released from the cask is assumed to be the product of the material released from the fuel pins to the cavity times the ratio of the pressure increase to total resultant pressure calculated as if the cask did not leak. This material would be released through the failed seal over a short time span, such as a minute. For purposes of analysis, this initial release has been defined as instantaneous.

Both the secondary forced coolant system and the neutron shield are assumed to be damaged in the accident to the extent that all liquid is lost. All heat must then be dissipated through the cask gamma shield across the now empty neutron shield and be dissipated to the atmosphere. A new equilibrium temperature distribution would develop in the cask, higher than for normal shipping conditions. The gas in the cask cavity, containing the remaining gaseous fission products that were not expelled following the instantaneous release, will expand to adjust to the new, higher equilibrium temperatures. In expanding, additional fission products will be forced out of the cavity with the helium coolant through the failed seal.

If the cask is involved in a 2-hr fire following the accident, cavity temperatures, for a time, will exceed the final equilibrium temperatures, causing the cavity gas to expand still more, releasing more activity than would be the case without a fire.

Temperatures of the fuel assemblies will also increase, but the temperature rise will be less than the temperature rise of the cavity wall owing to the importance of radiation in transferring heat throughout the fuel assembly and from the assemblies to the cavity wall. The maximum increase in the maximum fuel pin temperature is estimated to be less than 100°F. No significant additional release of fission products is expected from the ruptured rods to the cavity as a result of this 100 F° temperature rise.

Cavity wall temperatures calculated at the midplane and end of the cask under accident conditions are given in Table. 4.5-18.

Table 4.5-18
INSIDE CAVITY WALL TEMPERATURES
UNDER ACCIDENT CONDITIONS

	Temperature (°F)	
	Peak Fire	Final Equilibrium
Cask Midplane ^a	560	260
Cask End ^a	650	190

^aNormal operating temperatures were maintained at 200°F at the cask midplane and 150°F at the cask end by the cask forced-coolant system; the average cask wall temperature was about 175°F.

Activity releases from the cask were calculated based on the assumptions given above and are given in Table 4.5-19. The releases were assumed to be in two stages: the initial instantaneous release at the time of impact followed by a longer term release. The longer term release was calculated assuming a fire was both present and absent.

Table 4.5-19
ESTIMATED RELEASE OF RADIOACTIVITY
IN AN ACCIDENT INVOLVING A SPENT-FUEL CASK IF ALL RODS ARE RUPTURED

Isotope	Instantaneous Release (Ci)	Long-Term Release (Ci)	
		No. Fire	2-hr Fire
Kr-85	1.9×10^3	1.7×10^2	9.3×10^2
I-129	4.3×10^{-3}	3.9×10^{-4}	2.0×10^{-3}
I-131	2.1×10^{-9}	1.9×10^{-10}	1.0×10^{-9}
Sr-89	7.7×10^{-7}	3.2×10^{-8}	1.75×10^{-9}
Sr-90	5.5×10^{-6}	2.4×10^{-7}	1.28×10^{-4}
Cs-134	8.1	2.4×10^{-3}	3.2×10^1
Cs-137	4.2×10^1	1.26×10^{-2}	1.70×10^2

The length of time over which the longer term releases occur depends upon how rapidly the peak temperatures in the cavity are reached. In the case of fire, peak temperatures are reached about 3 hr after the accident occurs. In the case of no fire, it takes about one-half day to reach the new temperature equilibrium.

As a result of the loss of the neutron shield, the external cask surface dose rates will increase from about 10 to 500 millirems/hr; at 3 ft from the cask surface, increase would be from 4 to 200 millirems/hr.²⁹ The cask surface dose rate is well within the limit of 1000 millirems/hr at 3 ft following an accident as specified in the AEC regulations.³⁰

An examination of Table 4.5-19 indicates that the largest release of radioactive material occurs immediately after the breach occurs, at the time of the accident. A long term release that occurs within 12 hr after the breach is lower than the instantaneous release, but the longer release is increased somewhat (and is finished in about 3 hr) if accompanied by a fire.

Table 4.5-20 gives a summary of activity releases postulated for category-5 accidents for all shipments except spent fuel and high-level waste. The table also indicates the probability of a category-5 accident occurring during the year, based on the expected annual shipping distance and the probability of a category-4 accident happening per mile, given in Table II.A-4. Table 4.5-21 presents annual accident probabilities obtained using the data in Tables 4.5-1 and II.A-4.

4.5.6 Environmental Effects of Postulated Accidents

When an accident involves radioactive material shipments, it can result in environmental impacts stemming from both the nonradiological and radiological natures of the shipment. The former impact occurs because the shipment was made (e.g., injury or death to a truck driver caused by the impact in an accident or by a chemical release), whereas the latter impact occurs because of the radioactive nature of the cargo (e.g., contamination of property and radiation exposure to the public). These impacts were dealt with separately.

4.5.6.1 Nonradiological Consequences

Nonradiological impacts that would stem from accidents would be localized; they would involve property damage and possibly injury or death. Appendix II.C gives a discussion of the accident statistics that permit an evaluation of these various impacts. A package that must make an empty return trip (e.g., a spent-fuel cask) would produce a nonradiological impact on both legs of the journey. Disposable packages (e.g., low-level beta-gamma waste containers) would produce a nonradiological impact only one way.

Table 4.5-20
ACTIVITY RELEASE AND PROBABILITY FOR
CATEGORY 5 ACCIDENTS

Shipments	Activity Release in Category 5 Accidents (Ci)	Predominant Element	Annual Shipping Distance ^a (miles)	Probability of a Category 5 Accident (accidents/year)
<u>Unirradiated material</u>				
UF ₆	3.21	U	1,500	1.5×10^{-10}
UO ₂	1.32×10^{-9}	U	1,900	1.9×10^{-10}
<u>Fuel Fabrication Plant</u>				
Alpha Waste	0.058	Pu	5,000	5.0×10^{-10}
Low-level β-γ	0.0059	Sr	22,000	2.2×10^{-9}
<u>Power Station</u>				
Low-level β-γ	5.93	Mn, Fe, Co	2,100	2.1×10^{-10}
Tritium	2000	³ H	600	6.0×10^{-11}
<u>Fuel Reprocessing Plant</u>				
Alpha	0.058	Pu	770	7.7×10^{-11}
Low-level β-γ	5.93	Mn, Fe, Co	3,200	3.2×10^{-10}
Noble Gases	900,000	Kr	250	2.5×10^{-11}
Iodine	1.83	I	8	8.0×10^{-13}

^aOne-way distances involving packages that contain radioactive material.

Table 4.5-21
ANNUAL ACCIDENT PROBABILITIES
FOR HIGH LEVEL WASTE AND SPENT FUEL SHIPMENTS*

Category	High Level Waste	Spent Fuel
2	3.4×10^{-4}	6.0×10^{-3}
3	9.0×10^{-6}	1.6×10^{-4}
4	2.2×10^{-8}	4.0×10^{-7}
5	1.1×10^{-10}	2.0×10^{-9}

* per reactor year

4.5.6.1.1 Uranium Hexafluoride

Atmospheric Dilution of HF Released in a Transportation Accident

The postulated transportation accidents are described in Section 4.5.5.1, and the characteristics of release are summarized in Table 4.5-22 for accident cases 1 and 2. Release is either totally (case 1) or partially (case 2) associated with a 200-MW fire (see Appendix II.A.3.1). For the case 1 accident, 2420 kg are released over 90 min for a source strength of 448 g/sec, and the fire elevates the HF to 230 m, according to plume rise calculations given in Appendix II.I.1. After the plume rises to an effective stack height of 230 m, air concentrations of HF were calculated according to the Gaussian plume model. Atmospheric dispersion of HF during plume rise was also evaluated for a release height of 25 m. This circumstance considers the simultaneous plume rise and lateral dispersion of HF for the meteorological conditions given in Appendix II.I.1.

Two variations are described in Table 4.5-22 for the case-2 accident, early cylinder failure (2a) and delayed cylinder failure (2b). In 2a, 10% of the HF (285 kg) is released over an interval of one minute for a source strength of 4750 g/sec, all of which would be elevated by the fire to 230 m before dispersion would occur. The remaining 90% (2565 kg) would be released at ground level over a 4-hr interval with a source strength of 178 g/sec. In case 2b, 80% of the HF (2280 kg) would be released in the fire with a source strength of 37,833 g/sec for one minute, and the remaining 20% (570 kg) would be released at ground level over a 4-hr period with a source strength of 40 g/sec. The x/Q values are presented in Figure 4.5-10, and values for selected distances are given in Table 4.5-23 for the case-1 accident, and in Table 4.5-24 for cases 2a and 2b.

Effect of HF on Vegetation

At a given air concentration of HF, the duration of exposure must be considered in determining the effect of HF on vegetation and humans. The duration of exposure is 90 min for the case-1 accident. The duration of exposure is 240 min for ground-level releases for the case-2 accident, and for elevated release (230 m), the duration is one minute. These durations of exposure, however, are subject to the same uncertainties (atmospheric stability, wind velocity, stagnation, inversion, diffusion into the canopy, boundary-layer drag) that are discussed for the postulated accident at a fuel-fabrication plant (Section 4.3.9.1.5). Consequently, periods of exposure which are longer or shorter than those selected may occur for the given air concentrations of HF.

Table 4.5-22

TOTAL RELEASE AND RELEASE RATES OF HF FOR TRANSPORTATION ACCIDENTS

Characteristics of Release	Duration of Release (min)	Total HF Released (kg)	Release Rate (g/sec)
<u>Case 1 (Cylinder failure by impact during accident)</u>			
All HF Released in 200-MW Fire	90	2420	448
<u>Case 2 (No cylinder failure by impact during accident)</u>			
Case 2a (early cylinder failure in fire)			
10% of HF Released in 200-MW Fire	1.0	285 (10% associated with fire)	4,750
Balance of Release Independent of Fire	240	2545 (90% at ground level)	178
Case 2b (delayed cylinder failure in fire)			
80% of HF Released in 200-MW Fire	1.0	2280 (80% associated with fire)	37,833
Balance of Release Independent of Fire	240	570 (20% at ground level)	40

Table 4.5-23

CALCULATED CONCENTRATION OF HF RELEASED TO THE ATMOSPHERE
IN THE TRANSPORTATION ACCIDENT - CASE 1

Mode of Release	Distance (meters)	x/Q (sec/m ³)	Concentration of HF in Air (µg/m ³)
From Effective Stack	1250 ^a	4.20 E-7 ^b	188
Height of 230 m	1609 (1 mile)	3.7 E-7	165
	3218 (2 miles)	1.7 E-7	76
	8045 (5 miles)	9.2 E-8	41
From Release Height of	2000 ^a	3.5 E-7	157
25 m plus Plume Rise	1609 (1 mile)	3.4 E-7	152
to about 230 m	3218 (2 miles)	3.0 E-7	134
	8045 (5 miles)	1.8 E-7	81

^aDistance from source for maximal concentration of HF in the air.^b4.20 E-7 = 4.20 x 10⁻⁷.

X/Q VALUES FOR HF RELEASES RELATED TO
TRANSPORTATION ACCIDENTS

Figure 4.5-10

Table 4.5-24
CALCULATED CONCENTRATION OF HF RELEASED TO THE ATMOSPHERE
IN THE TRANSPORTATION ACCIDENT - CASE 2

Distance from Source (m)	From Fire		From Ground-level Release	
	x/Q^a (sec/m ³)	HF Concentration in Air ($\mu\text{g}/\text{m}^3$)	x/Q (sec/m ³)	HF Concentration in air ($\mu\text{g}/\text{m}^3$)
Case 2a (early cylinder failure) ^b				
100			5.1 E-3 ^c	9.1 E5
200			1.4 E-3	2.5 E5
500	1.1 E-9	5.2 E0	2.5 E-4	4.4 E4
1000	1.8 E-7	8.6 E2	6.8 E-5	1.2 E4
1609 (1 mile)	3.4 E-7	1.6 E3	3.0 E-5	5.3 E3
2000 ^d	3.5 E-7	1.7 E3	2.2 E-5	3.9 E3
3218 (2 miles)	3.0 E-7	1.4 E3	1.0 E-5	1.8 E3
8045 (5 miles)	1.8 E-7	8.4 E2	2.8 E-6	5.0 E2
Case 2b (delayed cylinder failure) ^b				
100			5.1 E-3	2.0 E5
200			1.4 E-3	5.5 E4
500	1.1 E-9	4.2 E1	2.5 E-4	9.9 E3
1000	1.8 E-7	6.8 E3	6.8 E-5	2.7 E3
1609 (1 mile)	3.4 E-7	1.3 E4	3.0 E-5	1.2 E3
2000 ^d	3.5 E-7	1.3 E4	2.2 E-5	8.7 E2
3218 (2 miles)	3.0 E-7	1.1 E4	1.0 E-5	3.9 E2
8045 (5 miles)	1.8 E-7	6.8 E3	2.8 E-6	1.1 E2

^aValue taken from Figure 4.5-10 (effective release height of 25 m, including plume rise).

^bSee Table 4.5-22 for characteristics of release.

^c5.1 E-3 = 5.1×10^{-3} .

^dDistance of maximal concentration for the elevated releases.

For the case-1 accident where the postulated release has an effective release height of 230 m, the maximum air concentration of HF would be $188 \mu\text{g}/\text{m}^3$ at 1250 m from the source (Table 4.5-23). The air concentration would be $165 \mu\text{g}/\text{m}^3$ at 1 mile and would decrease to 76 and $41 \mu\text{g}/\text{m}^3$ at 2 and 5 miles respectively.

The calculated concentrations for the 230-m release at distances of 1250 m to 1609 m (188 to $165 \mu\text{g}/\text{m}^3$) exceed the threshold for effect on green plants for a 90-min exposure.³¹ A 50% reversible inhibition of photosynthesis was observed with barley and alfalfa for a 2-hr exposure at a concentration approximately 200 ppb ($160 \mu\text{g}/\text{m}^3$). Bennett and Hill³¹ reported foliar necrosis (permanent damage) at a concentration of 150 ppb ($120 \mu\text{g}/\text{m}^3$), but inhibition of photosynthesis was reversible at concentrations as high as 200 ppb. There are few additional reports of effects on green plants induced by short-term exposure to HF (i.e., exposures of 1 to 2 hr). With foliar markings as an endpoint, an extrapolation of McCune's³² graphs indicates that a threshold concentration for damage to common sensitive plants (conifers, fruit trees, sorghum, tomato, citrus) would range from 300 to $1000 \mu\text{g}/\text{m}^3$ for a 90-min exposure. The threshold concentration for foliar damage to corn would be 100 to $200 \mu\text{g}/\text{m}^3$. Thus, a temporary effect would be expected on sensitive green plants between 1000- and 2000-m distances; damage from HF would be limited to a 0.2-square mile area. Effects on barley, corn, and alfalfa would be expected. The calculated air concentration of HF would be less than the threshold for damage to other sensitive and resistant species. No effect on green vegetation would be expected at distances less than 0.5 miles or at distances 2 miles or greater from the release.

If an effective release height of 25 m vs 230 m is postulated, maximum ground level HF concentrations occur at 2000 m from the source compared with 1250 m for the 230-m release height (Table 4.5-23). The principal effect of the lower release height is manifested at 2 to 5 miles from the source where concentrations would be a factor of two greater and thus temporary inhibition of photosynthesis may occur over a larger area (i.e., 4.1 square miles compared with 0.2 square miles when the effective release height is 230 m).

Either 20% (case 2b) or 90% (case 2a) of the HF released in the case-2 accident would be at ground level, and the release would not be associated with a fire (Table 4.5-22). Maximum HF concentrations for a ground-level release with no plume rise would be at the point of release. The maximum HF concentration ($910,000 \mu\text{g}/\text{m}^3$) in air at 100 m (Table 4.5-24) would be 70 times greater than the maximum calculated for an HF release associated with the fire ($13,000 \mu\text{g}/\text{m}^3$ at

2000 m, case 2b). The contribution of the HF released in the fire to the total HF in air is negligible to a distance of 500 m; HF from the ground-level release is the major contributor. At 100 m, HF concentrations expected in the case-2 accident (200,000 to 900,000 $\mu\text{g}/\text{m}^3$) probably will severely damage if not kill plants exposed during the 4-hr period of release.³³ At 1 mile, HF concentrations of 1300 to 5300 $\mu\text{g}/\text{m}^3$ exceed the threshold for foliar damage (Figure II.J-1, Appendix II.J). Annual plants, including agricultural crops, probably would be severely affected by the HF to a distance of 1 mile. Leaves and fruits of perennial plants would be severely damaged or destroyed in the same area, but plant survival would be expected at distances of 1 mile or greater. Foliar necrosis of sensitive plants and temporary inhibition of photosynthesis for sensitive and moderately resistant plants can be expected to a distance of 5 miles. If the released HF were confined to a single sector, the affected areas would be 0.2 square miles where plants would be killed or severely damaged and an additional 4.7 square miles where plants would be temporarily affected but recovery would be expected.

Dispersion of HF from the fire component in the case-2 accidents results in maximum concentrations (1,700 to 13,000 $\mu\text{g}/\text{m}^3$) at 2000 m. The duration of exposure is given as 1 min, the postulated duration of release. Reference data are not available for evaluating effects on vegetation at these concentrations for a 1-min exposure. Thus damage to vegetation at such high concentrations and short exposures cannot be predicted.

Accumulation of Fluoride by Forage

The accumulation of fluoride in vegetation was calculated for the respective intervals of exposure given in Table 4.5-22, 1.5 hr for the case-1 accident and 4 hr and 1 min for the case-2 accident. Ground concentrations of fluoride were calculated from the product of air concentration for the residence time of the plume and a deposition velocity of 0.003 m/sec (see Appendix II.L). Fluoride accumulation in forage was estimated at selected distances beginning at 100 m for ground-level releases and beginning at 1000 m for elevated releases (Tables 4.5-25 and 4.5-26).

Table 4.5-25

CALCULATED CONCENTRATION OF FLUORIDE IN VEGETATION
AT SUCCESSIVE DISTANCES FROM THE SOURCE FOR A 1.5-HR
EXPOSURE TO HF FROM THE CASE-1 TRANSPORTATION ACCIDENT

Fluoride Concentration				
Distance from Source (m)	Deposition Rate ($\mu\text{g}/\text{m}^2/\text{sec}$)	In the Air ^a ($\mu\text{g}/\text{m}^3$)	On the Ground ($\mu\text{g}/\text{m}^2$)	In Vegetation ^b (ppm)
Effective Stack Height of 230 m				
1250	0.56	188	3020	20
1609 (1 mile)	0.50	165	2700	18
3218 (2 miles)	0.23	76	1240	8.3
8045 (5 miles)	0.12	41	650	4.3
Release Height of 25 m plus Plume Rise to about 230 m				
2000	0.47	157	2540	17
1609 (1 mile)	0.46	152	2480	16
3218 (2 miles)	0.40	134	2160	14
8045 (5 miles)	0.24	81	1300	8.6

^aValues for x/Q and for fluoride concentration in air are taken from Table 4.5-22

^bStanding crop of $150 \text{ g}/\text{m}^2$ is assumed

At the calculated air concentrations of HF for either release height (25 m or 230 m) associated with the case-1 accident, fluoride will not accumulate in forage to levels which would be unsafe for livestock. For a 1.5-hr exposure, the maximum fluoride concentration would be 20 ppm at 1250 m for the source (Table 4.5-25, 230 m release height). The concentration would be less than 20 ppm at distances closer to or farther from the source. Fluoride concentrations in forage not exceeding 30 to 50 ppm are considered safe for consumption by livestock (Table II.J-3).

For the case-2 accident, the calculated fluoride accumulation in forage over a 4-hr exposure period will range from a high concentration of 58,000 to 260,000 ppm at 100 m to a concentration of 40 to 140 ppm at 5 miles from the release. The concentration at 5 miles exceeds the limit (30 to 50 ppm) which is considered safe

Table 4.5-26

CALCULATED CONCENTRATION OF FLUORIDE
AT SUCCESSIVE DISTANCES FROM THE SOURCE FOR THE CASE 2 TRANSPORTATION ACCIDENT

Distance from Source (m)	Fluoride Concentration (ppm)				
	In Air (as HF) ^a		On the Ground (as Fluoride)		In Vegetation
	From Elevated Release	From Ground-Level Release	From Elevated Release	From Ground-Level Release	
Case 2a: Early Cylinder Failure					
100	0	9.1 E5 ^b	0	3.9 E7	2.6 E5
200	0	2.5 E5	0	1.1 E7	7.2 E4
500	5.2 E0	4.4 E4	9.0 E-1	1.9 E6	1.3 E4
1000	8.6 E2	1.2 E4	1.5 E2	5.2 E5	3.5 E3
1609(1 mile)	1.6 E3	5.3 E3	2.9 E2	2.2 E5	1.5 E3
3218(2 miles)	1.4 E3	1.8 E3	2.5 E2	7.8 E4	5.2 E2
8045(5 miles)	8.6 E2	5.0 E2	1.5 E2	2.2 E4	1.4 E2
Case 2b: Delayed Cylinder Failure					
100	0	2.0 E5	0	8.6 E6	5.8 E4
200	0	5.5 E4	0	2.4 E6	1.6 E4
500	4.2 E1	9.9 E3	7.5 E0	4.3 E5	2.8 E3
1000	6.8 E3	2.7 E3	1.2 E3	1.2 E5	8.1 E2
1609(1 mile)	1.3 E4	1.2 E3	2.3 E3	5.2 E4	3.6 E2
3218(2 miles)	1.1 E4	3.9 E2	2.0 E3	1.7 E4	1.3 E2
8045(5 miles)	6.8 E3	1.1 E2	1.2 E3	4.8 E3	4.0 E1

^aValues for the concentration in air are taken from Table 4.5-24.

^b9.1 E5 = 9.1×10^5 .

for continuous consumption by livestock. A concentration of 1200 ppm, when fed continuously to beef heifers for 60 days, caused health to deteriorate so that the animals had to be slaughtered.³⁴ Fluoride accumulation in forage would exceed this level (1200 ppm) to a distance of 0.5 to 1.0 miles from the release. To permit grazing on HF-contaminated pastures would be inadvisable because fluorosis symptoms would be expected, and prolonged ingestion of forage with fluoride exceeding 1200 ppm may be lethal to cattle.

The fluoride concentration in forage would be depleted by weathering processes according to the function $e^{-.032D}$ (Appendix II.L). By applying this function

to the initial concentration at 1000 m, for example, the fluoride concentration would decrease to 45 to 196 ppm after 90 days, the estimated length of a growing season. Thus, to a distance of 1000 m from the release, the contaminated forage crop should not be utilized for forage until the next growing season. At 5 miles from the release, the fluoride in forage will have decreased to 18 to 46 ppm, after a weathering period of 35 days. Thus, cattle could safely graze contaminated pastures after allowing for a 35-day weathering period.

Effect of HF on Humans

Where a 200-MW fire elevates the HF plume to a height of 230 m (case-1 accident), no detrimental effect on humans is predicted. The maximum calculated air concentration at ground level is $188 \mu\text{g}/\text{m}^3$ ($0.19 \text{ mg}/\text{m}^3$) which is less than 10% of the limit for a daily 8-hr occupational exposure ($2.0 \text{ mg}/\text{m}^3$).³³

The air concentration of HF for the case-2 accident would range from 200 to 900 mg/m^3 at 100 m from the release. These concentrations exceed the highest tolerable exposure to 100 mg/m^3 (Table II.J-2, Appendix II.J) for a 1-min duration. The duration of release for this case-2 accident is 240 min.

The HF concentrations at 1000 m are 2.7 and 12 mg/m^3 for delayed and early cylinder release respectively. The maximum HF at 1000 m is 50% of the tolerable concentration for a several-minute exposure and exceeds the recommended public emergency limit ($8 \text{ mg}/\text{m}^3$) for a 10-min exposure.³³ At 1 mile the air concentration for the early cylinder failure ($5.3 \text{ mg}/\text{m}^3$) would exceed the public emergency limit of $4 \text{ mg}/\text{m}^3$ for a 60-min exposure. Clearly, a failure of the HF cylinder, in which the HF is not elevated in the plume rise, would constitute a hazardous condition where evacuation of people from the path of the plume would be necessary. If the plume would be confined to a single sector, evacuation may be required for a distance of 1 to 2 miles.

Summary of Impact of HF from a Transportation Accident

For the case-1 accident, where the HF is elevated to 230 m, a temporary 50% reduction in the photosynthesis of sensitive plants is expected at distances of 1 to 2 miles from the source of release. Foliar damage would be expected for the most sensitive crop plants (barley, alfalfa, corn), but not for deciduous forest species. The affected area would be less than 0.5 square mile if the effective release height was 230 m, but an area as large as 4.1 square miles would be affected if the release height was 25 m. Fluoride accumulation in forage would not exceed

levels which would be considered unsafe for consumption by livestock. No detrimental effect on humans is anticipated when the HF plume rises to 230 m as the result of the fire.

For the case-2 accident the air concentration at 100 m would be nearly 5000 times greater than when the HF is elevated by fire to 230 m. The resultant concentrations of HF (200,000 to 900,000 $\mu\text{g}/\text{m}^3$) probably will severely damage if not kill sensitive plants during the 4-hr period of release. At 1 mile from the release, the HF concentration (1300 to 5300 $\mu\text{g}/\text{m}^3$) will exceed the threshold for damage to foliage; agricultural crops would be damaged. Foliar necrosis and temporary inhibition of photosynthesis would be expected to a distance of 5 miles; plants would be temporarily affected in a 4.7-square mile area but recovery would be expected.

A high concentration of 58,000 to 260,000 ppm in forage at 100 m decreasing to a low of 50 to 140 ppm fluoride at 5 miles is expected for releases from the case-2 accident. Concentrations of fluoride in forage at 0.5 and 1.0 miles would cause serious deterioration in the health of livestock if they were permitted to graze on forage that possesses more than 1200 ppm fluoride. It would be necessary to hold cattle temporarily from pasture in a single sector for distances of 0.5 to 5.0 miles from the accident. At 5 miles, cattle could be returned to pasture after a 35-day weathering period, but longer withholding periods would be necessary at lesser distances.

Concentrations of HF in air to a distance of 1 mile resulting from the case-2a accident (early cylinder failure) exceed the public emergency limit of 4 mg/m^3 for a 60-min exposure. Where the HF release is not accompanied by plume rise it would be necessary to evacuate people from the path of the plume for a distance of 1 to 2 miles.

Accidental UF_6 Releases to Aquatic Environments

Toxicity of UO_2F_2 , HF, and F^- An analysis of the accidental release of UO_2F_2 and HF to aquatic environments must be limited in scope because no direct studies have been made on the effects of these fluorine compounds on aquatic organisms. Numerous experiments have been conducted on the toxicity of sodium fluoride³⁴⁻³⁶ and uranyl fluoride^{37,38} to mammals, and on the effects³⁴ of gaseous HF on terrestrial plants and mammals. Investigators concluded that the toxicity of uranyl fluoride was due primarily to its fluorine content.^{37,38} Anemia, loss of weight, and bone and kidney changes could be duplicated by ingestion of equivalent amounts

of fluorine as sodium fluoride. The analyses reported here are based on the assumption that the toxicity to aquatic organisms of F^- from sodium fluoride is about equal to that of F^- from ionization of HF.

A rather limited amount of data is available on the toxicity of sodium fluoride³⁹⁻⁴⁶ to aquatic biota. These data are summarized in Table 4.5-27. These data indicate that the toxicity of fluoride is dependent on many factors; the most important appear to be species composition, exposure time, water chemistry, and temperature. Apparently there are two classes of fluoride tolerance for fishes: less susceptible, warmwater forms, such as goldfish and carp; and more susceptible coldwater forms, such as trout. For example, the 20-day median tolerance limit (TL_m) for carp and rainbow trout is 75 to 91 ppm and 2.7 to 4.7 ppm respectively.⁴⁵ A similar situation is observed in toxicity responses for chlorine and its chemical residuals.⁴⁷

The value of the data in Table 4.5-27 for analysis of the impact of UO_2F_2 and HF is limited by several considerations. First, the toxicity of HF is expected to be greater than that of the F^- ion (as observed for terrestrial organisms).³⁴⁻³⁶ At concentrations ≤ 10 ppm in aquatic systems, fluorine should be present mainly as F^- ion. At concentrations > 10 ppm, a portion of the dissolved fluorine should be present in the more toxic HF form, as a consequence of the very low dissociation constant ($K_{diss} = 3.5 \times 10^{-4}$).⁴⁸ Second, the studies on toxicity of sodium fluoride to biota have been conducted for widely varying exposure times (12 to 825 hr). None of these experiments provide direct data on acute (~ 3 hr) or long-term chronic (> 100 days) exposures to elevated fluoride levels.

Angelovic et al.⁴⁶ observed that rainbow trout exposed to a fluoride concentration of 25 ppm suffered complete mortality within 26 to 86 hr. Trout exposed to 13 ppm fluoride exhibited 70 to 100% mortality within 32 to 118 hr. It appears that 50% of the trout would be killed in 72 hr by exposure to about 10 ppm fluoride ion (under similar conditions of water chemistry).

Accident Analysis Cases

This section addresses the effects of several representative cases for release of UF_6 into flowing freshwater environments. For all cases, it is assumed that the cylinder contains a $UO_2F_2 + HF$ solution (from the reaction of $UF_6 + H_2O$) which completely empties in 2.4 hr.¹⁶ Three cases are considered which demonstrate that a relatively small change in the accident configuration can result in changing the impact from negligible to locally acute.

Table 4.5-27

TOXICITY OF SODIUM FLUORIDE (NaF) TO FRESHWATER BIOTA

Test Organism	Concentration of NaF (ppm)	Remarks	Reference ^a
<u>Daphnia magna</u> (cladoceran)	504	Toxicity threshold or dose to immobilize in 48 hr, 25°C	1
	270	Toxic threshold 48 hr, 23°C	2
<u>Gambusia affinis</u> (mosquito fish)	1240	24-hr TL ^b	3
	925	48-hr TL _m ^m 21-24°C	
<u>Carassius auratus</u> (goldfish)	1000	Killed in 60 to 120 hr, hard water	4
	1000	Killed in 12 to 29 hr, soft water	
	100	No kill in 4 days, hard water	
	120	Killed in 4 days	5
<u>Tinca vulgaris</u> (tench)	678	Lethal dose	6
<u>Cyprinus carpio</u>	75 to 91	20-day TL _m ^m , 18.3 to 23.9°C	7
<u>Salmo gairdneri</u> (rainbow trout)	5.9 to 7.5	10-day TL _m ^m ; 7.2°C	8
	2.6 to 6.0	10-day TL _m ^m ; 12.8°C	
	2.3 to 7.3	10-day TL _m ^m ; 18.3°C	
	2.7 to 4.7	20-day TL _m ^m ; 12.8°C	7
		(fingerlings)	
	222 to 273	424-hr TL _m ^m ; 7.8°C, eggs	
	242 to 261	214-hr TL _m ^m ; 12.8°C, eggs	
	237 to 281	167-hr TL _m ^m ; 15.6°C, eggs	
	61 to 85.3	825-hr TL _m ^m ; 12.8°C, eggs	

^aReferences:

1. B. G. Anderson, Sewage Works J. 18(1): 82-87 (1946).
2. G. Bringmann and R. Kuhn, Gesundheits-Ing. 80(4): 115-120 (1959).
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4. M. M. Ellis, Bull. Bur. Fisheries 48: 365-437 (1946).
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6. P. Simonin and A. Pierron, Compt. Rend. Soc. Biol. 24: 133 (1937).
7. J. M. Neuhold and W. F. Sigler, Trans. Amer. Fish Soc. 89(4): 358-370 (1960).

^bTL (median tolerance limit) = 50% mortality within the specified time period for continuous exposure to the stated concentration.

Case 3a: The solution flows into a deep pool in the stream. It is assumed that the pool has a surface area of $\sim 100 \text{ ft}^2$ and that there is a 2-ft-deep quiescent zone between the surface of the $\text{UO}_2\text{F}_2 + \text{HF}$ solution and the flowing zone. The assumed streamflow is 100 cfs. Mass transfer from the solution is assumed to occur by diffusion alone.

For these conditions, the bulk of the $\text{UO}_2\text{F}_2 + \text{HF}$ solution remains in the pool for a relatively long period of time, or until the first major flood. The estimated in-stream concentrations over that period would be on the order of 3.4 ppb UO_2F_2 and 0.9 ppb HF. No significant effects either on stream biota or water use would be expected under the above conditions outside the immediate pool area. All biota residing in the pool would be expected to suffer complete mortality. It might be possible to recover the solution from the pool within a relatively short period of time.

Case 3b: The solution arrives in a shallow pool essentially undiluted. For a stream velocity of 1 fps over the top of the pool, the Richardson number is very high (~ 10 to 100). This produces an extremely low entrainment rate (about 10^{-5} to 10^{-6}).⁴⁹ Thus, the solution would be entrained into the stream over a period of 10 to 100 days.

The resulting fluorine concentrations are shown in Table 4.5-28 for representative streamflows⁵⁰ in the United States. As shown in Table 4.5-28, no significant effects either on stream biota or water use would be expected under the above conditions outside the area of the pool in streams with a flow ≥ 100 cfs. As in case 3a, all biota in the pool would suffer complete mortality.

Case 3c: Assume the solution travels ≥ 25 ft along the river bottom before reaching a pool. According to Nielson's formula,⁵¹ this leads to a dilution of approximately 10^3 and results in a density difference which is insufficient to prevent almost complete mixing. It is assumed that complete mixing occurs with the streamflow.

The resulting concentrations are shown in Table 4.5-29. As in case 3b, the effects vary widely depending on stream size. Virtually complete mortality of sensitive species would occur in a soft water stream having a flow of 100 cfs. A streamflow well in excess of 1000 cfs would be required to assure no severe adverse effects on stream biota. In addition, ingestion of a single mouthful (about 100 ml) of stream water by man during the 2.4-hr release period (100 cfs flow) would result

Table 4.5-28

CONCENTRATIONS OF FLUORINE IN A STREAM (CASE 3b)

Stream Flow (cfs)	Time to Empty Pool of UO_2F_2 + HF Solution (days)	Concentration of Fluorine, as HF or F^- (ppm)
100	10	1.3
100	100	0.13
1000	10	0.13
1000	100	0.013

Table 4.5-29

CONCENTRATIONS OF FLUORIDE IN A STREAM (CASE 3c)

Stream Flow (cfs)	Concentration of Fluorine, as HF or F^- (ppm)
100	130
1,000	13
10,000	1.3

in the intake of 0.013 g of fluoride (less than 1% of the lethal dose for an adult).³⁶

An essentially infinite number of cases could have been considered in the UF_6 accident analysis, e.g., estuaries, marine systems, lakes, and impoundments, of different sizes and chemical characteristics. However, it can be shown, from an analysis of the above three cases chosen to describe UF_6 release into a stream, that the resultant impact may vary from an almost negligible effect to localized acute effect on man and the ecosystem. To attempt to discuss what might be a reasonable accident case is clearly impractical owing to the uncertainties imposed by environmental variables and by lack of data on the effects of F^- and HF on aquatic biota. It would appear to be more reasonable to prevent the potential acute environmental effect of such an accident by changing the method of transport, increasing safeguards, using a different chemical form, or requiring shipping containers of greater integrity.

4.5.6.1.2 Unirradiated Fuel

All unirradiated-fuel-material shipments considered here would be made by truck. These materials would be expected to be transported in the public domain much as any other commodity is handled and would be subject to the same potential accident environment. Safety during shipment would not depend on special routes, use of escorts, and the like, but on the strength and reliability of the packages used to transport the material. Appendix II.M gives a discussion of the rigid standards to which the packages would be designed and built.

A summary of the expected costs plus injuries and deaths is given in Table 4.5-30. See Appendix II.C for more detailed statistics on risks.

4.5.6.1.3 Irradiated Fuel

All spent-fuel shipments would be assumed to be made by rail to the fuel-reprocessing plant. If the reactor should not have a rail spur, the cask would be trucked to the nearest railhead. Table 4.5-31 indicates the postulated injuries and deaths, based on rail round-trip shipments only.

4.5.6.1.4 Waste

Shipment of waste would be made both by rail and truck. Shipment would be made to Federal repositories or licensed burial grounds. Where the waste containers are disposable, only one-way distances were used in estimating the nonradiological

Table 4.5-30

SUMMARY OF THE PROJECTED NONRADIOLOGICAL IMPACTS INVOLVED
IN SHIPMENTS OF UNIRRADIATED MATERIAL

Material	Shipments per Year	Average Distance per Shipment (miles)	Total Miles ^a	Projected Accidents per Year ^b	Projected Injuries ^b per Year	Projected Deaths ^b per Year	Projected Property Damage Costs (\$/year) ^b
UF ₆	2.01	750	3,000	7.7×10^{-3}	4.0×10^{-3}	3.5×10^{-4}	18
UO ₂	2.58	750	3,900	9.5×10^{-3}	5.3×10^{-3}	4.4×10^{-4}	23
PuO ₂	2.91	750	4,400	1.0×10^{-2}	5.6×10^{-3}	4.7×10^{-4}	24
Fresh Fuel	20.1	750	30,000	7.5×10^{-2}	3.8×10^{-2}	3.2×10^{-3}	170

^aRound-trip miles.^bSee Appendix II.C for accident statistics.

Table 4.5-31

SUMMARY OF PROJECTED NONRADIOLOGICAL
IMPACTS OF IRRADIATED-MATERIAL (SPENT FUEL) SHIPMENTS

	Value
Shipment Characteristics	
Number per Year	26.75
Distance, miles	
Average per Shipment	750
Total Annual (roundtrip)	40,000
Projected Impacts per Year ^a	
Accidents, number	9.0×10^{-2}
Injuries, number	2.5×10^{-1}
Deaths, number	2.5×10^{-2}
Property Damage, \$	550

^aSee Appendix II.C for accident statistics.

consequences. Where empty casks had to be returned to the point of origin, round-trip distances were considered. Table 4.5-32 summarizes the projected deaths, injuries, and property damage from these shipments.

4.5.6.2 Effects of Radioactive Releases

Potential doses to individuals and to a population were calculated for accidents postulated to occur during transport and which would cause release of radioactive material (see Tables 4.5-33 and 4.5-34). Source terms for these calculations are in Tables 4.5-19 and 4.5-20. The methodology and dose factors used in these calculations were the same as those used to calculate doses that would result from radionuclide releases from plants during normal operation (Appendix II.E). For normal releases from plants, the dose per year of operation was calculated. For accidental releases from transportation accidents, the total potential dose per accident was calculated.

The meteorological conditions assumed to exist during and for a short time after the accident were indicated in Appendix II.I, along with the assumed population distribution. These meteorological conditions and population distribution were also used in calculating doses due to radionuclide releases from normal plant operation. However, the accidental release was assumed to occur when the wind was blowing in a single direction. Thus, the radioactive emissions would stay within a single 22.5° sector during the entire diffusion and transport of the radioactive material. The releases were assumed to occur at an effective stack height of 230 m because of the fire assumed to accompany the accidents.

All ground deposition was assumed to have occurred at the time of cloud passage. Deposition velocities were the same as those used in dose calculations associated with normal plant operation. The doses calculated here are within the AEC accident guidelines of 10 CFR 100 (25 rem or less whole-body dose to an individual and 300 rem or less dose to the thyroid of an individual).

Potential radiation doses to individuals and a population for accidents involving breaching of a spent-fuel shipping cask were calculated for each release situation and are given in Table 4.5-35. This table includes doses assuming both the presence and absence of cesium because of the uncertainty as to whether this element would escape from the cask. The same uncertainty exists for strontium. However, the tabulated doses were not affected by the presence of the strontium because of the relatively small amount available for release.

Table 4.5-32

SUMMARY OF NONRADIOLOGICAL IMPACTS OF WASTE SHIPMENTS

Type of Shipment	Shipments per Year	Average Distance per Shipment (miles)	Total Miles ^a	Projected Accidents per Year ^b	Projected Injuries per Year ^b	Projected Deaths per Year ^b	Projected Property Damage Costs (\$/year) ^b
<u>Fuel-Fabrication Plants</u>							
Alpha Waste	5	1000	10,000	2.2×10^{-2}	6.4×10^{-3}	6.4×10^{-3}	150
Low-Level Beta-Gamma Waste	45	500	22,000	5.5×10^{-2}	2.9×10^{-2}	2.4×10^{-3}	130
<u>Power Stations</u>							
Low-Level Beta-Gamma Waste	4.22	500	2,100	5.2×10^{-3}	2.9×10^{-3}	2.3×10^{-4}	12
Tritium	0.60	1000	600	1.5×10^{-3}	7.8×10^{-4}	6.3×10^{-5}	3
<u>Fuel-Reprocessing Plants</u>							
Alpha Waste	0.77	1000	1,500	3.3×10^{-3}	9.6×10^{-3}	9.5×10^{-4}	20
Alpha-Beta-Gamma Waste	25.57	1000	54,000	1.2×10^{-2}	3.5×10^{-1}	3.4×10^{-2}	710
Low-Level Beta-Gamma Waste	6.3	1000	3,200	8.0×10^{-3}	4.3×10^{-3}	3.5×10^{-4}	18
Cladding Hulls	1.3	1000	2,600	5.8×10^{-3}	1.7×10^{-2}	1.7×10^{-3}	35
High-Level Waste	0.75	1500	2,250	5.0×10^{-3}	1.5×10^{-2}	1.4×10^{-3}	30
Noble Gases	0.164	1500	500	1.2×10^{-3}	6.4×10^{-4}	5.4×10^{-5}	
Iodine	0.0076	1000	15	3.7×10^{-5}	2.0×10^{-5}	1.6×10^{-6}	

^aAll waste packages except those that involve low-level beta-gamma material and tritium are calculated on a roundtrip basis. Disposable low-level waste is shipped one way only.

^bSee Appendix II.C for accident statistics.

Table 4.5-33

POTENTIAL INDIVIDUAL DOSES
RESULTING FROM ACCIDENTAL RADIONUCLIDE RELEASES DURING TRANSPORT^a

Shipments	Inhalation and Submersion in the Passing Cloud						
	Total-Body Dose			Highest Reference Organ Dose			
	Dose (rems)	Critical Radio-nuclides	Percentage Dose from Critical Radionuclides	Dose (rems)	Reference Organ	Critical Radio-nuclides	Percentage Dose from Critical Radionuclides
UF ₆	2.6 E-4 ^b	U-234 U-238	11 87	1.1 E-2	Lung	U-234 U-238	11 87
UO ₂	<1 E-6			<1 E-6			
Alpha Solid Waste	2.5 E-5	Pu-238 Pu-240 Pu-241	34 12 45	1.1 E-3	Bone	Pu-238 Pu-240 Pu-241	31 11 50
Low-Level Beta-Gamma (fuel-fabrication plant)	<1 E-6	Sr-90	100	8.0 E-6	Bone	Sr-90	100
Low-Level Beta-Gamma (power station)	1.0 E-6	Mn-54 Fe-55 Co-58	60 23 13	4.8 E-5	Lung	Mn-54 Co-58	65 23
Tritium	8.6 E-6	H-3	100	8.6 E-6	Total body	H-3	100
Noble Gases	3.0 E-5	Kr-85	100	0			
Iodine	1.2 E-6	I-129	100	5.2 E-4	Thyroid	I-129	100

^aCalculations are for category-5 accidents. Effective release height was 228 m. The highest potential doses to an individual occurred at about 1250 m from the accident, and these doses were about 16% higher than the ones tabulated here for 1000 m.

^b2.6 E-4 = 2.6×10^{-4} .

Table 4.5-34

POTENTIAL POPULATION DOSES
RESULTING FROM ACCIDENTAL RADIONUCLIDE RELEASES DURING TRANSPORT^a

Shipments	Inhalation and Submersion in the Passing Cloud						
	Total-Body Dose			Highest Reference Organ Dose			
	Dose (man-rem)	Critical Radio- nuclides	Percentage Dose from Critical Radio- nuclides	Dose (man-rem)	Reference Organ	Critical Radio- nuclides	Percentage Dose from Critical Radio- nuclides
UF ₆	0.78	U-234 U-238	11 87	31	Lung	U-234 U-238	11 87
UO ₂	<1 E-6 ^b			<1 E-6			
Alpha Waste	7.1 E-2	Pu-238 Pu-240 Pu-241	34 12 45	3.2	Bone	Pu-238 Pu-240 Pu-241	31 11 50
Low-Level Beta-Gamma (fuel-fabrication plant)	1.6 E-3	Sr-90	100	2.3 E-2	Bone	Sr-90	100
Low-Level Beta-Gamma (power station)	3.0 E-3	Mn-54 Fe-55 Co-60	60 23 13	1.4 E-1	Lung	Mn-54 Co-58	65 23
Tritium	3.4 E-2	H-3	100	3.4 E-2	Total body	H-3	100
Noble Gases	0.4	Kr-85	100				
Iodine	3.6 E-3	I-129	100	1.55	Thyroid	I-129	100

^aCalculations are for category-5 accidents. Effective release height was 228 m.

^b1 E-6 = 1×10^{-6} .

Table 4.5-35

INHALATION AND SUBMERSION DOSES
FROM TRANSPORTATION ACCIDENTS THAT BREACH THE SPENT-FUEL CASK

Accident Description	Cs Included ^c		Individual Dose (rems) ^a		Population Dose (man-rems) ^b	
	Yes	No	Dose	Highest Organ Dose (inhalation)	Dose	Highest Organ Dose (inhalation)
Short-time Release at Ground Level	✓	✓	4.3 E0 1.3 E-2	5.4 E1 ^d (lung,insoluble) 4.2 E-2 (thyroid)	2.1 E0 6.5 E-3	2.7 E1 (lung,insoluble) 2.1 E-2 (thyroid)
Continuous Release at Ground Level (no fire)	✓	✓	2.4 E-3 1.2 E-3	1.6 E-2 (lung,soluble) 3.8 E-3 (thyroid)	1.2 E-3 6.0 E-4	8.0 E-3 (lung,insoluble) 1.9 E-3 (thyroid)
Continuous Release at 230 m above Ground Level (fire included)	✓	✓	5.3 E-4 <1.0 E-6	6.7 E-3 (lung,insoluble) <1.0 E-7 (thyroid)	6.8 E0 2.5 E-3	8.6 E1 (lung,insoluble) 7.8 E-3 (thyroid)

^aCalculated for an adult standing 50 m from the accident for a ground-level release and 1200 m from the accident for the continuous release at 228 m.

^bPopulation beyond 50 m and within 50 miles of the accident.

^cCesium was assumed soluble for whole-body dose calculations and insoluble for doses to the lung. This results in the highest contributions from cesium to each entry in the table.

^d5.4 E1 = 5.4×10^1 .

For the long-term release involving a fire, the effective release height was calculated to be 230 m because of the large vertical thermal currents produced in a 2-hr fire. The short-time release and continuous release without a fire were assumed to occur at ground level. This assumption is judged to be particularly conservative since many situations occur in real accidents that tend to increase the effective release height above ground level. If the cask remained on the transporting vehicle, for example, a release height of a few meters above ground level would be realistic. However, because of lack of definitive information on activity releases not accompanied by a fire, only ground level releases were assumed.

The dose to an individual (given in relative units) vs distance from the accident is shown in Figure 4.5-10 for several effective release heights. This figure shows that the highest dose to an individual and the distance from the source at which the peak exposure occurs are extremely dependent upon the release height. The maximum individual doses for each release height are always less than the same doses for a ground-level release, but they asymptotically approach the ground-level release dose with increasing distance from the accident. The population or man-rem dose is not as dependent on the stack height since only a small fraction of the assumed population distribution is within about one kilometer of the accident, where the calculated doses are so strongly dependent upon the effective release height of the activity.

Individual doses reported in Table 4.5-35 are at a distance of 50 m from the accident for ground-level releases and at 1200 m for the continuous release with a fire; assuming an effective release height of 230 m this latter distance gives the maximum dose to the individual (Figure 4.5-10). The calculated doses for a ground-level release condition given in Table 4.5-35 can be converted to doses resulting from the same activity releases at other effective release heights by use of the conversion factors given below in Table 4.5-36.

Table 4.5-36
FACTORS TO CONVERT TABULATED DOSES FOR A GROUND-LEVEL
RELEASE TO DOSES FOR OTHER RELEASE HEIGHTS

Release Height (m)	Divide Individual (ground-release) Doses by	Divide Population (ground-release) Man-rem by
3	4.1	~1.1
10	53	~1.7
100	6,600	~2.0
230	32,000	~2.5

The calculated doses given in Table 4.5-35 are almost exclusively the result of inhalation of, rather than submersion in, the radioactive gases. The maximum doses for both an individual and the population occur as a result of the initial, instantaneous release at the time of an accident and are dependent to a large extent on the assumption of the presence of cesium in the release of activity.*

The doses calculated for cesium were based on the assumption that it was both soluble (when computing whole-body doses) and insoluble (when computing doses to the lungs). This inconsistent, but conservative, treatment of cesium was necessary since the solubility of whatever cesium might be released is not known. If the cesium were soluble, the tabulated reference organ doses to the lung would be reduced by a factor of about 3.5.

The doses given in Table 4.5-35 can be compared to the accident guidelines set forth by AEC in 10 CFR 100, that is, 25-rem whole-body dose and 300-rem thyroid dose. It is clear that much of the uncertainty in the dose calculations for breach in a spent-fuel cask is due to uncertainties in the amount (if any) of cesium released and its solubility after release.

Some organizations, such as Natural Resources Defense Council, are of the opinion that the basis on which these dose numbers were developed is not correct and do not reflect the maximum dose rate to the population under the worst conceivable conditions. They suggest that worst weather conditions should be used in conjunction with high population density and the worst (category-5) accident accompanied by the maximum release of radioactive material.

*EPA (Comment Letter 53, p. 20) suggested that ruthenium should be included in the source term for radioactive releases from transportation accidents and that the dose due to inhalation of ruthenium should be included in assessing the consequences of such a release. The AEC does not agree with this suggestion. Unlike cesium, in its metallic state ruthenium exhibits a negligible vapor pressure at the temperatures under consideration in the cask (ruthenium melts at about 4450°F). The oxides of ruthenium are volatile, with RuO_3 being the most volatile. However, ruthenium must be exposed to an oxidizing atmosphere before it can convert to an oxide form. The cask atmosphere and the blanket gas in the fuel rods are not oxidizing under normal conditions. Only in an accident in which cask containment was breached could the cask cavity become oxidizing and then only slowly unless the entire lid of the cask was forced off of the cask. The events required to force the cask lid off the cask completely in an accident appears to be of such low probability as to be considered incredible. Therefore, assuming violation of containment, leakage should occur slowly. The cask would pressurize as the fuel rods ruptured and so gas transport would tend to be from inside out to the environment, keeping the atmosphere non-oxidizing. Significant releases of ruthenium would not be expected in transportation accidents.

Such a dose number might be useful if it could be correlated with a probability of the event occurring under the conditions assumed. Only in this way could a rational approach be used in judging whether the risk is high or low.

In this regard, the probabilities developed in the category-5 accidents (see Appendix II.A) have been applied to the releases estimated when various shipping containers are involved in a category-5 accident. Results are given in Section 4.5.7.

4.5.6.3 Nuclear Energy Centers

The term "nuclear park" or "nuclear energy center" applies to a nuclear industry complex or cluster in which the nuclear fuel is fabricated, used, and reprocessed on the same or contiguous sites. This requires that fuel-fabrication and fuel-recovery facilities be located in the cluster with the nuclear power plant. In such a cluster, transportation of unirradiated and irradiated nuclear fuel for the power plant would be limited to movement on the site.

The probability of a severe accident involving both unirradiated and spent fuel would be reduced by virtue of the facts that transport distances would be considerably shortened and that vehicle speeds would be lower. Vehicle-Vehicle collisions could be almost eliminated because of the controls that might be exercised over the shipments. In addition, trained personnel and equipment would be readily available should an accident occur.

However, the packages used to ship fresh and spent fuel are expected to be very well engineered, based on many years of experience in testing and shipping LWR fuel. The probability of an extremely severe accident, the type which might contribute to releasing radioactive materials to the environment, appears to be very low (see Appendix II.A). Reducing these numbers still further by the establishment of nuclear energy centers may produce only limited benefits. In addition, nuclear energy centers may not reduce the need for the shipment of wastes and surplus plutonium that is produced in the center. Thus, some offsite shipments of radioactive material would probably be required.

4.5.7 Evaluation of Risk of Radioactive Material Shipments

Section 4.5.7 has been added to provide additional information on the public health consequences of transportation accidents. Risk is defined as the product of the probability that something will happen and the consequences, or results, if the event does happen. Mathematically,

Risk = probability times consequences.

Since information on the probability, or frequency, with which an extremely severe, category-5 accident may occur is available (Appendix II.A) and the dose to a uniform population from such accidents is presented in Tables 4.5-34 and 4.5-35, the risk associated with these shipments may be estimated.

EPA uses as a guide an estimate that the dose of one million man-rems annually to a large population may produce an excess of health effects over other causes equal to the following:

- (1) 200 fatalities if the dose is to the whole body,
- (2) 200 non-lethal cancers if the dose is to the whole body, and
- (3) 300 serious health effects if the dose is restricted to the reproductive organs.

Health consequences may therefore be estimated once a dose to the general population has been determined.

For example, the probability of a category-5 accident occurring has been estimated at 1×10^{-13} accidents/vehicle mile (see Table II.A-4). Should such an accident occur to a cask containing spent reactor fuel, and as a result of that accident a leak occur in the cask, the surrounding population might receive an estimated total of 27 man-rems to the lung over a short time span and an additional 86 man-rems to the lung should the accident involve a 2-hr fire that disperses leaking fission products over a wide area (see Table 4.5-34).

Most of the estimated dose comes from the presence of radioactive cesium, a fission product. This cesium would be associated with shipments of core fuel assemblies as opposed to blanket assemblies. Table 4.5-1 indicates that for each 1000-MWe LMFBR there will be 22.83 shipments involving the core assemblies and that each shipment might travel 750 miles. Risk is then calculated as follows:

$$\begin{aligned} \text{Risk} &= (\text{accidents/vehicle mile})(\text{miles/shipment})(\text{shipments/reactor year}) \\ &\quad (\text{max. man-rems/accident})(200 \text{ deaths}/10^6 \text{ man-rem}) \\ &= (1 \times 10^{-13})(750)(22.83)(27 + 86)(200/10^6) \\ &= 3.86 \times 10^{-11} \text{ deaths/reactor year} \end{aligned}$$

Using the same approach, Table 4.5-37 was developed for all accidents in which radioactivity was postulated to be released.

The numbers presented in this table are based on the conservative assumption that every category-5 accident will result in the releases specified in Table 4.5-20 and 4.5-19. From that point of view the results for category-5 accidents considered are conservative. It should be pointed out that the risks, or indication of health effects, per reactor year presented in Table 4.5-37, are not totals since possible releases from other, less severe (categories 1 to 4) accidents are not included. With the exception of category-4 accidents, releases from such accidents are likely to be small and insignificant with regard to risk.

Table 4.5-37
THE RISK ASSOCIATED WITH CATEGORY-5
ACCIDENTS AND THEIR POTENTIAL RELEASES

Material Shipped	Risk per Reactor Year		
	Deaths	Cancers	Health Effects
UF ₆	9.3×10^{-13}	9.3×10^{-13}	1.4×10^{-12}
UO ₂	3.9×10^{-21}	3.9×10^{-21}	5.8×10^{-21}
Total Alpha Waste	3.7×10^{-13}	3.7×10^{-13}	5.6×10^{-13}
Total Beta-Gamma Waste	1.8×10^{-14}	1.8×10^{-14}	2.7×10^{-14}
Tritium	4.1×10^{-16}	4.1×10^{-16}	6.2×10^{-16}
Noble Gases	1.9×10^{-15}	1.9×10^{-15}	2.8×10^{-15}
Iodine	2.3×10^{-16}	2.3×10^{-16}	3.4×10^{-16}
Spent Fuel	3.9×10^{-11}	3.9×10^{-11}	5.8×10^{-11}
	4.0×10^{-11}	4.0×10^{-11}	6.0×10^{-11}

In general, only the most severe, category-5 accidents were evaluated in Appendix II.N. However, should releases occur in some category-4 accidents, the quantities released might, in some cases, approach the releases estimated for category-5 accidents. Since the category-4 accident frequency is 200 times that of the category-5 accident frequency (Table II.A-4) the risk might approach a factor of 200 times that shown in Table 4.5-37. However, it should be pointed out that the probabilities used here (accidents per reactor year) are based on the assumption that there is a release for each accident. This is obviously a very conservative assumption, and the risks in Table 4.5-37 are at least several orders of magnitude too high for this reason.

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4.6 RADIOACTIVE WASTE MANAGEMENT

4.6.1 Introduction

As is typical of all nuclear power plants, the nuclear processes that take place in the LMFBFR reactor result in the formation of radioactive fission products and actinide elements within the fuel rods. Although this formation is the principal source of the radioactivity produced by LMFBFRs, radioactive wastes are generated in essentially all segments of the LMFBFR fuel cycle.

Radioactive wastes generated during fuel cycle operations are divided into two primary categories--high level and other than high level. High-level liquid waste is defined in Appendix F, 10 CFR 50,¹ as the aqueous waste resulting from the 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. The product from solidifying this waste is "high-level solid waste." All other waste, regardless of its source, falls into the other-than-high-level category. This category has been further subdivided for purposes of waste management considerations into beta-gamma solid wastes, alpha solid wastes, and noble gas wastes.

The high-level waste from the reprocessing of the LMFBFR fuel, since it contains essentially all of the fission products and a fraction of the fuel materials, is the most significant waste material from the standpoint of hazard and difficulty of handling. It generates sufficient heat to require substantial cooling. It emits large amounts of potentially hazardous ionizing radiation, and it must be carefully isolated or contained essentially forever to prevent significant quantities of the more highly toxic radionuclides from entering man's environment. Present AEC regulations¹ require that the liquid high-level waste from fuel reprocessing be converted to a stable solid material within five years after separation in the fuel-reprocessing step and that it be encapsulated and shipped to a Federal repository within ten years of its production for long-term management by the AEC. The annual high-level waste production for reprocessing of fuel from one 1000-MWe LMFBFR power plant when converted to solid form will occupy a volume of approximately 55 ft³.

The volume of packaged other-than-high-level waste that will be generated in the various steps in the fuel cycle is considerably larger than the volume of high-level waste. During fuel fabrication operations, waste consisting of such things as rags, sweepings, boxes, and piping contaminated with uranium and plutonium

will be generated. The average concentration of radionuclides in the fuel fabrication waste will be low (from a few thousandths to as high as several curies per cubic foot), and the volume of packaged other-than-high-level waste amounts to several thousands of cubic feet per year for the model LMFBR annual fuel requirement. On the order of one or two thousand cubic feet per year of solid other-than-high-level waste will be produced at the reactor itself in the form of filters, spent resins, laboratory residues, and other material. These wastes will contain small amounts of fission products and activation products. A few cubic feet of tritium waste, probably in the form of Ca(OH)_2 , will also be produced annually at the reactor. Fuel reprocessing operations generate wastes consisting of general laboratory waste, small tools, gloves, clothing, and other material, contaminated with fission products and/or fissile materials. The volume of solid other-than-high-level waste generated as a result of reprocessing the spent fuel from the model LMFBR will also amount to several thousand cubic feet per year, depending on the use of compaction and/or other volume reduction methods at the fuel reprocessing plant.

Because of the long-term potential risk from release of radioactive waste produced in nuclear reactors, the successful implementation of a waste management system requires very careful development and planning.² The two major protective measures utilized in the management of radioactive waste are isolation and containment, although substantial interaction can and does exist between the two. Isolation is used to mean factors that influence the time required for migration to man's environment. Typical factors could include distance, the ion exchange capacity of the earth, or the lack of water. If waste is adequately isolated so that the migration times are greater than the time for substantial radioactive decay, isolation alone provides adequate protection. Conversely, if adequate containment is provided by man-made barriers--again for the length of time for decay, the waste could be placed in many selected locations in man's environment. Here, the word "barrier" is used to include the matrix for the waste (for example, silicate glasses), enclosures such as metallic casings, and man-made containment such as heavy concrete buildings.

The near-term waste management program for high-level waste discussed in this statement assumes retrievable surface storage for safekeeping until a safe and acceptable ultimate disposal method has been selected and tested. The surface storage facility is considered to be a safe method for all credible occurrences. It does, however, impose the requirement of human surveillance requirements for future generations. Thus a substantial program of analyses of potential disposal

alternatives is going forward as is a development program on disposal in geological formations, one of the possible ultimate disposal alternatives.

Current regulations indicate that wastes other than high-level wastes can be disposed of in commercial burial grounds. In the case of cladding hulls, noble gas cylinders, and long-lived alpha-emitting plutonium and other transuranic wastes, future regulations will probably require storage in repositories. Development programs are going forward to identify the most appropriate methods of disposal for these materials.

Clearly, the general problems associated with radioactive waste management apply to all fission reactor fuel cycles. Therefore, while the following material focuses on the LMFBR fuel cycle wastes, it also applies in large part to management of LWR and HTGR wastes.

4.6.1.1 Brief History of High Level Waste Management Developments

During the historically brief span of World War II and shortly thereafter, the nuclear fuel cycle, and its attendant generation of radioactive wastes, emerged essentially full blown. The characteristics of the wastes generated in the course of producing plutonium for nuclear weapons are basically the same as those of wastes from the nuclear power industry. From the outset, the high-level liquid radioactive wastes generated in the fuel reprocessing step have been discharged to underground storage tanks. Large quantities (multimillions of gallons) of such wastes have been accumulated, and they reside in the form of liquids, sludges, or damp salt cakes in underground tanks at the AEC's Hanford Works near Richland, Washington, and at the Savannah River Plant near Aiken, South Carolina. Smaller volumes of dry, solid high-level waste are stored within stainless steel bins in underground vaults at the National Reactor Testing Station (NRTS) in Idaho. Leaks have occurred in some of the oldest tanks. The most recent and largest leak is described and discussed in Appendix II.F.

The use of tank storage for liquid wastes has always been considered an interim measure. Since the earliest days of the atomic energy program, research and development work has been conducted with the objective of providing techniques for the more permanent storage of high-level wastes in a way that: (1) ensures isolation from the environment and (2) requires little reliance on human surveillance and intervention during the time the waste could be a potential biological hazard.

In 1955, at the request of the AEC, a committee of prominent earth scientists was established by the National Academy of Sciences--National Research Council (NAS-NRC) to consider the disposal of high-level radioactive wastes within the United States. In their report,³ which was issued in 1957, this committee recommended that the AEC follow two general approaches: (1) a continuation of the efforts that had been initiated on the development of processes for the solidification of high-level liquid wastes and (2) an exploratory program on the possible use of naturally occurring salt formations as places for long-term isolation of these wastes.

With respect to waste solidification, development work has successfully demonstrated the technical feasibility of converting liquid wastes to stable, solid forms by several processes.⁴ One of these processes has been in practical use at the NRTS for over a decade. Over half of the high-level liquid wastes generated at the Idaho Chemical Processing Plant have been converted to a granular solid that is now stored in high-integrity underground bins at the Idaho site. Solidification not only decreases the potential for the accidental release of this kind of waste to the environment but also reduces volumes to be stored considerably and contributes to safety in the transfer of these materials to permanent disposal sites.

On the basis of the results in waste solidification developments and other factors, the AEC announced, in July 1970, a policy requiring commercial fuel reprocessors to solidify high-level liquid wastes within five years of their generation and to ship the solidified wastes to a Federal Repository within ten years of their generation at the reprocessing plant.

The bedded salt deposits, suggested by the NAS-NRC committee for the long-term isolation of high-level radioactive wastes, underlie large areas of the continental United States. They are the residues of ancient seas that have dried up, have been deeply covered by sedimentation processes, and have been isolated from the biosphere for over 200 million years. Other geological formations, such as salt domes and granites, also have been effectively isolated from the biosphere for geologic periods and could provide permanent storage places. Thus, the concept of using bedded salt, or other suitable geologic formations, for the underground depositing of solidified wastes--followed by backfilling and/or sealing off selected repository areas as they are filled--seems highly promising as a means of permanently disposing high-level wastes with minimum impact on present and future generations.

In 1957, the AEC initiated a program to test the feasibility of storing high-level wastes in some of the massive salt deposits underlying large areas of this country.

At that time, there was no assurance that any of the solidification processes being actively pursued would be successful. Consequently, initial studies were concerned with the storage of liquid waste in salt caverns. A field demonstration was carried out in a mine in Hutchinson, Kansas, using simulated, though not radioactive, liquid wastes. These tests showed that there were many practical problems to be resolved in developing this concept.

More recently, the feasibility of storing solidified high-level waste in deeply buried salt was explored in an experiment known as Project Salt Vault,⁵ conducted by the Oak Ridge National Laboratory in an unused salt mine near Lyons, Kansas, between 1963 and 1966. In this experiment, containers of highly radioactive test reactor fuel, used to simulate the radioactive and thermal properties of solidified waste, were buried (and later retrieved) and effects on the salt measured. This experiment obtained data on the engineering properties of salt at elevated temperatures and indicated that there were no gross detrimental effects on the stability of the salt as a result of exposure to heat or radiation and no minor effects other than those which had been predicted.

Based on the encouraging results of the waste solidification program and Project Salt Vault, the AEC gave serious consideration to the use of the abandoned salt mine at Lyons, Kansas, as a pilot facility and an initial Federal Repository for the storage of solidified high-level wastes. Toward this end, the AEC initiated detailed studies of the geology and hydrology of the Lyons site to evaluate its adequacy for the long-term isolation of radioactive wastes. These studies revealed possible routes for the entry of water into the Lyons mine, namely: (1) three of 29 old oil and gas bore holes in and near the site may be difficult to plug adequately against water seepage and (2) the mining activities of a nearby operating salt mine include plans for expanded operations involving the use of water to dissolve and remove salt.

The difficulties encountered with the Lyons site as a specific candidate for a waste repository emphasize the need for slow, deliberate care and thoroughness in the evaluation and selection of prospective repository areas, to which the isolation of high-level wastes would be entrusted for geologic periods. Accordingly, in May 1972, the AEC dropped plans for any immediate selection of a permanent repository site. It chose instead to broaden the development program to include assessment of a broader range of alternative options and to provide time for any or all of the more promising options to be thoroughly assessed through experimental and proof-testing activities. The content of the program as currently conducted and planned is described in

documents cited among the references for this section.^{2,6} One of the current plans contemplated is a proof-test facility in bedded salt, or other type geologic formation, will be placed in operation in the early 1980's, at a site yet to be selected. (For additional discussion of waste disposal options, see Section 7.3.5.)

As an interim measure for storing solidified wastes from commercial operations, the AEC (also in May of 1972) announced plans for the design and construction of a Retrievable Surface Storage Facility (RSSF). The RSSF is intended to be available for use when the first quantities of solidified waste are ready for delivery from commercial reprocessing plants in the mid-1980's. The RSSF will be designed to store these wastes safely for as long as a century, if necessary, but as its name implies, the AEC intends that these stored wastes be retrieved and moved to permanent storage facilities when such facilities are established. The goal of the geologic storage development program is to have permanent disposal available in two to three decades.

Because retrievable surface storage is the high-level waste management technique certain to be in operation during the period covered by this Statement, the major part of this section is addressed specifically to such techniques and their foreseeable impacts. Accordingly, it seems appropriate to identify and discuss, at this point, the more general commentary received in response to the draft version of this Statement on the subject of high-level wastes and their management. Such discussion provides an opportunity to add perspective to this important subject, which might not otherwise arise in the subject matter of this Statement.

4.6.1.2 Critiques and Responses

Most of the commenters expressing views on waste management in an RSSF made points with which the AEC has little or no quarrel. The following selected quotations are representative of such comments:

The matter of man-made vaults at or near the surface for long term storage (500 years) is...uncertain. Their maintenance depends upon the existence of social and political institutions, the permanence of which cannot be guaranteed over time.*

We abolished slavery in this country a hundred years ago. It is highly ironic that we are now setting up a system of future slavery, because we are placing in bondage future generations to look after our radioactive wastes....**

*Comment Letter 38d (NRDC), p. 8, quoting the Union of Concerned Scientists, The Nuclear Fuel Cycle, p. 9 (October 1973).

**Comment Letter 8 (G. A. and I. H. Younghein), p. 6.

The addition of the preceding subsection, and its cited references, to this Statement is intended to convey the message that the AEC has, from its beginning, recognized the full implications of high-level wastes to present and future generations and has conducted a continuous development effort in the matter of ultimate disposal methods that will not burden mankind with caretaker functions. This effort is continuing, and its details are stated in documents^{*+} available to the general public. It can only be re-emphasized that the RSSF expedient is intended to accommodate relatively near-term needs, while providing time for careful and thorough pursuit of longer-term goals. There is, after all, a major difference between a facility designed to last 100 years, if needed, and a facility designed to take care of the first 100 years of a vastly longer-term need. Semantic difficulties may account for many disparate points of view on this subject.

On the other hand, a major line of commentary expresses concern about the risks and problems associated with high-level wastes and questions the availability of acceptable long-term solutions. Some selected, but representative comments in this category are:

I consider it irresponsible to plan a huge system of future nuclear power plants, when the problem of storage of wastes is still unresolved, and there is no real solution in sight.*

The draft environmental statement is seriously deficient because it does not put into clear focus the need for a societal decision on the larger question of whether or not to impose an unprecedented burden on this and thousands of succeeding generations....**

...the AEC apparently has no idea when its long term solutions to disposition of the LMFBR wastes may be "ultimately proved to be practical" ... The draft EIS fails to consider whether our "obligations to future generations" might be better served by refraining from generating the wastes until a long-term waste management strategy is proved and adopted.†

...a very large production of nuclear energy necessarily means the mass production of radioactive poisons in quantities which are terrifying.

This is the basic reason for the opposition against the use of atomic energy, which has become a worldwide controversy.

No general solution for the isolation of long-lived radioactive wastes from the biosphere, for the necessary many thousands of years, is yet in hand: that is, despite a wide variety of proposals, "experts" still disagree on

*Comment Letter 16 (J. T. Edsall), p. 1.

**Comment Letter 38d (NRDC), p. 6.

†Comment Letter 51 (Environmental Impact Assessment Project), pp. 52, 53.

††Comment Letter 38d (NRDC), p. 4, quoting Hannes Alfvén, "Fission Energy and Other Sources of Energy," Science and Public Affairs, p. 6 (January 1974).

*+See references 9, 10 and 12 of Section 7 (available for public inspection at AEC public document room, Washington, D. C. and other locations).

whether any of them will suffice.... It is impossible to be complacent about expansion in the use of nuclear power without having a solution in hand.*

There is clearly a body of opinion that the risks and problems associated with high-level radioactive wastes are so large and so unprecedented that society ought to forego the use of fission energy, or at least stop using it until the long-term waste disposal problem is solved. In principle, these options are available to society, but if decisions were made to abandon fission energy temporarily or permanently, society would not divest itself of immediate and long-term waste disposal problems--it would simply confine the problem to those amounts of fission products, plutonium, and other transuranics in existence when the last fission reactor (of any type--nuclear power or plutonium production) would have been shut down.

Stated differently, a societal decision on whether or not to impose an unprecedented burden on future generations appears to be foreclosed by the large quantities of fission products and plutonium already in existence. For the avoidance of caretaker burdens on future generations, present-day society will have to continue the quest for long-term management techniques, whether fission energy is continued in use or not. (See Section 5.4.2.2 for additional discussion of societal impacts.)

As previously stated, the AEC posture in the development of high-level waste-disposal methods is to take the time necessary to gain a high degree of confidence in the validity of methods that may ultimately be used. In this sense, the timing of the establishment of a permanent waste repository, or other disposal method (excluding surface storage), remains indefinite. Nevertheless, decisions can quite reasonably be expected within the next two decades on the suitability of stable geologic formations, such as bedded salt, for the permanent isolation of wastes from the biosphere. A decade or so of measurements and observations in the pilot facility(ies) to be in operation in the 1980's should provide a reasonable basis for such decisions.

In conclusion, many organizations and individuals expressed concern because an ultimate disposal method for "high-level" wastes is not available now.** This

*Comment Letter 38d (NRDC), p. 8, quoting the Continuing Committee of the Pugwash Conferences on Science and World Affairs, Pugwash Newsletter 11, p. 23 (July and October 1973).

**Comment Letters 8 (G. A. and I. H. Younghein), p. 6; 16 (J. T. Edsall), p. 1; 22 (Mid-America Coalition for Energy Alternatives), p. 10; 32 (R. W. Freedman), pp. 1, 2; 38d (NRDC), pp. 4-9, 21, 22; 39 (Minnesota Pollution Control Agency), p. 2; 51 (Environmental Impact Assessment Project), pp. 1, 58, 59, 61; 53 (EPA), p. 25; 55 (G. I. Rochlin and J. P. Holdren), p. 4; 58 (C. R. Lewis), p. 1; 60 (C. P. Brauman), pp. 2, 3; Hearing Letter 21 (Scientists' Institute for Public Information), p. 1; Pre-draft Letter 15 (NRDC), encl., p. 4.

concern is a common one and frequently appears to be based on a misunderstanding to the effect that AEC's program to develop the RSSF was also an abandonment of interest in permanent disposal. The AEC has, indeed, stated that high-level waste could be safely stored in an RSSF, using any of the design concepts now under consideration, for as long as society wished to provide surveillance and maintenance. However, the AEC has not stated that indefinite RSSF storage is either desirable or necessary. On the contrary, the evaluation of promising geological formations and sites leading to a pilot geological repository, which in turn could lead to permanent disposal, is a major part of the AEC high-level waste program (see Section 7.3 for further discussion of alternatives to the RSSF). Retrievable surface storage would be eliminated if an alternative that does not require perpetual guardianship were developed and proven to be acceptable. Use of an RSSF for an interim period of time (up to 100 years if necessary) would not preclude transfer of wastes from the RSSF to a permanent radioactive waste repository. Extensive surveillance would be required only for that period of time (less than 100 years) prior to use of an acceptable alternative to retrievable surface storage.

4.6.2 Radioactive Waste Generation

4.6.2.1 Reactor Wastes

The radioactive materials in the primary coolant system of the LMFBR power plants will be of two principal forms: those that will exist at some equilibrium condition in the liquid sodium and volatile fission products, most of which will escape from the liquid sodium to the inert cover gas.⁷ Radioactive materials in the coolant will consist of fission products and neutron-activated structural materials that are released to the coolant by corrosion processes. Some of the gaseous or volatile fission products such as iodine and most of the tritium will be captured and retained in the coolant through chemical interaction with the liquid sodium. Approximately 95% of the tritium in the fuel elements will diffuse through the stainless steel cladding of the fuel assemblies into the coolant and be converted to sodium tritide (Na^3H). Those gases not retained by the sodium will be collected in the cover gas repurification system. The liquid sodium in the primary system will be continually processed to remove these radioactive materials for collection and disposition as other-than-high-level waste.

The radioactive gas removal systems that will be installed in the LMFBR plant will be designed to remove virtually all fission products from the primary cover gas system. The radioactive fission-product gases are principally krypton and xenon isotopes. All but Kr-85 have very short half-lives. These gases will be temporarily stored for decay in a holdup system until the radioactivity from all gases

but Kr-85 has decayed to an insignificant level. Assuming a 1% leakage to the primary coolant system of gaseous fission products from the fuel in the reactor, approximately 1500 Ci/year of long-lived Kr-85 activity will be collected by the cover gas cleanup system. Filled cylinders of krypton can be stored at the site or shipped to a possible future waste repository designed for long-term storage of gaseous wastes.

All liquid wastes generated at the reactor as a result of laboratory and decontamination operations will be treated to remove radioactive material by methods such as evaporation or ion exchange. The concentrate containing the radionuclides will be solidified and stored as solid wastes.

Solid wastes generated at the reactor will consist typically of filters from the heating and ventilation system, deactivated primary coolant sodium cold traps, analytical laboratory and liquid waste treatment residues, contaminated tools and parts, and waste such as plastic bags, footcovers, paper towels, and protective clothing.* These wastes will be compacted and packaged in 55-gal drums which are sealed and then shipped to a low-level waste burial ground. About 6 ft³ of tritium waste per year per reactor, in the form of $\text{Ca}(\text{O}^3\text{H})_2$, will be included in these solid wastes. Based on experience at existing commercial power reactors and making allowances for the differences in plant characteristics, between 1000 and 2000 ft³ of beta-gamma solid waste per year is estimated to be generated at a 1000-MWe LMFBR nuclear power station. This waste, which would be contained in 140 to 280 55-gal drums, would utilize between 200 and 400 ft²/year at a burial ground.

4.6.2.2 Reprocessing Plant Wastes

Most of the radioactive wastes generated in the LMFBR fuel cycle will end up in the reprocessing plant wastes. The radioactive noble gases produced in the LMFBR fuel reprocessing plant are expected to be contained and stored in that plant for eventual offsite shipment to a repository.

High-level liquid wastes generated as a result of processing LMFBR fuel will be converted to solid form and shipped to a Federal repository. Constituents of high-level liquid wastes for LMFBRs are shown in Table 4.6-1. Characteristics of solid wastes will vary depending upon the solidification process used (see Table 4.6-2).⁸ After five years of decay, 10 million Ci of radioactivity will remain

*This range of materials is also typical of the low-activity solid wastes generated at reprocessing and refabrication plants.

Table 4.6-1

CONSTITUENTS OF HIGH-LEVEL LIQUID LMFBR WASTES

Constituent	Grams/Metric Ton of Fuel ^a
Reprocessing Chemicals and Corrosion Products	
H	1,261
Fe	26,229
Ni	3,272
Cr	6,919
BO ₃	98,040
NO ₃	243,555
Subtotal	379,276
Fuel Components	
U ^b	4,322
Pu ^b	475
Other Actinides	1,560
Fission Products	32,961
Subtotal	39,318
TOTAL	418,594

^aMixed core and blanket, with boron as soluble poison, 10% of cladding dissolved, 1249 liters/metric ton, 37,100 MWd/metric ton exposure, and 41.8% thermal efficiency.

^b0.5% product loss to waste.

Table 4.6-2

CHARACTERISTICS OF SOLIDIFIED HIGH-LEVEL WASTE^a

	Pot Calcine	Spray Phosphate Ceramic	Phosphate Glass	Borosilicate Glass ^b	Fluidized Bed Calcine
Form	Scale	Monolithic	Monolithic	Monolithic	Granular
Description	Calcine Cake, Friable	Ceramic Hard, Tough	Glass Hard, Brittle	Glass Hard, Brittle	Calcine, Mean Particle Diam. about 100 to 500 μm
Bulk Density, g/ml	1.2 to 1.4	2.7 to 3.3	2.7 to 3.0	3.0 to 3.5	1.0 to 1.7
Wt% Fission Product Oxides (Max.)	90	30	25	50	50
Thermal Conductivity, $\text{W}/(\text{m}^2)(^\circ\text{C}/\text{m})$	0.3 to 0.4	1.0 to 1.4	0.8 to 1.2	1.0 to 1.4	0.2 to 0.4
Leachability in Cold Water, g/cm ² -day	10.0 to 10^{-1}	10^{-3} to 10^{-5}	10^{-4} to 10^{-6} ^c	10^{-5} to 10^{-7}	1.0 to 10^{-1}

^aSource: K. J. Schneider et al., "Advanced Waste Management Studies--High-level Waste Disposal Alternatives," vol. 1, Report BNWL-1900, Battelle-Northwest, Richland, Washington, May 1974; see also "High Level Waste Disposal Alternatives," USAEC Report WASH-1297, May 1974 (a summary of vols. 1 through 9 of Report BNWL-1900).

^bProduced by either spray or fluidized-bed calcining followed by melting or in-canister vitrification processing.

^cDevitrified phosphate glass exhibits increased leachability (leach rates = 10^{-2} to 10^{-3} g/cm²-day).

in the high-level wastes produced as a result of reprocessing one year's fuel from a 1000-MWe LMFBR. When converted to solid form, this waste will occupy about 55 ft³ and could be contained in nine cylinders, each 12 in. in diam and 10 ft long.

Other solid wastes from the reprocessing plant include fuel cladding hulls and hardware and low-level waste similar to that generated at the reactor. The cladding hulls, which will amount to about 165 ft³/year for a 1000-MWe LMFBR, will be packaged in disposable steel containers and sent to a licensed burial facility or possibly to a future repository.

The low-level solid wastes from the reprocessing plant will include both beta-gamma and alpha wastes and will amount to 5000 to 10,000 ft³/year from a 1000-MWe LMFBR. This waste may be packaged in either 55-gal drums for disposal at a commercial burial ground or, if contaminated with plutonium or other transuranic elements and if storage at a repository is required, could be packaged in 25-ft³ containers, about 2 ft in diam by 10 ft long. In the former case, 700 to 1400 55-gal drums would be used. If sent to a repository, 200 to 400 containers would be used.

4.6.2.3 Fabrication Plant Wastes

The solid wastes generated at an LMFBR fuel fabrication plant will consist primarily of paper, plastic, gloves, and other wastes--some of which is contaminated with low levels of plutonium. Although the activity levels are low, the long-lived alpha-emitting plutonium and other transuranic elements that are the source of some of the contamination are expected to lead to a requirement that these alpha wastes be stored in a repository rather than in a commercial waste burial ground. Based on current experience with fabrication of plutonium and uranium oxide fuels, 10,000 to 30,000 ft³ of solid waste are expected to be produced as a result of fabricating the fuel for one year's operation of a 1000-MWe LMFBR with about a fifth of this waste being identified as alpha waste.

4.6.2.4 Comparison of Radioactive Solid Wastes

The estimated quantities of both high-level and other-than-high-level wastes generated in one year by 1000-MWe reactors and associated fuel cycle components are compared in Table 4.6-3 for the LMFBR and LWRs with plutonium recycle.

4.6.3 Waste Management Facilities

The technology and facilities established for management and disposal of radioactive waste from the LWR nuclear power industry will be directly applicable to the

Table 4.6-3

COMPARISON OF ESTIMATED ANNUAL QUANTITIES OF
RADIOACTIVE SOLID WASTES FROM AN LWR AND AN LMFBR

	1000-MWe LWR with Plutonium Recycle	1000-MWe LMFBR
<u>Produced at Reactor Site</u>		
<u>Other-than-High-Level</u>		
Cubic Feet	8,000-12,000	1,000-2,000
Number of 55-Gal Drums	560-1,120	140-280
Burial Ground Area Used, sq ft	800-1,600	200-400
<u>Noble Gases</u>		
Cylinders	-	5
<u>Produced at Reprocessing Plant Site</u>		
<u>High-Level Solid</u>		
Cubic Feet	55	55
Number of 12-In.-Dia. x 10 Ft. Long Containers	9	9
RSSF Repository Space Required, sq ft	11	11
<u>Cladding Hulls</u>		
Cubic Feet	60	165
Number 3.5 ft ³ containers	17	48
Repository ^a or Burial Space Required, sq ft	12	35
<u>Other Solid Wastes</u>		
Cubic Feet	600-4,000	5,000-10,000
Number of 55-Gal Drums	80-540	700-1,400
Burial Ground, sq ft	100-800	1,000-2,000
<u>Noble Gases</u>		
Cylinders		1
<u>Produced at Fabrication Plant Site</u>		
<u>Other-than-High-Level Including Plutonium Contaminated Wastes</u>		
Cubic Feet	10,000-30,000	10,000-30,000
Number of 55-Gal Drums	1,400-4,200	1,400-4,200
Repository ^a Space, sq ft	2,000-6,000	2,000-6,000

^aIf required by future regulations.

management of radioactive waste generated by the LMFBR and its fuel cycle. With the exception of technology under development to improve the retention capability for gaseous fission products and plutonium wastes, the containment and management of LMFBR wastes are based on methods that, at the time of the commercial entry of the LMFBR, will have been utilized on a routine basis for several years for LWR plant wastes⁹ and LMFBR demonstration plant wastes.

4.6.3.1 Retrievable Surface Storage Facility for High-Level Radioactive Waste

On May 18, 1972, the AEC announced that it would use the technique of retrievable surface storage to manage solidified high-level radioactive waste.^{10,11} A reference design of the proposed facility has been developed that uses a water-cooled basin for storage of the sealed waste canisters. Alternative concepts being considered include storage in air-cooled vaults and storage out of doors in sealed thick-walled steel casks. The water basin technique has been used for the past 30 years for storage of irradiated fuel, both from the AEC and commercial reactors. It is a thoroughly proven, safe, flexible technique that offers a high capacity heat sink, good shielding, and the advantage of visual surveillance.

The RSSF will include provisions for receiving, inspecting, and safely storing packaged wastes from fuel reprocessors. The facility will be designed to hold safely, for at least 100 years, all of the commercial high-level waste produced through the year 2000.¹² It will be designed to:

- (1) Confine radioactivity and assure the health and safety of the public and the protection of the environment.
- (2) Allow complete retrieval from stored waste for transfer to ultimate disposal.
- (3) Provide required waste cooling.
- (4) Ensure safe operation even under emergency conditions plus safe recovery from unplanned events.
- (5) Meet applicable local and Federal codes and regulations.
- (6) Provide containment for radioactive materials under normal and postulated abnormal occurrences and withstand all credible natural forces.

Regardless of the concept selected, the repository will be designed to receive and store all the high-level waste produced through the year 2000 and could be expanded incrementally to accommodate such wastes for an even longer period if desired.

Concern was expressed regarding the cost of interim RSSF storage. EPA* stated regarding the RSSF:

Our concern is that this construction cost, together with the cost of removing and transporting the wastes and decommissioning the interim facility, may be so expensive that this interim storage facility may, indeed, for economic reasons become a permanent storage facility.

The AEC does not agree that, for economic reasons, the RSSF would become a permanent storage facility. Studies are under way regarding the establishment of a fee schedule that would cover the costs of RSSF storage and provide a fund for later removal and transfer of the high level wastes from the RSSF to some ultimate disposal facility or site.

4.6.3.1.1 Site Considerations

The site for the repository has not been selected. A number of sites are being considered. For purposes of examining possible environmental impacts in this state-ment, the repository is assumed to be located within the Hanford reservation in the southeastern part of the State of Washington. Based on studies to date, the AEC concludes that construction and operation of the proposed repository would have no significant impacts on the environment.

Population in the surrounding area is sparse, averaging only about 47 persons/square mile within a radius of 30 miles. The population to the north, east, and west consists primarily of farmers and farming communities. Three small cities are located 25 to 30 miles southeast of the reservation and represent the major population concentrations in the area with a combined population of approximately 65,000. The predominant use of lands outside the reservation within 30 miles of the RSSF site is agricultural. The nearest farms are located about nine miles west of the reservation. There are no known archeological or historic sites within or near the Hanford reservation.

Basalts as much as 12,000 ft thick in total underlie the reservation. The sediments beneath the site are deposits laid down in Pliocene times and compacted to very high densities; therefore, settling of these materials due to seismic effects will be negligible.

An underlying aquifer is considered adequate to supply high capacity wells for emergency cooling water in case of temporary loss of the regular RSSF supply from

*Comment Letter 53, p. 25.

the river. Wells for this purpose would be included in the initial project. The RSSF could be located where the maximum probable flood as estimated by the Corps of Engineers would result in a river level no higher than 200 ft below the elevation of the RSSF site.

The area is subject to rather wide seasonal temperature variations. The average summer temperature is 73.7°F; the average winter temperature is 32.4°F, while the minimum and maximum recorded temperatures in the area were minus 27°F and 115°F, respectively. Precipitation averages approximately 7 in./year mostly occurring during the winter months. Northwest winds predominate. Over a 25-year period, the peak wind gust measured was 80 mph at the 50-ft level above the ground. The geographical area has a very low frequency of tornadoes; less than one every two years is expected within a radius of 100 miles.

The reservation is located in Zone 2 of the modified Seismic Risk Map of the Uniform Building Code (revised May 1970) and the Seismic Risk Map of the National Oceanographic and Atmospheric Administration (1969). Future earthquakes within 75 miles represent the principal seismic hazards. The design basis earthquake for the RSSF is based on analysis of earthquakes in the western United States associated with normal faulting and is postulated as magnitude 6.8 on the Richter scale with its focal point six miles deep on a postulated fault ten miles from the RSSF.

The natural vegetation in the vicinity of the reservation is dominated by desert shrubs. The most common mammals on the reservation are pocket mice, deer mice, jack rabbits, coyotes, and mule deer. Birds and reptiles are not abundant. Meadow-larks, horned larks, and loggerhead shrikes are the most conspicuous birds. The side-blotched lizard is the most common reptile. The gopher snake and Pacific rattlesnake are occasionally encountered. Because the proposed facilities would occupy such a small fraction of the total area of the reservation, there would be no detectable change in the ecological balance of the reservation.

4.6.3.1.2 RSSF Water Basin Concept

The water basin RSSF concept is one of three basic RSSF concepts under consideration by the AEC.¹³ The water basin concept has been used as the reference design for this Environmental Statement. Two alternative concepts, the sealed storage cask concept and the air-cooled vault concept, are described in subsequent Sections 4.6.3.1.3 and 4.6.3.1.4.

The water basin concept consists of three major elements: the waste receiving and handling facility, the storage facility (a series of water-filled concrete basins in which the waste canisters would be placed for cooling and long-term surveillance), and the heat rejection facilities (a series of forced-draft cooling towers associated with the basins, from which the waste heat would be dissipated to the atmosphere).¹² There would also be support facilities and services, e.g., an oil-fired steam plant. Figure 4.6-1 shows an artist's cutaway view and Figure 4.6-2 shows a conceptual layout of the concept. Modular construction of the actual storage area would be planned (Figure 4.6-3) so that the storage capacity would keep pace with the rate at which waste is expected to be delivered.

In the basic concept, bare waste canisters are used. A variation being considered calls for sealing each canister in another container, called an overpack, prior to storing in the water basins.

The facilities for receiving and handling the incoming shipping casks, removing the waste canisters from the casks, and monitoring and inspecting the waste canisters will be similar regardless of the approach selected for subsequent waste canister treatment and storage.

The receiving system would include facilities for receiving and unloading casks from trucks and railroad cars; removing waste canisters from shipping casks; inspecting, cleaning, and cooling canisters; and transferring canisters to the storage areas. Waste collection and processing systems plus building services and controls would be located in the receiving area to handle requirements for both that area and the storage area.

The receiving cell (Figure 4.6-4) would be heavily shielded and equipped with manipulators, crane, and viewing windows for remote operations. In this cell, canisters would be removed from the shipping cask and examined, decontaminated, and repackaged if necessary. The cell could also be used to handle other operational tasks requiring remote operating capability.

The inspected and cooled waste canisters would be transported from the receiving cell to the storage area in a water-filled shielded transfer cask, lowered into basins, and transferred from the cask to basin storage racks as shown in Figure 4.6-5.

4.6-19

WATER BASIN CONCEPT - CUTAWAY VIEW

Figure 4.6-1

4.6-20

WATER BASIN CONCEPT - FACILITY LAYOUT

Figure 4.6-2

4.6-21

WATER BASIN CONCEPT - SYSTEM EXPANSION

Figure 4.6-3

WATER BASIN CONCEPT - RECEIVING CELL

Figure 4.6-4

4.6-23

WATER BASIN CONCEPT -
CANISTER STORAGE BASIN

Figure 4.6-5

The basins would be stainless-steel-lined concrete cells 36 ft deep, each with a nominal capacity of 500 waste canisters. They would contain demineralized water to cool the waste and provide radiation shielding and a confinement barrier.

An insulated metal superstructure over the water basins would contain the bridge crane and service facilities. It would provide personnel working area, ventilation confinement, and protection from the weather.

The cells would be covered to allow free personnel movement in the working area and to provide temperature and humidity control in the building plus added contamination control in the event of a canister leak. Water depth of 13 ft over the waste canisters would provide adequate radiation shielding under normal conditions, but concrete cover blocks would be used to cover the cell in the event of a canister rupture and resulting increased radiation levels in the basin.

Radiation detection equipment would be installed in the cooling water recirculation line for each cell so that leaking canisters could be detected early and removed for repackaging.

The storage facility would be built in modules to meet increasing inventory requirements. The plan for expansion of the storage basins was illustrated previously in Figure 4.6-3. Approximately 10% spare basin capacity would be maintained at all times.

Heat would be removed by the constant circulation of the basin water in direct contact with the waste canisters; it would be transferred to the secondary cooling loop and rejected to the atmosphere from cooling towers situated just outside the storage cells. Separate coolant loops would prevent the release of radioactive materials to the environs through the cooling tower if waste leakage from a canister should contaminate the primary (basin water) cooling system. A diagram of the system is shown in Figure 4.6-6.

Each basin would be equipped with a pump and heat exchanger system to maintain basin water temperature below 120°F. These systems would be housed within the basin confinement structure and mounted on a support frame as a unit that could be replaced in 4 hr or less by means of the service crane. This arrangement would allow a safety margin of at least 10 hr before the basin water would begin to boil. Water as the cooling medium would provide the benefit of a large heat "sink" and would continue its primary function of cooling regardless of any

4.6-25

WATER BASIN CONCEPT - HEAT REMOVAL SYSTEM

Figure 4.6-6

temporary loss of circulation and secondary cooling. A portion of the coolant would circulate through a small filtration and demineralizer system. In the event of gross contamination, larger ion exchange units would be placed in the pool to remove contamination.

Each basin and cooling system would be isolated from the others to prevent cross-contamination in the event of waste canister or component failure within a storage system. An emergency water addition system would provide water for shielding and cooling under all credible occurrences.

Ventilation air would flow from occupied regions toward operating areas of increasing radioactivity. Air would be exhausted from the receiving cell and the waste canister handling area through three sets of high-efficiency particulate air (HEPA) filters in series.

Supply air to the storage buildings would be prefiltered to minimize dust and algae collection in the basin water and would be preconditioned as necessary for personnel work in the area. This storage area would normally be operated as an uncontaminated area, and the ventilation air would normally be discharged directly to the atmosphere through the ventilation stack. However, in the event of a contamination incident, monitoring instrumentation would sense the radioactivity and automatically divert the exhaust air flow through a set of filters, including at least one HEPA filter, before discharge.

During all operations, a portion of the exhaust air would be withdrawn from immediately above the pool area to prevent accumulation of hydrogen from radiolysis of the water.

A waste treatment system near the receiving cell would process liquid radioactive wastes from waste canister cooling and decontamination. Contaminated basin water might be routed to this same system for processing, or it might be treated by temporarily installing an ion exchange unit in the basin. The decontaminated water would be recycled. Other low-level liquid wastes would be cemented in 55-gal drums for burial.

4.6.3.1.3 RSSF Sealed Storage Cask Concept

The sealed storage cask RSSF concept is one of two alternatives to the water basin RSSF concept reference design being considered by the AEC. Several variations of the sealed storage cask concept have been considered. In one variation, the

canisters would be sealed in 2-in.-thick, low carbon steel casks that would be stored vertically outdoors on concrete pads inside 34-in.-thick concrete shields to provide additional gamma and neutron shielding. The casks would be spaced in a rectangular array on approximately 20-ft by 20-ft centers. Heat would be dissipated from the casks by natural convection air flow through the annulus between the cask and the neutron shield. With ambient air at 80°F, the expected maximum cask surface temperature would be about 360°F.

The three major elements of the concept are the waste receiving and handling facility, the storage cask welding and testing facility, and the outdoor waste cask storage areas.¹² Support facilities would include an oil-fired steam plant, offices, analytical laboratories, and maintenance facilities. To meet the forecast rate of increase in receiving waste shipments (more than 130 canisters/week by the start of calendar year 2010), expansion of facilities for handling and encapsulating would be required periodically. The receiving and cask welding stations could be expanded in modular components.

The cask receiving area would be a building to which the loaded railroad car would be brought and unloaded. It would be served by an overhead bridge crane, and space would be provided to set down the shipping cask for cooling and monitoring purposes.

The receiving cell would be a heavily shielded hot cell in which waste canisters would be handled remotely. It would be equipped with an overhead crane, manipulators, and lead glass viewing windows. It would contain a spray chamber decontamination station for cleaning waste canisters received in a contaminated condition. Beneath the receiving cell and connected through a cell port would be a storage cask cell equipped with a hydraulic lift table.

An empty storage unit consisting of a storage cask and its concrete neutron shield would be moved into the storage cask cell and raised by the hydraulic lift table to couple with a seal attached to the underside of the cell port. The clean waste canister would be lowered into the storage cask, and an air seal and a storage cask shield plug would be installed. The lift table would then be lowered and the storage unit moved to the cask welding and testing facility.

The cask welding and testing facility would be connected to the receiving cell building by the cask transfer corridor and would contain the cask welding and nondestructive testing stations. At the facility, the cask lid would be welded into the top of the cask. The initial module would have one welding station.

As modules were added to handle the increasing receipts, the total number of welding stations would rise to about 54 by the year 2010. The controls for welding and nondestructive testing would be used to check the welds made in the facility. Should flaws be detected, acoustical holography could be used to analyze the defects.

After completion of welding and weld testing, the storage cask would be moved to the storage unit assembly area for installation of a concrete neutron-shield cover to complete the assembly of a storage unit. Each cask would be placed vertically on an individual concrete pad as shown in Figure 4.6-7.

The storage area would be surfaced with crushed rock as required to prevent wind erosion and to provide access for the transport vehicle to all the storage cask locations. Area monitors and samplers would be provided to detect any radioactivity above the normal background level. A security fence would enclose the area to provide isolation and prevent inadvertent entry by personnel.

The (design) heat load in the typical canister assumed for this concept is 5 kWt. This heat would be dissipated directly to the atmosphere from the sealed cask containing the canister. The concrete neutron shield would be designed and positioned so as to allow free air flow around the cask to carry the heat away by natural convection (Figure 4.6-8). The maximum expected total heat output of the combined wastes would be 200 MWt in the year 2010.

Ventilation systems for the radioactive areas would be balanced and controlled so that air would flow from occupied regions toward operating areas of increasing radioactivity. Supply air would be exhausted from the receiving cell and the waste canister handling area through three HEPA filters in series, one in the receiving cell and two in the exhaust system.

A waste treatment system would be installed near the receiving cell and would process liquid radioactive wastes from canister decontamination. These liquid wastes could be cemented in 55-gal drums for burial.

4.6.3.1.4 RSSF Air-Cooled Vault Concept

In the air-cooled vault concept, each waste canister would be encapsulated in another 0.5-in.-thick carbon steel container called an overpack. These overpacks would then be placed in concrete vaults and cooled by natural draft convection as shown in Figure 4.6-9.

4.6-29

THICK WALL SEALED STORAGE CASK CONCEPT -
STORAGE AREA

Figure 4.6-7

SEALED STORAGE CASK CONCEPT -
VERTICAL STORAGE ASSEMBLY

Figure 4.6-8

AIR-COOLED VAULT CONCEPT

Figure 4.6-9

The three major elements of the concept are the waste receiving and handling facility, the overpack welding and testing facility, and the overpack-canister storage cells.¹² Expansion of facilities for handling and encapsulating (overpack) would be required periodically. The receiving and welding stations could be expanded in modular components.

The cask-receiving building and cask-receiving cell would be essentially identical to facilities required for the sealed cask storage concept. Similar support facilities would also be required.

Equipment for sealing the overpacks by welding and for non-destructive testing would be provided in the welding and testing facility. The controls for both welding and non-destructive testing would be located in a shielded operating gallery.

Ventilation systems for the receiving and handling facility and the welding and testing facility would be balanced and controlled so that air would flow from occupied regions toward operating areas of increasing radioactivity. Potentially contaminated exhaust air would be discharged through multiple HEPA filters in series.

A waste treatment system would be installed near the receiving cell and would process liquid radioactive wastes from canister decontamination. These liquid wastes would be cemented in 55-gal drums for burial.

The storage facility would be built in modules to meet increasing inventory requirements. Each storage cell (Figure 4.6-9) would have the capacity for 500 storage units suspended vertically in sleeves. The sealed overpack-canister would be placed in a transport shield for movement to a storage vault. A concrete cap in the cell ceiling would be removed while the overpack-canister was inserted, after which the cap would be replaced (Figure 4.6-10). The canister would hang suspended from the cell ceiling by means of a shoulder in the overpack.

Cooling air would enter through bottom inlets and a distribution plenum and pass upward through annuli formed by the storage units and sleeves. The air would be heated to about 210°F while maintaining a storage unit surface temperature below 500°F. Heat would be removed from the storage units by natural convection plus radiation to the sleeves. This concept would utilize the decay heat of the waste and the engineered design of the vault and the short exhaust port to induce air

AIR-COOLED VAULT CONCEPT -
STORAGE UNIT IN CELL

Figure 4.6-10

flow by natural draft to maintain permissible temperatures. The heated air would be discharged through the short exhaust port to the atmosphere without filtration.

The structure would provide all requirements for biological shielding and protection from natural phenomena. The high integrity overpack would provide complete confinement and containment of the waste. Overpack corrosion rates indicated for this environment are less than 0.002 in./year.

4.6.3.1.5 RSSF Environmental Effects

Estimates of environmental effects are based on the assumed use of the Hanford reservation as a reference site. The environmental effects of constructing the RSSF would be similar to but less extensive than those described for the reactor plant (Section 4.2.4), and similar steps would be taken to minimize impacts. Specific impacts can be summarized as follows:

- (1) Construction would produce noise and dust. The soil would be stabilized during and after construction to minimize effects of winds.
- (2) Construction and operating manpower would be primarily local forces; fluctuations in local populations would be temporary and small.
- (3) For the water basin and air-cooled vault concepts, approximately 100 to 150 acres of land would be used for the facilities for up to 100 years, with moderate probability that some part of it might be permanently restricted. A quarter-mile wide peripheral "exclusion area" around the water basin and air-cooled concepts would be restricted by fencing for the life of the RSSF. This land would be available for other possible use following RSSF decommissioning. For the sealed storage cask concept, land requirements could range from 1100 to 2300 acres for up to 100 years with moderate probability that less than 10 acres might be permanently restricted.

The environmental effects of operating the RSSF would be as follows:

- (1) For the water basin concept, water would be drawn from the Columbia River and consumed at the maximum rate of 2800 gpm at peak demand in the year 2010. Approximately 1500 gpm of water would be discharged to the atmosphere through cooling towers by the year 2010, and 1000 gpm would be discharged to the ground surface as blowdown from the cooling towers. The balance of the water would be used for cleaning shipping casks and waste containers and for various sanitary and operating uses. Consumption

would drop to 600 to 800 gpm by the year 2100. There would be no direct return of water to the Columbia River.

- (2) For the sealed storage cask or air-cooled vault concepts, about 150 gpm would be drawn from the river and used during the operating life of the RSSF for cleaning shipping cars and waste containers and for various sanitary and industrial uses.

Any contaminated water would be processed for removal of radioactive material. Uncontaminated water would be discharged to surface ponds or to a septic tank and tile field from which, except for that portion that evaporates, it would eventually return to the river as the groundwater. There would be no direct return of water to the river.

- (3) For the water basin concept, an average of approximately 10,000 kW of electric power would be used at peak demand in the year 2010. For the sealed storage cask concept, electrical power usage would be about 3000 kW in the year 2010, and for the air-cooled vault concept, electrical power usage would be about 2100 kW in the year 2010.
- (4) Approximately 200 MWt of radionuclide decay heat would be dissipated to the atmosphere at peak basin waste load but would drop to 30 MWt by the year 2100.
- (5) In the case of the water basin concept, heat dissipation through open cooling towers could cause a minor increase in the occurrence of fog and ice during winter months.
- (6) Occasional radioactive releases to the cell air might come from contaminated casks, the failure of a canister, or the cleanup operations in a contaminated area. The exhaust air from areas where these releases could occur, i.e., the receiving and handling areas and welding and testing area, would be routinely passed through three HEPA filters in series, one in the cell and two in the exhaust system. Release data from operating nuclear facilities having radioactivity containment capability similar to the RSSF indicates that normal radioactive material releases would be less than 1 mg annually, or approximately 1 mCi for solidified LMFBR waste.¹⁴ Radiation exposure estimates for this release were calculated using procedures¹⁵⁻²⁰ similar to those described in Appendix II.I but were specific for the Hanford site. For a member of the general public at the reservation boundary, who would be at least five miles from the RSSF site, total-body radiation doses from external exposure would be less than 0.001 millirem/year. The dose to the skin would be about the same. The 50-year bone dose commitment from inhaled particulates for a member of

the general public at the reservation boundary would be 0.008 millirem for each year of exposure. The population dose (total body) from external exposure within a 50-mile radius would be 8×10^{-8} man-rem/year. The 50-year bone-dose commitment to the population for inhalation would be 0.016 man-rem for each year of exposure. Radiation exposure from food chain pathways was also investigated and found to be very small. For example, an adult eating 200 kg/year of potatoes grown at the reservation boundary would receive a 50-year bone dose commitment of only 0.003 millirem. A child drinking one liter of milk per day from cows pastured nine months per year at the reservation boundary would receive a 50-year bone dose commitment of only 0.003 millirem.

- (7) With the sealed storage cask concept, radiation dose rates (primarily from neutrons) ranging up to 20 rem/hr, depending on the type and age of waste, would be encountered in the annulus between cask and concrete shield. Radiation fields and the high temperature (probably up to 360°F in the early years) would probably be fatal to small mammals intruding into the annulus of the vertical casks through the air flow ports at the base of the shield. Proper fencing would exclude large wide-ranging animals from the cask storage area. Although the heat would probably be enough to deter entry or close approach by most of the local fauna, the potential for adverse environmental impact is recognized. Minor loss of life would occur among small fauna, but despite the damage to individuals, the losses would be very small relative to the large regional fauna population. Flora would be affected only by the site-clearing operation.
- (8) In the case of the sealed storage cask concepts, exposure to direct radiation from the surface storage area would be a concern. A fenced exclusion zone around the storage area would reduce radiation levels to less than 1 millirem/hr at the exclusion zone fence, and no direct radiation from the casks would reach the public at the reservation boundary at least five miles distant. There would be a small amount of neutron-induced radiation formed in the soil under the storage casks. Dust carried into and out of the storage area by wind could be inhaled by man or deposited on vegetation and ingested by man and other fauna. Based on a worst potential exposure case of 100 years of radiation and on radioactive dust suspended in air at a constant concentration of $100 \mu\text{g}/\text{m}^3$ and inhaled year round by an individual in the uncontrolled zone, the individual's radiation dose rate would be less than 0.1 millirem/year either to the lungs or gastrointestinal tract.

A radiological monitoring program similar to that described for the reactor plant (Section 4.2.6) will be carried out at the RSSF to check routinely for possible radioactive releases. Samples would be routinely collected for evidence of radioactivity that would indicate a containment failure in the RSSF. In the case of the sealed cask concept, additional emphasis would be placed on dosimeters to monitor direct radiation in the vicinity of the RSSF.

Information on the safety analysis of the RSSF water basin concept is provided to aid in evaluating the safety of the concept. A preliminary safety evaluation of the RSSF water basin concept was made during conceptual design to evaluate engineered features that prevent or mitigate potential unacceptable safety hazards.²¹ The safety considerations were primarily those having potential for loss of radioactive material confinement and/or having capability of exposing either operating personnel or the general public to a radiation dose in excess of acceptable limits. Conditions that could result in radioactive material discharges were categorized as normal, abnormal, and upper limit accidents.

Normal Operations

Normal operations involve releases of very small quantities of radioactive materials under routine operating conditions. Releases of radioactive material during normal operations were previously discussed.

Abnormal Operations

Abnormal operations are those that might result from equipment failures, operator errors, or unplanned process variations that may occur sometime during the plant operating life. This category includes not only expected events (e.g., mechanical equipment failure) but also events that are not expected to occur during the plant operating life, but which have sufficient probability for occurrence to warrant design features to mitigate their potential consequences. A summary of the postulated abnormal operating occurrences is presented in Table 4.6-4. For each occurrence, the potential effects, mechanisms, mitigating facility design features, and recovery procedures are shown.

Upper Limit Accidents

The evaluation included analyses of highly improbable "upper limit" accidents that are not expected to occur during the plant operation life; however, design features to further reduce the probability and possible magnitude of waste confinement loss resulting from these events are justified by their potential consequences. The accidents included in this category are as follows:

Table 4.6-4
SUMMARY OF ABNORMAL OPERATIONS ANALYSIS FOR THE
WATER BASIN CONCEPT RSSF

System	Safety Consideration	Mechanism	Probable Effect ¹	Detection	Safeguard	Recovery Procedure
Receiving	Receipt of externally contaminated shipping cask.	Cask leakage. Insufficient cleaning.	Slight atmospheric discharge.	Radiation survey. Smear test.	Leak-resistant shipping cask. Ventilation exhaust filtration.	Decontaminate cask. Decontaminate receiving area.
	Receipt of contaminated or failed canisters.	Canister leakage. Transportation accident. Insufficient cleaning.	Slight atmospheric discharge.	Cask air sample. Canister smear test. Visual. Exhaust air radiation alarm.	Leak-resistant canister. Ventilation exhaust filtration.	Decontaminate canisters. Repackage canister (if failed). Decontaminate receiving cell. Decontaminate cask.
	Canister failure in receiving cell. ²	Drop canister. Other canister impact.	Slight atmospheric discharge.	Known cause. Visual. Exhaust air radiation alarm.	Impact-resistant canister. Ventilation exhaust filtration. Reliable canister lifting mechanism.	Repackage canister. Decontaminate receiving cell.
Transfer	Long holding period in transfer cask.	Equipment failure. Operator error. Earthquake or tornado.	Increased canister temperature.	Known cause.	Reliable transfer equipment. Water heat sink. Adequate heat transfer from dry transfer cask.	Periodically replenish transfer cask water. Complete transfer when possible.
Storage	Canister failure in storage basin.	Corrosion. Impact.	Contamination of basin water. Increased risk of slight discharge.	Known cause. Basin water radiation alarm.	Corrosion-resistant canisters. Water chemistry control. Impact-resistant canisters. Basin water as confinement barrier. Ventilation exhaust filtration. Ion exchange facilities.	Install shielding blocks as needed. Locate failed canister. Decontaminate basin water. Repackage failed canister. Complete basin decontamination.

¹The consequences of abnormal operations do not exceed the radiation dose and effluent radionuclide concentration guide limits presented in CFR 10, Part 20.

²The improbable worst-case consequences of canister failure in the receiving cell cause this event to be considered as the mechanism for the maximum credible accident.

Table 4.6-4 (continued)

System	Safety Consideration	Mechanism	Probable Effect ¹	Detection	Safeguard	Recovery Procedure
	Failure of storage basin liner.	Corrosion. Impact of massive object.	Expense of repair. Continued confinement.	Known cause. Liner sump leak detection alarm.	Structural integrity of basin and liner. Corrosion-resistant liner. Water chemistry control. Impact resistant basin and liner. Liner leak collection sump.	Pump out liner leakage collection sump. Transfer canisters to spare basin. Empty basin. Repair liner.
	Dropping transfer cask into cask well.	Equipment failure. Operator error.	Risk of basin liner failure.	Known cause. Liner sump leak detection alarm.	Impact absorber on cask well floor. Impact-resistant basin and liner. Reliable crane and cask ball.	Retrieve dropped cask. Inspect canister for damage. Determine possible liner damage. Replace impact absorber.
	Dropping canister into storage basin.	Equipment failure. Operator error.	Risk of canister failure.	Visual. Basin water radiation alarm.	Reliable crane and canister connector. Impact-resistant canisters and basin. Physical lift-limit on crane. Basin water as confinement barrier.	Retrieve dropped canister. Inspect suspect canisters for damage.
Heat Removal	Failure of primary cooling loop.	Failed heat exchanger. Failed circulation pump. Piping system failure.	Increasing waste temperature.	Visual. Observed leaks. Basin high temperature alarm. Other miscellaneous instrumentation.	Readily replaceable components. Reliable equipment. Water chemistry control. Emergency electrical power. Routine contact maintenance. Basin water as heat sink. Leakage collection sump.	Replace failed component or system.
	Failure of secondary cooling loop.	Failed circulation pump. Piping system failure.	Risk of increasing waste temperature.	Visual. Observed leaks. Basin high temperature alarm.	Once-through secondary cooling capability. Readily replaceable components. Adequate cooling tower blowdown. Reliable equipment. Emergency electrical power. Routine contact maintenance. Basin water as heat sink.	Utilize once-through secondary cooling if needed. Replace failed component or system.

Maximum credible accident
Dropping concrete cover block into storage basin
Contamination of secondary cooling water
Failure of ventilation exhaust air filtration system
Fire
Explosion
Tornado
Earthquake
Sabotage
Prolonged loss of cooling
Criticality
Massive missile impact
Flood

Table 4.6-5 is a summary of the upper limit accidents with the intended means of assuring continued waste confinement following each event briefly itemized.

The postulated maximum credible accident was selected after evaluation of the abnormal and upper limit accident conditions. These accident conditions can be categorized with regard to mechanism of potential waste confinement loss as follows:

- (1) Inadequate removal of radioactive decay heat (meltdown),
- (2) Gross contamination of storage basin water,
- (3) Gross contamination of dry waste handling facilities.

An analysis of sequence of events and consequences led to identification of the third mechanism as the maximum credible accident.

The maximum credible accident in the RSSF is postulated to involve a major rupture of a waste canister dropped during handling or inspection and releasing the assumed dispersible solids contents from the canister. The accident is assumed to occur in the waste receiving cell with solidified high-level LMFBR fuel waste. On release from the canister, a small fraction of the waste would be suspended in the air in the cell (volume 400 m^3). On the basis of six air changes per hour, the contaminated cell air would be exhausted in 10 min following the accident. The air would pass through the series of receiving cell and exhaust system HEPA filters and be discharged to the atmosphere. Under the postulated conditions, 0.02 mg of waste per cubic meter of air would escape through the HEPA filters, thus resulting in a total release of 8 mg, or approximately 10 mCi.

Table 4.6-5
SUMMARY OF UPPER LIMIT ACCIDENT ANALYSIS FOR THE
WATER BASIN CONCEPT RSSF

Upper Limit Accident	Means of Providing Continued Waste Confinement
Dropping concrete cover block into storage basin	Structural integrity of crane and cover block lifting bail. Cover block support design reduces possibility of cover block falling into basin. Basin water as confinement barrier. Cover block used infrequently.
Contamination of secondary cooling water	Basin contamination normally slight. Heat exchanger acts as confinement barrier. Secondary water pressure exceeds primary water pressure at heat exchanger.
Failure of ventilation exhaust air filtration system	Tornado dampers and blow-out panels. Emergency fire protection systems. At least two HEPA filters installed in series. Routine filter testing and replacement.
Fire	Noncombustible materials of construction. Minimized use of combustible supplies in contaminated areas. Emergency fire protection system.
Explosion	Absence of explosives in contaminated areas. Venting of radiolytic hydrogen gas. Safe operational control of ion exchange systems.
Tornado	Waste receiving building and storage basins designed to withstand tornado forces and missiles. Basin water as confinement barrier in case of vertically directed tornado missiles. Tornado dampers and blow-out panels protect ventilation air exhaust filtration systems.
Earthquake	Systems essential to waste confinement designed to withstand the maximum anticipated earthquake.
Sabotage	Engineered design features and administrative controls for sabotage prevention.
Prolonged loss of cooling	Emergency water source. Standby electrical power system. Water as heat sink provides time to install new temporary water supply system.
Criticality	No credible mechanism for criticality has been postulated. Waste acceptance criteria prevent receipt of waste that could become critical under any circumstances.
Massive missile impact	Low probability of occurrence; event considered incredible. Site location would not be in commercial aircraft flight path.
Flood	Site location will be above the maximum probable flood level.

Radiation exposure estimates for this accidental release were calculated using procedures described in refs. 16, 17, and 18. An individual 1000 m from the RSSF would receive a total body dose of 4.3×10^{-5} millirem and a bone dose (50-year commitment) of 85 millirems. A member of the general public, who would be at least five miles away, would receive a total body dose of 4.5×10^{-6} millirem and a bone dose (50-year commitment) of 3.6 millirems.

An accidental event with the potential for serious environmental impact is the meltdown of waste and canister. In all of the concepts, the RSSF would be designed to maintain continuity of waste cooling capability and electric power during any credible incident. Meltdown would be prevented by a combination of engineered features, redundant power and water supply systems, and appropriate safety and administrative control procedures.

In the case of the water basin concept, temporary loss of basin water cooling capability could result from many different system failure conditions. The systems and equipment whose failure could cause loss of cooling are the electrical power system, the water supply system, the primary cooling loop system, and the secondary cooling loop system.

The prolonged loss of cooling capability could result only from an incredible loss of primary and emergency water sources. For example, at the reference Hanford site, prolonged loss of cooling would require sudden disappearance of the Columbia River and the groundwater supply beneath the site. There is adequate time for equipment failures, such as those listed, to be corrected or otherwise compensated for before increasing waste temperatures could adversely affect confinement capability.

The sequence of events that would occur from extended loss of cooling are summarized as follows:

<u>Event</u>	<u>Time After Cooling Loss</u>
Basin water begins to boil	16 hr
Basin water completely evaporated	6-7 days
Waste begins to melt, canisters and basin liners melt, and concrete calcines and/or melts	1 to 2 weeks

The plant design includes features that minimize the frequency of temporary loss of cooling, as pointed out in other accident analyses in this section. Moreover, a fully redundant emergency water supply system is included in the design which will provide water in case the primary water supply becomes inoperable. Each basin is equipped with its own primary water circulation system, so that only one basin will be affected by an individual equipment failure in these systems. The water supply capability will be adequate for operation of one basin with once-through secondary cooling if equipment failure prevents heat dissipation to the atmosphere through a cooling tower. Each basin is equipped with a basin water high temperature alarm to alert personnel of inadequate heat removal. The design basis limits the normal maximum basin water temperature to 120°F; since the water temperature would not exceed 150°F until more than 4 hr after loss of cooling, most maintenance work on an individual basin will be performed without resorting to once-through secondary cooling. Use of quick-disconnect pipe and electrical fittings will facilitate such maintenance. Because no piping connections are attached to the basin lower than 1 ft below the normal water level, basin drainage due to piping leaks cannot occur.

Recovery procedures for an accident in which cooling capability is lost will begin with determination of the causative equipment failure and an estimate of the time required to repair or replace the failed equipment. If more than 4 hr is required for secondary cooling loop repairs, preparations will be made to operate the basin with once-through secondary cooling while repairs are made. The basin can be operated with boiling water, with the water added as necessary to replace evaporative losses, even if several basins are affected. Even if both the primary and emergency water supply systems are inoperative, time is available to install whatever temporary system is required to resume the water supply before the basin water has evaporated to a hazardous level.

4.6.3.2 Facilities for Other-Than High-Level Waste

Over 99.9% of all nonvolatile radioactive isotopes discarded from the LMFBR fuel cycle will be channeled to the high-level waste category. The remaining radioactive wastes are classified as other-than-high-level waste and itemized with their estimated activity in Table 4.6-6. For purposes of discussion, they can be divided into the following categories:

Table 4.6-6

ESTIMATED ACTIVITY OF OTHER-THAN-HIGH-LEVEL WASTES

<u>Fabrication Wastes</u>	<u>Radioactivity (Ci/ft³)</u>
Alpha	28
Beta-gamma	0.01
<u>Reactor Wastes</u>	
Beta-Gamma	1
Tritium	3100
<u>Reprocessing Plant</u>	
Alpha	28
Beta-Gamma	0.01
Alpha-beta-gamma	7
Cladding Hulls	1×10^4
Iodine	9

4.6.3.2.1 Beta-Gamma Waste

These wastes will be primarily miscellaneous solid materials contaminated with β and γ emitting materials but with very low level concentrations of plutonium and other long-lived alpha emitters and will include materials such as laboratory waste and residues, contaminated tools, plastic bags, paper towels, and protective clothing. Most of these wastes will be compacted and packaged in 55-gal drums and sealed for disposal in a licensed burial facility. Some of these wastes will require shielded containers. Discarded equipment that has become contaminated through use will be packaged for burial on a custom basis.

Other specific types of beta-gamma wastes include:

- (1) Tritium. Approximately 95% of the tritium produced during the reactor operations will diffuse through the stainless steel cladding of the fuel and will be converted to sodium tritide (Na^3H). It will be separated routinely from the sodium coolant and converted to a stable solid such as $\text{Ca}(\text{OH})_2$ and sealed in suitable containers, probably 55-gal drums, for disposal in a licensed burial facility. This form of storage was selected because of the known properties of both $\text{Ca}(\text{OH})_2$ and tritium, but the stability of tritiated calcium hydroxide has not yet been tested. The stability will be confirmed by pilot tests before any large-scale storage is undertaken.

Ca(OH)_2 is a stable and quite insoluble compound. It is stable up to 1076°F (580°C) where it decomposes into CaO and H_2O . It is only slightly soluble in water, 0.185 g per 100 g H_2O at 0°C and 0.077 g per 100 g H_2O at 100°C . The proposed concentration of tritium would be only 2 g (19,000 Ci) in 12 ft^3 of Ca(OH)_2 . This quantity of tritium would generate only 0.7 W of heat, and there is no conceivable way that decomposition temperatures could be reached. Because of the low solubility of Ca(OH)_2 , escape of any significant amount of tritium, even if exposed to seepage, to groundwater in a burial ground, is highly improbable. Furthermore, since the half-life of tritium is about 12 years, it will decay relatively quickly (only 6% of the initial activity would remain after 50 years).

- (2) Iodine. A small quantity of the long-lived iodine isotope, I-129 with a 1.6×10^7 year half-life, will be recovered during fuel reprocessing. This isotope emits only low energy beta and gamma radiation but is of concern because of its rapid uptake in the food chain and its long half-life. It will be converted to a stable form and packaged for burial or possible storage in a repository.
- (3) Sodium. Radioactive sodium coolant purged from operating reactors or discarded from reactors that have reached the end of their useful life and have to be shut down will be converted to inert solid compounds suitable for burial, such as sodium carbonate or a silicate glass formed by calcining a mixture of sodium hydroxide, lime, and silica. This material will be sealed in 55-gal drums and sent to a licensed burial facility.

Site Considerations for Land Burial Facility

At the present time, there are six commercial land burial facilities authorized to receive and bury other-than-high-level wastes. Five of the sites are on land owned by states, and the other site is on land owned by the Federal Government and leased to the state. Licensees who bury waste received from others must do so on land owned by either a state or Federal Government. The purpose of this requirement is to assure long-term control of the site in the event of default or abandonment of the site by the commercial operator. After radioactive wastes are buried at the site, the land will not be used for any other purpose. The states have the obligation for perpetual care and maintenance of land burial sites to assure their dedication to waste burial and to assure their restriction from

other uses. Authorization to operate a commercial land burial facility is based on an analysis of the nature and location of potentially affected facilities; of the site topographical, geographical, meteorological, and hydrological characteristics; and of groundwater and surface water use in the general area which must demonstrate that buried radioactive waste will not migrate from the site. To date, there have been no reports of migration of radioactive material from commercial burial sites. In the unlikely event that there would be such a finding, several courses of action could be taken, including: (1) a halt to burial operations, (2) removal of the radioactive material from the burial area in which it originated, (3) grouting of the site from which the radioactive material originated, or (4) other such procedures that might be necessary, depending on the extent of migration of radioactive material from the site.

Existing commercial land burial facilities are located in the States of Washington, Nevada, Illinois, Kentucky, South Carolina, and New York. Currently, about 1.5 million ft³ of packaged radioactive waste materials are buried per year. The annual volume has been estimated to increase to about 2 million ft³ by 1975 and 4 million ft³ by 1980.

The commercial land burial sites are located in rural, sparsely settled areas or in virtually uninhabited areas such as the Armagosa Desert in Nevada. The sites themselves are generally isolated so that public access to them is limited. The areas in which wastes are buried are enclosed by fences, and access to the sites is carefully controlled by the site operator.

The suitability of a site for burial of radioactive wastes depends on the geological and hydrological characteristics of a site and its ability to retain the radioactive material so that there will be no migration of the radioactive material from the burial site. No credit is taken for containment provided by the packages once they are buried. The purpose of the packages is to provide ease of handling, to minimize personnel exposure, and to prevent loss of the radioactive material to the surroundings during transport and prior to burial.

Description of Model Land Burial Facility

A model land burial facility²² for other-than-high-level wastes consists of 100 acres of land located in a rural, sparsely settled area. The site location is near a highway so that there is reasonable access to the site. The site characteristics are such that the groundwater is well below the bottom of the deepest trench.

The site hydrology is such that flooding of trenches and leaching of buried radioactive material are minimal. The soil provides good ion exchange characteristics to minimize percolation of any radioactive material that may be leached from the solid waste to the groundwater. There is no nearby use of groundwater or well water downstream from the site.

Buildings on the site provide space for offices, a laboratory, temporary storage of packaged radioactive wastes, and other waste handling as may be necessary, such as solidification of low activity liquids. The storage space is also used for storage of vehicles, earth moving equipment, forklifts, and other equipment necessary for preparation of trenches and handling of packages containing radioactive wastes.

The portion of the site in actual use is fenced so that access to trenches and buildings is prevented. The fence also serves to minimize entry by area wildlife. The operator of a site has a program for environmental monitoring that includes sampling of air, water, and vegetation to determine migration, if any, of radioactive material from the actual location of burial. Records are maintained of the monitoring results.

An average burial trench is about 300 ft long, 30 ft wide, and 20 ft deep and has a volume of 180,000 ft³. The volume is not completely utilized since there are voids between packages and between packages and earth fill. About 100,000 ft³ of waste can be stored in a trench. Assuming 50% usage of available land for trenches, a burial site will provide for disposal of approximately 5 ft³ of waste per square foot of burial site.

The prime operations at a licensed commercial land burial facility are the receipt, temporary storage, and burial in trenches of packaged radioactive wastes. The packages that are shipped to the burial facilities must meet the requirements of the Department of Transportation and the AEC for transportation of radioactive material.

The packages are normally buried as received with no reprocessing or repackaging of package contents. 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 the particular materials involved. In those instances, the primary package is removed from the reusable overpack before burial.

Trenches that are completed are cared for to minimize erosion and are marked to specify the contents. Packages in partially filled trenches are frequently covered over with earth to minimize access to the waste, to provide shielding during placement of packages in the trench, and to minimize entry of water into the trench.

The model facility keeps temporary storage of packages to a minimum. Packages are placed in a trench and covered with earth as soon after receipt as possible.

Environmental Effects of Land Burial Facility Construction

The environmental effects of the waste burial facility construction are (1) the temporary dust and noise during construction of the offices and laboratory, (2) the periodic excavation and filling of trenches, and (3) the presence of the exclusion fence around the area.

Environmental Effects of Land Burial Facility Operation

The normal operations in the burial of other-than-high-level radioactive wastes in solid form are not expected to result in any effluent releases to the offsite environment. Due to the very low concentrations of radioactive materials in the other-than-high-level wastes that result from nuclear fuel cycle operations, there is essentially no heat release to the environment from these wastes.

The volume of other-than-high-level packaged radioactive waste generated during fuel cycle operations is variable. However, the total quantity of radioactive material contained in such wastes will be small. The nuclear fuel cycle activities required to produce the fuel required annually for a reference LMFBR will generate relatively few curies of radioactive material in other-than-high-level wastes.

The total waste volume of low level waste for burial from a 1000-MWe LMFBR is estimated to be on the order of 25,000 ft³/year. A total area of about 5000 sq ft (approximately 0.1 acre) of land surface will be required each year per 1000 MWe. If storage of alpha wastes in repositories is required, land burial would be reduced to about 10,000 ft³/year per 1000-MWe LMFBR.

Environmental Effects of Accidents

There have been no accidents at burial grounds under Federal or state jurisdictions that have resulted in offsite environmental effects. Some minor accidents have occurred involving materials not utilized in any fuel cycle.

Accidental opening of a waste package prior to burial with release of the package contents might occur at a commercial land burial facility. Normally, the scattered waste materials would be confined to the fenced facility and the impact on the environment would be negligible. Even if as much as one-third of the waste contained in a 55-gal drum were released to the unrestricted area, the effect on the environment would be insignificant. The volume of a 55-gal drum is approximately 7.4 ft^3 . If a waste concentration of 0.001 Ci/ft^3 is assumed, the total amount of radioactivity involved would be about 2.5 mCi. If the waste were uniformly dispersed over an area of 1 acre, the concentration of the radioactive material would be about $0.06 \text{ } \mu\text{Ci/sq ft}$ and the dose would be less than 0.003 millirem/hr total body dose for a person standing in the center of the area.* Dispersion over a greater area would result in a proportionally lower concentration. Much of the dispersed material would be in the vicinity of the site, which is normally uninhabited and unused. Recovery of the bulk of the scattered waste could easily be accomplished. Radiation surveys could be made and the area decontaminated.

Monitoring

Monitoring wells are maintained at each burial site and are sampled periodically to determine if there has been any migration of radioactive material. In addition, air and vegetation samples are taken around the site and analyzed for radioactivity. Should any sample reveal an increase in the concentration of radioactive material in an aquifer above that measured prior to commencement of burial operations and if the radioactive material is determined to have originated in the burial ground, its significance must be analyzed, corrective actions developed, and the entire matter reported. Corrective actions are subject to approval by the AEC or an agreement state, and additional corrective measures would be required as necessary to protect public health and safety.

4.6.3.2.2 Plutonium Bearing or Alpha Waste

These wastes will be solid materials and discarded equipment containing low levels of plutonium or other long-lived alpha-emitting radioactive material. These materials will originate primarily from the plutonium fuel-fabrication operations and the plutonium processing sections of the LMFBR fuel reprocessing facility. Because of the high toxicity and very long half-lives of these materials, future regulations are expected to require that these materials be stored in repositories.

*Assuming (worst case) that the radioactive material is entirely Cs-137.

A specific type of potential alpha wastes is the fuel cladding hulls. These materials will be compressed and sealed in disposable steel containers and under present regulations may be sent to a licensed burial facility. Because of possible plutonium residues in these hulls, future regulations may require that these materials be sent to an alpha waste repository.

Alpha Waste Storage Facility

A preliminary concept for an alpha waste storage facility or repository is described in this section. The description is based largely on a facility description included in a study²³ of the management of hazardous wastes prepared for the Environmental Protection Agency. The necessary development work required for this conceptual facility could be completed by the time of commercial application of the LMFBR technology in the mid-1980's. A plan view of the conceptual facility is shown in Figure 4.6-11.

The facility will include provisions for (1) receiving and sorting the incoming waste shipments, (2) decontamination of transport equipment, waste containers, and work areas, (3) volume reduction and encapsulation of the waste, and (4) a high integrity interim storage warehouse. The facility would be designed to hold the stored alpha wastes for at least 100 years. However, the stored alpha wastes would probably be moved to an ultimate disposal site, as in the case of high level waste, within a much shorter time period. The design criteria would be the same as for the RSSF, specifically:

- (1) Confine radioactivity and assure the health and safety of the public and the protection of the environment.
- (2) Allow complete retrieval from stored waste for transfer to ultimate disposal.
- (3) Provide required waste cooling (probably negligible).
- (4) Ensure safe operation even under emergency conditions plus safe recovery from unplanned events.
- (5) Meet applicable local and Federal codes and regulations.
- (6) Provide containment for radioactive materials under normal and postulated abnormal occurrences and withstand all credible natural forces.

Figure 4.6-11

Site Considerations for the Alpha Waste Storage Facility

An isolated site with characteristics similar to the Hanford site described for the RSSF would be desirable to minimize population exposure in the event of any accidental release of activity. Access by both rail and highway would be required.

A detailed survey of the geology of the site with regard to groundwater flow and aquifer characterization, structural and long-term stability characteristics of the underlying strata, tectonic stability, and frequency of earthquakes would be required before the plant is sited.

Prevailing meteorological conditions (wind velocity and direction versus time, air "dilution" factors, and precipitation) and background radioactivity would be well established before site selection and monitored during site operation. Because of the long storage time anticipated, population trends would be anticipated, and the existence of the storage facility should be considered in long-range local and regional plans. Arid regions of low population density with deep water tables would be preferred but are not essential. An exclusion distance of at least one mile around the activities within the site, or especially from the ventilation stack location, would probably provide sufficient protection for the public.

Description of the Alpha Waste Storage Facility

Design features for improbable but possible spills and accidents in the receiving area would be provided. Unloading would occur within an enclosed area with well-controlled ventilation to prevent direct release of radioactive material to the environment which might result from surface contamination of containers or rupture of containers during handling. Routine radioactive decontamination procedures would be followed for isolation and cleanup of spills. High radiation levels in wastes would be contained by special shielding. All other material with low radiation doses would be treated in hoods or glove boxes in the same way plutonium is handled in nuclear fuel plants. Direct monitoring of personnel for radioactive contamination and indirect monitoring of radiation doses would be routinely conducted.

Because of the nature of the alpha wastes, no significant heat generation problem is anticipated.

Alpha wastes would be transported to the receiving facility by rail cars and trucks. The receiving facility would be fully enclosed and sealed to ensure that any spill of radioactive material during transfer operations from the common carrier to the

facility is contained within the facility and does not enter the environment. The receiving area within the handling and transfer area would be designed for easy decontamination (e.g., with nonporous surfaces and a minimum of crevices and other obstructions). As with the rest of the treatment facility, ventilation air would flow from this area to the next higher potential contamination area. All outside access to the receiving facility would be through airlocks (separated from the outside by sealed doors) to prevent radioactive contamination from moving within or outside of the facility.

The receiving facility would be capable of handling all shipping canisters approved by the Department of Transportation for solid radioactive wastes. These facilities would be capable of removing wastes from bulk transfer canisters, decontaminating the canisters, and returning them to the common carrier.

When a shipment of alpha wastes is received at the disposal site, the records of its contents would be checked, the external surfaces checked for contamination, and the radiation levels measured. The disposition of the wastes through the site would be determined from this information.

All materials would be handled within secondary enclosures such as hoods, glove boxes, plastic-covered pipes, or hot cells. Materials with high radiation fields would be handled semi-remotely behind concrete shielding using shielded viewing windows, remote manipulators, and cranes. Other materials could be handled more directly without shielding or with portable shielding on the transport devices.

Transport of heavy packages or packages with high radiation dose rates would be done by remotely controlled cranes, including the final placing in the storage warehouse. Other materials would be handled by forklift trucks, by cranes, and by other conventional techniques.

Nearly all of the alpha wastes will be reduced to a noncombustible form by incineration facilities at their source, that is, at the fabrication or reprocessing plants. However, small quantities of combustible alpha wastes would be generated at the storage facilities, and small quantities of very low level combustible alpha wastes may be shipped to the storage facility. Thus, an air incineration facility would be needed at the storage facility.

The wastes would be segregated or sorted into combustible and noncombustible fractions before processing.

Most ferrous materials can be extracted from the solid waste stream by magnetic separation. Some success has been achieved with air classifiers in the separation of paper, plastics, and other combustibles from heavy materials such as glass, ceramics, and metals. However, due to the present state-of-the-art, solid materials would probably be segregated by hand in glove boxes or shielded facilities, depending on the radiation level.

The receiving areas for the trucks and rail cars would have provisions for decontamination cleaning of modest amounts of external contamination by spraying with steam or water. These reusable equipment items would be decontaminated to acceptable levels. Packages of radioactive wastes with large amounts of external contamination would be taken into a decontamination "hot cell" where external contamination would be removed remotely. In more severe cases the packages would be enclosed in a temporary secondary wrapping (such as plastic) and taken to an adjacent "hot cell" facility where the entire original package would be put into a permanent second steel container and sealed. This package would then be taken to the storage warehouse by the conventional handling techniques for waste packages. Decontamination of work areas, when required, would be accomplished by using combinations of chemicals (steam, detergents, acids, bases, and oxidants), elevated temperatures, and mechanical assistance (scrubbing, pressure blasting with liquids or solids such as ice or sand, or ultrasonic baths). Care would be taken in the selection of chemicals and materials for use in decontamination operations because these materials eventually become wastes and would have to be processed to inert solids within the facility.

Volume reduction of solid wastes depends on the form of the wastes. Mechanical compaction can usually achieve volume reductions of ten to one. However, if combustible wastes are compacted, a danger of fire hazard is always present in the stored wastes. In this design concept, all combustible wastes are incinerated before storage either at their source or at the storage facility. Significant volume reductions are achieved without compaction but compaction may also be utilized for some materials at the disposal facility.

Chemical leaching (such as acid digestion) of solid wastes is a possible alternative to incineration. Pyrolysis (incineration at low temperatures, up to about 800°C, under less oxidizing conditions) is another possible future alternative to incineration. However, neither of these processes is sufficiently developed to be used for radioactive waste application at present.

Incineration is currently used for some wastes at several AEC facilities and is the method selected for this conceptual facility. Of the many types of commercial incinerators, only the rotary kiln and the ordinary fire-box incinerator, which was selected here, appear to be readily adaptable to burning radioactive wastes. The ordinary incinerator has the disadvantage of incomplete combustion due to liquefied plastics dropping through the grate into the ash box. For radioactive wastes, off-gas cleaning is a significant problem because the off-gas equipment must be remotely operable, designed to be critically safe by geometry at all times, and able to remove submicron-size dust and corrosive and contaminated gases. Thus the large volume of off-gas must be treated to achieve radioactive decontamination factors of up to 10^7 for fly ash. A low-efficiency high-capacity filter system, a wet-scrubbing system, and a high-efficiency filter make the off-gas treatment auxiliaries the most expensive part of an incinerator system.

The ashes from the remotely operated incinerator would be conveyed to the incinerator loading station. Here, the ashes would be loaded into standard steel cans. The cans would be welded shut, monitored, labeled, decontaminated if necessary, and sent to the storage area.

Alternative methods for solidification or immobilization of radioactive wastes without volume reduction, such as mixing with concrete or asphalt, may also be used for some waste materials. A significant increase in waste volume generally results from the use of such procedures.

The solids processing systems would also be designed to handle all noncombustible solid wastes capable of fitting into the standard 4 ft by 4 ft by 6 ft metal canisters. The facility would probably not be capable of processing large pieces of equipment for disposal (these will have been reduced in size and packaged at their source).

The encapsulated and solidified alpha wastes would be stored in two types of warehouses. One type would include up to 3 ft of concrete shielding (including removable cover blocks) to allow for storage of wastes with high penetrating radiation fields. The other type would have essentially no shielding for wastes with very low penetrating radiation fields. The wastes in either case would be stored in rectangular, sealed steel canisters 4 ft by 4 ft by 6 ft. Transfer of wastes to the shielded warehouses would be done semi-remotely by crane whereas transfer to the unshielded warehouses would be done directly by forklift.

Each warehouse would be provided with its independent supply of treated and prefiltered ventilation air. The compartmented ventilation exhaust from each warehouse would be separately filtered and monitored prior to joining the exhaust air from the other warehouses. This compartmentation would facilitate identification of canisters that have developed leaks during storage. All exhaust air from each warehouse would be combined and passed through a bank of high efficiency filters and monitored before being exhausted through the central facility stack.

The wastes would remain in the warehouses untouched unless removed for recanning due to leakage or until transferred to other disposal facilities. Routine visual inspection would be made of each canister and continual monitoring of the exhaust ventilation air from each of the compartments in the warehouses would be maintained.

Environmental Effects of Facility Construction

The environmental effects of constructing the alpha waste storage facility would be similar to but considerably less extensive than the impacts described for the reactor (Section 4.2.4). Specific impacts can be summarized as follows:

- (1) Construction would produce noise and dust, but the soil would be stabilized during and after construction to minimize effects of winds.
- (2) Construction and operating manpower would be primarily local forces; fluctuations in local populations would be temporary and small.
- (3) Approximately 50 to 100 acres of land would be used for the facilities for 50 to 100 years.
- (4) A quarter-mile wide peripheral "exclusion area" would be restricted by fencing for the life of the facility but would not otherwise be disturbed; this land would be available for other use following decommissioning of the facility.

Environmental Effects of Operation

Up to 100 gal of water per minute would be used for operation of the off-gas scrubbers and various sanitary and operating purposes. Electric power usage would be less than 500 kW. Exhaust air from waste processing areas would be routinely filtered through three HEPA filters in series. Only very small amounts of radioactive material would be released, and exposure to a member of the public at the site boundary (one mile away) would be less than 0.1 millirem/year.

Monitoring Program

A major problem in handling alpha wastes is the determination of the amount of radioactivity in the incoming wastes. Waste producers will be required to inventory the radioactivity in the wastes they send in terms of curies or grams of fissile materials and curies or grams of other radioactive materials.

The incoming wastes must be monitored for total radiation level, external contamination, and total fissile content (to determine the potential for achieving criticality). To reopen cans already adequately treated and packaged for storage should not be necessary. The initial monitoring will result in segregation on the basis of radiation level, need for container decontamination, and need for treatment. Normal radiation monitoring and isolation techniques should be used in the sorting area with the assumption that all materials contain plutonium.

All effluents of the volume reduction processes will be monitored (stack gases, retention pond effluent) before release. Because of the possibility of accidental releases during processing, environmental monitoring of air, water, land, biota, and people around the site would be conducted routinely.

The solid waste materials would be monitored before canning to provide an estimate of the content of specific nuclides, which will require some sampling. The technology for low-level continuous inventorying of solids will have to be developed to perform this task, because such technology is not available at this time. A continuous program of air sampling and can monitoring would be used to check for failed cans.

Once the wastes were safely canned and stored in the warehouses, accountability of radioactive materials would be largely a matter of restricting unauthorized access to the facility, keeping adequate records of inventories placed in storage, properly labeling each can for identification, and periodically estimating real-time and projected inventories of radionuclides that build up and decay as time progresses. This last item is desirable to help define the current and future potential hazard of the waste and to provide input data for decisions that might be necessary regarding future disposition of the waste. Duplicate records should be kept in permanent, offsite storage centers so that future generations can be better assured of access to them.

Environmental Effects of Accidents

Safety problems associated with the storage facility are: (1) can rupture followed by release of radioactive material, (2) criticality, (3) overheating, and (4) natural disasters. Can rupture will be detected by ventilation gas sampling, followed by isolation and removal procedures. Criticality-potential detection equipment will be located in the storage array. Similarly, temperature recording will detect unusual radioactive heating. These instruments will be supplemented with automatic fire alarms and automatic fire fighting equipment. Fire should be avoidable in the storage areas because combustible wastes will have been previously incinerated. The facility will be constructed to survive tornadoes and earthquakes and located to avoid floods and ground subsidence. Procedures to prevent sabotage or theft are not yet well developed beyond the usual military-type guard system, a factor that will be considered in detail during the final design phase. In any event, a high fence will surround the total property of the disposal site, and another fence will probably be installed around the processing and warehouse area. The total site will have appropriate audio visual surveillance.

4.6.3.2.3 Noble Gases

The noble gas fission products, krypton and xenon, will be released from the fuel during reprocessing, recovered, concentrated, and bottled in standard 50-liter gas storage cylinders. Although there are no present regulations controlling disposal, future regulations are expected to require that these materials be stored in specially designed repositories. The only significant radioactive isotope at the time of storage will be krypton-85.

Noble Gas Storage Facility

A preliminary concept for a facility for long-term storage of noble gases was developed for the purposes of this Environmental Statement. The scope of analysis was limited to establishing the major factors of engineering feasibility. The overall criteria for the design are derived from those previously cited for the high-level waste RSSF. The facility would include areas for receiving, handling, and inspecting noble gas cylinders received from fuel reprocessors and for transferring and storing the cylinders in the final storage area.

The concept is based upon holding safely for at least 50 years all the noble gas waste produced by commercial nuclear reactors through the year 2000. It would be designed to:

- (1) Confine the radioactive gases and assure the health and safety of man and the protection of his environment.
- (2) Allow complete retrieval of the stored noble gases.
- (3) Provide required cooling and isolation.
- (4) Ensure safe operation even under emergency conditions plus safe recovery from unplanned events.
- (5) Meet applicable Federal, state, and local regulations.
- (6) Provide containment for radioactive gases under normal and postulated abnormal occurrences and withstand all credible natural forces.

The noble gas storage facility as conceived would be a single repository; however, it could be built as multiple facilities if desired.

Facility Concept Description

The noble gas storage facility would be designed to receive and store all the noble gases produced through the year 2000. Noble gas cylinder receipts are based on the forecast nuclear-fueled power generation through the year 2000 and the projected number of required gas containers, assuming recovery from all reactor fuel starting in 1985.

The fuel reprocessor would remove the noble gases and store them in 50-liter gas cylinders, 9 in. in diam by 50 in. high, and transfer them to the storage facility in batches of six cylinders per shipment. It is assumed that DOT-3AA2265 standard cylinders would be used.

A maximum of approximately 17,000 gas cylinders each containing 90 kg of noble gases at 2200 psi are required through the year 2000. The estimated maximum accumulation of radioactivity in the year 2000 (allowing also for decay) would be 1700 MCi, and the estimated maximum accumulation of radioactive decay heat from these cylinders in the year 2000 would be approximately 2800 kWt. The storage facility would consist of a waste gas cylinder receiving and handling area, inspection area, and final storage area. A plan view of the facility is shown in Figure 4.6-12 and section view is shown in Figure 4.6-13.

The cylinder receiving area would be a building into which the loaded truck carrying the shipping cask would be unloaded. It would be served by an overhead bridge crane. Space would be provided for setting down the shipping cask for radiation monitoring. The shipping cask would be unloaded underwater in a basin located adjacent to the truck dock. The cylinders would be placed in individual transfer

4.6-60

CONCEPTUAL NOBLE GAS STORAGE FACILITY -
PLAN VIEW

Figure 4.6-12

4.6-61

CONCEPTUAL NOBLE GAS STORAGE FACILITY -
SECTION VIEW

Figure 4.6-13

baskets in the basin for movement to the final storage area. The cylinders in their transfer baskets would be moved from the basin to a point inside the cylinder inspection area by an inclined conveyor.

The cylinder inspection and gas transfer area would be connected to the final storage area. It would contain equipment for testing the cylinders for leaks, for inspecting the cylinders for corrosion or physical damage, and for transferring gas between cylinders. After inspection, the cylinders would be transported by a remotely operated overhead crane to the final storage area.

Every gas cylinder in its transfer basket would be placed into a storage rack in the storage area. This storage rack, along with the transfer basket, would protect the cylinders from damage by failure of an adjacent cylinder.

To minimize the impact of leaks occurring during storage, the capability would be provided for transferring gas from a leaking cylinder to a sound cylinder in the cylinder inspection facility. Radiation detection equipment would be provided so that leaking cylinders could be detected early and removed for repackaging.

The assumed design heat load for a single gas cylinder is 245 W, and the temperature at the surface of the cylinder would be less than 100°C. The heat would be dissipated to the atmosphere through a stack. The maximum expected total heat output of the accumulated noble gas wastes would be approximately 500 kW in the year 2000 from LMFBRs and approximately 2800 kW from the total nuclear industry.

The facility will be designed to conform with all Federal, state, and local safety regulations with respect to protection from natural disasters.

Alternative storage concepts that might be considered include storage of the gas cylinders in stainless steel containers underwater or in a forced-air ventilation and cooling facility.

Environmental Effects of Facility Construction

The environmental effects of constructing the noble gas storage facility would be similar to but considerably less extensive than the impacts described for the reactor (Section 4.2.4). Specific impacts can be summarized as follows:

- (1) Construction would produce noise and dust, but the soil would be stabilized during and after construction to minimize effects of winds.

- (2) Construction and operating manpower would be primarily local forces; fluctuations in local populations would be temporary and small.
- (3) Approximately 50 to 100 acres of land would be used for the facilities for 50 to 100 years.
- (4) A quarter-mile-wide peripheral "exclusion area" would be restricted by fencing for the life of the facility but would not otherwise be disturbed; this land would be available for other use following decommissioning of the facility.

Environmental Effects of Facility Operation

The environmental effects of operating the noble gas storage facility might be as follows:

- (1) About 10 to 20 gpm of water would be used during the operating life of the facility for cooling and shielding the gas containers being processed (before storage) and for various sanitary and industrial uses. The cooling and shielding water would be circulated internally, and some would be released to the environs through a small cooling tower. The balance of the water would go mainly to surface ponds or to a septic tank and tile field from which it would eventually return to the source.
- (2) Electrical power usage would be no more than 200 to 300 kW in the year 2000.
- (3) Approximately 2.8 MWt of radionuclide decay heat would be dissipated to the atmosphere at peak load but would drop to 0.1 MW by the year 2050.
- (4) Normal radioactive material releases would be very small and would be exclusively to the air. At the site, radiation exposures from the facility would be less than 0.01 millirem/year for a member of the general public.* The corresponding population dose would be 0.2 man-rem/year.

Monitoring Program

An air monitoring program will be carried out at the noble gas storage facility to check routinely for possible radioactive releases. Air sampling stations will be installed at the time the facility is constructed. Samples would be taken routinely at the stations for evidence of radioactivity that would indicate a containment failure.

*Dose at 1000 m from stack due to release of contents of one entire gas bottle (150,000 Ci) in one year.

Environmental Effects of Accidents

The maximum credible accident in the facility is postulated to involve a major rupture of a single gas cylinder with the resultant rapid and total escape of the noble gas from the cylinder (Kr-85 is the only radioactive component). The accident is assumed to occur in the cylinder inspection area with the noble gases being released to the storage facility ventilation system.

Based on the assumed meteorological conditions and calculational procedures described in Appendix II.I for other fuel cycle facilities, the individual whole body dose at 1000 m would be 0.13 millirem. The population dose would be 0.37 man-rem.

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4.7 Plutonium Toxicity

4.7.1 Introduction

The purpose of this section is to examine whether the human health effects from plutonium and other transuranic elements released to the environment constitute a significant risk inherent in the operation of an LMFBR fuel cycle. This risk is an unfamiliar one, with certain unusual features, and for this reason it is a cause of public concern and deserves the special attention provided in this section. Among the unusual features of this risk are the following: (1) large quantities of toxic materials will be produced, but under circumstances such that their containment can be rigorously insured, (2) radioactive half-lives are long, so that risks may persist for over a hundred thousand years, (3) although the toxicity of these materials is well demonstrated in experimental animals, there is no direct knowledge of effects in man, and (4) such effects as might conceivably occur will be indistinguishable from the normal ills of mankind. These and related concerns were expressed by many who commented on the Draft Statement.*

The effort to estimate the health consequences of exposure to plutonium and other transuranic elements released during operation of the LMFBR fuel cycle follows the model portrayed in Figure 4.7-1. The model starts with an input of radionuclides, the source term, from the LMFBR fuel cycle. These radionuclides are dispersed to air, soil, and water, and are then followed along pathways leading to man. After inhalation or ingestion by man, there is a distribution to organs of concern, in which the radionuclides decay to produce a calculated radiation dose.

The model of Figure 4.7-1 oversimplifies the real-life processes and, as described in the caption, is even further simplified in its application by neglecting certain pathways. Justifications for this simplification, and for the choice of values assumed in the quantitative application of the model, are summarized in this Section and supported by more detailed information in Appendix II.G.

The choice of values for the many parameters involved in the quantitative application of the model is often difficult. Where data are not available to support the choice of a "most probable" value, estimates have been made that are clearly "conservative."

*Comment Letters 1 (B. Simon), p. 2; 3 (W. Beaver), p. 2; 7 (R. Daifuku), encl., p. 14; 8 (G. Younghein and I. Younghein), p. 1, encl. 1, pp. 6-8; 23 (N. Dawkins), p. 1; 59 (C. P. Brawman), p. 1; 61 (C. Benson), p. 1. Hearing Testimony - Pat Hoffman, p. 174; E. A. Martell, p. 232. Predraft Comment Letters 8 (NRDC), encl. p. 1; 11 (NRDC), p. 8; 13 (Zero Population Growth), p. 1; 16 (J. Legakes), p. 1; 17 (P. Tebow), p. 2; 18 (Center for Science in the Public Interest), p. 2.

THE HEAVY LINES INDICATE PATHWAYS
AND COMPARTMENTS EVALUATED IN THIS SECTION. THE DASHED LINES INDICATE
PATHWAYS WHICH ARE CONSIDERED IN THE TEXT BUT DEEMED INSIGNIFICANT FOR
THE PURPOSES OF THIS RISK EVALUATION.

SCHEMATIC DIAGRAM SHOWING PRINCIPAL PATHWAYS FOR ACCUMULATION OF TRANSURANIC ELEMENTS BY MAN

Figure 4.7-1

This approach can be misleading because it overestimates potential health consequences, yet the results obtained predict minimal health effects despite the multiplicative conservatism of the procedure.

The Environmental Protection Agency (EPA) has suggested that estimates based on the most probable values, with an indication of the range of uncertainty, would be more useful for public health evaluations, or, where conservative values are used, the degree of conservatism should be stated.* This is a valid philosophy and one with which an effort has been made to comply. But it is often not possible, on the basis of current knowledge, to provide defensible estimates of the most probable values, or even estimates of degrees of conservatism. In such cases a very conservative estimate is used. Similarly, the reliability of conservative estimates where data are not available was questioned.** The use of very conservative estimates in the absence of extensive data does not necessarily invalidate the conservatism. For example, lung retention in the infant as a function of particle size is not known; the assumption of 100% retention, although unrealistic, is conservative, yet the degree of conservatism is not known.

The estimation of health consequences, based on the model's prediction of radionuclide accumulation in the environment and in man, is an uncertain procedure because of the very low exposure levels predicted, the total lack of direct experimental data on effects at these low exposure levels, and the lack of knowledge of any well established mechanism of effect on which basis one might extrapolate from data obtained at much higher exposure levels. Recognizing this uncertainty, predictions of risk have been made, based on comparison of predicted radiation doses in man with the dose-response data from animal experiments with plutonium, and with the dose-response data from human exposure to other forms of radiation. Relative risk comparisons involving fewer extrapolation uncertainties have been made between the predicted doses due to releases from the LMFBR fuel cycle and occupational exposures to plutonium, exposure to particulate plutonium from weapons test fallout and exposures to natural background radiation. These comparisons of health consequences are summarized in this section and described in greater detail in Appendix II.G and in responses to Comment Letters.

Many of those who commented on the Draft Statement expressed general concern regarding the adequacy of current knowledge to identify and quantitate potential

*Comment Letter 53 (EPA), pp. 2, 5.

**Comment Letter 32 (R. W. Freedman) p. 1.

health consequences, or the adequacy of the approach taken in the Draft Statement to estimate health consequences.* These concerns can be generally grouped into several categories, and are addressed specifically where appropriate in the appendices: inclusion of nonroutine releases of transuranic elements, addressed in Appendix II.G.1; adequacy of specific dispersion treatment, transfer coefficients, and metabolic modeling, addressed specifically in Appendices II.G.2, II.G.3 and II.G.4; validity of the estimation of health effects, addressed in Appendix II.G.5; and the "hot particle" issue, addressed in II.G.6.

Other commentors have raised questions regarding the adequacy of current standards.** Estimates of health effects in this Statement do not involve assumptions based on radiation standards or maximum exposure limits recommended by national and international bodies or specific regulations and operating procedures based on these standards. It is assumed that radiation exposures will be kept as low as practicable, in conformance with the basic guidance provided by these organizations and AEC policy.

4.7.2 Source Term

The source term is defined in Table 4.7-1. A release of 0.36 mCi of alpha-emitting transuranic elements is estimated for each 1000 MWe-year of LMFBF operation.[†] The derivation of this source term is summarized in Appendix II.G.1 and is based on transuranic release rates discussed in preceding parts of Section 4. The major fraction of the release occurs from the fuel reprocessing operation. Although relatively high releases may result from an individual accident, such accidents are predicted to occur so infrequently that they contribute insignificantly to the total source term. The release is airborne and predominantly particulate in nature, with an assumed activity median aerodynamic diameter (AMAD) of 0.3 μ m.

*Comment Letters 6, (NRDC) pp. 1-40; 23 (N. Dawkins), p. 1; 27 (D. P. Geesaman), pp. 6-12; 32 (R. W. Freedman), p. 1; 38 (NRDC), pp. 2-12; 51 (EIAP), pp. 16-37; 53 (EPA), pp. 2, 8-10; 57 (J. L. Menke), p. 1; 58 (C. R. Lewis), p. 1.
Hearing Testimony - Pat Hoffman pp. 183-185; E. A. Martell pp. 232-247.

**Comment Letters 7 (R. Daifuku), p. 10; 51 (EIAP), p. 26; 53 (EPA), pp. 6, 11.

†Predraft Comment Letter 6 (NCI), p. 1; 8 (NRDC), encl., p. 6.

†The Draft Statement employed a source term of 1 mCi per 1000 MWe-year. This included an erroneous value of 0.7 mCi for Cm-242 released rather than the correct value of 0.07 mCi. The corrected value for total alpha-activity becomes 0.36 mCi. We have carried this two-digit number, rather than rounding to 0.4, or 1.0, because it is the sum of numbers for individual isotopes, and calculations have been made for each isotope separately.

Table 4.7-1

QUANTITIES OF TRANSURANIC ELEMENTS RELEASED TO THE
ATMOSPHERE PER 1000 MWe-YEAR^a

Radionuclide	Half Life (years)	β -Activity (mCi)	α -Activity (mCi)	Mass (μ g)
Pu-238	86.4		0.18	10
Pu-239	24,400		0.04	650
Pu-240	6,600		0.05	220
Pu-241 ^b	13.2	5.4 ^b		48
Am-241 ^b	458		0.01 ^b	3
Cm-242	0.44		0.07	0.02
Cm-244	17.6		0.01	0.01
Total		5.4	0.36	930

^aThe quantities enumerated for each radionuclide are estimates of the routine release from fuel reprocessing operations. Quantities released in other operations or in accidents make a comparatively insignificant contribution to the total (Appendix II.G.1).

^bIn evaluating health consequences, the β -radiation from Pu-241 is insignificant compared to the α -radiation from other transuranic elements. However, the 5.4 mCi of Pu-241 decays to produce an additional 0.16 mCi of Am-241, which is included as a part of the source term, where appropriate.

Opposing points of view raised by the Natural Resources Defense Council (NRDC), Environmental Impact Assessment Project (EIAP), EPA, Mid-America Coalition for Energy Alternatives (MACEA), and others suggest that this source term does not account for, or adequately treat, the unscheduled event.* The potential accident situation, based either on experience or on probabilities, is addressed in Appendix II.G.1.

Deliberate dispersal of plutonium in the public domain might be attempted following a diversion from the LMFBR fuel cycle, or through sabotage of LMFBR facilities or transport. The development of a plutonium source term associated with deliberate

*Comment Letters 8 (G. Younghein and I. Younghein), encl. 1, pp. 2-7, encl. 2, pp. 5-9; 22 (MACEA), p. 7; 32 (R. W. Freedman), p. 3; 38 (NRDC), pp. 3, 6-11; 51 (EIAP), pp. 23, 24, 53-59, 74, 75; 53 (EPA), p. 6; 57 (J. L. Menke), pp. 1, 2; 66 (Office of Environmental Affairs, DHEW), p. 1.

releases would require prediction of the quantities and frequency of the releases. At present, there is no body of historical data on the occurrence of such acts, nor is there a method to predict, in quantitative terms, the effectiveness of the safeguards system in limiting their frequency and severity. While the systematic approach described in Section 7.4.8 for the development of future safeguards may enable a degree of quantification, it is unlikely that complete quantification of system effectiveness will be possible. Considering these points, and taking into account the intent to provide safeguards which will continue to assure a very high level of protection, it is concluded that no incremental plutonium source term should be assigned for deliberate releases.

It is also possible to conjecture that a foreign enemy attack, including attack on U.S. nuclear facilities, would result in substantial additional releases of radioactive materials, including plutonium, to the environment. There is no way to predict the probability of such an attack. Also, as indicated in Section 7.4.4.1, the overall consequences of a massive strategic attack would mask any incremental effects resulting from facility damage. For these reasons, it is not possible to justify a plutonium source term associated with acts of war.

4.7.3 Environmental Dispersal

For the purposes of this generic Environmental Statement, all of the source term is assumed to be released to the atmosphere. The initial release to the aqueous system, as well as all subsequent transfers from air to water or from soil to water, is ignored. While such a simplification might not be appropriate if one were concerned with the siting of a specific plant, it seems amply justified in this treatment of the general case, because of the small fraction initially released to water, the very low solubility of most transuranic compounds, and the great dilution volume provided by the world's oceans. Consistent with this approach is the assumption that terrestrially deposited transuranic elements are not depleted in the soil, but remain for their total radiological lifetime, fully available for root-uptake by plants. Environmental dispersal is considered in detail in Appendix II.G.2.

Atmospheric Dispersal

For modeling the dispersal of the source term within the atmosphere and its deposition from the atmosphere, a single point of release in the North Central

United States is assumed. This choice was based on the availability of a model developed for, and the availability of actual data from, this site.¹ It should be noted that there is a relatively large population downwind from this site and therefore the population density considered is higher than the average U.S. population density by about a factor of 10.

Application of the meteorological model, as detailed in Appendix II.G.2, results in the prediction that a fraction of 4×10^{-6} of the source term will be inhaled by the exposed population of the United States before the material reaches the ground. After reaching the ground, the material may be resuspended, and it is estimated that an additional fraction of 9×10^{-6} of the activity originally released will eventually be inhaled as a consequence of this resuspension, about one-fourth of this within 2 years following the initial release. A total fraction of about 1.3×10^{-5} of the source term is therefore predicted to reach the exposed population via inhalation, half of this during the first 2 years following release and the remainder at later times, extending for the lifetime of the radionuclides involved.

This estimate of the total fraction of the source term inhaled by the total exposed population of the United States is, of course, sensitive to the point of release; it will be higher for points of release closer to downwind population centers and lower for sites located upwind of sparsely populated areas. It is based upon average meteorological conditions and upon somewhat uncertain theory, particularly as regards the resuspended fraction. As discussed in Appendix II.G.2.4, a smaller fraction of the fallout plutonium from weapons tests seems to have been inhaled by man.² While the behavior of this fallout plutonium cannot be assumed to exactly match that of the transuranic releases from LMFBR operations, it does provide an important real-world check on the theoretical calculations - a check which suggests that these calculations may overestimate the fraction inhaled.

Terrestrial Dispersal

For the purpose of estimating man's uptake of the terrestrially dispersed source term via food chains, the entire source term is assumed to be uniformly distributed over the surface of the United States, where it attains a uniform concentration in the soil to a depth of 20 cm. It is retained for the lifetime of the radionuclides involved. There is no downward movement in the soil beyond the root zone of plants and no loss due to runoff. However, fallout plutonium from weapons tests have, in

Several measured instances, already penetrated to soil depth of greater than 20 cm.³ The assumption of uniform distribution over the United States will, of course, underestimate the deposition immediately downwind from the source; and if this area is used extensively to produce food crops, may underestimate the transuranic elements reaching man via food chains. An opposite effect will occur if the heavier deposition occurs on land unsuitable for agricultural use.

4.7.4 Transport Via Food Chains

Transuranic elements in soil may reach man via a number of food chain routes, the more important of which are indicated in Figure 4.7-1. Of these, the most significant on the basis of present information would seem to be the direct pathway involving absorption of the radionuclide from the soil by plants and ingestion of the plants by man. For estimating ingestion by man, it is assumed that plants will incorporate transuranic elements to a concentration that is 10 percent (plant wet-weight basis) of the concentration of these elements in the soil. This is likely a conservative estimate, approximating the plant uptake of americium under near optimum conditions and exceeding published uptake values for plutonium in agricultural crops by several orders of magnitude.^{4,5} This value is used because of the very long time period that the plutonium isotopes will remain in the soil (half-life of Pu-239, 24,400 years), and because there is no certain way of predicting possible changes in the availability of plutonium to plant roots over such a time period. There are, in fact, some data suggesting increases in availability over time spans of 2 to 5 years.^{6,7} (See Appendix II.G.3.)

Because of this uncertainty regarding the long term availability of plutonium in soil, and because of the conservative treatment of root uptake dictated by this uncertainty, no attempt is made to model the finer points of food chain transport to man. It is assumed that all plant-derived food ingested by man will have a concentration of LMFBR-released transuranic elements equal to 10 percent of the concentration of these elements in the top 20 cm of soil. This leads to a total ingestion by the U.S. population, over the lifetime of the radionuclides involved, of about 10^{-2} of the released activity; this treatment leads to total ingestion values that contribute significantly to human exposure, when compared to the contribution via inhalation. A more detailed consideration of the data supporting these assumptions is included in Appendix II.G.3.

4.7.5 Metabolism and Dosimetry in Man

With a few exceptions noted below, the assumptions of the International Commission on Radiological Protection (ICRP) are employed for modeling the behavior of transuranic elements in man and for calculating radiation doses to organs.^{8,9,10} These ICRP procedures and their application are described in more detail in Appendix II.G.4.

For the case of inhaled radionuclides, the ICRP Lung Model distinguishes between various solubility categories.⁹ For calculating dose to lung, the LMFBF transuranic particles are assumed to fall in the least soluble class, thus maximizing retention in lung and thoracic lymph nodes, and radiation dose to these organs. For calculating dose to other organs, an intermediate solubility is assumed for the transuranic particles, which results in more rapid clearance from lung to blood and somewhat larger doses to bone and liver. This ensures that dose to a specific organ will not be underestimated because of uncertainty as to the chemical nature of the material inhaled. But since the same material is assumed to be in two places at the same time, these assumptions necessarily overestimate the combined dose to all organs.

In accordance with the ICRP, it is assumed that 3×10^{-5} of ingested plutonium will be absorbed from the gastrointestinal tract to blood.⁸ For americium and curium an absorption fraction of 10^{-3} is assumed.^{10,11} Transuranic elements reaching the blood, whether absorbed from the lung or from the gastrointestinal tract, are assumed to deposit 45% in bone and 45% in liver; the material in bone is retained with a half-time of 100 years and the material in liver with a half-time of 40 years; all of these numbers conform to ICRP recommendations.¹⁰ A deposition from blood of 0.05% in the testes and 0.01% in the ovaries, with total retention, is also assumed. These values are not specified by ICRP but are based on published and unpublished data on gonadal content of transuranium elements in experimental animals and humans.^{11,12}

Employing the dose calculation procedures of the ICRP, estimates were made of the total 70-year dose commitment to several organs resulting from the inhaled and ingested fractions of the initial source term release. These procedures are described in Appendix II.G.4. Briefly, they take account of the physical decay and biological retention of the radionuclide, including daughter radionuclides where pertinent, and sum the dose over 70 years following initial deposition. Since exposure occurs at all ages there is an inherent conservatism in the use of a 70-year dose commitment, because the remaining life expectancy for ages beyond

infancy is progressively less than 70 years. These dose estimates for lung, bone, liver, gonads,* and thoracic lymph nodes are shown in Table 4.7-2, expressed in units of man-rem. These represent the total cumulative dose equivalent for all people for all time from transuranic elements released per 1000 MWe-year of LMFBR operation.

4.7.6 Estimates of Health Effects

The various estimates made of exposure to transuranic elements released from the LMFBR fuel cycle are summarized in Table 4.7-2. Relating these estimates of exposure to health effects must necessarily be done indirectly, since there are no observed health effects from transuranic elements in man. The relationship must also be quite uncertain, since the estimated levels of exposure are far lower than can be studied experimentally, and because no proven theory exists to support the prediction of radiation effects at such levels. A number of comparisons can be made, however, which give an indication of the probable magnitude of effects (see also Appendix II.G.5).

Comparison with Current Levels of Alpha-Emitters

Table 4.7-3 compares the exposure estimated to result from the operation of LMFBRs during the Year 2020, with various measures of exposure from (1) natural background radiation, (2) naturally occurring alpha-emitting radionuclides, (3) fallout plutonium present in our environment as a result of nuclear weapons tests, and (4) human plutonium depositions resulting from occupational exposures. The Year 2020 is taken as representing a matured, and perhaps maximum LMFBR program, with an installed LMFBR capacity of 2,200,000 MWe, or more than five times the current total installed electric capacity of the United States. The data of Table 4.7-3 are considered in greater detail in Appendix II.G.3 and Appendix II.G.5.

The numbers in Table 4.7-3 afford many interesting comparisons with the estimated exposures due to Year 2020 operation of 2,200,000 MWe of LMFBR capacity. The most extensive comparisons are those which can be made with the fallout plutonium from weapons tests. This fallout plutonium exceeds by a factor of 20,000 the estimated Year 2020 LMFBR release.³ Its measured concentration in plant-derived foods (largely external contamination) is 2,000 times the estimated contamination of foods by the Year 2020 LMFBR release.¹³ As one proceeds down the LMFBR column in Table 4.7-3, to estimates farther removed from the source term, the comparison

*A 30-year dose commitment is calculated for gonads based on the assumption of an average reproductive age of 30 yr.

Table 4.7-2

SUMMARY OF PRINCIPAL MODEL ASSUMPTIONS AND PREDICTIONS

	Reference to Appendix Tables	For Generating Capacity of 1000 MWe-year	For Year 2020 Generating Capacity of 2,200,000 MWe-year
Transuranics Released to Air (Ci)	II.G.1-1	0.36×10^{-3}	8×10^{-1}
Initial Transuranic Concentration in Soil (Ci/g - all in top 20 cm)	II.G.3-2	1×10^{-22}	2×10^{-19}
Initial Transuranic Concentration in Food (Ci/g)	II.G.3-2	1×10^{-23}	2×10^{-20}
Transuranics Ingested by U.S. Population (Ci)	II.G.3-2	4×10^{-6}	9×10^{-3}
Absorbed from G.I. Tract (Ci)	II.G.3-2	4×10^{-10}	9×10^{-7}
Transuranics Inhaled by U.S. Population			
Directly Inhaled (Ci)	II.G.2-2	1.4×10^{-9}	3×10^{-6}
Inhaled after resuspension			
During first 2 years (Ci)	II.G.2-2	0.8×10^{-9}	2×10^{-6}
After first 2 years (Ci)	II.G.2-2	2.5×10^{-9}	6×10^{-6}
Total Inhaled (Ci)	II.G.2-2	5×10^{-9}	1×10^{-5}
Radiation Dose to U.S. Population from Transuranics			
To lung (man-rem)	II.G.4-2	4	9×10^3
To bone (man-rem)	II.G.4-3	26	6×10^4
To liver (man-rem)	II.G.4-4	11	2×10^4
To thoracic lymph nodes (man-rem)	II.G.4-5	200	4×10^5
To gonads (man-rem)	II.G.4-6	0.4	9×10^2

Table 4.7-3

COMPARISON OF ESTIMATED EXPOSURE FROM LMFBR TRANSURANIC ELEMENT
RELEASE WITH EXPOSURES FROM OTHER SOURCES

	Units	Transuranic Release from LMFBRs in Year 2020 ^a	Plutonium- Exposed Workers ^b	Fallout Plutonium from Weap- ons Tests ^c	Naturally Occurring α -Emitters ^d	Total Natural Radiation ^d
Total Released to Air	Ci	0.8		16,000		
Concentration in Soil	10^{-18} Ci/g	0.2		4,000		
Concentration in Plant- Derived Food	10^{-20} Ci/g	2.0		400		
Total Inhaled by U.S. Population	10^{-5} Ci	1.				
Total Deposited in U.S. Population	10^{-5} Ci		0.5	60	10,000	
Dose Equivalent to U.S. Population ^e						
Lung	10^4 man-rem	0.9		300	80,000 ^f	
Bone	10^4 man-rem	6.		140	50,000 ^f	200,000 ^f
Liver	10^4 man-rem	2.		80		
Lymph nodes	10^4 man-rem	40.		10,000		
Gonads	10^4 man-rem	0.09			5,000	60,000

^aEstimated release for an assumed generating capacity of 2,200,000 MWe-year. Numbers taken from Table 4.7-2.

^bNumbers derived in Appendix II.G.5.1.

^cNumbers derived in Appendix II.G.3.

^dNumbers derived in Appendix II.G.5.2.

^eDose equivalent from LMFBR release is a 70-year dose commitment (30 years for gonads) to all persons exposed for all time. Dose equivalent from fallout plutonium is a dose commitment to the Year 2000 for all persons exposed from 1954 to 1972. Dose equivalent from natural radiation is calculated for 70-year exposure of 2×10^8 people (30-year exposure for gonads).

^fDose to bone-lining cells.

with the comparable fallout plutonium measurements becomes closer; e.g., while releases differed by a factor of 20,000, estimates of dose equivalent differ by only about 100. This would indicate that transuranics released from the LMFBR fuel cycle are estimated to be more effectively accumulated by man than is fallout plutonium; or, alternatively, that the LMFBR numbers are overly conservative. Both factors may be involved.

Radiation doses from the naturally occurring alpha-emitting radionuclides dwarf those from either fallout plutonium or predicted LMFBR releases.¹⁴ The distribution of this natural dose in bone and lung may not be exactly comparable to that from the transuranic elements, however, the differences are so large that this factor seems unimportant. The natural background dose to the gonads is especially noteworthy, since it exceeds the estimated LMFBR transuranic release dose by a factor of $\sim 10^6$ ($\sim 5 \times 10^4$ for naturally occurring alpha-emitters).

It is of interest that the estimated total human inhalation of transuranic elements from Year 2020 LMFBR operations is about equal to the total amount of plutonium deposited in a few thousand plutonium workers. These workers were exposed, mostly via inhalation, to forms of plutonium that may closely simulate the LMFBR release. Thus far no effects have been seen which are directly attributable to the plutonium deposition.

The comparisons of Table 4.7-3 provide no measure of the absolute risk of health effects from transuranic releases from the LMFBR fuel cycle. They argue convincingly, however, that such effects will be small compared to whatever effects may result from fallout plutonium. They also suggest that any effects from LMFBR releases (or from fallout plutonium) would be totally obscured by the much larger incidence of effects to be expected from natural background radiation, if, indeed, any effects are to be expected from any of these sources.

Comparison with Animal Toxicity Studies

Direct information on the toxicity of transuranic elements is available only from studies in experimental animals. The radiobiological literature suggests that the biological effects observed in such animal experiments will approximate those that would occur in man exposed under the same conditions. For this reason, it is justifiable to look to the results from extensive animal experimentation for guidance in estimating the health risks from exposure to transuranic elements.

These studies, reviewed in Appendix II.G.5.3, suggest that bone and lung cancers are the most important effects of exposure to the lowest levels of transuranic elements thus far studied.¹⁵ The levels employed in these studies were very much higher, however, than the estimated exposure from LMFBR releases. Thus, the lowest average tissue dose that has shown a significantly increased cancer incidence in experimental animals is about 300 rem to lung and 1500 rem to bone, doses that are about 10 million times higher than the estimated average lifetime dose to a resident of the United States from Year 2020 LMFBR operations. The dose to any given individual may be larger or smaller than the average but is, in any case, very much smaller than doses that have led to observed effects in animals. To extrapolate over this range of dose, without a proven theory relating dose to effect, is clearly a very uncertain operation. At present there is little choice but to assume that health effects are linearly related to dose, as discussed in Appendix II.G.5.5. Such linear projections were employed in obtaining the cancer risk estimates summarized in Table 4.7-4. The data on which these estimates are based are summarized in Appendix II.G.5.3.

The reasonable agreement among the few species for which data are available, particularly with respect to bone cancer incidence, lend some credence to the usefulness of the risk estimates in Table 4.7-4. One is aware, however, that quantitative extrapolation of the results of animal experiments to man is uncertain. For example, in the case of radium-226, where cancer risks can be computed for both experimental animals and man, the risk per rem to man is a factor of ten lower than the risk to dogs or mice.¹⁶

Animal data on liver cancers are too limited to permit dose-response estimates, but liver cancers seem less probable than bone or lung cancers. Despite the relatively high radiation doses to thoracic lymph nodes, there is no indication from animal studies of significant tumor production in the lymphatic system.

Comparison with Human Cancer Risk Data

The National Academy of Sciences - National Research Council Advisory Committee on the Biological Effects of Ionizing Radiations (BEIR) recently reviewed the extensive data on human cancer mortality risks from exposure to ionizing radiation.¹⁷ From these data they derived factors relating dose to estimated risk for various cancer types. Their derivation assumed that the human experience, largely with external irradiation at relatively high dose rates and high total doses, could be linearly extrapolated to zero effect at zero dose. While these cancer risk estimators from the BEIR report have the advantage of being based on human data, they share with

Table 4.7-4
CANCER INCIDENCE PREDICTIONS BASED ON PLUTONIUM TOXICITY STUDIES IN EXPERIMENTAL ANIMALS

1	2	3	4	5	6	7
Animal Data ^a			Predictions for LMFBR Transuranic Releases			
Tumor Type - Animal	Compound and Route of Administration	Increased Cancer Incidence per rem	For Generating Capacity of 1000 MWe-year		For Year 2020 Generating Capacity of 2,200,000 MWe-year	
			Man-rem ^b	Cancer ^c	Man-rem ^b	Cancer ^d
Bone			26		6 x 10 ⁴	
Dog	Citrate - intravenous	7 x 10 ⁻⁵		.002		4
Rat	Various	1 x 10 ⁻⁵		.0003		0.6
Mouse	Citrate - intravenous	2 x 10 ⁻⁵		.0005		1.2
Lung			4		9 x 10 ³	
Dog	Oxide - inhaled	7 x 10 ⁻⁵		.0003		0.6
Rat	Various - inhaled	6 x 10 ⁻⁵		.0002		0.5
Rat	Pu-238 oxide - inhaled	70 x 10 ⁻⁵		.003		6

^aCondensed from Tables II.G.5-8 and II.G.5-9, Appendix II.G.5.3.

^bFrom Table 4.7-2.

^cProduct of Columns 3 and 4.

^dProduct of Columns 3 and 6.

the animal studies the uncertainty of linear extrapolation from high dose observations, and have the additional disadvantage that the observed effects did not result from internally deposited transuranic elements.

Table 4.7-5 lists cancer mortality predictions for LMFBR transuranic releases, based on the BEIR report risk estimators; the procedures employed in arriving at these predictions are described in Appendix II.G.5.4. The rather close agreement between these estimates and the estimates from animal data in Table 4.7-4, should not be taken as evidence for the absolute accuracy of the estimates. The agreement more likely reflects the assumptions made in the derivation of both estimates, especially the assumption of a linear mechanism of effects. Nevertheless, the agreement increases the confidence with which the results from animal experiments are applied to the prediction of health effects in human beings. However, to establish the validity of this requires further animal and human data from comparable exposure situations.

Comparison with Genetic Risk Data

The BEIR report also provides estimates of the genetic risk from irradiation. Such risks are difficult to quantify, since they range from dominant single gene mutations whose effects are categorically recognizable, to subtle genetic contributions to disease conditions that are predominantly of non-genetic origin. To cope with this range of effects, the BEIR report made separate estimates of risk for specific genetic defects and for defects with complex etiology, with a 10-fold uncertainty assigned to the first category and a 100-fold uncertainty to the second. Some of this uncertainty may have been resolved by recent data reported by Newcombe, who suggests a total risk of hereditary defects of 20×10^{-6} per man-rem.¹⁸ The derivation of these risk estimates and the uncertainties involved are discussed in Appendix II.G.5.4. Table 4.7-6 lists predictions of genetic defects attributable to LMFBR transuranic releases, based on the two BEIR categories and on Newcombe's more recent data.

4.7.7 Consideration of Special Problems

In arriving at the estimates of health effects just presented, it was necessary to make many simplifying assumptions. While these assumptions have been defended as conservative, are there perhaps non-conservative aspects of what may be generally conservative assumptions? Several of these special problems are considered in the following discussion.

Table 4.7-5

CANCER MORTALITY PREDICTIONS BASED ON HUMAN IRRADIATION EXPERIENCE

1	2	3	4	5	6	7
Tumor Type	Risk Model ^a	Cancer Deaths Per Man-Rem ^b	For Generating Capacity of 1000 MWe-year		For Year 2020 Generating Capacity of 2,200,000 MWe-year	
			Man-Rem ^c	Cancer Deaths ^d	Man-Rem ^c	Cancer Deaths ^e
Lung	Absolute	16×10^{-6}	4	0.00006	9×10^3	0.1
	Relative	110×10^{-6}	4	0.0004	9×10^3	1.
Bone	Absolute	2×10^{-6}	26	0.00005	6×10^4	0.12
	Relative	17×10^{-6}	26	0.0004	6×10^4	1.02
Liver	Absolute	1×10^{-6}	11	0.0001	2×10^4	0.02
	Relative	7×10^{-6}	11	0.00008	2×10^4	0.1

^aThe different assumptions of these two models are explained in the BEIR report and in Appendix II.G.5.4.

^bThe risk estimates for lung and bone are taken directly from the BEIR report. No risk estimate for liver cancer is given in the BEIR report; the values listed were derived by procedures described in Appendix II.G.5.4.

^cFrom Table 4.7-2.

^dProduct of Columns 3 and 4.

^eProduct of Columns 3 and 6.

Table 4.7-4

PREDICTIONS OF GENETIC DEFECTS

1	2	3	4	5	6
Type of Risk	Incidence Per Man-Rem	For Generating Capacity of 1000 MWe-year Man-Rem to Gonads ^a	Genetic Defects ^b	For Year 2020 Generating Capacity of 2,200,000 MWe-year Man-Rem ^a to Gonads ^a	Genetic Defects ^c
Specific Genetic Defect ^d					
lower limit of estimate	50×10^{-6}	0.4	0.00002	900	0.04
upper limit of estimate	500×10^{-6}	0.4	0.0002	900	0.4
Defects with Complex Etiology ^d					
lower limit of estimate	10×10^{-6}	0.4	0.000004	900	0.009
upper limit of estimate	1000×10^{-6}	0.4	0.0004	900	0.9
Newcombe Estimate of Total Genetic Defects ^e	20×10^{-6}	0.4	0.000008	900	0.02

^aFrom Table 4.7-2.^bProduct of Columns 2 and 3.^cProduct of Columns 2 and 5.^dEstimates from BEIR report.^eEstimate of H. B. Newcombe.

Problems of Averaging Activity and Dose

Throughout this document calculations have been made as though the radionuclide release was infinitely divisible, whereas, in fact, it is composed of particles with an assumed AMAD of 0.3 μm . These particles are not of a single diameter, but exhibit a distribution of sizes and shapes. There is no way of predicting the continuing particle size or composition; in fact, the particles originally released may contain uranium or other diluting material. Therefore it would be unreasonable to attempt to employ a "particle model" for distribution of the source term beyond the initial atmospheric phase.

If the transuranic elements entering soils as particles remain in this form, transuranic availability would be diminished compared to the case where the particles are subdivided and dissolved by physical and chemical weathering processes or otherwise made unavailable. This occurs because increased surface area is exposed for chemical reaction on physical subdivision of the particle. On the basis of present information, it is not possible to quantify the extent of particle subdivision and chemical dissolution which will occur as a function of the time periods of concern. It would appear likely, however, that entrance into the soil as an oxide particle would have the effect of reducing at least initially transuranic availability to plants relative to entrance into the soil in soluble form.

Of greater interest are the possible consequences of inhaled particles. Is the intense radiation dose to tissue surrounding a particle more hazardous than the dose from the same quantity of radionuclide dispersed more uniformly throughout the lung? This question is addressed in Appendix II.G.6. While it can be argued on theoretical grounds that particles of a particular activity-range might be more - or less - hazardous than the same quantity of alpha-emitter uniformly distributed, an abundance of experimental animal data indicates that particles are less hazardous.¹⁹ The lack of observed health effects in workers that have inhaled plutonium (usually in particulate form) also argues against any surprisingly large hazard due to these particles. The use of an average lung dose in our estimation of health risks is therefore considered an appropriate procedure, which does not underestimate risks.

EPA has questioned several of the bases of this approach.*

- (1) They state that the guidance of national and international standards-setting bodies is not applicable to this problem because (a) they have

*Comment Letter 53 (EPA), pp. 6, 9, 10, 11.

never established a maximum permissible lung burden for plutonium and (b) it is difficult to relate dose and lung burden for particulates. A report of the National Council on Radiation Protection and Measurements (NCRP) published in 1953 (National Bureau of Standards Handbook Number 52) stated a maximum permissible lung burden for plutonium. Subsequently the dose concept was utilized, but the NCRP established and recommended procedures for converting dose to organ burdens and/or body burdens. It is clear from their reports that both the NCRP and the ICRP at various times have considered the "hot particle" question, and, although cautions are stated, have not thus far recommended any method of calculation other than the average dose to the organ.

- (2) They question the relevancy of data from animal experiments in addressing health consequences in humans from inhaled plutonium. It is a commonly accepted practice to investigate the harmful effects of toxicants in animals in order to gain insight into the behavior and effects of the toxicants in man. This usually involves a multiple species approach to avoid basing interpretations upon biological processes specific to a single species. Inhaled plutonium has been studied in mice, rats and dogs, and, despite the limited information on the distribution of inhaled plutonium in the human, there is no reason to believe that it would differ qualitatively in the human. Inhaled plutonium accumulates primarily in the subpleural and peribronchial areas of the lung, exposing alveolar epithelial cells and causing bronchiolar-alveolar adenocarcinoma. This differs from the bronchogenic carcinoma arising from the basal epithelial cells of the bronchial airways which is seen in the uranium miners. This has led some to conclude that irradiation of the human lung will result in bronchogenic cancer and not bronchiolar-alveolar carcinoma. However, studies of the human lung following radiation therapy show that the parenchymal portion of the lung is more susceptible to radiation damage than is the tracheobronchial area (the site of bronchogenic cancer). Therefore, it is believed that results from studies with inhaled plutonium in animals may be more pertinent to the consideration of potential health effects in man than are the data from the uranium miner lung cancer cases and from acute exposure of humans to external radiation, and that the pathological changes in man should be qualitatively similar to those noted in animals.
- (3) They suggest that a number of the statements made and evidence presented in the Draft Statement in support of the conclusion are inadequate, inappropriate or not convincing. The discussion of the "hot particle" issue has been revised in the Final Statement (Appendix II.G.6) to

reflect recent reviews of the subject¹⁹ and to more clearly identify the evidence upon which the conclusion is based.

The NRDC assumes that the entire normal release consists of particles conforming to their definition of a "hot particle," and that the risk of lung cancer is one for every 2000 hot particles.* Other commentators have supported or referred to the NRDC position.** Estimates of health consequences based upon this position exceed those presented in this Impact Statement. The stated risk per hot particle is not valid in the opinion of the AEC because the hypothesis upon which this risk estimate is based is not supported by any available experimental evidence. Considering the assumptions stated in the source term and the assumptions inherent in the risk factor justification, the AEC cannot accept the value for potential health consequences claimed by the NRDC as resulting from a 1000 MWe-year. Further discussion is provided in Appendix II.G.6 and in the AEC responses to Comment Letters 6 and 38.

Martell, representing the EIAP, presented oral testimony,[†] subsequently submitted as a part of the EIAP written comment,^{††} suggesting that tobacco smoke contains small insoluble particles in which radioactive decay products of natural radon are aggregated, and that alpha particles emitted from those particulates which reside in the bronchial bifurcation of the lung are a significant cause of lung cancer in smokers. This suggests that there is a significant alpha-irradiation related health risk in smokers, that insoluble particles containing a picocurie of plutonium would be an unacceptable public exposure, and that present levels of plutonium from fallout and that postulated low level releases from nuclear fuel cycle operations can be expected to result in serious health consequences. Martell published data suggesting that small quantities of lead-210 were concentrated in small insoluble particles in cigarette smoke; however, no evidence is presented supporting the assumptions that these are retained and accumulated in the bronchial epithelium of the lung, or for the suggestion that this may be the primary etiological agent for lung cancer. Physiological evidence indicates that long term residence at the bifurcation is unlikely. In addition, the temporal patterns of excess risk observed in epidemiological studies of lung cancer in smokers are not compatible with the suggestion that radioactive particles are the principal cause of lung cancer in smokers. Additional discussion can be found in the AEC response to Comment Letter 51.

*Comment Letters 6 (NRDC), pp. 1-40; 38 (NRDC), pp. 2-11. Predraft Comment Letter 8 (NRDC), pp. 2-6.

**Comment Letters 7 (R. Daifuku), encl. pp. 15-16; 16 (J. T. Edsall), p. 1; 23 (N. Dawkins), p. 1; 27 (D. P. Geesaman), pp. 6-18; 59 (C. Brawman), p. 1. Hearing Testimony (P. Hoffman), p. 184.

†Hearing Testimony (E. A. Martell), p. 232ff.

††Comment Letter 51 (EIAP), pp. 19-22.

Problems of Averaging People

The relatively small number of particles available for inhalation, the fact that these particles are released from a small number of fuel reprocessing plants, plus the fact that most of the direct inhalation of particles will occur within 50 miles of the release point (Appendix II.G.2), all combine to insure that LMFBR-derived transuranic element concentrations in the lungs of different people will vary over a range of hundreds or thousands. Whether this fact alters the acceptability of the very small total risk of health effects is a question pertinent, primarily, to decisions concerning the siting of individual fuel reprocessing plants.

Another type of people-averaging is implicit in the lack of separate consideration of the fetus, infant, or child. Data bearing on this problem are considered in Appendix II.G.4, and it is concluded that hazards to the very young are adequately covered by adult criteria.

Problems of Averaging Time

Estimates of health effects have been noted, as in Tables 4.7-4 to 4.7-6, without regard to their distribution in time. It is important to have an estimate of the total effect over the lifetime of the radionuclides involved, but it is also important to know how these effects are distributed in time. From Table 4.7-2 it is evident that about half the intake of the source term by man occurs by inhalation during the first 2 years following release. It would therefore follow that, except for genetic defects, at least half of the total predicted effects would occur during the first generation. Beyond 2 years the model postulates a constant rate of uptake, but at a diminished level, modified only by radioactive decay of the source term. The second half of the predicted effects should therefore be spread over the generations subsequent to the one initially exposed, at a much diminished rate, decreasing with radioactive decay. Because of the greater conservatism reflected in the long-term aspects of the model, it is probable that considerably more than one-half of the total non-genetic effects will be evidenced in the first generation.

4.7.8 Conclusions

Throughout this section attention has been directed to the uncertainties involved in the various assumptions. At this point, in arriving at final conclusions, the cumulative uncertainty of these conclusions must be of particular concern.

Thirty percent of the estimated inhalation of transuranics by man occurs during the primary phase of atmospheric dispersion (Table 4.7-2). The assumptions involved

in estimating this initial inhalation will principally affect the pattern of dispersion. Since the areas downwind from the reference release site are fairly populous, the quantity ultimately inhaled is relatively insensitive to these assumptions. It is therefore felt that the total quantity inhaled from the dispersed plume by the exposed U.S. population is a reasonable estimate for the site selected, but might vary over a 10-fold range depending upon the exact location of the release site with respect to population centers. This is primarily a matter to be evaluated in specific plant siting. The remaining 70% of the total estimated inhalation occurs as a result of resuspension, and is probably an overestimate, since it assumes no loss of availability subsequent to an initial 2-year weathering.

The estimated fraction of the LMFBR release that is inhaled by the exposed U.S. population is 25 times larger than the fraction of fallout plutonium inhaled (Table 4.7-3). It is reasonable that the LMFBR estimate should be larger, because of the higher population density of the heavy dispersal area for the LMFBR release, as compared to the U.S. average, and because the long-term inhalation of resuspended material is not reflected in the 20-year experience with fallout plutonium. These factors might not explain the total discrepancy; however, the agreement is reasonably good.

The estimated fraction of the LMFBR release that is ingested by the exposed U.S. population is estimated as 0.01. This rather unreasonable value results from a combination of conservative assumptions necessitated by the absence of pertinent data. Thus, it is assumed that the transuranic elements remain available for root uptake, in the upper 20 cm of the soil, for their lifetime; no sink is provided. It is further assumed that food plants will contain a transuranic element concentration equal to 10% of the concentration of these elements in soil. This level of accumulation has been observed in laboratory experiments and might occur under certain agricultural conditions. The models employed indicate that organ radiation doses from ingested transuranics are comparable to those from inhaled transuranics.

With regard to the assumed absorption and distribution of the transuranic elements in man, the ICRP models employed are, in general, "best estimate" models. Some conservatism results from the duplication of exposure involved in maximizing both retention in and movement from the lung, and from the practice of assuming

a 70-year period for accumulation of dose, regardless of age at deposition. These are minor uncertainties, however.

A major uncertainty is that involved in translating estimates of dose to estimates of specific health effects. Without doubt, the most valid comparisons are those gross measures which avoid the uncertainties of extrapolation far beyond observed effect. Thus, the most convincing comparisons are those in Table 4.7-3, which say that the predicted LMFBR releases for the Year 2020 are less by a factor of 20,000 than plutonium deposited in the U.S. from weapons tests; and that the conservatively estimated inhalation of transuranic elements in the total exposed population from Year 2020 releases is one ten-thousandth of the naturally occurring alpha-emitters in this population; and that the conservatively estimated doses to lung, bone and gonads are lower than those from naturally occurring radiation by factors of 10^4 to 10^6 . All of these numbers place the predicted LMFBR-derived transuranic exposures (even though probably overestimated) in a category of insignificance when compared with presently accepted and unavoidable risks.

The effort to place absolute numbers on specific risks must be looked upon as an exercise lending qualitative support to the conclusions implicit in Table 4.7-3. The numbers of cancers and genetic defects listed in Tables 4.7-4 to 4.7-6 have no precise quantitative significance. They are based upon conservative estimates of exposure, multiplied by conservative estimates of the risk from this exposure. Whether the actual risk approaches these numbers, or is zero, can in no way be inferred from our present knowledge; nor would such a choice appear to be of practical importance to decisions with regard to the LMFBR, considering the very low values estimated for these risks.

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VOLUME II

SECTION 5

**ECONOMIC, SOCIAL AND
OTHER IMPACTS**

SUMMARY FOR SECTION 5

The indirect effects on society due to the introduction of the LMFBR are found to be very similar to, or natural extensions of, those effects of present-day reactor systems. Two groups of indirect impacts are considered in this section: the economic impacts (Sections 5.2 and 5.3) and the sociopolitical impacts (Section 5.4).

The LMFBR could place at the Nation's disposal an abundant domestic source of electrical energy that will not be subject to rising fuel costs and will cause a minimum additional environmental disruption due to mining. Although the benefits of energy to society are difficult to quantify, there exists a close relationship between energy use per capita and economic activity. The industries most affected by changes in the prices of electricity are the extractive and heavy chemical industries.

In the industrial sector, the utilities will be most directly affected. Although the very large capital requirements that are anticipated will stretch and probably change utility financing methods, other electric energy generation systems have comparably large capital needs. The LMFBR will also require increased technical sophistication on the part of the utility industry, which will thus result in changes in its manpower requirements.

Although the national economic impact of the anticipated LMFBR construction program is significant, the impact on the local or regional scale can be even greater. The possibility of placing large numbers of LMFBRs in energy centers emphasizes the need for early regional energy planning.

Many comments were received expressing concern about the effect of the development of nuclear power on the structure and institutions of society. These included:

Will implementation of a safeguards program create a police state?

Waste management requires that man issue guarantees for events far into the future.

The probability of the conversion of an LMFBR to military production by a foreign government in violation of international agreements must be considered.

The implementation of an effective safeguards program would, during normal operations, have very little public impact. Emergency measures following a hypothetical successful diversion would concentrate on rapid recovery of the stolen material.

Such emergency situations, however, can be expected to be extremely infrequent and of short duration. Consequently, using potential or attempted diversion as an excuse for the application of widespread measures to suppress liberty or privacy is highly unlikely.

The accumulation of long-lived radioactive wastes raises unique questions of social responsibility to future generations. There are several options for permanent disposal of high level wastes open to contemporary society, all of which need to be carefully evaluated on both technological and societal grounds, in view of the unprecedented period of future time involved. Societal questions are at least as important as technological considerations. Assuming, for example, that high level wastes are to be sequestered in geologic formations, should the choice be one of foreclosing future accessibility to the extent possible? What options might future generations like to find left open to them? The technological testing of disposal options, as currently planned, will probably extend into the next two decades before serious proposals would be entertained for commitment to a permanent disposal method. An interim Surface Storage Facility for solidified high level wastes seems necessary for the sound management of wastes, now in existence and being generated, until permanent solutions can be adopted.

The problem of the proliferation of nuclear weapons is believed to be only marginally affected by the U.S. decision to develop the LMFBR. Any country with a medium-scale industrial base and any of several current fission energy conversion systems can, if it desires, produce nuclear weapons. In addition, foreign breeder reactor development programs are already well advanced.

5.1 INTRODUCTION

In other sections of this report, the first-order consequences of the introduction of the LMFBR into the Nation's energy system are examined in detail. The "first-order consequences" mean the direct environmental impacts resulting from the operation of the LMFBR plants and attendant fuel cycle, the resources consumed, and the economic costs of the program. However, the history of technological societies since the industrial revolution shows very clearly that some of the most important impacts of new technologies are often not those first-order effects but rather the consequences for society of those effects. For example, some of the first-order effects of the automobile are increased speed of travel, air pollution, and the use of land for highways and parking lots. Important as they are, these effects are overshadowed by the profound ways in which the automobile has altered the way that our society structures itself. Almost every facet of the structure and character of our society--the way it is distributed geographically, the relationship between generations, the prevailing moral codes--has been influenced by the existence of the automobile.

In the past, society has been particularly inept at anticipating the second-order consequences of new technologies. Admittedly, even if effects such as those of television or the automobile could have been anticipated, what society could or would want to do about them is not clear. In the case of innovations promoted and instituted by the government, however, it is desirable to attempt to assess second-order consequences in order to anticipate the full ramifications of policy decisions. Such an analysis would have two benefits: first, it could contribute to decisions regarding the development of the technology itself; second, it could help to mitigate any social, institutional, or economic difficulties attendant upon the large-scale introduction of the technology. Accordingly, this section focuses primarily on the second-order consequences of the introduction of the LMFBR.

There are three factors that determine the impact a new technology can have upon society. First is the degree to which the new technology is different from the one that it replaces. Second is the extent of interaction of the technology with the general public. Third is the extent to which the new technology is distributed throughout society.

Under the first criterion, nuclear power, not the LMFBR, represents a major change from previous (fossil-fuel) technology. As shown in the succeeding portions of this section, as well as in other sections of this Statement, many of the impacts of the breeder reactor are the same as those of light water reactors (LWRs). Both fuel

cycles use fissionable material, require a more sophisticated technology and a high level of quality assurance, and incur radioactive effluents and waste products. Larger quantities of plutonium will be handled in the fuel cycle of an LMFBR compared with that of an LWR of the same generating capacity fueled with recycled plutonium. Moreover, in a breeder the total amount of plutonium fuel available will increase more rapidly as plutonium is bred and more LMFBRs come on-line. This quantitative difference is central to many of the issues raised about LMFBRs and will be discussed in Section 5.4, as well as in Section 7.

The second criterion aims at distinguishing technologies that interact directly with important parts of society (as, for example, television and the automobile interact directly with the general public) from those that interact only indirectly (such as a radically new means of extracting iron ore). The technologies of the second sort can have profound effects, but those effects tend to be softened due to the "distance" between the technology and the public. Such is the case for the LMFBR whose primary product is electricity which, from the viewpoint of the consumer, is indistinguishable from electricity produced from fossil fuels.

The third criterion distinguishes between those technologies that are potentially widespread throughout society and those that are limited by geographical, economic, or other factors. Unlike some other energy forms, such as geothermal energy from natural steam sources, the LMFBR has few practical geographic or other constraints, apart from requiring, like LWRs, a site relatively remote from large population centers. It, therefore, has very wide potential application. Thus, while the first two criteria tend to lessen the potential second-order consequences of the LMFBR, the third emphasizes the importance of those consequences that can be identified.

In discussing various impacts of the introduction of the LMFBR, comparison of those impacts in relation to those of alternative technologies is appropriate. For example, to estimate the industrial impact, an examination of the demands on other industries relative to those of other central station technologies will be informative. In other cases (e.g., sociopolitical impacts), consideration of all possible alternative technologies will not be possible, except in a very general sense. For some effects, most alternatives will have similar impacts (e.g., capital requirements for central station electric generation). Other impacts (e.g., those related to plutonium safeguards) will apply quite uniquely to all nuclear reactors (LWRs and LMFBRs). As mentioned above, many of the impacts that will be discussed are natural extensions of the impacts of current LWR technology.

A word should be said about the general character of many of the issues raised in this section. They are the kinds of issues that are not often dealt with in conventional cost-benefit analyses, because they are not easily quantified. Some of these issues deal with equity questions relative to different socioeconomic groups or different generations. Others are basically ethical in nature. Thus, one cannot expect any resolution of these particular issues. One can merely raise the issues as they apply to the LMFBFR to provide an input to what must be a societal decision process. The preparation and review of this Statement is an important mechanism to ensure public participation in this decision.

The underlying question being addressed through this analysis is: Should society invest the funds necessary to bring into being an LMFBFR technology? Are the total benefits to society that result from the program greater than the total costs? The costs and benefits include all effects on society and must be compared with the costs and benefits of providing energy through other current and future technologies. One might ask, however, if the frame of reference has been too narrowly limited. What about other uses for the funds that are required to bring a commercial LMFBFR into existence? Wouldn't they be better spent in building hospitals or schools or producing food for this country or others?

The achievement of long-term social and national goals is, however, related to energy. A statement frequently made is that a national purpose should be improved quality of life rather than increased material consumption. However, when considering energy consumption and national output, how a country allocates priorities on the uses of its productive resources is a moot point. Building and operating schools and hospitals and attaining higher environmental standards use energy and expend national production just as manufacturing and marketing more consumer goods do. The maintenance of an abundant, domestic electric energy supply for the next 40 years is, in itself, a worthwhile social goal.

Thus, once the need for a certain level of energy production is conceded, a more limited analytical framework can be assumed. The question then becomes one of finding the best ways of providing that energy from a total societal perspective. Admittedly, the larger framework must be considered at some stage. It enters, for example, when decisions are made as to what total funds are to be devoted to energy research and development relative to other Federally funded activities. The allocation of priorities within the Federal budget is a complex social question that our political system is designed to address. That larger framework need not

be applied here--which is fortunate, for the job defined by the narrower framework is quite difficult enough.

5.2 NATIONAL IMPLICATIONS OF THE AVAILABILITY OF ELECTRICITY FROM THE LMFBR

5.2.1 The Role of Energy in Economic Growth

Energy is such a basic and pervasive commodity in advanced industrial economies that a detailed analysis of its importance in terms of simple cause and effect relationships is virtually impossible. The differential between the energy available and that needed to merely sustain life has given man the power to run the machinery of our society. This differential has enabled us to transport ourselves all over the globe and beyond, control our indoor temperatures, cleanse our effluents, improve our food supply, and diagnose our diseases. In the U.S. about one-fourth of all the energy currently used is in the form of electricity. However, energy consumption patterns are shifting and by the turn of the century, electricity will probably supply about half of the total energy in the U.S.

Some idea of the fundamental role of energy can be obtained by analyzing the growth of energy use and other economic indicators in individual countries, the relationship between energy and gross national product (GNP) for different countries, and the relationship between productivity and energy use in given industries as a function of time.

Figure 5.2-1 shows the historical relationship between energy consumption and GNP in the U.S. since 1920.¹ The most striking feature of the curve is the close correlation between economic growth and energy consumption.

As shown on Figure 5.2-2,¹ the energy/GNP ratio gradually declined from 1947 through about 1966; at this point the trend reversed until 1970 when the ratio started to decline once again.²⁻⁴ Even though this ratio is again declining moderately, its level is significantly above the historical trend line and its decline is at a lower rate than in the past. Many explanations have been given for the reversal of the energy/GNP ratio occurring between 1966 and 1970. Some of the more prominent explanations of the upward trend between 1966 and 1970 are:

- (1) Decrease in the historical improvement in the efficiency of generation of electricity,
- (2) Increased use of electricity as an intermediate energy form, particularly for air conditioning,

Source: U.S. Department of the Interior
ENERGY CONSUMPTION AND GROSS NATIONAL PRODUCT,
1920-1970, WITH PROJECTIONS TO 1990

Figure 5.2-1

Source: U.S. Department of the Interior
 RATIO OF U. S. ENERGY CONSUMPTION TO REAL GNP

Figure 5.2-2

- (3) Increased non-energy use of fuel resources,
- (4) A decline in automobile mileage per gallon.⁵

The influence of the last two factors is shown by comparing the lower curve of Figure 5.2-2, which does not include power used by utilities, with the upper curve. Clearly, the electrical sector accounts for only part of the increase. The above explanations, however, fail to account for the current resumption of the energy/GNP decline.

In the industrial sector, the key role of energy can be seen by analyzing productivity increases in given industries as a function of time. The latter analysis reveals particularly the role of electrical energy in increasing labor productivity. In fact, in many cases there appears to be a linear relationship between value added and the product of kilowatt-hours consumed and man-hours used.⁶ Thus, a decrease in the use of electricity, brought about by higher prices, for example, could be compensated for by an increase in the use of labor. The result may be a lower labor productivity. Clearly, the nature and extent of this kind of effect is very much dependent on details of specific industries.

Figure 5.2-3 shows the relationship between energy consumption per capita and GNP per capita for a number of countries in the world as of 1965.⁷ Once again, the correlation between economic activity and energy use is striking.⁸

Figure 5.2-3 also shows a large range of energy use in different countries at a fixed level of GNP. A comment* on the Draft Statement requested further discussion of this disparate aspect of energy consumption. No systematic analysis has thus far been made of the many factors that could influence this relationship. Nevertheless, a number of important factors such as the size of the country (influencing transportation requirements), the climate (influencing heating and cooling requirements), natural resources, degree of urbanization, and industrial-agricultural mix are not significantly a matter of choice. Another feature of the data underlying Figure 5.2-3 is that the amount of energy used per dollar of GNP in the U.S. is not high relative to that in many other countries.

5.2.2 Economic Effects of the Price of Electricity

For evaluation of the secondary effects of the introduction of the LMFBR on the U.S. economy, an examination of the current and projected role of electricity as a "fuel"

*Comment Letter 22, pp. 7-8, (D. Tegtmeier, Mid-America Coalition for Energy Alternatives).

Source: Resources for the Future
 PER CAPITA WORLD ENERGY CONSUMPTION AND GROSS NATIONAL PRODUCT (1965)
 Figure 5.2-3

in industry and for the American consumer is essential. For example, an important question revolves around the significance of electricity price increases for different industries and for different consumer groups.

What industries are most sensitive to changes in the price of electricity? Table 5.2-1 presents a list of those industrial sectors for which the cost of electricity is greater than 1% of the value of their final product. The figures are based on detailed 1963 inter-industry input-output coefficients from the U.S. Department of Commerce, the most recent detailed coefficients available.⁹ A significant conclusion from Table 5.2-1 is that the major industrial uses of electricity are the primary extractive and heavy chemical groups. For those basic industrial groups, the cost of electricity is 2 to 10% of each industry's contribution to the GNP. Prices of these fundamental materials are reflected throughout the economy. Maintaining an abundant and reliable supply of reasonably priced electric energy for these commodities has significance for sustaining a reasonable cost of living in the United States.

In the consuming sector, the statistic of primary interest is that, until recently, electricity purchases accounted for about 2.5% of the income of the average U.S. household. Recent fuel-price increases have raised that percentage somewhat. In terms of the socioeconomic impact of the price of electricity, an important question is the amount paid for energy by various income groups. Although the number of studies in this area is very limited, two features stand out. One is that electricity consumption is strongly correlated with income. In the western U.S., for example, families with disposable incomes below \$5000 a year expend, on the average, about \$50 a year on electricity whereas households with incomes above \$15,000 spend about \$130.¹⁰ The other feature of the effect of income on energy use is that lower income groups spend a larger fraction of their income on electrical energy. For example, a recent study of energy use in New York City shows that families with annual incomes in the \$3000 to \$5000 range spend about 5% of their income for electricity. For those with incomes above \$20,000, the corresponding figure is 1%.¹¹ These figures strongly suggest that the demand for electricity is inelastic among low income groups. Another conclusion is that increases in the cost of electricity have the greatest impact on families in the lowest income brackets. For this reason, increased attention has been focused recently on rate structures with varying suggestions for alleviating the disparate effects.

These data are significant for two reasons. First, improvement in standard of living is generally associated with increased energy use--and particularly with

Table 5.2-1

MAJOR INDUSTRIAL USERS OF ELECTRICITY^a

Industrial Sector	Purchases from Electric Utilities		
	Dollars (millions)	Percentage of Value Added by Sector ^b	Percentage of Sector's Gross Domestic Output ^c
Primary Aluminum Manufacturing	243.4	9.4	3.1
Iron and Ferroalloy Ores Mining	33.8	4.2	2.2
Nonferrous Metal Ores Mining	53.4	5.1	2.4
Stone and Clay Mining and Quarrying	77.3	4.1	2.4
Chemical and Fertilizer Mineral Mining	26.6	4.4	2.7
Petroleum Refining and Related Industries	191.2	2.2	1.0
Coal Mining	106.1	4.1	2.4

^aThe numbers in this table do not reflect the sharp rise in energy costs since October 1973.

^bValue added is defined as the industry's contribution to GNP.

^cGross domestic output includes both inter-industry sales and all final demand uses such as sales to domestic consumers, net exports, and net inventory damage.

increased electrical energy use. Second, a proportional change in the price of electricity affects lower income groups more strongly than higher ones. These are the principal factors that relate the cost and availability of energy to questions of social welfare. Although consideration of less direct relationships, such as the effect of energy cost on the prices of goods¹² and the differential effects of energy externalities on various socioeconomic groups, would be useful, such effects are currently not well understood. There is general agreement, however, that increased energy costs have been a major contributor to recent inflationary trends.

5.2.3 Non-Economic Factors, National Welfare and the LMFBR

There are a number of factors having to do with energy supply that are not easily translatable into economic factors but are of direct importance to a nation's welfare. In 1972, 39% of electricity production was based on the use of oil and natural gas although a significant portion of this country's original resource of both of these fuels had already been consumed. The continued use of oil and gas to produce electricity seems particularly unwise considering the dwindling supplies and the other uses for which these resources are uniquely suited. However, the difficulties involved in quickly shifting to other energy sources makes it likely that oil will continue to be used in electrical power plants during the next decade. The economic incentives for expediting, whenever possible, the substitution of nuclear energy for oil in electric generation are high. Ten 1000-MWe base-loaded nuclear plants would generate in one year the same quantity of electricity as would be produced by the combustion of about 100 million barrels of oil. At a current world price of \$10/bbl of oil, this amount would translate into a balance-of-payments credit of about one billion dollars.

The introduction of nuclear power, and particularly the breeder, has significant national value from the point of view of the management of our natural resources. Such considerations were central to the rationale for the nuclear energy program described in the report from the AEC to President Kennedy in 1962¹³ and more recently, for the breeder program, in President Nixon's Energy Message of June 4, 1971.¹⁴

Events in the Middle East at the end of 1973 have made the Western World well aware of the hazards of dependence on other countries for fuel supplies. Not only has the international political influence of the "oil weapon" been clearly demonstrated, but the domestic economic impact of curtailed energy supply has also been shown. Major economic dislocations caused by doubling and redoubling of international oil prices

are already evident. Thus, an energy form based on domestic resources, such as nuclear power, has immediate, significant national benefits.

At this point, a benefit of LMFBR technology should be mentioned. For the LMFBR, leveled fuel costs will be very small. Fuel cycle costs (in 1974 dollars) for LMFBRs coming on line in the 1990's are projected to be 1 to 2 mills/kWhr. On a similar basis, estimated fuel cycle costs in 1990 for LWRs will be 2 to 3 mills/kWhr, and for coal systems they are projected to be about 8 to 10 mills/kWhr. (See Section 11 for information about comparative electric generation costs.)

The U.S. now accounts for some 35% of world energy consumption. As per capita energy consumption rises elsewhere in the world during the remainder of this century, the competition will increase for ever-decreasing amounts of fossil fuel. Thus, the resource-related arguments for the LMFBR will inevitably strengthen in the future.

As pointed out in Sections 11.2 and 4.1, depleted uranium tails from enrichment plants, a byproduct of LWR fuel cycles, will be the source of uranium for breeders for many decades. Furthermore, the amount of uranium consumed by the LMFBR is so small (about one metric ton of uranium per year for a 1000-MWe plant) that any further mining requirements after the stockpile of uranium tails has been exhausted will be very low. The LMFBR places at the Nation's disposal a source of electrical energy that will not be resource-limited for centuries and will require minimum further environmental disruption due to mining.

There are implications for the country in terms of "second-order" effects of the availability of such a source of electrical energy. One such implication is the possibility of substituting electricity for other forms of energy, including electrification of certain forms of transportation. In addition, steam from nuclear plants could be used directly for industrial processes. The shift to electricity would diminish environmental pollution and, when combined with the use of effluent controls at the electric generation site, could greatly improve environmental conditions throughout the U.S.

5.3 SPECIFIC ECONOMIC IMPACTS OF THE LMFBR PROGRAM

5.3.1 Effect on the Utility Industry

One of the industries most directly affected by the introduction of the LMFBR will be, of course, the electric utility group. As in a number of other instances, some impacts of the LMFBR will be a natural extension of the impact of the introduction

of LWRs in the 1960's. Others will reflect the energy system context in which the trend to LMFBRs is occurring. Two areas of effect on utilities will be discussed briefly here--the implications of the very large amounts of capital required to finance the anticipated LMFBR construction program and the implications of the increasing sophistication of electrical generation technology.

5.3.1.1 Financial Requirements

The first major impact on the utility industry will be in the area of generating sufficient capital to finance the program. Rough estimates by the Atomic Industrial Forum of the capital requirements through 1990 to finance electric utility growth are \$274 billion for new generating capacity; \$213 billion for transmission, distribution, and other facilities; and some \$11 billion for nuclear fuel. Total capital requirements (in 1970 dollars), including refinancing of maturing bond issues, would be approximately \$512 billion.¹⁵

The Federal Power Commission, in the 1970 National Power Survey, estimated a total requirement for financing for plant and distribution facilities of from \$400 to \$500 billion in the period from 1970 to 1990. The rate of expenditure was estimated to be \$20 billion in 1980 (in constant 1970 dollars). A more current estimate by First National City Bank of New York placed the annual figure for 1980 at \$26 billion.¹⁶

In the initial appraisal by the Nuclear Task Group of the National Petroleum Council, an estimate of \$4.9 billion is given for the investment required in the nuclear fuel cycle during 1971 to 1985.¹⁷ These estimates are given only to indicate the order of magnitude of the financial requirements anticipated in the electric utility industry.

The growth in utility capital expenditures for new generation, transmission, and distribution capacity has been tremendous over the last 14 years. In 1962 the total was \$4.5 billion, in 1967 \$8 billion, and by 1973 \$16.7 billion. The figures in the above paragraph indicate a continuation and acceleration of these historic trends.¹⁸

The magnitude of the required funding in this single industrial sector raises the possibility of a major impact on capital markets. The ability of the industry to generate sufficient capital, as well as the resultant impact on financial markets, can be evaluated as the program progresses. The larger initial capital cost burden for the utility industry imposed by nuclear power is compensated for by significantly lower fuel costs over the life of the plants.

The ability of the electric utility industry to generate sufficient financing for the LMFBRs involves questions about whether sufficient capital is available in

financial markets and whether the earnings of the utility industry are sufficient to attract this capital. The problems connected with attracting sufficient capital for the expansion of electric generating capacity from all energy sources, including the LMFBFR, must be resolved by the utility industry.

Table 5.3-1 compares projected average annual requirements for utility investment (including interest) for additions to central station capacity with the projected GNP from 1985 to 2020. Total requirements for thermal reactors and fossil plants are, to varying degrees, understated due to elimination of plant replacement costs for this comparison.

For this analysis, note that in 1970 the total of funds raised in all credit markets was \$115 billion or 12% of the GNP. In 1970 private utilities accounted for 7% of total investment. This amount includes funds raised by financial sectors such as savings institutions and commercial banks as well as by nonfinancial sectors such as government, households, and noncorporate businesses. The figures for the year 2000 in Table 5.3-1 indicate that the LMFBFR alone will require 1% of the projected GNP. If the 1970 ratio of total credit market funding to GNP persists, the LMFBFR alone would require about 8% of such funds.

Only private capital has been considered here, because the electric utility industry in the U.S. has, in the past, relied principally on private financing. In addition, there may be increased public involvement in the future through expansion of public systems such as TVA and through loan guarantees, interest subsidization, and tax relief. This involvement might also include state or even municipal governments.

A comment* was received concerning the impact of the large capital requirements for LMFBFRs:

Do these words mean that because of the LMFBFR program that less money will be available for other goods and services for the citizens of the United States?

An adequate response to this question involves two major ideas. The first point is that energy is required to provide many of the "goods and services" to which the commenter alludes. The second idea involves a comparison of the capital requirements for the breeder with the capital requirements of other energy systems, as well as a projection of the total monetary benefits which would accrue to society because of the introduction of LMFBFRs into the electrical generation economy.

*Hearings Transcript, Testimony of Mrs. Pat Hoffman, p. 178.

Table 5.3-1

PROJECTED COMPARATIVE CAPITAL REQUIREMENTS
(billions of 1974 constant dollars)

Year	GNP ^a	Annual Capital Requirements ^b		
		Coal	Thermal Reactors ^c	LMFBR
1985	1486	3.9	10.8	0
1995	2124	6.2	12.8	13.5
2000	2543	16.9	20.2	24
2020	5174	26.4	8	58

^aBased on 1974 GNP of \$1007 billion and an assumed real growth rate of 3.6% per year.

^bAll costs based on 4000-Mwe, 2-unit plants using mid-1974 dollar values. Interest during construction is included. Capital costs and installation rates are based upon Case 3 assumptions and results of the cost-benefit analysis of this Statement (See Sections 11.2.3 and 11.2.4.)

^cThis column includes LWR and HTGR installations. After year 2000, all LWR additions will be plutonium burners. The breeder produces plutonium surplus to the system's need after about year 2005, and use of this surplus plutonium in LWRs is economical.

For example, solar energy has been suggested as an alternative to the LMFBR. Capital costs of solar energy systems are discussed in Section 6A.5.7. Assuming only over-night storage capacity, the capital costs of the solar-thermal conversion systems with mass-produced components would be more than twice that of LMFBR systems. (Even for the southwestern U.S., a minimum of 4 days of storage would be required--Section 6A.5.7.1.1.) Again, for photovoltaic conversion systems, Section 6A.5.7.1.2 states that solar costs would exceed conventional costs by a factor of 4 in the year 2000 and by a factor of 1.7 in the year 2020.

Capital cost trade-off analyses among better-defined technologies (coal and nuclear thermal converters) were considered in the cost-benefit analysis (Section 11.2). A conclusion of that analysis was that, compared with LWR systems, introduction of the breeder would defer large capital expenditures required to mine and mill uranium and to provide separative work capacity. In 1974 dollars, these deferred expenditures, cumulative through the year 2020, for uranium mining and milling would be about \$100 billion and for separative work they would be about \$60 billion (see Figure 11.2-16).

A conclusion of the cost-benefit analysis is that about 1.7 trillion dollars (undiscounted) will be available through the year 2020 to the citizens of the U.S. for "goods and services" if the LMFBR is developed for electric energy generation, instead of relying on other generation systems.

5.3.1.2 General Shift to a High Technology Industry

In the past, technology developmental efforts in the electric utility industry were focused principally on evolutionary improvements in fossil fuel technology. Until the last decade or so, electric utilities were not heavily involved in electric generation research; the major research and development work was performed by the equipment manufacturers. With the advent of nuclear power, the utility industry is rapidly becoming highly technological, and it is now exploiting some of the most complex technology of modern civilization.

In the future the utilities will require the most capable technical talent available. This prospect was recognized by the Utilities R&D Goals Task Force. In its report to the Electric Research Council, the Task Force emphasized the active role that the representatives of the utilities must play in creating the kind of talent they require through "utility system R&D programs, and as participants in programs of continuing education."¹⁹ Such programs should contribute to making the industry more vigorous and technically competent. In fact, highly trained manpower will be required throughout the utility industry from top management to technical,

operating, and maintenance personnel. Manpower requirements for the LMFBFR are discussed in more quantitative terms in Section 10.2.

What are the broader implications of this introduction of more technically trained personnel into the utility industry? For one, this shift is already under way, stimulated by the construction of light water nuclear power plants. The introduction of the breeder will require that this process be accelerated. Other new electrical fuel conversion developments, from low Btu coal gasification to fusion, will also require personnel familiar with advanced technologies.

The result of these new requirements should be a gradual change in the qualifications of utility management personnel. Top staff members will be more likely to have technical backgrounds in addition to legal and financial competence. This trend could well lead to a more active utility role in stimulating the introduction of new technologies and in the support of energy research. The current trend in this direction has been recently emphasized by the formation of the Electric Power Research Institute. With a new dedication to high technology brought about by its utilization of nuclear power, the electric utility industry is returning to the innovative role it played in the late 19th century when it introduced to society a new and versatile energy form.

Concern* has been expressed about the ability of utilities to meet the new requirements imposed by safeguards regulations that will pervade the nuclear industry, particularly with regard to:

The attitude and response of the nuclear industry to the significant "upgrading," "modification," "expansion," and "strengthening" of the safeguards program which the AEC asserts in the Draft Statement as an essential feature of the LMFBFR Program.

To be effective, the "upgraded" AEC safeguards regulations must be implemented by knowledgeable and sympathetic utility managers. They should understand the unique aspects of the use of nuclear fuel and should initiate and maintain training programs to assure that their employees have the competence, the proper attitude, and the sense of responsibility required for continued compliance with safeguards regulations.

Adherence to safeguards procedures is only one aspect of a complete industrial commitment which is a prerequisite of an effective safeguards program. An involvement with safeguards begins as soon as a utility announces its plans to build a nuclear

*Comment Letter 20, pp. 13, 15, (D. E. Abrahamson, Natural Resources Defense Council).

facility. Safeguards and security considerations are included in the design of the plant and such systems are reviewed during extensive licensing procedures. As described in the preceding paragraph, safeguards procedures are implemented at each facility and compliance is assured by AEC inspections. The AEC safeguards program is described further in Section 7.4.

5.3.2 Effects on Other Industries

In Section 5.2, the question of the impact on the national economy of the product of the LMFBFR--available and relatively inexpensive electricity--was addressed. In the previous subsection, the effect on the industry most directly influenced by the breeder program--the utilities--was analyzed. Here the implications of the breeder program are considered in terms of the sectors of the economy that provide inputs to the construction and operation of LMFBFRs.

Clearly, the construction of LMFBFRs will require inputs from a large variety of industries. Included, among others, would be the industries involved in raw material extraction and refining, the manufacturers of plant components, the fuel cycle industries, and the service industries.

There will also be major impacts in support industries such as rail transportation, where large quantities of coal are now transported. All electric energy growth projections call for an increased use of coal as an energy source, and the total transportation requirements for coal are likewise expected to increase. Quantification of such impacts depends not only on the rate of energy growth and the rate of introduction of the breeder reactor but also on the fuel mix in the remainder of the energy sector.

The long time scale over which this analysis has been made precludes any very detailed analysis of industrial impact. Nevertheless, a rough indication of these impacts can be obtained through three analytical approaches: (1) analyzing the primary material requirements of the LMFBFRs to be installed, (2) analyzing component requirements relative to current and future demand for other central station conversion technologies, and (3) looking at the overall demand on other industries by the electrical utility sector as revealed by an input-output model of the U.S. economy. The combination of these approaches will serve to indicate what major industrial demands will be stimulated by the program.

As a basis for these analyses, a projection of central station installed capacity of LMFBFRs of 1500 MWe in 1987, 28,000 MWe in 1990, and 400,000 MWe in 2000 has been used.²⁰

5.3.2.1 Materials Requirements

The materials required for these plants have been estimated on the basis of data from LWR projects currently under evaluation by the AEC,^{21,22} as well as data from the LMFBR conceptual design as developed by Atomics International²³ and from the Central Station Power Plants Investment Cost Study by United Engineers and Constructors, Inc.²⁴ More recent material requirement estimates have been provided by Oak Ridge National Laboratory.²⁵

First, the impact on the uranium mining industry implied by the LMFBR program is considered. A general consensus is that LWRs will make up about 30% of the country's electrical generating capacity in 1985,²⁶ as compared with about 7% for 1974.²⁷ The annual requirements for uranium will increase rapidly during the 1980's to fuel a growing LWR economy but will level off as breeder reactors are introduced and will decrease during the 21st century as breeder reactors assume the nuclear power growth load. The stockpiles of depleted uranium tails at uranium enrichment plants can supply uranium for a growing breeder industry for many decades. Without a doubt, the breeder will reduce the demand for uranium; the extent of the effect on the uranium mining and milling industry is difficult to predict. The effect will be determined by the date and rate of introduction of breeders, the price of uranium ore, the cost of uranium enrichment, and other speculative factors. Considerable concern about the difficulty of increasing U.S. uranium mining and milling capacity sufficiently over the next few years, to accommodate the growth of LWRs, has been expressed recently.²⁸ Some attention should also be given to long-term management of uranium extraction to minimize the effect of a demand that may peak sharply. These effects could include both unemployment in mining areas and an increase in uranium prices caused by the industry's need to write off investments at an accelerated rate.

The possible adverse impacts due to a short-term demand for uranium that could, indirectly, be attributed to the breeder would appear to be small when compared with the benefits associated with lowering the long-term demand for uranium. A reduced demand for uranium means that the environmental impact of ore extraction to fuel a nuclear economy will be kept within reasonable bounds.

Major direct material input requirements for LMFBRs include pure or chemically combined forms of aluminum, concrete, copper, lead, magnesium (used as insulation), sodium, stainless steel, and carbon steel. Table 5.3-2 compares the estimated

quantities of these materials required for the LMFBR on an annual basis with total demand in 1970 and with total United States consumption projected to the year 2000. These projections indicate that the LMFBR Program will impose no severe additional demands on the major mining industries.

Industrial impacts related to the fuel cycle are discussed in Sections 4 and 10 and are generally comparable to those of the LWR, with regard to fuel fabrication and reprocessing facilities. The gradual replacement of LWRs with breeder reactors would, of course, appreciably reduce the need for new enrichment plants.

5.3.2.2 Component Requirements

Another way of gauging the impact of the LMFBR on other industries is to ask what sort of manufactured components will be required for the construction of LMFBR plants. This analysis is best made in terms of a comparison between alternative types of central station power plants. In Table 5.3-1 the annual capital requirements for breeders, thermal nuclear reactors, and fossil-fueled plants are shown for reference years between 1974 and 2020.

The largest difference in costs between types of plant components results from the replacement of the boiler in the fossil-fueled plant with the nuclear core and associated equipment in nuclear generating stations. Otherwise the differences are very slight. Among the nuclear options, the increased capital cost of the LMFBR is almost entirely accounted for by its more expensive core components.

The nature of LMFBR components, in terms of manufacturing tolerances, reliability, and other requirements, will also not be radically different from those required for LWRs and HTGRs. The nuclear industry does, in general, require a high degree of sophistication in the design and manufacture of components. Stringent performance requirements would be imposed on components by the necessity to safeguard the huge investments in nuclear power plants as well as by operational safety controls. Since the inception of the atomic energy program, adherence to the highest performance and safety standards has been required for nuclear power plants to protect the general public. These requirements have two aspects. First, standards and criteria must be established and administered to assure that the public is, indeed, protected. These standards involve very basic engineering and safety concerns that are the subject of other sections. Second, the existence of

Table 5.3-2

MATERIALS REQUIREMENTS FOR LMFBR RELATIVE TO TOTAL U.S. DEMAND

Material	Reference 1000-MWe LMFBR (short tons)	Domestic U. S. Annual Consumption ^a (thousand tons/year)		Projected Year 2000 Demand for LMFBRs	
		1970	2000 (estimated)	Thousand tons/year	Percent of Total U.S. Demand
Aluminum	20	4,500	33,000	1.4	<0.01
Carbon Steel	36,000	117,000	188,000	2,500	1
Concrete	196,000	570,000	2,500,000	14,000	0.5
Copper	765	3,000	12,000	54	0.5
Magnesium	1,000	2,400	45,000	71	0.5
Lead	51	1,400	3,000	4	0.2
Metallic sodium	2,200	171	593	154	25
Stainless steel	2,400	1,300	4,400 ^b	166	3.8
Uranium	60	7,000	130,000	0 ^c	0 ^c

^aSources: U.S. Department of Interior, "Minerals Yearbook-1970," vol. I, 1972; U.S. Department of Interior, "Mineral Facts and Problems," 1970; U.S. Department of Commerce, "U.S. Statistical Abstract-1972," 1972; U.S. Atomic Energy Commission, "Nuclear Power 1973-2000," USAEC Report WASH-1139(72), Office of Planning and Analysis, 1972.

^bStainless steel use projected at 2.5% increase per year (proportional to GNP growth).

^cIn year 2000 uranium for LMFBRs will come from depleted tails stockpiles at enrichment plants.

such stringent requirements and their wide incorporation into industrial practice will have a net beneficial impact on U.S. industrial sophistication and capability. Widespread adherence to these high performance standards may be a significant second-order benefit of the LMFB program.

An example of the industrial "spin-off" that might be expected from the LMFB program would be the development of a new high-temperature sodium technology. The development of piping, pumps, valves, and heat exchangers for handling high-temperature sodium could lead to the development of other industrial applications that might exploit the excellent heat transfer properties of this material.

5.3.2.3 Impact on Other Industrial Sectors

A third way of estimating the impact of the LMFB program on other industries is a partial check on the previous two methods. In the immediately preceding sections, the direct material and component inputs into the construction of LMFBs have been analyzed. One might also ask whether there are other industrial sectors particularly affected by changes in the utility sector as it is now constituted. Economic input-output analysis allows one to identify the industries that are affected both directly and indirectly by changes in the output of a specific industry, in this case the utility industry.

On the basis of 1963 data,⁹ with the U.S. economy divided into 99 sectors, there is no sector, other than those involved in producing and transporting fuel, in which direct and indirect sales to utilities exceed 2%. Thus, even if installation of the LMFB radically altered the kinds of purchases made by utilities (which is unlikely given the composition of capital expenditures for the LMFB and their similarity to the LWR), it would not, generally, have a concentrated impact on any other economic sector. The exception to this conclusion is, of course, those industries directly involved in fuel extraction and transportation.

5.3.2.4 Indirect Research and Development Requirements

Although the direct requirements for research and development for the LMFB are discussed elsewhere in this report, it is useful to ask here whether there are indirect research requirements implied by the breeder program. A number of such requirements can, in fact, be identified. Although in the following discussion they are expressed in terms of research requirements, they have clear implications relative to eventual industrial production.

The LMFBR is the most prominent of a number of new energy technologies that, of necessity, operate through the electrical sector. The increasing shortage of fuels appropriate for direct burning and the improving relative economics of electricity generation will lead to an increased fraction of final energy use being supplied by electricity. This increased electrification emphasizes the need for research on devices that utilize electricity--from more efficient appliances to electric automobiles. Traditionally, research on utilizing devices has been slighted in favor of research on fuel extraction and large-scale conversion devices.

A second research area whose importance is increased by the LMFBR program is electrical transmission. This facet is a function of the increased electrification discussed above and of the potential clustering of power plants. Energy centers are discussed in Sections 7.2.6.4, 7.4.9, and 4.5.6.3 of this Statement. Current projections of the land area required for electrical transmission and the need to transmit even larger blocks of power emphasize the desirability of developing high capacity transmission facilities that can be placed underground, if possible.

A third research and development requirement is for large-scale energy storage technology. The LMFBR is obviously a baseload generating technology, as are many of the other new electrical generating technologies. There is a need to develop technologies that can allow capital-intensive generating capacity to operate continuously and can at the same time supply a time-varying demand. This problem is exacerbated by the fact that modern baseload plants (LWR and high-pressure fossil-fueled plants) are not adaptable to shifting to intermediate load operation as they "grow old," as was the traditional pattern. New energy storage technologies are also required to accompany intermittent electrical energy sources such as solar and wind systems.

5.3.3 Regional and Local Economic Impacts

Since a single 1000-MWe LMFBR power plant and associated equipment will cost about half a billion dollars, it will have a significant impact on the local community and region in which it is built. Furthermore, since urban centers are currently excluded as sites for LMFBRs, breeder installations will probably be built in locations that, in general, are not characterized by intensive capital investment or construction activity. The total number of such plants projected for construction during the last decade of this century and the first two decades of the 21st century indicates that the aggregate of these local effects will be significant during that period.

The impact on local communities of the six- to eight-year construction cycle of the LMFBR will depend critically on the size of the community, the number of transient vs resident employees in the LMFBR labor force, the existing supply of housing, and retail trade and service establishments available. The evaluation of these local effects and the government or private actions required to mitigate problems will necessarily differ with each project and must be evaluated on a case-by-case basis. As the LMFBR program develops, however, significant effort must be devoted to the understanding of the general nature of such effects, so that the procedures and mechanisms for their alleviation can be established before problems arise.

A significant mechanism for evaluating regional and local impact is through state and local fiscal effects. These effects fall into two categories. The first is the impact on tax revenues, both directly in terms of additional income and sales and indirectly in the increase of the base of assessed valuation for property tax purposes. The second, the expenditure category, is determined by the required additional public services resulting directly and indirectly from the project. These services include roads, utilities, schools, hospitals, and police and fire protection. Such an analysis can be carried out for specific plants and, in fact, has meaning only for specific situations.

One obvious generalization can be made on the program level regarding fiscal impact. Implementation of the program would result in the construction of a large number of plants, each requiring six to eight years for completion, with major new service requirements generated for local communities from the beginning of construction. Tax revenues from the plants would not be received until a considerable time after local government services would be needed. The magnitude and seriousness of this problem in timing should be evaluated for possible government action.

A trend already apparent favors the siting of several nuclear power plants in a single, large exclusion area. A transition from clustered sites for nuclear power plants to nuclear energy centers^{29,30} may occur, and aspects of the potential transition to nuclear energy centers are discussed in Sections 7.2.6.4, 7.4.9, and 4.5.6.3 as well as in Section 5.4.1.

5.4 SOCIOPOLITICAL IMPACTS

5.4.1 The Regional Dimension

The appropriate economic size of power plant facilities and the impact of such facilities on the environment have no relationship to state boundaries. The importance of

the regional dimension in formulating plans for land and resource use increases as the scale of power generation increases. There are incentives for siting a number of LMFBRs at a single location to serve a multi-state area. Unfortunately, the geographical scale of the Nation's political institutions does not match the scale of its energy planning requirements. The importance of the regional scale in energy planning was emphasized in a study by the Office of Science and Technology some years ago.³¹ Expectation of large-scale commercial use of LMFBRs accentuates the need to identify the legal and institutional changes that may be necessary to provide for regional energy development in the most socially acceptable way.

Comments* were received concerning the possible disruptive effect caused by the LMFBR on both the quality of life in rural areas as well as on the lives of the large numbers of construction workers and their families temporarily living at rural construction sites.

The vision of a rural or semi-rural area being chosen as the site for a nuclear park that would perhaps include one or more LMFBRs does not appeal to me. Nor do I think it will appeal to many people who live in such areas. -- I wonder what the farmers and other inhabitants of rural and semi-rural America are going to think about the nuclear park concept?

In the future, energy centers containing one or more LMFBRs may possibly be located in rural or semi-rural areas. Such centers would inevitably change lifestyles in nearby communities. However, if one agrees to the need for a fixed amount of additional generating capacity, the question becomes one of assessing whether the concentration of disturbance in one rural or semi-rural area is preferable to its dispersion over several areas with, however, a greater cumulative land disturbance. Minimum land disturbance for the production of a fixed quantity of electrical energy occurs in nuclear power systems. Land use is compared in Table 9.1-11 for various well-defined 1000-MWe energy conversion systems (coal, oil, gas, LWR, LMFBR, HTGR). Minimum land disturbance (about one square mile) would occur in the 1000-MWe LMFBR system; this disturbance would be even less on a per megawatt basis for the case of a nuclear energy center. The commenter* quoted above advocated the development of solar energy to meet the Nation's needs. If one were to follow this course, land disturbance for solar energy systems that would generate a quantity of electrical energy equivalent to the generation of a 1000-MWe nuclear station would be 10 to 30 square miles, depending on whether thermal or photovoltaic conversion systems are used. (See Sections 6A.5.3.1 and 6A.5.3.2.)

*Hearings Transcript, Testimony of Mrs. Pat Hoffman, p. 193.

Additional comments* concern the effect of the LMFBFR construction program on the lives of construction workers:

The majority of workers to be employed under the LMFBFR program are construction workers. These workers face the prospect of having to move their families to a semi-rural area for a five year construction period, and then when the project is completed, move back to the city or to another power project. (Because the LMFBFRs are to be located in non-urban areas, there are few alternative employers to take up the slack when the project is finished.)

To the construction worker with a growing family in need of schools and services, the decision to sign onto a LMFBFR project will indeed represent an "irreversible commitment of resources" in terms of his career...

In the past, certain types of construction work have been recognized as mobile occupations and have tended to attract people who prefer this lifestyle. Many construction projects, such as building power plants of any kind, laying pipelines erecting transmission lines, will always require a non-stationary labor force. However, the increasing trend toward locating several power plants at a single location promises to alleviate the type of problem suggested by this comment. As an extension of the "multiple siting" concept, nuclear energy centers would offer opportunities for stability for the construction labor force. Workers could adopt a more settled lifestyle, and obtain advantages such as permanent housing and better schools, if they chose to do so. There would be a phased construction activity associated with the sequential addition of power generating stations at an energy center in addition to the construction labor requirements for the anticipated adjacent industrial development. Opportunities would also exist for construction workers to become production and operation workers at the industrial and power plant facilities.

5.4.2 The National Dimension

The unique characteristics of nuclear power---its vast promise and its difficulties-- have been clear from its inception. As it became a viable energy source, the need for government regulation to ensure public health and safety was recognized. Such regulation was readily accepted because of the recognition of the unique characteristics of nuclear power and because of the origins of nuclear energy in a military context. A comprehensive and effective regulatory review procedure now exists for the construction and operation of nuclear facilities.

*Comment Letter 65, pp. 2-3, (R. A. Karasek, Mass. Inst. of Technology).

The other sections and volumes of this report address the "difficulties" of nuclear power from a technical standpoint. In this section, a different perspective is taken, and questions related to the sociopolitical implications of nuclear power systems are explored. First, the safeguards problem is discussed; then the waste storage problem is analyzed; and finally some implications for the political system are considered.

Many comments expressed concern or raised questions about the sociopolitical effects of safeguards. Responses to these comments may be found in two subsections below (Sections 5.4.2.1 and 5.4.2.3). The comments include:

...the extent to which the existence of the LMFBR program, which cannot of course be divorced from other civilian fission programs, will impose the necessity for behavioral, social, and other controls is among the most profound and grave of all of the many questions surrounding the development of the LMFBR Program. What are the changes, not in technology, but in our very social institutions that may be dictated by a decision to proceed with the LMFBR Program?...*

Having raised the issue of less political freedom with a Pu economy, some discussion of implications should be included.... Are such dangers as Pu weapons manufacture and the operation of a Pu black market so credible that discussing them with only unclassified, available information will lessen our security?...**

Intolerance of minorities, pervasive domestic intelligence, centralization, technological priesthoods, human engineering, specialization, inequity, epidemic paranoia are dilemmas already stressing the institutions of a disoriented society. Nuclear technology will affect these dilemmas, and if the effects compound the stresses to some ultimate threshold, then the apocalyptic means of resolution will bear no resemblance to the arid bureaucratic cant that appears in the draft statement.+

the secondary impacts of the safeguards program include...full discussion of: (1) the threat to personal liberty and individual privacy that would occur incident to nuclear theft or sabotage and attempts to recover illicitly obtained nuclear materials; and (2) the threat to domestic and international political order and governmental processes that would result from the successful diversion of fissile material.++

Sabotage is a terrifying prospect where we may be creating a quasi-military-private army system under government directives, which may be impossible to totally monitor.§

*Comment Letter 20, pp. 15-16 (D. E. Abrahamson, Natural Resources Defense Council).

**Comment Letter 22, p. 8 (D. Tegtmeir, Mid-America Coalition for Energy Alternatives).

+Comment Letter 27, Attachment #3, p. 2 (D. P. Geesaman, Univ. of Minn.).

++Comment Letter 38f, p. 2 (J. G. Speth, Natural Resources Defense Council).

§Hearings Transcript, Testimony of Dr. J. W. Stryker, p. 129.

The commenters quoted above have expressed fears and apprehensions about the implications of nuclear power. Issues such as these are not amenable to quantitative resolution. One can hope to place them into a manageable perspective so that, where possible, problems can be identified, addressed, and eventually resolved.

5.4.2.1 Nuclear Safeguards

All energy systems, except perhaps highly decentralized ones, are potential concentrated targets for widespread social damage. Fission reactor systems add another dimension to this potentiality in that they create the means, in theory at least, of an entirely new form of antisocial behavior. A major difference between an electrical energy system based on nuclear fission and one based on alternative systems derives from the presence of plutonium in a number of industrial locations in the former case. The quantities of plutonium involved will be more substantial in the case of LMFBRs than for LWRs. The possibility of the use of stolen plutonium in a nuclear weapon is perhaps less likely than its use, or the threat of its use, in more mundane ways--dispersal in water or in air by use of chemical explosions, for example. These possibilities are discussed in Section 7.4.3. The need to protect against misuse of weapons and dangerous chemicals has existed in the past, of course, but plutonium is a special case because of its high toxicity per unit weight and because large quantities will be in commercial circulation.

The programs necessary to assure adequate safeguards against diversion of nuclear materials are discussed in Section 7.4. Independent analysis has indicated that adequate control can be accomplished at costs that correspond to a small portion of the total cost of nuclear power.³² (Also refer to Section 7.4.9.) This section discusses what the social costs of these measures may be. In particular, three types of effects will be discussed: (1) effects on employees and potential employees of nuclear industries, (2) effects on the general population during normal periods arising from implementation of safeguards, and (3) effects on the public during recovery operations that would follow theft of nuclear materials.

The AEC has established safeguards regulations that must be complied with not only by the electric utilities but also by the industrial sector engaged in the storage, processing, or transportation of nuclear materials. These regulations are rigorous, but their impact on employees is still comparatively unobtrusive except in two respects. Armed guards restrict access to particular areas and personnel must submit to search upon entering or leaving "vital or material access" areas. However, instruments exist which could reduce the need for personal search. Similar procedures are common in other industries handling sensitive materials. (See Section

7.4.7 for a discussion of safeguards regulations.) In addition, companies must file various operational reports and adhere to structured nuclear materials accountability requirements. The regulations will apply to companies that, in the past, may not have experienced comparably intensive interaction with government.

Only authorized individuals, perhaps only those with security clearances,³³ will be permitted entry to nuclear materials storage areas. This imposition of security controls will be an increase in the role of the Federal Government in the private industrial sector. The significance of this measure will be minimal because of the very small fraction of the labor force involved. Nevertheless, as the LMFB program progresses and as it affects a larger segment of industry, there will be restrictions on employment opportunities in these operations. It must be reiterated that people engaged in nuclear activities do so voluntarily and, in addition, some form of background check is currently accepted as a prerequisite for employment in a large number of positions. For 30 years, the AEC has implemented a security program for employees in the nuclear industry who have access to restricted data without committing violations of their basic civil liberties or privacy. Finally, the purpose of the clearance program is to assure the public that reliable persons are in fact employed in nuclear industries.

Many comments* have been received concerning the security force that would be required to safeguard special nuclear material. Some typical comments include:

The Atomic Energy Commission may be forced into the politically undesirable position of a powerful and autonomous policeman. Security arrangements necessitated by a concern for public safety could create a police state.**

... Means of safeguarding against diversion through increased preventive police activity poses its own problems. Increased surveillance (both direct and electronic), infiltration, and other techniques to reduce the probability of criminal action might be carried out by the FBI, the CIA or a special nuclear police force. ---there are both monetary and social costs involved.+

With large quantities of plutonium around, it will be a matter of survival to increase security and military protection to keep it from getting into the wrong hands.++

*Comment Letter 38, p. 6 (J. G. Speth, NRDC); 22, p. 8 (D. Tegtmeier, Mid-America Coalition for Energy Alternatives).

**Comment Letter 40, p. 3 (Environmental Action of Colorado).

+Comment Letter 51, pp. 46-47 (M. F. Baldwin, Environmental Impact Assessment Project).

++Comment Letter 8, Attachment 1, p. 7 (G. and I. Younghein).

The security precautions required to prevent theft of nuclear materials will normally have very little public impact. They will, after all, be in effect at relatively few locations in very specialized industries. Under normal conditions, their effect on personal freedom and liberties will be minimal. The most visually obvious effects will be the presence of armed guards required during the transport of plutonium fuel among the reprocessing facility, fuel fabrication plant, and reactor and possibly during transport of high level wastes. The function of these armed guards is limited to safeguarding during transit valuable material in sealed containers. In the case of nuclear fuel, shipping containers are very heavy and their seals are inspected before and after shipment. Independent sampling and analyses are performed for more concentrated and purer materials, potentially more attractive to diverters, to validate the contents and verify material balances. An estimate of the size of the security force required to safeguard these shipments can be made by referring to Table 4.5-1 of this Statement. For one year's operation of one 1000-MWe reactor and its associated fuel cycle facilities, the following shipments of hazardous material will take place:

Unirradiated material containing PuO_2	20 shipments/year
Irradiated fuel containing UO_2 , PuO_2 , and fission products	27 shipments/year
Cladding hulls and high level wastes from reprocessing plants	2 shipments/year

Assuming that none of these shipments will take place within an energy center, that each shipment will be of three days duration, and that there will be 400 LMFBRs (each 1000-MWe in size) operating in the year 2000, one can calculate that there will be about 200 shipments in transit at any time during the year 2000. Assuming that four guards will be required per shipment and that there will be three shifts per day, safeguarding the shipment of hazardous material for LMFBRs can be accomplished by a security force of a few thousand carefully selected and well-trained individuals. A similar estimate of the size of the required security force is shown in Table 7.4-13. Again, the numbers involved and the tenuous nature of the effects do not justify exaggerated concern.

Safeguards activities need not be concealed, and, in fact, a fairly obvious, highly visible security activity could well increase public confidence in safeguards and, at the same time, reduce the probability of theft.³⁴

Security forces are generally required when material or facilities are of great monetary value or where misuse may endanger the public. Considerable experience exists today with security forces that have routinely performed such functions; for example, in protecting against industrial sabotage, against theft of money and jewels, and against hijacking of aircraft. Such forces can be either civilian or governmental. If, indeed, a quasi-independent nuclear material security force is established, its responsibilities will have to be circumscribed to assure that the force does not assume illegal authority.* The responsibilities that could be entrusted to this force might be significant but are comparable to the responsibilities routinely entrusted to police or members of the armed forces. Thus, a security force to safeguard shipments of special nuclear materials can be expected to function reliably and not pose a threat to society.

How do these considerations change once a theft has occurred? Concern has been expressed that a successful diversion could be used as an excuse for the repression of personal liberty and individual privacy.** The nature of any credible diversion activity, however, would inherently limit this sort of impact, because the fraction of society that might conceivably be interested in, or capable of, stealing nuclear material is very small. Thus, using nuclear diversion as an excuse for direct widespread suppression of liberty or privacy is not credible.

Comment letters of a highly speculative nature have been received concerning the possible consequences of a successful diversion. They include:

A threat to domestic and international political order and governmental processes would result from the successful diversion of fissile material.+

After a successful or even unsuccessful attempt at plutonium diversion, how will the relationship between small and large blocks of societal power be altered?++

The responses that follow are, like the comments themselves, highly speculative. There is no inference that the situations alluded to in the following responses will prove to be realistic.

Because of a recognition of the magnitude of the possible consequences of successful theft, the thrust of the Commission's safeguards program is to prevent diversion and, in the case of successful diversion, to recover rapidly any stolen material.

*Hearings Transcript, Testimony of Dr. J. W. Stryker, p. 129.

**Comment Letters 8, p. 2 (G. and I. Younghein); 20, p. 2 (D. E. Abrahamson, NRDC).

+Comment Letter 38, p. 6 (J. G. Speth, NRDC).

++Comment Letter 27, Attachment 3, p. 2 (D. P. Geesaman, Univ. of Minnesota).

The increasing incidence of terrorism is the basic reason for applying more strict measures for safeguarding strategic nuclear materials. There is a wide range of conceivable terrorist activities that might include either the threat or actual use of diverted material. Materials control and accountability systems are used to verify diversions, and detection devices can be employed to help recover nuclear material. The response of the authorities to terrorist activities would depend on the nature of the threat and would be graded to the assessed level of public risk. Because the number of people potentially interested in and able to divert nuclear material is small, and because there are numerous safeguards in force to prevent diversion, successful diversions are expected to occur rarely, if ever. Material recovery activities following a serious threat might conceivably be obtrusive and stringent. Such events are likely to be localized and of short duration, and recovery activities would be taken in the public interest.

Concern about permanent obliteration of freedoms due to the need for organized emergency action in infrequent emergency situations would not seem to be a realistic sociopolitical possibility. In the past, there have been immediate and vigorous public reactions when basic freedoms were threatened. The fears of permanent adverse sociopolitical impacts of infrequent emergency measures to recover stolen plutonium do not appear to be well-founded.

One commenter attempted to draw a comparison between efforts to control drug traffic and efforts to control traffic in diverted plutonium:

For example, to control traffic in heroin, there was a request for no-knock search laws. Is this an important freedom that will have to be abandoned once a significant plutonium theft has occurred?*

The AEC does not believe that this comparison is valid for a number of reasons. For example, the availability of heroin from poppies cannot be compared with the availability of plutonium from safeguarded facilities and vehicles. Moreover, because of its toxicity, special precautions must be taken in handling plutonium. The comparison will be pursued, however, despite its tenuous validity.

Drug traffic usually involves many people widely dispersed in society with a strong or even compulsive desire for the commodity. There is a widespread market for consumer use of heroin. Therefore, a request for certain control measures, which may intrude upon privacy or infringe on personal liberties, might conceivably be

*Comment Letter 38f, p. 6, (J. G. Speth, NRDC).

supported on the grounds that these measures are necessary for successful control of heroin traffic. This issue has been an important one in contemporary society where police activities to control drug traffic have been monitored and restricted by the courts. Adverse effects on personal liberties that could be attributed to recovery efforts following a hypothetical successful theft of plutonium cannot be compared with the social problems posed by narcotics control. A diversion of plutonium would be a localized event involving a small number of people. Therefore, widespread and prolonged application of repressive control measures is unlikely either to be sought or, if sought, to prove effective.

Concern* has been expressed that, should special nuclear material be successfully diverted, a black market in such material (with very serious secondary impacts) would rapidly develop.

The Commission believes that the application of stringent safeguards procedures to prevent diversion in the first place is the best means to preclude effectively the growth of such a black market. Procedures to prevent theft are discussed in Section 7.4. For reasons similar to those given in the preceding paragraphs concerning the nature of comparisons between traffic in heroin and diverted plutonium, a hypothetical black market in plutonium cannot be compared with illegal commerce in other commodities.

The general conclusion is that although the indirect impacts of safeguards require serious attention, they are manageable. It should be emphasized that only the indirect impacts are being discussed here. The discussion is not meant to make any inference concerning the very serious direct consequences of a diversion of nuclear materials and of safeguards against diversion. (See Section 7.4 for a discussion of the direct issues.)

5.4.2.2 Radioactive Waste Management

The disposal of high-level radioactive wastes, with their plutonium and other trans-uranic nuclide content, raises quite different issues relative to second-order impacts. The questions here revolve around the very long time periods that must elapse before such wastes become, of themselves, innocuous to man and to other living species. For decay to innocuous levels, the direct radioactive products of fission impose a time scale of centuries. The higher weight elements created by

*Comment Letters 22, p. 13, (D. Tegtmeier, Mid-America Coalition for Energy Alternatives), 51, p. 47, (M. F. Baldwin, Environmental Impact Assessment Project).

neutron interactions with uranium in the nuclear core (particularly Pu-239, with a half-life of 24,360 years) impose a time scale much longer than man's recorded history. Because of the serious consequences of improper handling of radioactive wastes and the time scale of concern, the Federal Government must play a strong role in the management and ultimate disposal aspects of nuclear wastes. In this connection, there is an obvious societal question of whether or not a possibly short-term use of fission energy by current and near-term future generations of man could impose an unprecedented caretaker burden on future generations for periods beyond man's experience.

The answer to this question depends on what is ultimately done to remove or isolate the long-lived, radiologically hazardous by-products of fission energy from man's environment. As discussed in Sections 4.6.1.1 and 7.3.5, the need and obligation of contemporary society to seek ways to dispose of nuclear wastes safely and permanently is clearly recognized. A number of prospective methods of accomplishing this purpose are under active study and consideration. The use of surface facilities for the storage of solidified high level wastes is considered to be a temporary expedient that could be used for many decades into the future, if needed. However, it is definitely not a permanent solution, and the present development program has the goal of making permanent disposal measures available on a time scale of two or three decades.

This discussion assumes that stable geologic formations, may eventually be used for the isolation of nuclear wastes from the biosphere. This assumption seems to be reasonable, pending identification of methods ultimately to be used, since there are many places in the earth--accessible to contemporary man through his technologies--which have been isolated from groundwaters and from the atmosphere for hundreds of millions of years. To make any absolute predictions on the continued future isolation of such formations is, of course, not possible, but the long geologic history of isolation gives promise, of the highest order, that such circumstances will continue far into the future.

As currently conceived, a geologic disposal waste repository would be located beneath Federally owned land. The first repository might be designed to receive and hold all high-level wastes generated by the commercial nuclear power industry through the year 2000. In broad concept, as solidified wastes were received, they would be imbedded in appropriately spaced arrays in the floors of pre-mined areas, a few thousand feet below the land surface. As prepared areas were filled with waste, they would be sealed by back-filling with previously mined material. The last solidified waste to

be buried in this assumed, initial repository would arrive in the year 2010. Sometime thereafter, all remaining tunnels and shafts would be backfilled, thus cutting off access to the deposited wastes except through re-mining or drilling.

Many important details have been omitted in this description, the purpose here being merely to invoke the image of a filled repository in order to give some thought to its impacts on future generations. The first obvious impact is that the fact of its existence would become a part of that body of knowledge that must be passed from one generation to another. In the future, persons who have occasion to learn, for example, about the geologic formations in the Permian Basin, a possible initial repository, would have to learn also about the existence of radioactive wastes imbedded in some 1500 acres beneath a Federal reservation at a depth of 1000 ft or so.

A second obvious impact would involve the permanent dedication of the overlying land to purposes that would preclude, or minimize the chance of, inadvertent intrusion of the deposit by mining or drilling activities. There are no obvious reasons why the surface land area could not be used for many purposes consistent with the no intrusion caveat. As a Federal reservation, such uses could be controlled, much as are National Park uses today. However, the time span for the radioactive decay of the transuranic nuclides is so vast that contemporary man can hardly be expected to foresee, let alone assure, that societal controls would be exercised over the necessary period.

The latter point is a matter of fairly general concern. It was frequently expressed in the comments received from reviewers of the Draft Statement.* While the matter was expressed in a variety of ways, the following, partially quoted statement probably catches the essence of the matter:

Fission technology requires that man issue guarantees on events far into the future, and it is not clear in most cases how this can be done.**

The general opinion is clearly representative of a high sense of social responsibility, but it is in some respects paradoxical and unrealistic. Even though there is

*Comment Letters 8, p. 6 (G. and I. Younghein); 16, p. 1 (J. T. Edsall, Harvard University); 22, p. 10 (D. Tegtmeier, Mid-America Coalition for Energy Alternatives); 32, p. 2 (R. W. Freedman); 38d, pp. 4-9, 21-22 (J. G. Speth, NRDC); 39, p. 2 (E. A. Pryzina, Minnesota Pollution Control Agency); 51, pp. 1, 4, 60, 62 (M. F. Baldwin, Environmental Impact Assessment Project); 58, p. 1 (C. R. Lewis); 60, pp. 2-3 (J. Huber).

**Comment Letter 38d, p. 9, quoting the Union of Concerned Scientists, The Nuclear Fuel Cycle, p. 9 (October 1973).

recognition, on the one hand, that the current generation of man is probably no more gifted than its predecessors in foreseeing events of the far future, there is also the feeling that contemporary society should abandon or avoid activities for which it cannot issue guarantees to all future generations that no future adverse effects will occur.

Among the many things that need to be considered along the way to making choices and committing to particular disposal methods, the various assumptions made about future generations should be high on the list. There is at least an undercurrent of thought, in some of the commentary received, that the burden on today's society is to protect future generations from their own inadvertent mistakes or from mistakes made as a result of possible future societal and institutional instabilities. Thus, for example, given a choice today between 1000-ft-deep bedded salt formations and five-mile-deep holes terminating in rock formations as places for depositing high level wastes, some people--perhaps many--would opt for the latter, because it seems to offer better protection against inadvertent intrusion in the future.

On the other hand, there are serious questions to be considered as to what options future generations might like to find left open to them. Given the assumption used in this discussion that wastes will be isolated in geologic formations, should the choice of geologic formation be guided by the assumption that accessibility to future generations should be foreclosed to the extent possible? Given the long time span in question, should credence be given to the possibility, however remote, that future circumstances--currently unforeseen--might make retrieval a matter of importance or even urgency? Should we assume that future generations may develop the means to make much more satisfactory disposition of the wastes than present-day society might choose and might wish to retrieve the wastes for this purpose? Should the possible value of the wastes after fission products will have decayed out, centuries hence, be a consideration?

The posing of these questions is not intended to reflect any current bias for a particular waste disposal method, but rather to suggest that complex societal, as well as technological, questions must be given serious consideration. The nature of the problem strongly recommends that present-day society give considerable additional thought and effort to the matter of permanent high-level waste disposal before choosing and permanently committing to a particular method. The history of the high-level waste management problem is traced in Section 4.6.1.1. The objectives of the AEC's current interim and long-term waste management programs are discussed

in Section 4.6.1.2 and particular attention is given in that Section to the many comments received on waste management policy.

The AEC's decision to build a retrievable surface storage facility will provide a safe means of holding wastes for many decades into the future and has the advantage of not foreclosing options that cannot be foreseen at this time. During those decades, there will be the opportunity to develop additional information and confidence in currently perceived options and perhaps to develop alternative approaches not currently under consideration.

Based on the information at hand, contemporary society may reasonably proceed on the assumption that it has the knowledge and the means to develop acceptable solutions to the problem of isolating nuclear wastes from the environment. No generation of man will ever be able to issue absolute guarantees on any activities whose impacts could extend into far distant future years any more than it can issue absolute guarantees on either the short-term or long-term effects of the dispersal into the atmosphere of non-radioactive materials whose biological hazards may not be as well understood as the effects of radiation. Society has no choice but to bear in mind its limitations and to act as wisely and as well as its best efforts will permit.

5.4.2.3 Federal Role in Energy Systems

The present energy situation has forced a broad energy role on Government. The complexity and extent of the energy situation is such that no body short of the Federal Government can orchestrate solutions on a national scale or cooperate in international arrangements to the same end regardless of the energy source. This growing role of the Federal Government can be taken no less seriously than its traditional responsibilities in fields such as interstate commerce, foreign relations, or national security. The Federal Government is now embarked on overall planning intended to assure an adequate energy supply for society within appropriate environmental and economic constraints.

To further these objectives, the recently created Federal Energy Administration (FEA)³⁵ has been charged with the responsibility to formulate a comprehensive national energy policy, as well as an integrated domestic and foreign energy resource policy. In this connection, the FEA has issued a study to determine how to optimize the use of domestic sources to satisfy the Nation's energy requirements. This study is called Project Independence Blueprint. The objective of the Project is the identification of the best options for incorporation into a national energy policy.

Again, consistent with a Federal role in the energy field, legislation has recently been enacted creating an Energy Research and Development Administration (ERDA).³⁶ It is envisioned as a complete energy research and development organization and will bring together into one agency major Federal programs of research and development for all forms of energy. Among its major efforts, it will continue the activities of the present Atomic Energy Commission in nuclear fusion and fission, expand fossil fuels research programs initiated by the Department of the Interior to capitalize on national reserves of coal and oil shale, and expand and upgrade research currently under the auspices of the National Science Foundation into making use of potentially important forms of energy, such as solar and geothermal sources. It will also move immediately into a substantial new research and development effort in energy conservation and utilization.

A new Nuclear Regulatory Commission (NRC)³⁷ will assume the licensing and regulatory responsibilities previously carried out under the Director of Regulation within the Atomic Energy Commission.

Included in the legislation specifying the responsibilities of ERDA and NRC is the authorization to establish a high-level interagency council concerned with energy policy, the Energy Resources Council (ERC).³⁸ It will be chaired by the Secretary of the Interior and will include 15 additional Cabinet members and Agency heads.

In connection with nuclear energy systems, Federal involvement in the operation of such systems is justified whenever there is a need to control and regulate elements of the energy system that pose potential dangers to the public. Energy fuels cannot travel unprotected when those materials can be of great danger to society. Waste management also presents unique problems of protecting the interests of present and future generations. Thus, the government, through research and development, regulations, and monitoring, must assure that the wider public good is served in the development of nuclear power.

Nuclear energy, of course, is not unique in creating new sociopolitical problems for society. The consequences of the growth of modern technology pose many similar dilemmas for society. Implications of the expanded Federal role in the operation of nuclear energy systems described above must be considered not only in the context of the expanded Federal role in the energy field but also in the context of similar forces created by the introduction into society of other new technologies. Many of these technologies impinge more directly on the public than the choice of an energy conversion system. The computer, for example, could be a powerful

instrument for the repression of personal liberty. Under many forms of government, especially authoritarian systems, the development of the computer would, at best, be of questionable social value. In a democracy, two considerations should guide the development of the computer. Its great potential for social benefits, through replacement of dehumanizing, uninteresting work, and vast improvement of mankind's productivity must be balanced against its potential for misuse. Nuclear energy systems, like computer systems, are among the new technologies that may pose new and unfamiliar social problems. The important fact is that basic to the American governmental system is the will to devise institutional mechanisms for assuring adequate safeguards against abuse of new technologies. The protection of individual rights and freedoms from a novel threat may not happen automatically, but it can be devised, if the public and its representatives are made aware of the nature of the threat and there is an appropriate response.

5.4.3 The International Dimension

5.4.3.1 Nuclear Proliferation

One of the most profound problems of modern times is how to limit the proliferation of nuclear weapons and other nuclear explosive devices. Since its establishment in 1957, the International Atomic Energy Agency (IAEA) has had a responsibility "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world"³⁹ and at the same time to administer a system of international safeguards to prevent the diversion of material suitable for weapons production from peaceful applications. However, technological control of nuclear materials in the absence of a genuine political commitment by the countries concerned to prohibit or limit the production of nuclear weapons would be ineffective. Accordingly, 83 countries have ratified the 1970 Nuclear Non-Proliferation Treaty. Recognizing that safeguarding hazardous nuclear material is a multinational problem, the treaty itself establishes the framework within which international safeguards, administered by the IAEA, are to operate. The two fundamental principles of IAEA safeguards are the prevention of the diversion of nuclear material by early detection and the minimization of interference so as not to impede peaceful advances.

A carefully defined IAEA safeguards procedure now exists, with provisions for design review, specification of material balance areas and key measurement points within facilities, and onsite inspections.

The concerns of IAEA safeguards are: (1) to deter national-level diversion of safeguarded nuclear material by the governments of the countries involved; (2) to place a responsibility on those governments to institute domestic programs that

would deter sub-national level diversion of safeguarded nuclear material (i.e., by individuals or groups acting contrary to Government policy); (3) to reduce specific international tensions by providing a degree of assurance among countries, especially hostile neighbors, that the safeguarded country is not developing nuclear weapons; and (4) more generally, to create an environment of confidence so that to the extent safeguards are being applied, international nuclear commerce to exploit the peaceful benefits of nuclear energy can be freely conducted without contributing to insecurity and tension among nations.

A number of comments* on the Draft Statement exhibited concern about the efficacy of IAEA safeguards. They include:

Regarding the international implications of a failure of the systems designed to "prevent the diversion of material suitable for weapons production from peaceful applications," we fully appreciate that there are developments in addition to the proliferation of LMFBR's that could lead to nuclear weapons capability being extended to virtually any nation-state.

The necessity to prevent the diversion of special nuclear materials both by nation-states and by sub-national groups is one of the most vexing of the environmental and social problems associated with the Liquid Metal Fast Breeder Reactor.**

There are really two issues here. One is the possibility of clandestine, illegitimate elements in a country diverting special nuclear material for use in a weapon or other device. The other issue is that of legitimate national governments gaining the capability of producing nuclear weapons. In the former case, the considerations discussed above relative to national safeguards apply to the international sphere (see Section 7.4.4.2). In cases where the U.S. exports LMFBRs, the importing national government will clearly have an interest in guaranteeing that no clandestine diversion is taking place. Furthermore, as a condition of sale, the U.S. will, as it does now with LWRs, impose safeguards requirements that are to be administered by the IAEA. Note that such an arrangement can be carried out even if the importing country is not a party to the Non-Proliferation Treaty.

The second case, the proliferation of "legitimate" nuclear weapons, is an entirely different matter. There is no reason to believe that the U.S. could significantly reduce the proliferation of nuclear weapons by abandoning its nuclear breeder program. Any country with even a medium-scale power industry based on any current

*Comment Letters 24, p. 8, (J. Legakes); 20, pp. 4, 5, 7, (D. E. Abrahamson, NRDC).

**Comment Letter 20, p. 23, (D. E. Abrahamson, NRDC).

fission system can produce nuclear weapons. The world's current nuclear arsenal, after all, was built up without the "benefit" of breeder reactors. Non-nuclear weapons states who are parties to the Non-Proliferation Treaty have agreed not to develop nuclear weapons. Such countries may be expected to continue as signatories only as long as their national interests are not adversely affected. Adherence to the treaty itself is, of course, voluntary with countries that have not signed it, such as India, China, France, and Israel.

To keep the potential impact of LMFBRs on proliferation in perspective, recognition must be given that, in the case of countries that will have LMFBRs, there may be other sources of special nuclear material (e.g., LWRs, HTGRs, or research reactors above 10 Mwt that utilize natural or slightly enriched uranium).

If a country has one or more LWRs available, it could use the plutonium produced in them for its weapons program. As with LMFBR fuel, this material would contain an undesirably high percentage of Pu-240, Pu-241, and Pu-242 resulting in a possibly low efficiency weapon. Discharged fuel from both LWRs and LMFBRs is highly radioactive and would have to be reprocessed to isolate plutonium. In the case of an LWR using recycled plutonium, as in the case of an LMFBR, there is plutonium in fresh fuel elements. Plutonium can be chemically separated from uranium and other material prior to irradiation with far less hazard than it can after irradiation. Hence, in the case of a country with an LWR facility with or without plutonium recycle and a reprocessing capability, an LMFBR would, at most, marginally increase the country's ability to acquire nuclear weapons.

Another related and important point is that many countries, especially those with advanced nuclear programs and good industrial bases, could acquire relatively pure Pu-239 and develop nuclear weapons on their own within a relatively short time. If such a country did not have unsafeguarded special nuclear material available and wanted to avoid breaking a safeguards agreement or related commitments, or if it simply wanted a source of relatively pure Pu-239, it could build a natural uranium reactor. Natural uranium is not difficult to obtain, and although under the treaty and the policies of many governments such material is subject to safeguards, there are many sources of natural uranium in the world. Moreover, if the goal is to build a crude nuclear weapon, the technology for processing this plutonium is readily available (see Section 7).

Breeder reactors may, nonetheless, make weapons production somewhat easier for a national government. Accepting that fact, the existence or non-existence of the

U.S. program will have little impact on the international availability of LMFBRs. As shown in Section 2, breeder programs in England, France, Germany, and the USSR are at least as far advanced as the U.S. program. The first breeders to be marketed internationally will probably be built in one or more of those countries.

5.5 GENERAL CONCLUSIONS

In the first part of this section, two purposes were identified for an analysis of the secondary impacts of a new technology such as the LMFBR. First it could contribute to understanding how society might accommodate the technology. Second, by anticipating the potential secondary social and economic impacts of the LMFBR, it could minimize the social cost and increase the social benefits of the technology.

As has been evident in much of the discussion above, many of the most serious issues entail value judgments as well as large uncertainties and complex technical factors. One cannot expect these issues to be dealt with in the black-and-white, dollars-and-cents vocabulary of cost-benefit analyses.⁴⁰ As the consideration of these issues cannot be black and white, so one cannot expect black-or-white conclusions. Unsatisfactory as this fact may be to those who want to prove acceptability or non-acceptability of a technology, no such proof is possible with regard to many of the issues dealt with in this section. Many of these issues, such as the question of safeguards and that of waste storage, have direct societal implications and should be the subject of continuing public discussion. By that discussion itself, society can contribute to its own protection and, at the same time, assure that its values are reflected in the evolution of the program. In its most general aspects, the intent of this document is to provide sustenance for that process of public discourse.

Although many of the issues discussed in this section cannot, by their nature, be resolved here, some measure of their total impact can be gained by considering the second-order costs and benefits along with the direct impacts and alternatives discussed in the rest of the Statement. On the benefit side, one is dealing with the social benefits of having available an abundant supply of electricity that is immune to cost increases caused by depletion of finite energy resources. On the cost side, one is dealing with implications similar to those encountered in a growing nuclear economy. The commitment of future generations to the management of radioactive wastes, the social implications of measures taken to safeguard plutonium, and the possibility of misuse of plutonium are problems that must be successfully resolved as the program develops if the LMFBR is to be a viable energy system option.

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Appendix II.A

PROBABILITY OF TRANSPORTATION ACCIDENTS

Protection of the public, of transportation workers, and of nuclear energy workers has been the overriding factor in the development of both international and national regulations that govern the shipping of radioactive material. Protection is achieved by limiting the contents of each package and by package design. Both the contents (if sufficient to cause a public hazard) and the package design must be approved by the AEC before the package can be used. A Safety Analysis Report is submitted that describes the behavior of the package under both normal and severe accident conditions. Usually, radioactive material is shipped in public commerce without restrictions, reliance for safety being placed on the package design.

II.A.1 ACCIDENT CLASSIFICATIONS

The consequences of an accident can be radiological or nonradiological in nature. Radiological consequences derive their significance from the fact that radioactive material is being transported, and, in some cases, a significant hazard can arise should this material be released from its package.

Nonradiological consequences depend on the size, weight, temperature, and other characteristics of the package during shipment rather than on the contents of the package.

Depending on the radioactive material being carried, two possible occurrences must be avoided: the assembly of a critical mass and the release of significant amounts of radioactive material.

The classification of accidents given in this analysis is the same as that elsewhere.¹ However, because of differences in some package designs, the consequences are different.

II.A.2 EXPERIENCE

In the past 25 years, several million packages of radioactive material, including about 3600 packages of irradiated fuel, have been transported in routine commerce. The Department of Transportation (DOT) estimates that about 800,000 packages are now shipped each year in the U.S. During that same period, only about 300 accidents were recorded^{2,3} in which radioactive material was involved. None of these⁴ resulted in serious injury of people as a result of

the radioactive nature of the material. In only about 30% of those accidents was there any release of radioactive material from the package or increase in the radiation levels outside the package.

II.A.3 TRANSPORTATION ACCIDENT STATISTICS

The probability of accidents by truck, rail, and barge are derived from statistics on accidents supplied by the DOT for 1969 and 1970.⁵⁻⁷ The conditions likely to be encountered in the accidents in terms of velocity of impact of the vehicle and of incidence and duration of fire were developed from analyses made by Leimkuhler,⁸ from various statistics on frequency of fires, and from information in the 1969 and 1970 accident statistics.

Accidents vary in frequency and severity. Most accidents occur at low vehicle speeds; the severity of accidents is greater at higher speeds, but the frequency is lower. Accidents generally involve some combination of impact, puncture, and fire effects.

For this analysis, accidents are divided into five categories: minor, moderate, severe, extra severe, and extreme.

The Natural Resources Defense Council expressed concern that the accident analysis suffers from a lack of important data.* It should be noted that the analysis of the entire spectrum of accidents given in this Appendix was based on data primarily available through agencies in the Department of Transportation and various previous studies. As was noted in the Draft Statement, some data are sparse. Where data are sparse or not available, best estimates were used and these estimates tended toward the conservative side. The AEC believes conservative estimates were made.

II.A.3.1. For Trucks

In 1969, 38,813 accidents involving death, injury, or property damage in excess of \$250 were reported for large motor carriers.⁵ The accidents included 19,682 injuries and 1497 fatalities, with an overall accident rate of 2.48 accidents per million vehicle-miles. For hazardous materials shipments, the accident rate was 1.69 per million vehicle-miles. The overall total accident rates per million vehicle-miles for previous years were: 1964, 3.2; 1965, 2.3; 1966, 2.4; 1967, 2.4; and 1968, 2.5. Of the reported accidents, 50% involved collision with autos or buses; 15.5%, collisions with other trucks; 14%, collisions with fixed objects;

*Comment Letter 38c, p. 15.

0.6%, collisions with trains; 9.5% were rollovers or runoffs; and 11.4%, other types of accidents. Fire occurred in 1.57% of the reported accidents.⁹

Severe damage to the package might be encountered in all types of truck accidents. Impacts that are likely to be most damaging are those on rigid stationary objects, such as concrete abutments or bridge structures. In collisions with an object, yielding or crushing of the vehicle or of the object with which the vehicle collides reduces the impact received by the package. Rollovers usually occur at high speeds and must be considered potential causes of major damage to a package.

A study in 1960 showed the percentages of accidents for four ranges of truck speeds (Table II.A-1).⁸ Those percentages were assumed to apply to the four slightly different ranges of speeds (0 to 30, 30 to 50, 50 to 70, and over 70 mph) used in the present analysis.

Data⁵ on truck fires indicate that fire is involved in about 0.8% of truck-truck collisions, 0.3% of truck-auto collisions, 0.6% of truck-fixed-object collisions, 2% of truck-train collisions, and 1% of rollover and runoff accidents. Most fires involve only the fuel from the vehicle's fuel tanks and last less than 1/2 hr unless other freight becomes involved. Only in the case of truck-truck collisions is a larger supply of fuel likely to be involved, for example, a collision with a gasoline tank truck or a truck loaded with paint. Some fires start from overheated tires or from accidental ignition of cargo. Truck-auto, truck-bus, and single-vehicle accidents were considered to be essentially free of fires that last longer than 1/2 hr.

Only for truck-truck accidents is there a likelihood that fires lasting longer than 1/2 hr would occur and then only if one of the trucks were carrying significant amounts of flammable liquids (e.g., tank trucks carrying gasoline or liquefied petroleum gas or van trailers carrying barrels of paint). For lack of data on the percentage of trucks carrying flammables, the assumption is made that at least one of the trucks in each truck-truck accident is carrying flammable cargo.

An estimate has been made⁸ that, of the fires that do occur, 1% last more than 1 hr, 10% last 1/2 hr and to 1 hr, and 89% last less than 1/2 hr. Although fires that last for several days do occur in transport, in most cases these involve the burning of only small amounts of fuel per unit time and are of little consequence in terms of heat output.

Table II.A-1

PERCENTAGE OF TRUCK ACCIDENTS AT VARIOUS SPEEDS

Type of Accident	0-32 mph	32-52 mph	52-72 mph	>72mph
All Accidents	23.7	56.0	19.8	0.5
Collisions with Autos and Buses	34	42	23	1
Collisions with Other Trucks	25	72	3	0.1
Overturns and Other Collisions	8	69	23	0

Table II.A-2

ACCIDENT PROBABILITY

Severity	Vehicle Speed (mph)	Fire Duration (hr)	Probability per Vehicle Mile		
			Rail	Truck	Barge ^a
Minor	0-30	<1/2	6 E-9	6 E-9	
	0-30	0	4.7 E-7	4 E-7	1.6 E-6
	30-50	0	2.6 E-7	9 E-7	1.4 E-7
		Total	7.4 E-7	1.3 E-6	1.7 E-6
Moderate	0-30	1/2-1	9.3 E-10	5 E-11	
	30-50	<1/2	3.3 E-9	1 E-8	8 E-9
	50-70	<1/2	9.9 E-10	5 E-9	2 E-9
	50-70	0	7.5 E-8	3 E-7	3.4 E-8
		Total	8.0 E-8	3.1 E-7	4.4 E-8
Severe	0-30	>1	7.0 E-11	5 E-12	
	30-50	>1	3.9 E-11	1 E-11	9.3 E-11
	30-50	1/2-1	5.1 E-10	6 E-12	1.3 E-9
	50-70	1/2-1	1.5 E-10	6 E-12	3.3 E-10
	>70	<1/2	1 E-11	1 E-10	
	>70	0	8 E-10	8 E-9	
		Total	1.5 E-9	8.2 E-9	1.6 E-9
Extra Severe	50-70	>1	1.1 E-11	6 E-13	2.3 E-11
	>70	1/2-1	1.6 E-12	2 E-13	
		Total	1.3 E-11	8 E-13	2.3 E-11
Extreme	>70	1	1.2 E-13	2 E-14	
		Total	1.2 E-13	2 E-14	

^aBarge accident probabilities are based on the duration of the fire and actuarial data on cargo damage. The impact velocities of all barge accidents were considered to be less than 10 mph, but for the purposes of this table, minor cargo damage is assumed to be equivalent to vehicle impact speeds of 0-30 mph, moderate cargo damage 30-50 mph, and severe cargo damage 50-70 mph.

In this analysis the source of the fire is assumed to be the petroleum being transported in bulk by a tractor-trailer combination. The volume of petroleum available under such conditions is assumed to be 10,000 gal. If it is to burn for 2 hr, the petroleum must be approximately 5 in. in depth, which in turn requires the pool to have an area of about 3000 ft². Gasoline, the most likely petroleum product, liberates about 20,000 Btu/lb of fuel. Over a period of two hours, a fire consuming 10,000 gal of gasoline will produce heat at a rate of about 2×10^5 Btu/sec, which is equivalent to approximately 200 MW.

Table II.A-2 lists the probabilities for truck accidents for the five severity categories.

II.A.3.2 For Railroad Cars

In 1969, for a total of about 61 billion car-miles, the rail industry⁶ reported a total of 8543 accidents that involved death, injury, or property damage in excess of \$750, of which 4971 were accidents other than grade-crossing accidents. The accidents included 23,356 injuries and 2299 fatalities.

In 1969, the total number of accidents per million train-miles was 9.89; in 1968, it was 9.16; and in 1967, it was 8.15.⁶ The average train length is about 70 cars. The overall accident rate is about 2.23 railcar accidents per million car-miles (see Appendix II.C).

Twenty-one percent of the reported accidents were collisions, 70% were derailments, and 9% were other types of accidents. About 1.5% of the rail accidents involved fire, most of them occurring in serious derailments in overland movements.

In rail accidents, severe damage to the cargo may be encountered in both collisions and derailments. Rail grade-crossing accidents (train-truck or train-auto) rarely involve significant damage to cargo. Other collisions that do not cause derailment are not likely to involve significant damage to a package. Accidents that have the highest probability of producing significant damage to shipment containers are overland derailment accidents that involve either impact of the packages on forward cars or impact of rearward cars on the packages.

An unpublished study by the DOT of the total accidents that occur at various speeds indicates that 58.5% of all train accidents occur below 30 mph, 32% occur between 30 and 50 mph, 9.4% occur between 50 and 70 mph, and 0.1% occur above 70 mph.

Fires other than those that involve ruptured tank cars of flammable liquids are unlikely to last longer than 1/2 hr because of lack of sufficient fuel. Data that relate major fires to train speed are sparse. The estimate is made that 1.5% of all rail accidents involve fire, of which 85% last less than 1/2 hr, 14% last 1/2 to 1 hr, and 1% last more than 1 hr. Table II.A-2 lists the probabilities for rail accidents. The same type of fire as described in Section II.A.3.1 is assumed to occur in the 2-hr rail fire.

II.A.3.3 For Barges

Records for fiscal year 1970 for domestic waterborne traffic⁷ show a total of 506 billion ton-miles of water traffic, with 548 cargo barge accidents reported. Data are not available that indicate the fraction of those ton-miles due to barge traffic. Barge ton-miles have been estimated to total 380 billion. According to a U.S. Coast Guard report,⁷ miscellaneous types of vessels, including cargo barges, were involved in accidents that resulted in 33 injuries and 33 fatalities during fiscal 1970.

The available data cannot be analyzed in the same way as the data for rail or truck transport. On the basis of discussions with the U.S. Coast Guard, the average net (cargo) weight of a typical barge is assumed to be about 1200 tons. The total number of barge-miles would then be about 310 million. This number, divided by the 548 cargo barge accidents reported, yields an accident rate of about 1.8 accidents per million barge-miles.

Very few data are available on the severity of accidents that involve barges. Barges travel only a few miles per hour; therefore, the velocity of impacts in accidents would be small. Because of the large mass of the vehicle and cargo, severe impact forces could be encountered by packages (e.g., spent-fuel casks) aboard barges. A forward barge could impact on a bridge pier and suffer crushing forces due to other barges being pushed into it. A coastal or river ship could knife into a barge. Fires could result in either case. An extreme accident, for example, an extreme impact plus a long fire, is not considered to be credible. The likelihood of a severe fire in barge accidents is small because of the availability of water at all times. Also, because casks could be kept cool by sprays or by submergence in water, loss of mechanical cooling could be compensated for.

The likelihood of cargo damage occurring in a barge accident is much less than in the case of rail accidents. For this analysis, the estimate is made, based on U.S. Coast Guard data, that about 90% of the barge accidents would result in minor or no damage to the cargo and would not involve fires. Moderate cargo damage due to

impact would result in 8% of the barge accidents and severe damage in 2%. Fire would be likely only in those accidents that involved moderate or severe cargo damage, and the estimated likelihood of a fire in severe accidents would be 10 times that in moderate accidents. Based on the 1970 data, with only one cargo fire reported, the estimate is made that fire would occur in 0.6% of the moderate accidents and in 6.5% of the severe accidents.

No data exist on the duration of fires in barge accidents; therefore, the rail figures were used: namely, 85% of all fires last less than 1/2 hr, 14% last 1/2 hr to 1 hr, and 1% last more than 1 hr. Table II.A-2 includes the probabilities for barge accidents. The same type of fire as described in Section II.A.3.1 is assumed to occur in a two-hour barge fire.

II.A.4 ACCIDENT-SEVERITY CATEGORIES

In Table II.A-3, accidents are categorized by degree of severity in terms of velocity of vehicle impact and incidence and duration of fire.

II.A.5 ACCIDENT PROBABILITY

Table II.A-2 shows the probability of an accident in each of the five accident-severity categories and for each of the three modes of transport. The probabilities were calculated on the basis of the data presented earlier. The differences between the probabilities of truck, train, and barge accidents, in terms of accidents per mile in each of the severity categories, are small. To estimate the risks in this analysis, a single value rounded off to one significant figure is taken for all three modes of transport (Table II.A-4).

II.A.6 RELATIONSHIP OF ACCIDENT SEVERITY TO PACKAGE DAMAGE

The amount of damage to a package in an accident is not directly related to the accident severity; that is, in a series of accidents of the same severity, or in a single accident that involves a number of packages, the amount of damage to the packages involved may vary from no damage to extensive damage.

Various factors limit the effect that accident conditions will have on a package.^{10,11} In relatively minor accidents, serious damage to packages can occur from impacts on sharp objects or from being struck by other cargo. Conversely, even in extreme accidents, damage to some packages may be minimal. In some cases, packages may be thrown free of the impacting vehicles or be so located in the vehicle that they are unaffected by the impact or by the fire that ensues. Package

Table II.A-3

ACCIDENT SEVERITY CATEGORIES

Vehicle Speed at Impact (mph)	Fire Duration (hr)	Severity
0-50	0	Minor
0-30	<1/2	
50-70	0	Moderate
30-70	<1/2	
0-30	1/2-1	Severe
>70	0-1/2	
30-70	1/2-1	
0-50	>1	
>70	1/2-1	Extra Severe
50-70	>1	
>70	>1	Extreme

Table II.A-4

ACCIDENT PROBABILITIES FOR TRUCK, RAIL, AND
BARGE PER VEHICLE-MILE FOR THE ACCIDENT SEVERITY CATEGORIES

Severity	Accident Probability per Vehicle Mile
Minor	2×10^{-6}
Moderate	3×10^{-7}
Severe	8×10^{-9}
Extra Severe	2×10^{-11}
Extreme	1×10^{-13}

damage depends on the amount of energy absorbed by the package and the ability of the package to withstand those forces. Also, the form and amount of the energy transmitted to the package in an accident depend on several factors, which vary according to the accident circumstances.

The ability of a package to withstand accident forces depends on the design of the package and on the quality assurance exercised in its manufacture, use, and maintenance.

DOT regulations and AEC regulations specify certain package accident-damage tests^{12,13} that provide a means for reproducing in the laboratory or in the field the same general type and degree of damage a package might reasonably be expected to sustain in a severe transportation accident. Any package that can be shown to meet those standards is called a type-B package and can be expected to withstand serious accidents without leakage or significant loss of shielding.

Four such tests are specified: a 30-ft free fall onto a flat unyielding surface, a 40-in. free fall onto a steel plunger, a thermal test, and a water-immersion test. To give a better understanding of the design requirements imposed by the accident-damage test criteria, the 30-ft free-fall test and the thermal test are discussed in some detail below.

Although the terminal velocity of the package in the 30-ft drop is about 30 mph, the test requires dropping the package, including the protective shield if it is part of the package, on an unyielding surface. Thus, all the kinetic energy of the package at impact must be translated into damage to the package and none to the impacted surface. In very few accidents does the vehicle collide with an unyielding surface. In a real accident, the forces on the package are mitigated by the angle of impact of the vehicle, which absorbs much of the impact, and by yielding of the object with which the vehicle collides.

As an example, in an instrumented full-scale test of a 15-ton cask on a semi-trailer in which the trailer was driven into an immovable barrier at 28.5 mph,¹⁴ the cask received only a fraction of the stress it was designed to withstand. The cask remained tied in place on the trailer and was undamaged, whereas the tractor was completely demolished.

As part of that same test series, a semitrailer truck loaded with several different types of drums was driven into the immovable target at 42 mph. Several of the drums

lost their lids, but none of the inner containers were released or opened. About 50% of the drums were not damaged.

In the thermal test, the entire surface of the package is subjected to the heat input from a radiant environment that has a temperature of 1475°F and an emissivity of 0.9 for 30 min.

Severe transportation fires seldom last more than 1/2 hr, except in ships and storage depots,¹⁵ because either the fuel is exhausted or the fire is extinguished by fire-fighting crews. Although flame temperature of liquids such as jet fuel or kerosene may reach 1800 to 2000°F, such peak temperatures are reached only very locally on the surface of material involved in the fire. Only under very unusual circumstances is more than 50% of a package surface likely to be exposed to the flame for as long as 1/2 hr. Even in a longer fire, the package may be in a location where the fire has little or no effect on it. In addition, packages in fires that have burning surface areas smaller than about 400 ft² are frequently able to "see" through flames to cooler surrounding areas. Fires that have burning surface areas greater than about 2000 ft² may suffer from oxygen starvation in the interior of the fire. In petroleum fires, both situations could cause reduced heat inputs to the packages.

For the above reasons, the conclusion is made that a package designed to meet the thermal requirements of a type-B package is likely to withstand the fire conditions in transportation accidents.

REFERENCES FOR APPENDIX II.A

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2. W. C. McCluggage, "The AEC Accident Record and Recent Changes in AEC Manual Chapter 0529," Report BNWL-SA-3906, Battelle Memorial Institute, Battelle Northwest Laboratories, Hanford, Wash., August 1971, pp. 108-129.
3. A. W. Grella, Today's Role of DOT in Regulating the Transport of Radioactive Material, USAEC Report CONF-710801, August 1971, pp. 25-34.
4. OHM Newsletter, Vol. 2, No. 8, 1972, Department of Transportation.
5. Federal Highway Administration, "1969 Accidents of Large Motor Carriers of Property," U.S. Department of Transportation, December 1970.
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10. W. de L. M. Messenger and A. Fairbairn, "The Transport of Radioactive Materials--Interim Recommendations for the Application of Environmental Tests to the Approval of Packaging," Report AHSB(S)R.19, United Kingdom Atomic Energy Agency, 1973.
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14. O. A. Kelly and W. C. Stoddart, "Highway Vehicle Impact Studies: Tests, Methodical Analysis of Vehicle, Package, and Tie Down Systems, Capable of Carrying Radioactive Material," Report ORNL-NSIC-61, Oak Ridge National Laboratory, Oak Ridge, Tenn., February 1970.
15. R. Gibson, The Safe Transport of Radioactive Materials, Pergamon, London, 1966.

Appendix II.B

CALCULATIONAL METHOD FOR DETERMINING POPULATION EXPOSURE RESULTING FROM TRANSPORTATION

For most transportation workers, exposures received as a result of shipment of radioactive materials in the LMFBR fuel cycle would result from reasonably close contact with the shipment. However, the general population would be far removed from such packages (at least 100 ft away); therefore, a different technique was used to determine their exposure. The calculation is similar to that presented elsewhere.¹

The radioactive shipment on the vehicle is assumed to be a point source for distances of 100 ft or more from the source. To determine the total accumulated population dose, the radiation doses to individuals at various distances from the passing source were calculated and summed.

The dose rate, $\dot{D}$, in millirems/hr, at a point can be approximated as follows:

$$\dot{D} = \frac{K}{r^2} e^{-\mu r} B(r)$$

where

K = constant dependent on source strength, (millirems-ft²)/hr,

r = distance between source and exposure point, ft,

$e^{-\mu r}$ = attenuation factor due to gamma interactions with air occurring between source and point; μ = linear absorption coefficient, 1.18×10^{-3} ft⁻¹,

$B(r)$ = buildup factor to account for scattered components returning to the point of exposure.

Figure II.B-1 shows buildup factors as a function of the atomic number, Z , of the absorbing medium and the distances between the source and the point for 4-MeV gamma rays. With values of B from Figure II.B-1 for $Z = 15$ and the assumption that $B(r)$ is a linear function, the buildup factor was estimated as follows:

$$\begin{aligned} B &= 1.5 && \text{at } r = 850 \text{ ft} \\ &= 2.0 && \text{at } r = 1700 \text{ ft} \\ B(r) &= mr + b \\ &= (6 \times 10^{-4})r + 1 \end{aligned}$$

EXPOSURE-DOSE BUILDUP FACTOR AS A FUNCTION OF ATOMIC NUMBER Z FOR A 4-MeV POINT ISOTOPIC SOURCE.

Figure II.B-1

The average gamma-ray energy for fission products is known to be about 1 MeV. However, the use of the easily available data for 4-MeV gamma rays will not result in an error that is large compared with the precision of the calculation.

The dose to an individual at a point is determined by integrating the dose received by that individual as the radiation source passes his position.

Let

v = velocity of vehicle, ft/hr

$= dx/dt$

x = distance along center line of shipping route;

then the total dose, D , in millirems, is

$$D = \int_{-\infty}^{\infty} \frac{1}{v} \dot{D} dx$$

and

$$D(d) = \frac{K}{v} \int_{-\infty}^{\infty} \frac{e^{-\mu(x^2+d^2)^{1/2}}}{x^2 + d^2} B[(x^2 + d^2)^{1/2}] dx,$$

where d = perpendicular distance from center line of shipping route.

To obtain a quantitative estimate of dose, the following assumptions were made:

- (1) The source strength, K , is such that the exposure rate is 10 millirems/hr at 10 ft.* That is; $10 \text{ millirems/hr} = K/10^2$ or $K = 10^3 \text{ (millirems.ft}^2\text{)/hr}$.
- (2) The vehicle travels 200 miles/day (this will lead to a conservative estimate)
 v (velocity) = 200 miles/day
 $= 200(5280)/24 \text{ (ft/hr)}$
 $= 4.4 \times 10^4 \text{ ft/hr}$.

Based on a uniform distance traveled each day and uniform distribution of persons along the route, the cumulative radiation dose to the population is the same whether the vehicle is moving all of the time at a constant rate of speed or is standing still part of the day.

*Due to gamma-rays and/or neutrons.

- (3) There are no people closer than 100 ft. The dose to persons farther than 2600 ft from the vehicle is negligible (about 700 times less than that at 100 ft).
- (4) The distribution density is 330 people/square mile, uniformly dispersed along the route.

Substitution of values gives:

$$D(d) = 4.5 \times 10^{-2} \int_d^{\infty} \frac{[6 \times 10^{-4} r + 1]}{r} \frac{e^{-ur}}{[r^2 - d^2]^{1/2}} dr$$

D is the total dose (millirems) a person standing a distance d from the center line of the shipping route would receive from the passing vehicle.

To obtain the man-rem dose, the assumption was made that on the average, in each mile of the shipping route, a total of 165 people would be uniformly distributed between 100 and 2600 ft on each side of the route. For ease of calculation, 1/26th of the 165 people were considered to be grouped at 100-ft intervals on each side of the route.

The calculated total man-rem dose per vehicle mile to the persons on one side of the route would be:

$$\begin{aligned} & (165/26) \text{ people/mile } [D(100 \text{ ft}) + D(200 \text{ ft}) + \dots + D(2600 \text{ ft})] \\ = & (165/26) \text{ people/mile } (5.8 \times 10^{-7} \text{ rem} + 2.5 \times 10^{-7} \text{ rem} + \dots + 1.3 \times 10^{-9} \text{ rem}) \\ & = 6.35 (1.4 \times 10^{-6}) \text{ man-rem/mile} \\ & = 9.0 \times 10^{-6} \text{ man-rem/mile} \end{aligned}$$

For both sides of the route, the cumulative dose would be about 1.8×10^{-5} man-rem/vehicle-mile.

REFERENCE FOR APPENDIX II.B

1. Directorate of Regulatory Standards, USAEC, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants," Report WASH-1238, December 1972.

Appendix II.C

RISKS IN TRANSPORTATION FROM COMMON (NONRADIOLOGICAL) CAUSES

Information on injuries, fatalities, and property damage is available from the DOT in the form of accident bulletins. Table II.C-1 summarizes information from the Bureau of Motor Carrier Safety¹ and from the Bureau of Railroad Safety.²

Table II.C-1

ACCIDENT STATISTICS - COMMON CAUSES

Transportation Mode	Accidents per 10 ⁶ Vehicle-Miles	Injuries per Accident	Fatalities per Accident	Average Property Damage per Accident (\$)
Truck	2.48	0.53	0.044	2335
Rail	2.23 ^a	2.90	0.286	6200 ^b

^aAssumes an average of 15% of all cars on the train are involved (damaged) in the accident.

^bSource: K. B. Stewart, "Rail Accident Statistics Pertinent to the Shipment of Radioactive Materials," Report HW-76299, Jan. 21, 1963.

REFERENCES FOR APPENDIX II.C

1. Bureau of Motor Carrier Safety, Federal Highway Administration, U.S. Department of Transportation, "1969 Accidents of Large Motor Carriers of Property," December 1970.
2. Federal Railroad Administration, U.S. Department of Transportation, "Summary and Analyses of Accidents on Railroads in the United States, Calendar Year 1969," Accident Bulletin No. 138.

Appendix II.D

TYPES OF PACKAGES

The type of packaging is specified in DOT regulations (49 CFR 173), according to the type and quantity of radioactive material (Table II.D-1).

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 and (2) "normal form," which applies to all radioactive materials that are not "special form." Normal-form radioactive materials are classified into seven groups of radionuclides based primarily on their radiotoxicity. Table II.D-1 shows package limits for the seven transport groups and "special form."

Table II.D-1

QUANTITY LIMITS AS A FUNCTION OF TRANSPORT GROUP

Transport Group	Examples	Exempt Quantity (Ci)	Type A Package (Ci)	Type B Package ^a (Ci)
I	Pu-239, Cm-242, Cf-252	10^{-5}	10^{-3}	20
II	Bi-210, Pb-210, Sr-90	10^{-4}	5×10^{-2}	20
III	Cs-137, Ir-192, I-131	10^{-3}	3	200
IV	As-76, C-14, Ca-45	10^{-3}	20	200
V	Noble gases, Kr-85	10^{-3}	20	5,000
VI	Ar-37, Xe-133, Kr-85 (uncompressed)	10^{-3}	1,000	50,000
VII	Tritium (as a gas or in luminous paint)	25	1,000	50,000
Special Form	Co-60 radiography source, Pu-Be neutron source	10^{-3}	20	5,000

^aA "large quantity" is defined as any quantity in excess of a type-B quantity. "Large quantity" packages must meet all requirements that type-B packages meet.

Small quantities of radioactive materials, certain concentrations, small quantities of radioactive materials in manufactured goods, and low-specific-activity materials may be shipped in strong industrial packages and are exempt from specification packaging, marking, and labeling with the radioactive-material label. The Postal Service regulations generally allow the exempt quantities to be shipped by mail in leakproof containers.

Type-A quantities of radioactive materials must be shipped in packaging, identified as type-A packaging, that will prevent loss or dispersal of the radioactive contents and will retain shielding efficiency and effectiveness of other safety features under normal conditions of transport. Standards for evaluation and testing with respect to normal conditions specified in AEC and DOT regulations include temperatures in the range -40 to 130°C, all surfaces except the bottom wet for 30 min, being subjected while wet to a 4-ft free fall, vibration normally encountered in transport, and external pressure reduced to 0.5 atm.

Quantities that exceed 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 of shielding efficiency and must be capable of retaining essentially all contents 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 ft onto an essentially unyielding surface, with the package landing in the orientation that does the most damage; (2) a free fall from a height of 4 ft onto a 6-in.-diam steel plunger long enough to do maximum damage, when the package is in the "worst-case" orientation; (3) heat input from 30-min exposure to a fire or to other radiant environment that has a temperature of 1475°F and an emissivity of 0.9, assuming the package surface has an absorptivity of 0.8; and (4) for fissile material packages, immersion in water to a depth of 3 ft for 24 hr. These test conditions make up the design-basis accident for type-B packages; that is, package designs that meet these criteria are considered to provide adequate protection to the public and to operating personnel in the expected transportation environment, which includes severe accident conditions on occasions.

The Natural Resources Defense Council stated, regarding these regulations*--

Without an explicit, documented analysis which shows, in fact, that the potential for shipping containers hitting very solid objects at speeds higher than 30 mph is very remote, it would seem prudent to test the ability of containers to withstand a free-fall onto an unyielding surface at heights well in excess of 30 feet.¹⁰

...We believe that the AEC should test the behavior of Type-B shipping containers to heat substantially more severe than provided for by the design-basis condition of 1,475°F for 30 minutes.

A thorough review of recent, very severe railroad, truck and barge accidents should be undertaken to determine the parameters of the accidents in terms of container impact and the intensity and duration of associated fires. These

*Comment Letter 38c, pp. 6 and 7.

conditions should then be simulated in live tests to determine whether or not they would cause spent fuel casks and other containers to leak significant amounts of radionuclides.

At the same time, studies should be initiated to find the 'limits' of existing shipping containers. That is, spent fuel casks and other packages should be tested to destruction in order to learn the nature of the conditions that would cause a significant leakage of radionuclides. This information would be very useful (1) in evaluating the possibility that existing or future transportation conditions could severely damage a shipping container, and (2) in learning the nature of the 'weak points' in the current containers so that future designs could be improved.

Regarding shipping container testing, the AEC assumes that the potential for shipping containers hitting very solid objects at speeds higher than 30 mph is low but calculable, and has in fact calculated it (WASH 1238). However, the evaluation of the results of a severe impact does not depend on the probability of whether or not that event will happen. A comparison, however, can be made which indicates the large difference in cask damage between a cask impacting from 30 ft. onto a solid, essentially unyielding, surface¹ and a 30 mph impact of a tractor-trailer carrying a similar cask against a solid, essentially unyielding surface. In the latter test there was essentially no damage to the cask. However, when dropped as required in the regulations there was structural damage to the outside of the cask. These tests provide an indication that real accidents can occur at normal driving speeds even in excess of 30 mph, and not necessarily produce damage in excess of that produced in the hypothetical accident conditions.

Regarding fire testing, the duration for accident category 5 was assumed to be 2 hours, four times the time duration specified in the regulations. While real fires generally have a flame temperature of around 2000°F, real fires do not necessarily expose all surfaces to that temperature. The maximum temperature occurs approximately 30 in. above the burning surface. Experience with real casks tested in real fires have shown the difficulty in maximizing heat inputs to the cask. If the cask is lying in the pool of burning liquid the bottom will be kept cool while the top is exposed to the flame. The temperature and emissivity specifications provide a reasonable method for estimating the total heat input to a cask which would occur in most transportation fires.

Regarding the study of transportation accidents, the AEC is conducting a study of serious transportation accidents. The study (at Sandia Laboratories) began in 1972 and is due to be completed in 1975. The progress of the study was reported at the International Symposium on Transportation and Packaging of Radioactive Materials in Miami Beach in September 1974. The AEC has conducted several destructive test series

on various packages, including one on plutonium packages in 1972. None of the packages leaked even though stressed far beyond the AEC regulatory test conditions. AEC is now conducting destructive tests on many obsolete casks to compare predicted damage with actual damage. It is expected that the LMFBR spent fuel cask will be tested under conditions more severe than required by the current regulations (a full scale prototype test is considered most desirable). It should be made clear, however, that most people working in the package testing field regard the test requirements as sufficiently damaging to conclude that a package that survives them will survive most transport accidents.

Large quantities must be shipped in type-B packaging that provides for adequate dissipation of heat. In addition, no loss of contents must occur at an external pressure of 25 psig, which is approximately equivalent to immersion in water to a depth of 50 ft.

With respect to heat dissipation, the regulations require that the package be designed so that the temperature rise due to decay heat will not adversely affect the package or its contents and will not cause excessive pressure. The temperature of the accessible surface of the package must not exceed 180°F.

EXPOSURE ASSUMPTIONS AND DOSE-CONVERSION FACTORS USED IN ESTIMATING RADIATION DOSE

The estimates of radiation dose given in Sections 4.2.5, 4.3.8, 4.4.9, 4.5.6, and 4.6.3 consider LMFBR power plants, fuel-element-fabrication and -reprocessing plants, the transportation of radioactive materials and radioactive waste management. The estimates were calculated according to the following assumptions regarding exposure and the dose conversion factors listed in Tables II.E-1 through II.E-5. The external-dose conversion factors for swimming in water contaminated with radionuclides, immersion in radioactive air, or irradiation by contaminated ground surfaces (Tables II.E-1 through II.E-3) were taken directly from the EXREM III computer code.¹ The models in the EXREM III code are adapted from standard texts.^{2,3} The dose rate above the contaminated land surface is estimated for a height of 100 cm. The models for immersion in air and submersion in water assume that the entire body surface is in contact with a large volume of air or water. For all external exposures, the dose calculated is for the air at the surface of the skin. The total-body dose is assumed to equal the dose from gamma radiation in air at the skin surface for all external exposures.

Factors for converting internal radiation exposure to estimates of dose were obtained with models and with data published by the International Commission on Radiological Protection (ICRP).⁴⁻⁷ Those models are programmed in the INREM and INDOS computer codes,^{8,9} which were used to complete the calculations for inhaled and ingested radionuclides. The basic reference document is Publication No. 2 of the ICRP.⁴ The dose models adapted for the INREM code are the ones used by the ICRP in calculating the maximum permissible body burdens and maximum permissible concentrations (radiation safety guides) presented in Table 1 of ICRP No. 2. The basic parameter values used to implement the models are those found in ICRP No. 2, Tables 5 and 8 through 12. Retention of the radionuclides in the reference tissue is assumed to be described by a single-exponential function. Uniform distribution of the radionuclides throughout the tissue is assumed for all reference tissues except the GI tract. In the case of the GI tract, uniform distribution in the contents of the tract segment is assumed, and the dose (other than alpha dose) to the GI tract wall is assumed, on the average, to be only half of the dose to the contents. For alpha particles in the GI tract, the dose to the walls is assumed to be only 1% of the dose to the segment contents. The dynamics of the ICRP No. 2 lung model are described in ICRP No. 2, Table 10, for both soluble and insoluble forms of the radionuclides. For insoluble materials in the lungs, the biological half time (T_b) is taken to be 120 days for all radionuclides except plutonium and thorium for which

Table II.E-1
DOSE CONVERSION FACTORS FOR
SWIMMING IN CONTAMINATED WATER^a

Radionuclide	Conversion Factor ^b ($\frac{\text{millirems/year}}{\mu\text{Ci/cm}^3}$)	Radionuclide	Conversion factor ($\frac{\text{millirems/year}}{\mu\text{Ci/cm}^3}$)
H-3	0.00E0	I-134	4.45E5
Mn-54	1.56E5	I-135	3.24E5
Fe-55	2.86E2	Cs-134	2.90E5
Co-58	1.82E5	Cs-137	1.05E5
Co-60	4.68E5	Cs-138	4.02E5
Br-80m	1.58E4	Ce-141	1.45E4
Br-82	5.01E5	Ce-144	3.61E3
Br-83	3.50E3	Pm-147	0.00E0
Br-84	3.47E5	Eu-154	2.39E5
Kr-83m	5.07E2	Eu-155	1.20E4
Kr-85	3.84E2	Th-228	6.60E2
Rb-88	1.27E5	Th-231	5.46E3
Rb-89	4.51E5	Th-234	1.40E3
Sr-89	0.00E0	Pa-234	5.81E5
Sr-90	0.00E0	U-232	3.34E2
Y-90	0.00E0	U-233	3.06E2
Y-91	4.52E2	U-234	2.85E2
Zr-95	1.36E5	U-235	2.89E4
Nb-95	1.43E5	U-236	2.46E2
Ru-103	9.05E4	U-237	2.82E4
Ru-106	3.48E4	U-238	2.22E2
Ag-110m	5.10E5	Np-237	1.57E4
Sb-125	8.33E4	Pu-236	3.27E2
Te-125m	6.63E3	Pu-238	2.80E2
Te-127m	1.70E3	Pu-239	1.10E2
Te-127	7.59E2	Pu-240	2.41E2
Te-129m	7.05E3	Pu-241	0.00E0
Te-129	1.24E4	Pu-242	2.38E2
I-129	4.62E3	Am-241	4.84E3
I-130	3.80E5	Am-243	9.35E3
I-131	7.08E4	Cm-242	2.73E2
I-132	4.22E5	Cm-243	1.34E3
I-133	1.25E5		

^aTime spent swimming assumed to be 1% of the year.

^bThe number after E is the power of ten by which the number should be multiplied; that is, 1.27E5 = 1.27×10^5 .

Table II.E-2
DOSE CONVERSION FACTORS FOR IMMERSION IN A RADIOACTIVE CLOUD

Radionuclide	Conversion Factor ^a (millirems/year) $\mu\text{Ci}/\text{cm}^3$	Radionuclide	Conversion Factor (millirems/year) $\mu\text{Ci}/\text{cm}^3$
H-3	0.00E0	I-133	5.93E9
Ar-39	0.00E0	I-134	2.12E10
Mn-54	7.43E9	I-135	1.54E10
Fe-55	1.36E7	Xe-133	3.98E8
Co-58	8.66E9	Cs-134	1.38E10
Co-60	2.22E10	Cs-137	4.99E9
Br-80m	7.50E8	Cs-138	1.91E10
Br-82	2.38E10	Ce-141	6.89E8
Br-83	1.66E8	Ce-144	1.72E8
Br-84	1.65E10	Pm-147	0.00E0
Kr-83m	2.41E7	Eu-154	1.14E10
Kr-85m	1.40E9	Eu-155	5.69E8
Kr-85	1.83E7	Th-228	3.14E7
Kr-88	1.78E10	Th-231	2.60E8
Rb-88	6.05E9	Th-234	6.63E7
Rb-89	2.15E10	Pa-234	2.76E10
Sr-89	0.00E0	U-232	1.59E7
Sr-90	0.00E0	U-233	1.45E7
Y-90	0.00E0	U-234	1.36E7
Y-91	2.15E7	U-235	1.37E9
Zr-95	6.45E9	U-236	1.17E7
Nb-95	6.79E9	U-237	1.34E9
Ru-103	4.30E9	U-238	1.05E7
Ru-106	1.65E9	Np-237	7.45E8
Ag-110m	2.42E10	Pu-236	1.56E7
Sb-125	3.96E9	Pu-238	1.33E7
Te-125m	3.15E8	Pu-239	5.21E6
Te-127m	8.07E7	Pu-240	1.15E7
Te-127	3.61E7	Pu-241	0.00E0
Te-129m	3.35E8	Pu-242	1.13E7
Te-129	5.92E8	Am-241	2.30E8
I-129	2.20E8	Am-243	4.45E8
I-130	1.81E10	Cm-242	1.30E7
I-131	3.37E9	Cm-244	6.37E7
I-132	2.01E10		

^aThe number after E is the power of ten by which the number should be multiplied; that is, 2.15E7 = 2.15 x 10⁷.

Table II.E-3

DOSE CONVERSION FACTORS FOR IRRADIATION BY
CONTAMINATED GROUND SURFACES^a

Radionuclide	Conversion Factor ^b (millirems/year) $\frac{\mu\text{Ci}/\text{cm}^2}{\text{year}}$	Radionuclide	Conversion Factor (millirems/year) $\frac{\mu\text{Ci}/\text{cm}^2}{\text{year}}$
H-3	0.00E0	I-134	3.61E6
Mn-54	1.31E6	I-135	2.49E6
Fe-55	3.45E4	Cs-134	2.41E6
Co-58	1.54E6	Cs-137	8.88E5
Co-60	3.62E6	Cs-138	3.00E6
Br-80m	1.41E5	Ce-141	1.23E5
Br-82	4.07E6	Ce-144	3.57E4
Br-83	2.67E5	Pm-147	0.00E0
Br-84	2.46E6	Eu-154	1.92E6
Kr-83m	5.93E4	Eu-155	1.08E5
Kr-85	3.25E3	Th-228	2.83E4
Rb-88	9.24E5	Th-231	2.72E5
Rb-89	3.38E6	Th-234	2.10E4
Sr-89	0.00E0	Pa-234	4.55E6
Sr-90	0.00E0	U-232	2.94E4
Y-90	0.00E0	U-233	1.76E4
Y-91	3.52E3	U-234	2.60E4
Zr-95	1.13E6	U-235	3.60E5
Nb-95	1.19E6	U-236	2.40E4
Ru-103	7.85E5	U-237	4.92E5
Ru-106	2.92E5	U-238	2.13E4
Ag-110m	4.13E6	Np-237	1.62E5
Sb-125	8.07E5	Pu-236	2.97E4
Te-125m	2.23E5	Pu-238	2.61E4
Te-127m	6.16E4	Pu-239	9.90E3
Te-127	6.49E3	Pu-240	2.24E4
Te-129m	9.87E4	Pu-241	0.00E0
Te-129	1.32E5	Pu-242	2.22E4
I-129	1.32E5	Am-241	1.20E5
I-130	3.16E6	Am-243	9.35E4
I-131	5.98E5	Cm-242	2.37E4
I-132	3.55E6	Cm-244	2.55E4
I-133	1.04E6		

^aHeight = 100 cm.^bThe number after E is the power of ten by which the number should be multiplied; that is, 3.2E3 = 3.52 x 10³.

Table II.E-4
DOSE-CONVERSION FACTORS FOR INHALATION OF
RADIOACTIVE MATERIALS (rem/ μ Ci INHALED)^a

Radionuclide	Total body	Bone	Muscle	Thyroid	Liver	Kidneys	Spleen	Testes	Ovaries	Lungs	GI tract
H-3	1.08E-4									1.08E-4	1.08E-4
Mn-54	2.60E-3				1.60E-2					1.81E-1	7.12E-3
Fe-55	9.45E-4	1.36E-3			4.12E-3		7.39E-3			6.44E-3	3.56E-4
Co-58	2.15E-3				2.10E-3	3.61E-4	5.06E-4			2.52E-1	7.12E-3
Co-60	5.97E-3				5.90E-3	1.04E-3	1.46E-3			7.44E-1	2.13E-2
Br-80m	1.03E-4									1.23E-3	4.27E-2
Br-82	1.84E-3									1.13E-2	3.56E-3
Br-83	2.95E-5									3.39E-4	
Br-84	4.01E-5									4.68E-4	
Kr-83m											
Kr-85											
Rb-88	2.68E-5									2.91E-4	2.13E-3
Rb-89	3.42E-6									2.43E-5	1.07E-3
Sr-89	1.16E-2	4.14E-1								1.79E-1	2.37E-2
Sr-90	2.22E-1	1.11E1								1.20E0	2.13E-2
Y-90	6.24E-4	2.35E-2								2.14E-2	5.34E-2
Y-91	8.96E-3	3.35E-1								2.11E-1	3.56E-2
Zr-95	1.60E-2	6.17E-2			2.60E-2	3.32E-2	1.99E-2			1.97E-1	2.13E-2
Nb-95	4.47E-3	1.31E-2			7.53E-3	8.18E-3	6.61E-3			6.45E-2	1.07E-2
Ru-103	7.71E-4	1.56E-3				6.45E-3				7.56E-2	1.07E-2
Ru-106	2.85E-3	2.04E-2				3.94E-2				1.16E0	7.12E-2
Ag-110m	2.27E-3	4.04E-3			3.96E-3	7.94E-3				6.28E-1	3.05E-2
Sb-125	3.66E-3	1.75E-2		1.64E-5	1.78E-4					2.42E-1	1.07E-2
Te-125m	7.16E-4	3.63E-3		1.20E-3	2.41E-3	2.05E-2	5.20E-3	4.43E-3		5.02E-2	2.13E-2
Te-127m	1.66E-3	1.23E-2		3.47E-3	6.35E-3	5.39E-2	1.37E-2	1.44E-2		1.64E-1	1.07E-2
Te-127	3.63E-5	1.67E-4		1.24E-4	8.07E-5	6.86E-4	1.74E-4	1.89E-4		8.55E-4	3.56E-3
Te-129m	4.38E-3	1.82E-2		6.72E-3	1.14E-2	9.15E-2	2.32E-2	2.22E-2		1.97E-1	3.05E-2
Te-129	1.99E-5	5.08E-5		4.26E-5	3.21E-5	2.54E-4	6.44E-5	6.17E-5		3.41E-4	1.07E-3
I-129	5.42E-3			3.961E0						5.28E-2	2.13E-4
I-130	6.59E-4			2.13E-1						4.11E-3	3.56E-2
I-131	2.63E-3			1.44E0						2.07E-2	1.07E-3
I-132	1.30E-4			5.31E-2						8.88E-4	2.37E-3
I-133	5.74E-4			3.92E-1						5.06E-3	2.13E-3
I-134	4.24E-5			2.49E-2						3.63E-4	2.13E-3
I-135	2.86E-4			1.22E-1						1.97E-3	2.13E-3
Cs-134	5.62E-2	3.73E-2	9.67E-2		9.95E-2	3.37E-2	7.52E-2			5.48E-1	2.13E-3
Cs-137	3.24E-2	6.07E-2	5.97E-2		7.87E-2	2.85E-2	6.66E-2			4.46E-1	1.07E-3
Cs-138	4.66E-5									3.94E-4	2.13E-3
Ce-141	1.65E-3	1.97E-2			1.35E-2	6.60E-3				4.17E-2	1.07E-2
Ce-144	6.50E-2	1.20E0			4.91E-1	3.03E-1				1.01E0	7.11E-4
Pm-147	6.92E-3	1.88E-1			2.28E-2	3.23E-2				6.71E-2	4.27E-3
Eu-154	1.95E-1	3.03E0			2.76E-1	1.64E0				9.26E-1	4.27E-2
Eu-155	1.35E-2	1.16E-1			2.58E-2	6.66E-2				8.75E-2	5.34E-3
Th-228	4.16E1	1.27E3			1.67E1	9.28E1				2.25E2	7.12E-2
Th-231	5.04E-5	1.13E-3			7.38E-5	3.66E-4				1.07E-3	4.27E-3
Th-234	5.74E-3	2.04E-1			9.35E-3	5.30E-2				1.66E-1	5.34E-2
Pa-234	8.06E-5	3.87E-4			9.26E-5	3.35E-4				1.52E-3	
U-232	7.30E0	1.06E2				1.13E1				2.30E2	3.56E-2
U-233	1.31E0	2.20E1				5.13E0				5.50E1	3.05E-2
U-234	1.33E0	2.11E1				5.03E0				5.39E1	3.05E-2
U-235	1.20E0	2.02E1				4.72E0				5.06E1	3.56E-2
U-236	1.23E0	2.02E1				4.82E0				5.17E1	3.05E-2
Ce-141	1.65E-3	1.97E-2			1.35E-2	6.60E-3				4.17E-2	1.07E-2
Ce-144	6.50E-2	1.20E0			4.91E-1	3.03E-1				1.01E0	7.11E-4
Pm-147	6.92E-3	1.88E-1			2.28E-2	3.23E-2				6.71E-2	4.27E-3
Eu-154	1.95E-1	3.03E0			2.76E-1	1.64E0				9.26E-1	4.27E-2
Eu-155	1.35E-2	1.16E-1			2.58E-2	6.66E-2				8.75E-2	5.34E-3
Th-228	4.16E1	1.27E3			1.67E1	9.28E1				2.25E2	7.12E-2
Th-231	5.04E-5	1.13E-3			7.38E-5	3.66E-4				1.07E-3	4.27E-3
Th-234	5.74E-3	2.04E-1			9.35E-3	5.30E-2				1.66E-1	5.34E-2
Pa-234	8.06E-5	3.87E-4			9.26E-5	3.35E-4				1.52E-3	
U-232	7.30E0	1.06E2				1.13E1				2.30E2	3.56E-2
U-233	1.31E0	2.20E1				5.13E0				5.50E1	3.05E-2
U-234	1.33E0	2.11E1				5.03E0				5.39E1	3.05E-2
U-235	1.20E0	2.02E1				4.72E0				5.06E1	3.56E-2
U-236	1.23E0	2.02E1				4.82E0				5.17E1	3.05E-2
U-237	2.84E-4	7.57E-4				3.57E-3				7.08E-3	
U-238	1.17E0	1.94E1				4.41E0				4.73E1	2.67E-2
Np-237	1.39E2	3.34E3			3.10E2	1.03E3				5.39E1	3.05E-2
Pu-236	9.11E1	3.22E3			2.80E2	2.09E2				2.66E2	
Pu-238	1.44E2	5.70E3			8.16E2	6.09E2				1.88E2	3.56E-2
Pu-239	1.59E2	6.56E3			8.96E2	6.76E2				1.77E2	3.56E-2
Pu-240	1.59E2	6.55E3			8.96E2	6.76E2				1.77E2	3.56E-2
Pu-241	2.55E0	1.24E2			6.41E0	1.22E1				1.65E-1	7.12E-4
Pu-242	1.53E2	6.08E3			8.62E2	6.51E2				1.70E2	3.05E-2
Am-241	1.36E2	2.07E3			7.17E2	1.02E3				6.26E1	3.56E-2
Am-243	1.32E2	2.06E3			6.98E2	9.99E2				5.94E1	3.56E-2
Cm-242	3.38E0	5.09E1			5.19E1	1.54E1				4.05E1	4.27E-2
Cm-244	7.45E1	1.26E3			5.42E2	3.48E2				6.48E1	3.56E-2

^aBlanks indicate no data.

Table II.E-5
DOSE CONVERSION FACTORS^a FOR INGESTION
OF RADIOACTIVE MATERIALS (rem/ μ Ci INGESTED)^b

Radionuclide	Total body	Bone	Muscle	Thyroid	Liver	Kidneys	Spleen	Testes	Ovaries	Lungs	GI tract
H-3	6.16E-5										6.16E-5
Mn-54	8.65E-4				4.56E-3						1.94E-2
Fe-55	3.15E-4	4.53E-4			1.34E-3		2.46E-3			7.80E-4	6.47E-4
Co-58	1.61E-3				7.35E-4	2.71E-4	3.79E-4				1.94E-2
Co-60	4.48E-3				2.06E-3	7.80E-4	1.09E-3				3.88E-2
Br-80m	1.38E-4										
Br-82	2.45E-3										6.47E-3
Br-83	3.94E-5										
Br-84	5.34E-5										
Rb-88	3.57E-5										4.85E-3
Rb-89	4.56E-6										1.94E-3
Sr-89	8.69E-3	3.10E-1									4.85E-2
Sr-90	1.66E-1	8.31E0									3.88E-2
Y-90	2.50E-7	9.26E-6									9.70E-2
Y-91	3.58E-6	1.32E-4									6.47E-2
Zr-95	6.39E-6	2.47E-5			9.12E-6	1.33E-5	7.96E-6				3.23E-2
Nb-95	1.79E-6	4.98E-6			3.39E-6	3.27E-6	2.64E-6				1.94E-2
Ru-103	8.57E-5	1.87E-4				7.74E-4					2.43E-2
Ru-106	3.17E-4	2.45E-3				4.73E-3					1.94E-1
Ag-110m	8.72E-5	1.56E-4			1.54E-4	3.18E-4					6.47E-2
Sb-125	4.07E-4	1.75E-3		1.85E-6	2.14E-5					1.36E-3	1.94E-2
Te-125m	4.71E-4	2.46E-3		7.86E-4	1.21E-3	1.37E-2	3.42E-3	3.02E-3			9.70E-3
Te-127m	1.09E-3	8.31E-3		2.28E-3	3.17E-3	3.60E-2	8.99E-3	9.80E-3			2.43E-2
Te-127	2.39E-5	1.13E-4		8.14E-5	4.04E-5	4.57E-4	1.14E-4	1.29E-4			6.47E-3
Te-129m	2.88E-3	1.23E-2		4.42E-3	5.72E-3	6.10E-2	1.52E-2	1.52E-2			6.47E-2
Te-129	1.31E-5	3.44E-5		2.80E-5	1.60E-5	1.69E-4	4.24E-5	4.20E-5			2.43E-3
I-129	7.22E-3			5.17E0							3.23E-4
I-130	8.79E-4			2.77E-1							6.47E-2
I-131	3.50E-3			1.88E0							1.94E-3
I-132	1.73E-4			6.92E-2							4.85E-3
I-133	7.65E-4			5.12E-1							3.23E-3
I-134	5.65E-5			3.24E-2							3.23E-3
I-135	3.81E-4			1.60E-1							3.88E-3
Cs-134	7.49E-3	3.59E-3	7.56E-3		1.20E-2	7.08E-3	8.15E-3			1.50E-2	3.88E-3
Cs-137	4.32E-2	8.10E-2	7.96E-2		1.10E-1	3.80E-2	8.77E-2			1.24E-2	2.43E-3
Cs-138	6.21E-5										4.85E-3
Cs-141	6.60E-7	7.89E-6			5.63E-6	2.64E-6					2.16E-2
Co-144	2.60E-5	4.81E-4			2.05E-4	1.21E-4					1.94E-1
Pm-147	2.77E-6	7.31E-5			6.84E-6	1.29E-5					9.70E-3
Eu-154	7.78E-5	1.21E-3			1.15E-4	6.57E-4					9.70E-2
Eu-155	5.26E-6	4.63E-5			1.08E-5	2.66E-5					9.70E-3
Th-228	1.66E-2	4.93E-1			8.34E-3	4.64E-2					1.94E-1
Th-231	2.02E-8	4.39E-7			3.69E-8	1.83E-7					9.70E-3
Th-234	2.30E-6	7.95E-5			4.68E-6	2.65E-5					9.70E-2
Pa-234	3.22E-8	1.58E-7			3.56E-8	1.34E-7					
U-232	2.93E-1	4.15E0				4.43E-1					6.47E-2
U-233	5.24E-2	8.64E-1				2.02E-1					6.47E-2
U-234	5.13E-2	8.29E-1				1.98E-1					6.47E-2
U-235	4.82E-2	7.95E-1				1.85E-1					6.47E-2
U-236	4.92E-2	7.95E-1				1.90E-1					6.47E-2
U-237	1.09E-5	2.97E-5				1.40E-4					
U-238	4.50E-2	7.60E-1				1.73E-1					4.85E-2
Np-237	5.54E-2	1.37E0			1.19E-1	4.12E-1					6.47E-2
Pu-236	1.09E-2	3.86E-1			3.32E-2	2.51E-2					
Pu-238	1.72E-2	6.85E-1			9.67E-2	7.30E-2					6.47E-2
Pu-239	1.91E-2	7.87E-1			1.06E-1	8.12E-2					6.47E-2
Pu-240	1.91E-2	7.86E-1			1.06E-1	8.12E-2					6.47E-2
Pu-241	3.06E-4	1.49E-2			7.59E-4	1.46E-3					1.94E-3
Pu-242	1.84E-2	7.30E-1			1.02E-1	7.81E-2					6.47E-2
Am-241	5.42E-2	8.21E-1			2.85E-1	4.06E-1					6.47E-2
Am-243	5.30E-2	8.19E-1			2.78E-2	4.00E-1					6.47E-2
Cm-242	1.35E-3	2.04E-2			2.08E-2	6.15E-3					9.70E-2
Cm-244	2.98E-2	5.02E-1			2.17E-2	1.39E-1					6.47E-2

^a The number after E is the power of ten by which the number should be multiplied; that is, 3.69E-8 = 3.69 x 10⁻⁸.

^b Blanks indicate no data.

$T_b = 1$ year and $T_b = 4$ years, respectively. This lung model does not take account of the size, shape, density, or the chemical form of the particulates. The lung model is adopted in INREM despite the fact that the newer, more complex, ICRP Task Group Lung Model (TGLM) was initially suggested in Health Physics as early as 1966.¹⁰ Use of the TGLM is complicated by the fact that since its introduction the model and associated parameter values have been subject to numerous revisions and interpretations by various authorities. The ICRP No. 2 lung model, in contrast, offers the advantage of its long-standing, almost universal acceptance. Furthermore, experience with the ICRP No. 2 model continues to show that the model provides dose estimates of suitable precision in view of the uncertainties (e.g., quantity of radioactivity, particle-size distribution, and chemical form) of the terms for which radioactivity releases and potential exposures of man are projected.

Internal doses from inhalation and ingestion of radioactivity were estimated according to the factors contained in the INREM⁸ and INDOS⁹ codes, as shown in Tables II.E-4 and II.E-5. For lungs, the inhaled material was assumed to be insoluble. For all other reference organs, the inhalation dose-conversion factors are based on the assumption that the inhaled radionuclides are soluble in form. The dose-conversion factors presented in Tables II.E-4 and II.E-5 for ²³⁸Pu, ²³⁹Pu, ²⁴⁰Pu, ²⁴¹Pu, ²⁴¹Am, ²⁴²Cm, and ²⁴⁴Cm differ from those found in Appendix 4.G, Table 4.G-11, of the Draft Statement. Table II.E-6 is included to facilitate comparison of the appropriate dose-conversion factors from the two appendices. The values tabulated in Table II.E-6 are the ratios of the appropriate dose-conversion factors from the two sources (Appendix II.G and Appendix II.E). Using the estimates of dose to bone due to inhalation of ²³⁸Pu as an example: (2730 rems per microcurie inhaled/5700 rems per microcurie inhaled) = 0.48. Many of the differences between the inhalation dose-conversion factors from the two appendices can be traced directly to the lung models used, each with its associated influences on the magnitude and dynamics of radionuclide redistribution from the lungs to other reference organs.* The higher inhalation dose-conversion factor in Appendix II.G for ²⁴¹Pu-liver exposure is unique in that it reflects the biological half time in liver assumed in that appendix for the ²⁴¹Pu daughter, ²⁴¹Am. The longer biological half time is also reflected in both the ingestion and inhalation dose-conversion factors from Appendix II.G for ²⁴¹Am-liver exposure.

*The length of the time period over which dose is integrated following radionuclide intake also differs between the two appendices: 50 years in Appendix II.E and 70 years in Appendix II.G.

Table II.E-6
COMPARISON OF THE DOSE CONVERSION FACTORS IN APPENDIX II.G
WITH THOSE PRESENTED IN THIS APPENDIX^a

Isotope	Class W Compounds								Class Y Compounds	
	Inhaled				Ingested				Inhalation	
	Bone	Liver	Kidneys	Gonads	Bone	Liver	Kidneys	Gonads	Lung	Lymph Nodes
²³⁸ Pu	0.48	1.4	0.24	0.30	0.88	3.1	0.41	0.59	4.9	b
²³⁹ Pu	0.50	1.5	0.25	0.29	0.89	2.8	0.49	0.53	4.9	b
²⁴⁰ Pu	0.50	1.5	0.25	0.29	0.89	2.8	0.49	0.53	4.9	b
²⁴¹ Pu	0.54	3.9	0.25	0.27	1.3	13	c	c	12	b
²⁴¹ Am	1.6	1.9	0.17	0.35	30	35	2.5	7.4	15	b
²⁴² Cm	0.98	0.50	0.19	0.30	20	9.6	4.9	7.4	6.3	b
²⁴⁴ Cm	1.1	1.2	0.24	0.40	22	230	4.3	6.7	14	b

^aThe dose conversion factors (rem per microcurie of intake) in Appendix 4.G, Table 4.G.11, of the Draft Statement differ from those in this appendix, Tables II.E-4 and II.E-5, by the tabulated factors. For example, the inhalation dose factor from Appendix II.G for ²³⁸Pu in bone is 0.48 of the comparable factor in this appendix.

^bDose conversion factors for lymph nodes not presented in this appendix.

^cDose conversion factors not presented in Appendix 4.G, Table 4.G.11, of the Draft Statement.

Differences in lung models and choice of lung mass (ICRP No. 2, 1000 g; TGLM, 570 g) explains the variation in lung dose estimates. Parameter values were not available to do organ-specific dose estimates for the gonads and lymph nodes using the models and the data base of Appendix II.E; thus no factors appear for gonads or lymph nodes in Tables II.E-4 and II.E-5. To complete Table II.E-6, the total-body dose-conversion factor was substituted for the gonad dose-conversion factor; thus the values appearing in Table II.E-6 in the columns entitled gonads represent comparisons of the Appendix II.G gonad dose factors to the Appendix II.E total-body dose factors. This practice was not adopted for the lymph nodes, because they are known to accumulate actinide radioisotopes to concentrations far exceeding concentrations of those isotopes observed in other soft tissues of the body, excluding the liver and kidneys. The most significant differences in the ingestion dose-conversion factors between the two appendices are attributable to the choice of uptake fraction (f_w , portion of the ingested radioactivity reaching the reference organ). For example, for ^{244}Cm -liver exposure the f_w in Appendix II.G is 110 times the f_w in Appendix II.E, with the remainder of the difference in dose-conversion factors in that case being due to variation in biological half times. The magnitude of some of the values in Table II.E-6 is significantly influenced by the rounding off (many to only one figure) of dose-conversion factors for entry into Table 4.G.11 of the Draft Statement.

Table II.E-7 lists the factors that describe the bioaccumulation of radioactivity by freshwater plants and animals. Except for the radioisotopes of tellurium, these factors are identical with those listed in USAEC Report UCRL-50564.¹¹ Table II.E-8 gives factors for daily intake of radioactivity, as estimated by the TERMOD¹² code, from continuous deposition onto crops.

To estimate radiation dose, the following assumptions were made regarding internal exposure from ingestion and inhalation:

- (1) Man's total food supply was derived from the nearby land.
- (2) Any transfer of radioactive material from the immediate environment by wind redistribution, by water runoff, etc., was negligible and could be neglected.
- (3) Daily intake by man via ingestion was 1.2 liters of water, 1 liter of milk, 0.3 kg of meat, 0.25 kg of vegetables, and 0.02 kg each of fish and shellfish. Intake via inhalation is 20 m^3 of air per day.
- (4) Dry weight of man's above-surface food per square meter of surface is 0.1 kg.

Table II.E-7
BIOACCUMULATION FACTORS FOR FRESHWATER
PLANTS AND ANIMALS^a

Radionuclide	Plants ^b	Mollusks and crustaceans	Fin fish	Radionuclide	Plants ^b	Mollusks and crustaceans	Fin fish
H-3	1.00E0 ^c	1.00E0	1.00E0	Np-237	1.00E3	4.00E2	1.00E1
Mn-54	1.00E4	4.00E4	1.00E2	Pu-236	3.50E2	1.00E2	3.50E0
Fe-55	1.00E3	3.20E3	1.00E2	Pu-238	3.50E2	1.00E2	3.50E0
Co-58	2.00E2	2.00E2	2.00E1	Pu-239	3.50E2	1.00E2	3.50E0
Co-60	2.00E2	2.00E2	2.00E1	Pu-240	3.50E2	1.00E2	3.50E0
Br-80m	5.00E1	3.33E2	4.17E2	Pu-241	3.50E2	1.00E2	3.50E0
Br-82	5.00E1	3.33E2	4.17E2	Pu-242	3.50E2	1.00E2	3.50E0
Br-83	5.00E1	3.33E2	4.17E2	Am-241	5.00E3	1.00E3	2.50E1
Br-84	5.00E1	3.33E2	4.17E2	Am-243	5.00E3	1.00E3	2.50E1
Rb-88	1.00E3	1.00E3	2.00E3	Cm-242	5.00E3	1.00E3	2.50E1
Rb-89	1.00E3	1.00E3	2.00E3	Cm-244	5.00E3	1.00E3	2.50E1
Sr-89	5.00E2	1.00E2	5.00E0				
Sr-90	5.00E2	1.00E2	5.00E0				
Y-90	5.00E3	1.00E3	2.50E1				
Y-91	5.00E3	1.00E3	2.50E1				
Zr-95	1.00E3	6.67E0	3.33E0				
Nb-95	8.00E2	1.00E2	3.00E4				
Ru-103	2.00E2	3.00E2	1.00E1				
Ru-106	2.00E2	3.00E2	1.00E1				
Ag-110m	2.00E2	7.69E2	2.31E0				
Sb-125	1.50E3	1.00E1	1.00E0				
Te-125m	1.00E3	1.00E5	4.00E2				
Te-127m	1.00E3	1.00E5	4.00E2				
Te-127	1.00E3	1.00E5	4.00E2				
Te-129m	1.00E3	1.00E5	4.00E2				
Te-129	1.00E3	1.00E5	4.00E2				
I-129	4.00E1	5.00E0	1.50E1				
I-130	4.00E1	5.00E0	1.50E1				
I-131	4.00E1	5.00E0	1.50E1				
I-132	4.00E1	5.00E0	1.50E1				
I-133	4.00E1	5.00E0	1.50E1				
I-134	4.00E1	5.00E0	1.50E1				
I-135	4.00E1	5.00E0	1.50E1				
Cs-134	8.00E1	1.00E2	4.00E2				
Cs-137	8.00E1	1.00E2	4.00E2				
Cs-138	8.00E1	1.00E2	4.00E2				
Ce-141	5.00E3	1.00E3	2.50E1				
Ce-144	5.00E3	1.00E3	2.50E1				
Pm-147	5.00E3	1.00E3	2.50E1				
Eu-154	5.00E3	1.00E3	2.50E1				
Eu-155	5.00E3	1.00E3	2.50E1				
Th-228	1.50E3	5.00E2	3.00E1				
Th-231	1.50E3	5.00E2	3.00E1				
Th-234	1.50E3	5.00E2	3.00E1				
Pa-234	1.13E3	1.13E2	1.13E1				
U-232	1.00E3	1.00E2	1.00E1				
U-233	1.00E3	1.00E2	1.00E1				
U-234	1.00E3	1.00E2	1.00E1				
U-235	1.00E3	1.00E2	1.00E1				
U-236	1.00E3	1.00E2	1.00E1				
U-237	1.00E3	1.00E2	1.00E1				
U-238	1.00E3	1.00E2	1.00E1				

^aAll bioaccumulation factors listed are from S. E. Thompson, C. A. Burton, D. J. Quinn, and Y. C. Ng, Report UCRL-50564, Rev. 1 (1972).

^bIncludes algae and macrophytes.

^c1.00E+00 is equivalent to 1.0×10^0 .

Table II.E-8

DAILY INTAKE OF RADIOACTIVITY THROUGH
THE TERRESTRIAL FOOD CHAIN^a

(μCi/day)

Radionuclide	Above-surface food ^b	Milk ^c	Beef ^d	Total
H-3	3.00E-3	9.89E-1	8.00E-3	1.00E0
Mn-54	4.60E0	1.26E-1	1.13E-1	4.84E0
Fe-55	4.75E0	6.39E-1	8.19E0	1.36E1
Co-58	4.05E0	4.45E-1	7.56E-1	5.25E0
Co-60	4.78E0	5.45E-1	2.97E0	8.29E0
Br-80m	6.55E-2	1.33E-1	<1E-3	1.99E-1
Br-82	4.80E-1	2.26E0	5.28E-3	2.74E0
Br-83	3.58E-2	4.70E-2	<1E-3	8.29E-2
Br-84	7.75E-3	2.66E-3	<1E-3	1.04E-2
Kr-83m	2.78E-2	1.19E-2	<1E-3	3.97E-2
Kr-85	4.80E0	1.13E1	4.83E0	2.09E1
Rb-88	1.55E-2	6.05E-3	<1E-3	2.16E-2
Rb-89	3.85E-3	<1E-3	<1E-3	4.26E-3
Sr-89	3.83E0	8.40E-1	3.00E-2	4.70E0
Sr-90	4.80E0	1.18E0	1.60E-1	6.14E0
Y-90	8.10E-1	<1E-3	<1E-3	8.10E-1
Y-91	3.90E0	<1E-3	2.20E-1	4.12E0
Zr-95	3.97E0	<1E-3	1.75E0	5.72E0
Nb-95	3.47E0	9.60E-1	7.80E0	1.22E1
Ru-103	3.60E0	<1E-3	1.28E1	1.64E1
Ru-106	4.65E0	<1E-3	6.09E1	6.56E1
Ag-110m	4.57E0	2.52E1	5.80E-1	3.04E1
Sb-125	4.72E0	7.90E-1	7.80E-1	6.29E0
Te-125m	3.90E0	4.29E-1	4.05E1	4.48E1
Te-127m	4.27E0	4.70E-1	6.63E1	7.10E1
Te-127	1.36E-1	8.40E-3	1.31E-2	1.58E-1
Te-129m	3.42E0	3.80E-1	6.20E-1	4.42E0
Te-129	1.72E-2	<1E-3	<1E-3	1.76E-2
I-129	4.80E0	1.27E1	7.62E0	2.51E1
I-130	1.80E-1	8.81E-2	2.78E-3	2.71E-1
I-131	1.81E0	1.39E0	4.10E-1	3.61E0
I-132	9.40E-1	6.87E-1	8.85E-2	1.71E0
I-133	2.88E-1	1.67E-1	7.23E-3	4.61E-1
I-134	1.30E-2	1.00E-3	<1E-3	1.40E-2
I-135	9.90E-2	3.63E-2	<1E-3	1.36E-1
Cs-134	4.72E0	7.39E0	7.80E-1	1.29E1
Cs-137	4.80E0	8.30E0	1.04E0	1.41E1
Cs-138	8.05E-3	<1E-3	<1E-3	8.85E-3
Ce-141	3.40E0	<1E-3	3.00E-2	3.43E0
Ce-144	4.60E0	1.00E-2	1.70E-1	4.78E0
Pm-147	4.75E0	<1E-3	9.50E-1	5.70E0
Eu-154	4.80E0	<1E-3	1.20E0	6.00E0
Eu-155	4.70E0	<1E-3	8.70E-1	5.57E0
Th-228	4.73E0	3.00E-3	7.84E2	7.89E2
Th-231	3.60E-1	<1E-3	4.60E-1	8.20E-1
Th-234	3.10E0	<1E-3	7.62E1	7.93E1
Pa-234	2.95E-4	4.10E-10	5.50E-5	3.50E-4
U-232	4.80E0	3.07E-1	1.09E1	1.60E1
U-233	4.80E0	3.17E-1	1.13E1	1.64E1
U-234	4.80E0	3.20E-1	3.00E-2	5.15E0
U-235	4.80E0	3.20E-1	3.00E-2	5.15E0
U-236	4.80E0	3.20E-1	3.00E-2	5.15E0
U-237	1.68E0	9.11E-2	1.29E-1	1.90E0
U-238	4.80E0	3.20E-1	3.00E-2	5.15E0
Np-237	4.80E0	3.17E-3	1.19E3	1.20E3
Pu-236	8.40E-3	<1E-3	<1E-3	8.64E-3
Pu-238	9.00E-3	<1E-3	<1E-3	1.00E-2
Pu-239	4.23E-1	1.00E-3	7.00E-3	4.31E-1
Pu-240	2.55E0	<1E-3	4.56E-3	2.57E0
Pu-241	9.58E-3	<1E-3	<1E-3	9.60E-3
Pu-242	9.94E-3	<1E-3	<1E-3	1.00E-2
Am-241	4.80E0	<1E-3	4.44E3	4.44E3
Am-243	4.80E0	<1E-3	1.19E3	1.20E3
Cm-242	4.45E0	<1E-3	2.16E2	2.21E2
Cm-244	4.80E0	<1E-3	3.72E3	3.72E3

^aValues were calculated on the assumption of a continuous unit deposition (1 μCi/m²) on the landscape per day.^b0.25 kg ingested daily.^c1.0 liter ingested daily.^d0.3 kg ingested daily.

- (5) Dry weight of a square meter of pasture is 0.15 kg.
- (6) Dry weight of grass consumed per day by a cow is 10 kg.
- (7) Amount of milk produced per day by a cow is 11 liters.
- (8) Fraction of deposited radionuclides initially intercepted by the principal receptor compartments: above-surface food crops, 0.1; soil surface in area where crops are grown, 0.9; pasture grass, 1.0.*
- (9) Environmental half-life of radioactivity deposited on vegetation is 13 days.

Assumptions regarding external exposure from immersion in air, from ground, and from swimming in water are as follows:

- (1) Dose-reducing factors such as shielding provided by dwellings and time spent away from the location, assumed for the calculation, are not considered.
- (2) Doses are estimated per year of plant operation for normal releases and per event for accidents.
- (3) Dose-conversion factors for external whole-body irradiation are those for photons only.
- (4) Exposure via swimming in water occurs 1% of the year.
- (5) Immersion in air occurs 100% of the year.
- (6) Exposure to ground surfaces occurs 100% of the year at 100 cm height.

The following are examples of dose calculations for U-235 in liquid effluents at a concentration of $1 \times 10^{-10} \mu\text{Ci}/\text{cm}^3$ or in gaseous effluents at $1 \times 10^{-15} \mu\text{Ci}/\text{cm}^3$. A deposition velocity of 10^{-22} m/sec was assumed in calculating ground deposition of radioactive material from the air.

External Dose Calculations

- (1) Swimming in water 1% of the year

$$1.0 \times 10^{-10} \mu\text{Ci}/\text{cm}^3 \times 2.89 \times 10^4 \frac{\text{millirems/year}}{\mu\text{Ci}/\text{cm}^3} \text{ (Table II.E-1)}$$

$$= 2.89 \times 10^{-6} \text{ millirem/year.}$$

*Information added in response to EPA comments, p. 93, concerning Appendices 4.E and 4.I in the Draft Statement.

- (2) Immersion in air 100% of the year

$$1.0 \times 10^{-15} \mu\text{Ci}/\text{cm}^3 \times 1.37 \times 10^9 \frac{\text{millirems}/\text{year}}{\mu\text{Ci}/\text{cm}^3} \quad (\text{Table II.E-2})$$

$$= 1.37 \times 10^{-6} \text{ millirem}/\text{year}.$$

- (3) Irradiation at height of 100 cm by ground surfaces 100% of the year

$$1.0 \times 10^{-15} \mu\text{Ci}/\text{cm}^3 \times 10^{-2} \text{ m}/\text{sec} \times 10^2 \text{ cm}/\text{m} \times 3.60 \times 10^5 \frac{\text{millirems}/\text{year}}{\mu\text{Ci}/\text{cm}^2}$$

$$(\text{Table II.E-3}) \times 3.15 \times 10^7 \text{ sec}/\text{year} = 1.13 \times 10^{-2} \text{ millirem}/\text{year}.$$

Internal Dose Calculations

- (1) Inhalation of radioactive material (50-year whole-body dose commitment)

$$1.0 \times 10^{-15} \mu\text{Ci}/\text{cm}^3 \times 20 \text{ m}^3/\text{day} \times 10^6 \text{ cm}^3/\text{m}^3 \times 365 \text{ days}/\text{year} \times$$

$$1.20 \text{ rem}/\mu\text{Ci} (\text{Table II.E-4}) \times 10^3 \text{ millirems}/\text{rem} = 8.76 \times 10^{-3}$$

$$\text{millirem}/\text{year}.$$

- (2) Ingestion of water that contains radioactive material

$$1.0 \times 10^{-10} \mu\text{Ci}/\text{cm}^3 \times 10^3 \text{ cm}^3/\text{liter} \times 1.2 \text{ liter}/\text{day} \times 365 \text{ days}/\text{year} \times$$

$$4.82 \times 10^{-2} \text{ rem}/\mu\text{Ci} (\text{Table II.E-5}) \times 10^3 \text{ millirems}/\text{rem} = 2.11 \times$$

$$10^{-3} \text{ millirems}/\text{year}.$$

- (3) Ingestion of fish

$$1.0 \times 10^{-10} \mu\text{Ci}/\text{cm}^3 \times 1 \text{ cm}^3/\text{g} \times 1.0 \times 10^1 (\text{bioaccumulation factor}$$

$$\text{from Table II.E-7}) \times 20 \text{ g}/\text{day} \times 365 \text{ days}/\text{year} \times 4.82 \times 10^{-2}$$

$$\text{rem}/\mu\text{Ci} (\text{Table II.E-5}) \times 10^3 \text{ millirems}/\text{rem} = 3.52 \times 10^{-4} \text{ millirem}/\text{year}.$$

- (4) Ingestion of terrestrial food

- (a) Above-surface crops

$$1.0 \times 10^{-15} \mu\text{Ci}/\text{cm}^3 \times 10^{-2} \text{ m}/\text{sec} \times 3.15 \times 10^7 \text{ sec}/\text{year} \times 4.80$$

$$\frac{\mu\text{Ci}/\text{day}}{\mu\text{Ci}/\text{m}^2\text{-day}} (\text{Table II.E-8}) \times 4.82 \times 10^{-2} \text{ rem}/\mu\text{Ci} (\text{Table II.E-5})$$

$$\times 10^3 \text{ millirems}/\text{rem} \times 10^6 \text{ cm}^3/\text{cm}^3 = 7.28 \times 10^{-2} \text{ millirem}/\text{year}.$$

(b) Milk

$$1.0 \times 10^{-15} \mu\text{Ci}/\text{cm}^3 \times 10^{-2} \text{ m/sec} \times 3.15 \times 10^7 \text{ sec/year} \times 3.20 \times 10^{-1} \frac{\mu\text{Ci}/\text{day}}{\mu\text{Ci}/\text{m}^2\text{-day}} (\text{Table II.E-8}) \times 4.82 \times 10^{-2} \text{ rem}/\mu\text{Ci} (\text{Table II.E-5}) \\ \times 10^3 \text{ millirems/rem} \times 10^6 \text{ cm}^3/\text{cm}^3 = 4.85 \times 10^{-3} \text{ millirem/year.}$$

(c) Beef

$$1.0 \times 10^{-15} \mu\text{Ci}/\text{cm}^3 \times 10^{-2} \text{ m/sec} \times 3.15 \times 10^7 \text{ sec/year} \times 3.00 \times 10^{-2} \frac{\mu\text{Ci}/\text{day}}{\mu\text{Ci}/\text{m}^2\text{-day}} (\text{Table II.E-8}) \times 4.82 \times 10^{-2} \text{ rem}/\mu\text{Ci} (\text{Table II.E-5}) \\ \times 10^3 \text{ millirems/rem} \times 10^6 \text{ cm}^3/\text{m}^3 = 4.55 \times 10^{-4} \text{ millirem/year.}$$

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12. R. S. Booth, S. V. Kaye, and P. S. Rohwer, "A Systems Analysis Methodology for Predicting Dose to Man from a Radioactively Contaminated Terrestrial Environment,": p. 877 in Proceedings of the Third National Symposium on Radioecology, May 10-12, 1971, Oak Ridge, Tennessee.

Appendix II.F

HANFORD WASTE TANK LEAK INCIDENT OF JUNE 8, 1973

II.F.1 SUMMARY OF OCCURRENCE

On June 8, 1973, at 4:25 p.m., the Production and Waste Management Programs Division, AEC Richland Operations Office (AEC-RL), was informed by the Atlantic Richfield Hanford Company (ARHCO) that a leak had occurred in tank 106 T in the T Tank Farm located in the 200 W (West Area) on the Hanford Reservation, located approximately 25 miles northwest of Richland, Washington. Tank 106 T is one of the twelve 533,000-gal underground tanks in the T Tank Farm. The T Tank Farm also contains four 55,000-gal underground tanks. All of these tanks are used for storage of high-level radioactive wastes in the waste management program conducted by ARHCO for the Atomic Energy Commission.

The total loss of fluid to the ground was subsequently determined to have been approximately 115,000 gal, containing approximately 40,000 Ci of cesium-137; 14,000 Ci of strontium-90; 4 Ci of plutonium; and various fission products.

A plan was immediately developed for pumping the remaining liquid waste from the tank into other available tank space. Necessary equipment was installed, and during the interval between 10:00 p.m. on Friday, June 8, 1973, and noon on Sunday, June 10, 1973, essentially all remaining liquid radioactive wastes (except for those interstitial liquors in the sludge and about 24,000 gals of supernate) were pumped from 106 T into tank 112 T, a tank within the same tank farm. This transfer was accomplished without problems.

There have been no injuries to onsite or offsite personnel as a result of the leak nor any damages to property other than incremental contamination of sediments beneath the tank farm.

On June 12, 1973, Mr. T. A. Nemzek, Manager, AEC-RL, appointed an Investigation Committee to obtain complete and factual information regarding this occurrence. The committee, under the chairmanship of Mr. Alex Fremling, Deputy Manager, AEC-RL, carried out the investigation and issued a report¹ in July 1973. This description of the incident is based primarily on information contained in the Investigation Committee report.

II.F.2 BACKGROUND HISTORY OF HANFORD WASTE MANAGEMENT OPERATIONS

For nearly 29 years, radioactive liquid wastes have been generated at Hanford as a result of the chemical processing of fuels from the reactors to separate out the plutonium, uranium, and by-product isotopes. In the early Hanford operations, the wastes were sufficiently dilute that the fission-product-generated heat contained in the waste did not present any thermal problems with regard to storage in underground tanks. Underground tanks constructed during the 1943-1950 period dissipated the heat generated from these "low heat wastes" without the liquid attaining a boiling temperature. (Although the heat output of these wastes is relatively low, they are characterized as high-level wastes because of the total fission-product content and in relation to other low-level wastes generated during chemical processing.)

Technology later improved to the point that a much smaller volume of waste was generated per ton of fuel processed. This "high-heat waste" was stored in modified tanks designed to permit boiling under controlled conditions. The vapors were condensed, and the condensate either discharged to a "crib" or returned to the tank as a cooled liquid to provide supplementary cooling.

In 1965, an evaporative process was implemented to solidify the "low-heat wastes" to a salt cake and render the wastes less mobile and less likely to migrate in the event a breach should occur in the containment structure. Earlier, an effort had been initiated to develop methods of removing enough of the long-lived heat-producing isotopes from the "high-heat wastes" so that they, too, could become "low-heat wastes" and could be solidified by the same evaporative processes.

In late 1967, a Fractionization Processing Facility (B Plant) was placed in operation which selectively removed the principal heat-producing isotopes, cesium and strontium, from the "high-heat wastes." In addition, the natural decay mechanism was relied on for removal of a large portion of the heat associated with short-lived fission products.

The resultant composite waste-solidification program entails removal of cesium and strontium followed by aging for about five years to convert the "high-heat wastes" to "low-heat wastes," followed by evaporation of the "low-heat waste" to produce an immobile salt cake that can be stored more safely than liquids in the existing tanks. The program also includes solidification of the removed cesium and strontium by other processes, and subsequent storage of the cesium and strontium in doubly encapsulated packages in a specially constructed water-cooled basin. A flow diagram of the composite waste management program is shown in Figure II.F-1.

II.F-3

FLOW DIAGRAM FOR HIGH LEVEL LIQUID
WASTE MANAGEMENT AT HANFORD

Figure II.F-1

Between 1943 and the present, 151 tanks were constructed for the storage of liquid high-level radioactive wastes generated at Hanford. Two more tanks are currently being constructed. All tanks constructed before 1965 were of concrete with a single carbon-steel liner on the base and sides of the concrete vessel. Two tanks completed after 1965 and the two tanks now under construction are stress-relieved (double-walled) tanks within a steel-lined concrete vessel.

These 151 tanks are located in 13 "tank farms" in the 200 East and West Areas on a plateau near the center of the 570-square-mile Hanford Reservation. They are of varying sizes, the smallest being 55,000 gal and the largest 1,000,000 gal. The total waste holdings in the tanks on the Hanford Reservation currently consist of approximately 42,000,000 gal of liquids and 22,000,000 gal of solids in the form of salt cake and sludges. There are approximately 12,000,000 gal of spare volume available for liquid storage within 20 tanks on the site.

The T Tank Farm is one of the oldest of the Hanford tank farms, having been constructed during 1943-44. There are 16 tanks in the farm, 12 having a capacity of 533,000 gal each, and 4 with a capacity of 55,000 gal.

As in the case with all tanks built prior to 1965, T Farm tanks are constructed of reinforced concrete with a single carbon-steel liner on the base and sides of the concrete vessel. They are cylindrical in shape with a domed top and are positioned underground with at least 6 ft of earth cover above the tank dome. Tank 106 T (shown in Figure II.F-2) is a 533,000-gal tank and is approximately 75 ft in diameter and 33 ft high. The top of the tank is approximately 7 ft beneath the surface of the ground, and the bottom of the tank is 162 ft above the water table.

Two systems for monitoring waste tanks for detection of leaks have been in use since storage of waste in liquid form was started at Hanford in 1944: (1) a liquid-level measuring system, and (2) a dry-well radiation monitoring system.

The liquid-level system involves direct measurement of the elevation of the surface of the liquid through a penetration in the tank dome. The first-generation liquid-level gauge was a conductivity tape. The accuracy was affected by the presence of foaming and wave action, and readings taken in succession could vary by 1 or more in. A new gauge with an accuracy of ± 0.1 in was developed and became available in 1971. These gauges were installed on 74 tanks, including 106 T, and

II.F-5

HANFORD 106-T WASTE STORAGE TANK

Figure II.F-2

became operational in November 1972. They require manual readout and a comparison of previous readings to determine a change in liquid level.

The dry-well monitoring system obtains radiation readings in vertical dry wells (cased openings) along the outside of the tanks. The monitoring device records on a strip chart showing variations in radiation levels as a function of ground depth.

The waste stored in tank 106 T on June 8, 1973, was originally generated at the PUREX Plant. It had been subjected to an appropriate decay period in boiling waste tanks during which the majority of the strontium precipitated, and the supernate was processed in B Plant by ion exchange for cesium removal. The ion exchange waste was pumped to tank 106 T for storage prior to subsequent salt cake solidification in the 242-T Evaporator system.

II.F.3 CAUSES AND RESPONSIBILITY FOR THE LEAK

The leak in question began on or about April 20, 1973, during filling of the previously "empty" (except for sludge, interstitial liquors, and residual supernate) tank. The leak was not, however, recognized as such until June 8, 1973. According to ARHCO calculations, had the leak been recognized at the earliest possible time under the existing leak detection mechanisms, the volume of leaked material could have been limited to between approximately 26,700 and 37,600 gal.

The exact cause of the leak is still unknown and may never be determined, but it most probably resulted from corrosion of the aging (29-30 years old) carbon steel tank by the caustic waste solutions to which it had been exposed during its lifetime.

The immediate cause of the failure to recognize the leak earlier and minimize the quantities of fluid lost was the failure of ARHCO employees to promptly review liquid level and radiation level data. The investigating committee determined that the following factors contributed substantially to the failure to detect the leak earlier:

- (1) Although there are formal Standard Operating Procedures and memoranda for obtaining the liquid-level and dry-well data in the 200 West Area, the procedures were deficient in that there was no apparent basis for the interval for leak detection monitoring. These procedures and memoranda did not specify:

- (a) the precise day during the month or week on which the data were to be obtained and recorded,
 - (b) the specific procedures by which the data were to be recorded, or
 - (c) the timing and the system for transmittal of the data to those persons responsible for the evaluation and review of the data.
- (2) Some written and oral instructions had been issued by ARHCO pertaining to the areas of deficiency listed in (1) above. However, these instructions had not been systematically developed; instead, they had evolved over the years based upon experience of working level and supervisory people.
- (3) The written and oral instructions in effect had neither been consistently applied nor completely understood by personnel responsible for following them.
- (4) There was no effective redundancy in the system to assure that a leak undetected by those workers primarily responsible for detection would be detected by someone else or to alert management's attention to any breakdown in the system.
- (5) There was no well-defined formalized training program for the tank farm operators nor was there any comprehensive audit performed of operator qualifications.
- (6) Although there was no evidence that faulty equipment or lack of maintenance contributed to the late detection of the leak in question, there were, at the time of the leak, no systematic preventive maintenance programs for monitoring equipment, and reliance had been placed on operators to judge when corrective maintenance was needed.
- (7) There was no evidence that the function of recording, transmittal, and evaluation of leak detection data generated in the tank farms had been reviewed by ARHCO management above the Tank Farm Subsection level. The Manufacturing and Operations Support Engineering Departments perform formal and "informal" audits, but these had been irregular and too informal and had not been conducted in depth.

AEC program management and safety reviews of the ARHCO tank farm operation did not reveal the absence of formalized leak detection procedures or the lack of application of informal methods of obtaining and evaluating leak detection data. Although the AEC's Richland Operations Office, beginning in early 1972, initiated action to require ARHCO to establish and implement detailed quality assurance programs for all critical operations, the absence of such programs at the time of the occurrence was a contributing factor to the failure to detect the leak earlier.

II.F.4 ENVIRONMENTAL EFFECTS OF THE LEAK

There have been 16 leaks in 15 years from the 151 storage tanks at Hanford. None of the radioactive liquids from these leaks has ever reached the water table or endangered the health and safety of either the general public or workers at the Hanford complex.

For reasons of security and safety, the Hanford Reservation was established in an arid, remote part of Southeastern Washington during World War II to produce plutonium for nuclear weapons. The plateau area where the surface is 200 to 300 ft above the water table was selected for the processing and storage of the liquid high-level radioactive wastes resulting from the chemical processing of nuclear fuel used in the Hanford reactors. Only one reactor is now operating; the others have been shut down.

In earlier leaks, the largest of which was 70,000 gal, the liquid penetrated the sediments beneath and around the tanks for several tens of feet, spreading laterally as it moved downward. In no case did the liquids penetrate to within 100 feet of the water table. At the time of the investigation committee report, studies of actual disposition of the fluid lost in the 115,000-gal leak had not been completed, but preliminary data indicated that the fluids were following the predictable pattern of fluid flow that had been observed and studied for more than 25 years at Hanford. Core sampling and radiation profiles from two dry wells drilled as of July 6, 1973, indicated that (1) the leaked fluids had moved downward with significant lateral spreading immediately beneath the tank and (2) the deepest measured penetration of radioactivity was 45 ft below the tank bottom. Existing groundwater monitoring wells in the immediate area showed no changes in radiation levels.

Since the initial investigation, a more detailed study has been completed and a report on the findings has been issued by the Atlantic Richfield Hanford Company.² Sixteen wells were drilled to delineate the contaminated zone. The deepest

penetration observed in the wells was 87 ft below the surface and approximately 47 ft below the tank bottom or 160 ft above the water table.

Long-lived radionuclides, such as strontium and cesium and the trace amounts of plutonium contained in the wastes, have always in the past been trapped in the sediments just a few feet below the tanks and have not been found in liquids that penetrate to the deepest point. This lack of movement proved to be the case in this instance also.²

The groundwater beneath the tanks moves so slowly toward the Columbia River that it would take at least 800 years to reach the river several miles away. In addition, the strontium-90 and cesium-137--even if they should reach the groundwater--would be further impeded due to the sorption characteristics of the sediments through which the groundwater flows.

Thus, any such radioactivity would have decayed to insignificant levels long before reaching the river.

II.F.5 STEPS TO PREVENT FUTURE INCIDENTS

After reviewing the investigative report, the Manager of the AEC's Richland Operations Office ordered that the following actions³ be taken:

- (1) Expedite the installation of an automatic leak detecting and warning system. This system will be centrally monitored so that corrective action can be taken immediately to minimize any future leaks.
- (2) Implement a procedure by ARHCO and the AEC to assure that there are independent verifications that the tank monitoring system is functioning properly.
- (3) Engage outside consultants to recommend improved methods for determining the structural integrity of existing tanks before refilling.
- (4) Accelerate the in-tank solidification program and determine the need for additional tank storage capacity.
- (5) Review the contractor organization and make recommendations concerning any appropriate changes required to improve the operation. Atlantic Richfield Chemical Company was asked to expedite a management review of its operation at Richland.

REFERENCES FOR APPENDIX II.F

1. USAEC, Richland Operations Office, "Report on the Investigation of the 106 T Tank Leak at the Hanford Reservation, Richland, Washington," Richland, Washington, July 1973.
2. Atlantic Richfield Hanford Company, "241-T-106 Tank Leak Investigation," Report ARH-2874, November 1973.
3. USAEC, Richland Operations Office, "AEC Issues Report on Investigation of Leak In Waste Storage Tank at Hanford," Press Release RLO-73-25.

Appendix II.G - Plutonium Toxicity

The sections of this appendix are titled to correspond to the major subsections of Section 4.7, and present more detailed information bearing on the discussions of that section. The appendix concludes with a comprehensive bibliography.

II.G.1 Source Term

In the LMFBR fuel cycle there are five prospective sources for the release of transuranic elements to the environment. These are the reactor, the fuel reprocessing plant, the fuel fabrication plant, the waste storage facilities, and transportation between these elements. In the following discussion releases are normalized to a generating capacity of a 1000 MWe LMFBR power plant operating for a year at an 80% capacity factor. Thus, for a fuel reprocessing plant designed to handle the fuel for eighty 1000 MWe reactors, the transuranic effluents released per year are divided by 80 to obtain the amount released per 1000 MWe-year equivalent of fuel reprocessed.

Two types of releases are considered: those predicted to result from routine operations and those resulting from accidents. The nature and magnitudes of these releases are considered elsewhere in Section 4. The treatment in this appendix is designed merely to draw these pieces together in the form of a single source term applicable to the estimation of health effects in the general population.

II.G.1.1 Releases Resulting from Routine Operations

The estimated releases of α -emitting, transuranic elements from normal operations of the LMFBR fuel cycle are summarized in Table II.G-1. The principal airborne release is that from the fuel reprocessing plant, totaling 0.36 mCi* per 1000 MWe-year; other airborne releases are small by comparison. Estimates of health effects have therefore been based on an atmospheric source term with the characteristics of the release from the fuel reprocessing plant.

Liquid transuranic effluents are assumed to occur only for the fuel fabrication plant and are estimated at 0.05 mCi per 1000 MWe-year. This release has been ignored in the estimation of health effects because it is small in comparison to the airborne release, and because the environmental dispersal of a liquid transuranic effluent is a site-specific problem that would, of course, need to be considered in connection with the environmental impact of a specific plant; it is

*In the Draft Statement this figure was erroneously stated to be about 1 mCi due to a decimal point error in the estimated release of curium-242.

Table II.G-1

ESTIMATED RELEASES OF TRANSURANIC ELEMENTS DURING NORMAL OPERATION OF THE LMFBR FUEL CYCLE

Facility or Process	Release Mechanisms	Protective Systems	Nature of Release	Radio-nuclide	Quantity (Ci/1000 MWe Yr)	Reference ^a
II.G-2	Fuel Reprocessing	Process equipment, cells, and HEPA ^b filters (CF ^e = 2 x 10 ⁹)	Airborne particles of PuO ₂ or Pu(NO ₃) ₄ AMAD ^c ~ 0.3 μm	Pu-238	.18 x 10 ⁻³	4.4.5.3 T 4.4-5
				Pu-239	.04 x 10 ⁻³	
				Pu-240	.05 x 10 ⁻³	
				(Pu-241) ^d	(5.4 x 10 ⁻³)	
				Am-241	.01 x 10 ⁻³	
				Cm-242	.07 x 10 ⁻³	
				Cm-244	.01 x 10 ⁻³	
				Total α	.36 x 10 ⁻³	
	Fuel Fabrication	Glove boxes, roughing and HEPA filters (CF = 10 ¹²)	liquid		zero	4.4.4.3
			Airborne particles of PuO ₂ AMAD ~ 0.3 μm	Pu-238	.36 x 10 ⁻⁶	4.3.5.3.1 T 4.3-5
				Pu-239	.07 x 10 ⁻⁶	
				Pu-240	.10 x 10 ⁻⁶	
				(Pu-241) ^d	(10 x 10 ⁻⁶)	
				Am-241	.02 x 10 ⁻⁶	
				Total α	.55 x 10 ⁻⁶	
			liquid	Pu-238	.030 x 10 ⁻³	4.3.5.3.1 T 4.3-6
				Pu-239	.006 x 10 ⁻³	
				Pu-240	.008 x 10 ⁻³	
				(Pu-241) ^d	(.91 x 10 ⁻³)	
				Am-241	.002 x 10 ⁻³	
				Total α	.046 x 10 ⁻³	
Reactor	Failed fuel to sodium to cover gas	Cover gas purification system			negligible	4.2.3.2.4

Table II.G-1 (continued)

Facility or Process	Release Mechanisms	Protective Systems	Nature of Release	Radio-nuclide	Quantity (Ci/1000 MWe Yr)	Reference ^a
Waste storage	Exhaust air from receiving and handling areas	HEPA filters			$< 1.5 \times 10^{-6}$ ^f	4.6.3.1
Transportation					zero	4.5.4 II.M II.N

^aReferences are to other sections of this Environmental Statement. "I" indicates table number.

^bHEPA = high efficiency particulate.

^cAMAD = activity median aerodynamic diameter.

^dBeta emitter not included in totals. The 5.4×10^{-3} Ci of Pu-241 decays to produce an additional 0.16×10^{-3} Ci of Am-241, which is included as a part of the source term, where appropriate.

^eCF = Confinement factor = Ci processed/Ci released.

^fThe estimated normal release of radioactivity from waste storage is 1 mCi/yr (4.6.3.1); this includes all radio-isotopes, the isotopic distribution of which will depend upon a number of variables. The RSSF is forecast to hold all of the commercial high-level waste produced through the Year 2000, including all types of reactors. In order to be conservative it is assumed that the entire 1 mCi consists of α -emitting transuranics solely from the LMFBR power capacity during a given year only (1999). (This ignores cumulative power production up to 1999 and radio-isotopes other than the α -emitting transuranics.) In 1999 LMFBR generating capacity is estimated at 178 1000 MWe-year [derived from the metric tons of uranium fabricated in that year (3090 - Sect. 4.1) and the metric tons of uranium required per 1000 MWe-year reactor (17.335 - Fig. 4.5-1)]. In addition, there will be 20,170 metric tons of uranium fabricated for other types of reactors (Table 4.1-1). Assuming that the quantity of high level waste is approximately the same from all types of reactors (see 9.1.2), then

$$1 \text{ mCi} \div \frac{3090/23260}{178} = 1.3 \times 10^{-3} \text{ mCi, or } < 1.5 \times 10^{-6} \text{ Ci.}$$

only in this generic consideration of the total fuel cycle that it can be considered as insignificant.

Plutonium-241, a beta emitter, has not been included in the summed release totals, since the energy of its emissions are insignificant, biologically, when compared to the α -emitters. It will, however, decay to produce Am-241, a long-lived α -emitter, in quantities that are significant (an additional 0.16 mCi/1000 MWe-year). This ingrowing americium is included, in dose calculations, wherever pertinent.

Although Table II.G-1 lists the chemical form of the particles as either PuO_2 or $\text{Pu}(\text{NO}_3)_4$, the actual form is more complex because of the uranium oxide which is part of the fuel, the other transuranics which are produced, and the variety of inert materials that might dilute the transuranic elements. Americium and curium are removed in the reprocessing plant and are not passed on to the fuel fabrication part of the cycle. The small amount of Am-241 formed by the radioactive decay of Pu-241 in the recycled plutonium is included in the effluents expected from fuel fabrication operations. Because of uncertainties as to the chemical composition and specific activity of transuranic particles released from the LMFBR fuel cycle, conservative assumptions regarding solubility are employed in estimating potential health consequences.

The released particles are assumed to have an activity median aerodynamic diameter (AMAD) of 0.3 μm .* Evidence supporting this assumption is included in Section 4.3.5.2 of this Environmental Impact Statement. The actual size, shape, density, and other characteristics of the particles will, of course, be highly variable, and will undoubtedly change following dispersal to the environment.

The Natural Resources Defense Council (NRDC) has questioned the use of this AMAD value.** They state:

...while actual measurements in an operating plant give an AMAD of 1.0 to 4.0 μm , this Draft EIS assumes a different AMAD that is derived from laboratory experiments with liquid aerosols of dioctyl phthalate (DOP).

The measurements referred to were made upstream of the final stages of HEPA filtration prior to exhaust from the buildings and included all material reaching

*This is a method of describing an aerosol that consists of a spectrum of particle sizes. Half of the radioactivity is associated with particles whose settling speeds in air are less than the settling speed of 0.3 μm unit density spheres, and half of the radioactivity is associated with particles of larger settling speeds.

**Comment Letter 38 (NRDC), pp. 3-6.

that point in the exhaust system. Exhaust ducts from all sources converged upstream from the sampling site; all of these ducts contained HEPA filters at their source. However, some of these prior HEPA filters had reduced efficiencies due to the glove box atmospheres in which they were used. Filtration theory suggests that 0.3 μ m is the size of concern because these particles will have maximum penetration through the filters; this is supported by Westinghouse data (Westinghouse Electric Corp., "Environmental Report - Westinghouse Recycle Fuels Plant," Docket No. 70-1432, pp. 3.6-1 to 3.6-3 and Fig. 3.5-2, July, 1973). This issue is further addressed in the AEC response to this NRDC letter.

II.G.1.2 Releases Resulting from Accidents

Several commenters have questioned the treatment of unscheduled events.* NRDC stated that

There is no justification for the assertion that accidental releases 'would make a negligible contribution to the source term.'**

EIAP states,

More realistic estimates would reflect experience at Rocky Flats as well as cumulative experience with transportation accident losses during uranium fuel transport and processing. The magnitude of the airborne release under 'normal operating conditions' indicated in Table 4.G.1 amounts to only one millicurie per year per 1000 MWe LMFBR, a release of less than one part in 10^9 (one billionth) of the total activity of the actinides reprocessed per year per reactor. Can the AEC really guarantee such a remarkable degree of containment?+

The EPA suggests that without consideration of unplanned plutonium releases, and the accident frequency,

there is no meaningful plutonium source term...++

*Comment Letters 8 (G. Younghein and I. Younghein), encl. 1, pp. 2-7, encl. 2, pp. 5-9; 16 (J. T. Edsall), p. 1; 22 (MACEA), p. 7; 32, p. 3; 38 (NRDC), pp. 3, 6-11; 40 (EAC), p. 2; 51 (EIAP), pp. 23, 24, 53-59, 74, 75; 53 (EPA), p. 6; 57 (J. L. Menke), pp. 1, 2; 66 (DHEW), p. 1.

**Comment Letter 38 (NRDC), p. 11.

+Comment Letter 57 (EIAP), pp. 23, 24.

++Comment Letter 53 (EPA), p. 6.

In response to these comments a discussion of accident source terms is included here and in Section II.G.1.3.

The various parts of Section 4 dealing with elements of the LMFBR fuel cycle include discussions of assumed accident situations, along with estimates of resulting radioactive releases. Estimated releases of transuranic elements for major accident categories from these sources are summarized in Table II.G-2. Other types of accidents not listed would release much smaller amounts than those tabulated. If these numbers seem small, it must be remembered that the facilities are designed to contain accidental releases, and only those quantities of material estimated to pass all containment barriers are listed in Table II.G-2.

There is little applicable experience bearing on the question of how frequently such accidents might occur. Techniques of fault tree and event tree analysis have been developed which allow the estimation of accident probability in reactor systems, based on frequencies of failure of individual components. In the development of these techniques, several investigators^{1,2} have estimated accident probabilities in model LWR's. A major study of the probabilities and consequences of accidents in both PWR's and BWR's, as built, has been conducted for AEC by a group under the direction of Dr. Norman Rasmussen of MIT.³ While comparable probabilistic analyses for LMFBR plants and fuel cycle facilities are planned, their actual accomplishment must await completion of detailed designs. In the interim, it may be noted that a probabilistic study of risk for an assumed model of a 1000 MWe LMFBR was made by P. Godbout⁴ as part of the UCLA project on the Public Health Risks of Thermal Power Plants. Six types of potential major accidents in a LMFBR were investigated and the probabilities and consequences of the accidents determined. Results were presented in terms of accident envelopes which describe the hypothetical size of the release as a function of the probability for that release. Predicted amounts of plutonium which could be released into the atmosphere range from one microgram with a probability of between 10^{-4} and 10^{-5} per reactor year, to several grams with a probability of 10^{-20} per reactor year.

Fault tree analysis, together with known transportation statistics, have been applied to determine the hazards from accidents in shipping spent fuel, recycled plutonium, and high level radioactive waste.⁵ The results of this study indicated that total plutonium released annually due to transportation accidents would average less than 10^{-6} Ci per yr between now and the Year 2020.

Table II.G-2

ESTIMATED RELEASES OF TRANSURANIC ELEMENTS DUE TO MAJOR ACCIDENTS ASSOCIATED WITH THE LMFBR FUEL CYCLE

Facility or Process	Type of Accident	Nature of Release	Quantity of Alpha-activity Released (10^{-3} Ci)		Reference ^a
			per Accident	per 1000 MWe-Yr	
Fuel Reprocessing ^b	Leakage of Highly Radioactive Material	to air	1.8	.002	4.4.8.2.3 T 4.4-15
	Solvent Fire	to air	0.11	.0001	4.4.8.2.3 T 4.4-16
Fuel Fabrication ^b	General Facility Fire	to air	3.7	.005	4.3.8.2.2
	Criticality	to air	0.18	.0002	4.3.8.2.2
Reactor ^c	Class 9 Core Melt-down with Breach of Primary Containment	to air	37	.00004	4.2.7.8.3
Waste Storage ^d	Rupture of Dropped Canister	to air	0.4	.002	4.6.3.1.5
Transportation ^e	Railcar Fire	to air	1.4	.0003	4.N.4 T 4.5.4.3

FOOTNOTES FOR TABLE II.G-2

^aReferences are to other sections of the Environmental Statement. "T" indicates table number.

^bBeta emitter not included in totals.

^cAssumes one accident of the designated type, per fuel cycle facility, per 10-year interval (discussed in text).

^cThe calculated release of plutonium to the environment from a hypothetical core disruption accident in a 1000 MWe LMFBR is 0.1 gram (Sec. 4.2.7.8.3). Assuming that the alpha-emitting transuranic radioisotopes released in a HCDA consist of the same isotopic distribution, by weight, as that normally released by a fuel reprocessing plant (Table II.G-1), the activity released is as follows:

²³⁸ Pu	18.7 mCi
²³⁹ Pu	4.3 mCi
²⁴⁰ Pu	5.3 mCi
²⁴¹ Pu	577.4 mCi
²⁴¹ Am	1.1 mCi
²⁴² Cm	7.3 mCi
²⁴⁴ Cm	0.1 mCi

Total	614.2 mCi
Total α -emitting transuranics	36.8 mCi

The probability of a core meltdown with a release from the primary containment vessel with subsequent leakage of 0.1 gram plutonium through secondary containment for a 1000 MWe yr is less than 10^{-6} . Assuming a risk of 1 in 10^6 per year, or 1 in 10^6 reactor-years, then $36.8 \text{ mCi} \div 10^6 \text{ reactor-years}$ equals $36.8 \text{ mCi} \times 10^{-6}$, or 36.8 mCi per reactor-year. A maximum credible accident is estimated to release approximately 10 mCi of activity to the environment (Sec. 4.6.3.1.5). Of the waste inventory anticipated in the Year 2000, 4.5% is expected to consist of α -emitting transuranic elements ("Retrievable Surface Storage Facility Alternative Concepts: Engineering Studies," Atlantic Richfield Hanford Company, July, 1974, ARH-2888 REV, Table 10-II). Therefore, assuming a similar isotope distribution, of the 10 mCi released, 0.45 mCi would be α -emitting transuranic isotopes. If LMFBR power capacity (1999) is 178 1000 MWe-yr (see Table II.G.1-1), then the release per 1000 MWe-yr is 2.5 μ Ci. (This ignores power produced in previous years and so is conservative on a per unit basis.)

^eOf the 58 mCi Pu release given in 4.5.5.4, 2.4% consists of α -emitting radioisotopes (Appendix II.N.4). The number assumes a total of six 1000 mile alpha waste shipments required per 1000 MWe-year (4.5, Table 4.5-12) and 2.23 accidents per 10^6 railcar miles (Appendix II.C). There would be only $6 \times (2.23 \times 10^{-3}) / 10^6 = 13 \times 10^{-9}$ accidents per year involving alpha waste shipments by rail. About 1.5% of all rail accidents involve fire (Appendix II.A). Thus, there would be about 2×10^{-4} alpha waste rail car fires per 1000 MWe-year. Alternatively, there would be one such fire for every 5×10^3 1000 MWe-years. Thus, the addition of this accidental release would add $1.4 \text{ mCi} / 5000 = 0.0003 \text{ mCi}$ to the routine release of 0.36 mCi per 1000 MWe-year. Americium-241 buildup from the decay of plutonium-241 would increase this to 0.0006 mCi per 1000 MWe yr.

A summary of operational accidents in AEC facilities, from 1943 to 1970, is given in WASH-1192.⁶ For those facilities and operations having a general resemblance to the various LMFBR fuel cycle steps, in the areas of fuel reprocessing, fuel fabrication, and scrap recovery, there have been a number of accidents of the type considered as major examples in Table II.G-2. Those which involved the possibility of environmental release include the following:

- (1) Five solution criticality events (1958--2 events, 1959, 1961, and 1962) in reprocessing or recovery operations involving highly enriched uranium or plutonium. All were of small consequence in terms of property damage or releases of radioactivity to the environment, but one fatality and several high radiation exposures occurred among operating personnel.^{6,7}
- (2) Chemical explosion in evaporator (1953),⁸ related to fuel reprocessing.
- (3) Explosion and fire in plutonium purification facility (1963).⁹
- (4) Metallic fuel fire (reactive metal) in process dissolver (1960),¹⁰ related to fuel reprocessing.
- (5) Spontaneous fire in radioactively contaminated, combustible waste (1951).¹¹
- (6) Two fires at the Rocky Flats plutonium fabrication and recovery facility, (1957)¹² and (1969).¹³ The 1969 fire caused \$45 million in property damage. Both fires are attributable to spontaneous ignition of plutonium metal, which is not involved in the LMFBR fuel cycle.
- (7) Fire around an anion exchange column, fuel reprocessing plant (1964 - ref. 6, p. 81).

Many of these accidents occurred during the early years of operations with new facilities and newly developed technologies. All were investigated and corrective actions were taken (e.g., design changes) to make the events unlikely of recurrence. Such corrective actions have been carried forward, where applicable, into design practices for new facilities, both government and commercial. During the past decade, criticality accidents have disappeared from the accident scene, and fire or explosion involving reactive metals has become the predominant major accident.

On a comparable basis, accident experience in commercial facilities to date include:

- (1) A solution criticality accident in recovery operations involving highly enriched uranium, fatal to operator (1964).⁷
- (2) An explosion in a metal fuel scrap dissolver (reactive metal - 1966).¹⁴

- (3) A series of dissolver "fires" (reactive metal), fuel reprocessing plant (1967-1968).¹⁵
- (4) Final HEPA filter bank failure (inadequate mechanical support), fuel reprocessing plant (1968).¹⁶
- (5) Fire in plutonium contaminated wastes, fuel fabrication facility (1973).¹⁷
There was no detectable release of plutonium to the environment.
- (6) Explosion in plutonium glove box, fuel fabrication facility (1972).¹⁸
About 5.6 μ Ci of alpha activity was released via the stack.

In this very limited (statistically speaking) experience, individual facilities - many in operation between 10 and 20 years - have experienced from zero to two major accidents of the type under consideration. It seems prudent to assume that LMFBR fuel reprocessing plants and fuel fabrication plants will have a comparable accident frequency experience, at least during early phases of operation. A generous estimate would be one such accident per fuel cycle facility per 10-year interval.

The last column of Table II.G-2 lists the contribution to the 1000 MWe-year transuranic source term from the tabulated accident types, assuming that fuel fabrication and fuel reprocessing accidents occur at 10-year intervals and that other accidents occur as described in footnotes to the table. The total estimated contribution from accidents is 0.010 mCi, which is small in comparison to the estimated routine release of 0.36 mCi. It is therefore concluded that the 0.36 mCi/1000 MWe-year source term accounts adequately for both routine and accidental transuranic releases.

II.G.1.3 Adverse Experience at the Rocky Flats Plant

E. A. Martell* points to experience at the Rocky Flats plutonium fabrication and recovery plant as indicating that high confinement factors are not achieved in practice.¹⁹ He makes reference to the out-of-doors storage of drums of plutonium-contaminated machine oil at the Rocky Flats site during 1958-68. Leakage of an estimated 85 grams of plutonium (5.3 curies) to the soil occurred before the drums were removed for reprocessing, and some (perhaps half) of the leaked plutonium was spread locally by wind and rain action.^{20,21,22,23}

In this context, Dr. Martell states:

Now, if you have a perfectly ideal containment system in an ideal technology, 10^{-9} might represent the airborne effluents that you

*Comment Letter 51 (EIAP), pp. 23, 24; Hearing Testimony (E. A. Martell), pp. 232 ff.

release. But that doesn't indicate the magnitude of releases for all accidental, all transportation accidents, all fires and explosions, human error, acts of God and other things. I think we should look at our experience in this area. (Ref. 19, p. 235.)

I suggest the impact statement would be improved if we based our discussion on experience at Rocky Flats, with transportation accidents and with accountability of fuels in the past. (Ref. 19, p. 237.)

A related point is made by G. A. Younghein* to the effect that human experience in general, and experience in the nuclear field thus far, shows that mankind cannot be relied upon to achieve the degree of perfection needed to assure the high degree of containment assumed in this statement. Mr. Younghein concludes that the risks with plutonium are just too great, as have others who have submitted comments on the Draft Statement.**

The adverse experiences in the nuclear field to date are a matter of record, and, in the context of this discussion, the experiences at the Rocky Flats plant are most prominent among them. There is no question that adverse experience must be taken into account when projecting future expectations, but the most constructive use of past adverse experience is as guidance for present and future actions. The assessment of future expectations properly includes an assessment of all pertinent experience, good and bad, along with an assessment of procedures, practices, regulatory guides, regulations, laws (e.g., NEPA), improved technology, etc. that have come into being and have an important bearing on future actions.

The measured and estimated quantities of long-lived alpha activity released from the Rocky Flats plant during its first 20 years of operation is summarized in Table II.G-3.²³ The removal of contaminated oil from the protected confines of the plant for long-term outside storage effectively compromised the plant's ability to confine alpha activity. It did so to a much greater extent than did the 1957 fire, which extensively damaged the containment filter system of the building in which the fire occurred. The clear message of these events has not gone unheeded.

Practices adopted for the design, construction, and operation of government facilities, and regulations (which have the force of law) governing comparable practices in licensed commercial facilities have evolved as a result of past experience. Among other things, they are intended to make past adverse experience, such as that at Rocky Flats, highly unlikely of recurrence. Moreover, recognizing

*Comment Letter 8 (G. A. Younghein & I. Younghein), encl. 2, pp. 5-9.

**Comment Letter I (B. Simon), p. 3; 3 (W. Beaver), p. 3; 16 (J. T. Edsall), pp. 1-2; 57 (J. Menke), pp. 1-2; 58 (C. R. Lewis), p. 2.

that human error and equipment failures are unavoidable in an absolute sense, considerable effort has gone into the continuing development of regulations and regulatory guides to provide criteria for the increased protection of the public, operating personnel and facilities in the event that such errors or failures do occur.

Table II.G-3

LONG-LIVED ALPHA ACTIVITY RELEASED FROM ROCKY FLATS

Date	Circumstances	Quantity (Assumed to be Pu-239)
1958-1968	Leakage of Pu contaminated machine oil stored at the Rocky Flats site	5.3 Ci to soil at drum storage area
1957	Fire in Bldg. 771 resulting in major damage to filter system; less than \$1 million property damage	60 mCi, airborne, mostly during fire
1969	Fire in Bldg. 776 - costs; \$10 M - decontamination \$16.5M - restoration of capability \$13.5M - safety improvements	0.2 mCi, airborne, over 6-day period during and after fire
1953-1970	Normal effluent releases (cumulative)	< 41 mCi, airborne ^a 91 mCi, liquid effluents

^a18 mCi could be subtracted from this amount as a contribution of the 1957 fire or, more specifically, as an indication of high samples observed in October 1957 from contamination in the ductwork and plenum following restoration of the filter system. Additionally it should be noted that daily stack samples for airborne alpha particulates were normally below detectable limits (see discussion in Section 4.3.5.3) so that the cumulative numerical value is derived by taking the minimum detectable quantity as the maximum possible release. One can infer that normal alpha particulate releases averaged something less than 1.3 mCi per year:

$$\frac{<41 - 18 \text{ mCi}}{18 \text{ years}} = < 1.3 \text{ mCi/yr.}$$

The specific and extensive modifications made to all plutonium handling facilities at Rocky Flats subsequent to the 1957 fire (especially the substitution of flame resistant filters for those formerly used, and the addition of fire protection in the filter banks and plenums) were clearly responsible for the vastly improved containment of alpha activity during the 1969 fire. The new plutonium recovery facility now under construction at Rocky Flats (as a replacement for older facilities)²² is being built under criteria (Ref. 22, Appendix A) that should provide even greater assurance that the facility will be able to confine plutonium releases to exceedingly

small values, even under severe abnormal circumstances - including natural events, such as tornadoes.

The criteria governing the design and construction of the new Rocky Flats facility apply to all comparable government facilities under construction or to be built in the future. Strictly comparable criteria exist as Regulatory Guides for commercial facilities that process or fabricate plutonium.^{24,25,26,27} The assumed characteristics of the various model LMFBR fuel cycle facilities, and the assessment of future expectations with respect to their performance under routine and accident circumstances, are based on the currently applied regulations, guides, criteria, and practices.

II.G.2 Environmental Dispersal

II.G.2.1 Atmospheric Dispersal

A release point at Morris, Illinois will be used for environmental dispersal calculations since actual sites have not yet been determined. This site is considered to be an appropriately conservative choice, tending to maximize population exposure estimates. The densely populated Chicago area is only about 40 miles downwind and the populous eastern seaboard also lies in the path of the plume. Plant sites in the less populous western states or the South Atlantic states would produce lower population exposures. For calculation of radionuclide concentrations downwind from the release point, the first two phases of a model for the dispersion of noble gases developed by Machta, et al.²⁸ was utilized; this model was recently revised to include deposition of particulate effluents.^{29,30} For this analysis five-day plume trajectories from Morris, Illinois were computer-calculated four times per day for one month out of each season to approximate mean annual plume dispersion. Air concentrations of non-depositing effluents are calculated along each trajectory using a gaussian plume formula for a continuous point source:

$$\text{where } x = \frac{Q}{\pi \sigma_y u \sqrt{2K_z t}} e^{-1/2(y/\sigma_y)^2} \quad (1)$$

where x = ground level air concentration (Ci/m^3)

Q = Emission rate (Ci/sec)

σ_y = crosswind standard deviation of the plume

K_z = vertical diffusion coefficient (m^2/sec) (Assumes $\sigma_z = \sqrt{2K_z t}$)

t = plume travel time (sec)

u = wind speed along trajectory (m/sec)

y = distance from plume centerline (meters)

II.G.2.2 Deposition

Since the transuranic release occurs in the form of small particles ($0.3 \mu\text{m}$ AMAD), these will deposit on the ground and other surfaces with resultant depletion of their concentration in air. This depletion will occur both by dry deposition and by precipitation scavenging.

The rate of dry deposition has been characterized in empirical studies by a "deposition velocity" which is the ratio of the rate of deposition per unit area to the concentration at a specified height (usually one meter).

Dry deposition (C_i/m^2) along each trajectory is calculated from

$$D_d = \chi V_d (\Delta t) \quad (2)$$

where χ = ground level air concentration (C_i/m^3)

V_d = deposition velocity (m/sec)

Δt = time interval (sec) represented by each trajectory (e.g., for 4 trajectories per day $\Delta t = 6 \text{ hrs}$ or $2.16 \times 10^4 \text{ sec}$).

Deposition velocity depends upon a number of variables including particle size, nature and roughness of the surface, and atmospheric conditions.

Simpson³¹ at Hanford, and Islitzer and Dumbauld³² at the National Reactor Testing Station (NRTS), using tracer particles with mean diameters of $2.5 \mu\text{m}$ and $1.0 \mu\text{m}$, respectively, (larger than estimated for LMFBR releases) estimated deposition velocities ranging from 0.2 to 9.2 cm/sec, with the lower values occurring at stronger stabilities. Studies of the deposition of fission products generated by an electrical arc, reported by Chamberlain,³³ yielded a range of deposition velocities from 0.07 to 0.3 cm/sec. These particles were of small size, $\sim 0.1 \mu\text{m}$. A study of fallout materials by Small³⁴ gave annual average deposition velocities for 1957 and 1958 at Kjeller, Norway, of 0.51 and 0.75 cm/sec.

Wind tunnel studies by Sehmel³⁵ demonstrate dependence of deposition velocity on particle size, wind speed, and surface type, and show that deposition velocities are at a minimum, and are nearly constant, over the particle diameter range from 0.1 to $1 \mu\text{m}$. For deposition on artificial grass with wind speeds in the tunnel of 5 mph and 30 mph, the deposition velocities were 0.08 cm/sec and 0.2 cm/sec, respectively, for $0.3 \mu\text{m}$ particles. A model based upon dimensionless correlation of the wind tunnel deposition velocity data for all particle diameters indicates

that deposition velocities increase rapidly with both increasing wind speed and increasing surface roughness and are a minor function of atmospheric stability.^{36,37} When projected to a natural roughness height of 3 cm, the model gives a deposition velocity of about 0.2 cm/sec for 1 m/sec winds and 0.8 cm/sec for 7 m/sec winds. A deposition velocity of 0.3 cm/sec was chosen as most appropriate for the sub-micron particles of interest here. (This was changed from 0.1 cm/sec in agreement with the EPA contention* that a deposition velocity substantially greater than 0.1 cm/sec would be more appropriate and might substantially influence the fraction entering man. The change did not significantly affect the fraction entering man via inhalation.)

Deposition by scavenging during periods of precipitation provides a second method of cloud depletion and ground deposition. Wet deposition was calculated using a scavenging ratio (sometimes called the washout ratio)³⁸ which is the ratio between the concentration in rain water and in air. That is, $E = x_w/x_a$

where,

x_w = concentration in rain water (Ci/m^3) at the ground.

x_a = mean concentration in the column of air from the ground to the top of the rain-bearing layer (Ci/m^3)

Wet deposition (Ci/m^2) along a trajectory is calculated from

$$D_w = \frac{x \sqrt{\pi K_z t}}{\Delta z} EP(\Delta t) \quad (3)$$

where $x \sqrt{\pi K_z t} / \Delta z$ = the mean air concentration in the column of height Δz

Δz = depth of rain bearing layer (m)

E = scavenging ratio

P = mean annual precipitation rate (m/sec)

Δt = time interval represented by each trajectory

The choice of an appropriate scavenging ratio is difficult since the ratio varies by a factor of ten or more and depends upon the synoptic rainfall type, the rate of rainfall, stability of the atmosphere, particle characteristics and other variables, none of which have been thoroughly explored. Some recent work in the St. Louis area using existing aerosols of Cu, Fe, Pb, Mg, Mn, and Zn indicate a change in the ratio with particle size.³⁹ A value of 500 (mass ratio) was chosen for this calculation since it is about at the middle of the range of measured

*Comment Letter 53 (EPA), p. 7.

values.³⁸ This is equivalent to a ratio of 4.2×10^5 by volume, as used in the model.*

Incorporating both wet and dry depletion into Eq. 1, the complete equation²⁹ for calculation of ground level air concentration along each trajectory is:

$$x = \frac{Q}{\pi \sigma_y u \sqrt{2K_z t}} e^{-1/2(y/\sigma_y)^2} e^{-2V_d \sqrt{t/\sqrt{\pi K_z}}} e^{-E P t / \Delta z} \quad (4)$$

where the second exponential term accounts for dry depletion and the third exponential accounts for wet depletion. Parameter values used in this calculation were

$$\begin{aligned} \sigma_y &= 1853 \text{ m/hr} \times t \text{ (hrs)} \\ K_z &= 5 \text{ m}^2/\text{sec} \\ V_d &= 0.3 \text{ cm/sec} = .003 \text{ m/sec} \\ E &= 500 \text{ (by mass)} = 4.2 \times 10^5 \text{ by volume} \\ P &= 40 \text{ in/yr} = 3.2 \times 10^{-8} \text{ m/sec} \\ \Delta z &= 4000 \text{ m} \end{aligned}$$

Figure II.G-1 shows the mean annual air concentration pattern obtained by averaging the contributions from all plume trajectories** at each grid point.

Population exposure to a distance of 70 miles was obtained by multiplying average air concentrations by population in annular segments formed by 22-1/2 degree sectors subdivided into 10-mile radial increments. Beyond 70 miles, the estimated average concentration for each state was multiplied by the population of the state. For the Morris, Illinois, site used in this analysis, more than half of the total population exposure occurs within 50 miles of the plant. With the assumed wet and dry deposition rates only about 1 percent of the material emitted remains airborne after the plume crosses the Atlantic so there is virtually no additional population exposure beyond the east coast of North America. Assuming a breathing rate of 10^4 m^3 per year† the amount inhaled is calculated to be 4.4×10^{-6} of the total annual emission. This result forms the basis for estimates of total inhalation in Section II.G.2.4.

This value is most likely an overestimate of the inhalation exposure from a typical reprocessing plant for several reasons. First, the population distribution about the plant site chosen for this calculation tends to maximize the total exposure.

*Using this value the fraction removed by wet deposition agrees closely with values calculated assuming a mean residence time of 3.3 days, as measured for radon daughter products.⁴⁰

**Includes wet and dry deposition.

†See footnote - page II.G-23.

MEAN ANNUAL SURFACE AIR CONCENTRATIONS AS A
FRACTION OF ANNUAL EMISSION (UNITS OF $10^{-20}/m^3$)

Figure II.G-1

Model Simulations²⁸ indicate, for example, that population exposure would be almost a factor of four lower for a plant site in South Carolina.

Second, the model does not take into account the initial height of the plume emitted from a reprocessing plant stack (about 100 meters). The plume height effect would tend to reduce ground level air concentrations near the source while increasing concentrations at greater distances since there would be less dry deposition near the source. The effect of plume height would be a net reduction in inhalation exposure, most likely by less than a factor of two.

As discussed earlier, there is considerable uncertainty concerning the most realistic values for wet and dry deposition parameters. However, the total population exposure is not very sensitive to these parameters. For example, Ferber, et al.²⁹ have made additional computer calculations, varying both wet and dry deposition parameters by a factor of ten. Resulting population inhalation fractions are shown in Table II.G-4.

Table II.G-4
POPULATION INHALATION FRACTIONS AS A FUNCTION
OF DRY AND WET DEPOSITION PARAMETERS

	Dry Deposition Velocity (cm/sec)	Wet Deposition Scavenging Ratio (E)	Time (days) to remove		Fraction Inhaled (10 ⁻⁶)
			50%	99%	
Low removal	0.1	100	6	60	6.6
Best guess	0.3	500	1	10	4.4
High removal	1.0	1000	0.2	3	2.5

With the high removal rate 99% of the effluent is removed from the atmosphere in 3 days while with the low removal rate it takes 60 days to remove the same fraction; yet the total amount*inhaled by the world population varies less than a factor of two from the "best guess" result used for this analysis.

Considering the uncertainties and site-dependent variability of meteorological factors, deposition parameters and population distribution, the fraction of the effluent inhaled is expected to lie in the range from 10⁻⁵ to 10⁻⁶, most probably in the lower part of that range. An additional increment to inhalation exposure, arising from resuspension of deposited material, is considered in the next section.

*Directly.

II.G.2.3 Resuspension

Particulate material reaching the ground can be resuspended by winds or by mechanical disturbance of the soil, thus allowing inhalation of particles long after cloud passage. Completely general models which might be used to predict the air concentrations resulting from resuspension are not developed to the point where they can be usefully employed for present purposes.^{41,42,43} Most data on resuspension have been reported in the form of a resuspension factor given by:

$$K = \frac{\text{Resuspended air concentration (Ci/m}^3\text{)}}{\text{Ground deposit (Ci/m}^2\text{)}}.$$

Application of this ratio involved the obviously questionable assumption that the material in the air originated at the point of measurement rather than upwind. It will be used for the present treatment, however, since the data are available in this form.

Resuspension factors have been reported for a variety of conditions;⁴⁴⁻⁵¹ values range from about $10^{-2}/\text{m}$ to $10^{-13}/\text{m}$. The higher values were measured in buildings with disturbances due to movement or working. Outdoor values range from $10^{-4}/\text{m}$ downward. This information has been reviewed and appropriate values recommended by several investigators. For early resuspension Stewart⁴⁵ recommended a value of $10^{-6}/\text{m}$ as appropriate for quiescent conditions outdoors and $10^{-5}/\text{m}$ for regions of moderate activity. Langham⁴⁷ suggested a value of $10^{-6}/\text{m}$ as a reasonable average value. Kathren,⁵² based primarily on the recommendations of Stewart and Langham, used $10^{-4}/\text{m}$ as a conservative value for standard-setting purposes.

Most measurements of resuspension factors have been made in arid country, frequently close to sources of mechanical disturbance. The application of such a factor to a well vegetated area, such as is considered here, is uncertain. One might expect that the initial factor would be higher than in an arid region due to deposition on foliage with a higher susceptibility for removal than from the ground. However, evidence from studies with fallout indicates that such material is removed from the vegetation with a half-life of days to weeks and is, presumably, transferred to the soil, which is protected from erosion or resuspension by the vegetation and root mat. The damp soil, often typical of areas in which the bulk of the population resides, should minimize resuspension.

A resuspension factor of $10^{-5}/\text{m}$ was chosen as adequately representing the initial conditions for the average population in the relatively well vegetated area of

interest. This corresponds to Stewart's recommendation for regions of moderate activity.⁴⁵

There is evidence from several experiments^{47,48,50} that the value of the resuspension factor decreases with time after deposition. This is interpreted as due to weathering processes causing the particles to bind more tightly to the soil or to move deeper into the soil profile. Data from these experiments cover only a relatively short period of time but are consistent with a resuspension factor half-life of 45-70 days over a period of up to 11 months following deposition.

The value which the resuspension factor approaches at long times following deposition can be estimated from measurements made above soil contaminated many years ago, and from consideration of other tracers such as naturally occurring ²³⁸U. Resuspension factors for plutonium of 10^{-9} and 10^{-10} /m were measured at the Nevada Test Site in an area contaminated 17 years previously.⁵³ Measurements of plutonium in the vicinity of the Rocky Flats plant,⁵⁴ several years after deposition, indicate a resuspension factor of 10^{-9} /m; however, when only the surface soil was measured by the use of sticky paper, the value approaches 10^{-6} /m, indicating that the lower value was due to penetration into the soil.

The EPA expressed concern regarding the variability in measured resuspension factors at Rocky Flats.* The previous discussion provides additional information.

The use of ²³⁸U as a more general tracer for estimating aged resuspension factors avoids the complications of sources exhibiting large horizontal concentration gradients; it suffers, however, from uncertainty with regard to the origin of the uranium measured in air, which may, in some cases, be largely from industrial processes. From Hamilton's⁵⁵ representative air and soil concentration measurements obtained at Sutton, England, a resuspension factor of 5×10^{-9} /m can be calculated. A similar value may be calculated from measurements of Sedlet at Argonne, Illinois.⁵⁶ A two-fold larger value is indicated by measurements in New York City, where industrial pollution is more likely.⁵⁷

Based on the above data, we have assumed that the initial resuspension factor of 10^{-5} /m would diminish with a half-life of 50 days until it reached a value of 10^{-9} /m, after 1.8 yr, at which level it remains for the life of the radionuclide. One would expect that this factor would continue to diminish with a lengthening

*Comment Letter 53 (EPA), p. 7.

half-life, but in the absence of data over long time periods, we have made the conservative assumption of constant resuspension.

For a continuously emitted plume which is diffusing, depositing and resuspending, the air concentration at any time, t , due to resuspension is considered the product of the surface concentration and the resuspension factor. The surface concentration, however, is an accumulation of deposits from the various elements of the plume passing by during succeeding periods of time up to the present. Each deposit has decayed, radioactively, over the period of time since being deposited and is subject to a resuspendability which has also decreased due to weathering over that same period of time. Integrating the contributions to resuspension from deposits made during all prior periods of plume passage, the ratio of resuspended air concentration to the initial plume concentration at some time T after releases began is found to be:

$$\frac{\bar{X}_R}{\bar{X}_0} = \frac{V_D K_0}{\lambda_D + \lambda_R} [1 - e^{-(\lambda_D + \lambda_R) T}] , T \leq t_a \quad (5)$$

$$\frac{\bar{X}_R}{\bar{X}_0} = \frac{V_D K_0}{\lambda_D + \lambda_R} [1 - e^{-(\lambda_D + \lambda_R) t_a}] + \frac{V_D K_m}{\lambda_R} [e^{-\lambda_R t_a} - e^{-\lambda_R T}] , T > t_a \quad (6)$$

The first term of Equation (6) is the contribution made by the deposits accumulated over the past 1.8 years (t_a), during which time the resuspension factor was considered to decay exponentially from $K_0 = 10^{-5}/m$, with a decay constant of $\lambda_D = 0.693/50$ days; λ_R is the radioactive decay constant of the radionuclide being considered. The second term of Equation (6) is the contribution made from material deposited more than 1.8 years previous, where the resuspension factor was considered constant at $K_m = 10^{-9}/m$.

The time integrated concentration (i.e., exposure, E) accumulated since the release was initiated until some time, T , must be considered together with population distribution and breathing rate to obtain the quantity of radionuclide inhaled from resuspension. The ratio of this exposure to that received directly from the initial plume is found to be, for the first 1.82 years after initiation of the plume,

$$\frac{E_R}{E_0} = \frac{V_D K_0}{\lambda_D + \lambda_R} \left[1 - \frac{1 - e^{-(\lambda_D + \lambda_R) t_a}}{(\lambda_D + \lambda_R) t_a} \right] , T = 1.82 \text{ yr} \quad (7)$$

For λ_R very small compared with λ_D and $e^{-(\lambda_D + \lambda_R) t_a}$ very much less than unity, this becomes, approximately,

$$\frac{E_R}{E_0} \approx \frac{V_D K_0}{\lambda_D}, T = 1.82 \text{ yr} \quad (8)$$

To account for the total inhalation during the full period of the decay of the radionuclide, T can be made equal to infinity, and the ratio of resuspended to initial plume exposure becomes,

$$\frac{E_R}{E_0} = \frac{V_D K_0}{\lambda_D + \lambda_R} [1 - e^{-(\lambda_D + \lambda_R) t_a}] + \frac{V_D K_m}{\lambda_R} e^{-\lambda_R t_a}, T \rightarrow \infty \quad (9)$$

or again, with λ_R small compared to λ_D ,

$$\frac{E_R}{E_0} = \frac{V_D K_0}{\lambda_D} + \frac{V_D K_m}{\lambda_R} e^{-\lambda_R t_a} \quad (10)$$

The second term can be considered as the contribution due to resuspension after 1.82 yr of weathering of all deposits.

For solution of these equations, a deposition velocity of 1 cm/sec was used to account for the sum of wet and dry deposition. At long distances this value is in reasonable accord with the sum of wet and dry deposition and the corresponding air concentrations. At closer distances to the source, an overall deposition velocity of about 0.4 cm/sec might be more appropriate. The use of a deposition velocity of 1 cm/sec may therefore overestimate exposure from resuspension.

The simplified Equation (10) is adequate for the radionuclides of interest, except Cm-242. The case of Am-241 is also somewhat more complex, since ingrowth from decay of Pu-241 must be taken into account.

The applicability of the above treatment of resuspension to special locations such as low rainfall areas, plowed fields, and urban situations, has been questioned by EIAP and EPA.* While such highly variable factors must certainly be taken into account when considering the location of any specific fuel cycle facility, the more general treatment described above is appropriate to this Environmental Statement and

*Comment Letters 51 (EIAP), pp. 24-26; 53 (EPA), p. 8; Hearing Testimony (E. A. Martell), p. 237.

will lead to generally conservative estimates of the fraction entering man by resuspension.

II.G.2.4 Estimate of Quantity Inhaled

The estimated total intake by direct inhalation was calculated from the population distribution and the annual air concentrations summarized in Figure II.G-1, by multiplying the air concentration in a given region by the number of people represented and the breathing rate per person ($10^4 \text{ m}^3/\text{yr}$).^{*} The total intake, due to resuspension, over the life of the radionuclide, can then be calculated using the exposure ratios derived in the preceding section. The resulting estimates of quantity inhaled are given in Table II.G-5.

The values from Table II.G-5 for the fraction of the LMFBR transuranic release inhaled may be compared to experience with fallout plutonium. An estimated 16,000 Ci of plutonium has been deposited on the United States.⁵⁹ If 4×10^{-6} of this had been directly inhaled, as predicted in Table II.G-5, this would amount to 0.06 Ci inhaled, or 3×10^{-10} Ci per person. Of this quantity inhaled, about 10% should be retained (see Appendix II.G.4) many years after exposure, or 3×10^{-11} Ci per person. This is about ten times higher than measured values in man⁶⁰ (Table II.G-16, Appendix II.G.5.1). Such a discrepancy might be expected, however, since fallout plutonium is more evenly distributed over generally less populous areas, and is more subject to precipitation scavenging because of its stratospheric origin. The fallout plutonium data may therefore be considered in general agreement with the estimates of direct inhalation in Table II.G-5.

New York City data on the ratio of plutonium surface air concentration to yearly deposition are shown in Table II.G-6.⁶⁰ Values for this ratio are an estimate of the maximum resuspension factor for plutonium deposited during the previous year. This estimate is too high by a factor of possibly 10, since most of the plutonium in air is known (from inventories) to be coming from the stratosphere rather than from the soil. Thus, from fallout experience, an average resuspension factor of $10^{-7}/\text{m}$ during the first year following deposition would seem to be a reasonable estimate. The estimates of inhalation from early resuspension in Table II.G-5 assume an initial resuspension factor of $10^{-5}/\text{m}$, which reduces to $10^{-7}/\text{m}$ only after some 330 days.

^{*}This is a conservatively high estimate of breathing rate. A value of $7300 \text{ m}^3/\text{year}$ is employed by ICRP.⁵⁸

Table II.G-5

INHALATION OF TRANSURANIC ELEMENTS BY EXPOSED U.S. POPULATION FOR 1000 MWe-YEAR OF LMFBR RELEASES

Radio-Nuclide	Released to Air (10^{-3} Ci)	Directly Inhaled (10^{-9} Ci)	Inhaled Following Resuspension			Total Inhaled (10^{-9} Ci)
			Early** (10^{-9} Ci)	Late*** (10^{-9} Ci)	Total (10^{-9} Ci)	
Pu-238	.18	0.72	0.44	0.03	0.47	1.2
Pu-239	.04	0.16	0.10	1.76	1.86	2.0
Pu-240	.05	0.20	0.12	0.60	0.72	0.9
Am-241	.01*	0.04	0.04	0.12	0.17	0.2
Cm-242	.07	0.28	0.12	0.00	0.12	0.4
Cm-244	.01	0.04	0.01	0.00	0.01	0.05
Total	.36	1.44	0.83	2.51	3.35	4.8
Fraction of Release		$[4 \times 10^{-6}]$	$[2 \times 10^{-6}]$	$[7 \times 10^{-6}]$	$[9 \times 10^{-6}]$	$[1.3 \times 10^{-5}]$

*Ingrowth of additional Am-241 from decay of 5.4 mCi of Pu-241 was considered in the calculation of inhalation following resuspension.

**Inhaled during first 1.8 years following deposition, while resuspension factor is decaying from 10^{-5} to 10^{-9} /m.

***Inhaled at times longer than 1.8 years following deposition, while resuspension factor is constant at 10^{-9} /m.

Table II.G-6

NEW YORK CITY PLUTONIUM FALLOUT DATA⁶⁰

1	2	3	4
<u>Year</u>	<u>Concentration in Surface Air (10^{-15} Ci/m³)</u>	<u>Annual Incremental Deposition (10^{-9} Ci/m²)</u>	<u>Ratio-Col. 2 to Col. 3 (10^{-6}/m)</u>
1963	1.68*	.62	2.7
1964	.91*	.41	2.2
1965	.33	.14	2.4
1966	.12	.05	2.4
1967	.051	.04	1.3
1968	.080	.04	2.0
1969	.063	.06	1.0
1970	.065	.03	2.2
1971	.060	.03	2.0
1972	.031	.02	1.6

*Estimated by comparison with Sr-90 measurements. Plutonium values after 1964 are measured values.

II.G.3 Transport Via Food Chains

Transuranic 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. Concerning many of these routes there is little direct experimental data on transuranic element behavior. This appendix will consider the data on food chain mobilization of transuranic elements and develop a model for predicting ingestion by man via these routes.

Routes Via Plants to Man

This would appear to be the most direct and therefore the most probable route by which transuranic elements in the soil might reach man. Experimental data on plant uptake of these elements has been recently reviewed.^{61,62,63} In laboratory studies employing a variety of soil types, chemical treatments and compounds of plutonium, americium, and curium, 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 Am-241 in beans and barley grown on several soils and in nutrient solution;⁶⁴ values of 1.4×10^{-3} for Am-241, and 2.2×10^{-3} for Cm-244 were reported for tumbleweed, and lower values for cheatgrass.⁶⁵ However, these latter values were based on a minimum estimate of contaminated soil. If the elements are assumed to be distributed over the entire volume of the soil used, calculated concentration factors would be up to a factor of 25 higher. Concentration factors up to 1.8 were obtained with DTPA chelates of americium.^{66,67}

Two studies have shown increasing uptake of plutonium with time following soil contamination.^{68,69} In one of these there was a seven-fold increase in the plutonium concentration in clover during a 5-year cropping sequence.⁶⁹ 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 plutonium in the roots than in the shoots of certain plants. Whether a long-term continuing increase in transuranic 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. Our model 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.^{59,70} 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 II.G-7 summarizes data from large samples of foods obtained in New York City in 1972.⁷¹ 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 $2.65 \times 10^{-13} \text{ Ci/cm}^2$.⁶⁰ If plutonium is distributed uniformly through a 20 cm depth of soil 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 is 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 five-fold 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. 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.⁷¹

Routes Via Animals to Man

The absorption of transuranic 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,⁷² 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.⁷³ 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 II.G-7 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

Table II.G-7

PLUTONIUM IN NEW YORK FOODS--1972⁷¹

Food Item ^a	Plutonium	Per Capita Annual Intake	
	Concentration (10 ⁻¹² Ci/kg)	Food (kg/year)	Pu (10 ⁻¹² Ci/yr)
Plant-Derived			
Bakery Products	0.0085	44	.37
Whole Grain Products	.0060	11	.066
Fresh Fruit	.0051	59	.30
Dry Beans	.0049	3	.015
Fresh Vegetables	.0043	48	.21
Root Vegetables	.0035	10	.035
Flour	.0028	34	.095
Rice	.0015	3	.004
Potatoes	.0013	38	.048
Macaroni	.0012	3	.004
Canned Vegetables	.0009	22	.019
Fruit Juice	< .0003	28	< .007
Canned Fruit	< .0002	11	< .002
Total Plant-Derived Diet		314	1.17 - 1.18
Animal-Derived			
Shellfish	.011	1	.011
Poultry	.0033	20	.066
Meat	.0026	79	.20
Fresh Fish	.0016	8	.013
Eggs	.0012	15	.019
Milk	< .0003	200	< .064
Total Animal-Derived Diet		323	.31 - .37
Total Diet			1.5

^aFood categories and quantities are based on Department of Agriculture food purchase surveys.

^bValues are calculated from total per capita annual intake of food and of plutonium.

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.⁷⁴ Data for gastrointestinal absorption of americium and curium are available only for the rat, with maximum reported fractional absorption values of about 10^{-3} .⁷⁵

In any case, the animal-derived diet estimated in Table II.G-7 contributes only one-fifth of the total radioactivity. In view of the manifold complexities and uncertainties of the various animal food chains, these routes are neglected in the Final Statement and the conservatism of the numbers employed for the direct plant-to-man route is relied upon to make up for minor contributions from other routes.

Aquatic Food Chains

Although liquid effluents constitute a minor fraction of estimated LMFBR releases of transuranic 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 transuranic elements of certain algae, seaweed and invertebrates.⁷⁶ 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 (Table II.G-7).⁷¹ While not a significant factor in the total context, concentration of transuranic elements along food chains in aquatic environments should not be ignored in specific siting problems.

Estimation of Transuranics Ingested by Man

For estimating the total quantity of transuranic elements ingested by man it is assumed that the total released radionuclides are uniformly distributed in the upper 20 cm of U.S. soil. This change from the site-specific model employed in estimating inhalation simplifies calculations, but it should be noted that localized deposition may occur. Furthermore, the selection of a larger volume of soil, perhaps appropriate in certain cases, would tend to give increased plant/soil concentration factors. All food-chain routes are neglected except the principal, direct route from soil, via plants, to man. For this route it is assumed that the ingested plant-derived foods, as consumed, will contain a concentration of transuranic elements equal to 0.1 times the concentration of these elements in soil. This "food/soil" factor would correspond to about a five-fold higher "plant/soil" concentration factor, as

earlier defined, since the plant/soil factor was based on dry weight of plant. A plant/soil concentration factor of 0.5 is within an order of magnitude of the highest published values for americium (in the presence of chelates) and of the lowest published values for this element (in the absence of chelates).⁶⁷ Since, to date, published plutonium uptake values do not generally exceed those for americium, this value may be considered a reasonable upper limit for plutonium uptake. However, the potential exists that localized deposition in combination with certain soil conditions and cropping practices may serve to increase the concentration factors for both elements. Since foliar deposition from stratospheric fallout may increase the plutonium concentration in plants above that from root uptake alone, the selected value should serve as a reasonable estimate of americium uptake and a conservative estimate of plutonium uptake, on the basis of present information.

The following equation is employed in the calculations:

$$C_i = \frac{S_i N I D}{A d w} \cdot \frac{(T_{1/2})_i}{\ln 2} \quad (1)$$

where: C_i = Curies of isotope i ingested by man per 1000 MWe-year
 S_i = Curies of isotope i released and deposited on soil per 1000 MWe-year
 N = Population of U.S. = 3×10^8 persons
 I = Plant-derived food ingested per person per year = 3×10^5 g (from Table II.G-7)
 $D = \frac{\text{Concentration of isotope } i \text{ in food}}{\text{Concentration of isotope } i \text{ in soil}} = 0.1$
 A = Soil surface area of U.S. = 10^{17} cm²
 d = Depth of soil in which isotope i is uniformly dispersed = 20 cm
 w = Density of soil = 2 g/cm³
 $(T_{1/2})_i$ = Radiological half-life of isotope i in years.
 $\frac{(T_{1/2})_i}{\ln 2}$ = Average lifetime of isotope i in years.

The assumed population value is intentionally high since it applies over an extended period of future time.

Values for the estimated quantity of each radionuclide ingested are tabulated in Table II.G-8. Although the total ingested is nearly 1000 times higher than the total inhaled (Table II.G-5), the fraction absorbed from the gastrointestinal tract (as discussed in Appendix II.G.4) is very small. When this fraction is applied to the quantity ingested, in the final column of Table II.G-8, ingestion is seen to be comparable to inhalation.

Table II.G-8

FOOD CHAIN MOBILIZATION OF TRANSURANIC ELEMENTS FROM SOIL TO MAN FOR 1000 MWe-YEAR OF LMFBR RELEASES

1 Radionuclide	2		3		4		5		6	
	Transuranic Input to Soil		Transuranics in Plant-Derived Food Ingested by Man		Transuranics Absorbed by Man from Food Ingested					
	Total (Ci)	Initial Conc. ^a (Ci/g)	Total ^b (Ci)	Initial Conc. ^c (Ci/g)	Total ^b (Ci)	Initial Conc. ^c (Ci/g)	Total ^b (Ci)	Initial Conc. ^c (Ci/g)	Total ^b (Ci)	Initial Conc. ^c (Ci/g)
Pu-238	$.18 \times 10^{-3}$	4.5×10^{-23}	$.05 \times 10^{-6}$	4.5×10^{-24}	$.02 \times 10^{-10}$					
Pu-239	$.04 \times 10^{-3}$	1.0×10^{-23}	$3. \times 10^{-6}$	1.0×10^{-24}	$1. \times 10^{-10}$					
Pu-240	$.05 \times 10^{-3}$	1.2×10^{-23}	$1. \times 10^{-6}$	1.2×10^{-24}	$.3 \times 10^{-10}$					
Am-241	$.17 \times 10^{-3e}$	4.2×10^{-23}	$.3 \times 10^{-6}$	4.2×10^{-24}	$3. \times 10^{-10}$					
Cm-242	$.07 \times 10^{-3}$	1.8×10^{-23}	$.0001 \times 10^{-6}$	1.8×10^{-24}	$.001 \times 10^{-10}$					
Cm-244	$.01 \times 10^{-3}$	0.2×10^{-23}	$.0006 \times 10^{-6}$	0.2×10^{-24}	$.006 \times 10^{-10}$					
Total	$.52 \times 10^{-3e}$	13×10^{-23}	4.4×10^{-6}	13×10^{-24}	4.3×10^{-10}					

^aAssumes uniform distribution in top 20 cm of U.S. soil of density 2.0, or 4×10^{18} g of soil. Because of the delayed input of most of the Am-241, the concentration based on total input slightly overestimates initial concentration.

^bCalculated using Equation (1) as described in text.

^cValues from Column 3 multiplied by plant/soil concentration factor of 0.1.

^dValues from Column 4 multiplied by a gastrointestinal absorption fraction of 3×10^{-5} for plutonium and 10^{-3} for americium and curium (Appendix II.G.4).

^eThis number is larger than the original source term because of an added 0.16 mCi of Am-241 which results from the complete decay of the β -emitting Pu-241.

Because of its greater gastrointestinal absorption, americium is the dominant element reaching man by ingestion. This is additional justification for the conservative treatment of ingestion, since less is known about americium than about plutonium. Beyond about 1000 years post-release, Pu-239 and Pu-240 become the dominant elements, because of their longer half lives.

II.G.4 Metabolism and Dosimetry in Man

The metabolic model employed to describe the distribution and retention of trans-uranic elements in man is shown schematically in Figure II.G-2. Values employed for the parameters that define quantitatively the movement along various pathways are listed in Table II.G-9. With a few exceptions, discussed below, the treatment conforms to recommendations in the most recent applicable publications of the ICRP.^{58,72,77}

The behavior of inhaled material is described in the ICRP lung model⁷⁷ as modified in the report of the ICRP Task Group on the Metabolism of Compounds of Plutonium and other Actinides.⁷² This model allows for variation in initial deposition as a function of particle size. As applied to the source term size of 0.3 μm AMAD, the model predicts deposition fractions of 0.08 for the nasopharynx region, 0.08 for the tracheobronchial region, and 0.42 for the pulmonary region. The remaining 0.42 of the inhaled material is immediately exhaled.

The ICRP lung model describes the behavior of three types of compounds, designated Class D, W, or Y. Class D consists of compounds which are removed from the lung over a period of days; Class W, over a period of weeks; and Class Y, over a period of years. No compounds of plutonium, americium, or curium are considered to fall into Class D. Because of uncertainty concerning the chemical nature of the source term and subsequently redistributed materials, calculations of internal distribution and radiation dose were made for both Class W and Class Y, and the most conservative value chosen for each organ of deposition; e.g., the longer retention of Class Y compounds in the lung is assumed when considering hazard to the lung, and the greater deposition in bone and liver of Class W compounds is assumed when considering the hazard to these organs. While this will sometimes put the same radionuclide in two places at the same time, and clearly overestimates the total risk, it is considered preferable to a procedure that may underestimate the risk to a major organ of deposition. Further justification for this procedure is found in reported results from the application of the ICRP lung model to data on fallout plutonium in air; calculations using Class W parameters greatly underestimated the actual measured lung values in man, while Class Y parameters slightly underestimated values in bone and liver.⁶⁰

MODEL EMPLOYED FOR DESCRIBING THE METABOLISM OF TRANSURANIC ELEMENTS IN MAN (VALUES FOR PATHWAY PARAMETERS ARE GIVEN IN TABLE II.G.4-1)

Figure II.G-2

Table II.G-9

VALUES EMPLOYED FOR PARAMETERS DESCRIBING THE
METABOLISM OF TRANSURANIC ELEMENTS IN MAN

1	2	3	4	5
Source Region	Pathway**	Further Qualifi- cations***	Regional Fraction†	Biological Half-Life (days)
Total Inhaled	(1) exhaled		0.42	
	(2) to nasopharynx region of lung		0.08	
	(3) to tracheobronchial region of lung		0.08	
	(4) to pulmonary region of lung		0.42	
Total Ingested	(5) to gastrointestinal tract		1.0	
Lung				
Nasopharynx Region	(6) to gastrointestinal tract	Class W	0.9	0.4
		Class Y	0.99	0.4
	(7) to blood	Class W	0.1	0.01
		Class Y	0.01	0.01
Tracheobronchial Region	(8) to gastrointestinal tract	Class W	0.5	0.2
		Class Y	0.99	0.2
	(9) to blood	Class W	0.5	0.01
		Class Y	0.01	0.01
Pulmonary Region* (570 g)	(10) to gastrointestinal tract	Class W	0.4	1
			0.4	50
		Class Y	0.4	1
			0.4	500
	(11) to blood	Class W	0.15	50
		Class Y	0.05	500

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Table II.G-9 (continued)

1	2	3	4	5
Source Region	Pathway**	Further Qualifications***	Regional Fraction†	Biological Half-Life (days)
	(12) to lymph nodes	Class W Class Y	0.05 0.15	50 500
Pulmonary Lymph Nodes* (15 g)	(13) to blood	Class W Class Y	1.0 0.9 0.1	50 1000 ∞
Gastrointestinal Tract	(14) to blood	Pu Am & Cm	0.00003 0.001	
Blood	(15) to bone (16) to liver (17) to gonads	male female	0.45 0.45 0.0005 0.0001	
Mineral Bone* (5000 g)	(18) excreted		1.0	36500
Liver* (1800 g)	(19) excreted		1.0	14600
Gonads* (testes, 40 g, ovaries, 8 g)	(20) no loss		1.0	

* Organs for which dose commitments were calculated. Organ mass employed in calculations is shown in parentheses.

** Number in parentheses refers to labeled pathway in Fig. II.G-2.

*** This column indicates where values apply to a Class W or Class Y compound (see text), or to a specific element or organ.

† The fraction of the radionuclide in the source region (column 1) that is following the pathway indicated in Column 2, with the half-time indicated in Column 3.

As described by the model, the radionuclides leave the various regions of the lung via the gastrointestinal tract, the blood or the lymph nodes. Of the plutonium reaching the gastrointestinal tract, from the lung or by ingestion with food, 3×10^{-5} is absorbed to the blood. This is the absorption fraction employed by the ICRP.⁵⁸ A lower value of perhaps 10^{-6} would be justified for plutonium oxide particles.⁷²

The conservative higher value is retained for lack of specific data on the gastrointestinal absorption of plutonium in foods. The absorption of americium and curium compounds is generally greater than that of plutonium compounds,^{72,75} and a larger absorption fraction of 10^{-3} is assumed for these elements. For calculation of dose to the gastrointestinal tract itself, radionuclides are assumed to reside in the various sections of the tract according to the model developed by Eve.⁷⁸ Gastrointestinal tract dose is insignificant compared with doses to other organs and is ignored in subsequent discussion.

Plutonium reaching the blood stream, whether absorbed from the lung or from the gastrointestinal tract, is deposited principally in bone and liver. The ICRP recommended deposition fractions of 0.45 in bone and 0.45 in liver, with biological retention half times of 100 years in bone and 40 years in liver, were employed.⁷² Equal distribution between bone and liver is supported by human autopsy data on fallout plutonium (Table II.G-16).⁶⁰ Distribution of the remaining 0.1 of plutonium reaching blood was not specifically defined in the ICRP Task Group report. A fraction of 0.0005 deposited in the testes and 0.0001 deposited in the ovaries, both retained without loss, is assumed for present calculations. These values for deposition are best estimate values, based on data from a variety of experimental animals, including dogs, rabbits, pigs and rodents, and man.^{75,79} The assumption of retention without loss is, of course, conservative, reflecting an absence of experimental data on this point.

Except for the gastrointestinal absorption fractions, americium and curium are assumed to be distributed and retained in the same manner as plutonium.⁷⁵

Employing the above described metabolic model, in conjunction with radioactive decay parameters, one may calculate a time-integral of retention for each radionuclide in each tissue. This integration is made over a period of 70 years, i.e., as though the total exposure occurred at the beginning of a 70-year life span.* This is an obvious factor of conservatism.

*A 30-year time-integral of retention was calculated for the gonads, since genetically transmitted effects will accumulate only to the end of the reproductive age.

The time-integral of retention is proportional to the number of radioactive disintegrations occurring within the tissue and may be converted to a dose equivalent, in units of rem, according to the equation,

$$D = 51 I E/m \quad (1)$$

where I is the time-integral ($\mu\text{Ci-days}$) referred to above; E is the effective energy, i.e., energy absorbed in MeV multiplied by a quality factor Q and a distribution factor N; and m is the mass of the organ in grams. The constant 51 represents gram rads per $\mu\text{Ci-day}$ per MeV.

Values of dose equivalent per 10^{-9} Ci inhaled or ingested were calculated for each source term radionuclide and for each organ of potential concern. These values are listed in Tables II.G-10 to II.G-14.

A more detailed account of the dose calculation procedures and the background for these procedures may be obtained from publications of the ICRP,⁵⁸ NCRP,⁷² and MIRD.^{66,67*} Decay scheme data were processed by the method of Dillman.⁸³ It was assumed that all alpha energy is absorbed in the tissue where the disintegration occurs. Weights assumed for the organs were those of ICRP reference man⁸⁴ and are given in Table II.G-9. The 570 g pulmonary lung includes the weight of the bronchial tree and of capillary blood. The 5000 g skeleton includes cortical and trabecular bone but excludes marrow, cartilage and periarticular tissue.⁸⁴

In arriving at the effective energy, E, for equation (1), a quality factor (Q) of 10 was used for alpha radiation.^{58,80} For calculations involving bone an additional organ distribution factor (N) of 5 was included so that $QN = 50$.^{58,80} The quality factor of 10 relates to the high linear energy transfer (LET), and consequent greater biological effectiveness of alpha radiation. The distribution factor of 5 takes account of the experimentally demonstrated greater risk of malignancies from plutonium as compared to radium-226. All of these procedures conform to current ICRP recommendations.^{58,80}

EPA commented that AEC has ignored recent theoretical considerations of alkaline earth metabolism that could modify the N factor.** These theoretical concepts are not yet sufficiently developed to justify a deviation from ICRP procedures. This is discussed in more detail in response to the EPA Comment Letter.

*Medical Isotope Radiation Dose Panel, Society of Nuclear Medicine, Inc.

**Comment Letter 53 (EPA), p. 8.

Table II.G-10

LUNG DOSE EQUIVALENT ESTIMATED FOR 1000 MWe-YEAR
OF LMFBR TRANSURANIC ELEMENT RELEASES

1	2	3	4
Radio- nuclide	Quantity Inhaled* (10 ⁻⁹ Ci)	Dose-Equivalent per 10 ⁻⁹ Ci Inhaled** (rem)	Total Dose Equivalent*** (man-rem)
Pu-238	1.2	0.91	1.1
Pu-239	2.0	0.86	1.7
Pu-240	0.9	0.87	0.8
Pu-241	41.	0.002	0.07
Am-241	0.2	0.92	0.2
Cm-242	0.4	0.26	0.1
Cm-244	0.05	0.91	0.05
Total			<u>4.0</u>

*From Table II.G-5.

**From modified ICRP lung and dose models, as described in text.

***Product of Column 2 and Column 3.

Table II.G-11

BONE DOSE EQUIVALENT ESTIMATED FOR 1000 MWe-YEAR OF LMFBR TRANSURANIC ELEMENT RELEASES

1	2	3	4	5	6	7	8
			Dose Equivalent per 10 ⁻⁹ Ci***		Total Dose Equivalent		
Radio- nuclide	Quantity Inhaled* (10 ⁻⁹ Ci)	Quantity Ingested** (10 ⁻⁹ Ci)	Inhaled (rem)	Ingested (rem)	Inhaled [†] (man-rem)	Ingested ^{††} (man-rem)	Total (man-rem)
Pu-238	1.2	50	2.7	0.0006	3.2	0.03	3.2
Pu-239	2.0	3000	3.3	0.0007	6.6	2.1	8.7
Pu-240	0.9	1000	3.3	0.0007	3.0	0.7	3.7
Pu-241	41.	----	0.067	-----	2.7	----	2.7
Am-241	0.2	300	3.3	0.025	0.7	7.	7.7
Cm-242	0.4	0.1	0.050	0.0004	0.02	nil	0.02
Cm-244	0.05	0.6	1.4	0.011	0.07	0.007	0.08
Total					16.	10.	26.

*From Table II.G-5.

**From Table II.G-8.

***From modified ICRP lung, metabolic, and dose models, as described in text.

[†]Product of Column 2 and Column 4.

^{††}Product of Column 3 and Column 5.

Table II.G-12

LIVER DOSE EQUIVALENT ESTIMATED FOR 1000 MWe-YEAR OF LMFBR TRANSURANIC ELEMENT RELEASES

1	2	3	4	5	6	7	8
Radio nuclide	Quantity Inhaled* (10 ⁻⁹ Ci)	Quantity Ingested** (10 ⁻⁹ Ci)	Dose Equivalent per 10 ⁻⁹ Ci***		Total Dose Equivalent		
			Inhaled (rem)	Ingested (rem)	Inhaled [†] (man-rem)	Ingested ^{††} (man-rem)	Total (man-rem)
Pu-238	1.2	50	1.1	0.0003	1.3	0.02	1.3
Pu-239	2.0	3000	1.3	0.0003	2.6	0.9	3.5
Pu-240	0.9	1000	1.3	0.0003	1.2	0.3	1.5
Pu-241	41.	----	0.025	-----	1.0	----	1.0
Am-241	0.2	300	1.3	0.01	0.3	3.0	3.3
Cm-242	0.4	0.1	0.026	0.0002	0.01	nil	0.01
Cm-244	0.05	0.6	0.64	0.005	0.03	0.003	0.03
Total					6.4	4.2	10.6

*From Table II.G-5.

**From Table II.G-8.

***From modified ICRP lung, metabolic, and dose models as described in text.

[†]Product of Column 2 and Column 4.^{††}Product of Column 3 and Column 5.

II.G-40

Table II.G-13

THORACIC LYMPH NODE DOSE EQUIVALENT ESTIMATED FOR 1000 MWe-YEAR
OF LMFBR TRANSURANIC ELEMENT RELEASES

1	2	3	4
Radio-nuclide	Quantity Inhaled* (10 ⁻⁹ Ci)	Dose-Equivalent per 10 ⁻⁹ Ci Inhaled** (rem)	Total Dose Equivalent*** (man-rem)
Pu-238	1.2	39	47
Pu-239	2.0	44	88
Pu-240	0.9	44	40
Pu-241	41.	0.37	15
Am-241	0.2	45	9
Cm-242	0.4	0.82	0.3
Cm-244	0.05	24	<u>1.2</u>
Total			200

*From Table II.G-5.

**From modified ICRP lung and dose models as described in text.

***Product of Columns 2 and 3.

Table II.G-14

GONAD DOSE EQUIVALENT ESTIMATED FOR 1000 MWe-YEAR
OF LMFBR TRANSURANIC ELEMENT RELEASES

1	2	3	4	5	6	7	8
Radio-nuclide	Quantity Inhaled* (10 ⁻⁹ Ci)	Quantity Ingested** (10 ⁻⁹ Ci)	Dose Equivalent per 10 ⁻⁹ Ci***		Total Dose Equivalent		
			Inhaled (rem)	Ingested (rem)	Inhaled [†] (man-rem)	Ingested ^{††} (man-rem)	Total (man-rem)
Pu-238	1.2	50	0.044	0.00001	0.05	0.0005	0.05
Pu-239	2.0	3000	0.046	0.00001	0.09	0.03	0.12
Pu-240	0.9	1000	0.046	0.00001	0.04	0.01	0.05
Pu-241	41.	-----	0.0007	-----	0.03	-----	0.03
Am-241	0.2	300	0.048	0.0004	0.01	0.12	0.13
Cm-242	0.4	0.1	0.001	0.00001	nil	nil	nil
Cm-244	0.05	0.6	0.030	0.0002	nil	0.0001	0.0001
Total					0.22	0.16	0.38

*From Table II.G-5.

**From Table II.G-8.

***From modified ICRP lung, metabolic, and dose models as described in text.

†Product of Columns 2 and 4.

††Product of Columns 3 and 5.

No special calculations of dose have been made for the fetus, infant or child. The available data on metabolic behavior of the transuranic elements in the fetus, infant, and child indicate that no significant underestimate of hazard will result from considering the total population as adults.

Because of the very long retention of transuranic elements in man, most of the radiation dose from radionuclides deposited in the fetus, infant, or child, will be delivered when the child has grown to a much larger size. The radionuclide is not only diluted by this growth process, but, in the important instance of bone, is buried under new bone growth and its alpha particles largely shielded from the radiosensitive cells on the bone surface. Thus, the smaller intake of radionuclide by the infant or child results, over the life span, in a very much smaller dose than the model predicts using adult intake parameters; this despite the fact that dose during the period of infancy may not be overestimated, since intake is roughly proportional to size.

Cross-placental transmission of plutonium is very low, as indicated by studies in mice,⁸⁵ rats⁸⁵⁻⁸⁷ and dogs.⁸⁸ In only one instance has a higher concentration been reported in a fetal tissue than in the mother; this was the case of rat fetal bone, which showed a threefold higher concentration of plutonium than the mother's bone, 24 hours after injection of the near-term mother with citrate complexed plutonium.⁸⁶ Americium shows a smaller cross-placental transfer than plutonium.⁸⁷

Absorption of plutonium from the gastrointestinal tract is as much as 100-fold greater in the newborn rat than in the adult.⁸⁹ A similar effect of enhanced absorption of normally non-absorbed substances is observed in the newborn of other animal species.⁹⁰ If such an effect occurs in the human infant, it will probably persist for only a few days following birth. The predominantly milk diet consumed during this period is a very poor source of transuranic elements, as indicated by the absence of detectable amounts of fallout plutonium in milk (Table II.G-7).⁷¹

There are no experimental data on the retention of inhaled transuranic elements in the lung of immature animals, or on the subsequent behavior of these radionuclides following deposition in the lung. The adult model (Fig. II.G-2 and Table II.G-9) provides for deposition of 58% of the inhaled transuranic particles in the three regions of the lung, three-fourths of which is subsequently lost via the

gastrointestinal tract. On the physiologically untenable assumption of 100% retention of inhaled transuranium elements by the infant, the increase over predictions of quantities deposited in adult would amount to a factor of seven. This would be compensated by the smaller intake; about 1/15 of adult intake in the case of the newborn, and 1/5 in the case of the one-year-old.

For all of the above reasons, it is concluded that the use of adult parameters in modeling the U.S. population has not resulted in a significant underestimate of population dose. There remains the question of whether dose to the fetus, infant, or child may be of greater significance to the development of health effects because of a greater radiosensitivity of the young. Such an increased sensitivity of the youthful members of the population is explicitly factored into the cancer mortality risk estimates taken from the BEIR report and employed in the cancer mortality predictions of Table 4.7-5.⁹¹

There are few experimental data relevant to the effect of age at exposure on the late effects of deposited transuranic elements. An experiment in rats showed a comparable risk of bone tumor induction, relative to cumulative bone dose, for animals exposed as newborns, weanlings, or adults; the lowest tumor incidence per rad occurred in prenatally exposed animals.⁹² Effects on life span were most marked in the weanling and adult groups. Observations on humans injected with radium-224 suggest about a two-fold greater sensitivity to bone tumor induction in those exposed prior to age 20;⁹³ but the short, 3.62 day, half-life of Ra-224 prevents significant burial within bone of growing individuals, such as would occur with longer-lived transuranic elements.

II.G.5 Health Effects

Data relevant to the health effects of transuranic elements are considered in this appendix under four general headings, (1) experience with transuranic elements in man, (2) experience with natural radiation in man, (3) data on effects of other forms of radiation on man, and (4) animal data on plutonium toxicity. With respect to the latter two areas, consideration will be given to the general problems of extrapolation from animal to man, and extrapolation from the levels of exposure and effect that can be observed, to the very much lower exposure levels of concern in the exposure of man.

II.G.5.1 Human Exposures to Transuranic Elements

No serious health consequences have been caused in man by exposure to transuranic elements. Scanty evidence of effects at the cellular level--e.g., histologically

evidenced effects surrounding a plutonium-contaminated wound^{94,95} and possibly increased chromosome aberrations following accidental exposure⁹⁶--cannot be related to either exposure or ultimate effect in any manner useful to the present analysis. 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 world-wide plutonium (fallout from atmospheric testing of nuclear weapons and other devices), and the accidental exposure of plutonium workers.

Exposure to Fallout Plutonium

As a result of atmospheric testing of nuclear weapons, there has been deposited on the surface of the earth an estimated 320 kCi of long-lived plutonium isotopes, of which about 250 kCi is deposited on the northern hemisphere and 16 kCi on the United States.^{59,97} Measurements of this plutonium in the air, in soil, in foods, and in man constitute a natural experiment of special pertinence to the estimation of the biological behavior of transuranic element releases from the LMFBR. Various portions of these fallout plutonium data have already been utilized in the considerations of Appendices II.G.2, II.G.3, and II.G.4.

Estimates of human organ burdens and doses resulting from fallout plutonium have been made by Bennett, employing the ICRP lung model and the data on New York City plutonium air concentrations (measured since 1965 and estimated prior to that date by analogy with measured strontium-90).⁶⁰ Assuming no intake subsequent to 1972 and calculating dose commitment to the Year 2000, he arrived at the estimates of dose-equivalent shown in Table II.G-15. These estimates assume no intake by ingestion and involve a cumulative inhalation intake from 1954 to 1972 of 42 pCi (1 pCi = 10^{-12} Ci) per person, or 8 mCi total for a present U.S. population of about 2×10^8 . For comparison, Table II.G-15 also lists the estimated LMFBR-release doses per 1000 MWe plant operating for a year and for an estimated Year 2020 operation of 2,200,000 MWe of LMFBR capacity. In comparing these numbers it must be kept in mind that the fallout dose equivalent is that estimated for a single generation with dose commitment calculated to Year 2000, i.e., for 28-46 years following initial deposition; whereas, the LMFBR estimates are for all succeeding generations (about half the dose is delivered to the first generation) and is calculated for a 70-year dose commitment following initial deposition.

Direct measurements have been made of fallout plutonium 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 reported from the Los Alamos Scientific Laboratory on samples from Colorado and New Mexico.^{60,98}

Table II.G-15

ESTIMATED MAN-REM EXPOSURE FROM FALLOUT PLUTONIUM AND FROM
LMFBR TRANSURANIC RELEASES

Organ	Dose Equivalent To Current Generation from Fallout Pu ^a (man-rem)	Dose Equivalent to all Subsequent Generations from LMFBR Transuranic Releases (man-rem)	
		For Generating Capacity of 1000 MWe-Year ^b	For Year 2020 Generating Capacity of 2,200,000 MWe-year
Lung	3.0×10^6	4	0.9×10^4
Bone	1.4×10^6	17	$4. \times 10^4$
Liver	0.8×10^6	7	$2. \times 10^4$
Lymph nodes	100×10^6	200	40×10^4
Gonads	----	0.23	0.05×10^4

^aEstimate of Bennett, based on New York City air concentrations.⁶⁰

^bDerived in Tables II.G-10 to II.G-14.

These determinations employed larger tissue samples than have usually been available and improved analytical procedures. The measured organ burdens are compared in Table II.G-16 with calculated organ burdens based on the New York plutonium air concentrations and the ICRP lung model. The agreement is quite good, except for 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.

Table II.G-16
FALLOUT PLUTONIUM IN MAN
(Colorado and New Mexico, 1970-71)⁶⁰

Organ	No. of Samples	Plutonium in Organ (10^{-12} Ci)	
		Measured	Calculated from New York Air Concentration ^a
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

^aCalculated by Bennett using ICRP lung model.⁶⁰

While these data on fallout plutonium in man cannot be linked to any measured effect, they do suggest that effects from the very much smaller levels of transuranic elements that may be deposited in man as a result of LMFBR operations would be insignificant compared to this unmeasured effect of fallout plutonium.

Occupational Exposure

An estimated 16,000 persons have been employed in operations that might lead to significant plutonium exposure (Table II.G-18).⁹⁹ About 3000 of these have shown some measurable evidence of internal plutonium deposition. Data on employees of contractors operating AEC-owned facilities are summarized for the period 1957-1970 in Table II.G-17, which lists the numbers of persons with estimated plutonium depositions in excess of 10 nCi, grouped according to deposition level.¹⁰⁰ If all

persons are assumed to contain the minimum of their range, the total deposition in these 203 persons is about 5 μ Ci. This may still be too high a value, since estimates of individual deposition, based on urine assay, are almost always high. However, if one considers the much larger number of persons with deposits of less than 10 nCi, the persons exposed prior to 1957, and the exposed licensee employees not tabulated, then it seems reasonable to conclude that at least 5 μ Ci of plutonium are deposited in approximately 3000 workers.

Table II.G-17

USAEC CONTRACTOR EMPLOYEES WITH ESTIMATED PLUTONIUM
DEPOSITIONS IN EXCESS OF 10 NANOCURIES FOR
THE PERIOD 1957-1970^a

No. of Persons	Deposition Range (10^{-9} Ci)
118	10 - 20
35	20 - 30
13	30 - 40
15	40 - 80
15	80 - 200
7	200 - 400
<hr/>	
203	

^aData from Ref. 100, supplemented by more recent data from the AEC Division of Operational Safety.

Efforts are being made to obtain the maximum amount of information from these exposed persons. The U.S. Transuranium Registry was established for this purpose in 1968; it is operated by the Hanford Environmental Health Foundation, with cooperation from Pacific Northwest Laboratory, Los Alamos Scientific Laboratory, and the Rocky Flats Plant. The Registry seeks to identify potentially exposed workers, obtain their health physics and medical records, and their permission for postmortem sampling. The current status of Registry operations is summarized in Table II.G-18.⁹⁹ 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.

Table II.G-18

STATISTICS ON OCCUPATIONAL EXPOSURE STUDIES OF THE U.S. TRANSURANIUM REGISTRY - JUNE, 1974⁹⁹

Location	Estimate of Potentially Exposed Persons	Number of Persons for whom Registry has:			
		Specific Identification	Release of Health Physics & Medical Data	Authority for Autopsy	Autopsy Performed
Hanford	2300	2199	2132	525	12
Rocky Flats	2500	1504	1489	167	30
Los Alamos	3000	259	259	127	1
Savannah R.	5000	1559	0	0	0
Mound	500	322	0	0	1
Oak Ridge	200	0	0	0	1
Licensees	2500	0	0	0	0
Total	16,000	5,843	3,880	819	45

II.G-49

One group of exposed persons is of particular interest because of their early exposure, their relatively high level of exposure, and the thoroughness of the followup studies.¹⁰¹ These 25 laboratory workers were exposed during 1944 and 1945 while working at what is now the Los Alamos Scientific Laboratory, under conditions that would be judged crude by today's standards. Their individual, current, systemic body burdens are estimated to range from 5 to 420 nCi, totaling, in the aggregate, about 2.5 μ Ci. These burdens have now been retained for nearly 30 years. One of the 25 died of a coronary occlusion at age 38. The others show no medical findings attributable to internally deposited plutonium.

In view of the relatively small number of persons with sizable plutonium depositions, it seems unlikely that statistically valid inferences with regard to toxic effects will ever be made. This would not be true, however, if plutonium were markedly more toxic than is currently believed. Thus, the 2.5 μ Ci presently retained by the Los Alamos subjects must have amounted to about 10 μ Ci originally deposited, which, assuming a reasonable particle size distribution, would have contained 10^7 particles larger than 0.6 μ m in real diameter.¹⁰² If the cancer risk for such "hot particles" were 5×10^{-4} per particle, as postulated by Tamplin and Cochran,¹⁰³ about 5000 cancers should have been produced.¹⁰² If no effects attributable to plutonium are ever demonstrated in the population of exposed workers, the autopsy data on plutonium distribution will still be an invaluable link between observations in animals and predictions for man.

II.G.5.2 Human Exposures to Other Types of Natural Radiation

Experience with transuranic elements in man is limited to the present generation. Other alpha-emitting elements, however, are a natural part of man's environment. He has lived with these internally deposited radioelements and with radiation from other natural sources throughout the history of the species. A comparison of the radiation doses from these natural sources with those predicted to result from LMFBR transuranic releases should be instructive.

Table II.G-19 summarizes data from the UNSCEAR Report on naturally occurring alpha-emitters in human bone.¹⁰⁴ These total to about 400 pCi per person. Considering that only 80 to 85 percent of radium,* and about 70% of polonium-210** is present in bone, the total body content of α -emitters may approximate 500 pCi. This corresponds to 0.1 Ci in the total U.S. population of 2×10^8 persons, or 10,000 times the 10^{-5} Ci estimate of transuranics inhaled as a consequence of Year 2020 LMFBR operations.

*From Ref. 104, Vol. 1, p. 32.

**From Ref. 104, Vol. 1, p. 36.

Table II.G-19

ALPHA-EMITTERS IN HUMAN BONE^a

Radionuclide	Concentration (pCi/kg)	Total Skeleton (pCi) ^b
U-238	4.5	22
Ra-226	7.6	38
+ daughters ^c	7.6	38
Ra-228 (β-emitter)	(3.8)	
+ daughters ^d	20.4	102
Po-210	40	200
Total α-emitters	80	400

^aBasic data from UNSCEAR Report, Vol. 1, p. 76, Table 9.¹⁰⁴

^bAssumes 5 kg of mineral bone.

^cAssumes 33% retention in bone of Rn-222 and its two short-lived α-emitting daughters.

^dAssumes equilibrium between Ra-228 and its daughters (5.36 alpha disintegrations per disintegration of Ra-228, Ref. 105, p. 9, Table 1.3).

The UNSCEAR Report also provides estimates of radiation dose from natural sources. Some of these estimates are summarized in Table II.G-20, recalculated as man-rem for the U.S. population of 2×10^8 persons over a period of 70 years (30 years for gonads). When compared with estimated man-rem from Year 2020 LMFBR transuranic element releases, the natural radiation dose from alpha-emitters is larger by factors of about 10^5 ; and the dose from all natural radiation sources is larger by factors of 10^5 to 10^6 . In making this comparison it must be noted that the estimated dose from LMFBR transuranic releases is delivered to all subsequent generations. About half of this dose is received by the initial generation, with very much smaller fractions delivered to subsequent generations, for the lifetime of the radionuclides involved.

II.G.5.3. Plutonium Toxicity in Animals

There is a substantial body of experimental data on the toxicity of plutonium in experimental animals. The acute toxicity of injected plutonium is due primarily to destructive effects on the blood-forming system, resulting from irradiation of the bone marrow by plutonium deposited on bone surfaces, or released from these surfaces into the marrow.^{106,107} In the case of inhaled plutonium, acute death

results from pulmonary edema, hemorrhage, and inflammation destruction of the functional tissue of the lung.^{107,108}

Table II.G-20

DOSE EQUIVALENT TO U.S. POPULATION FROM NATURAL
RADIATION SOURCES COMPARED WITH ESTIMATED DOSE
EQUIVALENT FROM YEAR 2020 LMFBR TRANSURANIC RELEASES

Organ	Dose Equivalent to U.S. Population (2×10^8 persons) - Units of 10^6 Man-Rem		
	From All Natural Radiation Sources	From Naturally Occurring Alpha-Emitters	From Estimated Year 2020 LMFBR Releases of Transuranic Elements ^a
Bone ^b	2000	500	0.06
Lung ^c		800	0.009
Gonads ^d	600	50	0.0009

^aFrom Table 4.7-2.

^bBasic data on natural radiation from UNSCEAR Report, Vol. 1, p. 83, Table 20.¹⁰⁴
A quality factor of 10 was used for alpha-emitters in converting rad doses to rem dose equivalents. Population dose was calculated for a 70-year period. The UNSCEAR data are for dose to bone-lining cells, which, as calculated using the method of Spiers,¹⁰⁵ results in a smaller dose than the average tissue dose estimated for LMFBR releases.

^cBasic data on naturally occurring alpha-emitters from UNSCEAR Report, Vol. 1, p. 35, paragraph 118. A quality factor of 10 was used in converting rad dose to rem dose equivalent. Population dose was calculated for a 70-year period.

^dBasic data on natural radiation from UNSCEAR Report, Vol. 1, p. 83, Table 20.¹⁰⁴
A quality factor of 10 was used in converting rad doses to rem dose equivalents. Population dose was calculated for an average generation time of 30 years.

Acute toxicity is conventionally expressed in terms of an "LD₅₀" 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 μ Ci/kg. A similar value was observed for mice. A somewhat lower value, 20 μ Ci/kg was observed in dogs; however, this was with injected hexavalent plutonium. The LD_{50/30} for inhaled plutonium in rats and dogs was not very different from the values for injected plutonium. All of these dose values will vary somewhat depending on the compound administered and the valence state of the plutonium. It seems unlikely, however, that acute death would result from an internally deposited dose of less than 10 μ Ci/kg, which translates to 700 μ Ci for a 70 kg man. This amounts to about 10 mg of plutonium-239, or about 40 μ g of plutonium-238.¹⁰⁷

Long-term effects 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 LMFBR releases. The incidence of cancer appears to be the most sensitive measure of these long-term effects. The organs in which plutonium is retained 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.

An informative experiment on the toxic effects of 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 extensive data are available on the toxicity of radium in man.

Table II.G-21 shows the status of the plutonium-injected animals in this experiment.¹⁰⁹ In the initial injected groups, all animals are now dead. In all of these groups, there was a very substantial incidence of bone cancers. With decreasing dose, the time to tumor appearance increased until, in the lowest dose group, the average life span was not significantly different from that of the controls. Additional groups at lower dose levels were exposed beginning in 1964; nearly all of these animals are still alive.

Table II.G-21
PLUTONIUM-INDUCED BONE CANCERS IN UTAH DOG STUDY¹⁰⁹

Injected Dose ($\mu\text{Ci}/\text{kg}$) ^a	Tumor Incidence	Dogs with Bone Cancer		
		Years to Death	Dose to Skeleton	
			Rad	Rem
2.9	7/9 = 78%	4.1	4900	245000
0.9	12/12 = 100%	3.6	1300	65000
0.3	12/12 = 100%	4.5	600	30000
0.1	10/12 = 83%	7.2	310	15500
0.05	9/13 = 69%	8.5	190	9500
0.016	4/13 = 31%	9.9	78	3900
Controls	0	11.5 (for all control dogs)		

^aAdditional studies are in progress at dose levels of 0.016, 0.006, 0.002, and 0.0006 $\mu\text{Ci}/\text{kg}$.

Of more interest than absolute cancer incidence figures, is the finding in the Utah studies that plutonium-239 is 5 to 10 times more toxic than radium-226, on the basis of the same total energy delivered to bone.¹¹⁰ This difference is attributable to the more hazardous localization of plutonium on bone surfaces, near the cells from which bone tumors originate. This finding supports the use of a distribution factor (N) of 5 in the calculation of dose equivalent for plutonium.

A number of long-term studies in rodents have also pointed to bone cancers as the most sensitive indicator of the toxicity of injected plutonium. Cancer incidence data from some of these studies are summarized in Table II.G-22. These data were selected from a more extensive tabulation by Bair,¹¹¹ and include only experiments or experimental groups meeting the following criteria: (1) radiation dose to bone was evaluated in the study, (2) more than one exposure level was studied, (3) cancer incidence did not exceed 50%, and (4) life shortening was not excessive. These criteria should help to exclude results showing a misleadingly low cancer incidence due to saturation effects or early death from other causes.

It should be noted that calculation of the rem doses listed in Table II.G-22 involves the use of a quality factor of 10 and a distribution factor of 5; rem doses are therefore 50 times the rad doses. The bone cancer incidence per rem shows rather close agreement, both within and between experiments, suggesting that cancer incidence is an approximately linear function of radiation dose to bone over the ranges studied, that it is not greatly influenced by the compound or route of administration, and that mice, rats, and dogs show a similar sensitivity, although dogs appear to be more sensitive than rodents. These similarities of behavior lend confidence to the use of these numbers for the estimation of effects in man.

The toxicity of inhaled plutonium may be illustrated with data from an experiment conducted at Pacific Northwest Laboratory, involving the inhalation of $^{239}\text{PuO}_2$ by beagle dogs.¹¹⁸ 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 postexposure, 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 II.G-3. Each point in this figure represents a dog. Most of the dogs died early, within a year or two following plutonium exposure, with symptoms of respiratory distress occasioned by severe pulmonary fibrosis. There appears to be about a 3-year minimum latent period before a lung cancer can develop; a longer latent period is associated with smaller doses. A line fitted to all of the these points intersects the normal

Table II.G-22

PLUTONIUM-INDUCED BONE CANCERS IN EXPERIMENTAL ANIMALS¹¹¹

Ref.	Species	Administration		Bone Dose Equivalent (rem)	Bone Cancer Incidence	
		Compound	Route		Fractional	per rem
112	Mouse	Citrate	Intravenous	2000	.039	2.0×10^{-5}
				4200	.08	1.9×10^{-5}
				6500	.180	2.7×10^{-5}
				20000	.430	2.2×10^{-5}
113	Rat	Citrate	Inhaled	830	.013	1.6×10^{-5}
				1640	.056	3.4×10^{-5}
				3820	.025	0.7×10^{-5}
				6700	.032	0.5×10^{-5}
				12200	.11	0.9×10^{-5}
				17800	.19	1.0×10^{-5}
114	Rat	Citrate	Oral	1650	.03	1.8×10^{-5}
				2850	.074	2.6×10^{-5}
114	Rat	Citrate	Intra- and Subcutaneous	2600	.04	1.5×10^{-5}
				25650	.25	1.0×10^{-5}
113	Rat	Carbonate	Inhaled	180	.01	5.6×10^{-5}
				320	.011	3.4×10^{-5}
				1260	.031	2.5×10^{-5}
				6000	.12	2.0×10^{-5}
115, 116	Rat	Nitrate	Intratracheal	2565	.016	0.6×10^{-5}
				5850	.04	0.7×10^{-5}
				20900	.09	0.4×10^{-5}
				44000	.13	0.3×10^{-5}
117	Dog	Citrate	Intravenous	3900	.31	7.9×10^{-5}
				9500	.69	7.3×10^{-5}
				15500	.83	5.4×10^{-5}

life span of the beagle dog at a deposition of about 5 nCi/g lung. This extrapolation is very uncertain, however. If one draws the best line through the closed circles (the tumor-bearing animals) a steeper slope is obtained. Clearly, more data are needed from dogs exposed at lower levels. Experiments to obtain these data are in progress.¹¹⁸

Geesaman questions the use of these data because of the relatively few animals in the study and the high exposure levels resulting in essentially 100% lung cancer incidence.* It is agreed that the utility of these data is limited with respect to dose-response relationships. It is included only to illustrate the relationship between latency prior to the appearance of lung tumors and the quantity of plutonium inhaled, a relationship which suggests that at lower concentration levels the latent period may exceed the normal life span of these animals.

Data on the induction of lung tumors in experimental animals are shown in Table II.G-23, as selected from the tabulation of Bair,¹¹¹ using the criteria previously employed for bone tumors. The rem doses for lung are 10 times the rad doses, reflecting a quality factor of 10, but no distribution factor. The data on lung tumors are less extensive than those for bone tumors. In general, the incidence per rem is substantially higher in the case of lung tumors. Of particular note is the high incidence per rem obtained in the only experiment employing Pu-238.¹¹⁹ This may be associated with the more uniform distribution of Pu-238 in the lung (Appendix II.G.6).

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.¹¹⁰ Small, benign bile duct tumors were incidental findings at autopsy in eight other dogs, but such tumors were also seen in controls at a somewhat lower incidence. The liver tumors showed a typically long latent period, which suggests that at lower dose levels, and lower incidences of bone and lung tumors, liver tumors might become more important relative to bone and lung tumors.

Lymph nodes draining the lung, or sites of intramuscular plutonium deposition, may accumulate plutonium concentrations many times higher than concentrations seen elsewhere in the body. Various histopathologic changes have been observed in the

*Comment Letter 27 (D. P. Geesaman), pp. 15-16. (NA 03T)

RELATIONSHIP BETWEEN THE QUANTITY OF $^{239}\text{PuO}_2$
DEPOSITED AND SURVIVAL TIME OF DOGS
Figure II.G-3

tracheobronchial lymph nodes of dogs with plutonium in their lungs, but these changes are not obviously detrimental. Tumors have been seen only rarely, and their relationship to plutonium exposure is uncertain. Indirect effects on immune capability or on lymph drainage are considered most unlikely at the low levels of exposure projected.^{111,118} The ICRP, while recognizing the higher burdens and dose commitments in the lymph nodes relative to the lungs, skeletal system, and liver, has not considered the thoracic lymph nodes to be a critical organ.⁷²

Table II.G-23

PLUTONIUM-INDUCED LUNG TUMORS IN EXPERIMENTAL ANIMALS¹¹¹

Ref.	Species	Compound Inhaled	Lung Dose Equivalent (rem)	Lung Tumor Incidence	
				Fractional	per rem
113	Rat	Citrate	470	.05	10×10^{-5}
			1200	.025	2×10^{-5}
			2300	.084	4×10^{-5}
			4700	.35	8×10^{-5}
113	Rat	Carbonate	410	.042	10×10^{-5}
			800	.05	6×10^{-5}
			1900	.13	7×10^{-5}
			5000	.36	7×10^{-5}
			11000	.43	4×10^{-5}
119	Rat	Oxide (Pu-238)	320	.20	63×10^{-5}
			3800	.25	7×10^{-5}
118	Dog	Oxide	12000	.88 ^a	7×10^{-5a}

^aBecause of the high tumor incidence (violating the selection criteria), the estimated incidence per rem is of questionable validity.

Effects on the production or survival of the various types of blood cells have been studied in many of the experiments on plutonium toxicity. A variety of such effects are noted at high exposure levels.¹⁰⁷ The most sensitive of these effects is probably the reduction in blood lymphocytes following deposition of plutonium in the lung.¹¹⁸ One cannot rule out the possibility of a relationship between this effect on lymphocytes, lymph node pathology, decreased immunological capability, and the pathogenesis of plutonium-induced lung tumors.

Since plutonium on bone surfaces will irradiate portions of the bone marrow, concern has been expressed that leukemia might be an important delayed effect of plutonium exposure. Although leukemias have, in rare instances, been reported to result from plutonium exposure, they have occurred only following rather large doses, and at a much lower incidence than bone tumors.^{106,120}

Effects in testes or ovaries have been observed only following plutonium doses much higher than the doses which would have resulted in other evidences of toxicity.^{121,122} Studies of cytogenetic effects in the testes of hamsters showed no significant increase in the frequency of chromosome aberrations after calculated radiation doses of 1 and 4 rads.¹²³ The exposures employed would result in significant life shortening and cancer induction, suggesting that genetic risks are small compared to somatic risks.

Estimates of cancer risk per rem from Tables II.G-22 and II.G-23 have been applied in Table 4.7-4 of Section 4.7 to the prediction of cancer risk in man from LMFBR transuranic releases. Uncertainties involved in the extrapolation of these animal data to man are discussed in Appendix II.G.5.5.

II.G.5.4 Effects of Other Types of Radiation in Man

In the absence of data on the effects of transuranic elements in man, some inferences regarding these effects may be drawn from observations of the effects from other forms of ionizing radiation in man. Such inferences would be based on data derived from the medical, occupational, accidental, or wartime exposure of humans to different radiation sources: external x-radiation, atomic bomb gamma and neutron radiation, radium, radon and radon daughters, etc. Such information was recently summarized by the National Academy of Sciences - National Research Council Committee on the Biological Effects of Ionizing Radiations⁹¹ and by the United Nations Scientific Committee on the Effects of Atomic Radiation.¹⁰⁴ Both groups arrived at comparable risk estimates for the expected mortality from radiation-induced cancer. The NAS-NRC document, commonly referred to as the BEIR report, presents these estimates in a form that is conveniently applicable to the estimation of risks in populations that include the normal age distribution, and is used as the source document for cancer risk estimates. The BEIR report is also employed to derive genetic risks that might be attributed to a retained burden of transuranic elements in the gonads. Risk estimates derived in the BEIR report are all linearly extrapolated from human experience at relatively high dose rates and total doses; the validity of such extrapolations will be discussed in Appendix II.G.5.5.

The extensive human data considered in the BEIR report will not be reviewed here; it should be noted, however, that effects of irradiation from external sources, in particular the Japanese atomic bomb survivors and irradiated spondylitics, were heavily weighted in arriving at risk estimates. The BEIR report makes estimates of both absolute risk and relative risk and for each of these assumes either a

30-year or a duration-of-life interval following the latent period, during which risk remains elevated. This leads to four risk estimates. Only the two estimates which lead to the lowest and the highest predictions of cancer mortality are considered here; the lowest being the absolute risk model with a 30-year plateau, referred to as the "absolute model," and the highest being the relative risk model with a lifetime plateau, referred to as the "relative model." Each of these models makes separate risk estimates for the in utero, 0-9 years, and 10+ years age periods, reflecting age differences in the sensitivity to irradiation. The derivation of these risk estimates and their application to the U.S. population is summarized in tables on pp. 169 and 171 of the BEIR report,⁹¹ where the excess deaths due to cancers other than leukemia for the U.S. population, per 0.1 rem, per year, are predicted as 1,210 by the "absolute model" and 8,340 by the "relative model."

The fraction of these non-leukemia cancers that are bone cancers is given as 0.04, and the fraction that are lung cancers is given as 0.26. No fraction is given for liver cancers, which fall in the BEIR report's "GI including stomach" category, which is 0.20 of the non-leukemia cancers. It is assumed that these cancers will be induced in direct proportion to their incidence in the Hiroshima-Nagasaki survivors, where primary liver cancer accounted for about 0.08 of the GI cancers. Therefore 0.016 (the product of 0.08 and 0.20) is considered the fraction of the total non-leukemia risk that is attributed to liver cancers. The Draft Statement incorrectly calculated the incidence of liver cancer from the BEIR numbers provided in the category of "other cancers." The BEIR report included liver cancers in the category of "GI including stomach" cancers, and the revised number given here reflects this correction.

Taking the fraction of the predicted non-leukemia cancers attributable to each cancer type and converting to a man-rem basis, one arrives at the risk estimates shown in Table II.G-24. These estimates are applied in Table 4.7-5 of Section 4.7 to the prediction of cancer mortality in man from LMFBR transuranic releases.

Leukemia risks were not estimated because dose to bone marrow from bone-deposited transuranic elements is small compared to the dose to other organs, and because leukemia has been only infrequently seen in animal experiments with plutonium.^{106,120}

The cancer risk from irradiation of thoracic lymph nodes was also neglected, even though these nodes, in experimental animals, receive a radiation dose from plutonium that is higher than the dose to any other organ.¹¹¹ The incidence of lymphosarcoma,

Table II.G-24

DERIVATION OF CANCER MORTALITY RISK ESTIMATES
FROM BEIR REPORT ESTIMATES

1	2	3	4	5
Cancer Type	Risk Model ^a	Fraction of Non-leukemia Cancers ^b	Predicted Excess Deaths per yr per 0.1 rem per yr Continuous Exposure of U.S. Population ^c	Predicted Excess Deaths per man-rem ^d
Non-leukemia	"Absolute"	1.0	1,210	
	"Relative"	1.0	8,340	
Lung	"Absolute"	0.26	315	16×10^{-6}
	"Relative"	0.26	2,168	110×10^{-6}
Bone	"Absolute"	0.04	48	2×10^{-6}
	"Relative"	0.04	334	17×10^{-6}
Liver	"Absolute"	0.016	19	1×10^{-6}
	"Relative"	0.016	133	7×10^{-6}

^a"Absolute" refers to BEIR report absolute risk model with 30-year plateau following latent period during which risk remains elevated. "Relative" refers to BEIR report relative risk model with lifetime plateau.

^bFractions given in BEIR report (p. 171) for age 10 or more assumed to apply to all categories. Fraction for liver cancers is not given in BEIR report, but estimated as explained in text.

^cNon-leukemia deaths taken from BEIR report (p. 169). Others calculated by application of fractions listed in Column 3.

^dCalculated from numbers in Column 4 by multiplying by 10 (converting to rem basis) and dividing by 2×10^8 (U.S. population total employed in derivation of Column 4 numbers).

reticulosarcoma, Hodgkin's disease, and/or multiple myeloma has been observed to be increased in Japanese atomic bomb survivors,^{124,125,126} in patients receiving radiotherapy for ankylosing spondylitis,¹²⁷ and in radiologists who entered practice in the days preceding current restrictions on occupational exposure.¹²⁸ For this reason, the lymphoid and reticular tissues were classified by an ICRP Task Group¹²⁹ as being "apparently" high in susceptibility to the carcinogenic effects of radiation; however, it was noted by the Task Group that at least some of the neoplasms included in this group (e.g., multiple myeloma) appear to arise in the bone marrow. Whether in fact any of these neoplasms is attributable to transformation of cells in lymphoid tissues, as opposed to transformation of precursor cells in the bone marrow, is debatable in light of current concepts of cell population kinetics in the lymphatic and reticular system. At present, therefore, in the absence of evidence that localized irradiation of lymphatic tissue is carcinogenic in human or animal populations, it is not possible to make numerical estimates of such cancer risks.

The genetic risks considered in the BEIR report include the full spectrum of genetic defects seen in the U.S. and other Western nations. Their effects upon the carrier may range from a lethal action occurring at any time of life (from before birth until death), 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 non-genetic origin. As a consequence, it is not appropriate to compare or equate estimates of genetic risk directly with cancer risks, where case incidence and case mortality are substantially one-to-one.

The BEIR report (pp. 54-57) summarizes its risk estimates for genetic defects in terms of a 5 rem per generation dose to a population of one million. Converting these numbers to a man-rem basis, one obtains the risk estimates of Table II.G-25. The range of these estimates reflects a 10-fold uncertainty in the value of the mutation rate doubling dose, which is assumed to lie in the range of 20-200 rem. There is a further uncertainty with regard to the magnitude of the genetic component of the defects with complex etiology.

An additional category of genetic risk discussed in the BEIR report is that concerned with general "ill health" of uncertain genetic determination. This risk was conservatively estimated as a 0.5 to 5.0 percent increase in the equilibrium

incidence of ill health per 5 rem per generation. Thus, for the total U.S. population, 10^9 man-rem per generation would increase ill health by 0.5 to 5 percent. Continuous LMFBR operation at the Year 2020 level, resulting in 500 man-rem per year (Table 4.7-2) or 15,000 man-rem/30 year generation, would, by this estimate, increase the equilibrium incidence of ill health by $(.75 \text{ to } 7.5) \times 10^{-5}$ percent.

Table II.G-25

DERIVATION OF GENETIC DEFECT RISK ESTIMATES
FROM BEIR REPORT ESTIMATES

1	2	3
Type of Risk	Predicted Defects per Million Persons per 5 Rem per Generation at Equilibrium ^a	Predicted Defects per Man-rem ^b
Specific Genetic Defects ^c	250-2500	$50-500 \times 10^{-6}$
Defects with Complex Etiology ^d	50-5000	$10-1000 \times 10^{-6}$

^aValues taken from Table 4, p. 57 of BEIR report.⁹¹

^bEqual to values of Column 2 divided by 5×10^6 .

^cIncludes dominant diseases, chromosomal and recessive diseases.

^dIncludes congenital anomalies, anomalies expressed later, constitutional and degenerative diseases.

Many uncertainties are involved in these genetic risk estimates, as reflected in the following statement from page 59 of the BEIR report:⁹¹

It is clear that these estimates are subject to great uncertainty. The ranges of plausible values are broad, and there is no assurance that the true values are within these ranges. We are well aware that future information will necessitate revisions. The estimates are presented, not as accurate scientific information (as scientists we would prefer to defer judgment until the information is solid), but as reasonable values based on current knowledge which, crude and uncertain as they are, may serve as a better guide to rational uses of radiation than no estimates at all.

Some of the uncertainty in genetic risk estimates may have been resolved by recent data from Canadian vital statistics records. As reported by Newcombe:¹³⁰

The numerous and important irregularly inherited diseases, like diabetes and cleft palate, are probably not mutation-maintained. Only the simple dominant diseases would be expected to increase in direct proportion with the mutation rate, but these appear from new data to be rarer than has been supposed, affecting only about 0.1 individual per 100 born alive instead of 1 per hundred as previously believed. A revised estimate of risk suggests that a population exposure of a million man-rem probably results in no more than 20 additional cases of hereditary defects.

This estimate of genetic risk is three times lower than the minimum BEIR estimate, and 75 times lower than their maximum estimate.

II.G.5.5 Problems of Extrapolation

Estimation of the human risks of very low-level environmental dispersal of transuranic elements involves several kinds of extrapolation. Whether based on observations in experimental animals or man, there is the problem of extrapolating to much lower exposure levels than are covered by the data. The human data are for types of radiations that do not include internally deposited transuranic elements; one must extrapolate, based largely on experience with external irradiation. Experimental animal data are available for transuranic element toxicity, but one must extrapolate from animal to man.

The EPA has questioned the use of animal data in extrapolations to man in view of the differences in radiosensitive cell populations, differences in detailed aspects of metabolic behavior, and the doubt that animal data can be used to predict effects in humans.*

The extrapolation of toxicity data from animal to man is without doubt the most familiar of these processes. That such extrapolation can be justified is an underlying assumption of most toxicological research. Where data from several animal species are in reasonable agreement, it is assumed, in the absence of conflicting evidence, that man will behave similarly. This assumption is more confidently made in the case of acute effects. For long-term effects, species differences have more opportunity to manifest themselves, and differences in life span may be of significance.

Fortunately, for the case of bone-deposited α -emitters, one can make direct comparisons between human and experimental animal toxicity data. Table II.G-26

*Comment Letter 53 (EPA), pp. 8-9.

compares data on radium toxicity in man and in experimental animals.¹³¹ For both dogs and mice the incidence of bone cancers per rad is about ten-fold higher than in man. For radium, the extrapolation error would thus result in a conservative overestimate of effect in man. While this same ratio could not be assumed to apply exactly to the case of transuranium elements, the agreement in radium data for animal and man is encouraging.

Table II.G-26

COMPARISON OF RADIUM TOXICITY DATA
IN EXPERIMENTAL ANIMALS AND MAN¹³¹

	Bone Cancer Induction (%/rad)
Human Data	
Ra-226 + Ra-228	0.0046
Ra-224 in adults	0.009
Ra-224 in juveniles	0.015
Experimental Animal Data	
Ra-226 in dogs	0.045
Ra-228 in dogs	0.13
Ra-226 in female mice	0.043
Ra-224 in female mice	0.12
Ra-224 in male mice	0.0073

Aside from the human radium data, experience with α -emitters in man is of little help in these evaluations. Data on α -radiation-induced lung tumors in uranium miners and liver tumors in thorotrast patients suffer from inherent dosimetric complexities which seriously limit their usefulness. Nevertheless the uranium miner data were considered in the BEIR report, and used along with data from our experience with external radiation exposures to derive risk estimates for radiation induced lung cancer.

The EIAP has expressed concern over the consequences of chronic exposure. It cites in this regard:

the increased incidence of bone cancer among uranium mine workers, and the high incidence of bronchial cancer among uranium mine workers... receiving a cumulative alpha radiation dose from the inhalation of radon daughters of as little as 60 rads in small volumes (a few grams) of tissue..., and the high incidence of liver and other soft tissue cancer...(and) leukemia among the 'Thorotrast' patients.*

The Final Statement does include reference to and data from the radium dial workers with respect to bone cancer induction. Recent studies have shown no significant excess of lung cancer in the dose range of 120 Working Level Months.^{91,132} This has been variously related to doses ranging from 60 rad⁹¹ to 1800 rad¹³³ to the basal cell layer of the respiratory epithelium, with 240-840 rad often being the range suggested.^{132,133,134} Although the Thorotrast injected patients do show an increased liver cancer and leukemia incidence, because of the lack of any generally accepted evaluation of the dosimetry, and the lack of unequivocal relationships between dosimetry and biological effects resulting in an acceptable dose-response curve, no risk estimate per unit radiation dose based upon this population is generally accepted by responsible bodies, e.g., the BEIR Committee, for risk assessment.

Atomics International suggests that the uranium miner experience and the individuals injected with Thorotrast provide evidence which shows "that the 'spatially averaged dose' is determining, and that average doses within the current standard (15 rad/yr) will not produce lung cancer."** While it is true that averaged lung doses at this level have not led to lung cancers in these populations, the complexities of the dosimetry, the marked differences in regional deposition in the respiratory tract, the fact that the alpha-emitters inhaled from mine air are single atoms attached to dust particles rather than particulate aerosols, and the incomplete studies of the Thorotrast patients preclude the use of these data as valid evidence supporting the conclusion suggested.

All observations on radiation effects, whether in animals or in man, have been made at radiation doses to tissues, and at radiation dose rates, that are very much higher than the average tissue doses and dose rates that might conceivably result from releases of transuranic elements from LMFBRs. The very low incidence of effects that might still be of concern when applied to very large populations are simply unmeasurable in either animal experiments or human epidemiological studies. It is necessary, therefore, to interpolate between the doses for which

*Comment Letter 51 (EIAP), pp. 20-21, 32.

**Comment Letter 54 (Atomics International), enclosure p. 5.

data on effects are available, and zero dose where zero effect can be assumed. For the estimates of health effects made in this Statement we have based this interpolation on an assumed linear dose-effect relationship. The BEIR report (page 97) justifies such a procedure in the following words:⁹¹

In view of the gaps in our understanding of radiation carcinogenesis in man, and in view of its more conservative implications, the linear, non-threshold hypothesis warrants use in determining public policy on radiation protection; however, explicit explanation and qualification of the assumptions and procedures involved in such risk estimates are called for to prevent their acceptance as scientific dogma. Furthermore, the linear hypothesis is the only one which permits the selection of the mean accumulated tissue dose to characterize the radiation exposure of a group under conditions of nonuniform exposure and exposure rate. The mean accumulated tissue dose is the only practical quantity that can be used to estimate the risk of cancer in such populations until the influence of the many interacting variables can be better specified.

While agreeing with the practical utility of this procedure, it is most important that the results not be accepted as "scientific dogma." It is not the objective of this Environmental Statement to "determine public policy on radiation protection." Its objective is to present all of the evidence, so that public policy, as eventually determined, may reflect all relevant risks and benefits. In this light, it must be emphasized that the interpolated risk estimates do rest on conservative assumptions, and that there is no experimental basis for choosing between these estimates and an estimate of zero effect.

From theory one may argue convincingly that the high LET of alpha radiation, and consequent lack of repair of cellular effects, demands a linear relationship at low dose between cells killed or damaged and alpha particles traversing the tissue. However, there is no accepted theory which describes the process by which dead or damaged cells lead to the production of cancers. This need not be a linear function of the number of cells affected.

II.G.6 Problems of Averaging Dose

Throughout this Environmental Statement, where necessary to consider a radiation dose to an organ, this dose has been calculated as an average value for the total organ. Such doses are, in fact, not uniformly distributed throughout the organ and this point has been raised as a criticism in several Comment Letters.

The EPA questions the validity of this approach* and points out that their use of this procedure has been qualified by the statement, "In the case of alpha emitters,

*Comment Letter 53 (EPA), pp. 6, 9, 10, 11.

such averaging is obviously inappropriate if only a few particles are present."* The NRDC based upon their analysis of the "hot particle" question,¹⁰³ concludes that averaging the dose is inappropriate and that the risk resulting from inhaled plutonium particles conforming to their definition of a "hot particle" is significantly underestimated in this Environmental Statement.**

Geesaman's discussion of nonuniform irradiation of tissues similarly suggested that the health effects derived in the Draft Statement were underestimated.[†] Relating alpha emitters contained in particulates entering and retained in the lung as a constituent of cigarette smoke to lung cancer in smokers, Martell and the EIAP suggest that the potential lung cancer incidence presented in the Draft Statement is underestimated.^{††} Other Comment Letters refer to these criticisms.[§] These issues are addressed in Section 4.7, in this Appendix, and, more specifically, in the AEC responses to the several Comment Letters.

This problem has been recently reviewed by Bair, Richmond, and Wachholz;¹⁰² their document may be consulted for a more detailed presentation of data bearing on the question.

Insoluble radioactive particles, immobilized in tissue, provide focal spots of high radiation dose. The range of alpha particles from the transuranic elements of concern is about 40 μ m in soft tissue of unit density, or about 200 μ m in lung tissue of assumed density 0.2. The actual range of individual alpha particles in lung tissue will vary depending upon the ratio of cells to air spaces along the path of the alpha particle. While the dose will be high within alpha particle range of a radioactive particle, the dose will be zero to tissues beyond alpha particle range. Thus, for a given quantity of particulate radionuclide, the larger the particles the smaller the fraction of tissue irradiated. This is illustrated in Table II.G-27, which shows the relationship between particle size, number and activity, and cells, or fraction of lung volume at risk.¹⁰²

*Comment Letter 53 (EPA), p. 10.

**Comment Letter 6 (NRDC), pp. 1-40; 38 (NRDC), pp. 2-11.

†Comment Letter 27 (D. P. Geesaman), pp. 6-18.

††Comment Letter 51 (EIAP), pp. 20-22, 29-34.

Hearing Testimony (E. A. Martell), pp. 239-247.

§Comment Letters 7 (R. Daifuku), enclosure pp. 15-16; 8 (G. Younghein and I. Younghein), enclosure 1, p. 4; 16 (V. T. Edsall), p. T; 23 (N. Dawkins), p. 1; 59 (C. Brawman), p. 1.

Table II.G-27

RELATIONSHIP OF PARTICLE SIZE TO NUMBER OF CELLS AT
RISK FOR A STATIC LUNG BURDEN OF 0.016 $\mu\text{Ci } ^{239}\text{PuO}_2^{\text{a},102}$

Particle Diameter (μm)	Number of Particles	Activity per Particle (pCi)	Number of Cells at Risk	Fraction of Lung at Risk
0.1	5.4×10^7	.0003	3×10^{11}	0.3
0.3	2.0×10^6	.01	1.3×10^{10}	0.01
0.7	1.8×10^5	.08	1.2×10^9	0.001
1.0	5.4×10^4	.3	3.6×10^8	0.0003

^aAssuming immobile particles in a structure-less human lung of uniform density 0.2 g/cm^3 with cells of average volume $10^3 \mu\text{m}^3$. Cells at risk are those within a sphere of radius equal to the alpha range ($200 \mu\text{m}$ at the assumed density).

The basic question, then, is whether the larger dose to the cells in close proximity to a radioactive particle is more, or less hazardous than the smaller dose to the larger number of cells exposed to a more uniformly distributed radiation source. This question is one that has received much attention from the ICRP and NCRP, as witnessed by the many references to the subject in their publications (quoted at length in ref. 102). The fact that these bodies have continued to base their standards on average dose to organs would suggest that they do not view this as a "non-conservative" assumption.

In the following discussion the arguments and evidence on this question will be briefly summarized under three general headings: (1) theoretical considerations, (2) data from animal experiments, and (3) evidence from human exposures.

Theoretical Considerations

It must be recognized at the outset that there is no universally accepted theory for the induction of cancer by radiation, except that such induction must be related to the physical absorption of radiation energy by the biological organism. The extremely dense ionization along the track of an α -particle is very disruptive and is usually lethal to a cell whose nucleus is traversed. But since a dead cell cannot itself become malignant, it is evident either that some irradiated cells must survive and express their injury by uncontrolled growth, or that dead or injured cells must be able to convey this characteristic of uncontrolled growth to neighbor cells. Cancer induction may be thought of as a complex competition between cell transformation and cell death.

If cancer induction required multiple events (within the same cell, or among nearby cells) one might postulate that a given amount of α -activity would be most effective when concentrated in discrete particles, rather than uniformly distributed throughout the tissue, since focal sources increase the frequency of multiple "hits" within discrete "targets." But this is not the only factor that must be considered. Injured cells within α -range of a particulate source are often killed by subsequent α -particle irradiation. Furthermore, the undamaged cells outside of the sphere of irradiated tissue may be able to assist the cells within to repair their damage and to prevent the expression of malignant growth. Thus, for particulate irradiation some factors may tend to increase the effectiveness of cancer induction, while others may decrease it.

It is not possible with present knowledge to predict confidently from theory whether particulate irradiation should be more or less effective than distributed irradiation in cancer induction. Fortunately, several animal experiments are directly relevant to the effect of non-uniform distribution of dose.

Animal Experiments

Before considering the results of specific experiments, factors should be noted that make all radioactive particle exposures more uniform than might be expected, and all uniform exposures more non-uniform than might be expected. A radioactive particle that moves around provides a more uniform exposure than an immobile particle. Although the mobility of particles in the lung has not been quantitated, such movement must occur as a result of biological efforts to clear the particle from the lung. Particles are engulfed by macrophages, which move around, and which concentrate many small particles to form, in effect, a larger one.¹³⁵ There are also solubilizing effects which tend to disperse particulate material; these seem especially important in the case of the radionuclides of higher specific activity, e.g., Pu-238, Cm-242, and Cm-244.¹³⁶ Thus, material present initially as particles of a particular size cannot be assumed to remain as such.

On the other hand, there is no way to provide a truly uniform exposure to transuranic elements. Except for certain organic chelates, all compounds of the transuranic elements will hydrolyze at physiological pH to form colloidal hydroxides. Tissue deposits in experimental animals, which may be initially quite uniform, as indicated autoradiographically, become more particulate with the passage of time, by cellular aggregation, or by chemical preference for particular deposition sites.

It is thus fair to say that all animal experiments and all human exposures have involved material that was particulate, to an uncertain degree, and that the estimation of dose on any basis other than an average tissue dose would be uncertain, at best.

Several animal experiments are directly relevant to the effect of distribution of dose in the lung. Experiments still in progress at the Los Alamos Scientific Laboratory have involved a variety of activities and numbers of plutonium-containing particles, deposited in the lungs of hamsters.^{137,138,139} More than 1000 of these animals have lived out their life spans or have been sacrificed to date. They contained, in the aggregate, about 5×10^6 particles, each with an activity in excess of 0.07 pCi (the "hot particle" of Tamplin and Cochran¹⁰³). Only three primary lung tumors have thus far been observed, an incidence very much lower than that predicted by the NRDC hypothesis.

A recent study measured the incidence of chromosome aberrations in the liver of hamsters as a function of the size of Pu-239 oxide particles administered.¹⁴⁰ For the same total quantity of plutonium administered, the more uniform dose was most effective in producing chromosome aberrations. The proportion of cells showing multiple aberrations was greater following particulate exposure, indicating multiple hits.

Studies with polonium-210, an α -emitter, have also compared the effectiveness of particulate vs. more uniform exposure.¹⁴¹ In this case the particulate exposure was achieved by adsorption of the polonium onto hematite (ferric oxide) particles, and compared to polonium administered without hematite. As confirmed autoradiographically, the treatment resulting in the most non-uniform distribution of polonium gave the lowest tumor incidence and the longest induction time.

Indications are that inhalation of a soluble plutonium compound, ^{238}Pu derived from crushed microspheres, by rats results in a substantially higher tumor incidence than observed in comparable experiments with Pu-239 oxide.¹¹⁹ The most reasonable explanation for this difference lies in the fact that aerosolized material contained small particles of ^{238}Pu and a high percentage of ultrafilterable plutonium. The plutonium in the lung was highly dispersed resulting in a relatively diffuse radiation exposure of the lung. Because of the short retention time of the plutonium in the lung 80% of the radiation dose was delivered within the first four months after the inhalation exposure. After a year less than 1% of the initial alveolar burden of plutonium remained in the lung and was concentrated with hemosiderin-like pigment

granules in peribronchiolar and perivascular regions of the lung. However, it appears that the majority of the lung cancers observed in the animals in this experiment originated in subpleural areas rather than the peribronchiolar or perivascular tissues, and were bronchiolar-alveolar adenocarcinomas. In another experiment in which rats inhaled $^{238}\text{PuO}_2$ or $^{239}\text{PuO}_2$ particles, lung burdens equivalent to those retained for long times in the experiment with the high percentage of ultrafilterable ^{238}Pu did not cause lung cancer in a total of 77 rats observed for about 600 days. Lung cancer was observed in rats which inhaled quantities of $^{238}\text{PuO}_2$ or $^{239}\text{PuO}_2$ up to twenty times greater.¹⁴² In beagle dogs exposed to inhaled particulate $^{238}\text{PuO}_2$, bone cancer has been the predominant cause of death; lung cancer occurred in only one dog which also had bone cancer.¹⁴³ Thus, experimental evidence to date indicates that diffuse distribution of ^{238}Pu in the lung was more effective in causing lung cancer than particulate sources.

All of the above-noted experiments in which there was a comparison of particulate and less particulate α -emitters, indicate that the more particulate material was less hazardous on an equivalent total activity basis. Many other toxicity experiments with plutonium, in which specific comparisons between particle sizes were not made, are nevertheless informative. Thus, in all of these experiments the plutonium was in a more-or-less particulate state, but in none of them was there an unusually high tumor incidence, i.e., higher than might have been expected from a uniformly distributed dose (Table II.G-23).¹¹¹ In fact, the highest reported incidence per rem comes from the experiment with inhaled Pu-238, which was initially present in a very finely divided state (count median diameter, 0.02 μm), and which appears to disaggregate, in vivo.¹¹⁹

Often cited in support of an enhanced effectiveness for particulate exposure are experiments of Albert, et al.,^{144,145,146} who irradiated the skin of rats and found that one tumor was produced for every 2000 to 9000 radiation-damaged hair follicles. These experiments are not relevant to the question of distribution of dose from α -emitters in the lung for the following reasons:

- (1) The radiation employed in these experiment was not from a particulate source. The hair follicle identified as the susceptible target could not, in fact, be irradiated throughout its length by α -radiation from a transuranic element particle, and irradiation of the total follicle and surrounding tissue was necessary for tumorigenesis.¹⁴⁷
- (2) A susceptible target such as the hair follicle has not been shown to exist in the lung. In fact, the results could not be repeated on the skin of the mouse¹⁴⁸ or another strain of rats.¹⁴⁹

- (3) Even if a susceptible target existed in the lung, of small enough size to be irradiated by a single radioactive particle, there must still be a mechanism for locating the radioactive particle adjacent to the susceptible target.

The chain of hypothetical improbabilities at this point becomes somewhat academic.

Evidence from Human Exposures

Data on the deposition of plutonium in man, from fallout and from occupational exposure, was presented in Appendix II.G.5.1. Of special relevance to the "hot particle" problem are the estimated 3000 workers with depositions totalling an estimated 5 μCi . These exposures were mostly due to inhalation of particulate plutonium. For the vast majority of these cases, one has no information on particle size; but for one incident, a fire at the Rocky Flats facility in 1965, a mass median diameter of 0.32 μm was measured.¹⁵⁰ This is similar to values measured by others during various plant and laboratory operations.¹⁵¹ If one assumes a log-normal particle size distribution with a mass median diameter of 0.32 μm , a standard geometric deviation of 1.83 and a density of 10 g/cm^3 , one can calculate that about 15% of the mass will be associated with particles larger than 0.6 μm in diameter,¹⁰² i.e., "hot particles" as defined by Tamplin and Cochran.¹⁰³

The group of 25 exposed workers being studied at Los Alamos,¹⁰¹ who now contain an estimated 2.5 μCi of plutonium, must have inhaled an initial quantity of at least 10 μCi when exposed nearly 30 years ago. If this plutonium had the size distribution described above, about 10^7 particles larger than 0.6 μm actual diameter (0.07 μCi or more) must have been retained initially in the lungs of these 25 workers.¹⁰² According to Tamplin and Cochran's tumor risk estimate of 5×10^{-4} per "hot particle," this exposure should have resulted in 5000 lung tumors. The human experience is clearly at variance with any such prediction.

Conclusions

A considerable number and variety of toxicity studies with α -emitters in experimental animals show no evidence for an enhanced effect of "hot particles." Theoretical considerations, while not conclusive, provide at least as many arguments for a diminished effect of "hot particles" as for an enhanced effect. Nearly 30 years experience with people exposed to plutonium "hot particles" in numbers far exceeding anything predicted to result from LMFBR operations, has resulted in no evidence of disabling effects. National and international bodies charged with the responsibility

for establishing conservative radiation protection standards, despite frequently expressed awareness of the problem of "hot particles," continue their practice of estimating dose to the lung on an organ-average basis. Such a procedure therefore seems appropriate for this report.

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II.H.1 THERMAL DISCHARGES

II.H.1.1 Directly Lethal Effects

The lethal effects of excess heat and sudden changes in water temperature are gross effects which can often be readily observed. Such effects, should they occur, will normally be confined to a relatively small volume of the receiving water, that is, immediately adjacent to the point of discharge.

The temperature extremes that will be directly lethal to a fish will depend, in general, on the species of the fish, its acclimation temperature, and the stage of its life cycle. Values for thermal death points, upper temperature tolerance limits, LT_{50} , and similar measures of temperature limits can be found in the literature for representative species of marine, estuarine and freshwater species.²⁻⁶ Most of these data have been derived from laboratory studies and it has been suggested⁵ that these limits should be lowered by 1 or 2°C (~ 3°F) when considering field applications. Roughly, upper temperature tolerance limits for fishes are:

marine and estuarine species	30°C (86°F)
cold-freshwater species	25°C (77°F)
warm-freshwater species	33°C (91°F)

Fish, being mobile organisms, ordinarily avoid lethal temperatures. Adults can apparently sense a thermal gradient and, if free to move, will seek preferred temperatures. Thermal death at heated discharges has been known to occur when the fish were trapped in the effluent canals^{7,8} or subjected to a sudden release of hot water.⁹ Juveniles of some species, however, may not avoid lethal temperatures. For example, a non-avoidance response was exhibited by young white perch (Morone americana) and striped bass (Roccus saxatilis).¹⁰ Mortality of young fish in the outfall area of the heated discharge may therefore occur if water currents do not carry them away. Breakdown of the avoidance response in adults can also occur. For example, summer water temperatures above 30.5°C (87°F) were actively avoided by 11 species of estuarine fishes. Short exposures to water at 34.4°C (94°F) resulted in breakdown of the avoidance response.¹¹ This phenomenon will ultimately result in death of the fish.

A thermal discharge into a spawning area will be directly lethal to most eggs and developing embryos if they are immersed in water at temperatures a few degrees greater than normal ambient fluctuations, but it will be lethal to adults entering the area to spawn despite elevated temperatures.¹² It has been suggested⁵ that the most serious effect of a heated effluent on marine fish life is the effect on the eggs. Hatching ranges of many marine species occur between 15 and 30°C (59 and 86°F); elevated temperatures can prevent hatching of some species.

A cause of mortality to both adults and juvenile fish, both resident and acclimated to the heated plume, is a rapid decrease in water temperature due to a sudden plant shutdown. In winter, decreases in temperature at the outfall of 8 to 16°C (15 to 30°F) in a matter of minutes will not be uncommon for a once-through cooling system on any body of water. Fish subjected to this cold shock would be severely stressed and death will very likely follow. Large fish kills in winter at the Oyster Creek power station, for example, have been reported following shutdown.¹³ The travel time of the circulating water in a cooling tower is on the order of a half-hour to an hour. This is insufficient time for gradual cooling of the blow-down, but because of the usually small volume of effluent involved, mixing zones are relatively small compared to a once-through system, and the adverse effects of cold shock are correspondingly reduced. Travel time of the circulating water in a cooling lake can be on the order of days before discharge to the receiving water, and in such a case cold shock should be virtually nonexistent.

Most benthic organisms, being relatively immobile, can not avoid lethal temperatures. Two exceptions appear to be the adult grass shrimp (Palaemonetes), an important member of the estuarine food chain, and the blue crab (Callinectes sapidus), an important food and sport species. In studies at the Oyster Creek generating station, avoidance behavior was exhibited by the grass shrimp at 89.7°F (~ 32°C) and by the blue crab at 99.5°F (~ 38°C). Breakdown of the avoidance response occurred at 97.5 and 104°F respectively.¹³

Certain species appear to tolerate relatively high temperatures. For example, in the South Branch of the Forked River, a tributary of Barnegat Bay in New Jersey, strong thermal gradients are natural phenomena associated with the layering of freshwater over saltwater. The bottom fauna included a mixture of freshwater and estuarine species. Such fauna must thus be able to withstand pronounced and rapid temperature changes associated with the tidal oscillation of the river.¹⁴ Geographic location is also of importance. In an Alabama river, where bottom temperatures can reach 90°F (~ 32°C), there was a diverse fauna which included

mosquito larvae, midge larvae, mayfly, dragonfly, and damselfly nymphs, fingernail clams, snails, water striders, water bugs, and water scorpions.¹⁴ Above 90°F in the Delaware River, however, an extensive loss in numbers, diversity, and biomass of benthos occurred in a heated area below a power plant discharge.¹⁵

The benthos-poor conditions sometimes associated with heated waters is in large part due to the high biochemical oxygen demand (BOD) and low dissolved oxygen characteristics of the water into which thermal discharges are made. For example, the elimination of fingernail clams and other shellfish from the Illinois River, a warm-water stream, has been attributed to the high BOD waste discharges from the Chicago metropolitan area.¹⁶

Marine benthos, particularly those associated with shore and intertidal areas, are subjected to large variations in their natural environmental temperatures. For example, on rocky surfaces the initial temperature of the wet substrate surface is that of the sea and lower than that of the air in summer. After tidal exposure, the surface temperature may rise along with air temperature as the day progresses at a rate that is highly dependent upon habitat variables such as rock type and algal cover. On exposed rocks where many species live, insolation in summer at low tide produces surface temperature conditions significantly above air temperature.¹⁸ For example, a July rock surface temperature adjacent to barnacles or limpets was 33°C (91.4°F) while the air three feet above the surface measured 24°C (75.2°F) and a few miles away 20°C (68°F).¹⁸ Internal temperatures of certain marine benthos can be as high as 35°C (95°F) in the mussel (Mytilus californianus) without signs of lethal effects in the population. These relatively high temperature conditions apparently can be tolerated only for short periods, and lethal effects can occur even under natural conditions. For example, the sea urchin (Strongylocentrotus purpuratus) which usually inhabits hemispherical burrows in rocks and is thus kept cool on the bottom side, shows a population mean internal temperature of 15 to 23°C (59 to 73°F) at low tide during the day in early summer in Oregon. On several successive days in May 1968, maximums of 27 to 30°C (81 to 86°F) were measured in some urchins. Successive days of this type of heat at low tide for periods of three to five hours led to a heat kill.¹⁸ Mass heat kills of reef echinoids in Puerto Rico have been reported, and such kills may occur even in cold-temperature seas.¹⁸ It can be expected, therefore, that effluent temperatures greater than about 25°C (77°F) may be lethal to marine benthos immersed continuously and to larvae of certain benthic species such as the oyster which may be free-swimming for a period of two weeks before anchoring to the substrates.

With the exception of a sinking plume which may occur under winter conditions, heated water usually floats. This will tend to decrease the number of benthos subjected to lethal temperatures. Compounding the temperature effect, and often of more significance, is the scouring action of the discharge, particularly a high-velocity jet close to the bottom. For example, the abundance of benthic invertebrates around four power plants on Lake Michigan was found to be slightly decreased close to the discharge structures. It was concluded that this was due to current scour rather than a purely thermal effect.¹⁷

II.H.1.1.1 Phytoplankton

Phytoplankton carried through the thermal plume are not likely to be killed unless temperatures exceed about 36°C (97°F). Temperatures found to be lethal to algae were 37 to 38°C (98 to 100°F) for large diatoms and about 44°C (111°F) for green algae. Freshwater algae are often able to endure temperatures which are adverse for growth by forming resting stages. After temperatures return to normal, the algae recover.¹⁹

Lethal temperatures for marine algae appear to be somewhat lower, in general, than those for freshwater species. Effluent temperatures greater than about 32°C (90°F) may be lethal to some species. At the Oyster Creek generating station, changes in phytoplankton abundance and species composition were noted at the discharge outfall but were considered insignificant compared to normal seasonal and annual variations.¹³

II.H.1.1.2 Zooplankton

The zooplankton population in natural bodies of water include species that belong to several groups (e.g., protozoans, rotifers, microcrustaceans, coelenterates, dislodged benthic organisms, fish eggs, and larvae). Larval stages of barnacles, certain annelids, molluscs, and other invertebrates are a part of the zooplankton of estuarine and marine waters. The zooplankters feed on phytoplankton, protozoa, bacteria, smaller zooplankton, and organic detritus. In turn, they are sources of food for juvenile and adult fish. Passage of zooplankton through a thermal plume at temperatures greater than about 5°F above ambient may be lethal, particularly if the ambient temperature is high. For certain species that reproduce throughout the year, populations are expected to recover downstream. In the Green River in Kentucky, the volume of zooplankton decreased markedly upon passage through the condensers of the Paradise steam plant, but populations recovered downstream. The recovery was attributed to seeding by organisms bypassing the plant and having

accelerated reproduction in the warm, mixed river water downstream.⁸ Zooplankton species, such as amphipods and euphausiids which have long generation times and reproduce only one season a year, may have limited capacity for population recovery.²⁰

II.H.1.2 Sublethal Effects

Of potentially greater environmental impact than lethal temperatures are sublethal temperature increases brought about by a general warming of a larger area and volume of the body of water. The effects of sublethal increases in temperature are often difficult to detect until irreversible change has occurred. Such changes are related to the effect of temperature on metabolic and physiological processes of aquatic organisms.

II.H.1.2.1 Fish Distribution

Water temperature is a major factor in the geographical distribution of fish. The so-called "cold-water" streams are those capable of supporting salmonids, and the "warm-water" streams are those intolerable to salmonids.

In large lakes where thermal stratification occurs, cold-water fishes such as trout, char, and whitefishes remain in deep water during summer, while warm-water fishes are restricted to the shallow upper warm-water layer.

In the Wabash River in Indiana, studies were made of the distribution of fish near the discharge of a power plant. The effluent zone harbored only about 50% of the species collected either above the plant or farther downstream. Most species which were absent during the summer returned in the fall when temperatures were lower.²¹

In general, estuarine species and most marine species of the Mid-Atlantic region cannot tolerate prolonged exposure to temperatures above about 32°C (90°F).²² The areas of such discharges, therefore, will very likely be devoid of most fish except during the cold months, when many species will likely collect around the discharge depending on their preferred temperatures. Such concentration may result in the secondary adverse effects of overcrowding.

In a lake, river, or estuary in which the collective effects of a number of thermal discharges do not raise the natural temperatures of a major portion of the water body outside normal variations, the redistribution of fish will be local. However,

if the number of thermal discharges to a particular body of water is sufficient to raise the ambient temperature by several degrees, many species will eventually be eliminated from such waters, and high-temperature tolerant species will ultimately predominate. Sensitive areas are those waters in which particular fishes are at the southern fringes of their ranges. It is doubtful that any natural body of water will become devoid of fish as a result of power plant thermal discharges alone. However, the fish species usually considered desirable by Western man are those that prefer cooler water, and this factor should be considered in the siting and choice of a cooling system for the LMFB.

II.H.1.2.2 Fish Migration

Movements to and from the sea and fresh water are an essential part of the life cycles of anadromous and catadromous fish species, and temperature is apparently involved in the stimulation and direction of migration. It has been shown that the migration time and pathway of marine species such as bonito (Sarda sarda), flounder (Pseudopleuronectes), eel fry (Anguilla japonica), and sockeye salmon (O. nerka) were correlated with temperature.⁵ A thermal block that spans a major portion of a waterway due to a high-temperature discharge may deter some species from their normal migration route due to their avoidance response to excessive temperatures. Such blockage need not occur, however, if this factor is considered in the design of the discharge and cooling systems. Studies of shad (Alosa sapidissima) in the Connecticut River indicated that thermal discharges had blocked neither the upstream migration of adults nor the seaward migration of juveniles. The latter swam under the thermal plume and apparently suffered no detrimental effects.²³ In the Columbia River, the seaward migration of young chinook salmon is not adversely affected by the thermal discharge at Hanford, due largely to the fact that the migration occurs during the spring when flows are high and water temperatures comparatively low. The majority of salmon juveniles apparently migrate along the banks, and the thermal plumes represent a small part of the river mid-channel.²⁴ Upstream migrations of adult salmon which occur in spring (chinook), summer (chinook and sockeye), and fall (chinook and coho) do not appear to be affected by the thermal discharges at Hanford. On this reach of the river, the adults showed a general preference for migration along the eastern shoreline, across the river from the Hanford Unit 2 outfall.²⁵

In addition to migration of certain species for purposes of reproduction, other species may migrate up- and downstream in response to other stimuli. A 1968-1970 study of fish movement in New Hope Creek, North Carolina, indicated that most of 27

species showed a consistent pattern of larger fish moving upstream and smaller fish moving downstream. Both upstream and downstream movements were greatest in the spring.²⁶ In a lake or reservoir, these types of migration may not ordinarily occur. A return of certain species to their spawning or stocking areas in a lake, however, is a form of migration that can be affected by a thermal discharge.

The necessity for restriction of the thermal plume to a minor width, area, and volume of a waterway is obvious. The extent of such restriction will depend on both the specific site chosen and the habits of the fish using the particular waterway.

II.H.1.2.3 Fish Reproduction

Gonad development and spawning are highly temperature-dependent, but because this dependence is species-specific,¹¹ a definitive statement of an "adverse" temperature is difficult to make. Freshwater species such as brown trout (Salmo trutta), brook trout (Salvelinus fontinalis), chinook salmon (Oncorhynchus tshawytscha), northern pike (Esox lucius), grayling (Thymallus vulgaris), minnow (Phoxinus phoxinus), and walleye (Stizostedion vitreum) require temperatures below 10°C (50°F) to spawn.³ Warm-water species spawn at higher temperatures: carp (Cyprinus carpio) and large-mouth bass (Micropterus salmoides) spawn at 15 to 25°C (59 to 77°F), smallmouth bass (Micropterus dolomieu) at 12 to 16°C (53 to 61°F), and yellow perch (Perca flavescens) at 10 to 15°C (50 to 59°F).³

Spawning activity in marine species appears to be triggered by a temperature stimulus and may occur at a temperature change of only 1 or 2°C (2 to 3°F).⁵ Generally, higher temperatures hasten spawning in marine species and may also protract the spawning season of some. However, any benefit gained by extending the spawning season may be negated by overgrazing of the food supply.⁵

Estuarine species appear to be less critically stenothermal in their spawning requirements than oceanic fish. For example, striped bass (Morone saxatilis) spawn in California estuaries at about 16 to 21°C (61 to 69°F).⁵

Premature spawning of any fish species as a result of heated water discharges may put the larval development out of phase with the development of the normal food supply (e.g., plankton). The percent survival of the young may thus be decreased due to lack of proper food if an adequate variety of other food is not available. By this mechanism, water temperatures even slightly warmer than ambient in a spawning area may eventually lead to elimination of a species. Often, more than one species may use the same nesting site; for example, spring-spawning suckers

(Catostomus) may use the same sites as fall-spawning brook trout (Salvelinus fontinalis).²⁷ This aspect of thermal discharges is thus of greater concern than the temperature of the hottest part of the discharge plume.

II.H.1.2.4 Fish Metabolism, Growth, and Physiology

Within limits, the feeding rates of fishes, food transport, absorption, and digestion apparently increase with increases in temperature;¹¹ thus, when food is not limiting, increased water temperatures should result in an increased growth of fish. This increase in growth rate, however, may not always be beneficial. Cyprinodon, a eurythermal fish, showed a better initial growth rate at higher temperatures (to 30°C or 86°F), but this rate was not maintained later in life; the slower-growing fishes located in lower temperatures grew larger and lived longer.⁵ Brown bullhead (Ictalurus nebulosus) and white catfish (I. catus), resident in the thermal plume in winter at a power plant on the Connecticut River, showed a decline in weight-length ratios (condition) despite an abundance of benthic invertebrates. (It was not established, however, that these organisms were desired as food by the fish.) Channel catfish (I. punctatus) in the same study showed no decline in condition.⁸

Of importance to marine fishes is the effect of increased temperatures on osmoregulation. Brackish water euryhaline species are better able to osmoregulate than are saltwater stenohaline species. For example, a relatively small temperature rise resulted in partial failure of osmoregulation in Pacific cottid fishes, while two species of estuarine fish survived in a wide range of temperatures and salinities.⁵

II.H.1.2.5 Vulnerability of Fish Fry to Predation

Laboratory studies with rainbow trout (Salmo gairdneri) and chinook salmon (Oncorhynchus tshawytscha) indicated that thermally shocked juveniles were selectively preyed upon by larger fishes. The relative vulnerability to predation increased with duration of sublethal exposure to lethal temperatures.²⁸ Similar results were obtained in laboratory studies of yearling coho salmon (O. kisutch) predation rates on sockeye salmon (O. nerka) fry; predation rates increased with increasing acclimation temperature.²⁹

Although laboratory studies may have limited application to field conditions, the potential for decreased survival of migrating or resident fry encountering a warm plume can not be discounted. It is thus necessary to stress the importance of

maintaining a major portion of a river or estuary free of temperature increases above normal variations.

II.H.1.2.6 Incidence of Fish Diseases

Reports have been cited^{5,11} that implicate elevated water temperatures in increased rates of infestation by fish diseases and parasites. For example, near obliteration of a run of sockeye salmon in the Columbia River in 1941 was attributed to the combined effects of high temperature and bacterial infection. Columnaris disease has increased in the same river, reportedly due to river warming. Higher water temperatures also apparently increased the effect of kidney disease, vibrio disease and columnaris in young hatchery-reared salmonids. No relation has been found, however, between the occurrence of columnaris disease and power plant thermal effluents, as far as is known. There have been no fish population control problems because of this disease in studies at the Allen S. King plant on the St. Croix river.³⁰

II.H.1.2.7 Plankton and Benthos Responses

A common concern related to thermal discharges into natural bodies of water is that shifts in algal species from the more desirable diatoms to the less desirable blue-green algae may occur. In oligotrophic waters, diatoms usually predominate the phytoplankton community at water temperatures below 30°C (86°F), green algae predominate between 30 and 35°C (86 and 94°F), and blue green algae above 35°C (95°F).¹⁹ The latter are considered undesirable because they are not generally utilized by herbivores. Certain species have a capacity to fix nitrogen and thus accelerate the eutrophication of a water body and blooms can result in taste and odor problems of the water. Experience at power plant thermal discharges has indicated that these shifts can occur in the periphyton growing in the discharge area,^{4,8} but that they do not always occur.⁸ In north temperate lakes or streams, shifts in algal populations from predominantly diatoms to predominantly blue-green algae are not likely to occur from the thermal discharges since temperatures greater than about 30°C (86°F) are not expected.

There is some evidence that ocean siting may be of more concern in this regard. At the Turkey Point power station, replacement of the normal turtle grass (Thalassia) and macroalgae community by blue-green algae was related to increases in temperature of about 5°C (9°F) above ambient. Since the grass and macroalgal community provided shelter for invertebrates and young fish, many of these species were also eliminated.³¹ The giant kelp (Macrocystis pyrifera) is sensitive to temperature

changes. At water temperatures above about 15°C (59°F), gametophytes remain vegetative. At water temperatures above 21°C (68°F), floating fronds may become detached.³²

The occurrence of red-tide blooms in temperate seas has been related to temperatures of 18 to 25°C (59 to 77°F). These organisms (Gonyaulax catenella and G. catenata) have been responsible for mortalities of some crustaceans, molluscs, and fish off Southern California. Other species of dinoflagellates have caused the death of fish and shellfish in Santa Monica Bay, also in California.³² If these natural phenomena occur at a proposed coastal site, they should be given serious consideration relative to the potential effects of the plant's thermal effluent.

Marine intertidal organisms may require the seasonal variations in temperature, including high and low extremes, rather than simply tolerating them.¹⁸ A thermal discharge to intertidal areas could possibly eliminate or shift such temperature variations and seriously interfere with basic biological processes of the intertidal biota. Elimination of even a few species by this means could cause breaks in the food chain.

It is improbable that most freshwater zooplankton and benthic species will be adversely affected by thermal discharge outside of a 5°F isotherm. One area of possible concern is insect pre-emergence during the cold months. The rate of growth of aquatic insects is related to their ambient temperature. A temperature increase of only 1°C (1.8°F) has been shown to cause hydropsychid caddis flies to emerge two weeks earlier downstream than upstream from the Hanford reactors.³³ In a laboratory study, 10 species of aquatic insects exhibited premature emergence when subjected to unseasonable high winter water temperatures. The same experiments also indicated that the time between emergence of males and females of some species was increased by increased water temperatures.³³ In areas where air temperatures remain fairly high throughout the year, insect pre-emergence may be unimportant. In northern latitudes, however, where air temperatures are near or below freezing for several months, adult insects that emerge earlier than normal may freeze to death or be inactivated such that mating is prevented. Mating would also be prevented if the emergence of one sex occurred much earlier than the emergence of the other.³³

Widespread elimination of insect species from a body of water as a result of thermal discharges from a single 1,000-Mwe power plant is not expected. Any

effect of this kind, if it occurs, will be limited to the area of the plume within the 3°F isotherm.

II.H.1.2.8 Other Temperature-Related Effects

II.H.1.2.8.1 Gas Bubble "Disease" of Fish

The formation of gas bubbles in the blood of fish can occur when water becomes supersaturated with air. This supersaturation sometimes occurs when water that is close to saturation is heated. The condition can also be brought about when a large algal bloom results in high oxygen production during the day, or when water cascades over a dam. If the degree of supersaturation is great enough, the fish may show external symptoms such as "pop-eye," caused by bubbles in the tissues in or behind the eye. Incidence of this "disease" in about a dozen species of warm-water fish in the heated effluent of a steam generating station has been reported.³⁴ This potential effect can be largely avoided by preventing fish access to the discharge canal and minimizing turbulence at the outfall.

II.H.1.2.8.2 Chemical Toxicity Synergism

Many chemicals, including pesticides, appear to affect aquatic life more acutely at higher temperatures,^{5,35,36} either by increasing uptake, by lowering resistance to toxins, or by lowering tolerance to low oxygen levels in the water. These effects may become evident on inland rivers and estuaries which receive run-off from agricultural areas or discharges from industries. However, future limitations on the use of pesticides and on the discharges of chemicals are expected to help prevent these adverse effects. Before the choice of a site for a power plant is made, chemical characterization of the receiving water is essential if these effects are to be avoided.

II.H.1.2.8.3 Dissolved Oxygen

Depletion of the oxygen content of the receiving water to levels adverse for aquatic life is not expected to occur as a result of thermal discharges. In some cases, the aerating action of cooling towers or sprays can, in fact, increase the dissolved oxygen concentrations in the discharge compared with the intake concentrations. Loss of dissolved oxygen occurs if water that is saturated or supersaturated with oxygen is heated. Even at a temperature of 40°C (104°F) the solubility of oxygen in water is about 6.5 mg/liter,³⁷ which should pose no threat to aquatic life. However, in a river that receives high BOD or COD waste from other sources, or in a cooling lake with large algal blooms, low dissolved oxygen levels (less than 3 mg/

liter) may occur, particularly at night due to phytoplankton respiration and the absence of photosynthesis. In these cases, elevated temperatures, which may increase a fish's metabolic rates and hence its oxygen requirements, can aggravate the stressed condition brought about by the low ambient dissolved oxygen. This condition is not expected to occur in an ocean or large oligotrophic lake, nor in a cooling pond or reservoir that is properly managed to prevent algal blooms. It is also not likely to occur in a river or estuary that is unpolluted with high BOD or COD waste.

II.H.1.2.8.4 Changes in the Physical and Chemical Characteristics of the Water

Discharge of the heated effluent into estuarine or tidal waters can affect the density and salinity of such waters, particularly when saline effluent is discharged into a fresh or less-saline segment of an estuary. Heated effluents in an estuary have been shown to produce layering that held the main effluent stream of warm saline water below the surface.⁵ Salinity affects developing fish eggs and larvae, the responses being species-specific. Changes in the salinity and temperature cause changes in water density and viscosity, which in turn may affect the depth to which fish eggs will rise. Higher subsurface densities due to a saline discharge may cause fish eggs to rise to the surface, where they will be subjected to high temperatures, wave action, and light, all of which tend to decrease survival. Here again, the necessity of avoiding the release of a heated discharge into a spawning area is stressed if adverse impacts are to be prevented.

Discharge of an effluent that is drawn from the hypolimnion of a thermally stratified lake and discharged to the epilimnion could cause the circulation of nutrients from the lower layers into the surface layers where algal blooms could result.

II.H.2 ENTRAINMENT AND CONDENSER PASSAGE

During condenser passage, entrained organisms may be subjected to many adverse conditions: thermal shock due to heat transfer in the condenser; mechanical shocks and abrasion due to turbulence, cavitation, and collision with pump impellers and irregular surfaces in the system; pressure changes, due to pumping and hydrostatic head differences; toxic chemicals, including chlorine or other biocides used for shock defouling of condenser tubes; and additional thermal stress, turbulence, and heavy predation in the discharge area.

When an organism becomes entrained in a power plant cooling system and is subjected to one of the time-temperature courses discussed by Coutant,⁸ the experience is more complex than the step-function single temperature changes on which Coutant's principles⁸ were based. Rather, the initial quick temperature elevation experienced by an entrained organism in the condenser tubes is followed by a decline back to ambient water temperatures. The rate of such decline depends mostly on the design and location of the plant's discharge. Assuming that the ambient water temperature is within the entrained organism's zone of thermal tolerance, and the maximum cooling water temperature reached is within the zone of thermal resistance, then the "thermal dose" experienced by the organism will depend both upon the amount by which the upper incipient lethal temperature is exceeded and the length of time during which it is exceeded. This "thermal dose" may be viewed as one critical determinant in the fate of an entrained organism with regard both to lethal and certain sublethal effects.

A number of power plant studies provide descriptions of entrainment effects regarding diverse combinations of cooling system designs (temperature changes, times of exposure, biocide use, types of discharges, etc.), types of aquatic ecosystems, and methods of investigation. These studies verify that in many instances, considerable damage to entrained organisms has resulted from thermal stress, as well as from other types of stresses and shocks listed previously. In other cases, however, measurable damage has been slight. Examination of these studies in the aggregate produces some general observations regarding the causes and extent of entrainment damage.

Most studies of the effects of condenser passage on phytoplankton have principally employed various indirect physiological measurements, including those of primary productivity (usually by some modification of the methods of Strickland and Parsons³⁸ or Steeman Nielsen³⁹) and chlorophyll a concentration.

Such increases and decreases in primary productivity due to condenser passage appear to primarily represent sublethal, physiological effects on the phytoplankton and to be produced mainly by thermal stress during condenser transit. Except during periods of condenser chlorination, chlorophyll a concentrations have not been observed to decline when reductions in productivity are measured.⁴⁰⁻⁴⁵ Since chlorophyll a degrades rapidly upon cell death, it appears that the cells become metabolically impaired in such cases, but not killed. Thus, these effects have been interpreted by most investigators as stimulation and inhibition of photosynthesis. The higher the discharge temperature or temperature rise at a given power

plant, the more generally pronounced is the photosynthetic inhibition observed. Gurtz and Weiss⁴⁶ found progressively greater inhibition as discharged phytoplankton flowed down the long discharge canal of the Allen power plant.

A number of investigators (e.g., studies at the Waukegan and Chalk Point power plants) have reported that photosynthetic inhibition in condenser-passed phytoplankton may not become fully evident immediately after discharge. In addition to such delayed expression, there also is evidence that inhibition may be persistent in the affected cells.

Although some cases of mechanical damage to phytoplankton have been reported,^{43,47-49} such effects seem not to be nearly as evident or significant as those apparently due to thermal stresses.

Severe reductions in productivity of phytoplankton entrained during condenser chlorination have been observed in many studies.^{41,43,50-52} In a number of such observations, observed cell damage and loss as well as reduction in chlorophyll a content of discharge samples accompanied the reduced productivity;^{40,41,44} the implication was that such chlorination practices were destroying phytoplankton rather than only producing sublethal metabolic depression.

Both thermal and mechanical stresses have been observed by various investigators to be lethal to entrained zooplankton. Neomysis mortality as a result of condenser passage and under simulated laboratory experiments has been described as being influenced more by absolute discharge temperature than by temperature change.⁵³⁻⁵⁵ The significance for zooplankton of the duration of condenser transit as well as of temperature rise was reported.^{43,56} The estimates of zooplankton mortality due to mechanical damage alone range from 6% (reported for Waukegan power plant) to 12% (reported for Palisades and Point Beach plants).

The entrainment damage resulting from lethal time-temperature relationships is skewed against stenothermal species. Similarly, at plants where damage is mostly mechanical, the mortality is skewed against the larger organisms. Delayed mortality and reduced reproductive success of zooplankton and reduced hatchability of zooplankton eggs has been reported as a consequence of condenser passage.^{42,57,58} High entrainment mortality of zooplankton during condenser chlorination has been observed in a number of studies.^{57,59,60}

Generally, various stages of the fish life cycle are more susceptible to entrainment mortality than phyto- and zooplankton. As with zooplankton, mechanical damage to entrained young fish increases with the increasing size of the fish.⁶¹⁻⁶³ Mechanical damage to entrained fish eggs can also be substantial.

In a preliminary report on his investigations at the Connecticut Yankee power plant, Marcy⁶⁴ stated that at condenser discharge temperatures of 28, 33, and 35°C, the 93-second condenser transit at the plant was survived by only 35, 17, and 0% respectively of entrained fish larvae and fry. In addition, none of the young fish were observed to survive subsequent transit down the 1.83km-long discharge canal in summer when discharge temperatures exceeded 30°C. In a later report, Marcy⁶² attributed 72 to 87% of the observed mortalities to mechanical damage, with thermal stress responsible for the rest. Studies conducted at the Millstone, Chalk Point and Indian Point plants generally are in agreement with these findings. Chadwick⁶⁵ has identified 30 to 32°C as the condenser discharge temperature range which, if exceeded, can result in mortality of most entrained striped bass larvae from thermal effects alone.

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ASSUMPTIONS AND MODELS USED TO ASSESS ENVIRONMENTAL EFFECTSII.I.1 METEOROLOGICAL DATA

The weather conditions used in estimating the expected ground-level atmospheric concentrations of radionuclides that originate from the various fuel-cycle operations were obtained by averaging meteorological data taken from 18 stations in the continental United States. The locations of these stations are shown in Figure II.I-1.

These data consist of the joint frequency of wind speed and atmospheric stability category, which is independent of wind direction. They were averaged to obtain mean joint wind speed-stability categories in a typical 22.5° sector. The results are presented in Table II.I-1 in terms of seven discrete wind speeds and four stability categories. (For simplicity, the A and B stability categories have been combined to form B, and the E and F categories have been combined to form F.) The overall average wind speed is 4.05 m/sec and the average within the most stable category, F, is 4.03 m/sec.

Table II.I-1
AVERAGE SINGLE-SECTOR JOINT FREQUENCIES OF
WIND SPEED-STABILITY CATEGORIES

Wind Speed (m/sec)	Frequency for Pasquill Stability Category			
	B	C	D	F
0.8	0.0336	0.0192	0.0448	0.0688
2.5	0.0672	0.0448	0.0912	0.1328
4.5	0.0464	0.0496	0.0720	0.1024
6.9	0.0192	0.0256	0.0512	0.0528
9.5	0.00496	0.00576	0.0288	0.0272
12.5	0.00091	0.00015	0.00224	0.00192
15.6	0.00048	0.00022	0.00256	0.00464
Total	0.173	0.145	0.293	0.390

The Gaussian plume model developed by Pasquill¹ and Gifford² is used as a basis for the concentration calculation. The concentration has been averaged over the cross-wind direction³ to give the estimated average ground-level concentration downwind of the source of emission:

$$\bar{X}(x, u, -h) = \frac{8\sqrt{2} Q}{\pi^{3/2} u x \sigma_z(x)} \exp \frac{-h^2}{2\sigma_z^2(x)} \quad (1)$$

II.I-2

METEOROLOGICAL STATIONS

Figure II.I-1

where

$\bar{X}(x, u, -h)$ = average ground-level concentration in a 22-1/2° sector at a distance x meters downwind (Ci/m³),

Q = source strength (assumed constant) (Ci/sec),

u = wind speed (m/sec) (in x direction),

x = distance downwind (m) (in x direction),

h = effective source elevation (m),

$\sigma_z(x)$ = standard deviation of the concentration in the vertical direction (m).

The quantity $\sigma_z(x)$, a function of both downwind distance and atmospheric stability, has been determined experimentally² and is a measure of the degree of atmospheric dispersion.

The effective source height, h , is the sum of the actual source height, h_0 ; and Δh is the elevation above h_0 to which the effluent plume rises because of initial momentum or buoyancy.

For the case of a momentum-dominated plume, the equation of Holland⁴ is used to obtain this plume rise:

$$\Delta h \approx 1.5dw/u \text{ meters,} \quad (2)$$

where

d = stack orifice diameter (m),

w = effluent velocity (m/sec),

u = wind speed (m/sec).

For buoyancy-dominated (hot) plumes, Briggs's formula^{5,6} has been used:

$$\Delta h \approx 2.9 (F/uS)^{1/3} \text{ meters,} \quad (3)$$

where the buoyancy flux F is given by

$$F \approx 3.7 \times 10^{-5} q \text{ (m}^4/\text{sec}^3\text{),} \quad (4)$$

q = heat input (cal/sec)

and the stability parameter S is:

$$S = \frac{g}{T} \frac{\partial \theta}{\partial z} \quad (\text{sec}^{-2}), \quad (5)$$

where

g = acceleration of gravity (9.8 m/sec^2),

T = ambient temperature ($^{\circ}\text{K}$),

$\frac{\partial \theta}{\partial z}$ = potential temperature gradient ($^{\circ}\text{K/m}$).

Effluent released in the form of particulate matter accumulates on the ground at a rate that depends upon its settling or deposition velocity. This mechanism depletes the airborne plume, thus reducing the atmospheric concentration. On the other hand, deposition of this material on the surface results in contamination that may find its way into food chains.

Deposition from the plume that is taken into consideration by using a method devised by Chamberlain⁷ can be expressed in the form:

$$\bar{\chi}_d(x, u, -h, V_g) = \bar{\chi}(x, u, -h) \exp - \left[\frac{V_g}{u} \int_0^x \frac{\bar{\chi}(n, u, -h) dn}{\int_{-h}^{\infty} \bar{\chi}(n, u, z) dz} \right] \quad (6)$$

where

V_g = deposition velocity (m/sec),

u = wind speed (m/sec),

$\bar{\chi}(x, u, -h)$ = ground-level concentration without deposition (Ci/m^3),

$\bar{\chi}_d(x, u, -h, V_g)$ = ground-level concentration with deposition.

The rate of ground deposition is merely:

$$R_g = V_g \bar{\chi}_d \quad [\text{Ci}/(\text{m}^2 \text{ sec})]. \quad (7)$$

The value of the deposition velocity will vary depending upon particle size and density; however, for many particulates, V_g averages about 0.01 m/sec , a value that has been used in this Environmental Statement.

II.I.1.1 Normal Operations

The joint wind speed-stability-category frequencies listed in Table II.I-1 are independent of direction. To include the effect of wind direction, data on wind

direction from approximately 70 locations in the continental United States were examined,⁸ and the frequency of prevailing winds was found to vary from a theoretical minimum of 6.25% (neglecting calms) to a maximum of 25% with a mean value of 12.72% (in two adjacent directions). Consequently, to obtain an estimated average joint frequency of wind direction-wind speed-stability category, the values in Table II.I-1 were weighted by the factor 0.1272 in two adjacent directions; in the remaining 14 directions, the frequencies were weighted by the factor:

$$(1 - 2 \times 0.1272)/14 = 0.05326 .$$

Calculations of the estimated averaged concentrations were then made in each of the 16 directions by calculating the value of $\bar{\chi}$ using each combination of speed and stability listed in Table II.I-1 and by weighting the results by the appropriate frequency and summing.

In the case of the fuel-fabrication plant, the effective stack height of the source was taken to be 10 m. In the case of the reprocessing plant, the effluent is assumed to be discharged from a 100 m stack having an orifice diameter of 2.1 m with a velocity of 20 m/sec. Thus, the effective height of this source is, according to Eq. (2),

$$h = 100 + (1.5 \times 2.1 \times 20/u) = 100 + 63/u \text{ meters} .$$

Values of χ/Q for actual stack heights of 0, 10, and 100 m are presented in Table II.I-2. These values are not weighted by frequency of wind direction.

II.I.1.2 Accident Cases

Because of the very nature of the occurrence (i.e., over a short time period), it is less valid to assume the existence of average meteorological conditions during an accident than it is for the case of routine operation. Nevertheless, since this study is generic and because each site will be specifically examined to determine accident effects prior to licensing, there is some useful information concerning the order of magnitude of the environmental impact of various postulated accidents to be gained by examining average conditions.

For this reason, accidents have been treated using the meteorological frequencies listed in Table II.I-2. In this case, however, the frequencies are not further weighted for wind direction, because it is assumed that the wind direction remains constant throughout the duration of the release.

Table II.I-2
SECTOR-AVERAGED METEOROLOGICAL DISPERSION FOR AVERAGE
CONDITIONS AND FOR VARIOUS ACTUAL STACK HEIGHTS, h_0

Distance from Source (m)	χ/Q^a (sec/m ³)					
	No Deposition			Deposition Velocity, 1 cm/sec		
	$h_0 = 0$	$h_0 = 10$ m	$h_0 = 100$ m	$h_0 = 0$	$h_0 = 10$ m	$h_0 = 100$ m
400	2.05 E-4	8.85 E-5	6.64 E-7	1.38 E-4	8.28 E-5	6.63 E-7
600	9.47 E-5	6.03 E-5	1.63 E-6	5.77 E-5	5.39 E-5	1.63 E-6
800	5.57 E-5	4.16 E-5	1.72 E-6	3.16 E-5	3.52 E-5	1.70 E-6
1,000	3.69 E-5	3.01 E-5	1.54 E-6	1.98 E-5	2.41 E-5	1.52 E-6
1,500	1.82 E-5	1.62 E-5	1.12 E-6	8.79 E-6	1.16 E-5	1.10 E-6
2,000	1.08 E-5	1.00 E-5	9.20 E-7	4.83 E-6	6.51 E-6	8.95 E-7
3,000	5.75 E-6	5.46 E-6	6.66 E-7	2.25 E-6	3.04 E-6	6.36 E-7
4,000	3.68 E-6	3.54 E-6	5.19 E-7	1.31 E-6	1.75 E-6	4.84 E-7
5,000	2.62 E-6	2.53 E-6	4.26 E-7	8.54 E-7	1.13 E-6	3.87 E-7
10,000	9.47 E-7	9.30 E-7	2.40 E-7	2.33 E-7	2.96 E-7	2.02 E-7
15,000	5.38 E-7	5.30 E-7	1.72 E-7	1.09 E-7	1.34 E-7	1.34 E-7
20,000	3.60 E-7	3.56 E-7	1.35 E-7	6.23 E-8	7.59 E-8	9.62 E-8
40,000	1.47 E-7	1.46 E-7	6.93 E-8	1.61 E-8	1.89 E-8	3.36 E-8
60,000	8.72 E-8	8.66 E-8	4.61 E-8	6.93 E-9	7.91 E-9	1.58 E-8
80,000	6.01 E-8	5.97 E-8	3.42 E-8	3.74 E-9	4.18 E-9	8.78 E-9

^a χ/Q is the ratio of the ground-level air concentration to the source strength.

The ground-level concentrations and surface concentrations are computed in exactly the same way as for the case of normal operation except that only a single direction is considered and the release is of finite duration.

Source heights used were identical to those used for normal operation in the case of fuel-fabrication and reprocessing plants. For those transportation accidents that do not involve a fire, the release is assumed to occur at ground level.

For transportation accidents involving a fire of sufficient severity to cause significant damage to a shipping cask, the plume rise due to buoyancy was estimated using the Briggs formula.

For this purpose, the potential temperature gradient $\partial\theta/\partial z$ was taken to be 0.027°K/m, based upon average values for E and F conditions found in AEC Regulatory Guide No. 23. The ambient temperature was chosen to be 300°K (about 80°F), so that

$$S = \frac{9.8}{300} \times 0.027 = 8.82 \times 10^{-4} \text{ sec}^{-2}$$

By putting this value into the Briggs formula and using the expression for the buoyancy flux from Eq. (3), we have

$$\Delta h \sim (q/u)^{1/3}.$$

The assumption made in Section 4.5.8.16 was that in order to generate a high enough temperature long enough to open one of the shipping casks, a value of $q \sim 5 \times 10^7$ cal/sec is required. The average wind speed is about 4 m/sec. Hence, $\Delta h \sim 230$ m for these cases.

II.1.2 DEMOGRAPHIC DATA

A site for a fuel-cycle facility is not likely to be selected within a heavily populated area. Many potential sites have populations of less than 500,000 within a radius of 50 miles; actual sites will probably be selected from among these areas. Since increases in population may occur over the years in the vicinity of a fuel-cycle facility, a population of 1,000,000 people uniformly distributed within a 50-mile radius was used in the calculations of population doses. The population distribution for each of sixteen 22.5° sectors emanating from the plant is given in Table II.I-3.

Table II.I-3

POPULATION DISTRIBUTION FOR EACH 22.5° SECTOR
FOR A TYPICAL FUEL CYCLE FACILITY SITE

Distance (mile)	Population
0-0.622 ^a	0
0.622-1	15
1-2	75
2-3	130
3-4	180
4-5	220
5-10	1,900
10-20	8,000
20-30	12,000
30-40	18,000
40-50	22,000
Total	62,520 x 16 = 1,000,320

^a1000-m plant boundary.

This population is equivalent to an average population density of ~127 persons/sq mile. Population densities (persons/sq mile) for several states in the United States in 1971 were: New York, 384; New Jersey (highest value), 969; South Carolina, 87; Virginia, 118; and Wyoming (lowest value), 3.5.⁹

II.I.3 RADIOLOGICAL-EXPOSURE PATHWAYS

Environmental transport links the source of release to the receptor by numerous exposure pathways. The most important pathways that result in the exposure of man from environmental releases of radioactivity are represented diagrammatically in Figure II.I-2. The resulting radiation exposures may be either external or internal or both. External exposures are those with the radiation source outside of the irradiated body, and internal exposures are those due to radioactive materials within the irradiated body.

Atmospheric releases of radioactive gases and particulates may result in external doses via immersion and exposure to a contaminated land surface and in internal doses via inhalation and ingestion. Man immersed in a radioactive plume receives an external immersion dose and an intake of radioactivity via inhalation as long as the plume or resuspended radionuclides are present. Deposition of radionuclides from the plume on vegetation and on the soil creates the possibility for man to be exposed externally to the contaminated surface and to be exposed internally via the food chain. Following initial deposition, the potential for exposure of man may persist, depending on the influence of environmental redistribution, long after the plume leaves the area. Radionuclide concentrations at the point of deposition normally are reduced by infiltration into the soil profile, loss of soil particles due to erosion, and transport in surface water and in groundwater. When the effects of these processes cannot be quantified, a conservative estimate of dose due to external exposure to the contaminated surface is obtained by assuming that the radionuclide concentrations are diminished by radioactive decay only. Man's dietary intake of deposited radionuclides is predicted with an expansion of the "terrestrial vegetation" and "animals" boxes in Figure II.I-2. The resulting model,¹⁰ a simplification of a real terrestrial food-pathway system, is incorporated into a computer code, identified by the acronym TERMOD, that is used to obtain information on food-chain transfers to man prerequisite to calculation of internal dose. The terrestrial food-chain model provides predictions of radionuclide intakes by man through consumption of milk, beef, and plants contaminated by direct deposition as well as by uptake from the soil. This generalized dynamic model is sufficiently versatile that it can be applied to many terrestrial environments and to all radionuclides.

PATHWAYS FOR EXPOSURE OF MAN
FROM RELEASES OF RADIOACTIVE EFFLUENTS

Figure II.I-2

Environmental release of radioactivity in liquid effluents as shown in Figure II.I-2 may result in external exposure of man by submersion in water during swimming or bathing and by exposure to a contaminated shoreline or to emergent sediments in a tidal area. Aquatic releases may also result in exposure of man via numerous food-chain pathways. Radionuclides released in liquid effluents are assumed to be diluted in the receiving water/body prior to the estimation of dose to man via drinking water and eating fish. Estimates of dose due to the use of this water for irrigation are obtained with the TERMOD computer code.¹⁰ Dose to man from exposure to a contaminated shoreline is estimated using a sediment model.¹¹

Additional external irradiation of man is possible due to exposure at some distance by gamma radiation. Pathways that exemplify this type of direct radiation exposure include plume overpassage, boating on, or standing near a body of contaminated water, and exposure from gamma radiation penetrating shipping containers.

II.I.4 MODEL FOR RADIOLOGICAL DOSE TO AN INDIVIDUAL

Radiation dose is estimated per year of plant operation for normal releases and per event for accidents. All exposures discussed are hypothetical, and the doses calculated and presented are estimates of the doses that could occur under the specified conditions. Doses are estimated both for individuals and for the population living within 50 miles of the facility. Assumptions and dose conversion factors used in the calculations are presented in Appendix II.E.

Factors for converting external radiation exposures to dose were obtained with a computer code EXREM,¹² which contains models adapted from standard texts.^{13,14} The dose rate above the contaminated land surface is estimated for a height of 100 cm. The models for immersion in air and submersion in water assume that the entire body surface is in contact with a large volume of contaminated air or water. For all external exposures, the value calculated is for the air dose at the surface of the skin. The total-body dose is assumed to equal the gamma air dose at the skin surface for all external exposures.

Factors for converting internal radiation exposure to estimates of dose were obtained with models and with data published by the International Commission on Radiological Protection (ICRP).¹⁵⁻¹⁸ Those models are programmed in the INREM and INDOS computer codes,^{19, 20} which were used to complete the calculations for inhaled and ingested radionuclides. Internal exposure continues as long as radioactive material remains in the body, which may be longer than the time period of

the individual's residence in the contaminated environment. The best estimates of internal dose resulting from an intake are obtained by integrating over the remaining lifetime of the exposed population; such estimates are referred to as "dose commitments." The remaining lifetime is assumed to be 50 years for an adult. Throughout the text and tables, use of the term "dose" should be understood as "dose commitment" whenever contributions from internal exposure pathways are included. All dose estimates given are for adults unless otherwise specified. The INREM computer code does provide the capability for estimating dose as a function of the age of the individual exposed. When nonadult dose estimates differ significantly from adult values, they are presented and discussed under the appropriate exposure pathway.

Dose is estimated for individuals located at and beyond the site boundary for the most significant exposure pathways. The location of maximum potential exposure is identified and evaluated, with that point assumed to have the highest concentration of radionuclides in the air and on the land surface. A further assumption made is that the exposed individual resides continuously at the location, with no allowance for possible protective shielding provided by a residence, and that this location is the point of origin for all food requirements. Estimates of dose were made for total body and for a number of reference organs, and those radionuclides that contribute large fractions of the total dose are identified. Where significant, the estimates of dose to organs other than the total body are discussed.

II.1.5 MODEL FOR RADIOLOGICAL DOSE TO THE POPULATION

The total dose received by an exposed population due to releases from a facility was estimated by summation of individual dose estimates within the population. The area within 50 miles of the facility was subdivided into 16 sectors (22.5° each) and into a number of annuli. The average dose for an individual in each subdivision was estimated, that estimate was multiplied by the number of persons in the subdivision, and the resulting products were summed across the entire area. Unless otherwise specified, the dose estimates summed are those for total body, and the unit used to express population dose is man-rem. Population dose estimates were calculated for a population assumed to be composed entirely of adults and is estimated only for those exposure pathways known to involve a significant number of persons. Unique aspects of a specific man-rem estimate are discussed in conjunction with estimates of individual dose for that particular exposure pathway.

II.1.6 ASSUMPTIONS FOR RADIATION EXPOSURE RESULTING FROM TRANSPORTATION UNDER NORMAL CONDITIONS

The radiation exposure to individuals from normal transportation of radioactive materials in the fuel cycle is normalized to one-way trip distances of 1000 miles and other information.²¹

The DOT has set radiation-level standards for shipments of radioactive materials under the "exclusive-use" conditions.²² These conditions specify that the vehicle is loaded by the consignor and unloaded by the consignee and also that the radiation level at 3 ft from the surface of the package is limited to 1000 millirems/hr, provided the radiation level does not exceed 200 millirems/hr at the surface of the vehicle, 10 millirems/hr at 6 ft from the outside surfaces of the vehicle, and 2 millirems/hr in either the driver's compartment or other normally occupied positions in the vehicle.

For the purpose of estimating the radiation dose to individuals and to the population resulting from normal transport operations, it is assumed that exclusive use conditions are met and those packages that require some shielding will have dose rates that coincide with the DOT limits.

Some packages in the fuel cycle (e.g., those containing UF_6 or tritium) do not have any significant amounts of penetrating radiation associated with them, and, therefore, contribute essentially nothing to accumulated radiation doses to workers or to the general public. Experience²¹ indicates that, for the LWR fuel cycle, shipments of material containing only very limited quantities of penetrating radiation contribute only about 1% of the exposures allowed by the DOT regulations. However, since no operating experience exists for shipping within the LMFBF fuel cycle, low-level items will be assumed to contribute 10% of the allowable exposure.

Most packages are put on one vehicle and are not transshipped; that is, they are not handled again until they reach their destination. However, spent fuel shipments, in some cases, may be different. Some reactor sites will not be equipped with a rail siding, so the cask may have to be trucked to the nearest railhead and transferred to a suitable flat car. This additional handling increases the dose to transport workers. Because of the lack of information about rail capabilities of each reactor site, it is conservatively assumed that all spent fuel is transshipped. Both the fuel-fabrication and fuel-reprocessing plants are assumed to have rail facilities.

Transport workers and the public will be exposed to radiation from these shipments. The exposures received by different groups are based on the dose rates described above and the average time that the various groups are associated with the shipment described below.

- (1) General public--onlookers. Ten persons per shipment are assumed to spend an average of 3 min at a distance of 3 ft from either a truck or a rail car.
- (2) General public--along the route. An average of 330 persons/sq mile is assumed to be uniformly distributed along the truck and train routes. An exclusion zone of 100 ft from either side of the truck or train route is also assumed; that is, no persons live within that zone.
- (3) Truck drivers. Two truck drivers during a 1000-mile trip will probably spend no more than 20 hr in the cab and about 1 hr outside the truck at an average distance of 3 ft from the cargo compartment.
- (4) Garagemen. For truck shipments, normal servicing will probably require two garagemen to spend about 10 min around the truck cab.
- (5) Brakemen. For rail shipments, train brakemen would be expected to spend about 10 min in the vicinity of the rail car; 10 different brakemen are assumed to be involved along the route.
- (6) Freight handlers. Freight handlers are assumed to be needed only in the transfer of the spent fuel cask from the truck to the rail car en route from the reactor to the fuel-reprocessing plant. In transferring the cask from truck to rail car, two workmen are assumed to work 1/2 hr at an average distance of 3 ft from the cask.

The results are given in Table II.I-4. The exposures are broken down into three categories. The first (called low level) involves shipment of radioactive material having essentially no penetrating radiation associated with it; the second (called high level) involves shipments in which the external dose rates were assumed to coincide with the DOT limits and which are not transshipped; the third involves irradiated fuel in which the external dose rates correspond with maximum DOT limits but which are transshipped.

Table II.I-4

ESTIMATED RADIATION DOSES PER 1000 MILES
OF SHIPMENT UNDER NORMAL CONDITIONS

	Exposure Time per 1000 Miles of Transportation (hr)	Exposure Rate per Shipping Mode (millirems/ hr)		Number of People Involved per Shipping Mode		Dose Per Shipping Mode per 1000 Miles (man-rems)	
		Truck	Rail	Truck	Rail	Truck	Rail
<u>Low Radiation Level Materials, Not Transshipped</u>							
Drivers							
Operation	20	0.2		2		0.008	
Cargo Inspection	1	5		2		0.010	
Garagemen	0.15	0.2		2		0.0001	
Brakemen	0.15		5		10		0.0075
Onlookers	0.05	5	5	10	10	0.0025	0.0025
Population		a	a	330,000	330,000	0.0018	0.0018
Total						0.0224	0.0118
<u>High Radiation Level Materials, Not Transshipped</u>							
Drivers							
Operation	20	2		2		0.08	
Cargo Inspection	1	50		2		0.10	
Garagemen	0.15	2		2		0.001	
Brakemen	0.15		50		10		0.075
Onlookers	0.05	50	50	10	10	0.025	0.025
Population		a	a	330,000	330,000	0.018	0.018
Total						0.224	0.118
<u>Irradiated Fuel, Transshipped</u>							
Drivers							
Operation	1		2		2		0.004
Cargo Inspection	0.25		50		2		0.025
Freight Loaders	0.50		200		2		0.20
Brakemen	0.15		50		10		0.075
Onlookers	0.05		50		10		0.025
Population			a		330,000		0.018
Total							0.347

^a 1.8×10^{-5} man-rem/1000 miles.

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SYNDROME OF HYDROGEN-FLUORIDE EFFECTS ON BIOTA

The most common end point used to gauge effects from hydrogen-fluoride (HF) exposure is the occurrence of chlorotic or necrotic lesions on leaves. The syndrome of HF-induced effects is given in Table II.J-1 and the sequence illustrates the progression of response beginning with incipient effects on cells and tissues and culminating as effects on organs and individuals. Chlorotic lesions, the most common symptom, are midway between incipient effects and death of the organism. The important implication of this sequence is that subtle physiological and biological effects have already occurred by the time gross pathological effects (chlorotic, necrotic lesions) develop. For example, inhibition of photosynthesis has been observed at exposures that do not cause foliar necrosis.¹ Plant vigor may be affected at HF concentrations less than those which produce visible symptoms.

Table II.J-2 summarizes the relative effects of fluoride on vegetation, cattle, and humans. For continuous exposure, visible effects on vegetation have been detected at fluoride concentrations as low as $0.2 \mu\text{g}/\text{m}^3$. Concentrations of HF in air that may affect man range from 0.1 to $100 \text{ mg}/\text{m}^3$. Generally, animals are not affected by the concentration of fluoride in air resulting from pollution sources. However, animals exhibit fluorosis and dental mottling (Table II.J-2) when the fluoride concentration of their forage routinely exceeds 30 to 50 ppm. Repression of food intake occurs when the fluoride concentration of forage regularly exceeds 60 ppm.²

Vegetation is damaged by HF in either continuous or intermittent exposures.³ Estimates of potential impact from continuous exposures to HF that exceed threshold concentration for effects can be estimated from Figure II.J-1. However, estimates of potential impact are less certain when the product of concentration multiplied by duration of each intermittent exposure does not exceed the threshold level, but the sum of intermittent exposures does exceed the threshold. Intermittent exposures may be summed and considered as a continuous exposure,³ but the prediction of effect from this type of exposure is not as straightforward as for continuous exposure. A comparison of two different concentration frequencies (i.e., two different patterns of intermittent exposure of pine) illustrated a 10-fold difference in effect (66 vs 6% injury). The 66% injury was manifested at 2 to 4 times higher HF concentrations for equivalent periods of exposure. The 6% injury occurred at intermittent exposure to concentrations of HF ranging from 0.6 to $9.5 \mu\text{g}/\text{m}^3$, but both dose-response trends tended to converge when the trees were exposed to a concentration of $1 \mu\text{g}/\text{m}^3$ for 225 hr of cumulative exposure. Thus, effects manifested at intermittent exposures may approximate those induced at

Table II.J-1

NATURE OF FLUORIDE-INDUCED EFFECTS IN PLANTS
AT FOUR LEVELS OF BIOLOGIC ORGANIZATION

Cellular	Tissue or Organ	Organism	Ecosystem
(Altered Cell Milieu?)			Increased F in ecosystem
Effects on Enzymes and Metabolites	Decreased assimilation Altered respiration		
Modification of Cell Organelles and Metabolism	Altered growth and development	Modified growth	Increased in F burden of animals
Pathway Disruption	Chlorotic lesions	Reduced reproduction	Fluorosis in animals
Cellular Modification		Decreased fitness for environment	Change in plant community
Disruption and Death of Cell	Necrotic lesions Death or abscission of leaf	Death of plant	Desolation

Source: National Academy of Sciences, "Biological Effects of Atmospheric Pollutants: Fluorides," U.S. Government Printing Office, Washington, D.C., 1971, Table 7-1.

Table II.J-2

SUMMARY OF THE EFFECTS OF FLUORIDE ON
VEGETATION, CATTLE, AND HUMAN SUBJECTS

Mean concentrations		Mean daily intake of fluoride, mg/day	Duration of exposure	Species	Effect reported
Hydrogen fluoride in air (mg/m ³)	Total fluoride in ration, diet, or water (mg/kg)				
0.00025	—	—	7 weeks	Chinese apricot	Less than 1% necrosis on foliage
<0.0005	—	—	18 months	Citrus	No effects on growth or physi- ology of plant
0.0006	—	—	1 week	Sorghum	Threshold for foliar markings
≈0.0007	—	—	2 weeks	Sorghum	Foliar lesions and reduced growth if exposed during anthesis
0.0009	—	—	17 weeks	Orchard grass and alfalfa	No effect on yield, no foliar lesions
0.0011	—	—	7 weeks	Chinese apricot	Some necrosis on foliage
0.0015	—	—	1 day	Gladiolus	Less than 2% necrosis
			1 day	Ponderosa pine	Less than 1% necrosis on young needles
			1 week	Sweet corn	Threshold for foliar markings
0.0018	—	—	2 weeks	Apple	Threshold for foliar lesions
0.0021	—	—	10 weeks	Bean	Significant effect on reproduc- tion, but no foliar lesions
0.0029	—	—	22 weeks	Tomato	No effect on growth nor injury to foliage
0.0050	—	—	1 day	Gladiolus	About 5% necrosis of leaves
0.0063	—	—	2 weeks	Tomato	No injury
0.010	—	—	1 day	Tomato and some tree fruits	Threshold for foliar lesions
≈0.1	—	—	Single exposure	Man	Perceptible odor concentration for hydrogen fluoride
2.5	—	—	Daily	Man	8-hr daily exposures ACGIH threshold limit value for hydrogen fluoride
≈25.0	—	—	Single exposure	Man	Tolerable, with conjunctival and respiratory discomfort only for several minutes
≈100.0	—	—	Single exposure	Man	Highest tolerable concentration for 1 min, marked conjunctival and respiratory irritation
	—	0.2–1.2	Daily	Man	Average daily intake from American diet and water supply
	2–8 ^a	—	Years	Man	Mottled enamel ^b
	20–30	—	Several years	Cattle	Threshold of discernible dental mottling ^b
	30–40	—	Several years	Cattle	Evident dental mottling ^b
	40–50	—	Several years	Cattle	Threshold of enamel hypoplasia, ^b moderate periosteal hypertosis
	>50	—	Several years	Cattle	Decreased milk production, significant incidence of lameness
	40 ^d	—	Several years	Man ^e	Loss of body weight
	>50 ^d	—	Months to years	Man ^e	Thyroid injury
	>100 ^d	—	Months	Man ^e	Renal injury
	—	20–80	10–20 years	Man	Crippling fluorosis
	—	2,500–5,000	Single dose	Man	Death in 2 to 4 hr

^aIn water.^bDental effects seen only if fluoride is ingested during period of tooth formation.^cBased on feeding experiments lasting 1.6–8 years.^dIn diet.^eBased on data from animals extrapolated to man.Source: National Academy of Sciences, *Biological Effects of Atmospheric Pollutants: Fluorides*, U.S. Government Printing Office, Washington, D.C., 1971, Table 11-1, pp. 238–239.

II.J-4

POSSIBLE AIR-QUALITY CRITERIA FOR ATMOSPHERIC FLUORINE,
WITH REFERENCE TO DIFFERENT PLANT SPECIES

Figure II.J-1

continuous exposure when the cumulative period is 10 days or more. Research is needed on the question of impact from intermittent vs continuous exposure for cumulative exposure periods greater than 10 days.

II.J.1 ACCUMULATION OF FLUORIDE BY PLANTS

Fluoride uptake by plants, particularly by forage species, is important because concentrations greater than 30 to 50 ppm in forage cause fluorosis in cattle as indicated in Table II.J-3. Dental mottling has been observed when livestock were continuously fed rations containing these concentrations of fluoride. At higher concentrations (approximately 50 to 300 ppm) lameness and repression of food intake have been observed.⁴ At higher concentrations of fluoride (300 to 600 ppm), feed consumption is reduced with resultant losses in weight and reductions in lactation.^{4,5} Low ambient concentrations (0.002 to 0.09 $\mu\text{g}/\text{m}^3$, Table 4.3.6.1, Section 4.3.6.2) by themselves are not deleterious to livestock, but, following long periods of exposure, toxic levels of fluoride accumulate in grass from deposition on foliage and assimilation by plant tissues.⁴ Unfortunately, accumulation and weathering processes have not been studied for chronic exposures at low ambient concentrations for longer than several weeks. Concentrations in orchard grass are given for HF concentrations $>2 \mu\text{g}/\text{m}^3$ and for 12-day exposures in Figure II.J-2.

There are insufficient data to permit extrapolation of the curves in Figure II.J-2 to consider accumulation for longer periods of exposure; for example, 10 to 100 days. Furthermore, the curves cannot be used to predict fluoride accumulation in grass at air concentrations less than 1 to 2 $\mu\text{g}/\text{m}^3$. Therefore, fluoride concentration in forage for selected periods of exposure was estimated from plume depletion, using the deposition velocity approach. A deposition velocity of 0.003 m/sec was derived from existing data; this value applies specifically to fluorine (Appendix II.K).

Table II.J-3
SAFE LEVELS OF FLUORIDE IN THE TOTAL RATION OF LIVESTOCK

Animals	Fluoride Content (ppm)	
	As Sodium Fluoride or Other Soluble Fluoride	As Phosphatic Limestone or Rock Phosphates
Dairy Cow	30-50	60-100
Beef Cow	40-50	65-100
Sheep	70-100	100-200
Swine	70-100	100-200
Chicken	150-300	300-400
Turkey	300-400	-

Source: National Academy of Sciences, "Biological Effects of Atmospheric Pollutants: Fluorides," U.S. Government Printing Office, Washington, D.C., 1971, Fig. 8-5.

RELATION OF FLUORIDE CONCENTRATION IN ORCHARD GRASS
TO CONCENTRATION OF HF AND DURATION OF EXPOSURE.

Figure II.J-2

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Appendix II.K

DEPOSITION VELOCITY FOR HF

A deposition velocity (V_d) of 0.003 m/sec was derived from existing data.¹ The advantage of using V_d for estimating ground concentrations of fluoride is that the function is easily coupled with the atmospheric dispersion model. The definition of V_d is

$$V_d = \frac{\frac{\text{surface deposition}}{\text{unit time}}}{\text{air concentration}} \quad \text{in terms of} \quad \frac{\frac{\mu\text{g}}{\text{m}^2 \cdot \text{sec}}}{\frac{\mu\text{g}}{\text{m}^3}}$$

This approach assumes that the plume is an infinitely large source of HF. An average V_d of 0.003 m/sec was derived from data¹ (Table II.K-1) and for a surface to mass conversion of 175 cm²/g for leaves at peak canopy development.² Fluoride concentrations in vegetation were determined by dividing ground concentrations by the standing crop of forage, 150 g/m², the same value used for evaluating radionuclide transfer in the soil-grass-cow-milk pathway.³

Table II.K-1
CALCULATED V_d BASED ON OBSERVED FUMIGATION CONDITIONS^a

<u>Fumigation Conditions</u>		<u>Net Fluoride Content of Forage (ppm)</u>	<u>Calculated V_d (m/sec)</u>
<u>Concentration ($\mu\text{g}/\text{m}^3$)</u>	<u>Duration (day)</u>		
5	1	27	.0036
40	3.125	767	.0041
40	3.125	477	.0025
$\bar{x}$			.0034

^aFumigation data taken from Tables 7-3 and 7-4 of ref. 1.

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Appendix II.L

ASSUMPTIONS AND PROCEDURES FOR CALCULATING FLUORIDE ACCUMULATION IN FORAGE

A deposition velocity of 0.003 m/sec transfers HF to the ground surface for selected time intervals. Fluoride concentrations in forage were evaluated for two discrete intervals: 5 days and 18 days. The 5-day interval would represent the maximum length of exposure under severe meteorological conditions. The fluoride concentration in vegetation is based on a forage density of 150 g/m². A loss rate function of $e^{-0.032 \cdot D}$ accounts for fluoride decreases in vegetation due to weathering losses, intraplant translocation, and dilution resulting from growth processes.¹ Net fluoride concentrations in forage (C in µg/g or ppm) were calculated from the relationship,

$$C = \frac{AV_d t e^{-0.032 D}}{150}$$

where

- A = concentration in µg/m³,
- V_d = deposition velocity, 0.003 m/sec,
- t = interval of accumulation in seconds,
- $e^{-0.032 D}$ = the weathering function (D = days), and
- 150 = the forage density in g/m².

REFERENCE FOR APPENDIX II.L

1. D. C. McCune, "Problems Involved in Devising Air Quality Criteria for the Effects of Fluorides on Vegetation," Am. Ind. Hyg. Assoc. J. 32(10): 697-701 (1971).

REGULATORY STANDARDS AND REQUIREMENTS

Packaging and transport of radioactive materials¹ are regulated at the Federal level by the AEC,² by the DOT,³⁻⁵ and by the U.S. Postal Service.⁶ Certain aspects, such as limitations on gross weight of trucks and transportation not subject to DOT, AEC, or the Postal Service regulations, are regulated by the states. Many states have adopted regulations that pertain to intrastate transportation of radioactive materials and require the shipper to conform to the DOT's packaging, labeling, and marking requirements to the same extent as if the transportation were subject to the rules and regulations of that agency.

II.M.1 PACKAGING STANDARDS AND REQUIREMENTS

The packaging standards and criteria are found in the regulations of the AEC (10 CFR 71) and in the regulations of the DOT (49 CFR 170-179).

These regulations are written so as to require all packages that carry significant quantities of radioactive materials to be capable of withstanding destructive forces and temperatures likely to be encountered in severe transportation accidents. The criteria specified in the regulations are a set of tests that can be made in either the laboratory or the field. If, from actual tests of similar packages, sufficient information is known about the behavior of the proposed package and materials of construction under conditions specified in the regulations, a comprehensive analysis is acceptable.

The accident criteria, which were first published by the International Atomic Energy Agency in 1964, have been adopted in many international and national transportation regulations and have served as the basis for the regulatory standards and criteria of the U.S. They were based on a detailed analysis of normal and accident conditions in worldwide transportation environments involving nonradioactive cargo and on nearly 20 years of experience in shipping many types of radioactive materials.

Regarding this analysis the Natural Resources Defense Council (NRDC) suggested that the criteria may be outdated and that revisions in the analysis may be required. They were also concerned about the impact the deterioration of the condition of the Nation's railroads would have. It should be noted that the analysis referred to involved the study by members of the various competent authorities of various countries in developing the original IAEA regulations. The AEC regulations were patterned after the International Regulations and are compatible with them. The

International Body of Competent Authorities meet at intervals to evaluate the adequacy of the regulations. They have recently issued revised international regulations which will undoubtedly spark changes in the U. S. regulations. In its latest revision, the International Body made no change in the 30 ft. free fall (9 meter), 1/2 hr. fire, or 40 inch (1 meter) puncture drop. The submergence criterion was modified to account for potential loss of packages in deep harbor areas. As to the concern about the nation's railroads, any deterioration of rail facilities would significantly raise the accident rate, which was considered through 1970 (see Appendix II.A). Also, the number of shipment miles would increase the number of accidents. This is already accounted for in DOT statistics. The number of shipment miles has increased. This effect is accounted for in the projection of shipments in future years in the Environmental Statement.

Type B package design (Appendix II.D) must be analyzed in accordance with the regulations and be reviewed by the AEC. The AEC not only considers the results of the analysis but also the methodology and codes used in the analysis. In most cases, the results are checked independently by use of the best available information that is applicable to the problem. The application for approval must also outline a quality-assurance program to ensure that the package will be built in strict accordance with the plans submitted to the AEC.

The NRDC stated, regarding package design review that it is not correct that all package designs are comprehensively analyzed and undergo rigorous experimental testing* -

First, not all package designs are actually subjected to the "design-basis" accident (10 CFR Part 71, Appendix B). Second, only initial versions of some packages are tested, i.e., after production is begun, each package that rolls off the production line is not tested. Third, some package designs are not subjected to the tests at all--reliance is placed solely on paper analyses.

The draft environmental statement asserts, without substantiation, that regulations ensure that a shipping container will not be manufactured or used in a manner inconsistent with its design.

By claiming, in essence, that every shipping container would pass the tests that are described in the regulations, the AEC, significantly, is able to ignore the possibility of a 'faulty' package in its accident analysis. Unfortunately, this optimistic assumption, which is very favorable to the AEC's point-of-view, is unwarranted. The GAO, for instance, has discovered that the

*Comment Letter 38c, pp. 9, 10.

AEC does not adequately police the manufacture and use of packages, thereby raising the possibility that some packages may be faulty and/or improperly used.

Obviously, since the tests are destructive, all casks cannot be tested. Full scale tests carry considerable weight in a PSAR (Package Safety Analysis Report) if they can be carried out practically. Model testing is considered next best. If no tests are carried out on a specific design then the methods of analysis must be detailed and considered acceptable by the AEC. Personnel in the Directorate of Regulation do an independent analysis of important results which appear in PSAR's.

It is intended that adequate QA programs be implemented to ensure the uniformness of all type B packages. The GAO investigation was carried out during the time the AEC had its own concerns about quality assurance and revised their regulations, for contractors, requiring the development of an adequate QA program. About six weeks before the GAO report was issued, AEC modified its own procedures to improve its quality assurance in package fabrication, maintenance, and use. On December 28, 1973 AEC proposed a regulation to impose comprehensive QA requirements on licensees, and is already evaluating licensee QA programs. The AEC intends to have uniformly fabricated packages. QA reviews of package designs in the field have started and it is reasonable to expect these reviews to continue at an accelerating pace.

The packaging must function in the normal transportation environment with a high degree of reliability. Systems selected to achieve the basic design functions (i.e., containment, shielding, heat dissipation, and nuclear-criticality safety) must provide a high degree of inherent safety under normal conditions and must have a high tolerance for malfunctions, off-normal conditions, and accidents if any should occur. Each shipping container is checked routinely to ensure that the "as-built" high quality is maintained throughout its lifetime.

Where the design includes mechanical systems necessary for satisfactory operation, a mechanical failure is assumed to be part of normal operating conditions, and therefore redundant mechanical systems are part of the design.

II.M.2 NUCLEAR-CRITICALITY SAFETY

Solid fissile material (i.e., U-233, U-235, and plutonium) in quantities that exceed 15 g/package require control in transport to ensure safety from accidental criticality. Nuclear-criticality safety in transport is provided by ensuring that the contents of each package of fissile material are subcritical when delivered to

a carrier for transport and that the package is so designed that it will remain subcritical under all conditions that may be encountered in transport, including accidents. In addition, either the contents must be limited or the package must be designed so that the number of packages likely to be accumulated in one vehicle or area will be subcritical under all conditions that may be encountered in transport, including accidents and handling errors.

The package design must be evaluated against the accident-damage test conditions discussed earlier for Type-B packages. The reactivity of the package must then be assessed on the basis of the assumption that the fissile material is in its most reactive configuration consistent with its chemical and physical form and on the assumption that water moderation and reflection exist.

A package for fissile material must be so designed and constructed and its contents so limited that the number of such packages shown in Table II.M-1 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. Table II.M-1 defines the various Fissile Classes.

The conditions for transport vary according to the Fissile Class. Packages that remain critically safe when stored in unlimited numbers can be assigned a Fissile Class I category providing at least 250 packages, damaged in accordance with the accident conditions specified in the regulations, remain subcritical when stored together.

Fissile Class II packages are limited by the carrier to an allowable number on a vehicle or in one handling or storage area. This control is achieved by the simple system of assigning to each package a number, called a transport index (TI), and requiring the carrier not to allow more than an accumulation of 50 TIs on a vehicle or area.

Fissile Class III shipments must be designated "exclusive use" (i.e., the consignor loads the shipment, the consignee unloads the shipment, and nothing is allowed on the vehicle other than the consignor's material) or must be accompanied by an escort provided by the shipper. Either the escort ensures that the shipment is kept separated from other fissile material, or some other procedure specifically approved by the DOT is followed.

II.M.3 PACKAGING-DESIGN REVIEW

The AEC now evaluates packaging designed to carry fissile, Type-B, and large-quantity materials and will, if the AEC finds that the packages meet the regulations, issue approvals to the person who requested the approval.⁸

Applicants for approval of a packaging design must provide a detailed analysis of that design to demonstrate that the design meets the packaging standards and criteria. The adequacy of the packaging design may be demonstrated by quantitative assessment, by tests of models, by extrapolation from test results for similar package designs or designs that employ similar construction features, by actual tests of components of the package, or by other evidence.

II.M.4 QUALITY ASSURANCE AND OPERATIONAL CONTROL

A package could inadvertently be constructed or used in a manner not in accordance with the design as approved. However, the likelihood of such errors is considered to be small in view of the regulatory requirements for quality assurance and for various observations and tests before each shipment.⁹⁻¹² For example, the regulations require that Type-B and large-quantity packages be tested prior to first use with respect to shielding and heat dissipation and be examined prior to each use for proper assembly, proper closing, temperature, pressure, and presence of neutron absorbers. Periodic checks of various functions are required after the package is put into service.

II.M.5 RADIATION-LEVEL LIMITATIONS

External radiation exposure of transport workers and of the general public in the transportation of packages of radioactive material is controlled during transport by several different methods.

The radiation emitted from individual packages of radioactive material is limited by the DOT regulations¹³ to no more than 200 millirems/hr on the surface (to limit the direct exposure to the person who handles the package) and to no more than 10 millirems/hr at a point 3 ft from the surface of the package (to limit the radiation level to which persons and property in the vicinity of the package would be exposed).

If a package is shipped in a closed truck or in a rail car under the "exclusive use" conditions, the radiation level at a point 3 ft from the surface of the package is limited to 1000 millirems/hr provided the radiation level does not exceed 200

millirems/hr at the surface of the vehicle, 10 millirems/hr at a distance 6 ft from the outside surfaces of the vehicle, and 2 millirems/hr in either the driver's compartment or other normally occupied positions in the truck or rail car. Most shipments in the fuel cycle are expected to be by the "exclusive-use" arrangement.

As a simple indicator of the radiation dose rate from an individual package, the regulations define one TI as being equal to 1 millirem/hr at 3 ft from the surface of the package. The regulations specify limits for aggregations of packages in terms of the sum of the TIs. 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 TIs does not exceed 50. This limitation prevents a large aggregation of packages, each with a significant radiation level, from producing a much higher total radiation level than is desirable because of the additive effect of the radiation levels from all of the packages.

Simple tables of minimum separation distances¹⁴⁻¹⁸ from people and from unexposed film are specified for packages of radioactive materials in storage and on vehicles in terms of the sum of the TIs in each group of packages.

The transport worker or carrier's agent is required to read each TI, add the total number of TIs, determine from the tables in the regulations the distance those packages must be kept from film and from continuously occupied areas, and ensure that those separation distances are maintained.

The TI system has also been adapted for limiting aggregations of packages that contain fissile radioactive materials to ensure nuclear-criticality safety (see Appendix II.M.2). The shipper determines, in accordance with specific criteria laid down in the AEC regulations, a TI figure that 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 TIs and, by complying with the limitation on the number of TIs in any one vehicle or location (50 TIs), limits the amount of fissile material in all types of packages to 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, several times the permitted total number of TIs in a collection of packages may be inadvertently placed together (see Table II.M-1).

Table II.M-1

MAXIMUM NUMBER OF PACKAGES OF FISSILE MATERIAL
FOR WHICH SUBCRITICALITY IS ENSURED

Fissile class	Conditions	
	Normal	Accident ^a
I	Any number	250 packages
II	Five times the allowable number ^b	Two times the allowable number ^b
III	Two times the allowable number ^b	The allowable number ^b

^aAll packages damaged in accordance with accident conditions specified in the regulations.

^bThe allowable number is the number of the same type of packages to be allowed in one shipment.

An important fact is that simultaneous consideration of nuclear-safety TIs with radiation-level TIs in the course of transport increases the margin of safety for both because they are not synergistic.

II.M.6 SURFACE-CONTAMINATION LEVELS

The DOT regulations¹⁹ also require the absence of significant removable surface radioactive 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 that the level is "not significant" if the radioactivity on the wipe does not exceed 10^{-11} Ci/cm² for beta-gamma emitters and 10^{-12} Ci/cm² for alpha emitters. Any fixed contamination of the surface is limited by the external-radiation-level limitations discussed in the previous paragraphs.

II.M.7 EXTERNAL TEMPERATURE

The DOT regulations²⁰ 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 as high as 180°F.

II.M.8 WARNING LABELS

Each package of radioactive material is required by DOT regulations^{21,22} to be labeled on two opposite sides with a distinctive warning label. Each of three label formats bears the unique trefoil radiation symbol. The label alerts persons who handle the package 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 for that package. However, if the background of the upper half

of the label is yellow, a radiation level that requires consideration may exist at the outside of the package, and an indication of what controls must be exercised for that package will be specified (e.g., the TI may be given). If the package bears a yellow label with three stripes, the rail or highway vehicle in which it is carried must be placarded.

II.M.9 PLACARDS

A truck or rail car that carries any package labeled with a "radioactive yellow-III" label must be placarded on the outside.^{23,24} The placard for rail cars bears the distinctive trefoil symbol, and for trucks, the placard reads "RADIOACTIVE" in letters large enough to catch the eye. The principal purposes of placards are to advise freight handlers of the presence of radioactive material with TIs inside the vehicle, to indicate the presence of special types of shipments (e.g., 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. This marking or placarding is intended to encourage persons not to remain unnecessarily long in the vicinity of the vehicle so as to reduce exposures that would otherwise result from loitering in the vicinity. Also, the placard will alert emergency crews to the need for taking appropriate precautions in case such vehicles are involved in accidents. Cars and trucks that carry carload or truckload lots of radioactive materials, packages with significant external radiation levels, packages that contain large quantities of radioactive material, or Fissile Class III shipments are required to be labeled with a "RADIOACTIVE" placard.

II.M.10 CAPACITY FOR COPING WITH ACCIDENTAL RELEASES

The consequences of an accident that involves radioactive material are mitigated by the procedures that carriers are required to follow.²⁵⁻²⁷ These procedures include: segregation of packages and materials from persons; immediate notification of the shipper and of DOT in case of an accident, a fire, or a leaking package; and a requirement that vehicles, cars, building areas, and equipment not be placed in service again until they have been surveyed and, if necessary, decontaminated.

Some reviewers* stated that emergency procedures to be used in case of accidents were not covered in the Draft Environmental Statement. Although this claim is incorrect, the following paragraphs and figure were added to present additional information on this important issue.

*Comment Letters 42, p. 19; 47, p. 1, enclosure 2.

There is a mechanism within some agencies of the Federal government that can provide immediate, competent, and experienced assistance to deal with radiological incidents such as those that might occur during the transportation of radioactive material.²⁸ The Atomic Energy Act of 1954 assigned general responsibility for protecting the health and safety of the public to the AEC. During the earlier stages of nuclear energy development and use, the AEC and Department of Defense (DOD) were the Federal agencies most deeply involved and, consequently, most knowledgeable in the safety ramifications of radiological accidents. The DOD-AEC Joint Agreement of February 27, 1958 formalized the mutual radiological assistance activities of the AEC. In June 1958, AEC formally initiated its Radiological Assistance Plan. This plan provides AEC with the specialized capability to deal with radiological incidents as needed and upon request. At about this time, the Federal Government recognized the expanding nonmilitary use of nuclear energy and the need to coordinate the growing radiological assistance capabilities of many other Federal agencies. This realization resulted in the Interagency Radiological Assistance Plan (IRAP), which assigns overall administrative responsibility for the plan to the AEC and defines generally the responsibilities of the other signatory agencies.

While the AEC continues to have a statutory responsibility to protect the public health when source, by-product, or special nuclear materials are involved, the aggressiveness of other Federal and state agencies in recognizing the need for adequate emergency response capability is adding to the overall radiological emergency capability to such an important extent that coordination is imperative. The increasing use of radiation-emitting materials and equipment not under AEC license or contractual control has required the U.S. Public Health Service and individual states to initiate comprehensive surveillance and control programs that provide a technical capability useful in planning for, assessing, and controlling radiological emergencies.

As stated in AEC Manual Chapter 0526, the agency's current policy is "to make available from its resources such radiological advice and assistance as may be appropriate to minimize injury to people and loss of property, to cope with radiological hazards and to protect the public health and safety, upon request from AEC contractors and licensees, Federal, state, and local agencies, private organizations, and persons cognizant of an incident believed to involve source, by-product, or special nuclear material." Practically interpreted, the AEC will provide assistance to anyone who requests it.

The eight AEC Regional Coordinating Offices and their geographical areas of responsibility are shown in Figure II.M-1. Each office has its own regional plan, and copies of the regional plan and its revisions are made available to signatory agencies and states in the region. The AEC radiological assistance function is solely advisory. The full responsibility for control of accidents actually rests with local and state authorities. The AEC's principal contribution is to provide the technical capabilities of personnel and instrumentation for assessing the significance and extent of an incident and for providing advice regarding the proper procedures to ensure that public exposure is minimized. The AEC also assumes responsibility and control if the occurrences involve its own radioactive materials; even in these cases, however, full cooperation is maintained with the states that are affected.

II.M.11 SHIPPER'S CERTIFICATION

Before delivering a package to a carrier for transport, the shipper must determine that no "significant" loose radioactive contamination is on the outside of the package, that the radiation levels on the surface of the package and at a distance of 3 ft from the package meet the specified regulatory levels, and that the marking and labeling comply with the requirements. The shipper also must certify^{29,30} in writing on the shipping papers that the radioactive materials are properly classified, described, packaged, marked, and labeled and are in proper conditions for transport according to the applicable DOT regulations.

U.S. ATOMIC ENERGY COMMISSION
REGIONAL COORDINATING OFFICES
 FOR
RADIOLOGICAL EMERGENCY ASSISTANCE
 AND THEIR
GEOGRAPHICAL AREAS
OF RESPONSIBILITY

REGION NO. and OPERATIONS OFFICE	POST OFFICE ADDRESS	TELEPHONE for ASSISTANCE	D D D AREA CODE
① NEW YORK	376 HUDSON STREET NEW YORK, NEW YORK 10014	989-1000	212
② OAK RIDGE	P. O. BOX E OAK RIDGE, TENNESSEE 37830	483-8611, Ext. 3-4510	615
③ SAVANNAH RIVER	P. O. BOX A AIKEN, S. C. 29801	N. AUGUSTA, S.C. 824-6331, Ext. 3333	803
④ ALBUQUERQUE	P. O. BOX 5400 ALBUQUERQUE, NEW MEXICO 87115	264-4667	505
⑤ CHICAGO	9800 S. CASS AVE. ARGONNE, ILLINOIS 60439	739-7711 Ext. 2111 duty hrs. Ext. 4451 off hrs.	312
⑥ IDAHO	P. O. BOX 2108 IDAHO FALLS, IDAHO 83401	526-0111 Ext. 1515	208
⑦ SAN FRANCISCO	2111 BANCROFT WAY BERKELEY, CALIFORNIA 94704	841-5121 Ext. 664 duty hrs. 841-9244 off hrs.	415
⑧ RICHLAND	P. O. BOX 550 RICHLAND, WASHINGTON 99352	942-1111 Ext. 6-5441	509

REGIONAL COORDINATING OFFICES FOR RADIOLOGICAL EMERGENCY ASSISTANCE

Figure II.M-1

II.M-1

REFERENCES FOR APPENDIX II.M

1. Directorate of Regulatory Standards, USAEC, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants," Report WASH-1238, pp. 61-74, December 1972.
2. 10 CFR 71.
3. 49 CFR 170-179 (for classification, packaging, labeling, and carriage by rail or truck).
4. 14 CFR 103 (for carriage by air).
5. 46 CFR 146 (for carriage by water).
6. 39 CFR 124-125.
7. 10 CFR 71.54 (Operating procedures, routine determination).
8. Fed. Regist. 38(62):8467 (1973).
9. 10 CFR 71.51-71.54 (conditions imposed by specific AEC licenses).
10. 49 CFR 171.
11. 49 CFR 173.
12. 49 CFR 178.
13. 49 CFR 173.393(i) and (j).
14. 49 CFR 177.842(b) (for motor vehicles).
15. 49 CFR 174.586(h)(2) (for railroad cars).
16. 49 CFR 175.655(j)(2) (for railroad cars).
17. 14 CFR 103.23(c) (for aircraft).
18. 46 CFR 146.19-35 (for ships).
19. 49 CFR 173.393(h).
20. 49 CFR 173.393(e).
21. 49 CFR 173.399.
22. 49 CFR 173.402(a)(8).
23. 49 CFR 174.54.
24. 49 CFR 177.823.
25. 49 CFR 171.15.
26. 49 CFR 174.566.

27. 49 CFR 177.861.
28. W. T. Thornton, "What Good is the AEC in a Radiation Emergency," pp. 196-208 in Conference on Transportation of Radioactive Materials held in Charlottesville, Va., Oct. 26-27, 1970, PB-197-566.
29. 49 CFR 173.427.
30. 49 CFR 173.430.

DETAILED ASSUMPTIONS AND CALCULATIONS FOR TRANSPORTATION ACCIDENTSII.N.1 PACKAGES OF UF₆

Depleted uranium for preparation of UO₂ and mixed-oxide pellets for LMFBR cores will be shipped in steel cylinders to the fuel-fabrication plants as solid UF₆. By current practice, such cylinders are shipped without a protective shipping package (PSP). However, PSPs are required for cylinders as large as the 48-Y type containing recycled uranium returned from the reprocessing plant because of the curies of radioactive material in these cylinders. Single cylinders of the 48-Y types,¹ having a nominal capacity of 14 tons (max. net 27,560 lb) and a tare weight of 5200 lb, are shipped by truck, but railroad cars have also been equipped to accommodate four of these cylinders on one car.

The characteristics of UF₆ are such that exposure to ordinary atmospheric conditions following an accident in which a hole is made in the steel cylinder will not result in the release of an appreciable amount of UF₆. The UF₆ is a nonflammable solid that slowly reacts with atmospheric moisture to form UO₂F₂ and HF.

The results² of small-scale experiments in which cylinders were submerged in water show that small holes that penetrate into solid UF₆ tend to plug with an insoluble mixture of UO₂F₂, H₂O and corrosion products from the metallic container. Where a larger interface between the water and the UF₆ is available, an insoluble product does not form.

The amount and rate of UF₆ released as a result of an accident is dependent on three principal variables: (1) the type of carrier, truck or rail, that determines whether one or four cylinders are involved; (2) presence or absence of damage to a cylinder resulting in a hole; and (3) the accident severity category (i.e., the impact velocity of the vehicle and duration of fire should one occur).

A cylinder carried by a truck involved in an accident of "minor" through "severe" classification (see Table 4.5-14) will be exposed to a fire of very short duration (less than one minute) and, even if the cylinder is punctured, there will be very little UF₆ released. A "severe" rail accident, however, would expose the rail car carrying four cylinders to a severe impact followed by a fire of an average duration of 0.47 hr. Accidents in the "extra severe" and "extreme" categories result in fires of essentially equal duration whether involving a truck or railroad car and the durations range from 1.7 to 2 hr. Obviously, fire will not occur in every

accident, but the five categories of accidents used here reflect an averaging of accident experience and include accidents in which fires did occur along with those in which they did not (see Section 4.5.5 for a discussion of accident categories).

If the cylinder is punctured in the gas space above the solid, some release of the UF_6 vapor will occur immediately, because solid UF_6 has a vapor pressure of 0.2 atm at 70°F. But the rate of release does not become substantial until the solid reaches a temperature of about 134°F where its vapor pressure is 1 atm. Subsequent release will be controlled by the rate at which heat is supplied to the UF_6 . A puncture at the bottom of a cylinder might be isolated from the gas phase, thus allowing the temperature of the UF_6 to rise to its melting point and permitting liquid UF_6 to be drained out of the hole. This sequence could result in a more rapid release of UF_6 vapor. The most rapid rate of release will follow the rupture of a cylinder due to either gaseous or hydrostatic pressure caused by heating in a fire.

By virtue of the large amount of UF_6 in a type 48-Y shipping cylinder and especially in transfer by rail, the potential exists for the release of a very large amount of UF_6 in an accident. The available statistics on accident frequency for carriers show that the chance of the occurrence of any of the accidents is comparatively small.

The probability (P) of an accident occurring in any one year to the UF_6 shipped to supply the fuel for one LMFBR is (see Appendix II.A):

Category 3 (severe)

$$P = 9.6 \times 10^{-6} \text{ accidents/reactor year}$$

or

$$1/P = 1.04 \times 10^5 \text{ reactor years of operation/accident}$$

Category 4 (extra severe)

$$P = 2.4 \times 10^{-8} \text{ accidents/reactor year}$$

or

$$1/P = 4.2 \times 10^7 \text{ reactor years of operation/accident}$$

Category 5 (extreme)

$$P = 1.2 \times 10^{-10} \text{ accidents/reactor year}$$

or

$$1/P = 8.3 \times 10^9 \text{ reactor years of operation/accident}$$

The following properties of UF_6 ¹ were used in calculating the UF_6 releases: (1) sublimation temperature (at 14.7 psia), 134°F; (2) heat of sublimation (147°F), 58.2 Btu/lb; (3) heat of fusion (147°F), 23.5 Btu/lb; heat of vaporization (147°F), 35.1 Btu/lb; and (4) specific heat² of solid (81°F), 0.114 Btu/lb/°F. In the

absence of a measured value, an emissivity of 0.5 was assumed for UF_6 . The specific heat of the steel cylinder is $\sim 0.12 \text{ Btu/lb/}^\circ\text{F}$.³

The cylinders measure 4 ft in diameter and are 12.5 ft long; the ambient temperature in the following cases is 70°F .

Three accidents have been examined that show the effect of cylinder damage or its absence, of fire, and of water submersion on the release of UF_6 . The resulting source terms are summarized in Table II.N-1.

An accident that resulted in both the puncture of the cylinder in the region of solid UF_6 and a fire was not analyzed because of difficulty in modeling the changing geometry and heat transfer; however, the total release is expected to be of equal or larger magnitude to that for a puncture in the void space, and it is also expected to occur more quickly.

II.N.1.1 Case 1

A truck carrying a 14-ton cylinder is involved in a wreck of Category 4 (extra severe). A hole measure 6-in. diameter is punctured in the cylinder above the solid. The resulting fire has a temperature of 1475°F , an emissivity $E_F = 0.9$, and a duration of 1.7 hr. The UF_6 will not melt under these conditions but will sublime directly to form a gas that will then flow out of the tank. Solid UF_6 is a very poor conductor of heat so its surface will quickly heat to the range of $134\text{--}147^\circ\text{F}$ where its vapor pressure varies from 15 to 22 psia. Heat will be transferred from the inside of the cylinder to the solid UF_6 by conduction, convection, and radiation. Conduction and convection mechanisms will control the heat initially, but the cylinder will rapidly rise in temperature as the peripheral UF_6 evaporates, forming a gap, after which radiation is taken as the controlling mechanism.

Table II.N-1
SUMMARY OF SOURCE-TERM DATA FOR UF₆ CYLINDER SHIPMENTS^a

Case	Nature of accident	Accident fire ^b		Material released (kg)			Release duration (hr)	Activity (Ci) and source ^e			
		Category ^c	Duration (hr)	UF ₆	UO ₂ F ₂ ^d	HF ^d		U-234	U-235	U-236	U-238
1	Collision and fire; hole in cylinder above UF ₆	4	1.7	10,022	8,769	2,278	1.7	1.1 E-1	1.4 E-2	5.7 E-3	1.0 E0
2	Collision and fire; cylinder intact ^f	4	1.7	1,253 11,275	1,096 9,865	285 2,562	1 min ^h 4	1.4 E-2 1.3 E-1	1.7 E-3 1.6 E-2	7.1 E-4 6.4 E-3	1.3 E-1 1.2 E0
2	Collision and fire; cylinder intact ^g	4	1.7	10,022 2,505	8,769 2,192	2,278 569	1 min ^h 4	1.1 E-1 2.8 E-2	1.4 E-2 3.4 E-3	5.7 E-3 1.4 E-3	1.0 E0 2.5 E-1
3	Collision and submersion in water; hole in cylinder above UF ₆	2	None	12,527	10,961	2,847	2.8	1.4 E-1	1.7 E-2	7.1 E-3	1.3 E0

^aDepleted UF₆ is shipped in type 48-Y steel cylinders that contain 27,560 lb (maximum) of UF₆. Single cylinders are shipped on a truck, and as many as four cylinders are shipped on a railroad car.

^bIntensity of fire is equal to 200 MW.

^cSee Table 4.5.5.1 for description of accident categories.

^dA kilogram of UO₂F₂ or of HF is the equivalent of the released UF₆, upon hydrolysis.

^ePercentage composition of depleted uranium: U-234, 0.000059; U-235, 0.2075; U-236, 0.0029; and U-238, 99.789.

^fEarly failure of cylinder with 10% of the UF₆ being molten.

^gDelayed failure of cylinder with 80% of the UF₆ being molten.

^hRelease from the cylinder at failure is instantaneous, but the resulting cloud of UF₆ is treated as though it were formed doing a 1-min release.

$$\text{Flux 1} = (0.173) (A_{\text{cyl}}) (F_{\text{cyl-fire}}) \left[\left(\frac{T_F + 460}{100} \right)^4 - \left(\frac{T_{s1} + 460}{100} \right)^4 \right]$$

$$\text{Flux 2} = (0.173) (A_{\text{hex}}) (F_{\text{hex-cyl}}) \left[\left(\frac{T_s + 460}{100} \right)^4 - \left(\frac{T_{\text{hex}} + 460}{100} \right)^4 \right]$$

If convective transfer is discounted, the cylinder will rise in temperature until Flux 1 = Flux 2.

$A_{\text{hex}} = (\frac{\pi}{2}) [D_{\text{hex}}^2 + 25 D_{\text{hex}}]$ is time dependent because it varies as UF_6 sublimes.

The terms that account for the varying area of the UF_6 and the emissivities of the cylinder and the UF_6 are given in Table II.N-2. The time dependence of UF_6 release is given in Table II.N-3.

Initial delay to
heat steel cylinder

$$\text{from } 70 \text{ to } 1390^\circ\text{F} \approx \frac{(1390 - 70)^\circ\text{F} (0.12) \text{Btu/lb}^\circ\text{F} (0.28) \text{lb/in}^3 (144) \text{in}^2/\text{ft}^2 (0.625) \text{in}}{\text{avg Flux 1 during heat up} \approx 2/3 (372 + 97) \text{Btu/ft}^2/\text{min}}$$

$$\approx 13 \text{ min}$$

The above analysis shows that the rate of UF_6 release will increase exponentially and peak as the cylinder reaches a temperature of 1300 to 1400°F. About 80% or 22,048 lb of UF_6 will be released in this accident. However, since an estimate of the emissivity for UF_6 was used, the calculated values may not be accurate. The UF_6 released will react with atmospheric moisture to form UO_2F_2 and HF. Basis for analysis:

Material: UF_6 , UO_2F_2 , and HF

Total Amount: $\text{UF}_6 = 22,048 \text{ lb (10022 kg)}$

UO_2F_2 equivalent of $\text{UF}_6 = 19292 \text{ lb (8769 kg)}$

HF equivalent of $\text{UF}_6 = 5010 \text{ lb (2278 kg)}$

Concentration: Released in the plume of a 200 MW fire during a period of 1.7 hr

Physical Properties: Soluble UF_6 , UO_2F_2 , and HF. UO_2F_2 powder ranging in size from submicron to about 10 μm diam.

Isotopic Content:

U-234 1.1 E-1 Ci

U-236 5.7 E-3 Ci

U-235 1.4 E-2 Ci

U-238 1.0 E 0 Ci

Table II.N-2

TABULATION OF TERMS WHICH ACCOUNT FOR VARYING AREA
OF UF₆ AND EMISSIVITIES OF STEEL AND UF₆

Fraction of UF ₆ sublimed	Dependence on diameter	D (ft) for given interval			$A_{\text{hex}}(\text{avg}) = \frac{\pi}{2}(D^2 + 25D)$	$\frac{1}{A_{\text{hex}} F_{\text{hex-cyl}}}$	$A_{\text{hex}} F_{\text{hex-cyl}}$	$\frac{A_{\text{hex}} F_{\text{hex-cyl}}}{A_{\text{cyl}}}$
		Beginning	End	Average				
0.25	$(D_{100}^2 - D_{75}^2)/D_{100}^2 = 0.25$	2.97	2.57	2.77	111	0.0194	51.6	0.284
0.50	$(D_{100}^2 - D_{50}^2)/D_{100}^2 = 0.50$	2.57	2.10	2.34	101	0.0212	47.2	0.259
0.75	$(D_{100}^2 - D_{25}^2)/D_{100}^2 = 0.75$	2.10	1.49	1.79	75	0.0280	35.7	0.196
0.85	$(D_{100}^2 - D_{15}^2)/D_{100}^2 = 0.85$	1.49	1.15	1.32	55	0.0377	26.5	0.146
0.90	$(D_{100}^2 - D_{10}^2)/D_{100}^2 = 0.90$	1.15	0.94	1.04	43	0.0479	20.9	0.115
0.95	$(D_{100}^2 - D_5^2)/D_{100}^2 = 0.95$	0.94	0.66	0.80	32	0.0639	15.7	0.086

Table II.N-3

TABULATION OF UF₆ LOSS AND ITS TIME DEPENDENCE

Fraction for period	UF ₆ loss		Average A_{hex} during loss period (ft ²)	Y_{avg}^a	Y_{avg}	T_s (°F)	Flux 1 = flux 2		UF ₆ sublimed (lb/min)	Lapsed time (min)	
	Running total (fraction)	(lb)					(millions Btu/hr/cyl)	(Btu/min/cyl)		For fractional loss	Running total ^b
0.25	0.25	6,890	111	101,120	17.83	1323	0.930	15,505	266	26	39
0.25	0.50	13,780	101	103,663	17.94	1334	0.830	14,496	249	28	67
0.25	0.75	20,670	75	110,678	18.24	1364	0.703	11,712	201	34	101
0.10	0.85	23,430	55	116,959	18.49	1359	0.553	9220	158	17	118
0.05	0.90	24,800	43	121,225	18.66	1406	0.452	7527	129	11	129
0.05	0.95	26,180	32	125,507	18.82	1422	0.350	5828	100	14	143

$$aY = \frac{T_s + 460}{100}$$

^bIncluding initial heatup period for cylinder.

II.N.1.2 Case 2

A truck carrying a 14-ton cylinder is involved in an accident resulting in a fire, but the cylinder remains intact. The fire has a temperature of 1475°F and an emissivity of 0.9. The effect of the fire on the container and its contents depends on the sequence of changes that the UF_6 will experience.

As heat is added to UF_6 in a closed container, its temperature first rises to 147°F, the point at which it melts and its vapor pressure is 22 psia. If heat is added slowly, as in a steam chest, the UF_6 solid and liquid will remain at 147°F until all of the solid melts, after which both the temperature of the liquid and its vapor pressure will again continue to rise. Uranium hexafluoride (UF_6) shows a large decrease in density on melting. The solid weighs 318 lb/ft³, but the liquid weighs only 228 lb/ft³ at the melting temperature. Thus, most of the void space in the cylinder becomes filled with liquid UF_6 at the melting point. The liquid continues to expand on heating until it reaches about 280°F when it will completely fill the container. The vapor pressure of UF_6 at this temperature is about 150 psia, well below the design pressure of a type 48-Y cylinder, but an additional increase in temperature will cause a very rapid rise in the system pressure because the liquid UF_6 has no more space into which it can expand. The cylinder would therefore fail as a result of hydraulic pressure.

Solid UF_6 is a poor conductor of heat [3.19 Btu/hr/ft²(°F/in) at (120°F)]. The conductivity of UF_6 , therefore, is similar to firebrick [2.90 Btu/hr/ft²(°F/in)] and dry sand [2.70 Btu/hr/ft²(°F/in)].⁴ In the subject accident, the effect of this poor thermal conductivity, which offers a high resistance to the flow of heat from the fire into the UF_6 mass, is to cause the cylinder and outer layer of liquid to rise to a much higher temperature than 147°F. The heavier solid would settle and force the molten UF_6 upward to form a pool covering the surface of the solid. The vapor pressure in the empty volume of the cylinder will depend on how hot the surface of the UF_6 liquid becomes.

Sufficient information is not available at present to predict the detailed behavior of the UF_6 in a bare container of the 48-Y type on exposure to a fire. However, a series of fire tests on small UF_6 -filled cylinders was done in 1965 at the Oak Ridge Gaseous Diffusion Plant.⁵ In these tests, the cylinders were exposed to flames from a pool of burning diesel fuel which provided a temperature of about 1500°F. The largest cylinder tested was made of nickel, contained 245 lb of UF_6 , and measured 8-in ID x 3/16-in thick x 48-in long (volume = 1.27 ft³). The cylinder failed explosively 8.5 min after the fire was started. The temperature profiles during

the test at several locations inside the cylinder and at the exterior of the wall on the top of the cylinder were obtained with thermocouples.

An unequivocal interpretation of the behavior of the UF_6 in the test cylinder cannot be made, but it is known from the temperature density characteristics of UF_6 that 245 lb of liquid UF_6 at 280-290°F would completely fill the container. A portion of the charge, therefore, had to remain cooler than the minimum recorded temperature of 320°F, supporting the possibility that part of the charge was solid at <150°F while surrounded by liquid UF_6 of substantially higher temperature. Another key temperature in the test was the 430°F registered by a thermocouple originally in the void region above the UF_6 . The critical temperature for the UF_6 is 446°F and the critical pressure is 669 psia. These values mean that the vapor rapidly acquired the noncompressible characteristics of a liquid immediately before failure. The melting of a relatively small additional increment of UF_6 would cause a rapid increase in system pressure once the critical temperature is reached causing failure of the cylinder.

The cylinder wall temperature over the void volume of the cylinder increased in a straight line for 8.5 min to 1025°F, at which time the cylinder ruptured. The strength of the nickel at the hottest part of the cylinder should be that of the annealed metal and thus have a yield strength of about 20,000 psi. If the hoop stress in the cylinder is limited by the yield strength then the pressure P at failure can be calculated.

$$\text{Yield strength} = 20,000 \text{ (psi)} = P \text{ (psi)} \frac{D \text{ (diameter-in)}}{2 T \text{ (thickness-in)}} = 21.3 P$$
$$P = \frac{20,000}{21.3} = 939 \text{ psi}$$

As a first step in using the behavior of the 8-in ID nickel cylinder to predict that of a 4-ft diam type 48-Y cylinder, the pressure to cause failure was calculated. Using the yield stress for annealed steel as the limiting fiber stress

$$\text{Yield strength} = 30,000 \text{ psi} = P \frac{(4)(12)}{(2)(5/8)} = 38.4 P$$
$$P = 30,000/38.4 = 781 \text{ psi}$$

Thus, the type 48-Y cylinder will fail at about the same pressure as the 8-in nickel cylinder.

An estimate of the time lapse to failure can be made by assuming that the rate of heat transfer into the two cylinders is the same per unit of surface area. Further, the amount of UF_6 that must be evaporated into the void volume to reach a given pressure will vary inversely as the size of the void volume. Since

$$\text{Area} = A_{\text{cyl}} = \pi\left(\frac{D^2}{2} + LD\right)$$

and

$$\text{Void} = V_v = (\text{void fraction})(\pi/4)(D^2)(L)$$

where

D = diameter

L = length

then

$$\text{Time to reach pressure, } P \propto \frac{V_v}{A_{\text{cyl}}} = \frac{(\text{void fraction})(D)(L)}{2(D + 2L)}$$

For 8-in ID x 48-in long cylinder:

$$\text{Time to reach pressure, } P \propto \frac{(0.39)(8)(48)}{(2)[8 + (2)(48)]} = 0.72$$

For 4-ft diam x 12.5-ft long cylinder:

$$\text{Time to reach pressure, } P \propto \frac{(0.39)(4)(12)(12.5)(12)}{(2)[(4)(12) + (2)(12.5)(12)]} = 4.03$$

By prorating, time to failure for 48-Y cylinder = $(4.03/0.72)(8.5) = 48$ min.

Within the uncertainty of the assumptions made, a bare type 48-Y cylinder might survive a Category 3 accident that resulted in a fire of 0.4-hr duration. It is virtually certain, however, that the cylinders would fail explosively in fires of 1.7-hr and 2.0-hr duration associated, respectively, with Category 4 and Category 5 accidents.

Release of UF_6 from this accident will be characterized by a puff at the time of cylinder failure followed by a diminishing rate of release with time. The longer the delay to failure the larger the puff. If the resistance to flow of heat into the cylinder and its contents is high, large temperature gradients will develop and failure will occur due to the high vapor pressure of the UF_6 before much of the UF_6 melts. This situation would give a relatively small initial puff. If good heat transfer develops due to convective movement of the liquid UF_6 within the cylinder, however, the entire contents might melt before failure occurs. In either event,

the contents of the cylinder will be expelled explosively causing any residual solid that may exist to break up and thus generate a large surface area for reaction with available moisture.

Two source terms are given below; the first represents early failure, the second represents a delayed failure.

Basis for analysis (early failure):

Material: UF_6 , UO_2F_2 , and HF

Total Amount: UF_6 = 27,560 lb (12,527 kg)

UO_2F_2 equivalent of UF_6 = 24,115 lb (10,961 kg)

HF equivalent of UF_6 = 6,263 lb (2847 kg)

Concentration: Instantaneous release of 10% or 2756 lb (1253 kg) of gaseous UF_6 equivalent to 2411 lb (1096 kg) of UO_2F_2 and 626 lb (285 kg) of HF. This material is released in the plume of a 200 MW fire. The remaining 24,804 lb (11,275 kg) equivalent to 21,704 lb (9865 kg) of UO_2F_2 and 5637 lb (2562 kg) of HF is broken up and thrown clear of the fire and will hydrolyze over a period of about 4 hr. The result could be a release of gaseous HF at ground level, leaving the resulting UO_2F_2 on the ground at the accident site.

Physical Properties: Soluble UF_6 , UO_2F_2 , and HF. Airborne UF_6 powder ranges in size from sub micron to about 10 μm .

Isotopic Content:

Airborne

U-234 1.4 E-2 Ci

U-236 7.1 E-4 Ci

U-235 1.7 E-3 Ci

U-238 1.3 E-1 Ci

On ground at accident site

U-234 1.3 E-1 Ci

U-236 6.4 E-3 Ci

U-235 1.6 E-2 Ci

U-238 1.2 E 0 Ci

Basis for analysis (delayed release):

Material, total amount, and physical properties are the same as for early failure.

Concentration: Instantaneous release of 80% or 22,048 lb (10,022 kg) of gaseous UF_6 equivalent to 9292 lb (8769 kg) of UO_2F_2 and 5011 lb (2278 kg) of HF in the plume of a 200 MW fire. The remaining 5512 lb (2505 kg) equivalent to 4823 lb (2192 kg) of UO_2F_2 and 1253 lb (569 kg) of HF is broken up and thrown

clear of the fire and will hydrolyze over a period of about 4 hr. This result could be release of gaseous HF at ground level, leaving the resulting UO_2F_2 on the ground at the accident site.

Isotopic Content:

Airborne

U-234 1.1 E-1 Ci

U-236 5.7 E-3 Ci

U-235 1.4 E-2 Ci

U-238 1.0 E 0 Ci

On ground at accident site

U-234 2.8 E-2 Ci

U-236 1.4 E-3 Ci

U-235 3.4 E-3 Ci

U-238 2.5 E-1 Ci

If the carrier in Case 2 were a railroad car rather than a truck, up to four cylinders of the 48-Y type could be involved. If each cylinder experienced the sequence described above, the result would lead to source terms larger by a factor of 4.

II.N.1.3 Case 3

A truck carrying a single type 48-Y cylinder skids off a roadway which parallels a river. The trailer rolls down a steep embankment into the river and stops in a wheels-up position. During the roll a hole is knocked in the gas volume of the cylinder. There is a delay of 24 hr before a crane reaches the accident site and lifts the trailer and cylinder from the river.

The water will react on contacting the UF_6 to form UO_2F_2 and HF, both of which are soluble. There is at present insufficient experimental information available to predict with certainty the rate at which the resulting solution will escape from the cylinder. Small-scale experiments¹ show, however, that UF_6 will rapidly dissolve and flow out of the opening of a small cylinder inverted in water. By using the solution concentration found for one of the laboratory-scale experiments, calculations showed that all of the UF_6 would be released from a cylinder in about 3 hr. In actuality, the release rate will vary over a wide range depending on the amount of cylinder damage and hole position. The rate of release could be larger than found above but is more likely to be smaller.

Basis for analysis:

Material: Aqueous solution of UO_2F_2 and HF:

Total Amount: $\text{UF}_6 = 27,560 \text{ lb (12,527 kg)}$

UO_2F_2 equivalent of $\text{UF}_6 = 24,115 \text{ lb (10,961 kg)}$

HF equivalent of $UF_6 = 6264 \text{ lb (2847 kg)}$

Concentration = total amount dissolved in 1366 ft^3 of aqueous solution in 2.8 hr

Isotopic Content

U-234 1.4 E-1 Ci

U-236 7.1 E-3 Ci

U-235 1.7 E-2 Ci

U-238 1.3 E 0 Ci

II.N.1.4 Protective Shipping Packages (PSP)

Current Federal Regulations⁶ for low specific-activity materials do not require a PSP for packages that are transported on vehicles assigned for the sole use of the consignor. The cylinders used to transport the depleted UF_6 from a separation plant to the fuel-fabrication plant, therefore, will be bare whether a truck or railroad carrier is used. The adverse effects of involvement in any of the accidents described above can be reduced to a very low level by use of a PSP, but will of course, require additional equipment expenditures and shipping costs. Tests of PSPs in simulated accidents have proved their value in minimizing damage due to impact and/or fire.⁷

II.N.2 PACKAGES OF UO_2

Packages, which have a phenolic foamed insulating material, used to ship UO_2 have been extensively tested and have been shown to meet all requirements of the hypothetical-accident conditions specified in the regulations. The assumption for accident-severity Category 5 is that only the lid and top insulation of one package in the shipment are lost, thus exposing the inner bolted lid to the radiant source of energy.

The following information was used in the calculation: weight of steel in inner container, 115 lb; and surface area of top flange of inner container, 0.92 ft^2 .

By assuming that the average heat input to the package is only $10,000 \text{ Btu}/(\text{hr-ft}^2)$, or equivalent to a lid temperature of 1050°F , the time required to increase the temperature of the inner container from operating temperature to the effective fire temperature of 1475°F (if only the top flange is exposed to the flame and if no heat is transferred through the insulation that surrounds the inner container) is given by

$$\text{Time} = \frac{(115 \text{ lb})[0.11 \text{ Btu}/(\text{lb-}^\circ\text{F})](1475^\circ\text{F} - 100^\circ\text{F})}{[10,000 \text{ Btu}/(\text{hr-ft}^2)](0.92 \text{ ft}^2)} = 1.89 \text{ hr} .$$

It is assumed that the gas expands and leaks out through the destroyed seal carrying with it particulates that exist in the gas space above the powder. The release estimated was based on the concentration of UO_2 in the gas space above the powder equivalent to 10 mg of uranium per cubic meter. By using the perfect gas law, the increase in pressure if the seal did not leak is given by

$$P_2 = P_1 \frac{T_2}{T_1} = 1 \text{ atm} \frac{1475 + 460}{560} = 3.45 \text{ atm abs.}$$

If the seal did leak after the 2-hr fire, it would release $3.45 - 1 = 2.45$ times the volume of free-gas space that exists above UO_2 .

Total volume of inner cylinder =	105,000 cm ³
-Total of metal in inner drums =	-5,000 cm ³
-Volume of 110 kg of UO_2 =	-10,500 cm ³
Total	89,500 cm ³ = 0.089 m ³ free-gas space volume of the package.

The loss of UO_2 from the package is therefore estimated to be

$$(0.089 \text{ m}^3)(2.45)(10 \text{ mg/m}^3) = 2.1 \text{ mg}$$

and the release of radioactivity is estimated to be

$$(2.1 \times 10^{-3} \text{ g})(6.26 \times 10^{-7} \text{ Ci/g}) = 1.32 \times 10^{-9} \text{ Ci of uranium.}$$

II.N.3 PACKAGES OF PuO_2

The assumption in the Category-5 accident is that the outer drum lid is lost, thus releasing the insulating cover, as in the case of the UO_2 package. The PuO_2 was assumed to be packaged in special-form containers and placed inside the inner package. To determine whether the temperature of the inner steel vessel reaches the fire temperature of 1475°F in the 2-hr fire, the following information was used:

Weight of steel in the inner steel vessel =	56.24 lb
Weight of steel in special form containers =	24.1 lb
	Total weight of steel = 80.34 lb
Area of top flange =	0.6 ft ²

The time required to heat all the steel to 1475°F is

$$\text{Time} = \frac{(80.34 \text{ lb})[0.11 \text{ Btu}/(\text{lb}\cdot^\circ\text{F})]}{[10,000 \text{ Btu}/(\text{hr}\cdot\text{ft}^2)](0.6 \text{ ft}^2)} = 2.02 \text{ hr}.$$

Therefore, the temperature of the inner PuO_2 cans will reach 1475°F in the 2-hr fire.

The special-form cans held in the inner steel vessel must be capable of surviving a 1475°F temperature for ten minutes to meet regulatory requirements. By assuming that the cans could meet the requirements for ten minutes they could remain at that temperature indefinitely, because the resulting maximum internal pressures of about 37 psig would remain constant and the cans would be capable of sustaining that internal pressure at that temperature. Therefore, even if exposed to a Category-5 accident, the containers would not rupture, and no PuO_2 would be lost.

II.N.4 ALPHA WASTE SHIPMENTS

Alpha-contaminated waste was assumed to be packaged in 55-gal drums and 4-ft by 4-ft by 6-ft boxes that would be placed in an 8-ft by 8-ft by 20-ft long cargo container. Two cargo containers would be placed in a rail car for shipment to a burial ground. The assumption made was that, after reduction of this combustible material to a noncombustible form, there might be as much as 0.1% of the material still capable of producing some gases should the package reach a high enough temperature (about 400°F).

Only the most severe impact and longest fire is assumed to be capable of providing a mechanism that could reduce some of the activity. An accident in Category-5, which would most likely involve a derailment and overturn of the car, was considered. Because of the strength of the vehicle underframe and internal bracing, the accident would probably create only rips or holes in the rail car, and the cargo containers would not be lost. If the vehicle were immersed in a 2-hr fire, some radioactivity could be released.

Rail car weight (60 tons)	120,000 lb
Cargo weight	<u>50,000 lb</u>
Total weight	170,000 lb

The rail car would be about 50 ft long and 10 ft on a side. If three sides are exposed to flame, the time for the waste to reach 400°F can be estimated.

$$\begin{aligned}\text{Time to reach } 400^{\circ}\text{F} &= \frac{(170,000 \text{ lb})[0.15 \text{ Btu}/(\text{lb}\cdot^{\circ}\text{F})](400^{\circ}\text{F} - 100^{\circ}\text{F})}{[17,000 \text{ Btu}/(\text{hr}\cdot\text{ft}^2)(3)(50 \text{ ft})(10 \text{ ft})]} \\ &= 0.3 \text{ hr}.\end{aligned}$$

The number of air gaps plus wooden insulation built into the car would probably delay the attainment of this temperature for over 1 hr. Any remaining combustible waste would be expected to undergo pyrolysis, releasing gases such as CO, CO₂ and water vapor. Some particulate matter might be entrained in these gas streams and would have to pass from the drum, through the cargo container, and through cracks and holes in the rail car. The velocity of gas passing out through these various gaps, due to the gas expansion by heat and gases generated by the waste, would probably be relatively low.

If 0.1% of the waste were combustible, the total weight loss from the car might be as much as $2 \times 10^{-4}\%$. By assuming conservatively that this release would carry a proportionate quantity of plutonium with it, the loss is estimated to be

$$(2 \times 10^{-6})(2500 \text{ g of Pu/car}) = 5 \times 10^{-3} \text{ g}$$

and the release of radioactivity is

$$(5 \times 10^{-3} \text{ gm})(11.6 \text{ Ci/gm}) = 0.058 \text{ Ci*}.$$

II.N.5 LOW-LEVEL BETA-GAMMA WASTE

Low-level beta-gamma waste would be transported in 55-gal drums and would contain waste that is slightly contaminated with fission products. The shipments would be by truck. This waste material is about 50% combustible and will burn at about 400°F.

The trailer is assumed to be 40 ft long and 8 ft on each side. Heat input to the trailer and contents will occur on three sides of the trailer.

Trailer weight = 15,000 lb

Cargo weight = 17,700 lb

Total weight = 32,700 lb

*97.6% Pu-241 (β emitter).

$$\text{Time} = \frac{(32,700 \text{ lb})[0.15 \text{ Btu}/(\text{lb}\cdot^{\circ}\text{F})](400^{\circ}\text{F} - 100^{\circ}\text{F})}{[17,000 \text{ Btu}/(\text{hr}\cdot\text{ft}^2)](3)(40 \text{ ft})(8 \text{ ft})}$$

$$= 0.09 \text{ hr}.$$

This result indicates that radioactivity might be released in accidents of Categories 3, 4, and 5 (see Table 4.5.5.1). The drums are assumed to stay with the vehicle in Category-3 and -4 accidents where 2% of the activity on the combustible waste might be released with ash and smoke. In Category-5 accidents, 25% of the containers are assumed to be thrown free of the vehicle and ruptured. In this case, 4% of the activity on the combustible waste might become airborne.

Categories-3 and -4

2% loss involving 50% combustible waste = 1% release.

Category-5

1% release of 3/4 drums plus 2% release of 1/4 drums = $0.75 + 0.5 = 1.25\%$ release.

By assuming each shipment contains 0.475 Ci of activity that is predominantly Sr-90, the release of radioactivity is:

<u>Category</u>	<u>Activity released, Ci</u>
3 and 4	$(0.475 \text{ Ci})(0.01) = 0.00475$
5	$(0.475 \text{ Ci})(0.0125) = 0.0059$

II.N.6 TRITIUM

Tritium is assumed to be converted to calcium hydroxide, $\text{Ca}(\text{OH})_2$, which would have a volume of $640 \text{ ft}^3/(10^6 \text{ Ci})$. The material is assumed to be packaged in steel cans that would, in turn, be placed in an insulated shipping container. Because physical testing of this type of package has shown that it will withstand more than the hypothetical-accident conditions, it is assumed that Category-1 through -4 accidents will not damage the package beyond its capability to maintain containment. A Category-5 accident, however, might be sufficiently severe to throw 25% of the packages free of the transporting vehicle. If 25% (i.e., four drums) of those packages released lose their lids and top insulation, exposure to fire might release the tritium.

The time for the inner container to reach 1076°F can be calculated if the top flange is exposed to a radiant heat source whose heat flux averages about $14,000 \text{ Btu}/(\text{hr}\cdot\text{ft}^2)$, corresponding to a top flange temperature of about 800°F . The inner

steel vessel of the insulated package weighs about 80 lb, and the flange has an area of 0.6 ft².

$$\begin{aligned} \text{Time} &= \frac{(80 \text{ lb})[0.11 \text{ Btu/(lb-°F)}](1076^\circ\text{F} - 100^\circ\text{F})}{[14,000 \text{ Btu/(hr-ft}^2\text{)}](0.6 \text{ ft}^2)} \\ &= 1.03 \text{ hr} . \end{aligned}$$

Under these conditions, the calcium hydroxide will reach a temperature of 1076°F in less than 2 hr and will therefore, lose the tritium in the form of water vapor. Four drums have a total capacity of 1.28 ft³ of Ca(OH)₂. The activity released is then

$$\frac{1.28 \text{ ft}^3}{640 \text{ ft}^3/10^6 \text{ Ci}} = 2000 \text{ Ci} .$$

II.N.7 ALPHA-BETA-GAMMA SOLID WASTE

Alpha-beta-gamma waste from a reprocessing plant would be volume-reduced, packed in disposable steel cylinders, and shipped in a shielded cask. All material would be handled remotely, and the steel cylinders would be remotely seal-welded. A small fraction of the waste carried in the welded steel cylinders might be combustible; if so, it could begin to gasify at 400°F.

Let x be the average temperature of the package at the end of a 2-hr fire. If the cask is 12.5 ft long and 5.4 ft in diameter, then:

$$\frac{(55,000 \text{ lb})[0.15 \text{ Btu/(lb-°F)}](x^\circ\text{F} - 100^\circ\text{F})}{[17,000 \text{ Btu/(hr-ft}^2\text{)}](\pi)(5.4 \text{ ft})(12.5 \text{ ft})} = 2 \text{ hr}$$

$$x^\circ\text{F} - 100^\circ\text{F} = 875^\circ\text{F}$$

$$x = 975^\circ\text{F}$$

The pressure inside the canisters due to gas expansion was calculated by using the perfect gas law.

$$\begin{aligned} P_2 &= P_1 \frac{T_2}{T_1} = 1 \text{ atm} \left(\frac{975 + 460}{560} \right) \\ &= 2.57 \text{ atm abs} = 1.57 \text{ atm gage} \\ &= (1.57 \text{ atm})[14.5(\text{lb/(in}^2\text{)})/\text{atm}] = 23 \text{ psig} . \end{aligned}$$

Inner welded canisters would be capable of withstanding this pressure plus any small decomposition pressure (assumed to be equal to that caused by gas expansion) at 975°F. Therefore, there will be no leakage in any accident category.

II.N.8 CLADDING HULLS

Stainless steel cladding hulls will be produced in the head-end facilities of the reprocessing plant and then placed inside a disposal steel cylinder for shipping.

The steel cylinders are assumed to be leak-tight to 30 psig internal pressure. This condition is equivalent to the following temperature

$$T_2 = \frac{P_2}{P_1} T_1 = \frac{3 \text{ atm abs}}{1 \text{ atm abs}} (560^\circ\text{R}) = 1680^\circ\text{R} \\ = 1220^\circ\text{F}$$

The following cask properties were assumed:

Weight = 188,000 lb
Heat capacity = 0.08 Btu/(lb-°F)
Surface area = $\pi(5.83)(12.5 \text{ ft}) = 229 \text{ ft}^2$

From this information, the time for the cask to reach 1220°F, if exposed to a 1475°F fire, is:

$$\frac{(188,000 \text{ lb})[0.08 \text{ Btu/(lb-°F)}](1220^\circ\text{F} - 100^\circ\text{F})}{[17,000 \text{ Btu/(hr-ft}^2\text{)}](229 \text{ ft}^2)} = 4.34 \text{ hr.}$$

This time period is longer than the 2-hr fire of Category-5 accidents. Therefore, no radioactivity will be released.

II.N.9 SOLIDIFIED HIGH-LEVEL WASTE

The solidified high-level waste shipping package is assumed to have the following specifications:

- (1) Inner steel can is 12 in. diam and 10 ft long and has a welded closure
- (2) Cask contains 12 cans, whose total volume is 6.28 ft³
- (3) Maximum surface temperature of center can in normal operation is 700°F
- (4) Center-line temperature of center can in normal operation is 950°F

- (5) Loaded cask weight is 229,000 lb
- (6) Surface area $= \pi(8.3 \text{ ft})(13 \text{ ft}) = 339 \text{ ft}^2$

The cask surface temperature under normal operating conditions is

$$\Delta T = \frac{Q}{hA} = \frac{(25,000 \text{ W})(3.413 \text{ Btu/hr-W})}{[2.5 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}](339 \text{ ft}^2)}$$

$$= 100^\circ\text{F}$$

The surface temperature is about 200°F for an assumed ambient temperature of 100°F.

The outer surface of this cask was assumed to be exposed to a radiant heat source of 1475°F for a period of 2 hr. A calculation was made to determine the maximum temperature of the solidified waste at the end of the fire.

Under normal operating conditions, the hottest waste will have a temperature of 930°F. At the end of the 2-hr fire, this temperature will rise to 1320°F.

However, the waste is exposed to temperatures over 1600°F in its preparation. Therefore, it would remain stable, no decomposition products would form to increase internal pressures, canister welds would not crack, and no release of activity in any accident category would occur.

II.N.10 NOBLE GASES

Noble gases, predominantly Kr-85, are assumed to be compressed into standard 2200-psi cylinders. For shipment, six of these could be placed into a water-filled shipping cask. The cask is assumed to be built of 3/4-in-thick steel plate and equipped with fusible plugs designed to relieve excessive steam pressure if the package is involved in a fire.

Fifteen percent of the volume of the cask is assumed to be taken up by steel Raschig rings, which are included to help protect the gas cylinders in severe impacts.

Weight of water in package	=	8,189 lb
Weight of Raschig rings	=	11,560 lb
Weight of steel in package	=	4,990 lb
Total weight	=	<u>24,740 lb</u>

$$\begin{aligned}
 \text{Water: } [1 \text{ Btu/(lb-}^\circ\text{F)}](212^\circ\text{F} - 100^\circ\text{F})(8190 \text{ lb}) &= 917,000 \text{ Btu} \\
 \text{Vaporization: } (970 \text{ Btu/lb})(8190 \text{ lb}) &= 7,940,000 \text{ Btu} \\
 \text{Steel: } [0.12 \text{ Btu/(lb-}^\circ\text{F)}](212^\circ\text{F} - 100^\circ\text{F})(16,550 \text{ lb}) &= 222,000 \text{ Btu} \\
 \text{Total} &= 9,079,000 \text{ Btu}
 \end{aligned}$$

$$\begin{aligned}
 \text{Category-3 heat input} &= [17,000 \text{ Btu/(hr-ft}^2\text{)}](112 \text{ ft}^2)(0.4 \text{ hr}) \\
 &= 761,600 \text{ Btu}
 \end{aligned}$$

Temperatures will not reach 212°F, there will be no water loss, and there will be no release of activity.

$$\begin{aligned}
 \text{Category-4 heat input} &= [17,000 \text{ Btu/(hr-ft}^2\text{)}](112 \text{ ft}^2)(1.7 \text{ hr}) \\
 &= 3,236,000 \text{ Btu}
 \end{aligned}$$

Temperatures would reach 212°F, about 25% of the water would be lost due to vaporization, the bottles would remain submerged, and no release of activity would occur.

In the very severe impact in a Category-5 accident, some water might be lost. Assume that 50% of the water could leak out from a penetration or rupture of the outer steel shell.

$$\begin{aligned}
 \text{Water} &= 458,500 \text{ Btu} \\
 \text{Vaporization} &= 3,970,000 \text{ Btu} \\
 \text{Steel} &= 222,000 \text{ Btu} \\
 \hline
 &= 4,650,500 \text{ Btu}
 \end{aligned}$$

$$(17,000 \text{ Btu/hr-ft}^2)(112 \text{ ft}^2)(2 \text{ hr}) = 3,808,000 \text{ Btu} .$$

Most of the water would be boiled away; the cylinders would be exposed to the hot outer steel shell and would absorb energy radiating to it. The cylinders would rupture due to increased pressure and lower the ultimate strength of the steel at the higher temperatures. Therefore, in this accident a 100% loss of radioactivity (equivalent to 900,000 Ci of Kr-85) from the package could occur.

II.N.11 IODINE

Iodine is assumed to be converted to potassium iodide (KI) for shipment to a repository. The material would be placed into canisters that would, in turn, be

placed into insulated shipping packages. Since few packages are produced per year, they could be shipped to the repository along with other waste.

The following specifications are assumed for the KI:

Melting point = 1333°F

Boiling point = 2426°F

In Categories-3 and -4 accidents, the package would have enough protection and insulation to prevent the inner containers from reaching the melting temperature of potassium iodide.

In the Category-5 accident several of the packages could be thrown free and cause one of them to lose its lid and insulation protecting the top flange.

The following specifications are assumed for the package:

Weight of steel = 80.34 lb

Weight of KI = 38.5 lb

C_p of steel = 0.11 Btu/(lb-°F)

C_p of KI = 0.07 Btu/(lb-°F)

It is assumed that there is a heat input of 17,000 Btu/(hr-ft²) and that the end flange of the inner steel vessel is exposed to the fire.

$$\begin{aligned} \text{Time} &= \frac{\{(80.34 \text{ lb})[0.11 \text{ Btu/(lb-°F)}] + (38.5 \text{ lb})[0.07 \text{ Btu/(lb-°F)}]\}(1333^\circ\text{F} - 100^\circ\text{F})}{[17,000 \text{ Btu/(hr-ft}^2\text{)}](0.6 \text{ ft}^2)} \\ &= 1.38 \text{ hr} . \end{aligned}$$

All the KI will be melted in a 2-hr fire, and none will be vaporized because of the high boiling point. The maximal release will be the activity in a package, or 1.83 Ci of iodine-129. The contamination will be localized since the temperatures are not high enough to cause significant vaporization.

II.N.12 CESIUM RELEASE IN SPENT FUEL SHIPPING ACCIDENT

In addition to the krypton, xenon, and iodine that are known to be available for release from the gas plenum of the fuel rod in a spent fuel shipping accident, it is possible that both strontium and cesium would also be available. The probable behavior of strontium and cesium under the postulated release mechanisms is not

well understood at this time. However, the amount of strontium is probably orders of magnitude lower than cesium. If any strontium or cesium are available as metal, they will produce a partial pressure in equilibrium with the temperature of the fuel rod and some could be released in the gaseous state. If they are chemically bound in the fuel pellets in a form such as the molybdate, uranate or other thermally stable compound, they will have a much lower partial pressure and not be released with the gas.

The cesium that might exist as vapor in the void of a fuel rod and be available for release was determined by evaluating the temperature distribution along the length of the hottest (center) rod in one assembly and studying the void space in different zones of the rod. Knowing the vapor pressure-temperature data, the grams and curies of cesium in the void space of the various zones can be calculated. All rods were assumed to have the same cesium activity available as the hottest one.

If the rupture of the rod occurs in the area of the core, cooler gas from the plenum might conceivably pick up heat and additional cesium as it passes the hotter core pellets on its way to being discharged from the rod. The dynamics of this release are complex, and they have not been studied in depth at this time. While it is conceivable the process described above might be capable of releasing more cesium than is reported in Table 4.5-18, the assumptions that were made in arriving at the cesium numbers are considered conservative. This view leads to the conclusion that the reported values are probably an upper limit.

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Appendix II.0

DOSES DUE TO RELEASE OF TRITIUM

II.0.1 Development of the Washout Fraction

A calculation of the fraction of tritium deposited on ground or water surfaces after release is described below. Deposition of tritium was estimated for conditions representing an average of 18 reactor sites in the United States (Appendix II.I):

(1) average annual rainfall of 36.6 in,¹ (2) average annual temperature of 56.3°F,¹ (3) average annual mean maximum mixing depth of 859 m,² and (4) an average wind speed of 4.05 m/sec (Appendix II.I). A precipitation coefficient reported by Dickerson and Crawford³ was used to estimate the tritium deposition:

$$\phi = \frac{R}{LW} \text{ sec}^{-1}$$

where

R = rainfall rate (g/cm²-sec),

L = height of precipitation system (cm),

W = absolute humidity (g/cm³).

The height of the precipitation system (L) is assumed to extend from the mean maximum mixing depth (859 m) to the ground. The absolute humidity (1.168 x 10⁻⁵ g/cm³) was calculated from the average temperature (56.3°F) by assuming that the relative humidity is 100% during rainfall. The calculated value of ϕ is 2.938 x 10⁻⁶ sec⁻¹. It takes 19,753 sec (t) for the plume to travel 50 miles at a wind speed of 4.05 m/sec. The fraction of tritium remaining in a plume after it has traveled t seconds is

$$f = e^{-\phi t}$$

and after traveling for 50 miles

$$f = 0.94.$$

So 6% of the tritium will be deposited on the ground by rainfall in the 50-mile radius.

II.0.2 APPLICATION TO THE FUEL-REPROCESSING PLANT

The atmospheric release of tritium is 1.52 x 10⁵ Ci/year from a 100 meter stack for the fuel reprocessing plant. Since 6% is deposited (washed out) within 50 miles, the average annual deposition is 4.49 x 10⁵ pCi/m². A suitable factor

for estimating the total dose to man for wet deposition of tritium in the temperate zone is 2.9×10^{-11} rem per pCi/m^2 of tritium. The model used to obtain this factor is based on an assumed 2500 g/day intake of water from consumption of food and drinking water. Using this same factor, the estimated average individual dose due to rainout of tritium within 50 miles of the facility is 1.63×10^{-2} millirem/year of plant operation. The estimated population dose due to tritium rainout is 16.3 man-rem/year of fuel-reprocessing plant operation. The dose due to tritium is 0.20 man-rem/year per 1000-MWe LMFBR annual fuel reprocessing requirement.

II.0.3 APPLICATION TO THE LMFBR POWER PLANT

The tritium releases at the LMFBR power plant are estimated to be 60 Ci/year into both the liquid and airborne effluents. Because the LMFBR airborne releases are near ground level, the assumption is that the airborne tritium will be totally deposited within the 50-mile radius from the power plant. Considering radioactive decay as the only removal mechanism and that an 8-m depth of ground water is available for mixing,⁵ the resulting steady-state tritium concentration in ground water is $6.6 \times 10^{-9} \text{ } \mu\text{Ci/cm}^3$. The liquid tritium release results in a river concentration of $2.2 \times 10^{-8} \text{ } \mu\text{Ci/cm}^3$ for a river flow rate of 3000 cfs. One-fifth of the population within 50 miles of the plant was assumed to obtain its drinking water from the river, with the remaining fraction utilizing the ground water.⁶ The concentration of tritium in the diet of the population was taken to be that of the ground water with the intake rate being equivalent to 1 liter/day of water. The atmospheric tritium concentration was based on a relative humidity of 50% derived from ground water. The tritium dose factors of Appendix II.E were used in determining the resulting doses. The estimated maximum annual dose received by any member of the population via the tritium releases of the LMFBR power plant would be less than 0.001 millirem/year. The annual population dose would be about 0.47 man-rem/year for the population within 50 miles of the power plant.

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PLANTS FOR REPROCESSING SHORT-DECAY LMFBR FUEL

II.P.1 INTRODUCTION

This appendix is included in response to inquiries* concerning the effects of short-decay fuel reprocessing, which is not possible without short-decay fuel shipping (see Section 7.3.4).

Certain aspects of reprocessing short-decayed fuel are discussed in Section 4.4.4.4.3. The purpose of this appendix is to expand upon that discussion by describing a concept of a plant designed to reprocess fuel decayed for 120 days or less; to estimate the emissions that would result from normal operation and from a variety of accidents. Because the supporting facilities and land area will remain essentially unchanged, these facilities and their environmental effects will not be discussed.

The general reprocessing technology applicable for short-decay fuel reprocessing is discussed in Sect. 4.4.1.2. However, the radioactivity confinement technology assumed for the short-decay fuel reprocessing plant is still in the development stage. The major difference between the confinement system proposed for short-decay fuel reprocessing and the one proposed for long-decay fuel reprocessing is the almost total recycle of cell ventilation air, process gas (used for mixing, level and specific gravity determination, and liquid pumping), and the air that inadvertently gains entrance to the process equipment. The processes proposed for short-decay reprocessing confinement are the same as those proposed for long-decay reprocessing confinement. Of major concern for reprocessing short-decay fuel is the retention in the plant of relatively large quantities of I-131. The quantity to be retained within the plant is an inverse function of fuel preprocessing decay time. The development of technology for trapping and storing large quantities of I-131 is of major importance. It is a matter that has received a great deal of attention during the past six years and will continue to be important in future years. The consequences of release of a large quantity of radioiodine are well known. The process and equipment must be designed so that mistakes and equipment failures do not result in large releases. Commercially-proven systems that accomplish this desired objective do not exist today. Although the basic technology required for trapping and storing the quantity of I-131 found in 120-day-decayed fuel is available now, final system development remains to be accomplished.

*From the Mid-America Coalition for Energy Alternatives and from the Environmental Protection Agency (Comment Letters 22 pp. 3-5; 53; p. 82).

Because shorter decay times rapidly increase the quantity of radioiodine, the development of the technology for decay time less than 120 days must involve more extensive testing and is less near-at-hand. The system that will be described and for which emission estimates will be made is a conceptual system that is expected to achieve the results indicated upon completion of the development program. The anticipated developmental program will end with pilot plant testing in which the system will be carefully observed, first at long preprocessing decay time, and then, as process performance warrants, at shorter decay times until the optimum decay time is reached.

The attainment of confinement factors necessary to justify use of very short preprocessing decay times cannot be guaranteed; however, the difficulty of their attainment, and the probability of success is seen as being of the same order of magnitude as many other technological objectives that are being undertaken. The final result is relatively well assured, provided appropriate manpower and financial resources are allocated to the effort, and time is available for performance of the work.

II.P.2 PLANT DESCRIPTION

The plant facilities and equipment are the same as those described in Section 4.4.4, Plant Description, with the exceptions discussed below.

II.P.2.1 Gaseous Confinement and Discharge Systems

As discussed in Section 4.4.4.3, reprocessing of short-decayed material will require, for practical reasons, that the process cells operate with low inleakage rate and with the recycle of all gas except the amount that leaks into the cells. Implicit in this Statement is abandonment of the practice currently followed in reprocessing plants of deliberately introducing large volumes of air into the process cells. Under the proposed process arrangement for short-decay processing, the opportunity for gaseous fission products to bypass an appropriate recovery system will be eliminated.

The processes for volatile fission product confinement that would be used in a plant reprocessing short-decayed fuel are anticipated to be the same as those stipulated for long-decayed fuel reprocessing. The principal difference is in the flow arrangement. Figure II.P-1 is a flow diagram of the gaseous confinement system as envisioned for a short-decay fuel-reprocessing plant. In this case, no cell ventilation gas is discharged directly to the atmosphere. Further, the cell atmosphere is planned to be oxygen-deficient, primarily to preclude the possibility

II.P-3

GASEOUS CONFINEMENT SYSTEM FOR A PLANT PROCESSING SHORT-DECAYED FUEL

Figure II.P-1

of an in-cell fire. The volume of gas from process equipment discharged to the atmosphere is estimated to be 100 cfm. In this case, a total of three HEPA filters, and perhaps a sand or packed glass fiber filter (not shown in Figure II.P-1) in series have been indicated, instead of two HEPA filters plus a sand filter.

The iodine trapping system in the model plant for short-decayed fuel utilizes the same trapping processes as those employed in the model plant for long-decayed fuel. However, the Iodex scrubbing column used on the gas stream that is eventually discharged to the environment, will have more gas-liquid contact stages, and greater iodine decontamination effectiveness which will be required under normal operating conditions. In this case, a decontamination factor (DF) of $\geq 10^6$ is the design objective. DF's of 9×10^5 have been obtained in a six-stage laboratory-scale column.¹ Because the DF per stage was not observed to decrease toward the top of the column, it is inferred that an additional factor of 10 to 100 could have been obtained through use of additional stages.

The Iodex scrub system will have a series of decay tanks in which the I-131 will decay. The number of tanks and the total decay period will be dependent upon the preprocessing decay time. These decay tanks will contain concentrated nitric acid and fission product iodine as HI_3O_8 . The tanks will have redundant means for heat removal. These tanks and the Iodex scrub columns will be installed in enclosures to provide mechanical isolation from each other and from other items of process equipment. Means will be available in these enclosures to collect any leakage of I-131 and transfer it to another vessel.

The gas leaving the Iodex scrub column will be compressed, treated to remove nitrogen oxides, dehumidified, filtered, treated with silver zeolite iodine sorbers, and finally treated for removal of noble gases.

The silver zeolite iodine sorbers are expected to have a total bed depth of as much as 8 in. and a normal face velocity of ~ 25 fpm. Under normal operating conditions, a DF of $\geq 10^3$ would be sufficient, although it would appear that larger DF's are attainable, barring future discovery of some unusually refractory iodine species. Under normal conditions with abnormally high loadings of iodine in the feed stream, a DF $\geq 10^6$ is expected.

Reliability for the iodine retention system would be achieved through use of redundant sources of supply of nitric acid for the Iodex system, by providing instruments to indicate the adequacy of performance of the Iodex and silver zeolite systems,

and by providing an abundant amount of silver zeolite in an operating and in a spare sorber. In the event of a failure of the Iodex system to function, alarms would actuate to direct the cessation of those operations that cause the iodine to be released to a gas stream. As an additional safety measure, the Iodex system provided for the recycle process gas would be piped so that a switch of a valve would allow the gas to be used on an emergency basis in treating the off-gas being discharged to the environment. The performance of the silver zeolite iodine sorbers would be checked regularly to assure the availability of a system capable of retaining an unusual surge of iodine.

The noble gas removal system is also expected to act as an iodine trap. Under normal operating conditions very little iodine will reach the noble gas system. However, if a substantial amount were to reach the noble gas recovery system, it would retain a substantial fraction of the iodine. No benefit from this phenomenon will be assumed in subsequent calculations of normal and accidental emissions, thus providing a degree of conservatism.

The filtration system for the process gas discharge to atmosphere is indicated as consisting of three HEPA filters and perhaps a sand filter in series. Under the assumptions used for HEPA filters in Section 4.4, these filters will provide a DF for particulates of 10^9 (as opposed to 10^8 for two HEPA filters and a sand bed) and will be operating on a gas stream that presumably has no greater particulate concentration than will be found in the process off-gas of the model plant for long-decay reprocessing plant. Furthermore, the volume of gas is only 0.1 times as great as that of the long-decay plant. Therefore, the logical conclusion is that the particulate emission via the process off-gas will be reduced by a factor of 1000. Inasmuch as there is no discharge of cell ventilation gas to the environment, the contamination component is totally eliminated.

II.P.2.2 Liquid Confinement and Discharge Systems

The liquid confinement and discharge system assumed for the model short-decayed fuel reprocessing plant is the same as for the model long-decayed fuel reprocessing plant except that the effluent water, if necessary, is retained in a series of tanks for a sufficient period of time to assure the virtual absence of I-131. At worst, the holdup time will probably be no greater than 6 months. The rate of I-131 introduction to the decay tanks will be $\leq 10^{-5}$ times the plant iodine input rate.

II.P.3 EFFLUENTS FROM NORMAL OPERATION

Only radioactive effluents will be considered in this section.

The radioactive nuclides to be considered were determined in the same fashion as described in Section 4.4.5.3. At decay times of 120 days or less, Xe-133, Ba-140, and La-140 must be considered in addition to the ones that were radiologically significant at 365-days decay.

The radioactive amounts contained in the plant feed material and the plant effluents are indicated in Table II.P-1 for the cases of feed material decayed for 30 and for 120 days. The plant confinement factors are also indicated in this table. Because there will be no cell ventilation off-gas per se, and because the quantity of process vessel off-gas is reduced by a factor of 10 from the quantity used for the model plant for 365-day-decayed fuel, the confinement factor for ruthenium, uranium, plutonium, and the particulates has been improved by a factor of 10. And because of elimination of a route for the noble gases to bypass the collection system, the confinement factor has been improved by a factor of 2. The I-129 confinement factor has been improved to 1×10^9 on the basis of more complete treatment and elimination of a by-pass route via the cell ventilation air. The I-129 confinement factor is not shown as being improved, because an opportunity still exists for its escape via the water stream being evaporated into the atmosphere. Conversely, I-131 will not escape via this route because of decay during a pre-discharge storage period. No change was indicated for the tritium confinement factor.

II.P.4 POSTULATED ACCIDENTS

Because the cell ventilation gas is not discharged directly to the environment, spills of radioactive materials can occur within the cells without producing a perceptible increase in discharge of these materials from the plant stack. The use of an oxygen-deficient atmosphere in the cells eliminates not only the possibility for a cell fire but also the potential for loss of radioactivity to the environment via this source. Accidents that could result in perceptible increases in plant emissions are failure of an Iodex scrubber by misoperation or by explosion, failure of the noble gas recovery system, and nuclear criticality.

Failure of an Iodex scrubber, whether as a result of explosion or of misoperation that totally negates its effectiveness, will result in greater amounts of iodine being fed to the next iodine removal system in the off-gas treatment train, namely, the silver zeolite sorbers. As described in Section II.P.2, the Iodex column will be

TABLE II.P-1

RADIOACTIVE CONTENT OF FEED AND GASEOUS EFFLUENT STREAMS FOR SHORT-DECAYED FUEL

Radionuclide	Input Radioactivity (Ci/ton)		Confinement Factor	Radioactivity Release (Ci/Year)	
	30 Days	120 Days		30 Days	120 Days
Volatile					
H-3	1.07E2	1.05E2	1	1.60E5	1.57E5
Kr-85	8.61E3	8.48E3	2E2	6.45E4	6.35E4
I-129	3.92E-2	3.94E-2	1E4	5.88E-3	5.91E-3
I-131	1.08E5	4.65E1	1E9	0.16E0	6.00E-5
Xe-133	5.64E4	4.08E-1	2E2	4.23E5	3.06E0
Semivolatile					
Ru-103	1.47E6	3.04E5	1E10	2.20E-1	4.56E-2
Ru-106	1.29E6	1.09E6		1.94E-1	1.64E-1
Uranium					
U-234	1.19E-1	1.33E-1		3.75E-8	3.99E-8
U-235	2.70E-3	2.70E-3	5.E9	8.10E-10	8.10E-10
U-236	7.77E-3	7.91E-3		2.33E-9	2.34E-9
U-238	2.87E-1	2.87E-1		8.61E-8	4.20E-2
Plutonium					
Pu-238	1.91E4	1.92E4		1.43E-3	1.44E-3
Pu-239	3.89E3	2.89E3	2E10	2.91E-4	2.91E-4
Pu-240	4.81E3	4.81E3		3.60E-4	3.60E-4
Pu-241	5.67E5	5.61E5		4.26E-2	4.20E-2
Particulates					
Sr-89	5.70E5	1.72E5		1.71E-2	5.16E-3
Sr-90	4.78E4	4.75E4		1.43E-3	1.42E-3
Y-90	4.78E4	4.75E4		1.43E-3	1.42E-3
Y-91	8.56E5	2.96E5		2.57E-2	8.88E-3
Zr-95	1.63E6	6.24E5		4.89E-2	1.87E-2
Nb-95	2.08E6	1.11E6		6.24E-2	3.33E-2
Te-127m	2.98E4	1.68E4		8.94E-4	5.04E-4
Te-129m	3.90E4	6.23E3		1.17E-3	1.89E-4
Ag-110m	9.29E2	7.26E2		2.79E-5	2.18E-5
Sb-125	9.29E2	2.41E4		7.68E-4	7.23E-4
Cs-134	2.74E4	2.49E4	5E10	8.22E-4	7.47E-4
Cs-137	1.27E5	1.26E5		3.81E-3	3.79E-3
Ba-140	4.29E5	3.28E3		1.29E-2	9.84E-5
La-140	4.94E5	3.78E3		1.48E-2	1.13E-4
Ce-141	9.72E5	1.42E5		2.92E-2	4.26E-3
Ce-144	1.17E6	9.43E5		3.51E-2	2.82E-2
Pm-147	3.45E5	3.25E5		1.04E-2	9.75E-3
Eu-154	1.44E3	1.42E3		4.32E-5	4.28E-5
Eu-155	5.35E4	4.86E4		1.60E-3	1.46E-3
Am-241	2.07E3	2.29E3		6.21E-5	6.87E-5
Am-243	5.19E1	5.19E1		1.56E-6	1.56E-6
Cm-242	6.64E4	4.53E4		1.99E-3	1.36E-3
Cm-244	1.27E3	1.25E3		3.81E-5	3.75E-5

capable of, and operated to achieve, a DF of $\geq 10^6$. The silver zeolite sorbers will be of a total depth of ≥ 8 in. and will provide a DF of $\geq 10^6$ during periods of high iodine loading in the feed gas. Instruments located in the off-gas stream will provide an alarm when the off-gas stream contains an abnormally high loading of iodine and will cause an immediate cessation of iodine-volatilizing operations. It is believed that the rate of evolution of iodine could be reduced by a factor of 100 in less than 15 min. A conservative assumption is that the evolution of iodine continues unabated for 2 hr.

Under the conditions described above, the quantity of iodine released as a result of an Iodex system failure is that amount contained in 2 hr of plant feed divided by the solid sorber DF of 10^6 . The amount of iodine released from a 1500-ton/year plant is, therefore, the amount found in 4.17×10^{-7} tons of feed. The amounts released would be as follows:

	<u>Preprocessing Decay Time</u>	
	<u>30 Days</u>	<u>120 Days</u>
I-129, Ci	1.63E-8	1.62E-8
I-131, Ci	4.50E-2	1.94E-5

Utter failure of the noble gas recovery system would result in a 100-fold increase in the release rate for these gases until such time as their evolution could be stopped. Using the same basis for plant shutdown time as used above for the iodine release, this failure would result in the release of the quantity of gas found in 0.417 ton of fuel. The amounts released would be as follows:

	<u>Preprocessing Decay Time</u>	
	<u>30 Days</u>	<u>120 Days</u>
Kr-85, Ci	3.59E3	3.57E3
Xe-133, Ci	2.35E4	1.70E-1

In the event of a catastrophic failure of a noble gas storage vessel, the gas would not be lost to the environment unless the failure occurred simultaneously with or after the failure of the noble gas trapping system, provided the container is within the normal noble gas confinement area, which vents to the cell atmosphere. In the case of a gas cylinder failure during an outage of the noble gas removal system, the gas would mix with the cell atmosphere. Because the time required to "displace" a cell volume will probably be several days at the anticipated rate of gas inleakage, gas would be very slowly liberated to the environment.

A nuclear criticality incident in a short-decayed fuel reprocessing plant would result in the release of the same amounts of noble gases and I-129 as would a

nuclear criticality incident in the dissolver of a long-decayed fuel reprocessing plant (see Table 4.4-17). The quantity of the short-lived halogens (including I-131) released to the environment would be decreased by a factor of 10^5 .

II.P.5 RADIOLOGICAL EFFECTS OF SHORT-DECAY FUEL REPROCESSING

Environmental effects evaluations of releases resulting from reprocessing 30-day and 120-day decayed fuel are confined to radiological effects only, because the nonradiological effects are expected to be essentially the same as those already discussed in Section 4.4. The same assumption for evaluation of environmental transport and dosimetry detailed in Section 4.4 and Appendices II.E and II.I is also used in this Appendix.

II.P.5.1 Radiological Effects of Normal Plant Operation

All radioactive effluents will be released to the atmosphere via the 100-m stack, including the water in the liquid confinement and discharge system. The quantities of each radionuclide considered in the radiological analysis are listed in Table II.P-1 of this Appendix.

Shorter decay times for the reprocessed fuel do not result in increased releases of radioactivity. Although the same general reprocessing technology is applicable for short-decay fuel reprocessing, the assumed radioactivity confinement technology includes more advanced concepts. In going to short-decay fuel reprocessing the assumptions include no change in confinement for tritium and I-129; however, noble gas confinement is increased by a factor of 2, I-131 confinement is increased by a factor of 10^5 , and confinement of all other radionuclides is increased by a factor of 10. Furthermore, since the growth of some of the daughter activities of the actinides at 30 days and 120 days is less than the activity that grows at 365 days, the potential doses are proportionately smaller for the shorter decay times. Increases in dose anticipated due to the larger inventory of relatively short-lived fission product radionuclides, such as I-131, Xe-133, Ru-103, Zr-85, and Sr-89, are off-set by the increased confinement factors assumed. As a net result, the dose estimates obtained for the short-decay fuel reprocessing, both the 30-day decay and the 120-day decay, are smaller in most instances than are comparable estimates for the 365-day decay fuel reprocessing.

Table II.P-2 summarizes the estimated total-body and critical-organ doses to adult individuals at locations of maximum exposure resulting per year of normal operation of an LMFBR fuel-reprocessing plant handling 30-day and 120-day decayed fuel.

Table II.P-2

SUMMARY OF THE ESTIMATED TOTAL-BODY AND CRITICAL-ORGAN DOSES TO ADULT INDIVIDUALS^a

Exposure Pathway	Dose (millirems)							
	Total-body		Bone		GI Tract		Thyroid	
	30-day	120-day	30-day	120-day	30-day	120-day	30-day	120-day
<u>Gaseous Effluents</u>								
Immersion ^b	0.60	0.004	0.60	0.004	0.60	0.004	0.60	0.004
Contaminated Ground Surface ^b	0.020	0.020	0.020	0.020	0.020	0.020	0.020	0.020
Inhalation ^b	0.43	0.43	0.88	0.88	0.42	0.42	0.44	0.43
Ingestion (terrestrial food chain) ^b	1.1	1.1	1.6	1.6	4.0	3.5	1.3	1.2
<u>Liquid Effluents</u>								
(No radio- active liquid effluents)	None		None		None		None	
<u>Direct Irradiation</u>								
From the Plant	Negligible		Negligible		Negligible		Negligible	
Total	2.2	1.6	3.1	2.5	5.1	4.0	2.4	1.7

^aAt locations of maximum exposure resulting per year of normal operation of an LMFBR fuel-reprocessing plant handling 30-day and 120-day decayed fuel

^bThese dose estimates are for a location on the site boundary 1000 m from the stack

Population doses to a distance of 50 miles from the facility are summarized in Table II.P-3 for short-decay fuel reprocessing. The immersion dose is almost entirely (99%) due to Xe-133 for 30-day-decay fuel, while for 120-day-decay fuel, the dose is nearly all (98%) contributed by Kr-85. Major contributors (over 80%) to the ground exposure dose are Ru-106 and Cs-137. The dose contributions due to ingestion and exposure are from Ru-106 and Cs-137. The dose contributions due to ingestion and inhalation are due largely to tritium, however, the GI tract is an exception, receiving approximately 75% of its ingestion dose from Ru-106. The relative uniformity of man-rem dose estimates among reference organs and decay times in Table II.P-2 reflects the fact that tritium, Kr-85, and Xe-133, each assumed to irradiate total-body, bone, GI tract, and thyroid uniformly, are the major dose contributors in most instances.

Comparison of the dose estimates in Table II.P-2 and II.P-3 with those obtained for 365-day decay fuel seems academic in view of the differences in confinement factors used. It is sufficient to say that the doses obtained for 30-day and 120-day decay fuels are equal to or less than those estimated previously for 365-day decay fuel.

II.P.5.2 Environmental Effects of Postulated Accidents Involving Short-Decay Reprocessing

The methodology used to estimate doses from accidental releases involving short-decay reprocessing was the same as used to estimate doses from reprocessing 365-day decayed fuel (see Section 4.4.9).

Dose estimates are summarized in Table II.P-4 for the two accidents identified as resulting in releases that would be different from similar accidents involving reprocessing 365-day decayed fuel. These accidents involve releasing an increased quantity of I-131 due to failure of the Iodex scrubber system and a release of Xe-133 due to failure of a noble gas storage system. The estimated maximum doses to individuals from inhalation and immersion in the overpassing cloud are not appreciably larger than the doses estimated for similar accidents involving the longer decayed fuel.

Assuming that an atmospheric release is confined to one 22.5° sector out to a distance of 50 miles from the stack, the total-body population dose from either type of accident is estimated to be less than 1 man-rem.

Table II.P-3

SUMMARY OF THE ESTIMATED TOTAL-BODY AND CRITICAL-ORGAN DOSES TO THE POPULATION^a

Exposure Pathway	Population Dose at Indicated Fuel-Decay Period (man-rems)							
	Total-Body		Bone		GI Tract		Thyroid	
	30-day	120-day	30-day	120-day	30-day	120-day	30-day	120-day
<u>Gaseous Effluents</u>								
Immersion ^b	14	0.088	14	0.088	14	0.088	14	0.088
Contaminated Ground Surface ^b	0.33	0.33	0.33	0.33	0.33	0.33	0.33	0.33
Inhalation ^b	8.9	8.9	18	18	8.7	8.7	9.1	8.9
Ingestion (terrestrial food chain) ^b	18	18	26	26	63	55	21	19
<u>Liquid Effluents</u>								
(No radio- active liquid effluents)	None		None		None		None	
<u>Direct Irradiation</u>								
From the Plant	Negligible		Negligible		Negligible		Negligible	
Total	42	28	59	45	87	65	45	29

^aFrom all pathways per year of normal operation of an LMFBR fuel-reprocessing plant handling 30-day- and 120-day-decayed fuel.

^bThese dose estimates are for a location on the site boundary 1000 m from the stack.

Table II.P-4

POTENTIAL DOSES RESULTING FROM RADIONUCLIDE RELEASES THAT ACCOMPANY
POSTULATED ACCIDENTS INVOLVING SHORT-DECAY FUEL REPROCESSING^a

Accident Description	Critical Radionuclides	Dose from Inhalation and Immersion in Passing Cloud at Indicated Fuel-Decay Period (rem) ^a			
		Total-Body		Thyroid	
		30-day	120-day	30-day	120-day
Failure of Iodex Scrubber System	I-129	5.8 E-4	5.7 E-4	4.3 E-1	4.2 E-1
	I-131	5.0 E-8	2.2 E-11	2.8 E-5	1.2 E-8
Failure of Noble Gas Storage Vessel	Kr-85	3.8 E-6	3.8 E-6		
	Xe-133	5.4 E-4	3.9 E-9		

^aStack height = 100 m. The highest potential doses to an individual occurred at about 1000 m from the stack.

REFERENCE FOR APPENDIX II.P

1. W. E. Unger et al., "Aqueous Fuel Reprocessing Quarterly Reported for Period Ending March 31, 1973," Report ORNL-TM-4240, pp. 27-28, June 1973.