ENVIRONMENTAL ASSESSMENT

**Operation of N Reactor
and
Fuels Fabrication Facilities**

Hanford Reservation

Richland, Benton County, Washington

August 1980

**U.S. DEPARTMENT OF ENERGY
Assistant Secretary for Defense Programs
Washington, D.C. 20545**

MASTER

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18

19

20

21

22

CONTENTS

LIST OF FIGURES.	vi
LIST OF TABLES	viii
LIST OF ABBREVIATIONS	xii
FOREWORD	xiv
I. SUMMARY.	I-1
I.1 DESCRIPTION OF SITE, PROGRAMS AND FACILITIES.	I-1
I.2 ENVIRONMENTAL EFFECTS	I-3
I.2.1 Routine Operations	I-3
I.2.2 Effects of Postulated Accidents.	I-6
I.3 UNAVOIDABLE ADVERSE EFFECTS	I-8
I.4 ALTERNATIVES	I-8
II. DESCRIPTION OF SITE, PROGRAM AND FACILITIES.	II-1
II.1 CHARACTERIZATION OF EXISTING ENVIRONMENT.	II-1
II.1.1 Hanford Reservation	II-1
II.1.2 Demography and Land Use.	II-1
II.1.3 Historical and Archaeological Sites.	II-3
II.1.4 Hydrology	II-3
II.1.5 Ecology	II-3
II.1.6 Meteorology.	II-4
II.1.7 Geology and Seismicity	II-5
II.2 DESCRIPTION OF PROGRAMS	II-14
II.2.1 Operational Relationships	II-14
II.2.2 Anticipated Benefits.	II-14
II.2.3 Safeguards	II-16

II.3	DESCRIPTION OF FACILITIES.	II-19
II.3.1	N Reactor and Facilities	II-19
II.3.2	Fuels Fabrication Facilities	II-102
	REFERENCES FOR SECTION II.	II-140
III.	ENVIRONMENTAL EFFECTS.	III-1
III.1	N REACTOR OPERATIONS.	III-1
III.1.1	Radiological Impact	III-1
III.1.2	Nonradiological Impact	III-12
III.1.3	Natural Forces.	III-21
III.1.4	Accident Impacts	III-24
III.2	FUELS FABRICATION FACILITIES	III-59
III.2.1	Radiological Impact	III-59
III.2.2	Nonradiological Impact	III-61
III.2.3	Natural Forces.	III-62
III.2.4	Accident Impacts	III-64
	REFERENCES FOR SECTION III	III-72
	APPENDIX III.A ANALYTICAL PARAMETERS USED TO EVALUATE RADIOLOGICAL EFFECTS OF ACCIDENTS	III.A-1
IV.	UNAVOIDABLE ADVERSE EFFECTS--N REACTOR AND FUELS FABRICATION FACILITIES	IV-1
IV.1	GENERAL	IV-1
IV.2	N REACTOR	IV-1
IV.3	FUELS FABRICATION FACILITIES	IV-2

V.	ALTERNATIVES--N REACTOR AND FUELS FABRICATION FACILITIES. . .	V-1
V.1	N REACTOR.	V-1
V.1.1	Alternative 1 - Shutdown of N Reactor, 1983 or 1990	V-1
V.1.2	Alternative 2 - Change Percent of Plutonium-240 Production	V-3
V.2	FUELS FABRICATION FACILITIES	V-5
V.3	DISPOSITION OF N REACTOR FUEL	V-5

LIST OF FIGURES

<u>Figure</u>	<u>Page</u>
II.1 Location of DOE Hanford Reservation Showing 50-Mile Radius from N Reactor.	II-2
II.2A Seismic Risk Map	II-8
II.2B Approximate Relationship Between Earthquake Magnitude, Intensity, Acceleration and Energy Release	II-9
II.3 Earthquake Zones of Washington	II-11
II.4 Map of Hanford Reservation.	II-20
II.5 100-N Area Layout.	II-21
II.6 100-K Area Plant Layout.	II-22
II.7 Elevation Versus Flow at Columbia River Mile 380.0.	II-23
II.8 Columbia River Bottom Contours in the Vicinity of N Reactor	II-23
II.9 Velocity Profiles at Low River Flows in the Vicinity of N Reactor	II-24
II.10 N Reactor Graphite Stack	II-28
II.11 Primary Piping Arrangement.	II-32
II.12 Plant Water Use Diagram.	II-37
II.13 Circulating Raw Water System for 100-N Area	II-40
II.14 Cross Section of the River Pump House	II-42
II.15 Traveling Water Screens.	II-44
II.16 Seal Wells, Spillway and River Channel Discharge - 1908 N	II-45
II.17 102-in. Outfall Line, 100-N Area.	II-47
II.18 N Reactor Liquid Effluent Flow Schematic	II-60

II.19	1301-N Facility Layout	II-61
II.20	Calculated Flow Paths Originating from the 1301-N Facility with Travel Time Lines.	II-63
II.21	Cross Section of the 1301-N Facility	II-64
II.22	100-N Atmospheric Release Points.	II-79
II.23	105-N Ventilation and Exhaust Systems	II-80
II.24	117-N Exhaust Filter and Diversion Facility	II-82
II.25	Fuels Fabrication - Aerial Photo.	II-103
II.26	Assembled Inner and Outer N Reactor Fuel Element	II-105
II.27	N Reactor Fuel Element Fabrication Process	II-106
II.28	Solar Evaporation Basin.	II-114
II.29	Maximum Water Usage in Fuel Fabrication Facilities.	II-116
II.30	Process Sewer System.	II-117
II.31	Sanitary Sewer System	II-118
II.32	Sources of Uranium Radioactivity UNI Fuels	II-120
II.33	Waste Acid and Uranium-Bearing Acid Recovery System	II-123
II.34	Outside Storage Tanks and Pipe Trenches	II-125
II.35	Fume Exhaust Systems.	II-126

LIST OF TABLES

<u>Table</u>	<u>Page</u>
II.1 Anticipated Benefits of N Reactor Operations.	II-16
II.2 Plant Water Use	II-38
II.3 Primary Coolant Radionuclide Composition	II-51
II.4 Spent Fuel Coolant Radionuclide Composition	II-53
II.5 Graphite and Shield Cooling System Radionuclide Composition.	II-54
II.6 Rod Cooling System Radionuclide Composition	II-56
II.7 Steam Generator Blowdown Radionuclide Composition	II-57
II.8 1310-N Facility Radionuclide Composition	II-59
II.9 Discharge of Radionuclides to 1301-N Facility and Seepage to River During 1978.	II-65
II.10 Physiochemical Characteristics of Radionuclides in Liquid Effluents Discharged to 1301-N	II-67
II.11 1301-N Facility Decontamination Factors	II-68
II.12 Radionuclide Burden of 1301-N Facility at End of 1978.	II-68
II.13 Releases to River Via 102-In. Discharge Line During CY-1978.	II-71
II.14 Radionuclide Discharge to River as a Result of Spill Cooler Failures	II-72
II.15 Ventilation Exhaust Location 105-N	II-76
II.16 Annual Radionuclide Releases to Atmosphere From N Reactor Operations.	II-77
II.17 Exhausts Vents - 109-N Building	II-78
II.18 100-N Area Liquid Effluents	II-86

II.19	Emissions to the Atmosphere from Oil Combustion.	II-88
II.20	N Reactor Effluent Monitoring and Sampling Program	II-93
II.21	Fuels Fabrication Facilities Water Usage	II-115
II.22	Summary of Uranium Activity Discharge from 300-Area Fuels Fabrication Facilities in CY 1978.	II-121
II.23	Chemicals Discharged to 100-H Area Solar Evaporation Basins, CY-1978.	II-129
II.24	Calculated Chemicals Discharged to the 300-Area Process Trenches, CY-1978	II-130
II.25	Emissions From Building 333 Exhausts.	II-132
III.1	Radionuclides Discharged to the Environment from N Reactor During CY-1978	III-2
III.2	Calculated Dose to the Maximum Individual During CY-1978 from Effluents Released from N Reactor During CY-1978	III-4
III.3	Calculated 50-Yr Dose Commitment to the Maximum Individual from Effluents Released from N Reactor During 1978	III-6
III.4	Calculated Population Dose During CY-1978 Due to Effluents Released from N Reactor During CY-1978	III-7
III.5	Calculated 50-Yr Population Dose Commitment from Effluents Released from N Reactor During CY-1978	III-7
III.6	Estimated Cumulative Population, Cumulative Annual Whole Body Dose and Average Annual Body Dose in Circular Areas Around N Reactor from Airborne Effluents Released from all Facilities in CY-1978.	III-8
III.7	Summary of Doses to Man from N Reactor Operations at Hanford During CY-1978	III-9
III.8	Whole Body Occupational Exposure--N Reactor and Fuels Fabrication Facilities Personnel	III-11
III.9	Historical N Reactor Accidents.	III-26
III.10	Radionuclide Release to 1301-N Facility Hypothetical Maximum Single-Element Cladding Failure	III-33
III.11	Radionuclide Releases Following the January 12, 1976, Fuel Cladding Failure	III-34

III.12	Transmission of Radionuclides from 1301-N Facility to Columbia River.	III-35
III.13	Radionuclide Release to Columbia River.	III-36
III.14	Radiation Doses from Hypothetical Maximum Single-Element Fuel Cladding Failure	III-36
III.15	Radiation Doses Following January 12, 1976, Fuel Cladding Failure	III-37
III.16	Primary Coolant System Radionuclide Inventory	III-42
III.17	Fission Product Releases From Small Pipe Break Accident . .	III-43
III.18	Population Exposure Resulting From Postulated Small Pipe Break Accident	III-44
III.19	Primary Coolant System Radionuclide Inventory	III-50
III.20	Fission Product Releases From Large Pipe Break Accident . .	III-51
III.21	Population Exposure Resulting From the Postulated Large Pipe Break Accident	III-52
III.22	Postulated Fuel Shipping Accident Fission Product Source Terms	III-54
III.23	Population Exposure Resulting From the Postulated Fuel Shipping Accident	III-55
III.24	Postulated Fuel Cask Drop Accident Fission Product Source Terms	III-57
III.25	Population Exposure Resulting From the Postulated Fuel Cask Drop Accident	III-58
III.26	Uranium Released to the Atmosphere from 300-Area Fuels Fabrication Facilities During 1978	III-59
III.27	Calculated Dose Commitment to the Public from 300-Area Fuels Fabrication Facilities During 1978	III-60
III.28	Average Inlet Chemical Concentration to 300-Area Process Sewer Trenches in 1978	III-62
III.29	Historical Incidents at the Fuels Fabrication Facilities. .	III-66
III.30	Source Terms for Postulated Uranium Fire	III-70
III.31	Calculated Doses for Postulated 300-Area Uranium Fire. . .	III-71

LIST OF ABBREVIATIONS

BPA - Bonneville Power Administration	g/d - grams per day
C - Centigrade	g/ml - grams per milliliter
cpm - counts per minute	gal - gallon
cfm - cubic feet per minute	gpd - gallons per day
cfs - cubic feet per second	gpm - gallons per minute
Ci/yr - curies per year	gr/day - grains per day
CRW - circulating raw water	HEDL - Hanford Engineering Development Laboratory
CY - calendar year	HEHF - Hanford Environmental Health Foundation
DOE - Department of Energy	HEPA - high-efficiency particulate air
DOT - Department of Transportation	HGP - Hanford Generating Project
DUN - Douglas United Nuclear	HMS - Hanford Meteorological Station
ECCS - Emergency Core Cooling System	hr - hour
EDTA - ethylene-diamine-triacetic acid	hp - horsepower
EPA - Environmental Protection Agency	HRC - horizontal control rod
ERDA - Energy Research and Development Administration	in. - inch
F - Fahrenheit	KE - K East Area
FFTF - Fast Flux Test Facility	kg - kilogram
fps - feet per second	kV - kilovolt
ft ² - square feet	kVA - kilovolt-ampere
ft ³ - cubic feet	kW - kilowatt
ft/min - feet per minute	kWhr - kilowatt-hour
g - gravity	

lb - pound	pCi/l - picocuries per liter
LMFBR - Liquid Metal Fast Breeder Reactor	pH - a measure of the relative acidity of alkalinity of a solution
mg/l ³ - milligrams per cubic liter	PMF - probable maximum flood
mm - millimeter	PNL - Battelle Memorial Institute Pacific Northwest Laboratories
mph - miles per hour	ppm - parts per million
mrem - millirem	psi - pounds per square inch
m/sec - meters per second	psig - pounds per square inch gauge
m - micrometer	PVC - polyvinyl chloride
g/m ³ - micrograms per cubic meter	RHO - Rockwell Hanford Operations
MSL - mean sea level	SNM - special nuclear material
MTU - metric tons of uranium	SRP - Savannah River Plant
MW - megawatt	SCFD - standard cubic foot dry air
Mwhr - megawatt-hour	ΔT - change in temperature
MWe - megawatts electrical	UBC - Uniform Building Code
MWt - megawatts thermal	UNC - UNC Nuclear Industries, Inc.
ND - not detected	V - volt
NLO - National Lead Company of Ohio	WHC - Westinghouse Hanford Company
NPDES - National Pollutant Discharge Elimination System	wk - week
O.D. - outside diameter	WPPSS - Washington Public Power Supply System

FOREWORD

Eliminate or UNI-1313
Revise - Scope is
To convert from 12 to
6% mode - NOT
"Continuing" operations.

This environmental assessment presents the impacts and alternatives of operating the N Reactor and Fuels Fabrication Facilities through 1990,

In December 1975, the Energy Research and Development Administration issued ERDA-1538, a final environmental impact statement of waste management operations, Hanford Reservation, Richland, Washington. ERDA-1538 reassessed the environmental impacts associated with the continuation of the Hanford waste management program. That document covered the wastes generated by production, research and development and other programs and activities at Hanford, but it was not a detailed statement on the operation of those programs.

ERDA-1538 stated that an environmental assessment would be prepared covering all of the operational aspects of the Hanford Site such as N Reactor operations. The purpose of this assessment is to determine whether the continued operation of the N Reactor and associated Fuels Fabrication Facilities may require the issuance of a supplement to ERDA-1538. This assessment covers the operational aspects of these production facilities, including associated waste management aspects, but does not cover the alternatives and/or costs and benefits with respect to the production of special nuclear materials.

ERDA-1538 presented extensive historical, site, construction, programmatic, impact, alternatives and other data pertinent to the waste generated during the operation of the facilities covered in this report. A concerted effort has been made in preparing this report to reduce redundancy and unnecessary material by referencing the applicable data already contained in ERDA-1538. Where appropriate for continuity or for the convenience of the reader, some material has been extracted from ERDA-1538 and included.

The discussion of environmental effects resulting from the continued operation of these production facilities updates the data presented in ERDA-1538. Potential environmental impacts are expected to be reduced as the planned modifications described in Sections II.3.1.8 and II.3.2.8 of this assessment are implemented.

Copies of ERDA-1538 are available for inspection at DOE public document rooms located in Washington, D.C.; Richland, Washington; Idaho Falls, Idaho; Aiken, South Carolina; and Oakland, California.

I. SUMMARY

Should be revised
to reflect
proposed action of
changing to 60% mode.

I.1 DESCRIPTION OF PROGRAMS, FACILITIES AND SITES

As part of the Department of Energy's (DOE) special nuclear materials programs, N Reactor produces plutonium for the Department of Defense's nuclear weapons program and for DOE research programs. Currently, byproduct steam from N Reactor is used by the Washington Public Power Supply System's (WPPSS) Hanford Generation Project to produce 4 to 5 billion kWhr of electricity per year. Thus, N Reactor, authorized by Congress in 1958, is a dual purpose reactor producing special nuclear materials and steam for the production of electricity.

The metallic uranium fuel used in N Reactor is produced by the Fuels Fabrication Facilities in the 300 Area. Currently these facilities fabricate 300 to 700 tons of N Reactor fuel each year in support of the dual program function of N Reactor.

After the metallic uranium fuel has been irradiated within N Reactor to the proper specifications, the spent fuel is stored first in N Reactor's spent fuel basin and later in the 105-K spent fuel basins. ~~Fuel reprocessing awaits startup of the Purex reprocessing plant in 1983.~~

N Reactor facilities occupy approximately a 90-acre site on the right bank (looking downstream) of the Columbia River. The reactor site is within a 5-1/2 mile radius exclusion area in the northcentral portion of the Hanford Reservation. The WPPSS Hanford Generating Project is adjacent to the 100-N Area, the reactor site designation.

The Fuels Fabrication Facilities, within the 300 Area, occupy a 15-acre site on the right bank of the Columbia River in the extreme southeastern part of the Hanford Reservation. The 105-K spent fuel storage basins lie along the right bank of the Columbia River, 2 miles west (upstream) of N Reactor, in 100-K Area.

All of these special nuclear materials facilities are located on the 570 square mile, semiarid Hanford Reservation. The major land use on the Hanford Reservation is for nuclear-oriented production and research.

There are approximately 236,000 persons living within a 50-mile radius of 100-N Area, all outside of the Hanford Reservation. Land use within the 50-mile radius includes homes, business outlets, light non-nuclear industry, other nuclear facilities and farming.

The area climate is mild and dry with occasional periods of high wind. The average monthly temperature is about 54 F with a record high of 115 F and a low of -27 F. The annual precipitation averages 6.28 inches per year with a record high of 11.45 inches and a low of 3.26 inches. The mean monthly wind speed averages 7.6 miles per hour with a record high of 80 miles per hour. In 33 years of observation only one tornado has been observed on the Hanford Reservation and it caused no damage. The wind speeds of that tornado were not measured.

Since 1942 on the Hanford Reservation, environmental measurements and comprehensive evaluation programs (treated in detail in ERDA-1538) in radiation biology, ecology, hydrology and meteorology have been maintained to measure the impact of facility operations on the environment. These program results are the basis for the projected environmental impacts described in this report.

Anticipated program benefits include electric power production and defense and research uses of N Reactor plutonium. N Reactor steam is converted to 4.5 billion kWhrs electricity each year by the Hanford Generating Project, an energy equivalent of 8,000,000 barrels of oil each year. The use of plutonium fuel to develop technology for reactor operations and for use in breeder reactor programs is an important benefit. The defense uses contribute to maintaining the security of the United States. The byproduct steam benefits include electrical production and reduction of heat discharged to the Columbia River.

In addition to the information presented in this report, supplemental data on facility descriptions, photographs, processing schematics and other supporting information are contained in Section II of ERDA-1538.

I.2 ENVIRONMENTAL EFFECTS

I.2.1 Routine Operations

Environmental data, calculations and analyses show no significant adverse radiological or nonradiological impacts from current or projected future operations resulting from N Reactor, Fuels Fabrication and Spent Fuel Storage Facilities.

Nonoccupational radiation exposures resulting from 1978 N Reactor operations are summarized and compared to DOE exposure limits ~~on the following page~~ in table S-1.

~~Besides the DOE limits, population dose limits from the Nuclear Regulatory Commission (NRC) and the Environmental Protection Agency (EPA) also exist for normal operation. The NRC design objective for an annual dose commitment to the total body is 5 mrem and 15 mrem to any organ of an individual in an unrestricted area. The EPA limits the annual dose equivalent to 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ of any member of the public. The NRC limits apply to the annual release from a light-water-cooled nuclear power reactor. The EPA limits apply to the annual release from a reactor's entire uranium fuel cycle operation.~~

*(DOE limits are only applicable limits $\Rightarrow$
 $\therefore$ don't need to discuss EPA or NRC.)

-
- (a) 10 CFR Part 50, Appendix I, Nuclear Regulatory Commission, January, 1980.
 - (b) 40 CFR Part 190, Subpart B, paragraph 190.10, published January 13, 1977, effective date December 1, 1979 (except for milling of uranium ore - December 1, 1980).

ANNUAL POPULATION DOSE COMMITMENT FROM N REACTOR OPERATIONS(a)

<u>Exposure Category</u>	<u>Calculated Exposures (mrem)</u>	<u>DOE(b) Limits (mrem)</u>
<u>WHOLE BODY</u>		
Max. Individual:(c)	0.16	500
Ave. Individual:(d)	0.006	170
<u>THYROID</u>		
Max. Individual:	0.36	1500
Ave. Individual:	0.013	500
<u>NATURAL BACKGROUND</u>		
All Organs:	100.0	NA(e)

<u>Exposure Source</u>	<u>50 Year Population Dose Commitment From 1978 N Reactor Operations to 236,000 People (Units in Man-Rem)</u>	
N Reactor Operations	Whole Body = 1.5	Thyroid = 4.7
Natural Background	Whole Body = 24,000	Thyroid = 24,000

-
- (a) Based upon 1978 data.
 (b) ERDAM, Chapter 0524.
 (c) Maximum Individual - A hypothetical member of the population assumed to reside 5 miles northwest of N Reactor, drinks all water from the Columbia River, drinks 275 liters of milk in 9 months from cows eating pasture grass near residence; eats 710 kg of produce grown near residence and irrigated with Columbia River water, eats 40 kg of fish per year caught from the Columbia River, and spends 500 hours per year on the river shoreline, 100 hours per year swimming and 100 hours per year boating.
 (d) Average Individual - Member of the population in Richland, WA., drinking Columbia River water, with more moderate fish and irrigated food consumption, and river use habits. See ERDA-1538.
 (e) Not applicable.

Since the nonoccupational doses resulting from present and future operations of N Reactor are: (1) such a small fraction of the most restrictive documented radiation protection standards, and (2) only 0.006 percent of the annual whole body dose (in man-rem) commitment and less than 0.02 percent of the thyroid dose commitment contributed by nature, the radiological impact of such operations is judged to be insignificant. Further, when compared to similar nonoccupational doses from other operating nuclear reactors, these population dose commitments are minimal (see Nuclear Safety, Volume 15, No. 1, January, February 1974).

Nonoccupational exposures, resulting from the operation of Fuels Fabrication Facilities in 1978, include: (1) a 50-year whole body population dose commitment of 0.000028 man-rem, (2) 0.000011 mrem to the maximum individual and (3) 0.000000064 mrem to the average individual. Thus, operations of the Fuels Fabrication Facilities do not add appreciable population exposures to the minimal values received from N Reactor operations. Due to the nature of spent fuel storage operations (total submersion and recirculating coolant), no calculable population exposures arise from the 100-K fuel storage facilities. Thus, the total impact, of the current and future reactor and fuels special nuclear materials operations, on population exposure from routine operations is insignificant.

In determining these population radiation exposure values, all significant exposure pathways are evaluated. Pathways include: (1) submersion in airborne effluent plumes, (2) consumption of drinking water from the Columbia River, (3) foods eaten that have been irrigated with Columbia River water, (4) ingestion of radioactive iodines through the cow-to-milk pathway, (5) consumption of fish from the Columbia River and (6) other less significant pathways.

A summary of the environmental assessment program designed to obtain population exposure baseline data and to assess possible changes in terrestrial and aquatic environs on and off the Hanford Reservation is provided in Section II.1.2 of ERDA-1538.

Nonoccupational population exposures to nonradioactive substances, such as chemicals, sewage and heat, have been evaluated, ~~and are determined not to have significant impact on the general population or the environment.~~

The river water temperature increase, in the dual purpose operating mode, is flow dependent and averages about 0.06°F at 120,000 cfs, the annual average flow. The water temperature could increase by 0.2°F should the river flow drop to a minimum flow of 36,000 cfs. Such temperature increases do not result in ~~significant~~ adverse environmental effects, and are less than allowable ΔT limits set by the State of Washington.

I.2.2 Postulated Accidents

Credible, but postulated, accidents within and between 100-K, 100-N and the 300 Area have been analyzed for radioactive and/or nonradioactive materials. The radiological accidents analyzed include: (1) failed fuel cladding, (2) reactor coolant pipe breaks, (3) waste management line or tank failure, (4) uranium fires, and (5) spent fuel cask and shipment accidents. Nonradiological accidents analyzed include: (1) postulated oil tank leaks, and (2) pipe breaks that would release stored process acids to the environs.

The most severe of the postulated accidents are those involving radiological events. Those accidents that would result in the largest population exposures are summarized on the next page, and are compared with DOE dose standards for normal operation.

COMPARATIVE EVALUATION OF ACCIDENTS AND STANDARDS

<u>Exposure(a) Source</u>	<u>Exposure(b) Category (Organ)</u>	<u>Max. Ind.(c) Dose (mrem)</u>	<u>DOE(d) Limit (mrem)</u>
1. Reactor large pipe break	Whole body Thyroid	0.9 370.0	500 1500
2. Cladding failure	Whole body Thyroid	0.0021 0.054	500 1500
3. Fuel storage cask drop	Whole body Thyroid	0.000007 0.0017	500 1500
4. Fuel shipment accident	Whole body Thyroid	0.00002 0.005	500 1500

(a) See subsections III.1.4 and III.2.4 for assumptions and calculations.

(b) The skin, as an organ, receives approximately the same dose as the whole body.

(c) The maximum individual normally receives 20 times more exposure than the average individual, and lives 5.5 miles from N Reactor.

(d) Manual Chapter 0524 for normal operations.

Only accident number 1 would exceed normal operations in terms of the maximum individual thyroid and whole body exposures. Further, most of the postulated accidents would not exceed the most stringent of population thyroid or whole body dose standards for normal operations; instead they would be a small fraction of such standards. It should be emphasized that the calculated doses for the worst accident (large pipe break) are but a small fraction of the NRC 10 CFR Part 100 guidelines of 25 rem whole body and 300 rem thyroid. Finally, none of these postulated population whole body or thyroid doses exceeds DOE standards for normal operation. ~~Consequently, the environmental impact from postulated credible accidents is judged to be insignificant.~~

I.3 UNAVOIDABLE ADVERSE EFFECTS

Continued operation of the special nuclear materials program has some small but unavoidable adverse effects. Three-tenths of 1 percent of the Columbia River chinook salmon fry passing the 100-N intake annually are killed as a result of drawing water from the Columbia River into the intake. The fry lost are equivalent to the number of fry resulting from two spawning pairs, and is not significant when considering the fishery as a whole.

The 236,000 people within a 50-mile radius of N Reactor receive a total whole body dose of 0.77 man-rem from N Reactor and fuels fabrication operations. However, this exposure rate does not contribute to observable or measurable ill health effects, and is lower than any published, nonoccupational exposure limits or values.

Approximately 5 million gallons of oil are used each year by the combined operations of the N Reactor and WPPSS Hanford Generating Project. However, since N Reactor steam is converted to approximately 4.5 billion kWhrs electricity each year by the Hanford Generating Project, an energy equivalent of 8,000,000 barrels of oil each year is saved.

Thus, while some minor unavoidable adverse effects will continue to occur with extended operation of N Reactor and Fuels Fabrication Facilities, no significant adverse effects are evident. Further, no other effects have been found nor has there been any notable changes in effects described in ERDA-1538.

I.4 ALTERNATIVES

The only alternatives considered in this environmental assessment for N Reactor are:

1. to continue operations in the current mode, producing plutonium-239 with 12% plutonium-240 and simultaneously producing steam for WPPSS to generate electricity;

2. to change the plutonium-240 assay from 12% to 6%; or

Proposed
Action

3. to shut down N Reactor.

The analyses of the alternatives in this assessment update the information discussed in ERDA-1538. All of the alternatives have minimal environmental impact. It should be pointed out, however, that because of the surveillance and maintenance activities, N Reactor in a shutdown mode would still have a minor impact on the environment.

The alternatives considered for the Fuels Fabrication Facilities are:

1. to continue producing fuel,
2. to increase production of fuel should the plutonium-240 assay change from 12% to 6%, or || Proposed Action
3. not produce fuel if N Reactor is shut down.

No observable effects on the environment are expected from any of these || alternatives.

II. DESCRIPTION OF SITE, PROGRAM AND FACILITIES

II.1 CHARACTERIZATION OF EXISTING ENVIRONMENT^(a)

II.1.1 Hanford Reservation

In early 1943, the U.S. Army Corps of Engineers selected the Hanford Site as the location for reactor and chemical separation facilities for the production and purification of plutonium for use in nuclear weapons (Manhattan Project). The semi-arid Hanford site occupies about 570 square miles of the southeastern part of the state of Washington (Figure II.1). A total of eight graphite-moderated reactors, using the Columbia River water for once-through cooling, and a new dual-purpose reactor (N Reactor), using a recirculating water coolant, were eventually built along the Columbia River.

Currently, N Reactor is the only plutonium production reactor at Hanford in operation. The by-product steam is used for electrical generation by the adjacent Hanford Generating Project, owned and operated by the Washington Public Power Supply System (WPPSS). The N Reactor facility is located along the Columbia River in the 100-N Area (see Figure II.4). The 300 Area, just north of Richland, is the location of the reactor fuel fabricating facilities.

II.1.2 Demography and Land Use

The population within a 50-mile radius of the Hanford Site varies from about 227,000 for a radius center at 100-N Area to about 164,000 for a radius center at the 300 Area.^(b) Land uses in the surrounding area include urban, commercial, industrial, and irrigated and dry-land farming. Of the irrigated crops, alfalfa hay uses 30% of the total area, wheat 15%, and sugar beets about 8%. Water removal rights, other than Hanford's, amount to about 800,000 acre-ft/yr within 50 miles downstream of the N Reactor, from an annual average river flow of about 120,000 cfs, or about 87,000,000 acre-ft/yr.

(a) For additional details see Section II.3 and Appendix II.3 of ERDA-1538.

(b) For conservatism in dose calculations and discussion, this figure is rounded up to 250,000.

FIGURE II.1. Location of DOE Hanford Reservation Showing 50-mile Radius from N Reactor

II.1.3 Historical and Archaeological Sites

Nine sites listed in the National Register of Historic Places are located on the Hanford Reservation. The remains of many Indian campsites and fishing grounds are found along the Columbia River within the Hanford Reservation boundary. Applications for listing two of these archeological sites on the Hanford Reservation have been submitted to the National Register of Historic Places. None of these sites are disturbed during operation of the 100 and 300 Area special nuclear materials facilities.

II.1.4 Hydrology

The surface water and groundwater hydrology of the Hanford area is described in Section II.3.8 and Appendix II.3-D of ERDA-1538. Descriptions of Columbia River flow, temperature and water quality characteristics are included in those sections. A system of test wells is in use throughout the Reservation for long-term monitoring of groundwater in both the confined and the unconfined aquifers. A map of these wells is presented in Appendix II.3-D of ERDA-1538.

The U.S. Army Corps of Engineers has postulated a probable maximum flood (PMF) for the Columbia River basin, based on a combination of most severe climatic conditions.^(4,5) Contributing factors of winter snow accumulations, spring melting and run-off season rainstorms were maximized for the study. The projected flow for the PMF is 1,440,000 cfs at the Hanford site. The PMF would reach an elevation of 423 ft MSL in 100-N and 100-K Areas and 383 ft MSL in the 300 Area. A seismic induced flood would not exceed the PMF.⁽⁶⁾

Neither 100-N Area or 300 Area would be located in the floodplain of the 100 year flood. Since the areas are not located in the floodplain, the floodplain analysis set forth in 10 CFR 1022 is not required.

II.1.5 Ecology

The Hanford Reservation is characterized as a shrub-steppe area, with sparse vegetation dominated by sagebrush, bitterbrush and cheatgrass. There are no significant tree growth areas other than a few willows near surface waters.

The anadromous salmon and the steelhead trout are the fish of economic importance in the Columbia River. About 9,500 fall Chinook salmon spawn in the Hanford reach of the river. The mule deer is the only big game mammal, and the cottontail rabbit is the most abundant small game mammal on the Reservation. Small mammals are abundant, particularly the Great Basin pocket mouse. The chukar partridge is the most important gamebird. Hunting is not permitted on the major portion of the Reservation. Controlled hunting, however, is allowed on State Department of Game managed land in the northeastern corner of the Reservation.

II:1.6 Meteorology

The Hanford Reservation is characterized as mild and dry with occasional periods of high wind. The average maximum and minimum temperatures occur in July and January at 91.8⁰F and 36.7⁰F, respectively. The average annual precipitation is 6.28 in.

Washington State averages about one tornado each year. Thirteen tornadoes have been recorded within a 100-mile radius of the Hanford Meteorological Station (HMS) since 1916. These were of relatively short duration, reaching ground level for only brief periods and causing only slight damage. One small tornado (June 16, 1948) and two funnel clouds have been observed along the Rattlesnake Mountain in 33 years (1945-1977) of Hanford observations. Only one small tornado has been sighted on the Hanford Reservation about 18 miles from the 100 and 300 Areas. This tornado caused no damage.

Several studies have been made of the maximum wind speed and the probability of a tornado striking a specific location on the Hanford Reservation. The maximum tornado wind speed for the southern portion of the Hanford Reservation is estimated to be 175 mph.⁽¹⁾ The 175 mph design basis tornado for the Hanford Reservation was selected in 1974 and reconfirmed in 1976 based on two studies performed by Dr. T. Theodore Fujita of the University of Chicago, a recognized authority on the study of tornado characteristics using DAPPLE techniques. The conclusions reached in both studies confirm that: (1) tornado alleys exist in the northwest, (2) the 100 and 300 Areas are situated between and well outside

the alleys, and (3) a maximum wind speed of 175 mph is a conservative design basis.^(7,8)

The conservatism in the selected design basis tornado is supported further by the Fujita observation that all northwest tornadoes from 1950 to the present have occurred within established tornado alleys. At a probability of occurrence of 10^{-7} within the Walla Walla alley, which is the nearest alley to the Hanford Reservation, the maximum wind speed would be about 175 mph. No tornado has been observed within 18 miles of N Reactor and the nearest tornado that caused even modest damage was 60 miles from 100-N and 100-K, and about 35 miles from the 300 Area.

Wind roses are included in ERDA-1538 for both the HMS (located near the center of the Reservation) and the N Reactor meteorological station. Based upon 1 year of meteorological station data at N Reactor, the wind at the two locations is similar except for less frequent calms and a more prevalent westerly wind at N Reactor. Diffusion characteristics for the 300 Area were derived from meteorological data gathered at the Fast Flux Test Facility (FFTF) meteorological station.

A peak wind speed of 80 mph at an elevation of 50 ft was recorded at the HMS on January 11, 1972. This is the highest wind speed ever recorded at Hanford. Winds of this magnitude are expected to have a recurrence interval of 30 years, and a gust of 96 mph is not expected to occur more than once each 500 years.

II.1.7 Geology and Seismicity

Eastern Washington is dominated by the Columbia Basin geologic province, which encompasses about 50,000 square miles. It is underlain by the vast field of lava flows of the Columbia River basalt group. Within the Columbia Basin are sub-basins that reflect the history of the region. The Hanford Reservation is situated within one of these, the Pasco Basin. The known geologic history begins in Miocene time with the extrusions of the Columbia River Basalts. Older rocks do not crop out in the Pasco Basin area. One deep hole drilled on the Hanford Reservation encountered Eocene age sediments and possible older basalts.

After cessation of the basalt flows, Ringold Formation sediments were deposited in Pliocene-Pleistocene time. The major development of long anticlinal ridges and associated faulting appears to have preceded the Ringold Formation. In Pleistocene time, the ancestral Columbia River was fed by glacial melt waters, and catastrophic floods occurred from glacier-dammed lakes. These events resulted in the deposition of alluvial deposits which covered most of the Pasco Basin and shaped much of the local topography. In more recent time, windblown silt and sand deposits have become dominant modifiers of the landscape.

The 100-K, N and 300 Areas special nuclear materials facilities are located on flat ground adjacent to the Columbia River in the north half of the Pasco Basin, a structural unit of the much larger Columbia Basin geologic province. Underlying these areas are an average of 50 feet of Quaternary glacio-fluvial sands and gravels, 470 feet of Pliocene-Pleistocene sediments, which are predominantly silt and clay but range to sand and gravel with cobbles, and more than 475 feet of interbedded Miocene-Pliocene basalt flows and sediments. All stratigraphic horizons are nearly flat and undeformed by folds or faults. The nearest known faults are 5 to 10 miles from N Reactor, the closest facility, and are not active.

Foundations of the deeply-founded plant structures are supported in very dense sands and gravels. Foundations of shallower structures are supported in compacted granular backfill adjacent to the deeply-founded structures. Both the sands and gravels and compacted backfill are competent soils and will adequately support conventional footing or mat foundations. For the deeply-founded structures, bearing capacities are not the controlling factor in the design of foundations. Instead, the differential settlements which the structures may undergo are of primary consideration.

The foundation soils are not susceptible to loss of strength, subsidence or liquefaction resulting from motions associated with the Safe Shutdown Earthquake. Also, existing slopes along the Columbia River in the vicinity of the special nuclear materials fabrication facilities have adequate stability against failure both under static and dynamic loading conditions that could be produced by postulated earthquakes.

On the basis of damage experienced since 1840, the U.S. Coast and Geodetic Survey, in 1955 and again in 1958, designated eastern Washington, as well as adjacent lands in Oregon and Idaho, as seismic Zone 2. See Figure II.2A for classification system and locations in accordance with the Uniform Building Code (UBC) publication. See Figure II.2B for approximate relationship between earthquake magnitude, intensity, acceleration and energy release.

N Reactor and 300 Area facilities were designed in accordance with seismic criteria equal to or greater than the UBC (1955 and 1958 Editions). The facilities are located in the UBC seismic Zone 2. The maximum earthquake for seismic Zone 2 has an intensity of MM-VII. The corresponding ground accelerations are shown in Figure II.2B.

N Reactor systems vital to nuclear safety were designed to withstand a ground acceleration of 0.20 g. Other systems critical to recirculating cooling were designed to withstand a ground acceleration of 0.10 g. The balance of the plant was designed to the minimum requirements of UBC seismic Zone 2.

The concepts of a Safe Shutdown Earthquake (SSE) and an Operating Basis Earthquake (OBE) for nuclear reactors were developed after N Reactor was constructed. Because of differences between the criteria for N Reactor design and those currently specified for a SSE, N Reactor's 0.20 g* design for nuclear safety systems provides protection equivalent to the current specified SEE of 0.25 g for the Hanford site. N Reactor design criteria limited the total combined stress to normal design stress limits whereas the current criteria for the SSE permits the total combined stresses to approach the yield strength of the material. Also included in the N Reactor design is a seismic trip set to shut the reactor down at an intensity of MM-IV or a ground acceleration of 0.015 g.

These design criteria were very conservative when compared to a maximum estimated earthquake intensity at Hanford of MM-IV to MM-V corresponding to a

* All g values are horizontal ground acceleration.

FIGURE II.2A. Seismic Risk Map

ROSSI-FOREL INTENSITY SCALE (0-9)	MODIFIED-MERCALLI INTENSITY SCALE (1930), WOOD AND NEUMANN MM I	Richter MAGNITUDE M	ENERGY E	GROUND ACCELERATION a	UBC SEISMIC ZONES
COL. 1	COL. 2	COL. 1	COL. 2	COL. 3	
	I Detected only by sensitive instruments.		E_{erg}	$\frac{cm}{sec^2}$ $\frac{g}{g}$	
I The shock felt only by experienced observer under very favorable conditions.	II Felt by a few persons at rest, especially on upper floors; delicate suspended objects may swing.	M=3	10^{10}	2	0.0025g
II Felt by a few people at rest; recorded by several seismographs.	III Felt noticeably indoors, but not always recognized as a quake; standing autos rock slightly; vibration like passing truck.		10^{11}	3	
III Felt by several people at rest; strong enough for the duration or direction to be appreciable	IV Felt indoors by many, outdoors by a few; at night some awakened; dishes, windows, doors disturbed; motor cars rock noticeably.	M=4	10^{12}	4	
IV Felt by several people in motion; disturbance of movable objects, cracking of floors.	V Felt by most people; some breakage of dishes, windows, and plaster; disturbance of tall objects.		10^{13}	5	0.005g
V Felt generally by everyone; disturbances of furniture, ringing of some bells.	VI Felt by all; many frightened and run outdoors; falling plaster and chimneys; damage small.	M=5	10^{14}	6	
VI General awakening of those asleep, ringing of bells, swinging chandeliers, startled people run outdoors.	VII Everybody runs outdoors; damage to buildings varies, depending on quality of construction; noticed by drivers of autos.		10^{15}	7	
VII Overthrow of movable objects, fall of plaster, ringing of bells, panic with great damage to buildings.	VIII Panel walls thrown out of frames; fall of walls, monuments, chimneys; sand and mud ejected; drivers of autos disturbed.	M=6	10^{16}	8	
VIII Fall of chimneys; cracks in walls of buildings.	IX Buildings shifted off foundations, cracked, thrown out of plumb; ground cracked; underground pipes broken.		10^{17}	9	
IX Partial or total destruction of some buildings.	X Most masonry and frame structures destroyed; ground cracked; rails bent; landslides.	M=7	10^{18}	10	0.25g
X Great disasters, ruins, disturbance of strata, fissures, rockfalls, landslides, etc.	XI New structures remain standing; bridges destroyed; fissures in ground; pipes broken; landslides; rails bent.		10^{19}	200	
	XII Damage total; waves seen on ground surface; lines of sight and level distorted; objects thrown up into air.	M=8	10^{20}	300	
			10^{21}	400	
			10^{22}	500	0.5g
			10^{23}	600	
			10^{24}	700	
			10^{25}	800	
			10^{26}	900	
			10^{27}	1000	1g
			10^{28}	2000	
			10^{29}	3000	
			10^{30}	4000	3g

FIGURE II.2B. APPROXIMATE RELATIONSHIP
BETWEEN EARTHQUAKE MAGNITUDE, INTENSITY,
ACCELERATION AND ENERGY RELEASE.

ground acceleration of 0.015 g to 0.03 g for the Milton-Freewater earthquake of July 15, 1936, or when compared to the maximum historical earthquake for this region of MM-VI corresponding to a ground acceleration of 0.06 g.

Eastern Washington is a region of low to moderate seismicity that lies between the active seismic zones of western Washington and western Montana. A number of earthquakes of intensity MM-VI or greater have occurred in western Washington. The strongest shock of historic record to occur in western Washington was the 1949 Olympia-Steilacoom earthquake originating in the Puget Sound channel about 150 miles from Hanford. The earthquake was rated MM-VIII, and intensities from MM-III to MM-VII were experienced at distances of 150 miles. In Richland and the Hanford area the shock was felt only slightly.

The great Alaska earthquake of 1964 and recent shocks in western Washington and in Montana were not felt as strongly at Hanford as in surrounding localities. Eight shocks in western Washington between 1920 and 1965 were reported as having maximum intensities of MM VII or greater. One of the strongest shocks to affect the Pacific Northwest occurred north of Lake Chelan on December 14, 1872, which could have produced an intensity of around MM-V in the Hanford area.

Eastern Washington and western Oregon earthquakes occurring in historic times have neither been as intense nor as frequent as those in western Washington. The largest historical earthquakes in the region occurred in 1893 and 1936 with intensities estimated from old newspaper accounts in the range of Modified Mercalli MM-IV to MM-VI and located 50 to 85 miles from N Reactor.

The 1918 Corfu earthquake, located about 12 miles north of N Reactor, may have reached MM-IV. Conservative estimates of the intensity of these quakes at N Reactor are in the range of MM-IV to MM-V. Recorded data on earthquakes of significance are limited to the 1936 Milton-Freewater quake with a magnitude of 5.8 (MM-VII) and located 75 miles southeast of N Reactor and 50 miles from the 300 Area and the 1971 Hanford quake with a magnitude of 3.2 (MM-III) and located 2.3 miles northeast from N Reactor. Since the startup of N Reactor in 1964, the reactor has not been shut down by the seismic trip system that is set at MM-IV.

Figure II.3 shows the active fault zones in Washington deduced from earthquake activity. No clear-cut relationships of epicenters to specific surface faults or structures are yet recognized in eastern Washington. (The few experienced earthquakes and the absence of detailed studies on the large-scale geologic features of the region preclude a comprehensive tectonic model of eastern Washington). A low rate of tectonic deformation in eastern Washington is indicated for probably more than 10 million years, and the deformations are much less severe than those in the active seismic zone of western Washington.

The site considered to be the largest potential earthquake generator near Hanford is located on the postulated Rattlesnake-Wallula fault at the southeast end of the Rattlesnake Hills. Field investigations and detailed literature reviews indicate that there are no significant faults closer to the site. If an

FIGURE II.3. Earthquake Zones of Washington

earthquake comparable to the 1893 Umatilla quake (MM-VII), which is about 54 miles from the 100-N Area, occurred at this location, ground acceleration of less than 0.13 g at the 100-N Area would be expected. This is consistent with the Zone 2 requirements.

A safe shutdown earthquake of 0.25 g maximum horizontal ground acceleration accompanied by a vertical acceleration of two-thirds the horizontal is assumed at Hanford for designing nuclear reactors and facilities that contain plutonium. A design basis of 0.25 g on the Hanford Reservation allows for an MM-VIII intensity earthquake epicentered at the same location, larger than any historical earthquake known to have occurred in eastern Oregon or Washington. An MM-VIII earthquake is consistent with Zone 3 of the seismic probability map, not Zone 2 as established for this area. The maximum earthquake intensity experienced in the Pasco Basin during the past 100 years is MM-IV to MM-VI corresponding to ground accelerations of 0.015 g to 0.06 g. The geology and seismology of the Hanford region has been thoroughly studied and is well documented. The Safety Analysis Reports for the Fast Flux Test Facility⁽¹⁰⁾, WPPSS Nuclear Project No. 2⁽¹¹⁾, and WPPSS Nuclear Project No. 1/4⁽¹²⁾, exemplify recent reviews of the geology and seismology of the Hanford region. All of these projects are 7 miles from the 300 Area.

A comprehensive study of the 100-N site was performed for the Washington Public Power Supply System (WPPSS).^(13,14,15) The WNP-1 project was originally planned to be located at 100-N Area, but was subsequently relocated to a different area on the Hanford Reservation.

In 1967, at the request of the AEC, the U.S. Coast and Geodetic Survey published an analysis of the seismicity of the Hanford area.⁽⁹⁾ The report concluded the maximum horizontal ground acceleration^(a) of 0.12 g on glacial-fluvial deposits at the Hanford site would be adequate for representing earthquake

(a) Ground accelerations reported in this document are the equivalent of free-field, surface-ground accelerations.

disturbances likely to occur within the lifetime of N Reactor, and that a maximum horizontal ground acceleration of 0.25 g would represent the maximum earthquake likely to affect this site. These values correspond to the current concept of an OBE of 0.12 g and a SSE of 0.25 g. A recent safety analysis report published in 1975 calculated a design basis earthquake for the FFTF located 20 miles southeast of N Reactor and 7 miles north of the 300 area to be a ground acceleration of 0.25 g.⁽¹⁰⁾

Though the SSE of 0.25 g is possible, it is judged to be highly improbable. However, because of the general acceptance of the more conservative OBE of 0.12 g and a SSE of 0.25 g, rather than the more plausible 0.06 g and 0.12 g, for the design of special nuclear materials reactors and plutonium facilities at Hanford, 0.25 g was used in evaluating those structures important to N Reactor's nuclear safety systems.

It is concluded that the 100-N site is suitable for a nuclear power plant in terms of the proposed Appendix A to 10 CFR Part 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants", and that the design basis for vibratory ground motion (Safe Shutdown Earthquake) should be 0.25 g without the need to design for surface faultings. It is also concluded that subsurface materials adequately provide vertical and lateral stability to all structures.

It is concluded that the 100-K site is suitable for special nuclear materials since the spent fuel basin has also been shown capable of withstanding the 0.25 g SSE earthquake.

It is concluded that the 300 Area fuel fabrication facilities, which were designed to UBC seismic Zone 2, are adequate for uranium fuel fabrication purposes.

It is also concluded that should an SSE occur, N Reactor, 100-K and the Fuels Fabrication Facilities would have a minimal environmental impact from spills and leaks of nonradioactive materials. There would be no release of radioactivity.

II.2 DESCRIPTION OF PROGRAMS

II.2.1 Operational Relationships

The N Reactor is currently operating to produce plutonium for DOE research and defense commitments. During this operation the excess steam is provided to the Washington Public Power Supply System (WPPSS) for the production of electricity. The N Reactor depends upon the Fuels Fabrication Facilities located in the 300 Area to provide the fuel needed for the reactor operations. This fuel is loaded into the reactor and irradiated to produce plutonium containing 12% ^{240}Pu . Following irradiation the fuel is discharged into large water-filled basins to cool and remain in storage to await processing. Annual discharge is about 315 MTU containing approximately 615 kg of plutonium (1950 g Pu/MTU).

The continued operation of N Reactor will impact on the waste management operations of Hanford. The impact on these operations has been evaluated and is discussed in Sections III and V of ERDA-1538.

For the production of plutonium containing 6% ^{240}Pu , annual reactor discharge is about 715 metric tons of uranium (MTU) containing approximately 700 kg of plutonium (985 g Pu/MTU).

	Discharge MTU	Pu (kg)
6%	715	700
12%	315	615

II.2.2 Anticipated Benefits

II.2.2.1 N Reactor Operations

Only two locations currently exist in the U.S. where plutonium of the necessary weapons-grade quality can be produced: Hanford and the Savannah River Plant near Aiken, South Carolina.

N Reactor is the only Hanford plutonium producing reactor remaining in operation. An assessment of the current and future requirements for defense material is beyond the scope of this report. Although the benefits are not assessed in this document, they contribute to maintaining the security of the United States.

Benefits are anticipated to be derived from the research programs that use the plutonium produced by N Reactor. The research uses of this product include the entire spectrum of life sciences programs that study and evaluate the significance of plutonium in man's environment. The use of plutonium in the developing technology of reactor operation and in breeder reactor programs is a prime anticipated benefit.

As a by-product of N Reactor operation, waste heat is used to generate steam that is sold to the WPPSS Hanford Generating Project for conversion to electricity. This by-product steam generates 4.5 billion kWhr/yr of electricity. Eight million barrels of oil would be required to produce this amount of electricity by an oil-fired plant.

This electrical energy is fed into the Bonneville Power Administration (BPA) grid system serving Montana, Idaho, Utah, Wyoming, Washington, Oregon and northern California. The electrical power generated by N Reactor steam is of particular benefit during the winter and early spring months when water supplies for hydroelectric plants in the Pacific Northwest are low and the demand for electricity is high.

Table II.1 summarizes the benefits of continued operation of N Reactor.

TABLE II.1. Anticipated Benefits of
N Reactor Operations

<u>Item</u>	<u>Benefit</u>
Plutonium Production	600-700 kg/yr ^(a)
Electrical Power Generation	4.5 billion kWhr/yr (equivalent to 8 million barrels of oil)

^(a)Depends upon fuel exposure.

II.2.2.2 Fuels Fabrication Facilities

The Fuels Fabrication Facilities manufacture the uranium metal fuel for the N Reactor that in turn produces the plutonium for use in research and national defense programs. The direct benefit to the reactor program by the continued operation of these facilities is the production of 300 to 700 tons of uranium metal fuel per year. This contributes to the anticipated benefits of the N Reactor program discussed in Section II.2.2.1.

II.2.3 Safeguards

The discussions on safeguards and special precautions apply not only to individual facilities but also to the Hanford operations in general. Each prime contractor, in coordination with DOE, is responsible for industrial security. Security programs include:

- nuclear materials management
- equipment safeguards

- employee access
- document control
- emergency procedures

A special measurement and record system is maintained to protect against undetected losses of special nuclear materials and to provide current, accurate information on the disposition and availability of all such materials received, processed, produced, stored, or shipped. Areas containing special nuclear materials are locked and controlled by authorized personnel. Surveillance is maintained by trained guards who also verify that designated "material custodians" are cleared and authorized to move the materials.

The 100 areas are located north of a manned check point on the Hanford Reservation. Each major DOE facility on the Reservation is provided with perimeter chain link fences with locked or guarded entries. Certain buildings, gates and rooms within these fenced areas are also locked and/or sealed to assure strict access accountability. The level of guard service provided depends on the activity status in controlled areas. In the 300 Area, which is fenced and has guarded entries, a magnetic key card building entry device is also employed as an alternate to the use of controlled keys for access to the Fuels Fabrication Facilities. Entry to locked deactivated areas and buildings requires an approved security badge, an entry permit and an escort.

Pre-employment investigations are conducted to clear an individual for work. Color-coded photo badges indicate the type of security clearance and a numbering system identifies area access for permanent employees.

Access to classified documents is limited by a need-to-know clearance and by established procedures governing the preparation, transmission, accounting, reproduction, storage and destruction of the documents. Documents that are not constantly attended by authorized personnel are stored in special locked repositories.

Each person entering DOE facilities is required to produce an appropriate badge. As a condition of access to protected areas, articles carried by persons and in vehicles are subject to inspection by a guard. Each employee receives an orientation and periodic training designed to motivate and maintain employee responsiveness to today's industrial security climate.

Local area guard forces are maintained at levels sufficient to provide immediate response to attempted theft, sabotage or emergency conditions and to assist the Emergency Director should evacuation of the affected area be necessary.

All personnel and vehicles entering and leaving the Reservation and the protected areas are subject to search. Business visitors are admitted only if they carry proper security clearance and identification and are accompanied by suitable official escort. Records are kept of all visitors.

Within all plants and facilities, access to special nuclear material (SNM) that has been adequately separated from other radioactive elements is under the direct control of a responsible assigned individual and is strictly limited to duly authorized persons. Rigorous administrative procedures, records and redundant control routines are designed to preclude the probability of accidental removal and/or deliberate diversion of SNM.

Visual inspection of people entering and leaving the protected areas and of the possessions and vehicles of such persons is done to detect and prevent deliberate theft and accidental diversion of SNM and to ensure that weapons or other prohibited articles are not brought into the areas.

Instruments and alarms of various types are strategically located to detect deliberate unauthorized attempts to intrude in vaults, areas or facilities or to gain access to SNM. Random checks of SNM storage locations are made on a 24-hr/day basis by roving security guards using appropriate surveillance techniques.

Armed guards protect against overt hostile action, physical violence and forced entry. The guard force is trained and equipped for immediate response to prevent or contain such a situation and to reestablish control without loss of SNM.

Security guards also provide escort and protection during the movement of SNM outside of secured facilities in the course of transferring these materials from one location to another on the Hanford Reservation.

II.3 DESCRIPTION OF FACILITIES

II.3.1 N Reactor and Facilities

II.3.1.1 General Description

The N Reactor and facilities occupy two adjoining areas known as the 100-N Area and the 100-K Area (shown in Figures II.4 through II.6). These areas are served by the Hanford railroad which connects to the south with the Burlington Northern Railroad and to the north with the Milwaukee Railroad.

The Hanford highway system serving the areas connects to State Highway 240, which crosses the western part of the Reservation from north to south. All Hanford roads except for State Highways 240, 243, and 24 are restricted, i.e., not open to the public.

The Columbia River flows past the N Reactor 380 miles above the mouth of the river. This area could be served by river navigation; however, the river channel is not maintained for such navigation and is not equipped with navigational aids in this immediate area. The water level is approximately 377 ft above mean sea level (MSL) at the minimum river flow of 36,000 cfs and 387 ft above MSL at the annual average flow of 120,000 cfs (Figure II.7). The Federal License for Priest Rapids Dam requires a minimum release rate of 36,000 cfs. The river bottom contours with elevations for river mile 380 are shown in Figure II.8. Velocity profiles near N Reactor at the minimum river flow are shown in Figure II.9. At the annual average flow of 120,000 cfs, the shape of the profiles would remain about the same, but midstream surface velocities would be increased by 5 to 7 fps.

FIGURE II.4. Map of Hanford Reservation

FIGURE II.5. 100-N Area Layout

FIGURE II.6. 100-K Area Plant Layout

FIGURE II.7. Elevation Versus Flow at Columbia River Mile 380.0

FIGURE II.8. Columbia River Bottom Contours in the Vicinity of N Reactor

FIGURE II.9. Velocity Profiles at Low River Flows in the Vicinity of N Reactor

The predominant features of the 100-N Area are the 105-N and 109-N buildings, which are 80 and 87 ft tall, respectively, with the operating floor level at 454.5 ft above MSL. These buildings are a combination of steel-reinforced concrete, concrete block and structural steel framing with insulated metal panels. The buildings are painted to harmonize with the surrounding landscape.

The predominant, tall structures in the 100-N Area are the 200-ft-high 116-N reactor ventilation stack, the 184-N powerhouse stack extending 70 ft above the building roof, and the 110-ft-high freestanding powerhouse annex stack.

Other significant features include the water and oil storage tanks, the steam transmission pipe lines to the adjacent WPPSS Hanford Generating Project and the spherical chemical waste storage tank, which lies behind an earthen retention and shielding dike.

The 102-in. line, carrying condenser cooling water, discharges to the middle of the Columbia River about 18 ft below the river surface. Springs along the river bank, occurring at about the river surface level, discharge seepage from the 1301-N crib and trench some 1,000 ft downstream from the 102-in. line.

The 105-N building that houses the reactor is the center of activity. The heat generated in the reactor is transported by the primary coolant to the 109-N heat exchanger building where the heat generates steam in the secondary coolant system. The steam is transferred to the WPPSS Hanford Generating Project through large overhead pipelines.

The 181-N river pump house building furnishes all of the area water needs, particularly the condenser cooling water for steam turbine drives and for condensation of excess steam production. The 183-N and 163-N buildings produce filtered and demineralized water respectively. The 182-N building houses various process and Emergency Core Cooling System (ECCS) pumps.

The 184-N plant service boiler house and its annex provide startup and backup steam power and about one-half of the plant electrical requirements.

The 117-N filter building and 116-N ventilation stack provide filtration and an exhaust system for the 105-N building ventilation air and confinement zones that contain airborne radioactive material.

The 1301-N crib and trench provide effective soil column treatment for the radioactive liquid wastes discharged from the reactor except for some very minor streams that discharge directly to the river and the chemical wastes that are held in the 1310-N spherical tank. The 1301-N emergency disposal basin, which has never been used, also provides soil column treatment for possible disposal of large quantities of reactor coolant.

The KE fuel storage facility is an integral part of the 105-KE reactor building and is approximately 2.3 miles upstream from 100-N Area (shown in Figures II.5 and II.6). The 105-KE reactor is a special nuclear materials production reactor that was shut down in January 1971 and is currently in retired status.

The fuel storage facility attached to the 105 KE-reactor provided shielding and coolant for fuel that was irradiated in the reactor during its many years of operation. This fuel storage facility has been modified to provide for safe storage of approximately 1,900 tons of irradiated N Reactor fuel.

The fuel storage facility consists of a reinforced concrete basin 125 ft long, 67 ft wide and 21 ft deep. The basin is filled with water used to cool the irradiated fuel and shield personnel working in the facility. The water level is maintained to provide a minimum of 10 ft of water over the fuel during all normal operations.

The KW fuel storage facility is identical to the KE fuel storage facility. The KW facility is an integral part of the 105-KW reactor building. The 105-KW reactor is a special nuclear materials production reactor that was shut down in February 1970 and is currently in retired status.

At each fuel storage facility the storage basin water is part of a closed, recirculating cooling system using water-to-water heat exchangers. Filters and ion exchangers are used to maintain basin water clarity and to control contamination levels. Expended filters and spent ion exchange resins are disposed of at the 200 Area waste storage facility.

II.3.1.2 Operational Programs

N Reactor is operated to produce plutonium with a 12% ^{240}Pu content. | State new
Proposed Action

II.3.1.3 Operational Systems

The N Reactor is a graphite-moderated, light-water-cooled, pressurized dual purpose reactor with a rated capacity of 4,000 megawatts thermal (MWt). The reactor and the component systems within the reactor building were designed by the General Electric Company and were built by Kaiser Engineers for the Atomic Energy Commission. The heat dissipation plant and auxiliary facilities, housed in separate buildings, were designed by Burns and Roe, Inc., and were also built by Kaiser Engineers.

The reactor core is constructed of interlocking, high-density, graphite bars that support the pressure tubes containing the fuel elements. The graphite provides moderation of the neutrons necessary to achieve a sustainable chain reaction. This core is contained in concrete that serves as part of the shielding to prevent escape of radiation.

Metallic uranium, clad in zirconium alloy, is used to fuel the reactor. The fuel elements are contained in 1,003 horizontal zirconium alloy pressure tubes running from front to rear through the graphite core.

The fuel is cooled by the recirculating primary coolant system which pumps high-purity water through each pressure tube. The coolant transports heat generated by the fission process in the fuel elements to steam generators located in the adjacent heat dissipation building (109-N). Steam produced in the steam generators is used to drive the primary coolant pumps, to produce electricity for inplant use, and to supply the WPPSS Hanford Generating Project.

FIGURE II.10. N Reactor Graphite Stack

Graphite Core

An isometric drawing showing the graphite block stacking arrangement is presented in Figure II.10. The stack's overall dimensions are 39 ft 5 in. long, 33 ft wide and 33 ft 1/2 in. high. Venting provisions, both in the longitudinal and transverse directions, are included in the stacking pattern to allow steam to escape from the primary coolant if a pressure tube fails.

A boundary layer of essentially solid graphite blocks, 20-in. thick in the front and rear and 48-in. thick on the remaining sides of the core, functions as a reflector for neutrons released in the fission process. Without the reflector these neutrons would be lost to the shielding that surrounds the core. The reflector also serves to attenuate the energy of the neutrons so that those escaping the active section of the reactor can be captured by the shielding.

Gas plenums are located between the active zone (that portion containing fuel) and the reflector graphite at the front and rear ends and at both sides of the stack. Helium is used as the normal stack atmosphere and is circulated through the active zone of the reactor from front to rear by blowers located in two gas dryer rooms. The helium gas dries the graphite moderator when necessary, provides a medium for detection of water and air leaks, prevents in-leakage of air by maintaining a slightly positive pressure in the core and assists in controlling moderator temperature by acting as a heat conductor between the graphite and the liquid coolant in the pressure tubes and the graphite coolant system.

The graphite reactor stack is penetrated by 1,072 horizontal, front-to-rear channels on an 8-in. horizontal and a 9-in. vertical lattice spacing; only 1,003 channels contain the zirconium alloy pressure tubes into which fuel elements are loaded. The graphite stack is also penetrated by 87 horizontal, side-to-side channels in a 32-in. horizontal by 36-in. vertical spacing for the control rod system, and by 108 vertical channels in a 32-in. square pattern for the ball safety system. In addition, there are 640 horizontal, side-to-side penetrations for the graphite coolant system.

The graphite cooling system is provided to maintain the graphite temperatures below 1325⁰F in order to limit long-term dimensional changes in the graphite. The 0.75-in. outside diameter zirconium-alloy graphite cooling tubes are located between each pressure tube row vertically (9-in. spacing) and on 16-in. centers horizontally.

Shielding

A thermal shield surrounds the graphite stack and intercepts most of the radiation and heat energy that escapes from the reactor core, thus reducing the heat load on the biological shielding. A 1-in. thick boron steel plate is used as the thermal shield material except where the pressure tubes penetrate the inlet and outlet ends of the reactor. Cast iron blocks, 8 in. thick, are used at the front and rear.

Concrete biological shielding prevents activation of equipment and components immediately adjacent to the reactor. It is composed of high-density concrete 40 to 65 in. thick and is used on all sides and on the top of the reactor; regular concrete is used on the bottom. The biological shield supports the thermal shield, fuel tube assemblies, control rods and ball hoppers. It also acts as the reactor gas atmosphere container.

Control Systems

The 84 horizontal control rods enter the left and right sides of the reactor. Approximately half of the rods are located on each side. The rods are hydraulically (water-oil emulsion) driven and can be extended through the active zone of the reactor. Each rod can be operated individually for reactor control, or all the rods can be "scrammed" into the reactor in less than 1.5 sec for rapid shutdown. The rods contain a neutron-absorbing material (boron carbide) that allows control of the reactor just at, or slightly above or below the critical state (self-sustaining chain reaction) or greatly subcritical when fully inserted.

The ball safety system provides a completely independent nuclear control system to rapidly shut down the reactor if for any reason the rod safety system fails

to function when called upon. The control elements for this system are 3/8-in.-diameter boron carbide balls for neutron absorption. Ball hoppers are embedded in the top biological shield over each of the 107 vertical channels through the reactor. The hopper gates, held closed by dc-powered solenoids, are designed to open within 0.75 sec upon receipt of a ball safety circuit signal and allow the balls to enter the channels by gravity. The ball system by itself has adequate neutron absorption strength to shut down the reactor and hold it subcritical.

The rod system provides automatic shutdown of the reactor upon detection of any abnormality in coolant supply, electric power supply and neutron flux level or rate of increase. The ball system serves to shut down the reactor in the event of excessive seismic activity and also acts as a backup to the rod system should its scram function not perform properly.

Reactor Fueling

The reactor fuel elements are contained in zirconium alloy (Zircaloy-2) pressure tubes that penetrate the graphite stack from front to rear and extend through the thermal and biological shielding where nozzle assemblies connect with the inlet and outlet cooling water manifolds. The tubes have a 2.7-in. inside diameter and a 0.250-in. wall thickness. The tube walls are relatively thick to withstand high temperature and high-pressure primary cooling water and other stresses. The nozzles on each end of the pressure tubes are equipped with a removable end cap to allow passage of fuel elements during charge and discharge operation.

The metallic uranium used in the fuel elements varies in enrichment levels from 0.95 to 1.25% ^{235}U and is contained in fuel elements clad with Zircaloy-2. The elements are arranged in a tube-in-tube configuration which provides three channels of coolant around the fuel: (1) between the pressure tube and the outer fuel tube, (2) between the two fuel tubes and (3) through the hole in the inner fuel tube. Fuel elements vary from 15 to 26 in. in length. (Detailed description of the manufacture of the fuel elements and a photo of an assembled fuel element is given in Section II.3.2.3).

The length of the fuel charge in a pressure tube is approximately 35 ft. Perforated, hollow, carbon steel spacers are used at each end of the fuel charge to position the charge in the active zone of the reactor.

Primary Coolant System

The primary coolant system transports the heated water from the reactor building (105-N) to the adjacent heat dissipation building (109-N). The portion of the system within the reactor building consists of 16 parallel lines, 18 in. in diameter, which conduct the coolant from an inlet coolant manifold to the reactor; each of these 16 lines terminates in a vertical riser that is connected to 54 to 66 individual pressure tubes. Similar outlet risers, 18 in. in diameter, and connector lines conduct the heated coolant from the pressure tubes to an outlet water manifold. The primary coolant system piping within the 105-N building is shown in Figure II.11.

FIGURE II.11. Primary Piping Arrangement

In the heat dissipation building, the primary coolant system consists of six cells in parallel, each containing two steam-generating heat exchangers and one primary coolant circulating pump. Each steam generator provides about 16,000 ft² of heat exchange surface. Five cells are used for reactor operation, and one cell is maintained in standby. The piping in the primary coolant system is constructed of carbon steel, and the system meets design conditions of 1825 psig and 600°F. Normal operating levels are ~395°F inlet, 535°F outlet temperatures and 1600 psig. The primary coolant circulating pumps are the single-stage, horizontal, centrifugal type with high-pressure injection seals to prevent primary coolant loss. The pumps are located inside the shielded cell walls housing the associated steam generator units; they are driven by steam turbines located outside the cells in the turbine bay. Steam for the turbines is supplied by the steam generators during normal reactor operation and by standby oil-fired boilers for reactor startup and shutdown. An electric motor is mounted on the outboard end of the pump drive shaft and is capable of driving the pump to supply sufficient coolant to the reactor during the scram and during shutdown conditions.

The primary system is pressurized to prevent boiling at any point in the system. An elevated pressurizer vessel is connected to the primary loop through a standpipe on the outlet manifold. Pressurization is accomplished by continuously operating submerged electric heaters within the pressurizer, thus creating a steam blanket in the upper portion. Pressure is reduced and controlled by injection of demineralized water into the steam blanket. This water is fed from either the reactor inlet water or from high-pressure injection pumps.

Secondary Coolant System (Steam Supply)

The secondary coolant system for the N Reactor removes the reactor heat from the primary coolant system by boiling secondary system water in the shell side of the steam generators. The steam produced is normally used to drive the primary-pump drive turbines and the in-plant turbine generator (located in 184-N building, the standby powerhouse) with the excess steam available to the WPPSS Hanford Generating Project. When steam is not exported to WPPSS it can

be routed to 16 dump condensers arranged in parallel and cooled by untreated river water. Condensate from the dump condensers is routed back to the steam generators for recycling.

In the current operating mode, N Reactor is not operated when WPPSS cannot accept the steam. The operation of N Reactor and the WPPSS Hanford Generating Project requires a coordinated effort. Each installation requires a thermal adjustment period when starting or stopping operations.

When the reactor is not operating, steam for N Reactor auxiliary systems is supplied by the station service boilers located in the 184-N building. Three boilers, one Foster-Wheeler unit rated at 575,000 lb/hr and two Combustion Engineering units each rated at 200,000 lb/hr, produce steam to drive the station service turbine generator that supplies B Bus power. They also supply steam to the drive turbines for the primary pumps for reactor startup and shutdown.

Electrical Power Systems

Power is supplied by two independent systems: the A Bus is fed by the Hanford 230-kV loop supplied from Bonneville Power Administration (BPA), and B Bus is fed by the 13.8-kV 184-N station service turbine generator.

The A and B Buses are electrically isolated from each other by redundant circuit breakers with each bus supplying the power to approximately half the pumps, blowers and other equipment necessary for reactor operation. Loss of either bus automatically signals a reactor scram, and either system is adequate to support a safe, orderly reactor shutdown. The normal load on each bus is approximately 10 MW.

Confinement System

In the 105-N building, the reactor core and the primary coolant system piping are contained within a secondary concrete enclosure that provides isolation of the reactor from the rest of the 105-N building during reactor operation. The part of the 105-N building contained within the secondary concrete enclosure is

referred to as 105-N Confinement Zone I. The surrounding areas are termed Confinement Zones II and III, each having a lesser potential for radioactive release. In addition, there are ventilation Zones IV and V for unrestricted occupancy.

During reactor operation, the pressure of the atmosphere within the structures is maintained slightly negative relative to that outside the secondary shielding wall and to that outside the 105-N building. Exhaust air from Zones I and II is routed through filter cells in the 117-N building. The filters are designed to remove radioactive particulates before release to the atmosphere through the 200-ft-high 116-N stack.

In the 109-N building, the steam generator cells and the gallery containing the primary coolant system piping are referred to as 109-N Confinement Zone I. The walls enclosing these areas are constructed much like the shielded walls in the 105-N building. The 109-N building has confinement Zone II areas similar to Zone III of the 105-N building that act as a buffer zone between the Zone I radiation and confinement areas and the Zone III unrestricted areas.

II.3.1.4 Water Use, Services, and Utilities

100-N Area

The principal water systems in operation at N Reactor are:

- circulating raw water system
- filtered service water system
- potable and sanitary water system
- demineralized water system
- fire protection system
- intake system
- discharge system

Each of these systems is described in the following pages. Table II.2 lists the quantities of water used during full-power operation and normal shutdown

conditions for the plant. Figure II.12 is a quantitative flow diagram showing water use rate during normal reactor operation at 3,760 MWt. (The reactor is rated at a power level of 4,000 MWt. During CY-1978, the normal power level of the reactor was 3,760 MWt.)

Circulating Raw Water System

The circulating raw water system provides coolant water for all major heat exchangers and steam condensers and is the source for all other water systems in 100-N Area. The system originates at the 181-N building from four pumps having designed ratings of 105,000 gpm each. Normal reactor operation requires that three of the pumps be in service with the remaining pump on standby. Pumps 1 and 2 feed into a 102-in. distribution header, and pumps 3 and 4 feed into a second distribution header of equal size.

Major raw water users are the dump condensers, turbine surface condensers and graphite coolant heat exchangers. Feeder lines to the 182-N building pump house and to the 183-N filter plant serve the other water systems.

Water for the 182-N building is supplied by either or both of the 102-in. supply distribution headers from the 181-N building. The effluent coolant water from the dump condensers is discharged into a 108-in. line; coolant water from turbine surface condensers, various heat exchangers and other sources in the plant is discharged into a 66-in. line. Each feeds into separate sections of a seal well. The discharge from the 66-in. line overflows its seal well to combine with the 108-in. line discharge. To achieve maximum dispersion, the combined flows are discharged through a 102-in. line from the seal well to a point in the middle of the Columbia River. In case of a line plug or collapse, overflow of the combined flows would discharge to the riverbank via an open concrete spillway.

FIGURE II.12. Plant Water Use Diagram

TABLE II.2. Plant Water Use

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System	Full-Power Operation		Normal Shutdown	
	Intake Quantity(gpm)	Intake Temperature(^o F)	Intake Quantity(gpm)	Intake Temperature(^o F)
Circulating Raw Water	315,000	River Ambient	105,000	River Ambient
Filtered Service Water	4,000	River Ambient	4,000	River Ambient
Demineralized Water	1,300	100	1,300	100
Portable and Sanitary Water	200	River Ambient	200	River Ambient
Fire Protection Water	4,500 When Needed	River Ambient	4,500 When Needed	River Ambient

The principal heat removal system in N Reactor is the circulating raw water (CRW) system. The CRW system transports river water, in a single pass mode, through the secondary system, auxiliary system heat exchangers and turbine condensers. Then the heated water is discharged to the river via a 102-in. outfall line that discharges to the center of the river.

The CRW flow rate is 315,000 gpm at a normal power level of 3,760 MWt. The maximum discharge water temperature of the combined or separate flows of the 102-in. and 66-in. lines will not exceed 83.4^oF. During full-power operation of the reactor, approximately 70% of the total heat generated is dissipated by the WPPSS CRW system. When WPPSS cannot accept the steam, N Reactor is not operated.

Will this
be the
case in the
future?

Another significant source that discharges to the river via the 102-in. line is the steam generator blowdown. The blowdown rate is highly variable but usually averages 100 to 300 gpm. The water temperature after mixing and dilution with the dump condenser discharge water will not exceed 83.4^oF.

Major services that dissipate heat to the CRW system are the 16 dump condensers, six primary coolant system drive turbine condensers and oil coolers, four graphite and shield coolant system heat exchangers, the turbine-generator surface condenser and the fuel cladding failure detection heat exchanger. The flow path of the CRW system is shown in Figure II.13.

Filtered Service Water System

The plant service water, also known as the filtered water system, is taken from the CRW system through three coagulator feed pumps in the 183-N filtration building. The feed pumps supply raw water to the chemical mix tank where flocculants are added before the water passes to the two settling basins. The water then flows through six conventional gravity flow filters and is collected in a 200,000-gal clearwell. Three turbine pumps discharge the filtered water into a 850,000-gal tank for storage and distribution. These pumps also supply backwash water to the gravity flow filters, which are backwashed once every 16 hr. The backwash water is discharged to a settling basin.

The filtered water storage tank distributes water to the potable and sanitary water system through the 182-N building. The filtered water distribution systems in the 182-N building consist of two complementary systems: a low-pressure pumping system and a high-pressure pumping system. The low-pressure pumping system provides low-pressure filtered water to the plant and provides normal suction supply water for the high-pressure filtered water pumps. The low-pressure pumping system is supplied by four pumps that distribute filtered water to the area through two lines. This system supplies major services such as the demineralized water plant, spent fuel storage basin and reactor shield coolant system. The main purpose of the high-pressure, filtered water system is to supply coolant water to the rod cooling heat exchanger. Three pumps, each rated at 1,200 gpm, supply the high-pressure system with filtered water.

Potable and Sanitary Water System

The potable and sanitary water system uses water from the filtered water storage tank. Chlorine is added prior to injection of the water in the potable or sanitary water supply system. The design capacity of the system is 200 gpm.

FIGURE II.13. Circulating Raw Water System for 100-N Area

Filtered water is supplied to the 10,000-gal potable water tank by two pumps and is then distributed as needed throughout 100-N Area. The waste flow, consisting of domestic sewage and kitchen waste, is disposed of through a septic tank and tile drainage field.

Demineralized Water System

The demineralized water system is designed to produce high-purity water for use in the primary, secondary, rod and graphite coolant systems with a normal flow of 1,300 gpm and a maximum rate of 2,300 gpm with all units operating. Filtered water, normally supplied by the low-pressure filtered water, is used as the feed water to the demineralizing plant.

This filtered water passes through cation and anion units in the 163-N building to produce demineralized water. The demineralized water is then stored in the 1,000,000-gal demineralized water storage tank. Demineralized water is also stored in two tanks of 500,000-gal and 200,000-gal capacities.

Fire Protection System

When required, water for the independent fire protection system is taken from the CRW system at the 182-N building. Two pumps, one electric motor driven and the other diesel engine driven, are available to supply water when a demand is signaled by low pressure on the pump discharge line. The discharge line is kept pressurized by a 1-in. high-pressure line from the high-pressure raw water system, but will not maintain the pressure when fire protection system water is being used. This decreased pressure signals the electric pump to start. Additional backup is furnished by the diesel engine driven pump.

Intake System

River water is supplied to 100-N Area by four deep-well axial-flow CRW pumps with ratings of 105,000 gpm each at 112 ft of head. Figure II.14 shows a cross section of the outdoor structure that houses the pumps. River water from the building forebay flows through bar screens and traveling screens before entering independent pump suction wells. Well bottoms are 61 ft below the operating floor. The wells are tied together by flood gates normally left in the open

position. Pumps 1 and 2 supply water to a 102-in.-diam distribution header, and pumps 3 and 4 supply coolant water to a second and parallel 102-in.-diam distribution header. The distribution headers are normally interconnected by two common valves with the dump condenser coolant water supply lines located on both sides of the junction. A 66-in. lateral line between these valves supplies the remaining raw water requirements of the plant. Only three river pumps are needed during normal plant operation.

Other intake facilities include conventional trash racks, traveling screens on the water intakes, stoplogs for isolating screen bays and pump bays and horizontal screen wash pumps taking suction from the CRW pump discharge header. Traveling screens are provided to remove material that might damage the pumps and to prevent fingerlings from entering the pump suction wells. The traveling screens are protected by bar screens (trash racks) that keep out large debris.

Six link-belt traveling screens are provided, four for the CRW pump suction wells, and two for the emergency raw water pump suction wells. A clean screen is capable of passing 70,000 gpm of water at minimum low water conditions with a maximum water velocity of approximately 1.25 fps. Each screen is made up of 68 sections, 10 ft wide by 2 ft high, which form an endless link screen, as shown in Figure II.15. The screen mesh is 1/8-in.-diam galvanized wire. Each screen housing is equipped with electric heaters for cold weather protection.

During screen operation, trash is backwashed from the screens by water jets to a sluice trench and is sluiced to a small sump. From the sump, the small trash, passing a 1-1/2 in. grid, and any entrapped fingerlings are transported back to the river through a 16-in. steel pipe.

Discharge System

Coolant water from the plant is discharged to the river through two seal wells as shown in Figure II.16. Seal well No. 2 is designed as a seal between the dump condenser circulating raw water return header and the river. Heated raw water from the dump condensers enters seal well No. 2 through a 108-in. line, flows over the weir and drops to the lowest section of the well. The 66-in. line discharges into seal well No. 1 and also flows over a weir into the lowest section of seal well No. 1. Effluent from both lines thus discharges through a 102-in. line into the main channel of the river to achieve maximum dispersion. The 102-in. outfall line, which extends 862 ft diagonally into the river, is buried below the river bed, covered with backfill and protected by concrete riprap as shown in Figure II.17. The outlet discharges water vertically through the pipe where it strikes a diffuser baffle. Should the 102-in. line become plugged or collapsed the flow from both the 108-in. and 66-in. lines would overflow a steel diversion wall in seal well No. 1, then into a concrete flume to the river's edge.

The 8 in. acid drain from 163-N, along with the water treatment plant filter backwash, is discharged to the settling basin. The steam generator blowdown is discharged into the dump basin from where it is normally pumped to the 1301-N Facility, but it could be pumped into the 102-in. discharge to the river.

FIGURE II.15. Traveling Water Screens

FIGURE II.16. Seal Wells, Spillway and River Channel Discharge

100-K Area

The 181-KE river pump house in the 105-KE Area contains two submersible pumps, either of which is used to pump raw water to the 183-KE filter plant. The pumps are rated at 10,000-gpm maximum capacity each.

Two sedimentation basins, having a total capacity of 6.3×10^6 gal, have been reactivated for servicing the water treatment plant. River water is filtered and treated for algae control in the water treatment plant. One of four service water pumps, rated at 5,000 gpm each, then supplies the treated water to the storage basin facilities at 105-KE and 105-KW. Storage basin water is recirculated through a heat exchanger and a filter.

The secondary coolant flow through the storage basin heat exchanger is less than 3,000 gpm. This treated water is discharged to the midpoint of the Columbia River via a submerged outfall consisting of two 84-in.-diam lines. Because the filtered water and the heat exchanger coolant water, usage rate is much less than the 10,000 gpm supplied by one river pump, there is an excess of at least 4,500 gpm of unused raw water discharged through the same outfall lines. The water treatment filter backwash discharges to a sedimentation basin where the solids are allowed to settle out. The clarified waters are discharged to the Columbia River. This is in accordance with NPDES permit No. WA-000374-3.

The primary function of the fuel storage basin coolant and cleanup system is to remove the decay heat generated by irradiated fuel elements that are stored in the basin and to remove particulate matter for water clarity.

At the design coolant water flow and temperature, the system can maintain the pool water temperature at, or below, 100°F when the basin contains the maximum number of fuel elements. The coolant system is designed to handle a heat load of 2,000 kW. The maximum heat output of the fuel in the fully loaded facility is expected to be approximately 1,860 kW. The maximum design temperature of the secondary coolant water in the water-to-water heat exchanger is 70°F. The maximum temperature rise of the secondary coolant water while passing through

FIGURE II.17. 102-in. Outfall Line, 100-N Area

the heat exchanger is less than 7⁰F. The treated secondary coolant water mixes with the excess raw water being returned to the river so that the temperature of the combined water discharged to the river will not be more than 3⁰F above the river temperature. The heat in this discharge is not expected to cause a measurable increase in river temperature. Thermal discharges from this facility are covered by NPDES permit, No. WA-000374-3, which allows a daily maximum discharge temperature of 75⁰F in the outfall line.

II.3.1.5 Radioactive Effluents and Solid Waste

This section describes the radioactive effluents and solid wastes for facilities in the 100-N and 100-K areas. Each area is discussed separately, with liquid effluents, airborne effluents, and solid wastes being the major subsections. Included in each subsection is a discussion of the origin of the waste, system inventories and/or quantities discharged, and waste treatments.

Section II.3.1.7 describes monitoring programs for the radioactive wastes and Section III discusses environmental effects resulting from the discharge of radioactive wastes from the 100-N and 100-K areas.

Liquid Effluents

Sources

Radioactive material is introduced into liquid systems at N Reactor primarily via cooling systems which are either in direct contact with fuel elements or pass through some section of the reactor core. The most significant of these primary sources are the reactor primary coolant and the spent fuel storage basin coolant. Less significant primary sources are the shield and graphite coolant system, the control rod coolant system, and the gas penetration cooling system. Additionally, radioactive material may be transferred from primary source systems to secondary source systems. The decontamination waste storage facility and the steam generator blowdown represent the major secondary source systems.

Figure II.18 illustrates the various contributory systems and flow pathways.

Discussion of each of the primary source systems and major secondary source systems are presented below. Except where noted otherwise, radionuclide inventories presented in this section are based on actual measurements.

Reactor Coolant System

Radioactive material may be introduced into the reactor coolant by the following processes:

- release of unirradiated uranium to the coolant system through a breach in the fuel cladding and subsequent irradiation of the uranium during circulation in the primary coolant system,
- release of fission products associated with irradiated uranium through a breach in the fuel element cladding,
- activation of materials within the reactor core and subsequent transport of the material by the coolant,
- activation of impurities that are dissolved or suspended in the primary coolant.

The quantity of radioactive material in the primary coolant is minimized through:

- demineralization of water prior to injection into the primary coolant system,
- addition of corrosion inhibitors to the primary coolant,
- rapid shutdown of the reactor and removal of fuel elements with failed cladding, upon indication of a cladding failure,
- continuous feed and bleed of coolant in the primary system,
- periodic decontamination of the piping surfaces within the primary coolant system.

Update! Table II.3 gives concentrations and inventories of radioactive material measured in the primary coolant system during equilibrium operating conditions in CY-1975.

Spent Fuel Storage Basin Coolant

Radioactive material may be introduced into the spent fuel storage basin as a result of discharge of failed fuel elements from the reactor into the

TABLE II.3. Primary Coolant Radionuclide Composition

<u>Radionuclide</u> ^(a)	<u>Type</u> ^(b)	<u>Source</u>	<u>Concentration</u> (pCi/l)	<u>Inventory</u> ^(c) (Ci)
H-3	FP, AP	Coolant, Fuel	2.0×10^4	0.015
Na-24	AP	Coolant	4.5×10^5	0.33
Cr-51	AP	Steel	4.9×10^4	0.037
Mn-54	AP	Steel	1.0×10^5	0.076
Mn-56	AP	Steel	2.1×10^7	16.00
Co-58	AP	Steel, Stellite	1.9×10^4	0.014
Co-60	AP	Steel, Stellite	1.1×10^5	0.083
Fe-59	AP	Steel	3.7×10^5	0.28
Zn-65	AP	Steel	1.6×10^3	0.0012
Sr-90	FP	Fuel	3.0×10^3	0.0023
Sr-91	FP	Fuel	1.4×10^6	1.11
ZrNb-95	AP, FP	Fuel, Zircaloy	3.9×10^5	0.29
Mo-99	FP	Fuel	6.6×10^5	0.50
Ru-103	FP	Fuel	1.4×10^4	0.011
Ru-106	FP	Fuel	3.4×10^3	0.0026
Sb-124	FP	Fuel	3.7×10^3	0.0028
I-131	FP	Fuel	6.2×10^5	0.47
I-132	FP	Fuel	2.0×10^7	15.00
I-133	FP	Fuel	1.1×10^7	8.3
I-135	FP	Fuel	1.6×10^7	12.00
Cs-134	FP	Fuel	5.6×10^2	0.0042
Xe-135	FP	Fuel	7.9×10^6	6.00
Cs-137	FP	Fuel	1.4×10^3	0.0011
BaLa-140	FP	Fuel	1.8×10^5	0.14
Ce-141	FP	Fuel	9.8×10^3	0.0074
Ce-144	FP	Fuel	4.2×10^4	0.032
Np-239	AP	Fuel	5.9×10^5	0.45
Pu-238	AP	Fuel	1.2×10^0	0.00000091
Pu-239/240	AP	Fuel	3.9×10^0	0.0000029

(a) Selected radionuclides.

(b) FP = fission product; AP = activation product.

(c) Primary coolant volume is 200,000 gallons.

basin, failure of fuel element cladding during manipulation of fuel elements in the basin, erosion of radioactive material from the surfaces of irradiated materials stored in the basin, reactor coolant flow during charge-discharge operations, and leakage of reactor coolant system water into the basin.

Table II.4 gives concentrations and inventories of radioactive material in the spent fuel storage basin cooling water.

Shield and Graphite Cooling System

Radionuclides are introduced into the shield and graphite recirculating coolant in the following ways:

- activation of materials within the system piping and subsequent dissolution of the activated material into the coolant;
- dissolution of materials within the system piping into the recirculating coolant and subsequent activation;
- release of fission products from natural uranium present in the coolant;
- activation of impurities in the cooling water.

Table II.5 provides the measured concentrations, calculated inventories, and type and origin of shield and graphite coolant radionuclides during equilibrium reactor operation conditions.

Control Rod Cooling System

Radionuclides are introduced into the recirculating control rod coolant in the following ways:

- activation of materials within the system piping and subsequent dissolution of the activated material into the coolant;
- dissolution of materials within the system piping into the recirculating coolant and subsequent activation;
- release of fission products from natural uranium present in the coolant;
- activation of impurities in the cooling water.

TABLE II.4. 105 N Reactor Spent Fuel Basin
Coolant Radionuclide Concentrations

<u>Radionuclide</u> ^(a)	<u>Type</u> ^(b)	<u>Source</u>	<u>Concentration</u> (pCi/l)	<u>Inventory</u> ^(c) (Ci)
H-3	FP, AP	Coolant, Fuel	2.3×10^4	0.072
Mn-54	AP	Steel	2.7×10^5	0.84
Fe-59	AP	Steel	9.6×10^4	0.3
Co-60	AP	Steel, Stellite	5.3×10^4	0.17
Sr-89	FP	Fuel	1.1×10^5	0.34
Sr-90	FP	Fuel	5.6×10^4	0.17
ZrNb-95	AP, FP	Fuel, Zircaloy	6.7×10^4	0.21
Ru-103	FP	Fuel	3.9×10^3	0.012
Ru-106	FP	Fuel	9.9×10^3	0.031
Cs-134	FP	Fuel	4.1×10^4	0.13
Cs-137	FP	Fuel	1.9×10^5	0.59
BaLa-140	FP	Fuel	3.0×10^3	0.0093
Ce-141	FP	Fuel	6.6×10^3	0.021
Ce-144	FP	Fuel	1.0×10^5	0.31
Pu-238	AP	Fuel	5.8×10^1	0.00018
Pu-239/240	AP	Fuel	2.9×10^2	0.0009

(a) Selected radionuclide.

(b) FP = fission product; AP = activation product.

(c) Coolant volume is 702,000 gallons.

TABLE II.5. Shield and Graphite Cooling System
Radionuclide Composition

<u>Radionuclides</u> ^(a)	<u>Type</u> ^(b)	<u>Source</u>	<u>Concentration</u> <u>(pCi/l)</u>	<u>Inventory</u> ^(c) <u>(Ci)</u>
Na-24	AP	Coolant	7.1×10^5	0.03
Mn-54	AP	Steel	3.2×10^3	0.00014
Mn-56	AP	Steel	2.3×10^5	0.0097
Co-58	AP	Steel	$< 1.2 \times 10^3$	< 0.000051
Co-60	AP	Steel	7.4×10^3	0.00031
Fe-59	AP	Steel	2.3×10^4	0.00097
ZrNb-95	AP,FP	Residual Uranium Zircaloy	$< 2.8 \times 10^3$	0.00012
MoTc-99	FP	Residual Uranium	4.4×10^4	0.0019
Ru-103	FP	Residual Uranium	$< 1.1 \times 10^3$	< 0.000047
Ru-106	FP	Residual Uranium	$< 1.1 \times 10^5$	< 0.0047
Sb-124	FP	Residual Uranium	$< 1.0 \times 10^4$	< 0.00042
I-131	FP	Residual Uranium	6.8×10^3	0.00029
I-132	FP	Residual Uranium	3.8×10^5	0.016
I-133	FP	Residual Uranium	1.2×10^5	0.0051
I-134	FP	Residual Uranium	9.4×10^5	0.04
I-135	FP	Residual Uranium	2.2×10^5	< 0.0093
Cs-134	FP	Residual Uranium	$< 1.4 \times 10^3$	0.000059
Xe-135	FP	Residual Uranium	4.4×10^4	0.0019
BaLa-140	FP	Residual Uranium	$< 3.8 \times 10^3$	< 0.00016
Te-132	FP	Residual Uranium	1.8×10^3	0.000076
Pu-238	AP	Residual Uranium	8.2×10^{-1}	0.000000035
Pu-239/240	AP	Residual Uranium	3.1×10^0	0.00000013

(a) Selected radionuclides.

(b) FP = fission product; AP = activation product.

(c) Coolant volume is 11,200 gallons.

Table II.6 provides the measured concentrations, calculated inventories, type and origin of rod coolant radionuclides during equilibrium reactor operation conditions.

Gas Penetration Cooling System

Radionuclides are introduced into the gas penetration coolant via activation of manganese in the piping walls ($^{55}\text{Mn} + n \rightarrow ^{56}\text{Mn}$, $T_{1/2} = 2.6 \text{ hr}$) and erosion into the coolant. The concentration of Mn-56 within the coolant is $4 \times 10^4 \text{ pCi/l}$. Other radionuclides originating in this system are below detection level.

Steam Generator Blowdown

Radionuclides are introduced into steam generator blowdown by way of leakage from the reactor coolant system into the secondary coolant system. Table II.7 provides the measured concentrations, calculated inventories, type and origin of blowdown radionuclides for normal operations over a full year of operation.

Decontamination Waste Storage Facility

In the mid and late 1950's, periodic chemical cleaning of internal piping surfaces of the eight Hanford nuclear materials production reactors proved to be a cost-effective method to reduce occupational exposure. Consequently, during design of N Reactor a method to chemically clean, collect, and process the decontamination waste was designed into the plant.

Decontamination is normally performed following shutdown of N Reactor for its annual maintenance outage. The decontaminate is inhibited phosphoric acid. 250,000 gallons of eight weight percent phosphoric acid in water is heated and flushed, on a once-through basis, through the reactor coolant loop, over the 17,000 fuel element assemblies, and into the 1310-N facility, a 900,000-gallon, spherical waste storage tank. Following the phosphoric acid flush, 350,000 gallons of water are used as rinse and likewise routed to the 1310-N facility. Sodium hydroxide is then added to the storage tank to neutralize the acid.

TABLE II.6. Rod Cooling System
Radionuclide Composition

<u>Radionuclides</u>	(a) <u>Type</u>	(b) <u>Source</u>	Concentration (pCi/l)	Inventory ^(c) (Ci)
Na-24	AP	Coolant	5.5×10^4	0.00025
Mn-56	AP	Steel	9.0×10^4	0.00041
Co-60	AP	Steel	2.8×10^4	0.00013
Fe-59	AP	Steel	2.9×10^4	0.00013
MoTc-99	FP	Residual Uranium	3.7×10^5	0.0017
I-131	FP	Residual Uranium	2.8×10^5	0.0013
I-132	FP	Residual Uranium	5.1×10^5	0.0023
I-133	FP	Residual Uranium	1.7×10^6	0.0077
I-134	FP	Residual Uranium	1.2×10^5	0.00054
I-135	FP	Residual Uranium	1.2×10^6	0.0054
Cs-134	FP	Residual Uranium	3.6×10^4	0.00016
Xe-135	FP	Residual Uranium	1.7×10^6	0.0077
BaLa-140	FP	Residual Uranium	4.1×10^4	0.00019
Pu-238	AP	Residual Uranium	7.0×10^{-1}	0.0000000032
Pu-239/240	AP	Residual Uranium	9.3×10^{-1}	0.0000000042

(a) Selected radionuclides.

(b) FP = fission product; AP = activation product.

(c) Coolant volume is 1,200 gallons.

TABLE II.7. Steam Generator Blowdown
Radionuclide Composition

<u>Radionuclides</u>	<u>Type^(a)</u>	<u>Source</u>	<u>Concentration (pCi/l)</u>	<u>Release^(b) (Ci/Mo.)</u>
Na-24	AP	Coolant	1.3×10^4	0.004
Cr-51	AP	Steel	5.1×10^4	0.016
Mn-54	AP	Steel	3.5×10^4	0.011
Mn-56	AP	Steel	3.8×10^4	0.012
Co-58	AP	Steel, Stellite	1.7×10^4	0.005
Co-60	AP	Steel, Stellite	9.0×10^4	0.028
Fe-59	AP	Steel	8.4×10^4	0.026
ZrNb-95	AP,FP	Zircaloy, Fuel	3.2×10^4	0.0099
MoTc-99	FP	Fuel	3.9×10^5	0.12
Ru-103	FP	Fuel	1.3×10^4	0.004
I-131	FP	Fuel	1.4×10^5	0.043
I-132	FP	Fuel	4.9×10^5	0.15
I-133	FP	Fuel	6.5×10^5	0.2
I-134	FP	Fuel	5.3×10^5	0.16
I-135	FP	Fuel	5.5×10^4	0.017
Cs1-34	FP	Fuel	1.9×10^3	0.00059
Xe-135	FP	Fuel	7.4×10^4	0.023
BaLa1-40	FP	Fuel	4.5×10^4	0.014
Pu-238 ^(c)	AP	Fuel	7.0×10^{-1}	
Pu-239/240 ^(c)	AP	Fuel	9.3×10^{-1}	

(a) FP = fission product; AP = activation product.

(b) Blowdown volume is ~82,000 gallons per month.

(c) Plutonium concentrations were calculated based on measured concentrations in comparable liquid systems.

Table II.8 provides the measured concentrations, calculated inventories, type and origin of selected radionuclides that were collected from the 1310-N Facility following the 1978 reactor coolant system cleaning.

Liquid Waste Treatment Systems

Figure II.18 provides a schematic of N Reactor liquid effluent streams, treatment facilities, and discharge points. The primary liquid waste treatment system for N Reactor is the 1301-N Facility. Two other facilities, the 1310-N and 1314-N tanks, are used to store and transfer liquid effluents containing radionuclides removed from cooling systems by chemical decontamination.

With the exception of the gas penetration cooling water and steam generator blowdown, all radioactive liquid streams are discharged to the 1301-N waste treatment facility during normal reactor operation.

1301-N Waste Treatment Facility

The 1301-N Facility system is the primary liquid radwaste disposal system for N Reactor. Radioactive liquid effluents from the reactor coolant system, spent fuel storage basin, periphery coolant systems, and various radioactive drain systems throughout the reactor facility are discharged to the 1301-N Facility. The 1301-N Facility is a ground disposal facility that makes use of the natural filtration properties of soil to remove radioactive material from effluent water.

The 1301-N Facility is located on the river bank bluff, some 60 ft above and 860 ft back from the Columbia River. Liquid streams from reactor systems discharge into a concrete trough then spill into a crib (see Figure II.19). The crib itself is a 125-ft by 290-ft rectangular basin that was created by excavating the existing soil and then surrounding the excavation with a soil and gravel embankment. The bottom of the crib is filled with a 3-ft layer of large stones. The water level in the crib is maintained well below the surface of the stones by limiting the flow to the crib and through the use of an extension trench connected to the north side of the crib.

TABLE II.8. 1310-N Facility Radionuclide Composition

<u>Radionuclide</u> ^(a)	<u>Type</u> ^(b)	<u>Source</u>	<u>Concentration</u> <u>(pCi/l)</u>	<u>Inventory</u> ^(c) <u>(Ci)</u>
Cr-51	AP	Steel	3.0×10^7	57.1
Mn-54	AP	Steel	1.8×10^7	36.5
Co-58	AP	Steel, Stellite	8.1×10^6	76.9
Co-60	AP	Steel, Stellite	1.4×10^8	290.6
Fe-59	AP	Steel	3.9×10^7	81.2
Nb-95	FP	Fuel	6.6×10^6	13.6
Ru-103	FP	Fuel	5.1×10^6	10.5
BaLa-140	FP	Fuel	8.8×10^5	1.8
Ce-141	FP	Fuel	4.6×10^6	9.6
CePr-144	FP	Fuel	4.9×10^6	10.1

(a) Other radionuclides not analyzed for.

(b) FP = fission product; AP = activation product.

(c) 600,000 gallons of 900,000 gallon capacity is used.

FIGURE II.18. N Reactor Liquid Effluent Flow Schematic

FIGURE II.19. 1301-N Facility Layout

The crib extension trench is an excavated ditch approximately 1600 ft long, 50 ft wide, and 12 ft deep, extending in a zigzag pattern. A wire mesh cover is provided over the entire trench to prevent larger animals and birds from obtaining food, water, or shelter within the radiation area. The water level in the trench is maintained at a nearly constant depth of approximately 3 ft.

The total flow to the crib and trench system is approximately 3 million gallons per day from (1) 36-in. radioactive drain line (about 2.5 million gpd during reactor operation), (2) 12-in. radioactive drain line (about 500,000 gpd during reactor operation), (3) 3-in. drain line from the waste disposal valve pit (about 350 gpd), and (4) 6-in. auxiliary decontamination line from 109-N building (intermittent).

Water discharged to the crib percolates through the underlying soil to an unconfined aquifer in communication with the Columbia River. During the movement of water in the ground underlying the crib and trench, radioactive materials are retained in the soil through precipitation and ion exchange reactions. Additional loss of radioactive material occurs by decay of the shorter lived radionuclides in the discharge water during transit to the river. Hydrological analyses have indicated that flow pathways to the river are a complex system in that the water may enter the river from beneath the river bed or from springs on the river shoreline, and that the pathways may lead directly to the river, or may parallel the river for several miles before entry. Radionuclide sampling studies have shown that the most radioactive spring water enters the river via the shortest crib-to-river pathway. Figure II.20 illustrates the results of an evaluation of hydrological factors influencing the flow pathways from the crib to the river using a mathematical simulation technique. Figure II.21 depicts a cross sectional view of the 1301-N Facility.

Table II.9 gives a summary of the sampling data for 1978 releases to the 1301-N Facility. The effectiveness of the disposal system is indicated by the degree of

FIGURE II.20. Calculated Flow Paths Originating from the 1301-N Facility with Travel Time Lines

FIGURE II.21. Cross Section of the 1301-N Facility

TABLE II.9. Discharge of Radionuclides to 1301-N Facility
and Seepage to River During CY 1978

Radionuclide	To the 1301-N Facility			Seepage from Facility to River Via Springs ^(a)			Decontamination Factor ^(b)
	Average Concentration (pCi/l)	Peak Concentration (pCi/l)	Total Released (Ci)	Average Concentration (pCi/l)	Peak Concentration (pCi/l)	Total Released (Ci)	
³ H	6.9x10 ⁴	2.9x10 ⁵	3.3x10 ²	5.8x10 ⁴	1.5x10 ⁵	3.3x10 ²	1
³² P	1.2x10 ⁴	6.6x10 ⁴	5.6x10 ¹	1.3x10 ¹	2.1x10 ²	2.9x10 ⁻²	1,900
⁵¹ Cr	7.9x10 ⁴	2.1x10 ⁵	3.7x10 ²	6.4x10 ¹	7.7x10 ¹	1.4x10 ⁻¹	2,600
⁵⁴ Mn	1.7x10 ⁵	2.8x10 ⁵	8.1x10 ²	ND	ND	ND	---
⁵⁹ Fe	1.6x10 ⁵	7.4x10 ²	7.6x10 ²	ND	ND	ND	---
⁵⁸ Co	1.4x10 ⁴	3.1x10 ⁴	6.4x10 ¹	7.8x10 ⁰	8.9x10 ⁰	1.7x10 ⁻²	3,800
⁶⁰ Co	2.0x10 ⁵	5.6x10 ⁵	9.4x10 ²	1.8x10 ²	3.9x10 ²	4.1x10 ⁻¹	2,300
⁶⁵ Zn	8.3x10 ³	1.3x10 ⁴	3.9x10 ¹	ND	ND	ND	---
⁸⁹ Sr	1.1x10 ⁵	1.8x10 ⁶	5.2x10 ²	1.8x10 ²	4.8x10 ²	4.0x10 ⁻¹	1,300
⁹⁰ Sr	2.5x10 ⁴	9.6x10 ⁴	1.2x10 ²	5.9x10 ²	1.5x10 ³	1.3x10 ⁰	90
⁹⁵ ZrNb	1.3x10 ⁵	1.1x10 ⁶	1.2x10 ³	ND	ND	ND	---
^{99m} MoTc	3.3x10 ⁵	5.6x10 ⁶	1.6x10 ³	5.9x10 ²	1.4x10 ³	1.3x10 ⁰	1,200
¹⁰³ Ru	6.3x10 ⁴	5.6x10 ⁵	3.0x10 ²	1.4x10 ²	4.3x10 ²	3.1x10 ⁻¹	970
¹⁰⁶ Ru	7.0x10 ⁴	1.6x10 ⁵	3.3x10 ²	2.2x10 ²	4.4x10 ²	4.9x10 ⁻¹	670
¹²⁴ Sb	1.3x10 ⁴	1.2x10 ⁴	6.2x10 ¹	2.8x10 ¹	8.0x10 ¹	6.2x10 ⁻²	1,000
¹²⁵ Sb	7.6x10 ²	7.5x10 ²	4.1x10 ⁰	7.7x10 ¹	3.1x10 ²	1.7x10 ⁻¹	20
¹²⁹ I	6.7x10 ⁻³	---	1.4x10 ⁻⁵	3.6x10 ⁻³	---	8.2x10 ⁻⁶	2
¹³¹ I	1.2x10 ⁵	1.0x10 ⁶	5.6x10 ²	1.7x10 ³	2.7x10 ⁴	3.8x10 ⁰	150
¹³³ Xe	1.1x10 ⁵	8.3x10 ⁵	5.2x10 ²	2.6x10 ³	8.3x10 ³	5.7x10 ⁰	90
¹³⁴ Cs	1.4x10 ⁴	4.6x10 ⁴	6.7x10 ¹	ND	ND	ND	---
¹³⁷ Cs	7.3x10 ⁴	2.4x10 ⁵	3.4x10 ²	7.1x10 ⁰	1.0x10 ¹	1.6x10 ⁻²	21,000
¹⁴⁰ BaLa	7.3x10 ⁵	1.4x10 ⁷	7.1x10 ³	ND	ND	ND	---
¹⁴¹ Ce	1.0x10 ⁵	1.2x10 ⁶	4.9x10 ²	ND	ND	ND	---
¹⁴⁴ CePr	1.4x10 ⁵	5.6x10 ⁵	6.6x10 ²	ND	ND	ND	---
¹⁵⁴ Eu	1.2x10 ⁴	1.3x10 ⁴	5.5x10 ¹	ND	ND	ND	---
¹⁵⁵ Eu	7.3x10 ³	7.3x10 ⁴	3.4x10 ¹	ND	ND	ND	---
²³⁸ Pu	5.5x10 ¹	1.2x10 ²	2.6x10 ⁻¹	5.7x10 ⁻²	3.3x10 ⁻¹	1.3x10 ⁻⁴	2,000
^{239/240} Pu	2.7x10 ²	5.6x10 ²	1.3x10 ⁰	1.1x10 ⁻¹	1.3x10 ⁰	3.0x10 ⁻⁴	4,000

(a) Estimated Flow = 900 gpm.

(b) DF = $\frac{\text{Total Released Facility}}{\text{Total Released Springs}}$

ND = Radionuclide not detected in springs but were measured in effluent to the facility.

decontamination provided by the facility. The variation in removal capability for different radionuclides is dependent on the half-life and the chemical properties of the radionuclide involved.

Table II.10 gives results from a study performed to determine the physicochemical composition of radionuclides discharged to the crib. In the study, radionuclides were partitioned into their particulate, cationic, anionic, and nonionic fractions. Since the 1301-N disposal facility utilizes a natural soil column composed of sand (particulate filtration) and clay (cation exchange), the column is noticeably more effective for radionuclides predominantly in the particulate and cationic forms. As indicated in Table II.9 the predominantly anionic and nonionic species such as iodine, ruthenium, and antimony were removed by the soil column to a lesser extent than the particulate and cationic radionuclides such as iron, cesium, zirconium, and manganese.

Some radionuclides exhibit characteristics indicating a mixture of the various physicochemical forms. Radionuclides of this type emerge at the river springs in essentially anionic and nonionic forms as their particulate and cationic fractions are retained by the soil column. Tritium, which is chemically indistinguishable from water, is not removed during transit to the river.

The capability of the 1301-N Facility to continue to provide satisfactory radionuclide removal during future operation of N Reactor has been an important factor in evaluating the need for an alternate liquid waste disposal system. Table II.11 gives results of liquid effluent transmission factors from 1974 to 1978. As shown in the table, the radionuclide removal capability remained fairly constant and showed no significant degradation.

The discharge of radioactive liquids to the 1301-N Facility since N Reactor startup in 1963 has resulted in the accumulation of significant quantities of long lived radionuclides in the soil on the bottom of the facility.

TABLE II.10. Physicochemical Characteristics of Radionuclides
in Liquid Effluents Discharged to the 1301-N(a) Facility
(Percent of Total)

MEASURED RADIONUCLIDE	APPROXIMATE HALF- LIFE	PARTICULATE	CATIONIC	ANIONIC	NONIONIC
Sc-46	84 d.	64	23	11	3.0
Cr-51	28 d.	85	6.7	7.7	.83
Mn-54	300 d.	10	89	.20	.34
Fe-59	45 d.	98	.70	1.3	.35
Co-58	71 d.	93	4.8	2.3	.14
Co-60	5.2 yr.	49	49	.98	.83
Zn-65	245 d.	24	77	1.3	.35
As-76	26 hr.	3.0	7.2	70	20
Sr-91	10 hr.	7.7	86	6.0	.39
Zr-95	66 d.	81	5.7	11	2.5
Nb-95	35 d.	41	47	8.0	4.5
Zr-97	17 hr.	81	9.4	5.6	3.8
Mo-99	67 hr.	.32	3.3	94	2.4
Ru-103	40 d.	66	17	13	4.6
Ru-106	1 yr.	17	49	30	4.7
Sb-122	3 d.	17	11	48	23
Sb-124	60 d.	29	35	23	13
Sb-125	2.7 yr.	23	46	18	13
I-131	8 d.	.39	19	79	1.7
I-133	20 hr.	.33	21	78	1.5
I-135	7 hr.	6.3	31	61	1.1
Cs-134	2 yr.	13	85	1.4	.08
Cs-137	30 yr.	15	85	.41	.03
Ba-140	13 d.	4.5	89	.46	5.6
La-140	40 hr.	6.3	93	.34	.41
Ce-141	33 d.	63	29	6.1	2.1
Ce-144	284 d.	50	42	5.0	3.0
Pm-151	28 d.	9.0	62	14	15
W-187	24 hr.	85	7.7	5.9	1.6
Np-239	2 d.	26	70	3.2	.42

(a) Ref. 12.

TABLE II.11. 1301-N Facility Transmission Factors (%)

<u>Radionuclide</u>	<u>1974</u>	<u>1975</u>	<u>1976</u>	<u>1977</u>	<u>1978</u>
³ H	100	100	100	100	100
⁶⁰ Co	0.06	0.08	0.11	0.07	0.04
⁹⁰ Sr	0.47	0.43	0.32	0.83	1.11
⁹⁵ ZrNb	0.01	(a)	(a)	(a)	(a)
⁹⁹ Mo	0.03	0.06	0.02	0.06	0.08
¹⁰⁶ Ru	0.63	0.45	0.33	0.29	0.15
¹³¹ I	0.47	0.42	0.22	0.77	0.67
¹³⁷ Cs	0.03	(b)	(a)	0.01	(b)
¹⁴⁰ BaLa	0.05	(a)	(a)	(a)	(a)

(a) Radionuclide detected at crib but not at springs.

(b) Not significant.

TABLE II.12. Radionuclide Burden of
1301-N Facility at End of 1978

<u>Radionuclide</u>	<u>Facility Burden (Ci)</u>
⁶⁰ Co	3100
⁹⁰ Sr	1400
¹³⁷ Cs	1800
²³⁹ Pu	11

Table II.12 lists quantities of fission and activation products contained in this soil as of 1978.

A series of wells have been drilled in the vicinity of the 1301-N crib-trench system to monitor the migration of radionuclides away from the soils directly underlying the facility. Regular sampling of these wells has provided no indication of migration of radionuclides contained in the soil.

1310-N Decontamination Waste Storage Facility

Approximately 600,000 gallons of liquid waste from the reactor coolant piping decontamination are routed to the 1310-N storage facility annually. The 1310-N facility is a 900,000-gal spherical storage tank located inside the northeast corner of the 100-N Area exclusion fence.

Reactor decontamination is a once-through process in which 250,000 gallons of 8% by weight phosphoric acid is flushed through the reactor coolant and routed to the 1310-N tank. Sodium hydroxide is added to the tank to neutralize the acid. The final volume of waste in the tank is approximately 600,000 gallons. The tank is emptied by shipping the waste via the 1314-N Facility into a railway tank car which carries the liquid to the 200-West Area for processing through evaporation so that only the solids are disposed of in designated burial ground after appropriate packaging. Table II.8 gives the radionuclide concentration and inventory of decontamination wastes initially stored in the 1310-N as a result of the 1975 reactor cooling system decontamination.

Most
Recent?
Update

Liquid Effluent Discharges

The two points of discharge of radioactive material to the environment from N Reactor liquid systems are the riverbank springs and the 102-inch discharge line.

Riverbank Springs

Figure II.18 shows the various systems contributing to the riverbank springs discharge and Table II.9 presents a summary of the quantities of radioactive material released to the Columbia River via the riverbank springs discharge point during 1978.

102-Inch Line

The 102-in. discharge line discharges minor amounts of radioactive material directly to the Columbia River. Figure II.18 shows the various systems contributing to the 102-in. line discharge and Table II.13 presents a summary of the quantities of radioactive material released to the Columbia River during 1978.

The primary source of radionuclides routinely discharged via the 102-in. line is the gas penetration coolant coil drain system. During reactor operations, the only identifiable radionuclide present in the gas penetration cooling system is the short lived activation product ^{56}Mn ($T_{1/2} = 2.6$ hr). Manganese-56 is discharged at an average concentration of 2 pCi/gal in the 102-in. line. This concentration is approximately 0.003% of the ERDAM-0524 concentration guide for releases of ^{56}Mn to uncontrolled areas. Cost-benefit analyses have indicated that alternate effluent discharge methods for this system are not economically practical.

Operational Upsets

A failure of a containment barrier between a radioactive liquid system and a nonradioactive liquid system could release the radioactive material directly to the river via the 102-in. line. However, operating procedures require shutdown of N Reactor upon indication of such an event in order to mitigate these releases.

On March 25 and on November 20, 1976, failure of two heat exchanger tubes in the reactor coolant system spill cooler resulted in the inadvertent release of radioactive material to the Columbia River at concentrations exceeding normal releases. This effluent stream has subsequently been rerouted to the 1301-N crib. Table II.14 presents a summary of the releases resulting from this equipment failure.

TABLE II.13. Releases to River Via 102-in.
Discharge Line During CY 1978

<u>Radionuclide</u>	<u>Average Concentration (pCi/l)</u>	<u>Total Released (Ci)</u>
⁵⁴ Mn	0.31	0.18
⁵⁶ Mn	2.1	1.2
⁵⁸ Co	0.026	0.015
⁵⁹ Fe	0.19	0.11
⁶⁰ Co	0.96	0.55
⁹⁵ ZrNb	0.44	0.26
¹⁰³ Ru	0.095	0.055
¹⁰⁶ Ru	0.14	0.084
¹³¹ I	1.7	0.96
¹³³ I	1.7	0.95
¹³⁷ Cs	0.026	0.015
¹⁴⁰ BaLa	7.0	4.1
¹⁴¹ Ce	0.06	0.035
¹⁴⁴ Ce	0.25	0.15
¹⁴⁷ Nd	0.29	0.17
^{239/240} Pu	0.15	0.085

TABLE II.14. Radionuclide Discharge to River
as a Result of Spill Cooler Failures

Radionuclide (a)	March 25, 1976		November 20, 1976	
	Maximum Concentration (pCi/l)	Released (Ci)	Maximum Concentration (pCi/l)	Released (Ci)
^{32}P	1.5×10^4	1.8×10^0	8.6×10^2	1.8×10^{-2}
^{54}Mn	6.8×10^3	8.7×10^{-1}	2.6×10^3	2.5×10^{-1}
^{56}Mn	1.2×10^6	6.4×10^1	9.3×10^4	8.9×10^0
^{58}Co	2.1×10^3	2.7×10^{-1}	ND ^(b)	ND
^{60}Co	8.2×10^3	8.1×10^{-1}	9.7×10^2	9.3×10^{-2}
^{59}Fe	2.4×10^4	1.8×10^0	1.0×10^3	9.7×10^{-2}
$^{85\text{m}}\text{Kr}$	1.2×10^4	6.9×10^{-1}	5.8×10^3	5.6×10^{-1}
^{87}Kr	3.5×10^4	2.8×10^0	1.6×10^4	1.5×10^0
^{88}Kr	3.4×10^4	2.9×10^0	1.3×10^4	1.3×10^0
^{89}Sr	1.3×10^3	7.9×10^{-2}	1.8×10^2	1.8×10^{-1}
^{90}Sr	1.4×10^2	9.9×10^{-3}	7.7×10^0	7.0×10^{-4}
^{91}Sr	1.9×10^4	1.6×10^0	8.8×10^3	8.5×10^{-1}
^{95}Zr	1.0×10^4	1.3×10^0	ND	ND
^{95}Nb	8.3×10^3	1.0×10^0	ND	ND
^{99}Mo	6.7×10^3	5.4×10^{-1}	5.0×10^3	4.9×10^{-1}
^{131}I	3.8×10^3	2.1×10^{-1}	1.5×10^3	1.4×10^{-1}
^{132}I	2.2×10^5	1.2×10^1	3.7×10^4	3.6×10^0
^{133}I	3.0×10^4	3.9×10^0	2.5×10^4	2.4×10^0
^{134}I	ND	ND	3.0×10^4	2.9×10^0
^{135}I	1.2×10^5	1.4×10^1	4.9×10^4	4.7×10^0
^{133}Xe	ND	ND	1.8×10^3	1.8×10^{-1}
^{135}Xe	4.8×10^4	3.6×10^0	9.5×10^4	9.1×10^0
^{138}Cs	5.1×10^5	4.1×10^1	2.2×10^5	2.2×10^1
^{140}Ba	ND	ND	9.5×10^2	9.2×10^{-2}
^{140}La	ND	ND	1.3×10^3	1.3×10^{-1}
^{187}W	3.5×10^4	3.9×10^0	ND	ND
^{239}Np	1.8×10^4	2.1×10^0	ND	ND
^{238}Pu	ND	ND	4.0×10^{-1}	$< 3.9 \times 10^{-5}$
$^{239/240}\text{Pu}$	4.8×10^0	4.6×10^{-4}	3.2×10^{-1}	$< 3.2 \times 10^{-5}$

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(a) With the exception of ^3H , radionuclides not reported were not identified in the effluent stream.

(b) ND = not detected.

Airborne Effluents

Sources

Airborne radioactive material is generated by the following processes:

- activation of impurities (primarily argon) within the inert reactor gas atmosphere;
- migration of noble gases and volatile fission products from irradiated fuel through leaks in the fuel element cladding;
- suspension of particulate contamination from the surfaces of radioactive materials.

Argon-41 is the primary radioactive airborne emission from N Reactor. During 1978, 120,000 Ci of ^{41}Ar were discharged to the atmosphere from N Reactor. Studies have shown that ^{41}Ar production is the result of the leakage of air into the reactor gas system and subsequent activation of the stable argon in the air.

Noble gases and volatile fission products, such as ^{133}Xe and ^{131}I , are introduced into the liquid streams through leaks in fuel element claddings. These gases and vapors are then vented from the liquid systems into the atmosphere through gas vents in the primary coolant system and at points of coolant bleed and leakage.

Nonvolatile particulate fission and activation products (such as ^{60}Co , ^{154}Eu , ^{99}Mo) become airborne as a result of primary coolant contacting exposed surfaces, then drying and becoming suspended in air currents.

Because of the nature of the generation of airborne contamination within the reactor systems, it is very difficult to quantitatively characterize the primary source terms. Sampling of all effluent discharge points at N Reactor provides data for assessment of atmospheric discharges of contaminated materials from the reactor buildings.

Airborne Effluent Discharges

The three general pathways for release of airborne radioactive material from 100-N areas to the environment are (1) 105-N reactor building ventilation exhausts, (2) 109-N steam generator building ventilation exhausts, and (3) radioactive liquid system vents. The locations of these discharge points are shown in Figure II-22.

105-N Reactor Building Ventilation Exhausts

The ventilation systems in 105-N are separated into five zones based on the potential for airborne radioactive material. Zone I is the primary confinement zone servicing the reactor block and reactor coolant system pipe spaces. Zone II services areas adjacent to Zone I, including the control rod rooms, the fuel cladding failure monitor rooms, and the ball safety system control room. Zone III serves the hot shops, fuel storage area, and equipment rooms. Zones IV and V serve offices and administration areas along with the reactor control room.

Table II.15 lists ventilation exhaust locations for the 105-N building.

The 116-N stack is the primary discharge point for airborne radioactive material from N Reactor. Figure II.23 shows the 105-N building ventilation exhaust systems that discharge to the atmosphere via the 116-N stack. Immediately preceding the 116-N stack is the 117-N filter and diversion facility through which the exhaust air is routed prior to release to the stack.

There are three relatively minor exhaust systems that also serve the 105-N building. The 105-N wash pad exhaust, 105-N elevation (+14 ft) decontamination hood exhaust, and the 105-N transfer area exhaust are each provided with a vent located on the roof of the 105-N building. Discharges through the first two vents are routed through high-efficiency particulate filters. The discharge is intermittent. The 105-N transfer area exhaust is continuous and unfiltered. The discharge of airborne material is negligible from each of these release points as indicated by periodic sampling.

Table II.16 gives the 1978 releases to the environment from the 105-N building. All 105-N systems not discharging via the 116-N stack are included in the column titled "all others".

109-N Steam Generator Building Ventilation Exhausts

The ventilation system in the 109-N building is separated into four zones similar to the 105-N ventilation system. Zone I is the primary confinement zone. Two separate and independent Zone I ventilation systems exist in the 109-N building. The 109-N Zone I cell 6 ventilation system provides ventilation exhaust from steam generator cell 6. The 109-N Zone I building ventilation system provides ventilation exhaust from steam generator cells 1 through 5, the auxiliary systems, the pipe gallery, the pressurizer penthouse, and the cell access corridor. Both systems use once-through flow that exhaust directly to the atmosphere. There are no cross ties from Zone I to any other ventilation zone or system, with exception of cross vents with 105-N Zone I, which allows diversion of 109-N Zone I through the 117-N filter system during an emergency.

Zones II and III represent areas of lower airborne radiation potential. Zone IV services offices and other non-radiation areas. Table II.17 lists the ventilation exhaust points for the 109-N steam generator building. Table II.16 gives the 1978 releases to the environment from the 109-N building. All 109-N systems not discharging via the Zone I vent or the cell 6 exhaust are included in the column title "all others".

Radioactive Liquid System Vents

Small but measurable quantities of radionuclides are released to the atmosphere from vents in the 1301-N waste treatment facility the 1310-N and 1314-N chemical decontamination facilities, and the main export steam lines. Figure II.22 shows the location of these facilities with the exception of the steam line vents which are located near the transformer area in the lower portion of the figure.

TABLE II.15
VENTILATION EXHAUST LOCATION 105-N

<u>Discharge Point</u>	<u>Zone</u>	<u>Area Served</u>
116-N Stack	I, II, III	Reactor Zone I, II, and all Zone III exhaust except those minor exhaust systems listed below. Particulate and charcoal filters used in the 117-N Facility used for Zones I and II.
105-N Roof Vent	III	Wash pad - 105-N. Particulate filter used.
105-N Roof Vent	III	Decontamination hood. Particulate filter used.
105-N Roof Vent	III	Fuel transfer area (no treatment).

TABLE II.16. Annual Radionuclide Releases to Atmosphere(a)
From N Reactor Operations
(Measured Activity)

Radionuclide Types	116-N Stack	<u>109-N Releases</u>		All Others	Total Released
		Zone 1 Vent	Cell 6 Exhaust		
<u>Noble Gases</u>		(Curies Released)			
Ar-41	120,000	230	410	ND(b)	120,640
Xe-135	1,500	660	88	ND	2,248
H-3	32	0.19	0.037	ND	32.23
Halogens and Particulates	116-N Stack	<u>109-N Releases</u>		All Others	Total Released
		Zone 1 Vent	Cell 6 Exhaust		
		(Curies Released)			
I-131	0.12	0.28	0.073	0.002	0.48
I-132	1.1	3.7	1.4	ND	6.2
I-133	0.46	2.0	0.6	0.0065	3.1
I-135	1.1	3.3	1.7	ND	6.1
Mn-54	0.0025	0.011	0.0068	0.0069	0.027
Mn-56	0.11	2.1	1.5	ND	3.71
Fe-59	0.0058	0.01	0.0069	ND	0.028
Co-60	0.0061	0.02	0.012	0.0085	0.047
As-76	4.0	2.6	0.043	ND	6.64
Sr-89	0.00049	0.016	0.0052	ND	0.022
Sr-90	0.000058	0.0005	0.00016	ND	0.00072
Mo-99	0.0012	0.81	0.17	0.0024	0.98
BaLa-140	0.0036	0.97	0.14	0.00068	1.11

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w/ Guide
(5480.1)
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(a) Selected radionuclides based upon Environmental Report for 1978 releases and assumptions including a total of 380,000 CFM exhaust rate from all release points.

(b) ND = Not Detected

TABLE II.17. Exhausts Vents - 109-N Building

<u>ZONE NO.</u>	<u>RATED FLOW (CFM x 10³)</u>	<u>AREA SERVED</u>	<u>DISCHARGE POINT</u>
I	128	Cells 1-5, auxiliary cell pipe gallery, and pressurizer	109-N roof
IA	14	Cell 6 and Cell 6 pipe gallery	109-N roof
II	23	Hot tool room, hot storage room	109-N roof
III	130	Turbine bay, mechanical & electrical equipment rooms, storage area, offices, lunch rooms, locker rooms	109-N roof
	2.4	Hot water quality labs	109-N roof
	3.1	Rms, 17, 112, 110, locker room, shower, toilet	109-N roof
	0.21	Exhausts hydrogen peroxide room to Zone III	To Zone III Main Exhaust
	0.275	Exhausts crud probe fume hood	To EF-11, 12
	0.25	Exhausting chemical cabinet in Room 207	109-N roof
	21	Cell 6 end of turbine bay	109-N roof
	1.6	Chemical Decontamination Solution Preparation Area	E1. 80'0" Decontamination Dry Chemical Storage Area Roof
	1	Same as above	Decontamination Dry Chemical Storage Area Roof

FIGURE.II 22. 100-N Atmospheric Release Points

FIGURE 11.23. 105-N Ventilation and Exhaust Systems

Samples taken from these facilities have shown that in each case, the releases are insignificant compared to the total airborne discharge from N Reactor. Discharges from these facilities are included in Table II.16 under the column marked "all others".

Airborne Waste Treatment Facilities

Figure II.24 is a diagram of the 117-N filter and diversion facility. Exhaust air from the 105-N building enters 117-N building through three exhaust ducts. The three ducts serve to maintain separation of air coming from Zones I, II, and III. The 117-N facility is composed of four separate air filtration cells. Exhaust air from 105-N Zone I is passed through two of the filtration cells. These cells each consist of a series of three filter banks. The first stage filter bank serves a moisture separator and is used to protect the remaining filters in the cell. The moisture separator consists of aluminum mesh screens designed to remove the entrained moisture that could be present in the air during an accident.

The second-stage filter bank consists of a high-efficiency fire-retardant particulate filter. The filter is designed for continuous operation at a maximum temperature of 200⁰F at a relative humidity of 100%. Minimum efficiencies for sorption of particulate matter is 99.9%. The third stage filter bank contains granular activated charcoal capable of removing 95% of the incident inorganic halogen gases that reach the cell.

Exhaust air from 105-N Zone II is passed through a single filter cell (see Figure II.24). The Zone II filter cell consists of a series of two filter banks: a high-efficiency particulate filter and an activated charcoal absorber.

Exhaust air from 105-N Zone III is normally routed through 117-N so that the filter cells are bypassed. However, in the event of an emergency, provisions exist for combining the Zone III exhaust with the Zone II exhaust so that filtration can be provided.

FIGURE II.24. 117-N Exhaust Filter and Diversion Facility

A fourth filter cell is provided in 117-N as a standby or emergency backup unit. The fourth cell contains a series of three filter banks similar to those in the cells used for Zone I exhaust filtration.

The exhaust air from Zone I, II, and III discharges into the 116-N stack which exhausts to the atmosphere 200 ft above ground level.

Solid Waste

The types of radioactive solid waste generated at N Reactor are diverse. For convenience, radioactive solid waste at N Area is divided into two categories: waste containing relatively high quantities of radioactive material and waste containing relatively low quantities of radioactive material.

Waste containing^a high quantities of radioactive material consists of irradiated reactor components such as fuel element support hardware, fuel element spacers, boron carbide safety balls, irradiated valves and piping, and other metallic refuse from the reactor and fuel storage basin. The level of activity in this waste is sufficient to require shielding for personnel protection. The waste is shipped to the 200 areas for disposal. Predominant radionuclides present in this waste are ^{60}Co , ^{59}Fe , $^{95}\text{ZrNb}$, and ^{154}Eu .

Typical shipments of waste containing lower quantities of radioactive material consist of large quantities of paper floor covering, wiping materials, plastic, tape, step-off pads, wood, pipe, and other standard industrial and construction refuse. Large pieces of machinery or equipment are also occasionally included. These materials are, on the average, contaminated to relatively low levels and do not require shielding for personnel protection. Radionuclides normally present in this waste are ^{51}Cr , ^{54}Mn , ^{59}Fe , ^{60}Co , ^{65}Zn , $^{95}\text{ZrNb}$, and $^{144}\text{CePr}$.

All radioactive solid waste generated at N Reactor is surveyed for dose rate and surface contamination levels before it is removed from the radiation zones.

Low-level waste is generally packaged in cardboard cartons and sealed with tape. Large-sized waste is either wrapped in plastic or, if the surface contamination levels are not prohibitive, is placed unwrapped in a covered transport vehicle. Waste that is shipped unwrapped consists of large sections of wood, piping, or large pieces of machinery or equipment. Low-level radioactive waste is shipped to the 200 areas for disposal.

Radioactive waste consisting of fuel element support hardware and other refuse from the 105-N fuel storage basin is packaged under water in the fuel storage basin using a special, shielded shipping cask. This cask is an upright cylinder having a capacity of 40 ft³. After the metallic waste has been loaded, the cask is removed from the basin, drained, and loaded onto a vehicle for truck or rail shipment to the 200 areas for disposal.

Other radioactive waste is packaged and transported using a variety of special methods designed in each case to prevent undue personnel exposure and spread of loose radioactive material outside the shipping container.

During CY-1978, 28,000 ft³ of radioactive waste containing over 5,700 Ci of activity were disposed to the 200 areas for burial from N Area. About 5,000 ft³ of radioactive waste containing about 170 curies from 105-K Area were also shipped to the 200 Area for disposal. The onsite shipment of radioactive wastes complies with the requirements imposed by DOE-M-RL 0529, as interpreted by UNI-M-29, "Shipment of Radioactive and Other Hazardous Materials Manual".

II.3.1.6 Nonradioactive Effluents and Solid Waste

100-N Area

Liquid Effluents

The types of material discharged as liquid effluents from 100-N Area, the source of the material and the annual quantities of materials used are given in Table II.18.

Sodium hydroxide is used to regenerate the anion exchange columns, and sulfuric acid is used to regenerate the cation exchange columns. The anion and cation exchange columns are used to deionize the water in the water purification process. These regenerative chemicals are used alternately, and their use rate depends on the required regeneration frequency. Approximately 2,000 lbs of caustic and acid are used per regeneration. The caustic injection lasts approximately 85 min, and the acid injection lasts approximately 32 min. Regenerations average one per day, but may occur up to four times per day. The spent chemicals, consisting mainly of sodium hydroxide and sulfuric acid, are discharged to a ground disposal facility, in compliance with NPDES Permit No. WA-000374-3.

Aluminum sulfate and polyacrylamide are added to the incoming raw water to form and retain aluminum hydroxide precipitate in the filter media. The precipitate removes the suspended solids and silt from the raw water. The aluminum hydroxide, polyacrylamide and filtered solids are also discharged to the ground disposal facility during filter backwashing. Filter backwashes occur approximately eight times daily, each lasting from 6 to 8 min and using an average of 40,000 gal of water.

TABLE II.18. 100-N Area Liquid Effluents

<u>Material</u>	<u>Annual Quantity Used (lb)</u>	<u>Source</u>	<u>Discharge Point</u>	<u>Annual Average Flow (gal)</u>
Sodium Hydroxide	1,000,000	Used to regenerate anion exchange columns	Ground Disposal Facility	1.5×10^8
Sulfuric Acid	1,700,000	Used to regenerate cation exchange columns		
Aluminum Sulfate	380,000	Used as a coagulant in the water filtration process		
Polyacrylamide	1,800	Added to water filters to aid in retention of particulates		
Hydrazine	9,000	Added to reactor secondary cooling water as an oxygen scavenger	102-in. line	1.4×10^{11}
Morpholine	460	Added to reactor cooling water and boiler water to control pH	102-in. line	
Chlorine(a)	18,000	Used as a biocide in filtered water	102-in. line 1301-N Facility	9.1×10^8
Ammonium Hydroxide	53,000	Added to reactor primary cooling water for pH control	1301-N Facility	

(a) Chlorine discharge is through two different discharge points because of the various process uses.

Ammonium hydroxide is added to the primary and graphite coolant systems to control pH. The rate of feed varies to maintain the specified pH ranges. Morpholine is added to the reactor secondary cooling water and boiler water to control the pH. Hydrazine is added as an oxygen scavenger. The concentrations of morpholine and hydrazine in the cooling water are increased when the reactor is not operating.

Chlorine is used in the filtered water as a biocide. The concentration of free residual chlorine is maintained at 0.05 ppm in the filtered water. The spent chlorine is discharged to the crib and the river in the form of chlorides. Chlorine is also added to the sanitary (potable) water system for bacteria control. This water is discharged to one of two septic tanks and then to a tile field. Heated water is discharged to the center of the river via a 102-in. outfall line.

Airborne Effluents

Nonradioactive materials discharged to the atmosphere from 100-N Area consist mainly of combustion products from the diesel and No. 2 grade oil used in the Foster Wheeler and two Combustion Engineering boilers located in the 184-N powerhouse. During CY-1978, approximately 4,011,000 gal of No. 2 fuel oil and 495,700 gal of diesel oil were consumed in the 100-N Area and resulted in the atmospheric discharges shown in Table II.19.

The effluent air flow rate from the 184-N powerhouse is estimated to be $1 \times 10^{10} \text{ ft}^3/\text{yr.}^{(2)}$ Measurements of the SO_2 and particulate concentrations in the 184-N Foster Wheeler stack, when corrected to 7% O_2 , range up to 920 ppm and 0.098 grains/SCFD (standard cubic foot dry air) respectively. Measurements of the SO_2 and particulate concentration in the effluent air from the Combustion Engineering boiler stacks, when corrected to 7% O_2 , range up to 770 ppm and 0.05/grains/SCFD, respectively.

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TABLE II.19. Calculated Emissions to the Atmosphere from Oil Combustion

<u>Material</u>	<u>Total Released from 100-N Oil Fired Boilers (lb/yr)</u>
Particulates	99,400
Sulfur Dioxide	1,102,000

Source: "Compilation of Air Pollutant Emission Factors", Environmental Protection Agency, 1973.

Washington State Air Quality Standards specify maximum allowable concentrations of 1,000 ppm of SO₂ and 0.10 grains/SCFD of particulates in the emissions from large combustion sources.⁽³⁾

Solid Waste

Approximately 50,000 cubic feet of nonradioactive solid waste generated annually in the 100-N Area is disposed of in the Hanford Central Sanitary Landfill. The majority of this waste consists of paper from office trash cans. The volume of the waste is reduced by approximately a factor of 3 through the use of compactor trucks.

105-KE Area

The chemicals used in the 105-KE fuel storage facility are for control of the cooling pool water pH and biocides for control of algae and other organisms. These chemicals are introduced into the recirculated basin coolant water, which is not discharged to the environment. Liquid alum, a flocculation aid (Seperan) and chlorine are used in water treatment plant. The alum and filter aid are combined with water impurities in the form of floc and are retained by the filters. This floc is later washed off the filters during the backwashing operation. The backwash water is routed to a concrete basin where particulate material settles out prior to river release in compliance with NPDES Permit No. WA-000374-3.

II.3.1.7 Effluent and Process Monitoring

100-N Area

ERDAM-0513 (Effluent and Environmental Monitoring and Reporting) requires a comprehensive radioactive effluent sampling program. The NPDES permit covering N Reactor liquid effluents requires sampling or monitoring of all liquid discharges from the 100-N Area to the Columbia River to verify compliance with the permit. State air quality standards also require sampling of certain effluents to the atmosphere. A comprehensive program has been established to fulfill these requirements. See ERDA-1538, Appendix III-G for biota monitoring.

Radioactive Effluent Monitoring and Sampling Program

Approximately 1,100 samples per year are taken of N Reactor effluents. Measurements are made of the radionuclide concentrations in effluent air and water. Analyses of the samples are performed by U.S. Testing Company and UNC. The following samples are routinely collected and analyzed.

- Liquid Discharge to the 1301-N Facility. Liquid discharge to the 1301-N facility is monitored continuously for flow. The flow monitor reading is sent to a flow proportional composite sampler. A small aliquot of the effluent stream is piped into a sample container after a preselected quantity of effluent has been discharged to the 1301-N Facility. The composite sample is then collected and analyzed once per week for radionuclide concentrations.
- Riverbank Springs. The radioactive liquid effluent that seeps through the crib soil column to the river appears at distinct riverbank springs. The springs are sampled by a timer-controlled composite sampler. A small aliquot of the effluent is piped to a sample container at preselected time intervals. The composite sample is then collected and analyzed once per week for radionuclide concentrations.
- 102-in. Discharge Line. The 102-in. line is equipped with a sampling pump that normally pumps 100 gal/week of water from the discharge line. The

large volume water sample passes through several particulate filters and resin beds that concentrate the radioactive material present in the sample water. The 102-in. line is continuously sampled with weekly replacement and radioanalysis of ionic resins.

Temperature and flow in the 102-in. line are measured and recorded in the 105-N control room. In addition, the temperature differential between the 102-in. discharge and the Columbia River is monitored.

- Airborne Effluents. Over 99% of N Reactor's airborne effluent is discharged through the 116-N stack, the 109-N Zone I cell 6 vent and the 109-N Zone I building vent. These discharge points are provided with continuous sampling systems. The sampling systems, located at the effluent discharge points, consist of an insulated, heat-traced, 1/2-in. stainless steel line that draws an isokinetic air sample (where feasible) from the center of each stack through a quick disconnect stainless steel sample cartridge containing: (1) membrane particulate filter, (2) 1-in. charcoal bed (GE 602 charcoal) and 2-in. charcoal bed (GE 618 charcoal). The sampling systems also contain a water cooling coil that condenses water vapor for tritium analysis. The sample cartridges and condensate are removed once per week for analysis.
- Monitoring of Radioactive Waste Disposed of in the 200 Area. Before releasing from 100-N Area the liquid waste associated with N Reactor decontamination or the solid waste, waste volumes and radioactivity levels are measured onsite. The radioactive waste is then transported to the 200 Area for disposal. Disposition of waste in the 200 Areas is discussed in Section 11.1 of ERDA-1538.

Fuel Cladding Failure Detection System

The N Reactor fuel cladding failure detection system is designed to monitor samples of effluent from each of the 1,003 reactor pressure tubes for traces of fission products indicative of a fuel cladding failure. The system can reliably detect a failure within 10 min, including a sample transit time of 45 to 75 sec, for concentrations of uranium and fission products greater than 1.0×10^{-7} g/ml.

Upon detection of a fuel cladding failure, the reactor is manually scrammed, and the failed element is discharged from the reactor. In addition, sampling frequencies are increased both at the point of discharge of primary coolant water and storage basin bleed to the 1301-N Facility and at the 100-N riverbank springs following a fuel cladding failure incident. The increased sampling program enables continuous tracking of the onsite and environmental discharges resulting from the fuel cladding failure.

Nonradioactive Effluent Monitoring

Nonradioactive effluents from N Reactor are currently monitored and sampled as required by the NPDES Permit. In FY-1978, automatic equipment was installed to facilitate the program. The analysis of nonradioactive samples is performed by Hanford Environmental Health Foundation (for suspended solids and grease), UNC (for pH and residual chlorine) and U.S. Testing Company (for slightly radioactive liquid samples). NPDES permit related monitoring equipment includes:

- 181-N Inlet Screen Backwash and Fish Return. A locally indicating, totalizing type of flowmeter is installed on the 3-in. fish return supply. Running time meters are installed on the backwash pump motor starters.
- 182-N Storage Tank Farm Overflow. The flow in the 36-in. drain line is measured by an ultrasonic level sensor. The level is converted to flow, totalized, and recorded by an instrument in an enclosure located inside of the perimeter fence. The discharge temperature is monitored continuously at the same location.
- 182-N Pumphouse Filtered Water and Floor Drain. The flow from the 42-in. drain line is collected in a funnel and run through a totalizing flowmeter that transmits the flow signal to an indicator located inside the perimeter fence.

- Temperature and flow in the 102-in. line are measured and recorded in the 105-N control room. In addition, the temperature differential between the 102-in. discharge and the Columbia River is monitored. A pH meter also monitors the 102-in. discharge.
- Discharge from the two boiler stacks are periodically sampled in accordance with the state air quality standards. The discharges are in compliance with the standards.

Table II.20 lists each of N Reactor's effluent (including radioactive) release points and the current sampling program.

105-KE Area

A routine liquid effluent sampling program is conducted to monitor the water discharged from the secondary cooling system; results routinely show natural background levels.

The secondary coolant discharge from the heat exchanger is sampled weekly for chlorine. The clarified backwash water from the sedimentation basin is measured, on a weekly basis, for suspended solids. The combined 100-K Area discharge is continuously monitored for flow and measured weekly for temperature and for pH.

A routine effluent air sampling program is conducted to monitor the air exhausted from the facility; results routinely show natural background levels.

II.3.1.8 Modifications to N Reactor and Facilities

Improvements Completed

100-N Area

Planned waste management improvements discussed in ERDA-1538 (pages V-16 and V-38) have been completed or essentially completed. The projects, along with their costs, are as follows:

TABLE II.20. N Reactor Effluent Monitoring and Sampling Program

<u>Nature</u>	<u>Released to</u>	<u>Discharge Point</u>	<u>Measurements</u>	<u>Periodic Sampling Program</u>
Chemical	Land	1301-N Facility	Flow	Semi-Annual Chemical Analysis, pH
		66-in. Line	Flow, Temperature	Semi-Annual Chemical Analysis, pH
	River	102-in. Line	Flow, Temperature, pH	Semi-Annual Chemical Analysis
		N-Springs	None	Monthly Chemical Analysis, pH
		Foster Wheeler Boiler Stack	Flow, % Opacity	Annual Particulate, SO ₂
		Combustion Engineering Stack	Flow, % Opacity	Annual Particulate, SO ₂
Sanitary	Septic Tank	N-Area Tank	None (22,500 gpd)	None
Thermal	River	102-in. Line	Flow, pH, Temperature	Free Available Chlorine (monthly)
		Water Storage Farm Overflow	Flow Temperature	Total Oils and Grease (monthly) Free Available Chlorine (monthly)
Solids	Land	Central Landfill (2.5 acres/year) Industrial Waste	None	Suspended Solids (weekly) pH (weekly)
Particulate	River	182-N Filtered Water and Floor Drain	Flow	Monthly Suspended Solids and Total Oil and Grease
		Inlet Screen Wash Water and Fish Return	Flow	Monthly Suspended Solids
Radioactive	Land	1301-N Facility (liquid)	Flow	Proportional to Flow Composite Sampler, Analyze Weekly for Radionuclides
		102-in. Line	Flow	Nuclides Large Volume Sampler Used
		N-Springs	None	Weekly Analysis for Radionuclides
		182-N Raw Water Drain		Annual Analysis of Gamma Emitters
	River	182-N Tank Farm Flume		Annual Analysis of Gamma Emitters
		116-N Stack	Flow, Temperature	Weekly Radionuclide Analysis
		Cell 6 Stack	Flow, Temperature	Weekly Radionuclide Analysis
		109-N Zone I Stack	Flow, Temperature	Weekly Radionuclide Analysis
		109-N Zone II	Flow	Quarterly Radionuclide Analysis
		EF-7 ^(a) , EF-8		
		109-N Zone III	Flow	Quarterly Radionuclide Analysis
		EF-10		
		EF-13		
		109-N Zone IV	Flow	Quarterly Radionuclide Analysis
		EF-14, EF-15		
		105-N Dummy Spacer Decontamination Facility Exhaust	Flow	Quarterly Radionuclide Analysis
		105-N 14 ft Decontamination Facility	Flow	Quarterly Radionuclide Analysis
		105-N Transfer Area	Flow	Quarterly Radionuclide Analysis

^(a) Ventilation exhaust fan numbers.

N Reactor Gravity Drain and Disposal Basin. Final cost, - \$493,000. The gravity drain feature of this project was changed to a diesel drive pump system.

N Reactor Ventilation Loop Seal Improvement. Final cost, \$210,000.

Other waste management improvements made during the interim period following preparation of ERDA-1538 include:

1314-N Tank Car Loading Station Modification, \$12,000. Modification of the 1314-N tank car loading station has been completed to minimize the potential for spill of radioactive contaminated solution during the tank car loading operation and to improve control of spills that do occur. Tank car liquid level and dome pressure instrumentation and an overflow catch tank have been installed. Local alarms and annunciation in the reactor control room have been provided for the liquid level and dome pressure instrumentation.

1310-N Waste Storage Facility, \$219,000. A precision level measuring device has been installed to read the liquid level in the 1310-N tank. The new gauge has an accuracy of ± 100 gal when the spherical tank is half full as compared to the previous accuracy of $\pm 1,200$ gal. Remote readout has been provided in the reactor control room. This greater accuracy will aid in detection of leakage in the tank or connecting piping should such a leakage occur. Cost of the installation was approximately \$5,000.

An epoxy protective coating, 60 mils thick, was applied to the interior below grade surfaces of the 1310-N tank to enhance the integrity of the tank walls. Cost of this treatment was approximately \$214,000.

RDR Valve 802-2 Valve Elimination, \$6,400. The 802-2 valve allowed diversion of the 109-N radioactive drain directly to the Columbia River instead of the 1301-N Facility. The valve was removed and the lines blanked during the 1975 summer outage to ensure that this potential source of radioactive material discharging directly to the river was eliminated.

36-in. Drain Line Isolation in Lift Station Pump Well, \$4,500. The 36-in. drain line from the 105-N lift station pump well to the Columbia River was blanked immediately downstream of the drain line valve during the 1975 summer outage. Installation of the blanking plate eliminates the pump well as a potential source of radioactive material discharging directly to the river.

Revised Horizontal Control Rod Heat Exchanger Coolant Drain System, \$35,000. The left and right side horizontal control rod coolant system heat exchanger drains were modified during the 1975 summer outage to segregate the primary and secondary cooling effluents. The primary coolant bleed now discharges to the 1301-N Facility, and the secondary coolant discharges to the Columbia River. This modification eliminates an inadvertent discharge of radionuclides directly to the Columbia river during emergency once-through cooling.

N Plant Filtered and Demineralized Water Plant Modifications, \$446,500. The N Plant filter backwash, ion exchange resin regenerative waste and the coagulator sludge sump waste formerly discharged impurities removed from the water along with water treatment chemicals to the Columbia River. The water plant uses gravity flow filters to produce filtered water. The six filters are each backwashed about every 16 hr to remove filtered solids.

The 163-N demineralized water plant cation resins are regenerated with a sulfuric acid solution and the anion resins with a sodium hydroxide solution. The spent chemicals were then discharged to the river.

The 183-N sludge sump receives sediment and water from the bottom of the coagulator ponds. The filter backwash waste, the resin regeneration waste and the coagulator sludge sump waste discharged approximately 8,000 lb of chemicals per day to the river via the 102-in. and 66-in. raw water return lines.

This project provides a new concrete sump and settling pond system. The N Plant filter backwash was diverted to the new settling pond. The 163-N ion exchange resin regenerative waste and the 183-N coagulator sump waste are pumped to this new settling pond. The clarified water seeps into the ground rather than being discharged to the river. Beneficial use of this project was achieved on September 2, 1977.

66-in. Raw Water Return Diversion Into the 102-in. Outfall Line, \$36,500.

The 66-in. raw water return line formerly discharged into a seal well and then to a concrete flume which extends to the shoreline of the Columbia River. The flow through this flume could reach 82,000 gpm during N Reactor operation. This shoreline discharge of heated liquid was terminated by July 1, 1977. To accomplish this, the seal well was modified to divert the 66-in. line seal well to overflow into the adjacent seal well and 102-in. outfall. In the event of an emergency (such as a blockage of the 102-in. outfall line) the combined flow of both the 102-in. and 66-in. lines will still be able to overflow into the concrete flume.

Foster Wheeler Boiler Modification, \$735,000. The N Plant boiler and steam system were modified to permit operation in a more efficient range of combustion. Various items of steam power equipment were modified to use reactor-generated steam (125 psig) instead of the 450-psig boiler-generated steam. This project was completed in September 1977. These modifications will result in a 30% decrease in future oil use, improved operating efficiency and reduced emission concentrations even further below state air quality standards.

105-KE Area

100-K Area Filter Backwash Water, \$148,000. The backwash water from the 183-KE water treatment plant contains suspended solids removed from raw river water and coagulant used in the water treatment process. This was formerly discharged directly to the Columbia River during filter backwashing. Completion of this project resulted in the backwash water being pumped to a concrete basin where the particulate material is allowed to settle out of the water.

The clarified water is then discharged to the river. Beneficial use of this project was attained on June 14, 1977.

Fuel Storage Facility. Final cost, \$2,526,000. The 105-KE fuel storage facility has been modified to provide for the safe storage of approximately 1900 tons of irradiated N Reactor fuel. A recirculating coolant system equipped with an ion exchange system, a sand filter, bypass, in-line filters, heat exchangers and instrumentation to monitor radiation levels, heat generation rate and pool water level fluctuations has been installed in the basin of 105-KE. This was originally a once-through system. Storage racks for the individual N Reactor canisters have also been installed on the floor of the basin.

Improvements Planned or Under Construction

100-N Area

Major improvement modifications proposed for the period from CY-1979 to 1981 and their estimated costs are described below.

N Reactor Irradiated Fuel Storage Basin Recirculation Coolant System, \$5,100,000. This project, currently scheduled for completion by the fourth quarter of CY-1981, allows the N Reactor irradiated fuel storage basin to be converted from the present single-pass cooling mode of operation to a recirculation cooling system mode of operation to reduce the volume of radioactive effluent currently being discharged from the N Reactor irradiated fuel storage basin to the 1310-N Facility. The storage basin recirculation coolant system will be designed to provide pumps, filters and heat exchangers needed to maintain the storage basin water temperature at less than 100⁰F during periods of maximum decay heat load and a level of water quality necessary for underwater handling and packaging of the irradiated fuel elements. An ion exchange system will be installed to control the storage basin water chemistry and to remove soluble radionuclides for personnel radiation exposure control.

N Reactor Fuel Cladding Failure Detection Sample Water ReInjection, \$800,000. This project provides further reduction in the volume of radioactive liquid effluents discharged to the 1301-N Facility from the N Reactor fuel cladding failure detection sample by reinjecting the reactor fuel cladding failure detection sample water back into the primary coolant system. The reinjection system will consist of pumps, associated piping, fittings and valves. Flow, differential pressure and temperature of the water will be monitored locally and also monitored and recorded in the 105-N control room. Two reinjection systems are needed, one for the left and one for the right side fuel cladding failure detection systems.

This project is proposed for completion by the fourth quarter of 1981.

1301-N Trench Cover, \$950,000. The presence of permanent water in the 1301-N trench enhances the growth of vegetation, which in turn attracts wildlife. The trench is surrounded by a cyclone fence and covered with a mesh screen to prevent larger animals and waterfowl from approaching and entering the trench. A solid cover is necessary to preclude possible entry by smaller animals and birds. A pressurized-air cover will be installed in FY 1981.

II.3.1.9 Reactor and Fuels Decommissioning Program Program Development

The Hanford decontamination, decommissioning and disposition (D&D) program was initiated in 1964 when F Reactor was shut down. F Reactor was the first of eight special nuclear materials Hanford reactors that were shut down between 1964 and 1971. All eight shutdown reactors are currently in retired status.

Concurrent with the 1971 shutdown of the eight special nuclear materials reactors, that portion of the Fuels Fabrication Facilities used to produce fuel for the eight reactors was also shut down.

The first step in the D&D for both the retired reactor and fuels facilities is called "lay away". In this step the radionuclides that are or could become loose are removed from equipment by decontamination and stabilized. Detailed radiation surveys are made, and a maintenance/surveillance program with procedures is documented for use in facilities that do contain stabilized radionuclides.

The second program step is to fully characterize radionuclides by location, matrix and quantity. The purposes of this step are: (1) establish the disposition of facilities and equipment that require radiological control, and (2) provide a basis for environmental assessment and safety analysis reports for dismantling and disposition of facilities.

The third step, decommissioning analyses, establishes the methods, costs and priorities for D&D of the facilities. This step involves extensive engineering analysis and evaluation of the facilities. The analyses include an environmental assessment and a safety analysis to insure that alternative methods, timing, costs and priorities consider all program aspects prior to final development and implementation.

The fourth program step accomplishes facility dismantling and disposition. This step is implemented using the detailed procedures developed in the preceding step.

The final step is site restoration. The objective of this step is to return the site to as near natural conditions as is practically achievable. The long half-life radioactivity will be removed to achieve unrestricted use levels. The remaining short half-life radioactivity will also be removed to the extent that the remaining activity will decay to unrestricted use levels in a few years.

The radionuclides in the eight retired reactors and in the shutdown fuel facilities have been stabilized and the radionuclides characterized. Thus, the first two steps of the D&D program have been completed for these facilities. All of the retired facilities are under a facility surveillance and maintenance program designed to insure that the (1) radionuclides remain stabilized and (2) facilities do not deteriorate such that they could impose safety problems. Thus, corrective maintenance action, based upon surveillance results, would be taken before a shutdown facility could become an onsite or offsite problem.

Each of the shutdown facilities are at various stages of completion within program step 3, decommissioning analyses. Some of the reactor facilities and equipment that contained no radionuclides have been removed.

In June of 1974, under the direction of ERDA, Battelle Memorial Institute Pacific Northwest Laboratories (PNL) became the program coordinator for D&D of all Hanford shutdown facilities. The purpose of this program is to establish the methods, costs and priorities for D&D. Phase I of this program includes the preparation of a D&D resource book for shutdown of Hanford facilities. The resource book (Ref. 18) is currently under development and will become a working document that will be modified and updated as new findings and methods are developed.

The DOE increased the scope of its D&D program to a national basis in fiscal year 1978. This program will assist in establishing national D&D priorities and provide a broad exchange of methods and procedures from the various DOE sites with nuclear facilities.

D&D Program Application to N Reactor and N Fuels Facilities

Upon shutdown of N Reactor and its Fuels Fabrication Facilities, the radionuclides would be stabilized and facility characterization would be initiated. The same general methods used successfully in earlier D&D program work would be adapted to the N Reactor and its fuels fabrication facilities.

The plan objectives for N Reactor and its ancillary facilities would include: (1) decontamination of all locations that contain smearable quantities of

radionuclides to specified levels, (2) removal of all spent fuel for storage and/or reprocessing, (3) stabilization of the 1301-N waste storage facility to allow radionuclide decay prior to long-lived radionuclide removal, and (4) characterization of all locations having radionuclides for the development of detailed area restoration procedures.

The plan objectives for fuels fabrication facilities would include:

(1) decontamination of uranium to specified levels, (2) removal of all new fuel for recycling, and (3) characterization of all locations containing measurable quantities of uranium for development of detailed restoration procedures.

II.3.2 Fuels Fabrication Facilities

II.3.2.1 General Description

The Fuels Fabrication Facilities are located in the 300 Area about one mile north of the Richland city limits on the west bank of the Columbia River. Roughly rectangular in shape, the area includes about 375 acres. The aerial photo, Figure II.25, shows that the area is bounded by the Columbia River on the east and the highway on the west.

Most facilities in the 300 Area, completed in 1943 and the years immediately following, were related originally to the fabrication of reactor fuels. These included many technical and service support facilities as well as a fuel manufacturing facility. As the Hanford production reactors were shut down, fuel manufacturing activities decreased. Over the past 15 years, research and development programs have become an increasingly important part of the activities in the 300 Area. The newer facilities house mostly laboratories and large test facilities in support of the peaceful use of plutonium, reactor fuels development, liquid metal technology, fast flux test reactor support, gas-cooled reactor programs and life sciences programs operated by Battelle-Northwest and Westinghouse Hanford Company.

Although most of the 300 Area buildings are industrial-type single story structures with widely varying construction, the appearances are compatible. The newer buildings harmonize with the landscape. Fuels Fabrication Facilities are located in Buildings 303, 304, 311, 313, 333, 334, 3701, 3707, 3710, 3712, 3713, 3716, 3717 and 3734 (see Figure II-30). These buildings and associated production activities account for the major part of the routine or potential environmental impact from this area.

II.3.2.2 Operational Programs

The Fuels Fabrication Facilities requirements for the anticipated operating program for N Reactor are approximately 328 MTU in the near future.

FIGURE II.25. Fuels Fabrication-Aerial Photo

II.3.2.3 Operational Systems

Bulk Chemical Storage

The outdoor process and waste chemical storage tanks serving the Fuels Fabrication Facilities are all located above grade. Some of these tanks have catch basins installed to divert any spills or leaks to the pipe trench, which in turn drain into the process sewer. Sewer monitoring stations detect acid or caustic leaks. The other tanks have concrete dikes with impervious bottoms installed to fully contain spills or leaks.

The following listed tanks have catch basins:

- 2 - 6,000-gal tanks for HNO_3 storage
- 1 - 6,000-gal tank for H_2SO_4 storage
- 2 - 10,000-gal tanks for NaOH storage
- 1 - 6,000-gal tank for waste acid storage

The tanks with diked containment are:

- 1 - 10,000-gal tank for perchloroethylene storage
- 1 - 4,000-gal tank for HNO_3
- 1 - 4,000-gal tank for neutralized acid waste
- 2 - 660-gal tank for uranium bearing waste
- 1 - 660-gal tank for used oil

Fuel Fabrication Process

Fuels Fabrication Facilities use a variety of mechanical, chemical and electrical processes in the conversion of uranium billets and assorted components to finished fuel elements ready for charging into N Reactor. The finished, assembled fuel elements are a tube-in-tube design with a metallic uranium core surrounded by a thin Zircaloy-2 cladding (see Figure II.26). The inner fuel tube is enriched to 0.95% ^{235}U . The outer fuel tube is enriched to either 0.95% ^{235}U or 1.25% ^{235}U . All actual fuel fabrication activities are carried on in the 333 building, although certain auxiliary procedures such as component fabrication and cleaning, chemical waste treatment, uranium recovery,

uranium and beryllium Zircaloy-2 scrap concretion and uranium scrap packaging are performed in Buildings 313, 304 and 3712.

FIGURE II.26. Assembled Inner and Outer
N Reactor Fuel Element

The fuel fabrication process consists of the sequence of operations shown in the process flow diagram in Figure II.27. The more significant subsystems are described below in the same order.

Component Cleaning

Uranium billets, Zircaloy-2 cladding shells and copper-silicon jackets preshapes and end plates are subjected to cleaning operations that include vapor degreasing in perchloroethylene, acid etching and cleaning, process and deionized water rinses and warm-air drying. Solutions routinely used in component cleaning include nitric acid, nitric-hydrofluoric acid, and chromic-nitric-sulfuric acid. Fumes from the cleaning tanks are removed by the central chemical bay exhaust system that discharges vertically 140-feet above ground level.

FIGURE II.26. N-Reactor Fuel Fabrication Process

Billet Assembly

The uranium core and components are assembled into billets immediately after cleaning to prevent oxidation, which would inhibit clad-core bonding. The copper-silicon end plate-to-jacket joints are fusion welded, helium-leak tested and sealed.

Occasionally materials other than fuel production billets are processed to form tubes or rods of copper, titanium, aluminum, copper-silicon alloys, and zirconium alloys. Billet assembly techniques for these materials are generally similar to techniques for uranium billets.

Billet and Tooling Preheat

Billet assemblies are preheated in four hot-air, pit-type, electrical furnaces capable of reaching 1450 F temperatures. Extrusion dies, mandrels, backing plates and other tools are also preheated in an electric furnace.

Extrusion

Preheated billet assemblies are extruded to the tubular fuel configuration in a 3,000-ton horizontal extrusion press, which has a stroke of 66 in. and a maximum speed of 30 in./min. The copper-silicon can that surrounds the uranium billet and Zircaloy-2 cladding serve the dual purpose of preventing oxidation of the metallic surfaces during heating and acting as a lubricant during the extrusion process.

Tube Cutoff

The extruded fuel tubes are cut into pieces of fuel element length by two semiautomatic abrasive cutoff saws. The sawing is performed under water to prevent burning of the small pyrophoric chips (fines) that are produced.

Saw fines are collected primarily in the saw tank. Some fines carry over in the water to a section of process trench that is equipped with a weir and stilling area where the dense fines settle to the bottom for periodic removal. All fines are then stored in water-filled 30-gal drums.

The saws are provided with shrouds and exhaust ducts connected to a counter current scrubber and 15-horsepower exhaust fan. Air is continually drawn past the water-cooled cutting zone and through the scrubber to prevent release of smoke or airborne uranium particles to the work areas.

The scrubber, rated at 7,200 cfm, is a packed-column type with a 5-ft-deep bed of Tellerette packing. A once-through water flow of approximately 30 gpm is routed to the same settling trench used for the saws. Interlocks are provided to prevent saw operation if the exhaust fan is not running and prevent exhaust fan operation if the scrubber water is turned off.

Counterboring

Both ends of the cut fuel pieces from the cutoff operation are machined simultaneously on an automatic, double-spindle, horizontal boring machine (counterbore lathe) to provide end recesses for installation of Zircaloy-2 end caps. The counterbore lathe has an integral coolant recirculation and chip separation system where chips are collected for periodic removal and storage in water-filled 30-gal drums.

Air is continually drawn from shrouds around the cutter heads by a centralized beryllium-uranium exhaust system. An interlock prevents operation of the machine if the exhaust system is not running.

Copper-Silicon Stripping

Following the counterbore operation, the fuel pieces are processed through hot nitric acid to remove the copper-silicon remaining from the billet jacket. This operation takes place in a covered 576-gal stainless-steel tank and is followed by a static water rinse.

Concentrated NO and NO₂ fumes generated by the reaction are drawn off below the cover and routed to a special 16-stage NO_x scrubber. The remaining fumes are then routed to the central chemical bay exhaust system. An exhaust hood behind the tank draws any fumes that escape from the tank to the same exhaust system.

Chemical Milling

A thin layer of uranium is left on the walls of the fuel ends in the counter bore operation and is removed by chemical milling. This operation is performed in a hot nitric-sulfuric acid bath in a special 910-gal stainless steel processing tank and is followed by static and flowing water rinses in stainless steel tanks.

The processing tank was designed to augment the 16-stage NO_x scrubber. A power-operated cover on each of the four compartments of the tank prevents air dilution of the concentrated fumes, thus enhancing scrubber efficiency. Fume exhaust hoods behind the tank draw any escaping fumes to the central chemical bay exhaust system. A leak catch pan and alarm are integral parts of the tank design.

Prebrazing Cleaning

The stripped and chemically milled fuel pieces are cleaned prior to installation of end caps by etching in nitric-hydrofluoric acid and in nitric acid-aluminum nitrate solution, followed by process water and deionized water rinses.

The cleaning facility consists of two polyvinyl chloride nitric-hydrofluoric acid etch tanks of 270- and 350-gal capacity, one 140-gal stainless steel nitric acid/aluminum nitrate tank, three stainless steel rinse tanks and a steam-heated warm-air dryer. Fumes from the cleaning line are removed by the central chemical bay exhaust system.

End Cap Braze

The recessed ends of the fuel pieces are sealed with Zircaloy-2 end caps that are vacuum brazed using a beryllium-Zircaloy-2 braze alloy.

Heat Treat

Following installation of end caps, the fuel pieces are heat treated to produce random orientation of the uranium grain structure and assure dimensional stability of the fuel during irradiation. The fuel pieces are heat treated in

a molten sodium-potassium chloride salt bath at 735°C (nominal), quenched in a sodium-potassium nitrate, sodium nitrite bath at 300°C to control the cooling rate, air cooled and finally quenched in water for further cooling and residual salt removal. Fumes from the heat treat operation are exhausted from the building through a 9,800-cfm rooftop exhaust vent.

End Machining

The ends of heat-treated fuel pieces are faced off in a lathe to remove excess clad and braze material and to prepare the fuel for end welding of the clad-to-end cap joint. Chips are periodically removed and stored in water-filled drums pending disposal by concretion and burial. A shroud encloses the cutting operation, and any smoke generated is routed to the central beryllium-uranium exhaust system.

Abrasive Blast

Following the end machining operation, the fuel pieces are vapor degreased in perchloroethylene to remove any oil and water. The fuel pieces are then air-abrasive blasted on both inside and outside surfaces with aluminum oxide grit to remove heat treat scale and any residual salt. Exhaust air from each machine is filtered through a fabric bag type of filter cabinet and routed through a cyclone separator prior to discharge to the atmosphere through a vertical stack 36 ft above ground. Exhaust flow is 1,900 cfm from the north machine and 1,600 cfm from the south machine.

Preweld Cleaning

The abrasive-blasted fuel pieces are etched in nitric-hydrofluoric acid, rinsed in process water and deionized water and dried in preparation for final end welding of the end cap-to-clad joints.

The preweld etch line consists of a 350-gal polyvinyl chloride tank containing nitric-hydrofluoric acid, three stainless-steel rinse tanks and a steam-heated warm-air dryer. Fumes from the line are removed by the central chemical bay exhaust system.

End Welding

The brazed end cap-to-clad joint is fusion (TIG) welded to provide the primary closure against water entry to the uranium.

An exhaust ring in the purge chamber continuously draws any beryllium and welding fumes into the beryllium-uranium exhaust system.

Autoradiograph

The final nondestructive test of the fuel element is a radiograph of the end weld. Waste solutions from the radiographic film developing process are discharged to the chemical waste system.

Support Welding

The individual finished fuel elements are equipped with support hardware to position and lock the inner and outer fuel elements together and to position the assembly in the reactor pressure tube. Four projection resistance welders are used to attach all support hardware. There are no fumes, smoke or waste products associated with this operation.

Final Cleaning

The fuel pieces are processed through one final acid cleaning operation to provide a clean nonoxidized surface, which promotes formation of a uniform oxide layer during autoclaving. The pieces are vapor degreased in perchloroethylene, water rinsed, etched in nitric-hydrofluoric acid, rinsed in process water and deionized water and stored in deionized water-filled drums until loaded into the autoclaves.

Fumes from the etch tank are removed by the central chemical bay exhaust system.

Autoclave

Fuel pieces are subjected to a 750°F, 150 psig steam environment for 48 hr in 12 electrically heated autoclaves for the dual purpose of proving the integrity of the cladding and developing a protective black zirconium oxide film on the surface.

Final Assembly

Following autoclave processing the fuel pieces are given a final inspection and joined into fuel element assemblies consisting of an inner and outer fuel piece that are locked together. Lubricant used during assembly is removed by manual steam cleaning with deionized water.

Component Cleaning

The component cleaning facility in the uranium recovery room of Building 313 is used for nonroutine or nonfuel production related cleaning jobs. The facility consists of a perchloroethylene vapor degreaser, a steam-heated warm-air dryer, two 280-gal polyvinyl chloride-lined stainless-steel nitric acid tanks, two stainless-steel water rinse tanks and a polyvinyl chloride waste acid transfer pump.

Fumes emanating from the tanks are removed through hoods at the rear of the tanks, scrubbed out in a water spray scrubber and exhausted through a vertical stack about 33 feet above ground level.

Copper-Silicon Casting Facility

A copper-silicon casting and machining facility for the production of billet preshapes and backing plates is contained in Building 313 and used in the extrusion billet assembly. The facility includes lathes, casting furnaces, ovens, saws, sanders and shears.

A central exhaust system removes gases and fumes from around the furnace and removes smoke and particulates from the band saw and grinder. This central exhaust system exhausts vertically about 30 feet above ground level.

Concretion Facility

Uranium and Zircaloy-2 fines and chips are encapsulated in masonry concrete billets in Building 304 for shipment to National Lead Company of Ohio^(a) or for burial. Equipment includes a cement mixer, scale and a cyclone-type dust collector.

(a) National Lead Company of Ohio at Fernald, Ohio, is the feed site for DOE uranium.

Solar Evaporation Facility

Four deactivated water treatment basins located in 100-H Area have been designated for use as solar evaporation basins for neutralized chemical waste. The useful capacity of each basin is 317,000 gal. The entire group of four basins is surrounded by a 6-ft-high chain link fence topped by three strands of barbed wire.

Two basins have been modified for chemical waste use by permanently plugging the drain, installing a tank truck discharge line and stringing wire over the top at 6-ft intervals in both directions to discourage waterfowl landings. The other two basins will be modified similarly and placed in service as needed.

Figure II.28 shows a view of one of the evaporation basins.

Sampling Data Used To Determine Uranium Inventory

The 300-Area process trenches are sampled weekly. A monthly composite sample is used to determine uranium releases. All uranium contributions to the process ponds and trenches are assumed to be from the Fuels Fabrication Facilities since none of the other 300-Area contractors handle significant quantities of uranium.

The 100-H solar evaporation basin uranium values are calculated for each load of solution transported to the basin. A sample is taken during the loading of the waste tanker from the pump discharge line.

II.3.2.4 Water Use, Services and Utilities

The 300-Area pumphouse supplies all water needs for the 300-Area from the Columbia River. Maximum and average water usage rates for the Fuels Fabrication Facilities in the 300 Area, in gallons per day, are given in Table II.21.

FIGURE II. 28. Solar Evaporation Basin

TABLE II.21. Fuels Fabrication Facilities Water Usage

Water <u>Inflow (gpd)</u>	<u>Water Outflow (gpd)</u>	
	<u>Process Sewer</u>	<u>Sanitary Sewer</u>
640,000 maximum	630,000 maximum	11,000 maximum
550,000 average	540,000 average	9,900 average

Water meters in Building 333 and in Building 313 are installed in each main water inlet line and are used to monitor water usage. Practically all of the water used in Fuels Fabrication Facilities in these buildings is for scrubbing, cooling, rinsing and other miscellaneous process operations. Maximum water usage occurs during day shift operation of fabrication facilities. The diagram in Figure II.29 shows maximum water usage rates for each of these operations. Used process water then drains into process sewer system (Figure II.30). Likewise, sanitary sewage flows into the sanitary sewer system (Figure II.31).

The cutoff saw exhaust from Building 333 is sampled by an isokinetic sampler installed in the exhaust stack. The sample filter paper is changed each working day, and the alpha and beta activity is counted in a Nuclear Measurements Corporation sample counter. Uranium releases are calculated using the counter results and the flow in the exhaust stack. Solid uranium scrap, uranium chips and saw fines are accounted for by using actual weights and uranium-to-Zircaloy-2 ratio factors established by Production Planning. Uranium-extrusion end crops, which contain uranium, copper-silicon, Zircaloy-2 and graphite, are accounted for on a "by-difference" weight -- the average billet factor weight minus all the measurable losses and quantities that are accounted for.

Sodium diuranate sludge is sampled after the sludge cake has been removed from the filter press and loaded into 30-gallon drums. Each drum is sampled and a composite sample is analyzed for uranium content. The uranium content percentage is applied to the net weight of the sludge from press clean-out for total uranium weight. Used filter press canvas containing uranium sludge is sampled to determine a factor weight for the uranium content.

FIGURE II.29. Maximum Water Usage in Fuel Fabrication Facilities (gpd)

FIGURE II.30. Process Sewer System
(Fuels Fabrication Facilities
are shaded)

FIGURE II.31. Sanitary Sewer System
(Fuels Fabrication Facilities
are shaded)

Finished fuel factor weights are determined for all models by statistically sampling and weighing the elements in water and air.

In-process storage of uranium fuel is carried on the books at the established factors for the various models and for the manufacturing step.

Uranium Material Flow Paths

Uranium billets are received by the Fuel Fabrication Operation from RMI Company of Ashtabula, Ohio. Current delivery is by exclusive-use motor freight. The billets are shipped in sealed wooden containers and are nearly always free of any smearable or loose oxides. After statistical weighing and dimensional measurements, each billet is enrichment tested, and the billets are then returned to their shipping boxes and stored in warehouses until scheduled for processing in Building 333.

Uranium processing and effluent sources follow four principal material flow paths as related to fuel manufacturing (Figure II.32). These are finished fuel, in-process storage, scrap returned to National Lead Company of Ohio for recovery and waste streams.

Scrap Returned to National Lead Company. Uranium metal scrap and C-6 sludge from uranium-bearing acids are returned to National Lead Company of Ohio for reprocessing. During CY-1978, 165,950 lb of scrap were returned. Uranium scrap sources include uranium chips and saw fines, 23,559 lb; solid metal scrap, 135,580 lb; and sodium diuranate (C-6) sludge, 6,811 lb.

Table II.22 summarizes the uranium activity in all waste streams from the Fuels Fabrication Facilities. (The weight of uranium in these waste streams is indicated in tables in Section II.3.2.6.)

DISPOSITION OF URANIUM IN FUEL FABRICATION PROCESS

YEAR - 1978

URANIUM BILLETS TO PROCESS

522,359 lbs

FIGURE II.32 Uranium Flow Path in Fuels Fabrication Process CY-1978.
(This figure presents the uranium flow and inventory in the Fuels Manufacturing Process during CY-1978.)

TABLE II.22. Summary of Uranium Activity Discharge from 300-Area
Fuels Fabrication Facilities in CY-1978

	<u>Volume</u>	<u>Uranium Activity</u>
Gaseous Discharges		
Cutoff Saw Exhaust	$2.9 \times 10^8 \text{ ft}^3/\text{yr}$	
Total Discharged ^(a)		$3.3 \times 10^{-5} \text{ Ci/yr}$
Weekly Average		$9.7 \times 10^{-7} \text{ Ci/wk}$
Maximum Weekly Discharge		$5.1 \times 10^{-6} \text{ Ci/wk}$
Average Concentration		$1.1 \times 10^{-13} \text{ Ci/ft}^3$
Liquid Discharges		
Process Sewer	$5.0 \times 10^5 \text{ gpd}$	
Total Discharged ^(b)		$3.3 \times 10^{-2} \text{ Ci/yr}$
Average Concentration		$2.8 \times 10^{-10} \text{ Ci/gal}$
Solar Evaporation Basin	$150,000 \text{ gal/yr}$	
Total Discharged ^(b)		$5.1 \times 10^{-2} \text{ Ci/yr}$
Average Concentration		$3.4 \times 10^{-7} \text{ Ci/gal}$

(a) UNI Reactor and Fuel Production Facilities 1978 Environmental Release Report, UNI-1261, March 1, 1979.

(b) Calculated using "one curie of natural U is equivalent to 3,000 kilograms or 6,615 pounds of natural U" from ERDA Appendix 0524, Annex A.

Note: These source terms are based on an operating time of 236 operating days per year.

II.3.2.5 Radioactive Effluents and Solid Waste Liquid Effluents

Chemical Waste Containing Uranium

A total of 560 lb of low-enrichment uranium was written off to uranium losses during CY-1978. A chemical waste disposal system is used to receive and dispose of all concentrated liquid chemical wastes including those containing uranium. The present waste handling system provides for collection (see Figure II.33), neutralization and transportation of the waste to concrete basins in the 100-H Area where the liquids evaporate to form a solid salt cake. Use of the solar evaporation process has eliminated the need to discharge chemical waste to the environment and allows for possible recovery of valuable chemicals such as copper salts. During CY-1978 this waste contained approximately 340 lb of uranium (5.1×10^{-2} Ci).

The only routine chemical waste or uranium particulates that enter the process sewer from Buildings 313 and 333 are from process water rinse tanks, air scrubbers, wash stations, cutoff saws and the concretion facility in Building 304. During CY-1978 this was estimated to be 222 lb of uranium (3.3×10^{-2} Ci).

Chemical solutions that contain appreciable amounts of uranium are collected in holding tanks, pumped to Building 313 and neutralized with sodium hydroxide. This forms a uranium precipitate of sodium diuranate. The precipitate is removed by filtering the slurry in a press-and-frame filter press and is shipped to National Lead Company of Ohio for uranium recovery. If the filtrate contains less than 200 ppm uranium, it is pumped to an outside storage tank for tank trailer shipment to the solar evaporation basin. When necessary, the filtrate is recycled to achieve the 200 ppm maximum uranium content so that the uranium can be shipped to National Lead Company of Ohio for recycle. Chemical solution wastes that contain no recoverable uranium are discharged to storage tanks, pumped to Building 313 for neutralization with sodium hydroxide and pumped to the outside tank for shipment to the solar evaporation basins.

FIGURE II.33. Waste Acid and Uranium-Bearing Acid Recovery System

Other aqueous chemical waste solutions, such as the x-ray film developer chemicals, the sulfamic acid film machine cleaner and the NaOH-KMnO_4 and EDTA (ethylenediamine tetraacetic acid) solutions used to decontaminate an autoclave following a fuel cladding failure, are transferred by barrels to Building 313 and added to the chemical waste system for disposal in the evaporation basin. The silver in the used x-ray developer solution is removed and recovered by iron displacement before the solution is transferred to Building 313.

Any large leaks in the uranium-bearing processing tanks, the holding tank or the transfer lines in the pipe trench (Figure II.34) would drain into the process sewer. Sewer monitor stations installed in the process sewers would detect such leaks.

Similarly, large leaks of chemical waste solutions containing no recoverable uranium from processing tanks, transfer lines or the outside overhead storage tank would drain into the process sewer. Two of these storage tanks are located in a 10-ft concrete pit within Building 334-A (see Figure II.34). Leakage from these tanks or lines is confined to the pit to determine the quantity of the leakage and appropriate disposal method.

All quantities of chemicals used or transferred are closely controlled. Any substantial loss of acid or caustic to the process sewer can be neutralized with the appropriate chemical to keep the process sewer trench at a pH of between 5 and 10 to prevent dissolution of copper and uranium compounds in the bottom of the process sewer trench. The 4,000-gal storage tank for neutralized waste, along with a 10,000-gal perchloroethylene and a 4,000-gal nitric acid tank, is surrounded by a dike with an impervious bottom that will hold 10,000 gal in the event of a tank failure. Waste that could leak into the diked area can be pumped into a tank trailer for disposal at the solar evaporation basin. The trailer is positioned over a catch pad that drains into the diked area.

Waste Water From Process Stations and Concretion Facilities

The cutoff, counterbore and heat treat area process drains consist of floor trenches that enter a sump before discharging to the process sewer. Weirs

FIGURE II.34. Outside Storage Tanks and Pipe Trenches

II-126

provide settling basins for suspended particles of uranium and other materials. These particles enter the trench system in the coolant from the three cutoff saws, the three spray rinse stations, the air wash on the air exhauster for the three cutoff saws and band saw and the two truck wash stations. Between different uranium enrichment processing runs, the accumulated fines in the trenches and sump are cleaned out, drummed and transferred to the concretion process for subsequent return to National Lead Company of Ohio for uranium recovery.

The only soluble non-radioactive wastes that routinely enter this drainage system are the nitrate and nitrite ions from the hot water rinse at heat treat (less than 5-1/2 lb nitrate ion, NO_3 , per day and less than 4 lb nitrite ion, NO_2 , per day).

When the water-filled barrels of uranium fines, uranium lathe chips, and beryllium-contaminated Zircaloy-2 lathe chips are received at the concretion facility, the water is poured out through a 50-mesh screen in the floor drain. The floor drain discharges into a metal catch pan within a water-filled settling basin before entering the process sewer. The area around the cement mixer is kept free of any spilled cement and fines by hosing down the floor and flushing the water through the screened floor drain. The screen in the floor drain, the catch pan and the settling basin are periodically cleaned of any debris, which is then included in the cans and barrels of concreted product.

Airborne Effluents

Ventilation systems are provided at process locations to collect and remove airborne uranium particulates and smoke and to discharge the filtered air outside the buildings (see Figure II.35). Building 333 uranium-contaminated exhaust systems are described below.

Uranium Cutoff Saw Exhaust A

An exhaust system equipped with a water spray scrubber is provided to remove the uranium-contaminated fumes and smoke from the three abrasive uranium cutoff saws and the band saw used for sectioning peel test samples. The exhauster discharges vertically 39 ft above ground. The discharge is continuously sampled

for uranium particles. The exhaust blower motor and water spray feed line are interlocked to the three uranium cutoff saws and the peel test band saw so that both the motor and the water spray must be operating before any of the four saws can be energized.

Beryllium Exhaust B

The beryllium exhaust system removes beryllium and uranium fumes from the end welders, the quality control metallurgy laboratory, the two end facing lathes, the counterbore lathe, the braze vacuum pumps and the beryllium maintenance hood. The air passes through a high-efficiency particulate air (HEPA) filter before being discharged vertically 34 ft above ground (8 ft above roof). The discharge is continuously sampled for beryllium and uranium particles.

The exhaust blower motor is interlocked to the end welders, quality control metallurgy laboratory equipment, the two end facing lathes, the counterbore lathe and the braze lines so that the motor must be operating before this equipment can be energized.

A gauge on the HEPA filters has low and high differential pressure alarm lights to indicate a plugged filter, filter breakthrough or a nonoperative blower. In CY-1978, the total quantity of beryllium-Zircaloy-2 braze rings used in the process was 1,330 lb, which averaged 5% beryllium.

Solid Waste

Solid waste contaminated with uranium or beryllium is packaged for transfer by truck to the 200-Area Plateau Disposal site. A total of 2,000 ft³ of solid waste was sent to the disposal site in CY-1978 and this waste contained less than 38 lb of beryllium in a beryllium-Zircaloy 2 alloy and trace amounts of nonrecoverable low enriched uranium.

II.3.2.6 Nonradioactive Effluents and Solid Waste

Liquid Effluents

100-H Area Solar Evaporation Basin

Most of the chemical waste generated in the 300-Area is handled in the chemical waste disposal system (see Section II.3.2.5) and is discharged into 100-H solar evaporation basins as shown in Table II.23.

TABLE II.23. Chemicals Discharged to 100-H Area
Solar Evaporation Basins, CY-1978

<u>Materials</u>	<u>Total Discharged (lb)</u>
Ammonium ion	250
Fluoride ion	9,400
Nitrate ion	154,400
Sulfate ion	46,400
Chromium	110
Copper	28,400
Manganese	270
Uranium	340

300-Area Process Trenches

The chemical waste and uranium particles that enter the process sewer from the Fuels Fabrication Facilities are shown in Table II.24. The 540,000 gpd of process water is diluted to 2,400,000 gpd in the 300-Area process sewer before being discharged into the two process sewer trenches.

TABLE II.24. Calculated Chemicals Discharged to the
300-Area Process Trenches, CY-1978

<u>Materials</u>	<u>Pounds Discharged</u>	<u>Average Concentration (ppm)</u>
Nitrate ion	39,200(a)	39.76
Fluoride ion	960(b)	0.98
Copper	370(c)	0.37
Uranium	220(d)	0.22

- (a) Based on nitric acid (HNO_3) use plus nitrate salt from hot water rinse at beta heat treat minus equivalent amount shipped to evaporation basin, amount discharged as nitrogen dioxide (NO_2) out chemical bay stack and amount in filter press sludge shipped to the feed site.
- (b) Based on 1.18×10^8 gallons flow into process trench and average fluoride ion (F^-) content into process trench minus average F^- content in inlet water. It is assumed that all F^- to process trench originates in UNC facilities.
- (c) Based on 1.18×10^8 gallons flow into process trench and average copper (Cu) content into process trench minus average Cu content in inlet water. It is assumed that all Cu to process trench originates in UNC facilities.
- (d) Based on 1.18×10^8 gallons flow into process trench and average uranium (U) content into process trench minus average U content in inlet water. It is assumed that all U to process trench originates in UNC facilities.

Waste Oils

About 4,200 gal of waste oil and water solution are generated annually by the extrusion press. The water-oil emulsion is applied to Hanford roads to stabilize dust.

Beryllium-contaminated oils and uranium-contaminated oils are absorbed in "DicaSorb" and sealed in metal containers. The containers are then encased in plastic for disposal. About 50 gal/yr of beryllium and low-level uranium-contaminated oil are disposed of in the 200 Area disposal site.

Sanitary Waste

Sanitary waste consists of liquid waste from sinks, showers, toilets, janitor sinks and lunchrooms. All sanitary waste from Fuels Fabrication buildings in the 300 Area is discharged into the common 300 Area sanitary sewer system operated by Westinghouse Hanford Company (see Figure II.31). The contribution to the average total 300 Area sanitary sewer flow of 360,000 gpd is 9,900 gpd.

Airborne Effluents

Ventilation systems are provided at process locations to remove harmful or annoying gases, fumes, heat and particulates from the work area and to discharge them outside the buildings (see Figure II.35). The major Building 333 nonuranium-contaminated exhaust systems are described below.

Chemical Bay Exhaust C

This exhaust system removes the gases and acid fumes evolved from the chemical processing tanks in the chemical bay and component preparation areas. The fumes are passed through water spray scrubbers before being discharged vertically 140 ft above ground (114 ft above roof). The flow rate is measured and recorded continuously with a totalizer to obtain the total volume of air discharge per week. An NO_x monitor continuously measures and records the total NO and NO_2 gas content as NO_2 , and a totalizer is provided to obtain the total pounds of NO_2 discharged per week. (See Table II.25 for emission data.)

Extrusion Press Exhaust D

The press exhauster removes most of the smoke and oil fumes from the burning extrusion lubricant at the extrusion press and tooling preheat furnace. The air is discharged vertically about 38 ft above ground (about 12 ft above roof). (See Table II.25 for emission data.)

TABLE II.25. Annual Emissions from Building 333 Exhausts

Source	Material	Concentration	Total Volume (ft ³)	Control Limits	Total Discharged (lb)
Chemical Bay Exhaust	NO _x	16.4 ppm	1.4 x 10 ¹⁰	None	19,600
Cutoff Saw Exhaust	U Particles	4.6 x 10 ⁻⁹ lb/ft ³	2.9 x 10 ⁸	4 x 10 ⁻⁵ Ci/wk(a) (4.3 x 10 ⁻⁸ lb/ft ³)	--
Heat Treat Roof Fan	Salt Particles	0.0010 grain/ft ³	3.1 x 10 ⁹	0.1 grain/ft ³ (b)	500
Extrusion Press Exhaust	Smoke Particles	0.0013 grain/ft ³	1.5 x 10 ⁹	0.1 grain/ft ³ (c)	290
Vacublast Exhausts					
South	Dust Particles	0.025 grain/ft ³	1.1 x 10 ⁸	0.1 grain/ft ³ (c)	330
North	Particles	0.003 grain/ft ³	9.3 x 10 ⁷	0.1 grain/ft ³ (c)	50
Beryllium Exhaust	Be Particles	Not Detectable	1.8 x 10 ⁹	10 grain/day(b)	-
	U Particles	Not Detectable		4 x 10 ⁻⁵ Ci/wk(c)	-

(a) DUN-M-31, Environmental Control Manual, Section 6.2.2.

(b) Washington General Air Pollution Regulations, WAC 18 04-060.

(c) 40 CFR Part 61, "National Emission Standards for Hazardous Air Pollutions," Fed. Regist., 36 (234):23243 (Dec. 7, 1971).

Vacublaster Exhaust F

The air from the filter bag chambers on the two vacublasters is discharged out the east wall through two different exhaust systems that are equipped with cyclone separators. After passing through the separators, the air discharges vertically about 36 ft above ground. The exhaust contains very fine particles of alumina grit, heat treat scale from the fuel cladding and zirconium oxide. (See Table II.25 for emission data.)

Roof Fan in Heat Treat Area Exhaust H

A roof fan above the heat treat furnaces removes the chloride salt and nitrate-nitrite salt fumes. The air is discharged horizontally at roof level. (See Table II.25 for emission data.)

Exhausts L,N,P,Q,S,T, and U

These are minor in both actual and potential discharge of radioactive contaminants. Each is provided with either a filter, scrubber or separator system.

Exhausts E,G,I,J,K,M,O,R,V, and W

These function primarily as ventilators to provide fresh air, air balance, and heat removal. All of these systems discharge directly to the atmosphere.

Solid Waste

Toxic and/or Pyrophoric

Toxic nonradioactive solid waste that contains asbestos, beryllium, mercury, cobalt, selenium, and cadmium is packaged and sealed in plastic before transfer to the Central Landfill Disposal site operated by the Rockwell Operations Office.

Finely divided uranium and Zircaloy-2 are pyrophoric and are stored in water-filled drums until being mixed with masonry cement and cast into metal containers or drums for disposal. The concreted uranium and Zircaloy-2 fines from the cutoff saws and the chips from the counterbore lathe are cast into 7-1/4 gal steel cans and shipped to National Lead Company of Ohio for uranium recovery. The Zircaloy-2 lathe turnings from the end-facing lathes contain

beryllium and are concreted in 55-gal drums and sealed in plastic before being transferred to the 200-Area Plateau Disposal site.

Salvageable Scrap

Salvageable scrap including lumber, plumbing parts and fixtures, electrical wiring, conduits and fixtures and other building fixtures and materials are surveyed for radioactive contamination and, if clean, are collected in appropriate containers throughout the 300 Area. Periodically, these containers are transported to the Central Stores excess yards where the salvaged materials are processed for public sale.

Trash

Solid waste generated in offices, lunchrooms and laboratories throughout the 300 Area includes paper products and food scraps but excludes material contaminated with uranium or noxious chemicals. The waste volume is reduced by a factor of 3 by compaction, and the reduced volume of waste is buried at the Central Landfill Disposal site.

II.3.2.7 Effluent and Process Monitoring

At the Fuels Fabrication Facilities, chemical tanks that are an integral part of the waste acid and uranium-bearing acid recovery system are routinely sampled to help maintain process information and to provide a warning if an abnormal change in tank level, pH, temperature or uranium concentration should occur. The principal gaseous release points are monitored continuously for uranium emissions or nonradioactive chemicals.

Waste Acid and Uranium-Bearing Acid Recovery System (Figure II.33)

Building 334-A

The tank pit is equipped with a leak detector, and the two 2,000-gal storage tanks are equipped with high-level detectors that actuate an audible alarm and light in Building 333 chemical bay and a light in Building 334-A.

Outside Auxiliary Storage Tank

The 6,000-gal tank is equipped with a high-level detector that actuates a light in Building 334-A and shuts off the transfer pumps in Building 334-A.

Waste Acid Neutralizer Tank in Building 313

The 1,500-gal neutralizer is equipped with three systems that will turn off the inlet transfer pumps in Building 334-A:

- a high-level detector,
- a high-temperature switch, and
- a low-pH trip (with pH meter).

A low-level detector will turn off the transfer-recirculating pump on the tank.

Uranium-Bearing Acid Storage Tank and Chemical Milling Tank in Building 333

The 1,770-gal storage tank and the 910-gal chemical milling tank are equipped with catch trays with leak detectors that actuate audible alarms and lights in the chemical bay.

Uranium-Bearing Acid Storage Tanks in Building 313

The two 450-gal storage tanks are equipped with high-level switches that shut off the transfer pumps at Building 333.

Uranium-Bearing Acid Neutralizer Tank in Building 313

The 370-gal tank is equipped with two systems that will turn off the inlet transfer pump: a high-level detector and a high-temperature switch. The tank also is equipped with a pH meter.

Filter Press Effluent Tanks in Building 313

The two 560-gal tanks are used to retain the effluent until a sample of the effluent is sent to the laboratory and is analyzed for uranium. If the uranium concentration is less than 200 ppm, the solution is pumped to the outside storage tank; if not, the solution is pumped to the feed tank and refiltered.

Outside Neutralized Waste Storage Tank

The 4,000-gal neutralized waste storage tank is equipped with a high-level switch that turns off the inlet pumps in Building 313.

Tank Trailer

The 3,000-gal tank trailer has a high-level switch that turns off the load pump. A sample port on the inlet line to the tank trailer is used to obtain a representative sample of each load of neutralized waste. Each load is analyzed for uranium, copper, pH and specific gravity. A monthly composite sample is analyzed for chromium, manganese; nitrate ion, NO_3^- ; sulfate ion, SO_4^{--} ; ammonium ion, NH_4^+ ; fluoride ion, F^- and pH.

100-H Solar Evaporation Basin

A 55-ft-deep 6-in.-diam sample well is located adjacent to the solar basin on the east side (river side). Samples of the water in the well are analyzed periodically for the following: monthly for nitrate ion, NO_3^- ; fluoride ion, F^- ; uranium; copper; sulfate ion, SO_4^{--} ; tritium and beta activity and quarterly for gamma activity.

Process Sewer Monitor Stations

Two sewer monitor stations (see Figure II.30) are installed in the process sewers from Fuels Fabrication buildings to detect any unusual acid or caustic discharges. Each monitor station is equipped with a sample pump and sequential sampler in Building 334 to take weekly samples for chemical analysis and a pH meter with high- and low-level alarms that will enunciate in Building 333. One station monitors discharges from Buildings 333, 334-A, 303-L, and 3720. The other station monitors discharges from Buildings 313 and 303-F and from the 311 tank complex.

The only Fuels Fabrication Facilities that will not be monitored by the two sewer monitor stations are Buildings 303-K and 304. No chemical solutions are used in Building 304. The Building 303-K decontamination facility is used infrequently where only small quantities of chemicals are used; therefore, release from the building is minor.

Outside Waste Oil Tank

The 660-gal waste oil tank on the west side of Building 333 has a high-level detector with an audible alarm and light in the press room in Building 333.

Sampling Program for Uranium Losses to the Process Sewer

Portable sequential samplers are used to take intermittent samples (depending upon fuel models being processed) of waste streams to the process sewer that may contain uranium to determine uranium loss factors. Streams that are sampled include the four process rinse tanks in the chemical cleaning areas, the floor trenches from the uranium machining and cut-up area in Building 333 and the drain from the floor sump in the Building 304 concretion facility.

Airborne Systems

Chemical Bay Stack

The effluent from the stack is continuously analyzed by a Beckman Model 951 Chemiluminescent NO/NO₂ Analyzer. Results are recorded on a strip chart in ppm NO₂. A totalizer indicates the pounds of NO₂ discharged. The stack flow rate is measured and recorded on a strip chart, and a totalizer denotes the total volume of air discharged in cubic feet.

Uranium Cutoff Saw Exhaust

The effluent from the cutoff saw stack is continuously sampled by an isokinetic sampler that collects the uranium particulates on a Gelman glass fiber Type A filter paper for alpha and beta analysis. Sample flow is measured with a rotometer and stack flow with a differential pressure Annubar gage. An operating time meter is used to determine the time the exhaust was used per week.

Beryllium Exhaust

The effluent from the beryllium exhaust is continuously sampled for particulates on a Millipore Type HA filter. This filter paper is removed for alpha, beta and beryllium analysis.

Incidental Stack Sampling

Stacks such as the vacublast exhausts, extrusion press exhaust and heat treat roof fan are sampled for particulates on a nonroutine basis to assure compliance with applicable effluent controls.

II.3.2.8 Modifications to Fuels Fabrication Facilities

Improvements Completed

Improvements in 300-Area waste management operations discussed or contemplated in ERDA-1538 have continued. The current status is described here.

Concentrated NO₂ Fume Scrubber; final cost, \$48,500.

A 16-stage scrubber was put into operation in August 1975 to treat the concentrated NO₂ fumes from the copper strip tank 10 and chemical mill tank 32 in Building 333. This new scrubber has reduced the average concentration of NO₂ in the chemical bay exhaust stack from 31 ppm to 13 ppm.

Improvements of Tank Farms; final cost, \$4,800.

In July 1975 a vent was installed on the discharge line from the H₂SO₄ storage tank to relieve any pressure buildup when the line is valved off. A redundant polyvinyl chloride line from the HNO₃ storage tanks was removed in October 1975 to prevent an accidental discharge of HNO₃ to the limestone pit (a stainless-steel line was already in place). In September 1975 the overflow drains from the two NaOH tanks were modified to discharge to the process trench instead of to the ground. Reliable level indicators were installed on the two 10,000-gal caustic tanks. This installation eliminated potential caustic overflows to the environs during loading and unloading operations.

Improvements to Scrubber for Coextrusion Cutoff Saws; final cost, \$4,400.

An Annubar flow element, an improved stack sampler for uranium discharges and an hour meter were installed on the Building 333 scrubber stack in October 1975. An interlock system was installed in February 1976 to prevent any of the four uranium saws from being operated without the exhaust scrubber in operation.

Replacement of Copper Strip Tank in Building 333; final cost, \$24,500.

The 205-gal copper strip tank having air sparger mixing and loose fitting lids was replaced in August 1977 by a 576-gal tank having mechanical agitators and tight lids to reduce air dilution of NO_x fumes to the 16-stage NO_x scrubber. The scrubber operates more efficiently with higher concentrations of NO_x fumes.

Installation of Sewer Monitoring Stations; final cost, \$22,500.

Sewer monitoring stations were installed in the two process drain lines leaving the Fuels Fabrication Facilities in November 1977. The station contains a pH alarm, continuous sampling system and a flow rate measuring device.

Catch Basins and Dikes for Chemical Storage Tanks; final cost \$43,000.

Catch basins and concrete dikes were provided in August 1977 for a number of above-ground chemical storage tanks located near the Fuels Fabrication Facilities in the 300 Area. These basins and dikes provide additional assurance that accidental discharge of acids, caustics, oils, and solvents to the process sewer or the ground does not occur.

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III. ENVIRONMENTAL EFFECTS

III.1 N REACTOR OPERATIONS

III.1.1 Radiological Impact

III.1.1.1 General

Radionuclides gain access to the environment via releases to the atmosphere, the ground and/or to the Columbia River during normal operation of N Reactor and the Fuels Fabrication Facilities. Detailed studies have been conducted at Hanford to determine the exposure pathways leading to man from these sources. A complete discussion of these exposure pathways and maximum individual and population doses from Hanford operations are presented in ERDA-1538, Section III.1. The dose calculations in that statement are based upon the cumulative annual radioactive releases from all Hanford operations.

The maximum possible dose to a member of the public (the "maximum individual") during 1978 and the 50-yr dose commitment to this maximum individual from 1978 radionuclide releases were calculated using special computer codes developed by PNL. These codes are described in ERDA-1538, Appendix III-2. The CY-1978 effluent data given in Table III.1 were used as source terms for calculations.

In this report operational dose calculations are based upon the radioactive releases from N Reactor and the Fuels Fabrication Facilities. Past studies at Hanford, combined with results of the environmental surveillance program, have enabled the development of characteristics of a hypothetical person whose dietary and recreational habits maximize, through each separate pathway, the dose he might receive from N Reactor operation. Such a hypothetical person is called the maximum individual. The habits and diet of the maximum individual comprise the maximum intake or exposure reported for each exposure pathway even though the maximum values are not, in actuality, attributable to the same person. The maximum individual is a person assumed to have the following characteristics:

TABLE III.1. Radionuclides Discharged to the Environment
from N Reactor During CY 1978

<u>Radionuclide</u>	<u>Half Life</u>	<u>Liquid (Ci)</u>	<u>Airborne (Ci)</u>
^3H	12.3 yr	330.	32.23
^{24}Na	15 hr	ND	ND ^(a)
^{32}P	14.3 days	0.029	ND
^{41}Ar	1.8 hr	ND	120,640
^{46}Sc	84 days	ND	ND
^{51}Cr	28 days	0.14	ND
^{54}Mn	303 days	0.18	0.027
^{56}Mn	2.6 hr	1.2	3.71
^{58}Co	71 days	0.032	0.019
^{60}Co	5.3 yr	0.96	0.047
^{59}Fe	46 days	0.11	0.028
^{65}Zn	245 days	ND	0.012
^{76}As	26.4 hr	ND	6.64
^{89}Sr	50.8 days	0.4	0.022
^{90}Sr	28 yr	1.3	0.00072
^{95}Nb	35 days	0.26	0.015
^{95}Zr	66 days	0.26	0.015
^{99}Mo	67 hr	2.6	0.98
^{103}Ru	40 days	0.37	0.017
^{106}Ru	368 days	0.57	0.06
^{124}Sb	60 days	0.062	0.0029
^{125}Sb	2.7 yr	0.17	0.0003
^{131}I	8 days	4.76	0.48
^{132}I	2.3 hr	ND	6.2
^{133}I	20.3 hr	0.95	3.1
^{135}I	6.7 hr	ND	6.1
^{134}Cs	2 yr	ND	ND
^{135}Xe	9.1 hr	ND	2,248.
^{137}Cs	30 yr	0.031	0.0023
$^{140}\text{BaLa}$	12.8 days	4.1	1.11
^{141}Ce	32.5 days	0.035	0.0073
^{144}Ce	284 days	0.15	0.057
^{238}Pu	87.4 yr	0.00013	0.00000037
^{239}Pu	24,390 yr	0.085	0.0000022

(a) ND = not detected.

- resides continuously 5 miles northwest of N Reactor
- obtains drinking water from the Columbia River downstream of N Reactor
- drinks 275 liters (73 gal) of milk during a nine-month period from a cow grazing near his residence
- eats 710 kg (156 lb) of produce grown along the Columbia River downstream of N Reactor and irrigated with Columbia River water
- eats 40 kg (88 lb) of fish per year caught from the Columbia River downstream of N Reactor
- spends as much as 500 hr/yr on the shoreline of the Columbia River downstream of N Reactor, 100 hr/yr swimming in the river downstream of N Reactor and 100 hr/yr boating in the river downstream of N Reactor

ERDA-1538, Section III.1.2, discusses the impact of radionuclide releases at Hanford on biota other than man. The discussion includes a summary of research and surveillance programs designed to assess the possible changes in the terrestrial and aquatic environment in and near the Columbia River since operations began in 1944.

III.1.1.2 Effluent Releases

Dose from Operation of N Reactor - 1978

Based on the assumptions for the maximum individual and on CY-1978 effluent releases as the source terms, the maximum fence post, maximum resident and total population doses were calculated.

Maximum "Fence-Post" Exposure Rate

The maximum fence-post exposure rate during CY-1978 was calculated to be 0.075 mrem/hr at a point on the Hanford shoreline of the Columbia River near N Reactor. Most of this exposure is attributable to ^{41}Ar (half-life of 1.8 hr) and ^{135}Xe (half-life of 9.2 hr) released from N Reactor.

Maximum Individual Dose

The calculated whole body dose received during CY-1978 by the hypothetical maximum individual from effluent released from N Reactor during CY-1978 was 0.16 mrem, as shown in Table III.2. Most of this dose is attributable to external radiation from ^{41}Ar released to the atmosphere.

TABLE III.2. Calculated Dose to the Maximum Individual During CY 1978
from Effluents Released from N Reactor During CY 1978

Pathway	Dose Commitment in mrem					
	Skin	Total Body	GI-LLI(a)	Thyroid	Bone	Lung
<u>Airborne Effluents(b)</u>						
Inhalation	----	2.8E-5	1.7E-4	1.2E-3	1.9E-5	1.5E-3
Ingestion	----	5.3E-5	4.8E-5	2.0E-2	5.8E-5	1.5E-5
Air Submersion	1.1E-1	8.0E-2	(8.0E-2)	(8.0E-2)	(8.0E-2)	(8.0E-2)
GRD Contamination	----	<u>4.4E-5</u>	<u>(4.4E-5)</u>	<u>(4.4E-5)</u>	<u>(4.4E-5)</u>	<u>(4.4E-5)</u>
<u>Total Air Pathways</u>	1.1E-1	8.0E-2	8.0E-2	1.0E-1	8.0E-2	8.1E-2
<u>Liquid Effluents</u>						
Drinking Water	----	3.3 X 10 ⁻⁴	2.6 X 10 ⁻⁴	4.7 X 10 ⁻²	5.9 X 10 ⁻⁴	1.4 X 10 ⁻⁴
Fish Consumption	----	7.9 X 10 ⁻³	4.2 X 10 ⁻⁴	4.9 X 10 ⁻²	5.1 X 10 ⁻³	1.8 X 10 ⁻⁴
Irrigated Foods	----	2.2 X 10 ⁻³	8.7 X 10 ⁻⁴	1.6 X 10 ⁻¹	8.7 X 10 ⁻³	2.0 X 10 ⁻⁴
Shoreline	----	4.6 X 10 ⁻⁴	(4.6 X 10 ⁻⁴)	(4.6 X 10 ⁻⁴)	(4.6 X 10 ⁻⁴)	(4.6 X 10 ⁻⁴)
Swimming	----	2.6 X 10 ⁻⁵	(2.6 X 10 ⁻⁵)	(2.6 X 10 ⁻⁵)	(2.6 X 10 ⁻⁵)	(2.6 X 10 ⁻⁵)
Boating	----	1.3 X 10 ⁻⁵	(1.3 X 10 ⁻⁵)	(1.3 X 10 ⁻⁵)	(1.3 X 10 ⁻⁵)	(1.3 X 10 ⁻⁵)
<u>Total Water Pathways</u>	----	1.1 X 10 ⁻²	2.0 X 10 ⁻³	2.6 X 10 ⁻¹	1.5 X 10 ⁻²	1.0 X 10 ⁻³
<u>Total Adult</u>	1.1 X 10 ⁻¹	1.6 X 10 ⁻¹	1.8 X 10 ⁻¹	3.6 X 10 ⁻¹	9.5 X 10 ⁻²	8.2 X 10 ⁻²

(a)Gastrointestinal--lower large intestine.

(b)Joint frequency wind data from the N Reactor site (1970) was used for stack releases at 70 m height (61 m stack and 9 m normal plume rise).

(c)Internal dose from external exposure indicated by parentheses.

The potential adult dose to the thyroid was calculated to be 0.36 mrem from the effluents released by N Reactor during CY-1978, as shown in Table III.2. The dose was primarily due to ^{131}I in milk. The iodine in milk results from the irrigation of pasture with Columbia River water and from deposition of airborne radioiodine on the pasture grass. Essentially all of the dose would be received during CY-1978 since ^{131}I has a half-life of eight days.

The 50-yr whole body dose commitment to the maximum individual from CY-1978 effluents was calculated to be 0.14 mrem, as Table III.3 shows.

50-Mile Radius Population Dose

The whole body population dose received during CY-1978 by the population residing within a 50-mile (80-km) radius of N Reactor and the 50-yr dose commitment to the same population from effluent released during 1978 were calculated for all of the radionuclides listed in Table III.1. Table III.4 lists the population dose received during CY-1978 by the whole body and thyroid (determined from critical pathway analysis to be the two organs of primary interest).

The calculated whole body population dose received by the approximately 236,000 people living within the 50-mile radius during CY-1978 was 1.5 man-rem or an average per capita dose of 0.006 mrem. Most of this dose is attributable to external radiations from ^{41}Ar released to the atmosphere.

The population thyroid dose during CY-1978 was calculated to be 4.7 man-thyroid-rem or an average per capita dose of 0.013 mrem. The dose to the thyroid is primarily due to radionuclides of iodine released to the atmosphere and the Columbia River.

Table III.5 lists the 50-yr dose commitment potentially received by the population of 236,000 persons from effluents released during CY-1978. The whole body population dose commitment was estimated to be 1.6 man-rem.

TABLE III.3. Calculated 50-Yr Dose to the Maximum Individual
from Effluents Released from N Reactor During 1978

Pathway	Dose Commitment in mrem					
	Skin	Total Body	GI-LLI(a)	Thyroid	* Bone	Lung
<u>Airborne Effluents(b)</u>						
Inhalation	----	3.2E-5	1.7E-4	1.2E-3	6.4E-5	1.5E-3
Ingestion	----	5.9E-5	4.8E-5	2.1E-2	7.4E-5	1.5E-5
Air Submersion	1.1E-1	8.0E-2	(8.0E-2)(c)	(8.0E-2)	(8.0E-2)	(8.0E-2)
GRD Contamination	----	4.4E-5	(4.4E-5)	(4.4E-5)	(4.4E-5)	(4.4E-5)
<u>Total Air Pathways</u>	1.1E-1	8.0E-2	8.0E-2	1.0E-1	8.0E-2	8.2E-2
<u>Liquid Effluents</u>						
Drinking Water	----	2.1 X 10 ⁻³	2.6 X 10 ⁻⁴	4.9 X 10 ⁻²	7.4 X 10 ⁻³	1.5 X 10 ⁻⁴
Fish Consumption	----	1.7 X 10 ⁻²	4.2 X 10 ⁻⁴	5.0 X 10 ⁻²	6.1 X 10 ⁻²	3.2 X 10 ⁻⁴
Irrigated Foods	----	3.6 X 10 ⁻²	8.8 X 10 ⁻⁴	1.6 X 10 ⁻¹	1.4 X 10 ⁻¹	2.2 X 10 ⁻⁴
Shoreline	----	4.6 X 10 ⁻⁴	(4.6 X 10 ⁻⁴)	(4.6 X 10 ⁻⁴)	(4.6 X 10 ⁻⁴)	(4.6 X 10 ⁻⁴)
Swimming	----	2.6 X 10 ⁻⁵	(2.6 X 10 ⁻⁵)	(2.6 X 10 ⁻⁵)	(2.6 X 10 ⁻⁵)	(2.6 X 10 ⁻⁵)
Boating	----	1.3 X 10 ⁻⁵	(1.3 X 10 ⁻⁵)	(1.3 X 10 ⁻⁵)	(1.3 X 10 ⁻⁵)	(1.3 X 10 ⁻⁵)
<u>Total Water Pathways</u>	----	5.6 X 10 ⁻²	2.1 X 10 ⁻³	2.6 X 10 ⁻¹	2.1 X 10 ⁻¹	1.2 X 10 ⁻³
<u>Total Adult</u>	1.1 X 10 ⁻¹	1.4 X 10 ⁻¹	8.2 X 10 ⁻²	3.6 X 10 ⁻¹	2.9 X 10 ⁻¹	8.3 X 10 ⁻²

(a)Gastrointestinal--lower large intestine.

(b)Joint frequency wind data from the N Reactor site (1970) was used for stack releases at 70 m height
(61 m stack and 9 m normal plume rise).

(c)Internal dose from external exposure indicated by parentheses.

TABLE III.4. Calculated Population Dose During CY-1978 Due to Effluents Released from N Reactor During CY-1978

Pathway	Whole Body Dose (man-rem)	Thyroid Dose (man-thyroid-rem)
<u>Gaseous Effluents</u>		
Inhalation	4.2E-3	1.8E-1
Ingestion	3.8E-3	1.4E0
Air Submersion	1.5E0	(1.5E0) ^(a)
Ground Contamination	4.4E-3	(4.4E-3)
<u>Liquid Effluents</u>		
Drinking Water	1.0×10^{-2}	1.4×10^0
Fish Consumption	7.3×10^{-4}	1.8×10^{-2}
Aquatic Recreation	2.3×10^{-3}	2.3×10^{-3}
Irrigated Foods (produce, eggs, milk)	2.2×10^{-3}	1.6×10^{-1}
TOTAL	1.5×10^0	4.7×10^0

(a) Internal dose from external exposure indicated by parentheses.

TABLE III.5. Calculated 50-Yr Population Dose Commitment from Effluents Released from N Reactor During CY-1978

Pathway	Whole Body Dose (man-rem)	Thyroid Dose (man-thyroid-rem)
<u>Gaseous Effluents</u>		
Inhalation	4.7E-3	1.8E-1
Ingestion	4.2E-3	1.5E0
Air Submersion	1.5E0	(1.5E0)
Ground Contamination	4.4E-3	(4.4E-3)
<u>Liquid Effluents</u>		
Drinking Water	6.4×10^{-2}	1.5×10^0
Fish Consumption	6.5×10^{-3}	1.9×10^{-2}
Aquatic Recreation	2.3×10^{-3}	2.3×10^{-3}
Irrigated Foods (produce, eggs, milk)	3.5×10^{-2}	1.6×10^{-1}
TOTAL	1.6×10^0	4.9×10^0

(a) Internal dose from external exposure indicated by parentheses.

The calculated 50-yr thyroid dose commitment is essentially the CY-1978 thyroid dose, since the iodine nuclides and major external exposure sources that contribute to most of the dose have short half-lives. Because the gaseous plume model is readily adaptable to assessing radiological impacts to population segments, Table III.6 is provided. It lists the calculated cumulative population doses for annular regions centered at N Reactor as a result of gaseous releases from N Reactor during CY-1978. The liquid pathways identified in Table III.5 account for approximately 4% of the calculated 50-yr population commitment.

TABLE III.6. Estimated Cumulative Population, Cumulative Annual Whole Body Dose, and Average Annual Body Dose in Circular Areas Around N Reactor from Airborne Effluents Released from all Facilities in CY-1978

<u>Radius (miles)</u>	<u>Cumulative Population (1981)</u>	<u>Cumulative Dose (man-rem/yr)</u>	<u>Average Annual Dose (mrem/yr per person)</u>
10	64	.0017	.027
20	6,229	.086	.014
30	45,945	.32	.007
40	139,753	.61	.0044
50	236,195	.74	.0031

Table III.7 presents a summary of the doses to the total population and to the maximum individual from N Reactor operation during CY-1978.

III.1.1.3 Solid Waste

There is no offsite dose from radioactive solid waste generated from N Reactor and associated facilities.

III.1.1.4 Occupational Doses to Employees

Based on thermoluminescent dosimeter readings for each employee, the annual average whole body dose received by the 800 employees engaged in the operation

TABLE III.7. Summary of Doses to Man from N Reactor Operations
At Hanford During CY-1978

	<u>50-Year Dose Commitment (mrem)</u>	
	<u>Total Body</u>	<u>Thyroid</u>
Average Individual Within 50-km radius	0.006	0.013
Maximum Individual	0.14	0.36

of the N Reactor and Fuels Fabrication Facilities during the 1974-1978 period ranged from 1.0 to 0.84 rem (Table III.8).

Projected occupational whole body exposures during the period 1977-1990 are based on the following assumptions:

- Conversion will be made to the plutonium production mode in which 6% instead of 12% of the plutonium yield is ^{240}Pu .
- Approximately ten weeks of summer outage will be scheduled each year for major maintenance.
- Outside contractor forces will be utilized to accomplish project-type work and to supplement maintenance forces as required.
- Periodic N Reactor primary coolant system decontamination will be continued as needed.
- The current occupational exposure guidelines will prevail throughout the period.
- 4.5 billion kWhr of electricity will be produced annually.

Conversion to the alternate plutonium production mode (in which 6% of the plutonium yield is ^{240}Pu) would increase the total work force by approximately 130 persons. However, the average annual exposure is projected to essentially remain constant at 0.84 rem.

III.1.1.5 Transportation of Radioactive Materials

Irradiated fuel from the 105-N Reactor building is shipped by rail to the 105-KE fuel storage facility. The fuel is shipped in shielded casks in amounts insufficient to form a critical mass.

Unirradiated fuel from the 300-Area fuels fabrication facility is shipped by truck to the 105-N Reactor building. The fuel assemblies are contained in a lattice arrangement to ensure nuclear criticality safety.

TABLE III.8. Whole Body Occupational
Exposure--N Reactor and
Fuels Fabrication Facilities
Personnel (rem)

<u>Year</u>	<u>Total Number of Employees</u>	<u>Annual Exposure</u>	
		<u>Total</u>	<u>Average</u>
1974	870	870	1.0
1975	910	870	0.96
1976	830	680	0.82
1977	850	780	0.92
1978	930	780	0.84
1979-1990	970	820	0.84

51%

For these onsite shipments procedures are in effect that specify the inspection, loading and unloading requirements. The combination of container requirements and administrative controls meets the requirements of ERDAM-0529 and provides safety equivalent to that afforded by shipping regulations adopted by the Department of Transportation (DOT).

The only radioactive material shipped offsite from N Reactor and the Fuels Fabrication Facilities is scrap uranium that is packaged in DOT-approved containers and shipped to National Lead Company at Fernald, Ohio. The uranium scrap is shipped by exclusive-use truck or railroad car. The nature of the material is such that essentially no radiation exposure is received by the general public during transport.

III.1.1.5 Seepage from the K-Area Fuel Storage Facilities

The KE and KW fuel storage facilities are located approximately 2000 ft. from the Columbia River (Figure II.6). Each facility was originally constructed with a secondary seepage barrier, in the form of an asphalt membrane, under the basin floor. This two-inch-thick membrane was installed under the entire storage basin except for the pickup chute area and was designed to collect leakage in perforated pipes that lie on top of the membrane. These pipes drain to a sump in which an automatic pump returns any seepage back to the basin.

During testing conducted in 1974 as part of the project to modify the 105-KE basin to a fuel storage facility, a 15-day basin drawdown established a 4 gpm seepage rate from the basin system. At least 90 percent of the seepage was thought to be collected by the membrane and sump. The remaining 10 percent of the seepage was assumed to discharge to the soil beneath the basin.

In early 1977, an unusually high seep rate (13.5) gpm was detected. Improved basin level monitoring was initiated at this time. Investigations determined the increased seepage to be due to reduced basin water temperature during the winter months. The lower temperature caused concrete contraction to expand minor cracks which exist in the basin. Further investigations showed that no radioactivity was being released in K-Area effluents and that no water was being collected in the membrane drainage sump. This indicated that all of the seepage was going to the ground either through or past the membrane.

Due to the above conditions, several steps were taken to reduce the basin seepage. Repairs to the basin and installation of a watertight dam separating the basin discharge chute from the main basin were completed. As a result, basin seepage was reduced to zero.

The KW fuel storage facility is not yet in use. During the modifications to the KW basin to prepare the basin for fuel storage all detectable seepage points were repaired. As an assurance against the loss of water, all floor drains in the basin have been sealed with concrete and the walls and floor of the basin have been coated with a pliable epoxy sealant.

Some small amount of seepage of radioactively contaminated basin water may be anticipated. The likelihood of such a release is being reduced to "as low as technically and economically practical". All detectable seepage points have been repaired or isolated.

Seepage from the basin (currently zero) adds to the current K Area soil inventory. Radionuclides released by seepage should be retained in the soil column immediately beneath the basin. Studies performed in the deactivated mile long K trench clearly show that over 95 percent of the Pu, Cs and Sr activity released to the trench is found in the first ten to fifteen feet of the soil column.

The maximum credible discharge rate to the soil from the KE or KW basin is 25 gpm and would be caused by a cask drop accident. This seep rate is very low compared to other ground disposal facilities that have been and are being used at Hanford. For example, the flow rate to the 1301-N crib system is 1800 gpm. The much lower seepage rate from the KE basin would permit more effective filtration and ion exchange reactions in the soil. In addition, none of the typically more soluble radionuclides (such as I-131) will be found in the KE basin water (with the exception of tritium).

The KE basin is located 2.5 times farther from the river than the 1301-N crib which increases the effective length of the soil column.

Due to the low seepage rate potential from the KE basin and the high retention rate of the area soil, operation of the KE basin has not, and is not expected in the future to produce any measurable nuclide concentrations at the Columbia River.

During the operating history of the KE and KW Reactors, radioactive material seeped from the basins to the soil beneath each facility through similar pathways that might develop today. No direct measurements have been taken of the radionuclide inventories beneath the basin; however, analyses of the sludge that accumulated on the basin floor before storage of N fuel have been

performed. It is reasonable to assume that the soil at the seepage site would have shown similar radionuclide concentrations as the sludge (since the sludge could have seeped through at similar locations).

The ultimate disposition of the soil beneath the basin will be determined by the Pu concentration in the soil because all other radionuclides will decay to releasable levels within several hundred years. Based on the level of Pu in the sludge and the assumed similar level in the seepage area soil, it was concluded that the soil beneath the KE and KW basins would have to be removed for ultimate disposal. Operation of the basins with seepage would not change this conclusion although it could increase the amount of soil that would have to be removed. This increase in volume would not significantly affect the cost of the ultimate D&D efforts.

III.1.2 Nonradiological Impact

III.1.2.1 Effluent Releases

The chemical discharges to the 1301-N Facility at 100-N Area are listed in Section II.3.1.6. The concentrations of chemicals found in the riverbank springs were measured in June 1976 and were compared to the N Reactor intake river water concentrations. Most of the chemicals measured at the riverbank springs were found at concentrations equal to or less than the concentrations measured in the intake river water. All of the remaining chemicals were only slightly higher than the intake river water, except for total phosphorous, which is approximately 100 times more concentrated.

Phosphorous is an essential nutrient for plant and animal growth. The concentration of phosphorous at the riverbank springs is 0.22 mg/l. Fish studies have shown that the total fish productivity may increase when phosphorous is increased above 0.2 mg/l but less than 1.4 mg/l.⁽¹⁾

The quantities of chemical released to the Columbia River, as shown in Section II.3.1.6, have minimal, if any, impact on nearby biota.⁽²⁾ Chemicals that are added to the river in concentrations slightly above ambient river

concentrations are quickly diluted by an average river flow of 120,000 cfs. Thus, any adverse effects would be limited to the immediate area of the discharge.⁽²⁾

The filter backwash water containing aluminum hydroxide, polyacrylamide, and suspended solids along with ion exchange regeneration wastes are discharged to a ground disposal facility. This facility became operational in July of 1977 in compliance with the National Pollution Discharge Elimination System (NPDES) under permit number WA-000374-3.

No adverse environmental effects have been determined from operation of the ground disposal facility and none are predicted.⁽³⁾ The infiltration rates of the facility soil accommodate the volume of liquid waste,⁽⁴⁾ and the groundwater level is not adversely affected.⁽⁵⁾

In summary, nonradioactive effluents from N Reactor will not lead to nor cause any known adverse effects on the environs.⁽⁶⁾

Airborne Effluents

Nonradioactive materials released to the atmosphere from 100-N Area consist primarily of combustion products such as nitrogen dioxide, sulfur dioxide and particulates. Samples taken of sulfur dioxide and particulate emissions from the 184-N powerhouse show compliance with Washington State Air Quality Standards (see Section II.3.1.6).

During CY-1978, the ambient air in and around the Hanford Reservation was monitored for nitrogen dioxide by the Hanford Environmental Health Foundation. Results continue to show that the concentration of nitrogen dioxide is well within the national ambient air standards.⁽⁷⁾ Releases of this chemical to the air produce no observable environmental effects.⁽⁸⁾

What is 60%

Heat Dissipation System

The N Reactor heat dissipation system is discussed in detail in Section II.3.1.3. Only a few of the operating parameters that determine the environmental effects of operation are summarized here.

Waste heat generated by N Reactor is dissipated to the environment in cooling water discharged to the Columbia River. The intake and outfall designs are shown in Figures II.13 through II.16 of Section II.3.1.4.

At a reactor power level of 3,760 MWt (the normal power level during CY 1978), approximately 460 MWt is discharged from N Reactor via the 102-inch outfall line to the center of the river. The remaining excess steam is exported to the WPPSS generating station where 860 MWe is produced, after which the residual heat (approximately 2,500 MWt) is discharged from the generating station to the river.

Thermal Effluent Effects and Standards

During the dual purpose operation of N Reactor the majority of the waste heat is discharged to the river in the coolant water from the WPPSS generating station. Although the WPPSS Hanford Generating Project and associated facilities are not part of DOE's Hanford operations, a brief description of WPPSS discharge and the resulting thermal plume is presented in Section III.1.3.2.3.1 of ERDA-1538. WPPSS has summarized the environmental impacts of this discharge in their "Environmental Impact Statement on the Continued Operation of the Hanford Generating Project", dated January 1977.

In the 100-N Area, condenser coolant water is discharged to the middle of the Columbia River via a 102-in. outfall line. The temperature of the water in the 102-in. line is about 5.5 C (10 F) above ambient river temperature.

The 102-in. line discharges into the main river channel at river mile 380. The line goes from a seal well on shore into the river channel, a distance of 862 ft. A 14-ft-diam velocity cap, located about 6 ft above the vertical discharge orifice, directs the discharge horizontally through a 360° arc.

The velocity cap is at an elevation of 359 ft, 16 ft below the low flow water elevation of 375 ft. The river width in the vicinity of N Reactor varies between 1200 and 1800 ft, depending upon flow rate. River velocities range from 3 fps to over 11 fps, depending upon river cross sections and flow rate.

At a minimum regulated river flow rate of 36,000 cfs, a routine release rate of 500 MWt via the 102-in. line is calculated to increase the average river temperature by only 0.21 F. At the annual average river flow rate of 120,000 cfs, the average increase in river temperature is calculated to be approximately 0.06 F.

Juvenile downstream migrant salmon experience no adverse impact as a result of the 102-in. discharge. A recent assessment of the Hanford Generating Project discharge (2600 MW compared to 500 MW from the 102-in. line) indicated that the fish would not be exposed to lethal temperatures⁽¹⁰⁾ (see discussion in Section V.1.3.5.2 on the Plutonium-Only Operation of N Reactor in ERDA-1538). The lack of significant impact due to heating of the Hanford reach of the Columbia River is also discussed in Section III of ERDA-1538.

The NPDES permit limits the combined discharge from the 102-in. line to 650 MWt and a maximum discharge temperature of 83.4 F. A 100-ft wide dilution zone extending from 50 ft upstream of the point of discharge to 300 ft downstream of the point of discharge is also specified in the permit.

A prediction of excess temperature downstream of a 650-MW waste heat discharge from the 102-in. line was performed for a river flow of 65,000 cfs with a river velocity of 3.0 fps and a critical ambient temperature of 66.7 F. The prediction indicated that the maximum temperature at the mixing zone boundary 300 ft downstream of the discharge will be about 1 F above ambient river temperatures. This calculated increase is in compliance with state water quality standards even at regulated and 7-day 1-in-10-year low flows at a critical ambient temperature of 66.7 F. The state water quality standards allow a ΔT of 1.8 F under these flow and river temperature conditions.

Intake Effects

The effects of Hanford intake structures on the aquatic ecological system were discussed briefly in Sections III.1.3.2.1 of ERDA-1538. The principal effects of the operation of N Reactor's cooling water intake system are impingement of fish on the traveling screens of the 181-N water intake structure on the Columbia River, entrainment of fish in the 181-N intake structure and destruction of drift organisms passing into the water withdrawal system and through the condenser cooling water system.

Juvenile chinook salmon are present in the Hanford reach most of the year. The fry of the fall run emerge from the gravel from March to May, and the average size is about 30 to 40 mm. Many of these fry are swept down river upon emergence from the gravel. Studies ⁽³⁶⁾ indicate that these fish are to be found predominately near the shore. Since the 181-N intake is a shore-line structure, some impact on these fry is expected.

The impacts of the N Reactor water intake structure were assessed in 1977 by PNL ⁽²¹⁾. Fish impingement and traveling screen passage were studied at the 181-N water intake structure from late April to August 1977. Species and numbers of fish affected were determined and compared to those at the adjacent Hanford Generating Project (HGP) since numerous studies ⁽¹¹⁻¹³⁾ have been conducted there. Methods included: (1) weekly beach seining to evaluate timing, size and relative abundance of juvenile chinook salmon occurring in the vicinity of the intakes, (2) sampling of the traveling screen washings and intake water behind the traveling screens to evaluate fish impingement and screen passage and (3) releases of marked rainbow trout fry at both intakes for direct comparison.

Some of the 181-N intake structure study conclusions are:

- Fewer chinook salmon fry were impinged at 181-N during the study period (345) than at HGP (9092).

- Virtually all of the impinged 345 chinook and 3592 perch at 181-N were killed, while over 95% of those fish impinged at HGP survived.
- Entrainment in the intake structure by passage of chinook salmon fry through or around the 181-N traveling screens may be equal to or greater than impingement.
- 181-N impinged more yellow perch fry than any other species during May through June 1977. An estimated 3592 yellow perch fry were impinged at 181-N. However, impingement of perch fry occurred only during June.
- Total estimated screen impingement and entrainment mortalities during May and June 1977 were less than 0.1% of the average vulnerable chinook salmon outmigration going by the 181-N intake.
- If the fish protective measures used at HGP were installed at 181-N, fish passage through the traveling screens may be reduced and survival of impinged fish may be about 95%.

Further interpretation of the data in the 1977 study ⁽²¹⁾ and comparisons to other studies reveal the following:

- The vulnerable outmigrant population in 1977 was 903,000 fry.
- Assuming the outmigrants moved downstream similar to previous years, the 1977 study covered half the migration season.
- The ratio of entrained fish to impinged fish was experimentally determined to be 10:3 and therefore the total number of passed fish was at least 2300.
- The total mortality due to the N Reactor intake was approximately 3000 salmon fry during 1977, 0.3% of the vulnerable outmigrants passing the N Reactor intake.

- The 3000 fry lost are equivalent to the number of swimup fry resulting from two spawning pairs and is not significant when considering the fishery as a whole.

Fish fry protection modifications to the 181-N intake similar to those mitigative measures taken by WPPSS at the HGP intake (low pressure wash system) ⁽¹³⁾ have been evaluated to determine if such a change would be cost effective for further reducing the small impact the N Reactor intake has on fish fry. Assuming N Reactor were to operate through 1990 and the low pressure wash system were to result in 100% survival of impinged fish, performing such modifications would cost approximately \$2.00 per chinook salmon fry saved (\$150/lb); state rearing and planting costs for hatching chinook fry are about \$0.024 per fry (\$1.80/lb). Given the small impact and the cost per fish saved, such a modification is judged to be unreasonable and the impact is judged to be acceptable.

The impingement of perch fry is not considered to be significant because they have virtually no recreational or food value in this area.

III.1.2.2 Solid Waste

Ordinary trash (primarily paper) is compacted to approximately one-third volume and buried in the Hanford Reservation central sanitary landfill. The N Reactor output is negligible in comparison with the 360,000 ft³/yr total for all of the Hanford Reservation.

III.1.3 Natural Forces

III.1.3.1 Tornadoes and Wind

In view of the low incidence of tornadoes at the Hanford site, reactor facilities were designed to withstand straight winds of severe magnitude. A peak wind speed of 80 mph at an elevation of 50 ft was recorded at the Hanford Meteorological Station (HMS) on January 11, 1972, with no damage to process buildings and associated structures at the 100-K and 100-N areas. This is the highest wind speed ever recorded at HMS. Winds of this magnitude are expected to occur about once every 33 years. The following wind pressures were used in the design of the two reactor facilities mentioned:

	<u>Elevation Above Ground (ft)</u>	<u>Wind Pressure (lb/ft²)</u>	
100-K Reactors	0 to 60	15	<i>Relate To wind speed.</i>
	60 to 100	20	
	Over 100	25	
100-N Reactor	Under 20	15	
	20 to 50	20	
	50 to 100	25	
	Over 100	30	

The 200-ft ventilation stack at 100-N Area was designed for a wind pressure of 25 lb/ft^2 on the projected area. The above values equal or exceed minimum design values for wind pressures presented in the Uniform Building Code for eastern Washington. These pressures correspond to maximum gust velocities of about 70 mph and 80 mph at 30 ft and 100 ft above the ground, respectively.

The reinforced concrete confinement structure surrounding the reactor is designed to withstand lateral forces about four times those generated by the Hanford standard design tornado. The Hanford design tornado has a maximum wind velocity of 175 mph and is made up of 150 mph rotational velocity and 25 mph translational velocity. The resultant velocity (175 mph) is applied uniformly over the full exposed height of the structures.

UNC also uses the fuel storage basins in the 105-KE and 105-KW areas to store irradiated N Reactor fuel elements. An analysis⁽¹⁵⁾ of tornado effects on the structures over these basins indicates that the wall siding and part of the roof covering would probably be destroyed, but the structural framing would not be damaged. No other structural effects on the basins are anticipated from a tornado, nor would the tornado be able to blow enough water out of the basin to expose any fuel elements.

III.1.3.2 Earthquakes

N Reactor was designed in accordance with the earthquake regulations in Section 2312 of the Uniform Building Code (1955 and 1958 editions). Except for certain items considered to be critical for reactor safety, the design was based on the requirements for seismic Zone 2.^(14,16) The basis ground acceleration was assumed to be 0.1 g horizontal and 0.05 g vertical. For the design of primary piping systems, an equivalent horizontal force of 0.2 g was used to allow for dynamic responses of piping to the earthquake forces.

Critical areas and systems were designed for seismic Zone 3 forces, using a basic ground acceleration of 0.2 g horizontal and 0.1 g vertical. For critical piping systems, a horizontal force factor of 0.5 g was used in lieu of a dynamic response analysis.

The concepts of a Safe Shutdown Earthquake (SSE) and an Operating Basis Earthquake (OBE) for nuclear reactors were developed only recently. Because of differences between the criteria for N Reactor design and those specified for a SSE, N Reactor's 0.20 g design is equal to 0.25 g for nuclear safety systems. N Reactor design criteria limited the total combined stress to normal design stress limits, whereas the criteria for the SSE permits the total combined stresses to approach the yield strength of the material. Also included in the N Reactor design is a seismic trip set to shut the reactor down at an intensity of IV (MM) or a ground acceleration of 0.015 g.

A recent review⁽¹⁵⁾ of the 100-KE and 100-KW storage basins shows that they would remain functional through an earthquake producing a maximum horizontal ground acceleration at the site of 0.25 g, which exceeds Zone 2 requirements. In the extremely unlikely event of a Uniform Building Code earthquake in the 300 Area, maximum environmental effects would occur from a pipe break in the chemical transfer lines. It is expected that the chemical solutions in damaged piping and tanks would enter the process sewer trench or ground at the building site. Solutions would be neutralized to minimize impact. They would then eventually reach the Columbia River as water at a neutral pH range.

III.1.3.3 Floods

The flow for the projected maximum flood of 1,440,000 cfs at the Hanford site would produce a river surface elevation of 423 ft MSL.⁽¹⁷⁾ This is 2 ft above the 181-N river pump house floor. However, the emergency diesel pump drives are elevated about 5 ft above the floor allowing 3 ft of freeboard. The 300-Area elevation varies between 383 and 405 ft MSL. The probable maximum flood river elevation at the 300 Area is calculated to be 383 ft MSL,⁽¹⁷⁾ which presents no problem to fuels fabrication. A seismic induced flood would not exceed the probable maximum flood.⁽¹⁷⁾ The 100 year flood and the 500 year flood would be less than the probable maximum flood.

The highest recorded flood stage at the N Reactor site occurred in 1894 and was calculated to be 415 ft MSL. The area structures with the lowest elevation are the river pump house, which is at 421 ft MSL, or 6 ft higher than

the highest recorded flood.^(16,18) The probability of a naturally occurring flood of this magnitude is small, and the consequences would be mitigated by the storage capacity in the series of dams upstream from the Hanford site.

III.1.4 Accident Impacts

The discussions of accident source terms presented in this section are based on actual measurements of the radionuclide inventories in N Reactor systems and effluents. The radionuclide inventories for the postulated low-, moderate- and high-severity accidents were established by averaging a number of sample results obtained during different operating periods and conditions at N Reactor. In several of the postulated accidents, radionuclide inventory differences are accounted for by the accuracy limits of the analytical techniques used to characterize the inventory.

Prevention of accidents and protection against the occurrence of postulated accidents at the N Reactor is provided through design and construction features and proper operation of the facilities. A formal quality assurance program is in place covering new construction, modifications, materials, procedures and operations of all reactor systems.

Accident analyses are also presented in chapter 15 of the N Reactor Updated Safety Analysis Report (NUSAR). In NUSAR, Class 8 events^(a) are used, together with highly conservative assumptions, as the design-basis events to establish the performance requirements of engineered safety features. The highly conservative assumptions and calculations used in NUSAR are not suitable for environmental risk evaluation, because their use would result in a substantial overestimate of the environmental risk. For this reason, Class 8 events, such as a large or small pipe break, are evaluated realistically in this Environmental Assessment and may differ from the NUSAR postulated accident evaluations.

(a) Class 8 events are defined in the proposed Annex to Appendix D, 10 CFR Part 50, which was originally published in the Federal Register December 1, 1971 (36 FR 22851).

This section discusses nine postulated credible accidents including the estimated environmental effects of varying severity accidents up to and including major loss of reactor coolant. The methods used to estimate the consequence of reactor and irradiated fuel accidents are based upon guidelines published in Regulatory Guide 1.4 as shown in Appendix III.A. The effects of low-severity accidents have been derived by extrapolating operating experience. Accident analyses results for small and large pipe break accidents reflect current N Reactor configurations and operating mode, and the state-of-the-art analysis techniques.

In arriving at the type of postulated accidents presented in this environmental report, numerous conceivable accidents were studied, with emphasis on the conditions or circumstances most likely to result in an accident or abnormal event. In addition to the historical incidents that have occurred at N Reactor, the following accidents were considered:

- postulated primary loop spill cooler failure
- postulated 1314-N liquid waste loadout facility railroad tank car leak
- postulated emergency dump tank leak
- postulated oil line leaks
- postulated N Reactor fuel cladding failure
- postulated small pipe break
- postulated 105-KE fuel storage facility cask drop
- postulated large pipe break
- postulated spent fuel shipping accident

For the postulated loss-of-coolant accidents, the primary coolant discharged from the small and large pipe breaks would contain radionuclides originating from activation of corrosion products in the coolant and from fission products remaining from fuel element cladding leaks. The fission product inventory, in the coolant at the time of the pipe break, would depend on the operating time since the last fuel cladding leak, the decay time for released radionuclides and the coolant feed-and-bleed rates. Two cases were considered: one in which the radioactive inventory in the coolant was essentially at the normal equilibrium

TABLE III.9. Historical N Reactor Accidents

Date	Description of Incident	Environmental Consequences
3-25-76 11-20-76	On two separate occasions the primary loop spill cooler equipment leaked because of the breakage of primary side cooling tubes. N Reactor was shut down.	Unplanned release of a minor amount of radioactivity contamination to the Columbia River. Fifty-year dose commitment from each incident was calculated to be less than 0.01 man/rem (whole body).
9-66 7-22-73	During September of 1966, a variance in the N Reactor diesel oil inventory prompted pressure testing of the lines, revealing a leak between the storage tanks and the 166-N building. In July 1973, an evaluation indicated a leak of No. 6 fuel oil from the line between the unloading station and the 166-N pumphouse.	An estimated 80,000 gal of diesel oil leaked into the soil around the tank in 1966 but was trapped in the trench and small amounts reached the Columbia River. Less than 200 gal of fuel oil leaked locally from a buried line in 1973, and none reached the river.
5 times/yr	N Reactor fuel elements fail by two mechanisms: (1) Wire fretting failures result in no measurable increase in the normal radioactive release rates from the reactor. These fretting failures generally result in small holes in the fuel cladding. (2) End-associated failures result in significant degradation of the fuel cladding, including tearing of the cladding, limited swelling of the fuel, and oxidation of a significant quantity of uranium.	As a result of wire fretting failures, releases of uranium and fission products are so small that they are immeasurable relative to the normal inventory in the N Reactor primary coolant system. In FY-1975 and 1976, three wire fretting failures occurred at N Reactor. When an end-associated failure occurs, the uranium and fission products released to the primary coolant system are eventually released to the 1301-N Facility via the coolant system bleed. During FY-1975 and 1976, eight end-associated failures occurred at N Reactor. (See Section III.1.4.6 for a complete discussion of the radioactive material involved in the calculated population doses from the worst fuel cladding failure observed at N Reactor.)
6-21-67	A pump lubricating oil ^(a) fire in Cell 3 of the 109-N building occurred in an area normally free of combustible materials. Oil escape was caused by the inadvertent closure of the pump housing drain line valves in the lubricating oil drain system. About 80 gal of oil overflowed the Cell 3 pump bearing vents into an area containing electrical equipment piping and sample lines. Shortly after the fire (estimated duration, 13 min) was reported, the N Reactor was manually scrammed. The ventilation supply and exhaust valves associated with Cell 3 were then closed manually from a 109-N building control panel.	No damage occurred to primary loop components, nor any structural damage to the 109-N building. Electrical equipment, instrumentation, piping, valves, the overhead crane, primary pump bracing and insulation were damaged in Cell 3. While repair and cleanup activities were in progress in Cell 3, N Reactor was operated with all of the other five steam generator cells in service. The steam generator internal heat exchanger tubes that provide separation between the radioactive primary coolant system and the secondary coolant system were not damaged by the fire.

^(a) Steam generator cells (each 50 ft x 94 ft x 50 ft high) are connected to the pipe gallery by large "windows" near the cell ceiling and by a sump that extends under the wall between the cells and the pipe gallery. Primary loop coolant lines extend from the cells through the sumps to the pipe gallery. Two cylindrical heat exchangers are located on pedestals in each of the six cells in the 109-N building. Each of the five primary coolant system pumps located inside the steam generator cells is provided with a self-contained lubricating oil system for the primary coolant pump bearings and pedestal bearings. Reconditioned low-pressure oil is circulated through the closed loop system, which includes a 500-gal makeup oil reservoir located outside the 109-N building.

level with no detectable fuel cladding leaks and one in which the pipe break was assumed to occur 24 hr after reactor startup following removal of a leaking fuel element. The latter case was selected because it maximizes the consequences of the postulated accidents.

The primary coolant radionuclide inventories for the selected cases were determined using the results of analyses of the primary coolant from January 1976 to April 1976. This period included reactor operations with normal radionuclide inventories and the highest fission product inventories experienced in N Reactor following fuel cladding failures.

III.1.4.1 Historical Accidents

A historical review of the accident experience at Hanford provides insight on the types and extent of the impacts that have occurred. The accident summaries and the associated environmental consequences given in Table III.9 are representative of the kinds of accidents that have occurred at N Reactor.

III.1.4.2 Postulated Primary Loop Spill Cooler Failure

The spill cooler is provided to reduce the temperature of the primary coolant bleed so that the coolant does not flash to steam upon pressure reduction. Inside the spill cooler, 560 heat exchanger tubes provide separation between the radioactively contaminated primary coolant system and the secondary cooling system. Unplanned discharge of small quantities of radioactive material to the Columbia River occurred in the two 1976 incidents over a 3- to 5-hr period. Future accidents could result in discharges to the Columbia River or to the 1301-N Facility, depending on the routing of the secondary coolant discharge.

Mitigating Conditions

A radiation monitor on the spill cooler secondary coolant water discharge line will activate an alarm in the control room should normal background radiation be exceeded. The calculated system response corresponds to a 1-gpm leak of primary water into the secondary coolant.

Spill cooler secondary coolant has been temporarily rerouted to the 1301-N Facility. Prior to this modification, N Reactor has been shut down when a spill cooler failure occurred. The spill cooler has failed on only two occasions during the lifetime of N Reactor. The cooler has been cleaned and the water velocity reduced to improve the integrity of the primary coolant tubes.

Source Term and Potential Environmental Impact

A single tube failure with a leak rate of about 210 gpm is assumed should a primary loop spill cooler failure occur. The population dose commitment represents less than 0.3% of that due to N Reactor's annual radioactive releases under normal operating conditions (normal operation dose commitment equals 0.77 man-rem).

III.1.4.3 Postulated 1314-N Liquid Waste Loadout Facility Railroad Tank Car Leak

Chemical decontamination of the primary system of the 105-N reactor generates radioactive liquid waste consisting mainly of water and neutralized phosphoric acid. This waste is collected in the 900,000-gal 1310-N chemical waste storage tank. The waste is then transferred from the 1310-N storage tank by the 1310-N recirculation pump to a 1,000-gal tank at the 1314-N loadout facility. From the loadout facility, the liquid waste is pumped to special 20,000-gal stainless-steel waste tanks mounted on railroad cars. These cars are transported to the 200 Area for processing and disposal of the waste. The 1314-N loadout station is equipped with a catch basin.

The decontamination tank cars are positioned over the catch basin during loading operations. The liquid waste consists primarily of a solution of neutralized phosphoric acid containing dissolved and suspended radioactive material from decontamination of the N Reactor primary coolant system. The used acid is neutralized in the storage tank by the addition of sodium hydroxide. An excess of sodium hydroxide is added to bring the pH of the solution to between nine and ten to assure compatibility with the waste treatment process operated by

Rockwell Hanford Operations. The maximum volume of waste and, therefore, the maximum inventory of radionuclides present at the 1314-N facility occurs when the railroad tank cars are full.

The postulated accident involves leakage of a full railroad tank car load of radioactive liquid waste. Up to 20,000 gal of radioactive liquid containing approximately 20 Ci of mixed corrosion, activation and fission products could be leaked to the 10,000-gal catch basin and surrounding ground areas if the loaded tank car developed a large leak in the bottom of the tank. However, a large leak in a fully loaded tank car is considered to be a highly unlikely event, because small leaks in a tank car are easily detected and corrected before the leak becomes large.

Mitigating Conditions

Catch basins constructed at the 1314-N loadout facility would contain approximately 10,000 gal of waste. Leaks with volumes in excess of this amount would overflow to the surrounding soil areas. Temporary diking could confine the liquid waste to a limited area. Liquid wastes would rapidly percolate into the soil, and the radionuclides would be removed and retained by the soil.

The catch basin drains to a 500-gal overflow catch tank. A leak from a tank car would be detected by the catch tank level indicator, which activates a visual and an audible alarm should the level in the tank exceed a preset alarm point. An annunciator in the 105-N control room would also be activated by these alarms.

At six-month intervals, the liquid waste loading instrumentation is tested with a variable pressure source. Both the tank car and catch tank level indicators are exercised. Trips are indicated by the horn and flashing red light at the 1314-N building and an annunciator in the 105-N control room. Snap Tite Couplers are pressure tested to assure that no leakage occurs when the transfer system is pressurized.

A large leak in a fully loaded tank car is considered to be a highly unlikely event.

Source Term and Potential Environmental Impact

In the event of such a high-volume leak, some of the radionuclides could possibly reach the Columbia River. Under the worst-case conditions, it is estimated that approximately 1,000 gal of liquid waste containing 1 Ci of radioactive material could potentially reach the river. The radiological consequences of such a liquid waste loadout facility accident would be less significant than the fuel cladding failure accident discussed in Section III.1.4.6.

III.1.4.4 Postulated Emergency Dump Tank Leak

In the extremely unlikely event that the N Reactor primary coolant system dump valves are actuated, a method for quenching the hot pressurized coolant must be provided to prevent the coolant from flashing to steam. The emergency dump tank has been designed to provide this quenching action and to contain the gases and fission products in the condensed primary coolant.

The emergency dump tank has a total capacity of 1,300,000 gal and is a steel-walled cylindrical vessel with a hemispherical dome. The tank rests on a reinforced concrete foundation. A minimum of approximately 680,000 gal of unheated water is maintained in the tank at all times to provide the required quenching should a primary coolant spill dump occur.

In this accident analysis, it is assumed that the steel shell of the tank fails and allows the normal 680,000-gal inventory of quench water in the tank to reach the Columbia River. Because of small leakage through the primary coolant system dump valves, the quench water in the tank normally contains a small inventory of radioactive materials.

Mitigating Conditions

The emergency dump tank was designed and fabricated according to strict performance standards that assure that the tank integrity will be maintained throughout the range of expected operating conditions. The dump tank receives a detailed internal inspection every three years as part of the N Reactor

equipment maintenance system. As part of the routine operations patrol, the dump tank level is checked every 8-hr shift. An overflow pipe is provided inside the dump tank to route excess quench water to the 1301 N Facility.

Source Term and Potential Environmental Impact

The radiological consequences of inadvertently releasing the entire 680,000 gal of quench water, primary coolant valve leakage mixture containing about 1.6 Ci of radionuclides, are considered to be no more severe than the fuel cladding failure accident discussed in Section III.1.4.6. The population dose commitment would represent less than 0.0066 man-rem.

III.1.4.5 Postulated Oil Line Leaks

In this postulated accident a break occurs in the diesel oil transfer line that connects the 105,000-gal oil tanks at 166-N to the 182-N building.

A daily review is made of the fuel oil inventory in the 100-N Area. Discrepancies in the oil inventory due to a leak would be discovered during this daily review. The diesel oil tanks are also equipped with level indicators and the tank level is checked every 2 hr.

During FY 1981, a diesel oil leakage control project will be completed at 100-N Area. The project will include improved level monitoring on the diesel oil storage tanks, upgraded administrative controls, a network of groundwater wells for early detection of diesel oil leakage infiltrating to the water table, and oil recovery and clean-up equipment to prevent any diesel oil that may leak to the river from traveling downstream of 100-N Area. The project is intended to further reduce the potential for an oil spill and to clean up any oil spills that may occur.

Source Term and Potential Environmental Impact

The maximum potential spill could be 100,000 gal. Some fraction of this oil would seep to the groundwater and to the river. However, this oil would be recovered at the riverbank and the impact of such oil seepage would not have significant impact on river biota or downstream potable water supplies.

III.1.4.6 Postulated N Reactor Fuel Cladding Failure

Most of fuel cladding failures that occur in N Reactor do not cause measurable increases in the normal release of radionuclides to the 1301-N Facility or the Columbia River. On occasion, a severe fuel cladding failure occurs and causes a measurable increase only in release rates to the 1301-N Facility. The most severe fuel cladding failure, in terms of offsite dose commitment, observed during the operational history of the N Reactor occurred on January 12, 1976. The following information describes this fuel cladding failure and a hypothetical worst case fuel cladding failure.

N Reactor fuel elements fail by two mechanisms: wire fretting failures and end-associated failures. Wire fretting failures are more common than end-associated failures but result in no measurable increase in the normal radioactive release rates from the reactor. These fretting failures generally result in small holes in the fuel cladding. Only minor amounts of uranium and fission products are released to the primary coolant system.

End-associated failures occur less frequently but are more severe in nature than wire fretting failures. Significant degradation of the fuel cladding (including tearing of the cladding), limited swelling of the fuel, and oxidation of a significant quantity of uranium have occurred. The uranium and fission products that are released to the primary coolant system are eventually released to the 1301-N Facility via the coolant system bleed.

Following detection of a wire fretting or end-associated failure, the reactor is rapidly shut down. The resulting rapid cool down of the primary loop minimizes the oxidation of uranium through the failed fuel cladding. The uranium and fission products released before cooldown of the primary coolant system are entrained in the coolant and carried to the 1301 N Facility in the coolant bleed. The effluent, once in the facility, then percolates through the soil and reaches the river after approximately five days, which is the shortest travel time. No measurable increase in release rates of radioactive material to the atmosphere have been observed following any fuel cladding failure in N Reactor.

For cladding failure events, a worst-case failure has been hypothesized. This worst-case failure establishes a very conservative upper boundary on the severity of a single-element cladding failure. The maximum hypothetical single-element failure was defined to be a 1300 g loss of highly irradiated uranium to the primary coolant system. The fission product inventory in the 1300 g of highly irradiated uranium was calculated using the RIBD computer code. The total inventory was assumed to discharge to the 1301 N Facility. This assumption is very pessimistic. Extensive studies of past fuel cladding failures in N Reactor show that large fractions of the inventories of the significant radionuclides are retained in the primary coolant system where they eventually decay.

The calculated releases of significant radionuclides from this hypothetical worst-case single-element failure to the 1301-N Facility are shown in Table III.10.

TABLE III.10. Radionuclide Release to 1301-N Facility
Hypothetical Maximum Single-Element Cladding Failure

<u>Radionuclide</u>	<u>Discharge to 1301-N (Ci)</u>
⁹⁰ Sr	14
⁹⁹ Mo	1100
¹³¹ I	700
¹³⁷ Cs	26
¹⁴⁰ Ba	1000
¹⁴⁰ La	1000

The measured releases to the 1301-N Facility following the worst fuel cladding failure observed at N Reactor (November 3, 1979) are shown in Table III.11 for comparison.

Table III.11 Radionuclide Releases Following the November 3, 1979
Fuel Cladding Failure.

RADIONUCLIDE	ONSITE DISCHARGE TO 1301-N CRIB (Ci)			SEEPAGE AT THE RIVERBANK SPRINGS (Ci)
	GROSS RELEASE	NORMAL RELEASE	NET RELEASE	ENVIRONMENTAL RELEASE TO RIVER (NET) (a)
H-3	10.0	5.0	5.0	5.0
P-32	15.0	2.2	13.0	.0068
Cr-51	34.0	2.5	32.0	.012
Mn-54	20.0	20.0		---(c)
Fe-59	25.0	15.0	10.0	---(c)
Co-58	3.3	1.0	2.3	.00061
Co-60	21.0	19.0	2.0	.00087
Sr-89	190.0	6.0	180.0	.14
Sr-90	8.2	1.1	7.1	.079
Zr-95	92.0	4.0	88.0	---(c)
Nb-95	110.0	6.0	100.0	---(c)
Mo-99	200.0	65.0	140.0	---(c)
Ru-103	97.0	2.0	95.0	.098
Sb-124	.30	.25	.05	.00005
I-131	340.0	17.0	320.0	2.3
Xe-133	36.0	10.0	26.0	.29
Cs-134	4.3	2.0	2.3	---(c)
Cs-137	17.0	8.0	9.0	.00043
Ba-140	170.0	9.0	160.0	---(c)
La-140	3300.0	150.0	3200.0	---(b)
Ce-141	200.0	2.7	200.0	---(c)
Ce-144	88.0	13.0	75.0	---(c)
Nd-147	77.0	10.0	70.0	---(c)
Sm-153	130.0	15.0	120.0	---(b,c)
Np-239	680.0	2.0	680.0	---(b,c)
Am-241	8.7	0.0	8.7	---(c)
Pu-238	.0078	.0036	.0042	.0000021
Pu-239/240	.042	.018	.024	.0000060
Short-lived Radionuclides (T-1/2 <24 hrs)	5200.	1600.	3600.	---(b)

- (a) Except as noted, results were calculated using 1977-1979 average transmission factors.
 (b) Short half-life, not identified at riverbank springs.
 (c) Transmission factor has not been calculated because this radionuclide has not been identified in spring water.

NOTE: Np-239 is produced from the exposed uranium which creates these daughter products more abundantly after a cladding failure.

Mitigating Conditions

Stringent quality control measures are used during the manufacture of N Reactor fuel to ensure that the fuel cladding has an extremely high integrity. Numerous tests are made of the cladding during the manufacturing process, and each completed fuel element is given a destructive test in an autoclave system before shipment to the reactor.

In addition, special procedures are used at N Reactor to minimize fuel damage during charge and discharge operations and during irradiation. These procedures include control of fuel charging pressure and removal of small debris from the primary coolant system in specially provided crud traps.

The fuels fabrication quality control program and the special fuel handling procedures at N Reactor combine to reduce the annual fuel cladding failure rate to a very low level.

Should a fuel cladding failure occur, N Reactor is provided with a fuel cladding failure detection system that can detect small cladding failures in any one of the 1,003 pressure tubes. The fuel failure detection system can detect such a failure within approximately 10 min after the cladding failure occurs. This rapid response allows a quick shutdown of the reactor following a fuel cladding failure. The rapid cooldown of the reactor greatly mitigates the quantity of uranium and fission products that are released to the primary coolant system and subsequently to the 1301-N Facility.

Source Terms and Potential Environmental Impact

The radionuclides released from the primary coolant to the 1301-N Facility are, for the most part, retained in the soil beneath the crib and trench. A small fraction of these radionuclides is transported through the soil to the Columbia River. Extensive studies have been conducted which establish the transmission factors for all radionuclides that are discharged to the 1301-N Facility. These

transmission factors are the ratio of the radionuclides reaching the Columbia River divided by the radionuclides released to the facility. The transmission factors for the significant radionuclides released to the facility following the hypothetical maximum single-element fuel cladding failure are listed in Table III.12.

TABLE III.12. Transmission of Radionuclides from
1301-N Facility to Columbia River
Following the Hypothetical Maximum Single-Element
Fuel Cladding Failure

<u>Radionuclide</u>	<u>Transmission Factor (%)</u>
^{90}Sr	4.0
^{99}Mo	0.05
^{131}I	1.0
^{137}Cs	0.2
^{140}Ba	0.08
^{140}La	0.08

Using these transmission factors and the releases to the 1301-N Facility from the hypothetical maximum single-element fuel cladding failure, radionuclide releases to the Columbia River have been calculated and are given in Table III.13.

TABLE III.13. Radionuclide Release to Columbia River

<u>Radionuclide</u>	<u>Maximum Predicted Failure (Ci)</u>
^{90}Sr	0.57
^{99}Mo	0.55
^{131}I	7.0
^{137}Cs	0.05
^{140}Ba	0.8
^{140}La	0.8

For purposes of comparison, the measured releases to the Columbia River following the January 12, 1976, failure are shown in Table III.11.

Doses resulting from intake and exposure to liquid radioactive effluents resulting from this hypothetical accident were calculated using the computer codes ARRRG and FOOD. These programs calculated the annual and 50-yr dose commitments to the maximum individual and the population resulting from the radioactive effluents from the fuel cladding failure. Detailed description of these computer codes is given in ERDA-1538, Appendix III-A.

The calculated doses due to the hypothetical, maximum, single-element fuel cladding failure are shown in Table III.14. The calculated doses resulting from the January 12, 1976, fuel cladding failure are listed in Table III.15.

TABLE III.14. Radiation Doses from Hypothetical Maximum Single-Element Fuel Cladding Failure

<u>Dose</u>	<u>Thyroid</u>	<u>Whole Body</u>
Maximum Individual, mrem	5.4×10^{-1}	2.1×10^{-2}
50-Yr Dose Commitment, man-rem/50 yr	2.8×10^0	2.6×10^{-2}

TABLE III.15. Radiation Doses Following January 12, 1976, Fuel Cladding Failure

<u>Dose</u>	<u>Thyroid</u>	<u>Whole Body</u>
Maximum Individual, mrem	6.4×10^{-2}	7.0×10^{-3}
50-Yr Dose Commitment, Total Population, man-rem/50 yr	3.3×10^{-1}	5.3×10^{-3}

III.1.4.7 Postulated Small Pipe Break (100-N Area)

The N Reactor coolant is distributed from the main supply headers through risers that supply connectors to each individual pressure tube in which the reactor fuel is located. In a similar fashion, the effluent coolant from each pressure tube is collected into risers and then into a return header for recirculation back to the steam generators and the primary coolant pumps (see Section II.3.1.3). This section considers an accident in which a break occurs in one of the 2 5/8 in. O.D. connector pipes in the part of the coolant system located in the 105-N building.

The break is assumed to be double-ended such that the coolant would flow from both sides of the break. The coolant loop would begin to depressurize as the coolant flashes partially to steam through the break, and the reactor would automatically shut down due to a low pressure tube flow trip. However, depressurization would not continue to the point of initiating emergency coolant flow since the coolant makeup system can compensate for the coolant loss through the break. In case of a front connector break, reverse flow through the stricken tube would occur because of the higher rear riser pressure. The coolant flow would be adequate to preclude damage to the fuel in the stricken tube.

The escaping steam would cause enough pressure increase in the confinement zone to actuate the confinement system. When this occurred, the 105-N building ventilation system would shut down, and the ventilation confinement valves would close. The pressure rise in 105-N would then accelerate, and the pressure in 109-N would also increase as the steam and air mixture flows through vents in the common barrier wall between the two buildings. As the pressure continued to rise, the 109-N confinement system would be actuated, as would the fog spray system in both buildings.

The pressure transient from the double-ended small pipe break would not cause any steam vent weather caps to open. The condensing and cooling effect of the fog sprays would keep the pressure within confinement limits. Twenty minutes

following the pipe break, the pressure would reduce to levels that would actuate re-opening of the ventilation valves. Re-opening of the ventilation valves would establish a filtered release flow path for the building atmosphere.

For this analysis the reactor is assumed to be in a normal operating mode at a power level of approximately 4,000 MWt at the time the pipe break occurs. The primary loop blowdown transients and the flow characteristics of the coolant discharged from the break have been addressed in previous studies. It is further assumed that all engineered safeguards and other safety features of the reactor function as designed.

The fog spray system, actuated automatically, would continue to operate until manually reset. In addition to condensing steam from the primary coolant, the system washes radionuclides from the confinement zone atmosphere.

The fog spray water, the condensed steam, the liquid coolant from the primary loop and the continuing leakage from the break would accumulate in the below-grade levels of the confinement structure. This water with its radionuclides would, after sufficient decay time, be pumped to the 1301-N Facility. The radionuclides contained in the water are considered as event source terms.

Mitigating Conditions

The process limits specified for N Reactor are well within the design capabilities of the reactor systems and are closely monitored at all times, both by automatic surveillance instrumentation and by operator surveillance. The process limits are strictly enforced. Foreseeable operational transients will not produce stresses that would initiate the postulated pipe break.

The piping system and associated equipment are subject to a stringent inspection program for material degradation, and the system is maintained according to a well-organized formal maintenance program with specified detailed criteria for equipment performance and detailed procedures for inspections. Over-pressure protection is provided by reactor shutdown instrumentation and relief valves set

to trip at pressures below the design rating of the primary coolant system. Quality Assurance staff members perform periodic audits of both the operation of the reactor and the maintenance of the reactor systems.

The design features of the reactor and its associated equipment take into account the foreseeable natural disasters that otherwise could initiate a small pipe break accident. The piping design meets the criteria for the design basis earthquake, and the confinement structure meets the Uniform Building Code requirements to withstand the maximum Hanford wind conditions, as described in Section III.1.3. If a probable maximum flood should occur, it would not prevent adequate reactor cooling (see Section III.1.3).

The above considerations indicate that the small pipe break accident is a highly unlikely event. However, if such a break does occur, the engineered safeguards provided in the reactor design substantially mitigate the environmental impact of such an accident. The emergency coolant system would not be activated by this accident, but if needed, could provide coolant for the reactor core from diesel-powered pumps (automatically actuated) that are supplied initially from stored, treated water and eventually from the Columbia River, when the stored, treated water is used up.

With adequate cooling provided from the coolant makeup system, fuel damage would not occur, and radioactive releases from the reactor would be limited to the radionuclide inventory in the primary reactor coolant. The fog spray system is supplied from either an electric or two automatic-start diesel-powered pumps; it reduces the effect of the accident pressure transient and washes volatilized radionuclides out of the air. The confinement system retains radionuclides on particulate and halogen filters to substantially reduce the releases to the environment. The emergency coolant and confinement systems incorporate redundant features such that the failure of an active component will not impair system operation.

Source Terms and Potential Environmental Impact

The primary coolant discharged from the break would contain radionuclides originating from activation of corrosion products in the coolant and from fission products remaining from fuel element cladding leaks. The fission product inventory in the coolant at the time of the pipe break would depend on the operating time since the last fuel cladding leak, the decay time for released radionuclides and the coolant feed-and-bleed rates. Two cases were considered: one in which the radioactive inventory in the coolant was essentially at the normal equilibrium level with no detectable fuel cladding leaks and one in which the pipe break was assumed to occur 24 hr after reactor startup from a shutdown following removal of a failed fuel element. The latter case was selected for analysis because it maximizes the consequences of the postulated accident.

The primary coolant radionuclide inventories were determined using the results of analyses of the primary coolant from January 1976 to April 1976. This period included reactor operations with normal radionuclide inventories and the higher fission product inventories experienced following fuel cladding failures. Radionuclides not reported in the coolant analyses were calculated by normalization of RIBD computer code⁽¹⁸⁾ information to measured radionuclide concentrations. These semiempirical radionuclide inventories were used for calculating the radioactive releases following the pipe breaks.

Primary coolant system discharge from the small pipe break was based on a calculated initial maximum flow from a guillotine break of a 2-1/2-in. pressure tube connector. The flow rate was assumed to decay exponentially until the entire primary coolant system inventory was ejected to the Zone I. This information was then used to calculate the quantity of steam released to the atmosphere through the confinement system leaks and that condensed by fog spray.

A modified version of the CORRAL computer code⁽²⁰⁾ was used to determine the time- dependent radioactive material plate-out and rain-out within the confinement system and the releases through the confinement system boundaries. This computer code was patterned after the CORRAL code developed for the Nuclear Regulatory Commission's Reactor Safety Study.⁽¹⁹⁾ Inputs to the code included information pertaining to the confinement releases, filter efficiencies and organic iodide formation.

The organic iodide formation for the conditions existing during the pipe break accident were calculated and compared to the assumptions specified in Regulatory Guide 4.2. The calculated values are more conservative than the assumptions used in Regulatory Guide 4.2 as used in the accident analyses discussed below. The dose calculations were made in general accord with Regulatory Guide 1.4 guidance.

The primary coolant radioactive material inventories and results of the CORRAL computer code analyses were used to determine the concentration of radionuclides in the airborne effluents from the ventilation stack effluent discharges to the atmosphere and from leaks in the confinement structure. These source terms are listed in Table III.16.

The calculated nonoccupational population radiation doses at the site boundary and to the low population zone within 50 miles of N Reactor are given in Table III.18. Individual and population doses were calculated using the guidance in Regulatory Guide 1.4 as described in Appendix III-A and the releases given in Table III.17.

TABLE III.16. Primary Coolant System Radionuclide Inventory

Radionuclide	Normal		Maximum	
	Conc (pci/l)	Inventory (Ci)	Conc (pci/l)	Inventory (Ci)
Kr-85m	0	0	4.28×10^8	3.24×10^2
Kr-85	0	0	3.83×10^7	2.90×10^1
Kr-87	7.8×10^6	5.9×10^0	5.83×10^7	4.41×10^1
Kr-88	1.5×10^7	1.1×10^1	8.60×10^7	6.51×10^1
I-131	6.2×10^5	4.7×10^{-1}	7.82×10^7	5.92×10^1
I-132	2.0×10^7	1.5×10^1	1.33×10^8	1.01×10^2
I-133	1.1×10^7	8.3×10^0	1.81×10^8	1.37×10^2
I-134	2.9×10^6	2.2×10^1	2.15×10^8	1.63×10^2
I-135	1.6×10^7	1.2×10^1	1.74×10^8	1.32×10^2
Xe-131m	0	0	5.40×10^5	4.09×10^{-1}
Xe-133m	0	0	2.47×10^7	1.87×10^1
Xe-133	1.5×10^6	1.1×10^0	1.72×10^8	1.30×10^2
Xe-135m	0	0	3.22×10^7	2.44×10^1
Xe-135	7.9×10^6	6.0×10^0	2.96×10^7	2.24×10^1

TABLE III.17. Fission Product Releases From Small Pipe Break Accident

<u>Radionuclide</u>	<u>Release From Confinement (Curies)</u>		
	<u>Steam Vent</u>	<u>Filtered Release</u>	<u>Building Leakage</u>
Kr-85m	0.	4.96×10^{-1}	3.87×10^{-2}
Kr-85	0.	7.32×10^0	3.87×10^{-1}
Kr-87	0.	3.15×10^0	1.68×10^{-1}
Kr-88	0.	7.97×10^0	4.23×10^{-1}
I-131	0.	6.64×10^{-1}	1.06×10^{-1}
I-132	0.	5.08×10^{-1}	8.31×10^{-2}
I-133	0.	1.38×10^0	2.20×10^{-1}
I-134	0.	3.56×10^{-1}	6.19×10^{-2}
I-135	0.	1.06×10^0	1.70×10^{-1}
Xe-131m	0.	1.03×10^{-1}	5.42×10^{-3}
Xe-133m	0.	4.50×10^0	2.38×10^{-1}
Xe-133	0.	3.26×10^1	1.73×10^0
Xe-135m	0.	2.00×10^{-1}	1.15×10^{-2}
Xe-135	0.	5.34×10^0	2.30×10^{-1}

TABLE III.18. Population Commitment Resulting From the
Postulated Small Pipe Break Accident(a)

<u>Population Location</u>	<u>Organ Dose in Rem</u>	
	<u>Whole Body</u>	<u>Thyroid</u>
Site Boundary	1.09×10^{-4}	1.67×10^{-2}
Low Population Zone	1.04×10^{-6}	2.34×10^{-4}

(a) See Appendix III-A for assumptions and dose calculation parameters.

III.1.4.8 Postulated Large Pipe Break (100-N Area)

This section considers an accident in which a double-ended break occurs in the 26-in. return pipe located in the 109-N pipe gallery. Six 26-in. pipes return coolant to the six cells from a single manifold connected to the 16 reactor outlet risers. (See Section II.3.1.3 for further description of the primary coolant system.)

It is assumed that the pipe break causes a displacement of the pipe ends so that flow from each end of the break is unrestricted. A rapid blowdown of the primary loop would be initiated with the primary coolant flashing to a mixture of steam and hot water. The mixture would be discharged into the 109-N pipe gallery beginning a pressure transient in the 109-N building Zone 1. The reactor would automatically shut down as a result of flow reduction to the pressure tubes or the primary loop pressure decreasing to the low pressure trip.

The emergency cooling system would be actuated by a combination of selected pressure tube low-flow, low-coolant pressure and high confinement zone pressure. As the pressure in the confinement zone of the building rises, closure of the confinement system would be initiated, first in the 109-N building, then in the 105-N building as the pressure transient progresses into 105-N through vents in the barrier wall between the two buildings. The confinement trip would close all exits from the confinement structure and shut down the building ventilation systems.

The continuing pressure transient would actuate the fog spray system, which partially condenses the steam to reduce the magnitude of the pressure transient, and washes radionuclides out of the building atmosphere. A further increase in pressure would remove weatherhead caps from steam vents on the building roofs to release the steam to the atmosphere before the confinement structure could be overpressurized. After the pressure transient subsides, butterfly valves in the steam vents would automatically close, and ventilation exhaust

valves in the 105-N building would reopen automatically on a timed sequence to establish a filter release path for the atmosphere within the confinement structure.

For this analysis it is assumed that the reactor is in a normal operating mode at a power level of 4,000 MWt at the time the pipe break occurs. The primary loop blowdown transients and the flow characteristics of the coolant discharge from the break were calculated on this basis using the RELAP4 code.⁽²⁴⁾

It is further assumed that all engineered safeguards function as designed. As discussed earlier, the fog spray system would wash most of the radionuclides out of the volatilized coolant discharged from the primary loop. The fog spray water, condensed coolant from the steam, and a continuing discharge of liquid coolant from the break would all accumulate in the below-grade regions of the 105-N and 109-N buildings. This water would eventually be pumped to the 1301-N Facility. See Appendix III.A for other parameters and assumptions.

Mitigating Conditions

The operating limits specified for the reactor are well within the design capabilities of the reactor systems and are closely monitored at all times both by automatic surveillance instrumentation and by operator surveillance.

The process limits are strictly enforced. Foreseeable operating transients will not produce stresses that would initiate the postulated pipe break. The piping system and associated equipment are subject to a stringent inspection program for material degradation, and the system is maintained according to a well-organized formal maintenance program with specified detailed criteria for equipment performance and detailed procedures for inspections.

Overpressure protection is provided by reactor shutdown instrumentation and relief valves set to trip at pressures below the design rating of the primary coolant system. Quality Assurance staff members perform periodic audits of both the operation of the reactor and the maintenance of the reactor systems.

The design features of the reactor and its associated equipment take into account the foreseeable natural disasters that could initiate a large pipe break accident. The piping design meets the criteria for the design basis earthquake, and the confinement structure meets the Uniform Building Code requirements to withstand the maximum Hanford wind conditions, as described in Section III.1.3. If a probable maximum flood should occur, it would not interfere with cooling the reactor (see Section III.1.3).

The above considerations indicate that the large pipe break accident is an extremely unlikely event. However, if such a break does occur, the engineered safeguards provided in the reactor design would substantially mitigate the environmental impact of such an accident. Although the pipe break would cause the loss of the recirculating coolant, the emergency coolant system would provide once-through cooling for the reactor core from diesel-powered pumps (automatically actuated). The pumps are supplied initially from stored, treated water and eventually from the Columbia River when the stored, treated water supply is used up. With the emergency coolant system in operation, fuel damage would not occur, and radioactive releases from the reactor would be limited to the radionuclide inventory in the primary reactor coolant.

The fog spray system, if needed, would also be supplied from the high-pressure raw water system (see Section II.3.1.4) with automatic-start diesel-powered backup pumps. The fog spray reduces the effect of the accident pressure transient and washes volatilized radionuclides out of the air. The confinement system retains radionuclides on particulate and halogen filters to substantially reduce the releases to the environment. The emergency coolant and confinement systems incorporate redundant features such that the failure of an active component will not impair system operation.

Source Terms and Potential Environmental Impact

The primary coolant discharged from the break would contain radionuclides originating from activation of corrosion products in the coolant and from fission products remaining from fuel element cladding leaks. The fission product inventory in the coolant at the time of the pipe break would depend on

the operating time since the last fuel cladding failure, the decay time for released radionuclides and the coolant feed-and-bleed rates. Two cases were considered: one in which radioactive inventory in the coolant was essentially at the normal equilibrium level and one in which the pipe break was assumed to occur 24 hr after reactor startup from a shutdown following removal of a failed fuel element. The latter case was selected for analysis because it maximizes the consequences of the postulated accident.

The primary coolant radionuclide inventory was determined using the results of analyses of the primary coolant from January 1976 to April 1976. This period included reactor operations with normal radionuclide inventories and the higher fission product inventories experienced following fuel cladding failures. Radionuclides not measured in the coolant analyses were calculated by normalization of RIBD⁽¹⁸⁾ computer code information to measured radionuclide concentrations. These semiempirical radionuclide inventories were used in calculating the radioactive releases following the pipe breaks.

The primary coolant discharged to the Zone I atmosphere following the outlet manifold break (large pipe break case) was determined from the RELAP4 computer code break analyses of the N Reactor coolant system. The break and initial operating conditions were selected to maximize the quantity of coolant discharged to the confinement system Zone I atmosphere. The RELAP4 computer code was also used to determine the distribution of steam flow between the confinement system steam vents, which discharge to the atmosphere, and the confinement system Zone I atmosphere.

A modified version of the CORRAL computer code was used to determine the time-dependent radioactive material plate-out and rain-out within the confinement system and the releases through the confinement system boundaries. The CORRAL computer code used in analyses was patterned after the CORRAL code developed for the Nuclear Regulatory Commission's NRC Reactor Safety Study.⁽¹⁹⁾ Inputs to the code included information pertaining to the confinement system and fog spray system, system leakage, RELAP4 code analyses results, filter efficiencies, and organic iodide formation. The organic iodide formation for the conditions

existing during the postulated pipe break accident was calculated and compared to the assumptions specified in Regulatory Guide 4.2.

These calculated values were more conservative than the values used in Regulatory Guide 4.2 and were used in the accident analyses described here.

The primary coolant radionuclide inventories and results of the CORRAL computer code analyses were used to determine the concentration of radionuclides in the gaseous effluents from the steam vents, the ventilation stack effluent and the discharge to atmosphere from leaks in the confinement structure. The primary coolant radionuclide concentrations and inventories are listed in Table III.19, and the releases in Table III.20.

The calculated radiation dose at the site boundary and to the population within 50 miles of N Reactor are given in Table III.21. The nonoccupational population doses were calculated using guidance provided in Regulatory Guide 1.4 and the source terms in Table III.18. See Appendix III.A for other parameters and assumptions.

III.1.4.9 Postulated Transportation Accident

This postulated accident assumes that a railroad car carrying irradiated fuel elements from the N Reactor storage basin to the 105-KE basin is derailed and the contents of three loaded casks are spilled onto the ground. The accident is assumed to occur halfway between the two areas, at a distance of four miles by road from the N Reactor site.

A fully loaded cask car carries three casks, each containing a normal full load of 42 fuel elements weighing 2,200 lb (1.0 metric ton). It is assumed that the derailment could tip all three of the casks onto the ground, causing the lid hasps to unfasten and spill coolant water and the elements on the ground. Other assumptions and calculation parameters are found in Appendix III.A.

TABLE III.19. Primary Coolant System Radionuclide Inventory

Radionuclide	Normal		Maximum	
	Conc (pci/l)	Inventory (Ci)	Conc (pci/l)	Inventory (Ci)
Kr-85m	0	0	4.28×10^8	3.24×10^2
Kr-85	0	0	3.83×10^7	2.90×10^1
Kr-87	7.87×10^6	5.9×10^0	5.83×10^7	4.41×10^1
Kr-88	1.5×10^7	1.1×10^1	8.60×10^7	6.51×10^1
I-131	6.2×10^5	4.7×10^{-1}	7.82×10^7	5.92×10^1
I-132	2.0×10^7	1.5×10^1	1.33×10^8	1.01×10^2
I-133	1.1×10^7	8.3×10^0	1.81×10^8	1.37×10^2
I-134	2.9×10^6	2.2×10^1	2.15×10^8	1.63×10^2
I-135	1.6×10^7	1.2×10^1	1.74×10^8	1.32×10^2
Xe-131m	0	0	5.40×10^5	4.09×10^{-1}
Xe-133m	0	0	2.47×10^7	1.87×10^1
Xe-133	1.5×10^6	1.1×10^0	1.72×10^8	1.30×10^2
Xe-135m	0	0	3.22×10^7	2.44×10^1
Xe-135	7.9×10^6	6.0×10^0	2.96×10^7	2.24×10^1

TABLE III.20. Fission Product Releases From
Large Pipe Break Accident

<u>Radionuclide</u>	<u>Release From Confinement (Curies)</u>		
	<u>Steam Vent</u>	<u>Filtered Release</u>	<u>Building Leakage</u>
Kr-85m	8.62×10^1	neg.*	neg.
Kr-85	7.71×10^0	neg.	neg.
Kr-87	1.17×10^1	neg.	neg.
Kr-88	1.73×10^1	neg.	neg.
I-131	1.57×10^1	8.17×10^{-2}	1.26×10^{-2}
I-132	2.69×10^1	3.28×10^{-2}	5.08×10^{-3}
I-133	3.64×10^1	1.41×10^{-1}	2.19×10^{-2}
I-134	4.34×10^1	2.25×10^{-2}	3.48×10^{-3}
I-135	3.51×10^1	8.51×10^{-2}	1.32×10^{-2}
Xe-131m	1.09×10^{-1}	neg.	neg.
Xe-133m	4.97×10^0	neg.	neg.
Xe-133	3.46×10^1	neg.	neg.
Xe-135m	6.49×10^0	neg.	neg.
Xe-135	5.96×10^0	neg.	neg.

* Negligible.

TABLE III.21. Population Commitment Resulting From the
Postulated Large Pipe Break Accident^(a)

<u>Population Location</u>	<u>Organ Dose in Rem</u>	
	<u>Whole Body</u>	<u>Thyroid</u>
Site Boundary	9.57×10^{-4}	3.7×10^{-1}
Low Population Zone	9.06×10^{-6}	5.05×10^{-3}

(a) See Appendix III.A for assumptions and dose calculation parameters.

Mitigating Conditions

No accident involving shipment of irradiated reactor fuel has ever occurred during the 30 yr of shipping operations at Hanford. The probability of this accident occurring is very low since detailed safety measures are provided for all irradiated fuel shipments.

Gasoline and diesel oil are delivered only infrequently when construction machinery is required onsite for special projects. The only road and rail intersection between the 100-N and 100-K areas is located just outside the 100-KE entrance. For these reasons, an accident along this route involving a fuel oil truck collision with a loaded cask car was considered incredible.

A criticality accident was not considered since studies show that 23,400 lb of fuel is required for the minimum critical mass.⁽²⁷⁾ Since only 6,600 lb is involved in this postulated accident, criticality is not possible.

Source Terms and Potential Environmental Impact

Using the assumptions outlined in the previous paragraphs, the source term for radionuclides released from three fuel casks to the environment are given in Table III.22.

Doses were calculated using the meteorology assumptions and pathway models described in Appendix III.A. A ground level release was assumed, and the dose to a maximum individual was computed assuming full-time occupancy at a distance of 5.5 miles northwest of the site. Calculated doses for the transportation accident would be an extremely small fraction of the dose standards specified in ERDAM-0524. Doses resulting from this accident are shown in Table III.23.

III.1.4.10 Postulated 105-KE Fuel Storage Facility Cask Drop

The maximum credible accident in the 105 KE storage facility consists of dropping a fuel transport cask loaded with irradiated fuel. It is assumed that:

TABLE III.22. Postulated Fuel Shipping Accident
Fission Product Source Terms

<u>Radionuclide</u>	<u>Initial Puff Release (Ci)</u>	<u>Oxidation After Breakage (Ci/hr)</u>	<u>Total Release (Ci)</u>
Kr-85m	0.	0.	0.
Kr-85	8.96×10^{-2}	8.96×10^{-2}	7.26×10^0
Kr-87	0.	0.	0.
Kr-88	0.	0.	0.
I-131	2.93×10^{-5}	2.93×10^{-5}	2.38×10^{-3}
I-132	2.22×10^{-13}	2.22×10^{-13}	1.80×10^{-11}
I-133	0.	0.	0.
I-134	0.	0.	0.
I-135	0.	0.	0.
Xe-131m	8.63×10^{-5}	8.63×10^{-5}	6.99×10^{-3}
Xe-133m	6.72×10^{-20}	6.72×10^{-20}	5.45×10^{-18}
Xe-133	1.94×10^{-7}	1.94×10^{-7}	1.57×10^{-5}
Xe-135m	0.	0.	0.
Xe-135	0.	0.	0.

TABLE III.23. Population Commitment Resulting From
the Postulated Fuel Shipping Accident (a)

<u>Population Location</u>	<u>Organ Dose in Rem</u>	
	<u>Whole Body</u>	<u>Thyroid</u>
Site Boundary	2.27×10^{-8}	5.18×10^{-6}
Low Population Zone	1.38×10^{-9}	2.52×10^{-7}

(a) See Appendix III.A for assumptions and dose calculation parameters.

- The cask would break and spill the 42 fuel elements into the open air.
- The peak temperature reached by the uncooled fuel would be 910 F,⁽²⁸⁾ which is well below fuel cladding melt temperature.

The spilling of a full transport cask of fuel presents no criticality problem since the amount of fuel in a cask is only 10% of the minimum hemispherical critical mass with optimum moderation and reflection.⁽²⁷⁾ Other assumptions and calculation parameters are found in Appendix III.A.

Mitigating Conditions

The transfer crane used to lift the loaded fuel transport cask has a rated capacity of 30 tons. A fully loaded cask weighs 24 tons. Strict operating and maintenance procedures assure the safest possible operation of this lifting equipment. Each operator is thoroughly trained in operating procedures in criticality, safety, and process standards.

A rigid inspection schedule of the crane assures its integrity. This schedule consists of:

- an inspection, prior to the day of use, covering limit switches, wire rope, load chains and hooks.
- a 30-day inspection and lubrication, including an operating load test and inspection of motor, brakes and hook engagement.
- a 90-day test, similar to the 30-day test, but including a more thorough inspection of components.
- annual magnaflux examination of all hooks.
- general biannual underwriters third party inspection.

Source Terms and Potential Environmental Impact

The source terms for a cask drop are given in Table III.24; they represent one-third of the values used for a transportation accident since only one cask, not three, is involved. All other assumptions are the same. Therefore, the calculated dose to the resident is as shown in Table III.25.

TABLE III.24. Postulated Fuel Cask Drop Accident
Fission Product Source Terms

<u>Radionuclide</u>	<u>Initial Puff Release (Ci)</u>	<u>Oxidation After Breakage (Ci/hr)</u>	<u>Total Release (Ci)</u>
Kr-85m	0.	0.	0.
Kr-85	2.99×10^{-2}	2.99×10^{-2}	2.42×10^0
Kr-87	0.	0.	0.
Kr-88	0.	0.	0.
I-131	9.78×10^{-6}	9.78×10^{-6}	7.92×10^{-4}
I-132	7.39×10^{-14}	7.39×10^{-14}	5.99×10^{-12}
I-133	0.	0.	0.
I-134	0.	0.	0.
I-135	0.	0.	0.
Xe-131m	2.88×10^{-5}	2.88×10^{-5}	2.33×10^{-3}
Xe-133m	2.24×10^{-20}	2.24×10^{-20}	1.82×10^{-18}
Xe-133	6.46×10^{-8}	6.46×10^{-8}	5.23×10^{-6}
Xe-135m	0.	0.	0.
Xe-135	0.	0.	0.

TABLE III.25. Population Commitment Resulting From the
Postulated Fuel Cask Drop Accident^(a)

<u>Population Location</u>	<u>Organ Dose in Rem</u>	
	<u>Whole Body</u>	<u>Thyroid</u>
Site Boundary	7.56×10^{-9}	1.73×10^{-6}
Low Population Zone	4.61×10^{-10}	8.41×10^{-8}

(a) See Appendix III.A for assumptions and dose calculation parameters.

III.2 FUELS FABRICATION FACILITIES

III.2.1 Radiological Impact

III.2.1.1 General

Uranium is the only radioactive material handled in the 300-Area Fuels Fabrication Facilities. Table III.26 gives the quantity of uranium released to the atmosphere from the Fuels Fabrication Facilities during 1975.

III.2.1.2 Effluent Releases

Although liquid effluents have been discharged to the ground in the 300 Area since operations began during World War II, no detectable increase in concentrations of radionuclides has been observed at the 300 Area forebay, which supplies drinking water for the 300 Area. The forebay is about 0.3 mile downstream of any potential entry of liquid effluents produced in the 300 Area. Samples obtained at the Richland Pumping Plant (the first downstream public water supply) have shown no detectable concentrations of uranium above natural river background levels.

TABLE III.26. Uranium Released to the Atmosphere
From 300-Area Fuels Fabrication
Facilities During 1978

<u>Nuclide</u>	(a)	<u>Release (Ci/yr)</u>
^{234}U		1.8×10^{-9}
^{235}U		2.3×10^{-7}
^{238}U		3.2×10^{-5}
Total		3.3×10^{-5}

(a) 0.95% enriched in ^{235}U .

Population Doses

A combination of wind velocity data at the Fast Flux Test Facility site and long-term stability data from the Hanford Meteorological Station were used in the dose calculations. Calculated maximum individual and population doses

resulting from inhalation of the uranium are given in Table III.27. The maximum individual residence is located directly across the Columbia River from the 300 Area, 1.24 miles (2,000 m) east of the Fuels Fabrication Facilities. In comparison with direct inhalation, the other pathways were insignificant (less than 1%) and are not listed in the table. Maximum individual doses were calculated assuming that the individual resides directly across the river 1.24 miles east of the 300 Area.

Assuming that 250,000 people reside within a 50-mile radius of the 300-Area facilities, the first-year whole body dose to the average individual was calculated to be 6.4×10^{-8} mrem.

TABLE III.27. Calculated Dose Commitment to the Public from 300-Area Fuels Fabrication Facilities During 1978

	Maximum Individual (mrem)		Population (man-rem)	
	First-Year Dose	50-Yr Dose Commitment	First-Year Dose	50-Year Dose Commitment
Whole Body	6.3×10^{-6}	1.1×10^{-5}	1.6×10^{-5}	2.8×10^{-5}
Lungs	4.8×10^{-4}	5.9×10^{-4}	1.4×10^{-3}	1.7×10^{-3}
Bone	5.3×10^{-5}	1.9×10^{-4}	1.3×10^{-4}	4.2×10^{-4}
GI	4.2×10^{-6}	4.2×10^{-6}	6.8×10^{-6}	6.9×10^{-6}

III.2.1.3 Solid Waste

There is no offsite environmental impact from radioactive solid waste because the material is packaged and transferred by truck to the 200-Area disposal site.

III.2.1.2 Occupational Doses to Employees

Fuels Fabrication Facilities personnel are also engaged in the operation of N Reactor facilities, and employee annual exposures indicated the total exposures from both operations. The data on these employees are presented in Section III.1.1.2 (Table III.8) and indicate that the annual exposures are approximately five times less than the ERDAM-0524 limit of 5 rem.

III.2.2 Nonradiological Impact

III.2.2.1 Effluent Releases

The chemical waste that enters the process sewer from the Fuels Fabrication Facilities is combined with waste from other 300-Area facilities in the process sewer before being discharged into one of the two process sewer trenches. (Shown in left foreground in aerial photo, Figure II.25.) Each trench is 1,500 ft long, and the surface area of each varies between one-third and three-fourth of an acre, depending upon the water depth in the trench (usually less than 4 ft deep). The inlet to the trenches is sampled routinely by Westinghouse Hanford Company and analyzed for pH and various chemicals. The inlet water pH ranged between 7.1 and 9.4 in CY-1978. The average concentration of chemicals that could partially originate from UNC facilities are within drinking water limits, as shown in Table III.28.

The water in the trenches seeps through the soil and forms a groundwater mound and eventually enters the Columbia River through several submerged riverbank springs. The shortest distance between the Columbia River and the process sewer trench is 1,000 ft, and the bottom of the trenches is about 15 ft above the average river level. A test well (about 50 ft deep) is located near the two trenches and is sampled monthly and analyzed by BNW for pH, fluoride ion, F^- ; nitrate ion, NO_3 ; uranium; copper; zinc; cadmium; lead; mercury and alpha, beta and gamma activity.

The neutralized chemical waste that is discharged into and retained by the concrete solar evaporation basin in 100-H Area has an average pH of 10.4. The high copper and chemical content of this waste prevents the growth of aquatic life, and since the trenches are alternately used and then dried, they are not suitable for most aquatic life. The concrete walls are topped by a 6-ft-high cyclone fence to prevent entry by large animals, and wires strung across the basin discourage entry by waterfowl.

Airborne Effluents

Atmospheric monitoring stations are located on the east side of the Columbia River with the closest station being about one mile east of the 300 Area

(Sullivan Ranch). The 1978 yearly average of NO_2 at the Sullivan Ranch was 0.006 ppm with a maximum 24-hr average of 0.025 ppm. These measurements include the NO_2 release from the 300-Area power plant, the predominant source. The standard for NO_2 concentration at such sites is 0.05 ppm⁽²⁹⁾ (the 1978 NO_2 releases from the chemical bay stack are discussed in Section II.3.2.6). The NO_2 releases from the 300 Area have no significant effect on man or the biota.

TABLE III.28. Average Inlet Chemical Concentration to 300-Area Process Sewer Trenches in 1978

<u>Material</u>	<u>Average Inlet Concentration (ppm)</u>	<u>Drinking Water Limits (ppm)^(a)</u>
Nitrate Ion	1.7	10
Fluoride Ion	0.51	1.8
Copper	0.46	1.0
Uranium	0.15	5 (as UO_2^{2+})

(a) U.S. Drinking Water Standards except for UO_2^{2+} which is from Industrial Pollution Control Handbook, H.F.²Lund, Ed., P. 4-34.

The Environmental Protection Agency specified a beryllium release rate of 10 g/day and an ambient concentration limit of 0.01 g/m^3 . The concentrations of beryllium in the 333 building exhaust stack have routinely been below the detectable level of 0.001 g/m^3 over the past five years.

III.2.2.2 Solid Waste

The nonradioactive solid waste from the Fuels Fabrication Facilities does not constitute an offsite environmental impact. See Section II.3.2.6 for a detailed discussion.

III.2.3 Natural Forces

Tornados

The Pacific Northwest is one of the areas of the country with the lowest frequency of tornados. On the average, the State of Washington experiences less than one tornado each year. Thirteen tornados have been recorded within a

100-mile radius of the Hanford Meteorological Station since 1916. These were of relatively short duration, reaching ground level only for brief periods and causing only slight damage. One small tornado (June 16, 1948) and two funnel clouds have been observed in 33 years (1945-1977) at Hanford. These were sighted along Rattlesnake Mountain.

Several studies have been made on the maximum wind speed and the probability of a tornado striking a specific location on the Hanford Reservation. These analyses conclude that the probability of a tornado hitting a particular facility onsite is six chances in 1 million during any given year or about one chance in 4,000 during a 40-year plant life. The maximum tornado wind speed for the Hanford site is estimated to be 175 mph.⁽³⁰⁾ Also see Section II.1.7.

Most 300-Area buildings were designed to withstand the winds and roof loadings expected in the Hanford area. The 333 building is designed to withstand a roof live loading of 40 lb/ft² and a wind load of 20 lb/ft² on all vertical projections. The latter corresponds to pressure resulting from wind speeds up to about 70 mph at an elevation of 30 ft.

Earthquakes

The Fuels Fabrication Facilities, which were built under the Uniform Building Code applicable at the time of construction, met the criteria for straight winds, which probably exceeded the Zone 1 seismic criteria applicable in 1949.

Some Fuels Fabrication Facilities in the 300 Area were built before the Uniform Building Code included seismic design provisions. Because Hanford was then designated a low-intensity minimum-occurrence earthquake zone, only small lateral earthquake design forces were considered. Other Fuels Fabrication Facilities were built according to the Uniform Building Code which contained seismic provisions, with Hanford designated as Zone 2, implying only moderate damage based on deductions from historical records.

Credible lateral wind forces were included in the design of both Buildings 333 and 313, based on the maximum straight lateral wind forces experienced in the

recorded history of the region. Natural forces beyond the limits of the Uniform Building Code were not considered. However, analytical studies suggest that at least some of the Fuels Fabrication Facilities may fail under a maximum credible (0.25 g) earthquake.

In the unlikely event of a severe Uniform Building Code earthquake in the 300 Area, maximum environmental effects would occur from a pipe break in the chemical transfer lines. It is expected that the chemical solutions in damaged piping and tanks would enter the process sewer trench or ground at the building site. Solutions would be neutralized to minimize impact. They would then eventually reach the Columbia River as water at a neutral pH range. However, considering the extremely low probability of a severe earthquake at the site, there appears to be very little risk to the public for the projected operating period.

Floods

The elevation of the 300-Area site averages about 400 ft MSL, providing adequate protection against Columbia River floods. The largest recent flood occurred in June 1948, with a flow of 693,000 cfs at Hanford compared to the estimated 740,000 cfs for the 1894 flood. It is not expected that a recurrence of the peak flood would inundate the 300-Area site, nor would the probable maximum flood level of 383 ft MSL flood this area. A seismic induced flood would not exceed the probable maximum flood. Finally, the 100 year flood and the 500 year flood would be less than the probable maximum flood.

III.2.4 Accident Impacts

Prevention of accidents and protection against the occurrence of postulated accidents at the Fuels Fabrication Facilities is provided through design and construction features and proper operation of the facilities. A formal quality assurance program is in place covering new construction, modifications, materials, procedures and operations of all the reactor systems.

This section discusses three postulated credible accidents including the estimated environmental effects of varying severity accidents up to and including a uranium-fines fire in a barrel.

In arriving at the types of postulated accidents to be presented in this environmental assessment, numerous conceivable accidents were studied, with emphasis on the conditions or circumstances most likely to result in an accident or abnormal event. In addition to the historical incidents that have occurred at N Reactor the following accidents were considered at the Fuels Fabrication Facilities:

- postulated small leak from the liquid chemical waste system
- postulated large leak from a bulk chemical storage tank
- postulated fire in a uranium fines barrel

III.2.4.1 Historical Incidents

The incidents listed in Table III.29 are representative of the types of accidents which have occurred at the Fuels Fabrication Facilities.

III.2.4.2 Postulated Small Leak from the Liquid Chemical Waste System

Nitric and sulfuric acid solutions are used in the manufacture of N Reactor fuel. As a result, some of the acid contains low concentrations of uranium. The used acid solutions are collected, neutralized and transported to a concrete-lined basin at 100-H Area where the liquids are evaporated to a solid salt cake. Small accidental spills of the acids could occur due to the failure of the processing tanks and/or associated piping or to operator error.

Although spilled acid could be discharged to the ground, the most likely discharge path would be to the 300-Area process sewer. All leaks from the processing tanks in Buildings 333 and 313 and in the transfer lines to and from these tanks would be collected in the process sewer. This sewer discharges to a trench located near the 300 Area (see Section III.2.2.1).

Uranium bearing acid solutions contain less than 116/gal of uranium, and the larger tanks are equipped with catch trays and leak detectors as described by Section II.3.2.7.

TABLE III.29. Historical Incidents at the
Fuels Fabrication Facilities

<u>Date</u>	<u>Description of Incident</u>	<u>Environment Consequences</u>
9-3-64 2-2-66	On September 3, 1964, an employee was removing the lid from a barrel of uranium chips near the Building 300 in the 300 Area when the chips ignited inside the barrel. In a similar event about two years later, about 50 lb of incompletely oxidized uranium located in the 303-L oxide burner and 40 lb of uranium on the oxide burner air filters ignited.	The 1964 uranium fire was contained by inverting a barrel over the burning uranium. No release of uranium to the environs was reported. The ground around many of the 300-Area buildings was surveyed after the incident in 1966. Contamination levels were reported as less than 1×10^{-6} Ci/cc. No uranium fires with the potential for environmental impact have occurred at the Fuels Fabrication Facilities since 1968. The 303-L oxide burner was retired from service in 1971.
12-30-70	On December 30, 1970, radio active waste material was reported outside of the 300-Area uranium burial trench radiation zone. An investigation of the problem revealed that several hundred peel test sample envelopes had been scattered northeast from the trench by high winds. The 4- x 8-in. paper envelopes are normally disposed of by placing them in sealed contaminated waste cartons, then transporting the cartons to the trench in a contaminated waste lugger. Wet weather and snow had destroyed the seal on the cartons and allowed the envelopes to be scattered by the wind. Activity levels on the envelopes were up to 30,000 counts/min.	The loose contaminated envelopes were picked up, returned to the burial trench and covered with soil. Ground contamination was limited to the area immediately adjacent to the burial ground. The 300-Area solid waste burial ground was covered and retired from service in 1972. All solid waste from the 300 Area is now transferred to the 200 areas for disposal.

Mitigating Conditions

The process tanks, pumps and transfer lines are constructed of inert materials that are compatible with the solutions handled. The transfer lines are routed between facilities in concrete trenches. Liquids that leak from the lines would drain into the trenches, then into the process sewer. The process tanks are also located such that any leaking liquid would be routed to the process sewer. Routine inspection and maintenance of all acid-bearing tanks and lines are conducted in the Fuels Fabrication Facilities to detect and rectify problems before a leak can occur. A pH monitor is provided in the process sewers from the Fuels Fabrication Facilities and will alarm if an acid or caustic leak has occurred. Acids or caustics that leak into the process sewer would then be neutralized by manually valving caustic or acid to the process sewer.

Hanford Engineering Development Laboratory routinely samples the process trench influent to assure that chemical concentrations are within administrative limits. PNL routinely samples test wells adjacent to the trench to detect migration of chemicals to the river and to evaluate environmental effects of these releases. To date, no significant adverse environmental effects have been identified as a result of routine operation of the trench.

Source Terms and Potential Environmental Impact

Leaking acid solutions would enter the process trench via the process sewers. After entering the trench, the acids would be neutralized and the resulting liquids would then percolate through the soil to the Columbia River. Retention of most of the chemicals in the process trench soil is assured through careful control of the pH of the effluent discharging to the trench. As a result of this pH control, no increase in the release of chemicals to the river would be observed.

Based on the very low level of uranium concentration in the acid solutions, no uranium would be released to the trenches or river. Consequently, no adverse environmental impact would result from a small acid spill to the process trench.

III.2.4.3 Postulated Large Leak from a Bulk Chemical Storage Tank

Nitric and sulfuric acid is stored in bulk above-ground storage tanks that have capacities up to 6,000 gal as described in Section II.3.2.3. Sodium hydroxide is also stored above ground in two 10,000-gal capacity tanks. These chemicals are used in the manufacturing of N Reactor fuel and are transferred to processing tanks as needed. The chemicals could accidentally be spilled should one of the tanks develop a severe leak. Spilled solutions would be caught by catch basins which drain into the process sewer and hence to the process trench located near the 300 Area.

Mitigating Conditions

Storage tanks and associated piping are constructed of inert materials that are compatible with the solutions stored in the tanks. Log books are kept for each bulk storage tank, and the volume in the tank is recorded after each addition or withdrawal. The log book records aid in determining how much solution was lost in the event of a leak and in establishing the appropriate corrective action.

Routine inspections and maintenance of all bulk storage tanks and lines are conducted in the Fuels Fabrication Facilities to detect and rectify problems before a leak can occur. Acids or caustics that leak to the process sewer would be neutralized by manually valving the appropriate chemicals to the process sewer trench. A pH monitor is provided in the process sewer from the Fuels Fabrication Facilities and will alarm if an acid or caustic leak occurs.

HEDL routinely samples the process trench influent to assure that chemical concentrations are within administrative limits. PNL routinely samples test wells adjacent to the trench to detect migration of chemicals to the river and to evaluate environmental effects of these releases.

Source Terms and Potential Environmental Impact

Spilled chemicals (acids or caustics) would enter the process trench via the process sewer. After entering the trench, the solutions would be neutralized, and the resulting liquids would then percolate through the soil to the Columbia

River. Retention of most of the chemicals in the process trench soil is assured through careful control of the pH of the effluent discharging to the trench. As a result of this pH control, no increase in the release of chemicals to the river would be observed. Thus, no significant adverse environmental impact would result from a large acid or caustic spill.

III.2.4.4 Postulated Fire in Uranium Fines Barrel

Uranium fines from the three cutoff saws, the three spray rinse stations, the air wash on the air exhauster for the cutoff saws and band saw and the two truck wash stations are collected in the trenches and sumps, which are cleaned out between enrichment campaigns. These fines are a mixture of uranium metal and oxides, Zircaloy 2 metal, zirconium oxides, copper oxides, silicon carbide grit, abrasive wheel matrix material and extrusion lubricant. The accumulated fines are drummed in 30-gal barrels according to weight, volume and enrichment limits as specified in DUN-5750, Nuclear Safety Specification 3.1.

Procedures specify that the fines be temporarily stored in the 30-gal drums for periods not greater than three months before being solidified for shipment to National Lead Company of Ohio for uranium recovery.

For purposes of this accident analysis, 419 lb of fines at 0.95% enrichment are assumed to be completely oxidized over a 3-hr period. Of the uranium in each barrel, 98.998% is ^{238}U , 0.045% is ^{236}U , 0.95% is ^{235}U , and 0.007% is ^{234}U .

Mitigating Conditions

Prior to 1968, experience with uranium fires suggested that only a very small fraction of the initial uranium mass escaped from the drums during a fire. The major mass remained inside the 30-gal drum containers.

Strict administrative controls affecting nuclear safety⁽³¹⁾ and operating procedure were introduced in 1968 to provide for spacing of temporary storage drums 44 in. from center to center on a curbed cement pad equipped with a drain; routine daily surveillance of water level in storage drums and outside

ambient temperature; water cover over fines to prevent rapid oxidation of uranium; use of a sprinkler system to keep outer barrel surfaces cool when ambient temperatures approach 100 F; and partially filled drums that are spaced to further reduce the probability of propagating a uranium fire.

Should a problem develop, equipment is available to restrict the oxidation rate and further reduce the potential for airborne release of the uranium oxide formed.

Source Term and Potential Environmental Impact

Notwithstanding precautionary measures, the conservative assumption is that in the unlikely event that a fire occurs in a barrel of uranium fines, a maximum of 0.10% of the uranium mass in the barrel could become airborne and the uranium would be completely oxidized within 3 hr.

This airborne release fraction falls within an envelope of past experience with uranium fires involving fines and other small uranium specimens.^(32,33,34,35) The resultant source terms are given in Table III.30.

TABLE III.30 Source Terms for Postulated Uranium Fire

<u>Nuclide</u>	<u>Percent</u>	<u>Curies Release</u>
²³⁸ U	99.998	6.26×10^{-2}
²³⁶ U	0.045	5.37×10^{-3}
²³⁵ U	0.950	3.85×10^{-3}
²³⁴ U	0.007	8.23×10^{-2}

Atmospheric dispersion of radionuclides released during the accident was described by a bivariate normal distribution model. Maximum individual dose calculations were performed assuming exposure at the plume centerline during moderately stable atmospheric conditions and a 3.3 fps wind speed. Population dose calculations were based on sector average atmospheric dispersion formulas using joint frequency distributions of wind velocity and atmospheric stability data from meteorological data obtained from the Fast Flux Test Facility site

and the Hanford Meteorological Station adjacent to the 200-W Area. Dose calculations were made for both external exposure from submersion in the passing clouds and internal exposure from inhalation.

External dose calculations were performed using the computer program KRONIC.⁽²²⁾ The program considered contributions from both beta and gamma radiations and calculated doses to internal organs in addition to the whole body.

The inhalation dose calculations were performed using the computer code DACRIN.⁽²³⁾ The material inhaled was considered insoluble when calculating organ doses.

Maximum individual doses were calculated assuming that the individual resides directly across the river 1.24 miles east of the 300 Area and possesses the habits attributed to the maximum individual described in Section III.1.1.1.

Intake and exposure to radioactive liquids were considered for this accident; however, compared with direct inhalation, the liquid pathways were insignificant and are not listed in Table III.31.

TABLE III.31. Calculated Doses for Postulated
300-Area Uranium Fire

	First-Year Dose		50-Yr Dose Commitment	
	Maximum Individual (rem)	Population (man-rem)	Maximum Individual (rem)	Population (man-rem)
External				
Whole body	7.1×10^{-11}	3.3×10^{-5}	7.1×10^{-11}	3.3×10^{-9}
Inhalation				
Whole Body	4.8×10^{-6}	1.7×10^{-4}	3.1×10^{-5}	1.1×10^{-3}
Lung	2.3×10^{-2}	8.0×10^{-1}	5.6×10^{-2}	2.0×10^0
Bone	4.5×10^{-5}	1.6×10^{-3}	5.1×10^{-4}	1.8×10^{-2}

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APPENDIX III.A

ANALYTICAL PARAMETERS USED TO

EVALUATE RADIOLOGICAL EFFECTS OF ACCIDENTS

APPENDIX III.A

ANALYTICAL PARAMETERS USED TO EVALUATE RADIOLOGICAL EFFECTS OF ACCIDENTS

1.0 Purpose

This appendix provides the assumptions, calculational models and values used to evaluate the radiological impacts that are common to the postulated accidents.

2.0 Data and Assumptions

2.1 Radioactive Sources

The data and assumptions used to estimate the radioactive sources from postulated accidents are:

1. For reactor pipe breaks;

- Power level, for both conservative and realistic cases, of 4080 MWt is based upon a 4000 MWt plus a 2% uncertainty allowance for calorimetric measurements.
- Fuel burnup is based upon operation at 200 kW/ft for 210 days.
- The reactor coolant activity before the accident assumes operation with maximized fuel cladding failures involving 1300 grams of uranium in the primary loop.

2. For spent fuel events;

- Fuel burnup is based upon operation of 200 kW/ft for 210 days.
- Fuel has been stored for 150 days.
- 1500 g of uranium/cask will become oxidized within 80 hours (accident life).
- Three broken elements per cask are broken at a 45 cleavage angle.

2.2 Radionuclide Release Assumptions

The data and assumptions used to estimate the amount of radionuclides released to the environment are:

1. For pipe break events;

- The confinement leak rate is 5 percent of the filter flow at the same pressure differential that exists between the confinement and outside atmospheres. Adsorption and filtration efficiencies of the confinement filters are 99.95% for particulates and 95% for elemental halogens.
- The confinement spray parameters are calculated via a computer model based upon fog spray data, particle sizes and compartment conditions that vary during the course of the postulated accidents.
- The confinement volume, for both realistic and conservative cases, is $3.08 \times 10^6 \text{ ft}^3$.

2. For spent fuel events;

- 100% of the noble gases and 50% of the iodines will be released from the oxidized fuel.

2.3 Radionuclide Dispersion Data

The data and assumptions used to estimate the environmental dispersion of released radionuclides are:

1. The distance to the site boundary and to the low population zone (LPZ) is 5.5 miles (8900 meters) and 20 miles (32,000 meters), respectively.
2. The X/Q values selected are based upon curves in Regulatory Guide 1.4 for ground level releases.

For all pipe break and spent fuel accidents, the X/Q LPZ boundary values were selected as those which could be exceeded 5% of the time as based upon 12 months of onsite meteorological data. The X/Q values for the site boundary were based upon wind speeds ranging from 1 m/sec to 3 m/sec depending upon the conditions for the individual accident analyzed. Thus, the X/Q values are:

<u>Time Period</u>	<u>1 m/sec</u>	<u>2 m/sec</u>	<u>3 m/sec</u>	<u>LPZ Boundary</u>
0-8 hr	2.6×10^{-5}	1.3×10^{-5}	8.67×10^{-6}	2.06×10^{-6}
8-24 hr	5.2×10^{-6}	2.6×10^{-6}	1.73×10^{-6}	6.15×10^{-7}
24-96 hr	3.37×10^{-6}	2.6×10^{-6}	1.73×10^{-6}	2.70×10^{-7}

2.4 Dose Calculations and Data

1. The selected X/Q values were based upon the wind speed that would result in the highest dose at the site boundary.
2. For whole body dose calculations the receptor was assumed to be immersed in a semi-infinite plume of uniform concentration. Specific equations are:

$$D_{\beta} = 0.23 \cdot X/Q \cdot \sum_i Q_i \cdot \bar{E}_i$$

$$D_{\gamma} = 0.25 \cdot X/Q \cdot \sum_i Q_i \cdot \bar{E}_i$$

where:

D_{β} is the whole body beta dose (rem)

D_{γ} is the whole body gamma dose (rem)

X/Q is the atmospheric dispersion parameter (sec/m^3)

Q_i is the quantity of isotope i released during the interval of interest (curies)

$\bar{E}_i$ is the decay energy from isotope i (MeV/disintegration)

3. The thyroid inhalation doses were calculated using the following expression during each period of interest.

$$D_t = X/Q \cdot B \cdot \sum_i Q_i \cdot DCF_i$$

where:

D_t is the thyroid inhalation dose

X/Q is the atmospheric dispersion parameter (sec/m^3)

B is the breathing rate (m^3/sec)

Q_i is the quantity of iodine isotope i released during the interval (curies)

DCF_i is the dose conversion factor for iodine isotope i (rem/curie inhaled)

4. See Table III.A-1 for the dose conversion assumptions.
5. The resultant doses are discussed in individual accident analyses.

TABLE III.A-1
PHYSICAL DATA FOR RADIONUCLIDES

Radionuclide	Decay Constant ^(a) (sec ⁻¹)	Gamma Energy ^(a) (MeV/Dis)	Beta Energy ^(a) (MeV/Dis)	Dose Conversion ^(b) Factor (Rem/Ci)
I-131	9.96×10^{-7}	0.375	0.209	1.48×10^6
I-132	8.02×10^{-5}	2.290	0.421	5.35×10^4
I-133	9.25×10^{-6}	0.636	0.403	4.00×10^5
I-134	2.20×10^{-4}	2.508	0.558	2.50×10^4
I-135	2.89×10^{-5}	1.457	0.475	1.24×10^5
Kr-85m	4.41×10^{-5}	0.156	0.101	_____
Kr-85	2.14×10^{-9}	0.0021	0.228	_____
Kr-87	1.48×10^{-4}	0.856	1.014	_____
Kr-88	6.95×10^{-5}	2.000	0.307	_____
Xe-131m	6.68×10^{-7}	0.022	0.135	_____
Xe-133m	3.49×10^{-6}	0.0239	0.191	_____
Xe-133	1.52×10^{-6}	0.0728	0.126	_____
Xe-135m	7.40×10^{-4}	0.440	0.090	_____
Xe-135	2.11×10^{-5}	0.248	0.303	_____

BREATHING RATES

<u>Time Period (hrs)</u>	<u>Breathing Rate (m³/sec)^(c)</u>
0-8	3.47×10^{-4}
8-24	1.75×10^{-4}
24-720	2.32×10^{-4}

- (a) Meek, M.E. and R. S. Gilbert, Summary of Gamma and Beta Energy and Intensity Data, NEDO-12037, 1970.
- (b) TID-14844.
- (c) Regulatory Guide 1.4.

IV. UNAVOIDABLE ADVERSE EFFECTS--N REACTOR AND FUELS FABRICATION FACILITIES

IV.1 GENERAL

About 3% of the Hanford Reservation is used for structures and waste disposal sites. The N Reactor and Fuels Fabrication operations discussed here involve only a relatively small portion of this land area. This land commitment does not constitute a significant adverse effect on the ecological community. Petroleum products, uranium and other small quantities of nonrenewable resources are consumed during the operation of N Reactor and the Fuels Fabrication Facilities. Quantities consumed are negligible in comparison with the overall national commitment of these resources.

IV.2 N REACTOR

The offsite radiological impact of N Reactor operations at Hanford during CY-1978 is discussed in Section III. Releases of radioactive materials to the environs resulted in an average per capita dose rate during CY-1978 of 0.006 mrem/yr. The whole body dose commitment for 236,000 people living within a 50-mile radius was 1.5 man-rem during 1978, or approximately 0.006% of the dose received by the population from naturally occurring radiation in the environment. The primary contributors to offsite doses from N Reactor operation are ^{41}Ar and ^{135}Xe , both relatively short-lived noble gases released from the reactor building. Upon shutdown of N Reactor, the release of these gases would cease and an immediate reduction in the annual offsite dose commitment of approximately 0.6 man-rem would be achieved. The discharge of radioactive liquids to the ground at 100-N Area would also cease. Population doses due to liquid releases would decrease from the current 0.02 man-rem per year as radioactive materials already present in the 100-N Area Facilities and in river sediments decay.

The practice of using the soil column beneath the 1301-N Facility at the N Reactor for removal of radionuclides from liquid effluents has resulted in the buildup of long-lived radionuclides in the soil. Continued use of this system for cleanup of liquid waste from N Reactor will increase the total quantity of radionuclides in the soil but will not alter its final disposition or surveillance requirements.

N Reactor operation also results in the discharge of relatively minor quantities of chemicals to the environment. Airborne effluents from 100-N Area are primarily combustion products from burning 5 million gallons of fuel oil in the plant service boilers. Spent chemicals from the water treatment process discharged to ground are primarily 1.7 million pounds of sulfuric acid and 1 million pounds of sodium hydroxide annually. The release of these chemicals produces no observable adverse environmental effects.

At the minimum river flow rate of 36,000 cfs, a routine release rate of 500 MW of heat via the 102-in. line is calculated to increase the average river temperature by 0.21 F. The average river temperature increase is calculated to be approximately 0.06 F at the average annual river flow rate of 120,000 cfs, and 0.12 F at the seven-day one-in-ten year low flow of 65,000 cfs. The lack of significant impact due to heating of the Hanford reach of the river is discussed in Section III.1.3.2.3.1 of ERDA-1538. The physical and ecological environment in the vicinity of the 100-N Area thermal discharges has been characterized. It is concluded, in ERDA 1538, that the discharge of thermal energy to the river results in little or no impact.

The principal effects of the coolant water intake system at the 100 N Area are impingement of some fish on the intake screens, entrapment or entrainment of some fish in the intake structure and destruction of drift organisms passing through the water intake systems into the condenser coolant water system. Three-tenths of 1 percent of the Columbia River chinook salmon fry passing the 100-N intake annually are killed as a result of drawing water from the Columbia River into the intake, the fry lost are equivalent to the number of fry resulting from two spawning pairs, and is not significant when considering the fishery as a whole. Previous studies reported in ERDA-1538 and current study results indicate that no significant effects on the ecological system result from cooling water withdrawal. The total use of Columbia River water by the entire Hanford operations, including the cooling water withdrawn for the N Reactor, is 500 cfs or only 0.4% of the average flow rate.

IV.3 FUELS FABRICATION FACILITIES

The Fuels Fabrication Facilities released 3.3×10^{-5} Ci of uranium to the atmosphere in CY-1978. This amount is the estimated annual release of uranium from these facilities for the duration of N Reactor operations. Upon shutdown of N Reactor, uranium releases from fuels fabrication operations would terminate. These releases contribute only 6.4×10^{-8} mrem first-year whole body dose to the average individual.

Most of the chemical effluents generated in the 300 Area are disposed of in a solar evaporation basin in the 100-H Area. A small amount of chemical effluent is discharged to the 300 Area process sewer trenches. Airborne effluents from the 300 Area consist of chemical vapors and fumes from process operations. These releases are discussed and quantified in Section II.3.2.6 and their impact is discussed in Section III.2.2. Releases of these chemicals to the atmosphere and those reaching the Columbia River produce no observable adverse environmental effects. Both release rates and ambient measurements are well within state standards.

V. ALTERNATIVES--N REACTOR AND FUELS FABRICATION FACILITIES

V.1 N REACTOR

The only alternatives to the current N Reactor operation programs considered are to shut down N Reactor or change the percent of ^{240}Pu in the product.

V.1.1 Alternative 1 - Shutdown of N Reactor, 1983 or 1990

An alternative to continued operation of N Reactor is to shut down the reactor. As stated earlier, the primary role of the reactor is production of plutonium and other materials for national defense and research and development activities. The need for this material is beyond the scope of this environmental assessment. This section updates the analysis in ERDA-1538 of the environmental impact of shutting down N Reactor.

Table V.1 provides a comparison of the changes resulting from the shutdown of N Reactor. The changes are expressed in terms of annual impact. The table compares the changes based upon the present mode of operation, operation with reinjection of the fuel failure detection system bleed and operation with recirculation of the spent fuel storage basin cooling water.

The shutdown of N Reactor would not substantially change the inventory of radionuclides currently stored in the 1301-N Facility. The quantity of long half-life nuclides not added to the 1301-N Facility is only a small portion of its current inventory of 6,350 Ci (1978). Most of these radionuclides are near the floor of the 1301-N crib and trench. The longer-lived constituents are 4,050 Ci of ^{60}Co , 1,700 Ci of ^{137}Cs and 600 Ci of ^{90}Sr . The 200 Ci/yr of tritium not added to the groundwater is only a few thousandths of one percent of the current groundwater tritium inventory. The 40,000 ft³/yr of waste not generated is about 0.4% of the solid waste already stored. The 1.5 man-rem/yr of radiation dose not received is about 0.006% of the radiation dose received from natural background each year.

TABLE V.1. Impacts That Will Not Occur Following N Reactor Shutdown

Impact(a)	Shutdown from Present Mode Of Operation	Shutdown from Operation With Fuel Cladding Failure Detection Monitor System Bleed Reinjection	Shutdown from Operation With Fuel Storage Basin Coolant Recirculation
When 6% of the plutonium yield is ^{240}Pu , uranium fuel which would not be irradiated (tons/yr)	717	717	717
When 12% of the plutonium yield is ^{240}Pu , uranium fuel which would not be irradiated (tons/yr)	315	315	315
When 6% of the plutonium yield is ^{240}Pu , Pu not produced (kg/yr)	700	700	700
When 12% of the plutonium yield is ^{240}Pu , Pu not produced (kg/yr)	615	615	615
Electricity not produced (kWhr/yr)	4.5 billion	4.5 billion	4.5 billion
^{90}Sr and ^{137}Cs not added to 1301-N soil (Ci/yr)	330	330	18.9
^{60}Co not added to 1301-N soil (Ci/yr)	370	284	284
Total activity not added to 1301-N soil (Ci/yr)	5,200	4,700	3,800
Total long-lived activity not released to Columbia River (Ci/yr)	1.7	1.5	1.4
Tritium not sent to groundwater (Ci/yr)	130	130	130
Solid waste not generated (ft ³ /yr)	40,000	40,000	40,000
^{41}Ar not released to atmosphere (Ci/yr)	54,900	54,900	54,900
Population dose not committed (man-rem/yr)	0.77	0.77	0.75

(a) Based on a normal operating year.

Note: Purex would be shut down when N Reactor fuel was processed.

One to two fewer double-wall tanks would be required through 1990.

V.1.2 Alternative 2 - Change Percent of Plutonium-240 Production

N Reactor is currently operated in such a manner that ^{239}Pu , the fissile isotope used in the defense program, is produced with a ^{240}Pu concentration of 12% (i.e., number of ^{240}Pu atoms/number of Pu atoms = 0.12). As an alternative to the present operating mode, the reactor could be operated so that the ^{240}Pu impurity level would be lower. This alternate mode involves limiting the burnup of the reactor fuel so that the impurity, ^{240}Pu , which builds up in the fuel during irradiation, is reduced to a maximum concentration of 6%.

TABLE V.2. N Reactor Production Alternatives

Plutonium Production Mode	Operating Days	Electrical Output	Refueling Days	Summer Maintenance Days
6%	252	4.5 billion kWhr	43	70
12%	250	4.5 billion kWhr	27	88

V.1.2.1 Resource Consumption

Operation of N Reactor in the 6% production mode would require the irradiation of about 2.3 times more uranium each year in order to produce an amount of ^{239}Pu equal to that currently generated in the 12% mode. Because of the increased shutdown time for the refueling and the requirement for comparable electrical power generation, the time allowed for the annual summer maintenance outage would be reduced. The shortened maintenance outage period would result in an increased number of maintenance personnel that would be required to accomplish an equivalent amount of maintenance work.

The increased shutdown-startup frequency resulting from the 6% operating mode would necessitate increased usage of fuel oil and process chemicals. Fuel oil consumption increases would result from the higher boiler operating levels required during each additional reactor startup. The increased consumption of

demineralized water during the more frequent refueling and steam generator blowdown operations would increase the demand for each of the water treatment chemicals (listed in Table VIII.1). Based upon an evaluation of operational aspects of the 6% production mode, it is estimated that fuel oil consumption would increase 6% and water treatment chemical usage would increase 20%.

V.1.2.2 Waste Generation

~~Routine releases~~ of radioactive contamination to the Columbia River via N Reactor liquid discharge streams would not be significantly affected by operation in the 6% production mode. Fuel element cladding failures, which contribute to radioactivity in liquid effluents, would be expected to increase as a result of the greater quantity of fuel handled in the reactor. However, because of the decreased irradiation time per fuel element, radioactivity inventories in the elements would be reduced so that it would be unlikely that significant changes in radioactive liquid discharges would occur as a result of fuel cladding failures.

The release of radioactive airborne effluents from N Reactor is essentially dependent on the megawatt-days of reactor operation. Operation in the 6% ~~production mode~~ would not result in significant changes in airborne releases.

Based on the assumption that irradiated fuel reprocessing facilities in the 200 area will be in operation by 1984, irradiated fuel storage facilities at 100-N Area and 100-K Area would be sufficient to accommodate the increased fuel throughput resulting from 6% operation. The discharge of fuel element spacers used to position fuel elements within the reactor could be expected to double in the 6% operating mode, requiring essentially double the burial or storage capacity currently required. Nevertheless; because of the reduced in-reactor residence time per spacer, the total activity buried annually would remain essentially the same.

V.1.2.3 Radiological Impact of Alternative 2

A change in N Reactor operation from the 12% production mode to the 6% mode would not significantly alter the radiation dose received by the population surrounding N Reactor as a result of N Reactor discharges.

V.2 FUELS FABRICATION FACILITIES

Alternatives related to the operation of the Fuels Fabrication Facilities follow directly from those considered for N Reactor, viz.:

- (1) Continue producing fuel in support of N Reactor operation;
- (2) Discontinue fuel production if N Reactor stops operating.

Based on the discussion in Section III-2, no observable effects on the environment are expected from the continued operation of the Fuels Fabrication Facilities. If the alternative to shut down N Reactor were adopted, no fuel for the reactor would be fabricated and there is no anticipated detrimental environmental effect in shutting down the fabrication plant.

V.3 DISPOSTION OF N REACTOR FUEL

Fuel irradiated at N Reactor is currently being stored in facilities at 100-N Area and 100-K Area. There are three potential alternatives for disposition of N Reactor fuel; namely:

- (1) Continued storage and possible canning of spent fuel elements;
- (2) Processing in PUREX to recover plutonium for weapons or reactor research and development, and uranium and neptunium for use in Defense Programs; and
- (3) Processing at the alternate site, the Savannah River Plant.

Changes from the present storage option will be fully addressed in subsequent environmental impact studies.