December 17, 1982

cp-1

Distribution
Docket File
NRC PDR
Local PDR
ORB#2 Rdg
D. Eisenhut
S. Norris
H. Nicolaras
OELD
SECY
L. J. HARMON 2
T. Barnhart 4
L. Schneider
D. Brinkman
ACRS 10
OPA Clare Miles
R. Diggs
NSIC
Gray
ASLAB Xtra - 5

see rpt

Docket No. 50-263

Mr. D. M. Musolf
Nuclear Support Services Department
Northern States Power Company
414 Nicollet Mall - Eighth Floor
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications in response to your May 1, 1979 application, revised by letter dated July 23, 1982 and subsequent discussions between the NRC staff and your staff. Changes to the proposed Technical Specification were discussed with, and agreed to, by your staff.

This amendment authorizes changes to the Technical Specifications 1) to implement the requirements of Appendix I of 10 CFR Part 50, 2) to establish new limiting conditions for operation for the quarterly and annual average release rates, and 3) to revise environmental monitoring programs to assure conformance with the Commission's regulations.

As part of the July 23, 1982 letter, you submitted copies of the Offsite Dose Calculation Manual and the Process Control Program. We have reviewed the Offsite Dose Calculation Manual and have found it acceptable. The Process Control Program is presently under review.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

8301040065 821217
PDR ADOCK 05000263
P PDR

Enclosures:
1. Amendment No. to DPR-22
2. Safety Evaluation
3. Notice of Issuance

* with comments

cc: w/enclosures
See next page

Amendment + FR Notice

	DL:ORB#2	DL:ORB#2	DL:ORB#2	DSI:RAB			DL:ORB#4
OFFICE	S.Norris	H.Nicolaras	D.Vassallo	Conget	Willis	W.Houston	Wagner
SURNAME	11/15/82	11/15/82	11/15/82	11/18/82	11/ /82	11/12/82	12/ 6/82
DATE						DL:ORB G.Lainas 12/13/82	OELD R.Bachmann 12/14/82

Mr. D. M. Musolf
Northern States Power Company

cc:

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and
Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Box 1200
Monticello, Minnesota 55362

Plant Manager
Monticello Nuclear Generating Plant
Northern States Power Company
Monticello, Minnesota 55362

Russell J. Hatling, Chairman
Minnesota Environmental Control
Citizens Association (MECCA)
Energy Task Force
144 Melbourne Avenue, S. E.
Minneapolis, Minnesota 55414

Ms. Terry Hoffman
Executive Director
Minnesota Pollution Control Agency
1935 W. County Road B2
Roseville, Minnesota 55113

Mr. Steve Gadler
2120 Carter Avenue
St. Paul, Minnesota 55108

Commissioner of Health
Minnesota Department of Health
717 Delaware Street, S.E.
Minneapolis, Minnesota 55440

Mr. D. S. Douglas, Auditor
Wright County Board of Commissioners
Buffalo, Minnesota 55313

U.S. Environmental Protection Agency
Region V Office
Regional Radiation Representative
230 South Dearborn Street
Chicago, Illinois 60604

James G. Keppler
Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 15
License No. DPR-22

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. This application for amendment by Northern States Power Company (the licensee) dated May 1, 1979 and revised by letter dated July 23, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Facility Operating License No. DPR-22 is hereby amended to read as follows:

   2. Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 15 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8301040070 821217
PDR ADOCK 05000263
P PDR

3. This license amendment is effective on January 1, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 17, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 15

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

1. Replace the following pages of Appendix A Technical Specifications, with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of changes.

REMOVE	INSERT
i	i
iii	iii
iv	iv
v	v
vi	vi
-	vii
5	5
-	5a
46	46
47	47
48	48
48a	-
62	62
68	68
124	124
192	192
193	193
194	194
195	195
196	196
197	197
198	198
-	198a to 198z
-	229g to 229s
239	239
242	242
244	244
246	246
-	246a to 246c
249	249
-	249a
250	250
251	251
-	251a
252	252
253	253
-	253a
254	254

2. Delete Appendix B in its entirety.

TABLE OF CONTENTS

		Page
1.0	DEFINITIONS	1
2.0	SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS	6
	2.1 and 2.3 Fuel Cladding Integrity	6
	2.1 Bases	10
	2.3 Bases	14
	2.2 and 2.4 Reactor Coolant System	21
	2.2 Bases	22
	2.4 Bases	24
3.0	LIMITING CONDITIONS FOR OPERATION AND 4.0 SURVEILLANCE REQUIREMENTS	26
	3.1 and 4.1 Reactor Protection System	26
	3.1 Bases	35
	4.1 Bases	41
	3.2 and 4.2 Protective Instrumentation	45
	A. Primary Containment Isolation Functions	45
	B. Emergency Core Cooling Subsystems Actuation	46
	C. Control Rod Block Actuation	46
	D. Deleted	
	E. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation	47
	F. Recirculation Pump Trip Initiation	48
	G. Safeguards Bus Voltage Protection	48
	3.2 Bases	64
	4.2 Bases	72
	3.3 and 4.3 Control Rod System	76
	A. Reactivity Limitations	76
	B. Control Rod Withdrawal	77
	C. Scram Insertion Times	81
	D. Control Rod Accumulators	82
	E. Reactivity Anomalies	83
	F. Scram Discharge Volume	83
	G. Required Action	83A
	3.3 and 4.3 Bases	84

Amendment No. ~~10~~ 15

	Page
3.8 and 4.8 Radioactive Effluents	192
A. Liquid Effluents	192
B. Gaseous Effluents	197
C. Solid Radioactive Waste	198e
D. Dose from All Uranium Fuel Cycle Sources	198f
3.8 and 4.8 Bases	198u
3.9 and 4.9 Auxiliary Electrical Systems	199
A. Operational Requirements for Startup	199
B. Operational Requirements for Continued Operation	200
1. Transmission Lines	200
2. Reserve Transformers	201
3. Standby Diesel Generators	201
4. Station Battery System	202
3.9 Bases	204
4.9 Bases	205
3.10 and 4.10 Refueling	206
A. Refueling Interlocks	206
B. Core Monitoring	207
C. Fuel Storage Pool Water Level	207
D. Movement of Fuel	207
E. Extended Core and Control Rod Drive Maintenance	208
3.10 and 4.10 Bases	209
3.11 and 4.11 Reactor Fuel Assemblies	211
A. Average Planar Linear Heat Generation Rate	211
B. Linear Heat Generation Rate	212
C. Minimum Critical Power Ratio	213
3.11 Bases	215
4.11 Bases	217
3.12 and 4.12 Sealed Source Contamination	219
A. Contamination	219
B. Records	221
3.12 and 4.12 Bases	222

	Page
3.13 and 4.13 Fire Detection and Protection Systems	223
A. Fire Detection Instrumentation	223
B. Fire Suppression Water System	224
C. Hose Stations	226
D. Yard Hydrant Hose Houses	227
E. Sprinkler Systems	227a
F. Halon Systems	227b
G. Penetration Fire Barriers	227b
3.13 Bases	228
4.13 Bases	229
3.14 and 4.14 Accident Monitoring Instrumentation	229a
3.14 and 4.14 Bases	229d
3.15 and 4.15 Inservice Inspection and Testing	229e
3.15 and 4.15 Bases	229f
3.16 and 4.16 Radiation Environmental Monitoring Program	229g
A. Sampling and Analysis	229g
B. Land Use Census	229i
C. Interlaboratory Comparison	229j
4.16 Bases	229s
5.0 DESIGN FEATURES	230
5.1 Site	230
5.2 Reactor	230
5.3 Reactor Vessel	230
5.4 Containment	230
5.5 Fuel Storage	231
5.6 Seismic Designs	231
6.0 ADMINISTRATIVE CONTROLS	232
6.1 Organization	232
6.2 Review and Audit	237
6.3 Special Inspection and Audits	243
6.4 Action to be taken if a Safety Limit is Exceeded	243
6.5 Plant Operating Procedures	244
6.6 Plant Operating Records	246
6.7 Reporting Requirements	248
6.8 Environmental Qualification	254

LIST OF FIGURES

Figure No.		Page No.
4.1.1	'M' Factor - Graphical Aid in the Selection of an Adequate Interval Between Tests	44
4.2.1	System Unavailability	75
3.4.1	Sodium Pentaborate Solution Volume-Concentration Requirements	97
3.4.2	Sodium Pentaborate Solution Temperature Requirements	98
3.6.1	Change in Charpy V Transition Temperature versus Neutron Exposure	133
3.6.2	Minimum Temperature versus Pressure for Pressure Tests	134
3.6.3	Minimum Temperature versus Pressure for Mechanical Heatup or Cooldown Following Nuclear Shutdown	135
3.6.4	Minimum Temperature versus Pressure for Core Operation	136
4.6.2	Chloride Stress Corrosion Test Results @ 500 F	137
3.7.1	Differential Pressure Decay Between the Drywell and Wetwell	191
3.8.1	Monticello Nuclear Generating Plant Site Boundary for Liquid Effluents	198g
3.8.2	Monticello Nuclear Generating Plant Site Boundary for Gaseous Effluents	198h
3.11.3	K_f Factor versus Percent of Rated Core Flow	218
6.1.1	NSP Corporate Organizational Relationship to On-Site Operating Organization	234
6.1.2	Monticello Nuclear Generating Plant Functional Organization for On-Site Operating Group	235

Amendment No. 15

LIST OF TABLES

Table No.		Page
3.1.1	Reactor Protection System (Scram) Instrument Requirements	28
4.1.1	Scram Instrument Functional Tests - Minimum Functional Test Frequencies for Safety Instrumentation and Control Circuits	32
4.1.2	Scram Instrument Calibration - Minimum Calibration Frequencies for Reactor Protection Instrument Channels	34
3.2.1	Instrumentation that Initiates Primary Containment Isolation Functions	49
3.2.2	Instrumentation that Initiates Emergency Core Cooling Systems	52
3.2.3	Instrumentation that Initiates Rod Block	57
3.2.4	Instrumentation that Initiates Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation	59
3.2.5	Instrumentation that Initiates a Recirculation Pump Trip	60
3.2.6	Instrumentation for Safeguards Bus Degraded Voltage and Loss of Voltage Protection	60a
3.2.7	Trip Functions and Deviations	70
4.2.1	Minimum Test and Calibration Frequency for Core Cooling, Rod Block and Isolation Instrumentation	61
3.6.1	Safety Related Snubbers	131
3.7.1	Primary Containment Isolation	172
3.8.1	Radioactive Liquid Effluent Monitoring Instrumentation	189i
3.8.2	Radioactive Gaseous Effluent Monitoring Instrumentation	198k
4.8.1	Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements	198m
4.8.2	Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements	198n
4.8.3	Radioactive Liquid Waste Sampling and Analysis Program	198p
4.8.4	Radioactive Gaseous Waste Sampling and Analysis Program	198s
3.11.1	Maximum Average Planar Linear Heat Generation Rate vs Exposure	214
3.13.1	Safety Related Fire Detection Instruments	227c

3.14.1	Instrumentation for Accident Monitoring	229b
4.14.1	Minimum Test and Calibration Frequency for Accident Monitoring Instrumentation	229c
4.16.1	Radiation Environmental Monitoring Program (REMP) Sample Collection and Analysis	229k
4.16.2	REMP - Maximum Values for the Lower Limits of Detection	229p
4.16.3	REMP - Reporting Levels for Radioactivity Concentrations in Environmental Samples	229r
6.1.1	Minimum Shift Crew Composition	236

Y. Shutdown - The reactor is in a shut condition when the reactor mode switch is in th shutdown mode position and no core rations are being performed. In this condition reactor scram is initiated and a rod block is inserted directly from the mode switch. scram can be reset after a short time delay.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

Z. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

AA. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling, also referred to as partial nucleate boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

AB. Pressure Boundary Leakage - Pressure boundary leakage shall be leakage through a non-isolable fault in the reactor coolant system pressure boundary.

AC. Identified Leakage - Identified leakage shall be:

1) Reactor coolant leakage into drywell collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

2) Reactor coolant leakage into the drywell atmosphere from sources which are specifically located and known not to be Pressure Boundary Leakage or which do not significantly impair the methods used to detect reactor coolant leakage.

AD. Unidentified Leakage - Unidentified leakage shall be all reactor coolant leakage which is not Identified Leakage.

AE. Process Control Program (PCP) - The PCP is the manual containing the current formula, sampling, analysis, test and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10CFR20, 10CFR71, and Federal and State regulations and other requirements governing the disposal of radioactive wastes.

AF. Solidification - Solidification is the conversion of wet radioactive wastes into a form that meets shipping and burial ground requirements.

AG. Offsite Dose Calculation Manual (ODCM) - The ODCM is the manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive liquid and gaseous effluents, in the calculation of liquid and gaseous effluent monitoring instrumentation alarm and/or trip setpoints, and in the conduct of environmental radiological environmental monitoring.

AH. Source Check - A Source Check is the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

AI. Purging - Purging is the controlled process of discharging air or gas from a confinement to maintain temperature pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

AJ. Venting - Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required.

AK. Dose Equivalent I-131 - Dose Equivalent I-131 is the concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or in NRC Regulatory Guide 1.109, Rev 1, October, 1977.

AL. Offgas Treatment System - The Offgas Treatment System is the system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

AM. Members of the Public - Means all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

AN. Site Boundary - Means a line within which the land is owned, leased, or otherwise controlled by the licensee. The site boundary for liquid releases of radioactive material is defined in Figure 3.8.1. The site boundary for gaseous releases of radioactive material is defined in Figure 3.8.2.

AO. Unrestricted Areas - Means any area at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial, institutional and/or recreational purposes.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
B. Emergency Core Cooling Subsystems Actuation When irradiated fuel is in the reactor vessel and the reactor water temperature is above 212°F, the limiting conditions for operation for the instrumentation which initiates the emergency core cooling subsystems are given in Table 3.2.2. C. Control Rod Block Actuation The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.3. D. (Deleted)	

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
E. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation 1. a. Except as specified in 3.2.E.1.b below, four radiation monitors shall be operable at all times. b. One of the two reactor building vent isolation monitors and one of the two radiation monitors on the refueling floor may be inoperable provided inoperable radiation monitor instrument channels are tripped. Upon loss of both channels, the Reactor Building ventilation system shall be isolated and the standby gas treatment system operated until repairs are complete. 2. The radiation monitors shall be set to trip as follows: (a) reactor building vent - refer to Specification 4.8.B.1.b (b) refueling floor $\leq$100 mr/hr 3. When irradiated fuel is in the reactor vessel and the reactor water temperature is above 212°F, the limiting conditions for operation for the instrumentation listed in Table 3.2.4 shall be met.	

Amendment No. 15

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
F. Recirculation Pump Trip Initiation 1. Whenever the reactor is in the RUN Mode, the limiting conditions for operation for the instrumentation listed in Table 3.2.5 shall be met. G. Safeguards Bus Voltage Protection 1. Whenever the safeguards auxiliary electrical power system is required to be operable by Specification 3.9, the Limiting Conditions for Operation for the instrumentation listed in Table 3.2.6 shall be met.	

Table 4.2.1 - Continued
Minimum Test and Calibration Frequency For Core Cooling
Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
3. Steam Line Low Pressure	Note 1	Once/3 months	None
4. Steam Line High Radiation	Once/week (5)	Note 6	Once/shift
HPCI ISOLATION			
1. Steam Line High Flow	Once/month	Once/3 months	None
2. Steam Line High Temperature	Once/month	Once/3 months	None
RCIC ISOLATION			
1. Steam Line High Flow	Once/month	Once/3 months	None
2. Steam Line High Temperature	Note 1	Once/3 months	None
REACTOR BUILDING VENTILATION			
1. Radiation Monitors (Plenum)	–	See Table 4.8.2	–
2. Radiation Monitors (Refueling Floor)	Note 1	Once/3 months	(4)
RECIRCULATION PUMP TRIP			
1. Reactor High Pressure	Note 1	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/Day
2. Reactor Low Water Level (Note 7)	Once/month	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/shift

Amendment No. 15

Bases Continued:

3.2 For effective emergency core cooling for the small pipe break the HPCI or Automatic Pressure Relief system must function since for these breaks, reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria is met. Reference Section 6.2.4 and 6.2.6 FSAR. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors measure radioactivity of the reactor building ventilation exhaust and on the refueling floor. Any one upscale trip will cause the desired action. Trip settings for the ventilation exhaust isolation monitors are based upon initiating normal ventilation isolation and Standby Gas Treatment System operation prior to exceeding the maximum release rate limit for the reactor building vent. Trip settings of 100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation stack but that all the activity is processed by the standby gas treatment system.

The recirculation pump trip description and performance analysis is discussed in Topical Report NEDO-25016, September 1976, "Evaluation of Anticipated Transients Without Scram for the Monticello Nuclear Generating Plant". (See September 15, 1976 letter from Mr L O Mayer, NSP, to Mr D L Ziemann, USNRC.) The pump trip is provided to minimize reactor pressure in the highly unlikely event of a plant transient coincident with the failure of all control rods to scram. The rapid flow reduction

analyzed for radioactive iodines of I-131 through I-135 during power operation. In addition, where steam jet air ejector monitors indicate an increase in radioactive gaseous effluents of 25 percent or 5000 uCi/sec, whichever is greater, during steady state reactor operation a reactor coolant sample shall be taken and analyzed for radioactive iodines.

(b) Isotopic analysis of reactor coolant samples shall be made at least once per month.

(c) Whenever the steady state radioiodine concentration of prior operation is greater than 1 percent but less than 10 percent of Specification 3.6.C.1, a sample of reactor coolant shall be taken within 24 hours of any reactor startup and analyzed for radioactive iodines of I-131 through I-135.

(d) Whenever the steady state radioiodine concentration of prior operation is greater than 10 percent of Section 3.6.C.1, a sample of reactor coolant shall be taken daily and prior to any reactor startup and analyzed for radioactive iodines of I-131 through I-135 as well as the coolant sample and analyses required by Specification 4.6.C.1.(c) above.

3.8 RADIOACTIVE EFFLUENTS

Applicability:

Applies at all times to the liquid and gaseous radioactive effluents from the plant and the solidification and packaging for shipment of solid radioactive waste.

Objective:

To implement the requirements of 10CFR20, 10CFR71, 10CFR50 Section 50.36a, Appendix A and Appendix I to 10CFR50, 40CFR141, and 40CFR190 pertaining to radioactive effluents.

Specification:

A. Liquid Effluents

1. Concentration

a. The concentration of liquid radioactive material released from the site (Figure 3.8.1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} uci/ml total activity.

b. When the concentration of radioactive material in liquid released from the site exceeds the limits in (a) above, immediately restore concentration within acceptable limits.

4.8. RADIOACTIVE EFFLUENTS

Applicability:

Applies to the sampling monitoring, and recording of liquid and gaseous radioactive effluents, verification of solidification, and verification of equipment operability.

Objective:

To implement the requirments of 10CFR20, 10CFR71, 10CFR50 Section 50.36a, Appendix A and Appendix I to 10CFR50, 40CFR141, and 40CFR190 pertaining to radioactive effluents.

Specification:

A. Liquid Effluents

1. Concentration

a. Radioactive material in liquid effluent released from the site shall be continuously monitored in accordance with Table 3.8.1.

b. The liquid effluent monitors having provisions for automatic alarming, as listed in Table 3.8.1, shall be used to limit the concentration of radioactive material released at any time from the site to the values given in 3.8.A.1.a. Setpoints shall be determined in accordance with the methods in the Off-site Dose Calculation Manual (ODCM).

c. Surveillance of liquid effluent monitoring instrumentation shall be performed as required by Table 4.8.1.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	d. Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.8.3. e. The results of radioactive analysis shall be used in accordance with the methods of the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.8.A.1.a.

AH. Source Check - A Source Check is the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

AI. Purging - Purging is the controlled process of discharging air or gas from a confinement to maintain temperature pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

AJ. Venting - Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required.

AK. Dose Equivalent I-131 - Dose Equivalent I-131 is the concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or in NRC Regulatory Guide 1.109, Rev 1, October, 1977.

AL. Offgas Treatment System - The Offgas Treatment System is the system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

AM.* Members of the Public - Means all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

AN. Site Boundary - Means a line within which the land is owned, leased, or otherwise controlled by the licensee. The site boundary for liquid releases of radioactive material is defined in Figure 3.8.1. The site boundary for gaseous releases of radioactive material is defined in Figure 3.8.2.

AO. Unrestricted Areas - Means any area at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial, institutional and/or recreational purposes.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. Dose	2. Dose
a. The dose or dose commitment to an individual from radioactive materials in liquid effluents released from the site (Figure 3.8.1) shall be limited: 1. During any calendar quarter to $\leq$ 1.5 mrem to the total body and to $\leq$ 5 mrem to any organ, and 2. During any calendar year to $\leq$ 3 mrem to the total body and to $\leq$ 10 mrem to any organ.	a. Cumulative dose contributions for the current calendar quarter and year from liquid effluents shall be determined in accordance with the ODCM monthly.
b. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, within 30 days submit to the Commission in lieu of any other report, a special report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce the releases and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits. This special report shall also include (1) the results of radiological analyses of the drinking water source, and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40CFR141, Safe Drinking Water Act.	

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3. Liquid Radwaste System a. The liquid radwaste treatment system shall be used used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent from the site (Figure 3.8.1) when averaged over one month would exceed 0.06 mrem to the total body or 0.2 mrem to any organ. b. With radioactive liquid waste being discharged without treatment in excess of the limit in (a) above, within 30 days submit to the Commission, in lieu of any other report, a special report which includes the following information: 1. Identification of the inoperable equipment or subsystems and the reason for inoperability, 2. Action(s) to be taken to restore equipment to operable status, and 3. Summary description of action(s) taken to prevent a recurrence.	3. Liquid Radwaste System a. Doses due to liquid releases shall be projected at least once each month in accordance with the ODCM.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
4. Liquid Holdup Tanks	4. Liquid Holdup Tanks
a. The quantity of radioactive material contained in each outside temporary tank shall be limited to $\leq$10 curies, excluding tritium and dissolved or entrained noble gases.	a. The quantity of radioactive material contained in each outside temporary tank shall be determined to be within the limit in 3.8.A.4.a by analyzing a representative sample of the contents of the tank at least weekly when radioactive materials are being added to the tank.
b. With the quantity of radioactive material in any outside temporary tank exceeding the limit in (a) above, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.	

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
B. Gaseous Effluents	B. Gaseous Effluents
1. Dose Rate	1. Dose Rate
a. The dose rate at any time due to radioactive materials released in gaseous effluents from the site (Figure 3.8.2) shall be limited to the following: 1. For noble gases to $\leq$500 mrem/year to the total body and $\leq$3000 mrem/year to the skin, and 2. For I-131, tritium, and radioactive particulates, with half-lives greater than eight days to $\leq$1500 mrem/year to any organ.	a. Radioactive material in gaseous effluents released from the site shall be continuously monitored in accordance with Table 3.8.2.
b. With the dose rate(s) exceeding the limits in (a) above, immediately decrease the release rate within acceptable limits.	b. The noble gas effluent monitors having provisions for the automatic termination of gaseous releases, as listed in Table 3.8.2 shall be used to limit offsite dose rates to the values established in Specification 3.8.B.1.a.1. Setpoints shall be determined in accordance with the ODCM.
	c. Surveillance of gaseous effluent monitoring instruments shall be performed as required by Table 4.8.2.
	d. The release rate of I-131, tritium and radioactive particulates with half-lives greater than eight days shall be determined by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.8.4. Following each analysis the dose rate due to I-131, tritium, and radioactive particulates with half-lives greater than eight days shall be determined to be less than the limit in Specification 3.8.B.1.a.2 in accordance with the ODCM.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. Dose from Noble Gases a. The air dose due to noble gases released in gaseous effluents from the site (Figure 3.8.2) shall be limited to the following values: 1. During any calendar quarter, to $\leq$5 mrad for gamma radiation and $\leq$10 mrad for beta radiation, and 2. During any calendar year, to $\leq$10 mrad for gamma radiation and $\leq$20 mrad for beta radiation. b. With the calculated air dose from radioactive noble gases in gaseous effluent exceeding any of the above limits, within 30 days submit to the Commission, in lieu of any other report, a special report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce the releases and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits.	2. Dose from Noble Gases a. Cumulative dose contributions for the current calendar quarter and year from noble gases in gaseous effluents shall be determined in accordance with the ODCM monthly.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3. Dose from I-131, tritium, and radioactive particulates with half-lives greater than eight days. a. The dose to any organ of an individual due to I-131, tritium, and radioactive particulates with half-lives greater than eight days released from the site (Figure 3.8.2) in gaseous effluent shall be limited to the following: 1. During any calendar quarter to ≤7.5 mrem, and 2. During any calendar year to ≤ 15 mrem. b. With the calculated dose from the release of I-131, tritium, and radioactive particulates with half-lives greater than eight days exceeding any of the above limits, within 30 days submit to the Commission, in lieu of any other report, a special report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce the releases and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits.	3. Dose from I-131, tritium, and radioactive particulates with half-lives greater than eight days. a. Cumulative dose contributions for the current calendar quarter and year from I-131, tritium, and radioactive particulates with half-lives greater than eight days in gaseous effluents shall be determined in accordance with the ODCM monthly.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
4. Offgas Treatment System	4. Offgas Treatment System
a. The offgas treatment system shall be in operation whenever the main condenser air ejector system is in operation. Components of the system shall be operated to provide the maximum holdup time obtainable except during periods of equipment maintenance.	a. Following each isotopic analysis of a sample of gases from the steam jet air ejector required by 4.8.B.5.c, verify that the maximum storage tank activity limit specified in 3.8.B.4.e cannot be exceeded using the method in the ODCM.
b. With gaseous waste being discharged for more than 7 days with a holdup time of less than 50 hours, within 30 days submit to the Commission, in lieu of any other report, a special report which includes the following information:	
1. Identification of equipment or sub-systems not functional and the reason.	
2. Action(s) taken to restore equipment to functional status.	
3. Summary description of action(s) taken to prevent a recurrence.	

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
c. The concentration of hydrogen in the offgas treatment system shall be limited to $\leq$ 2% by volume downstream of the recombiners. With the concentration of hydrogen in the offgas treatment system > 2% by volume, but $\leq$ 4% by volume, restore the concentration of hydrogen to < 2% by volume within 48 hours or suspend operation of the compressed storage subsystem.	
d. The hydrogen monitors shall be operable as specified in Table 3.8.2 and set to automatically close the recombiner inlet valves at $\leq$ 4% hydrogen by volume.	
e. The quantity of radioactivity after 12 hours holdup contained in each gas storage tank shall be limited to $\leq$ 22,000 curies of noble gases (considered as dose equivalent Xe-133).	

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
5. Main Condenser Offgas Activity	5. Main Condenser Offgas Activity
a. The gross gamma radioactivity release rate measured at the steam jet air ejector shall be limited to $\leq 2.6 \times 10^5$ uci/sec following a 30-minute decay.	a. The activity of radioactive material in gaseous form removed from the main condenser shall be continuously monitored by the steam jet air ejector monitors in accordance with Table 3.8.2.
b. When the limit in (a) above is exceeded, restore the gross gamma radioactivity release rate to within the limit within 72 hours or be in at least hot shutdown within the next 12 hours.	b. The steam jet air ejector monitors shall be set to automatically terminate offgas flow within 30 minutes at the limit established in Specification 3.8.B.5.a.
	c. The gross gamma radioactivity of noble gases from the main condenser air ejector shall be determined to be within the limit specified in 3.8.B.5.a at the following times by performing an isotopic analysis of a representative sample of gases: 1. Once every month. 2. Within 24 hours following an increase in the continuous monitor reading of 50% after factoring out increases due to power level.
6. Containment Venting and Purging	6. Containment Venting and Purging
a. Except for inerting operations following startup and deinerting prior to shutdown, containment venting and purging above cold shutdown shall be via the 2-inch bypass flow path using the Standby Gas Treatment System.	a. Except for inerting operations following startup and deinerting prior to shutdown, the containment shall be determined to be aligned for venting or purging through the Standby Gas Treatment System within 4 hours prior to start of and at least once per 12 hours during venting or purging of the containment above cold shutdown.
b. Containment inerting following startup and deinerting prior to shutdown shall be via the Reactor Building plenum and vent.	b. Prior to containment venting or purging, the sampling and analysis requirements of Table 4.8.4 shall be met.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
C. Solid Radioactive Waste	C. Solid Radioactive Waste
1. Solid Radwaste System	1. Solid Radioactive System
a. A solid radwaste system shall be Operable and used, as applicable, in accordance with a Process Control Program for solidification and packaging of radioactive wastes to ensure meeting the requirements of 10 CFR Part 20 and of 10 CFR Part 71 prior to shipment of radioactive wastes from the site.	a. The Process Control Program (PCP) shall be used to verify the Solidification of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g. filter sludges, spent resins, and chemical solutions).
b. With the packaging requirements of 10 CFR Part 20 or 10 CFR Part 71 not satisfied, suspend shipments of defectively packaged solid radioactive wastes from the site.	b. If any test specimen fails to verify Solidification, the Solidification of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative Solidification parameters can be determined in accordance with the PCP, and a subsequent test verifies Solidification. Solidification of the batch may then be resumed using the alternative Solidification parameters determined by the PCP.
	c. If the initial test specimen from a batch of waste fails to verify Solidification, the PCP shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate Solidification. The PCP shall be modified as required, as provided for in Section 6 of the Technical Specifications.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
D. Dose from All Uranium Fuel Cycle Sources 1. The dose or dose commitment to any member of the general public from all uranium fuel cycle sources is limited to $\leq$25 mrem to the total body or any organ (except for the thyroid, which is limited to $\leq$75 mrem) over a period of 12 consecutive months. 2. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.8.A.2.a.1, 3.8.A.2.a.2, 3.8.B.2.a.1, 3.8.B.2.a.2, 3.8.B.3.a.1, or 3.8.B.3.a.2, prepare and submit within 30 days of special report, in lieu of any other report, to the Commission which defines corrective actions and calculates the highest radiation exposure to any member of the general public from all uranium fuel cycle sources (including all effluent pathways and direct radiation). Unless this report shows that exposures are less than the 40 CFR Part 190 standard, either apply to the Commission for a variance to continue releases which exceed the 40 CFR Part 190 standard or reduce subsequent releases to permit the standard to be met.	D. Dose from All Uranium Fuel Cycle Sources 1. Cumulative dose contributions from all liquid and gaseous effluents shall be determined in accordance with Specifications 4.8.A.2.a, 4.8.B.2.a, and 4.8.B.3.a and in accordance with the ODCM.

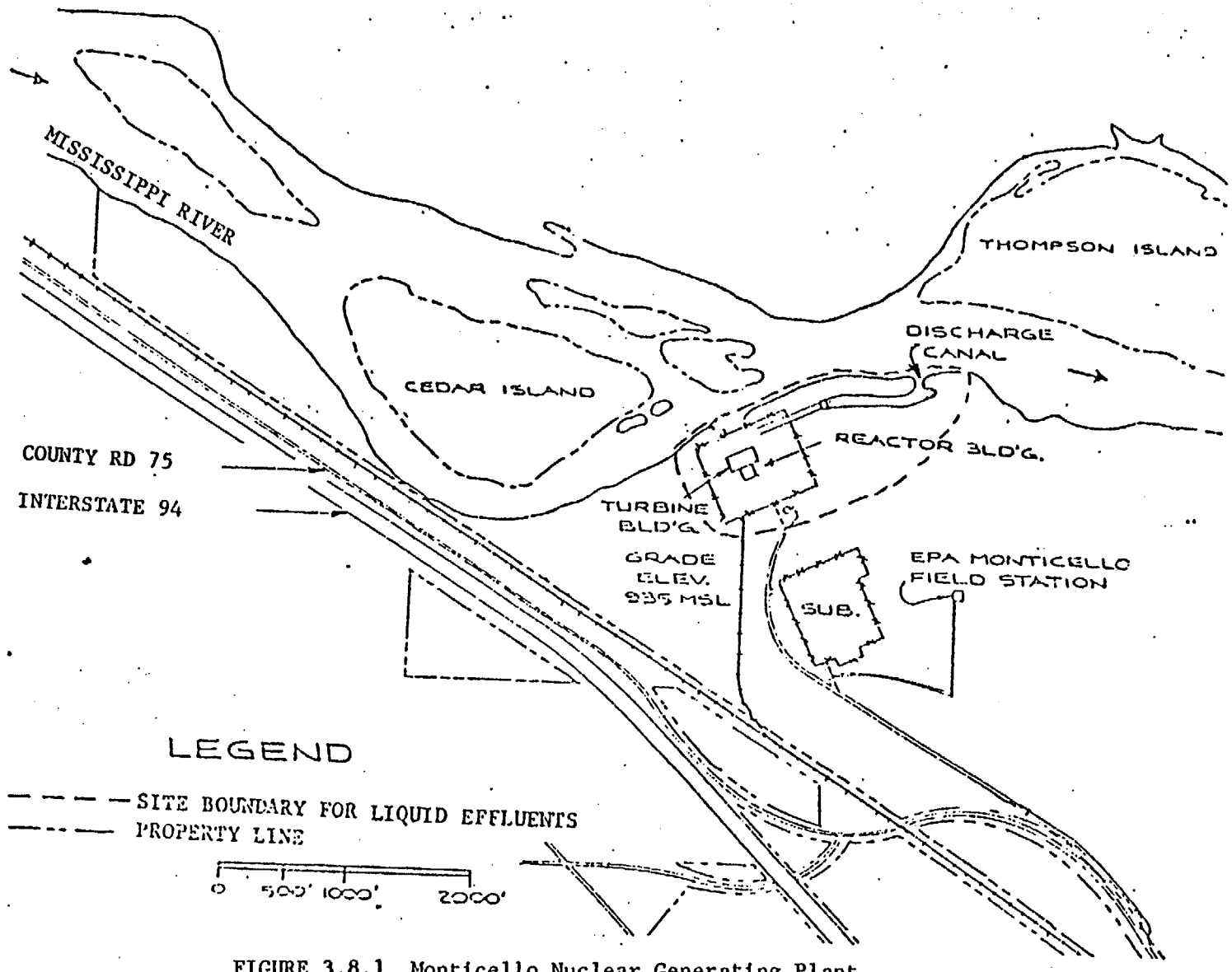

FIGURE 3.8.1 Monticello Nuclear Generating Plant
Site Boundary for Liquid Effluents

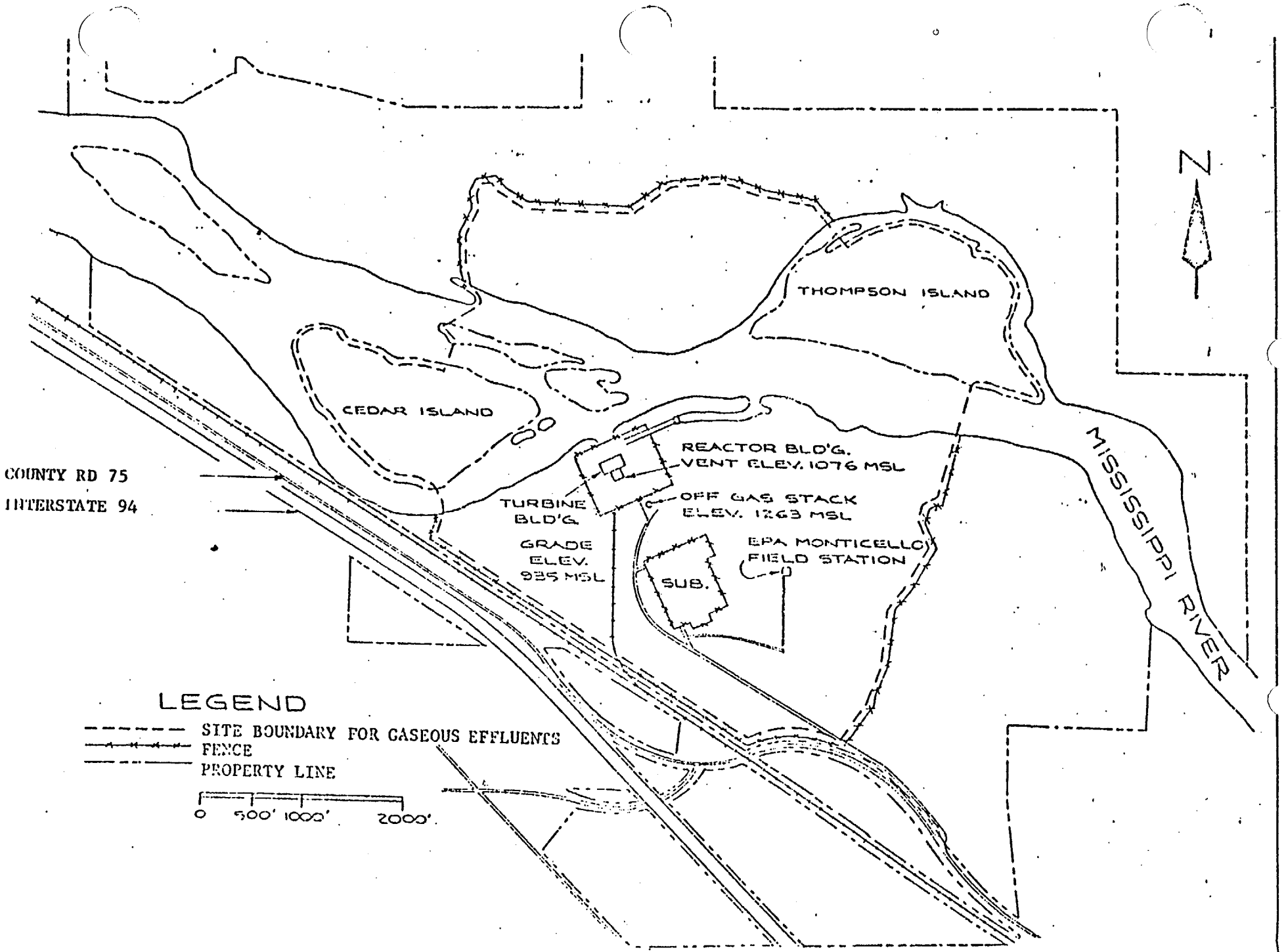

FIGURE 3.8.2 Monticello Nuclear Generating Plant
Site Boundary for Gaseous Effluents

TABLE 3.8.1 - RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION
(Page 1 of 2)

Instrument	Minimum Channels Operable	Applicability	Action if Minimum Channels not Operable
Liquid Radwaste Effluent Line Gross Radioactivity Monitor*	1	During release of liquid radwaste	Effluent releases may continue for up to 14 days provided that prior to initiating a release: a. At least two independent samples are analyzed in accordance with Specification 4.8.1.d b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving; Otherwise, supsend release of radioactive effluents via this pathway.
Liquid Radwaste Effluent Line Flow Instrument	1	During release of liquid radwase	Effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once every four hours during actual releases. Pump curves may be used to estimate flow.
Discharge Canal Flow Measurement Open Cycle Mode Closed/Helper Cycle Mode	 1 1	 During release of liquid radwaste	Effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once every four hours during actual releases. Pump curves may be used to estimate flow.
Discharge Canal Gross Radioactivity Monitor*	1	At all times	Liquid radwaste releases may continue for up to 30 days provided that at least once every 8 hours a grab sample shall be collected and analyzed for gross beta and gamma radioactivity at an LLD of 10^{-7} uCi/ml.
Service Water Discharge Pipe Gross Radioactivity Monitor*	1	At all times	Service water discharge may continue for up to 30 days provided that at least once every 8 hours a grab sample is collected and analyzed for gross beta and gamma radioactivity at an LLD of 10^{-7} uCi/ml.

TABLE 3.8.1 - RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION
(Page 2 of 2)

Instrument	Minimum Channels Operable	Applicability	Action if Minimum Channels not Operable
Turbine Building Normal Drain Sump Monitor*	1	At all times	Liquid sump releases may continue for up to 30 days provided that at least once every 8 hours a grab sample shall be collected and analyzed for gross beta and gamma radioactivity at an LLD of 10^{-7} uCi/ml.
Level Monitors for Temporary Outdoor Tanks Holding Radioactive Liquid	1	When tanks are in use	Liquid additions to a tank may continue for up to 30 days provided the tank liquid level is estimated during all liquid additions.

* - Indicates monitor provided with automatic alarm.

TABLE 3.8.2 - RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
(Page 1 of 2)

Instrument	Minimum Channels Operable	Applicability	Action if Minimum Channels not Operable
Main Condenser Air Ejector Noble Gas Activity Monitor	2	During air ejector operation	From and after the date that one of the two steam jet air ejector off-gas radiation monitors is made or found to be inoperable, continued reactor power operation is permissible provided the inoperable radiation monitor instrument channel is tripped. Upon loss of both steam jet air ejector off-gas radiation monitors, be in Hot Standy within six hours.
Main Condenser Offgas Treatment System Hydrogen Monitors	2#	During air ejector operation	Operation may continue for up to 14 days with one Operable channel per operating recombiner train. With all channels inoperable, be in Hot Standby within six hours.
Reactor Building Vent Isolation Noble Gas Monitors*	-	-	See Specification 3.2.E
Plant Stack			
Noble Gas Activity Monitors*	1	At all times	Releases via this pathway may continue for up to 30 days provided grab samples are taken and analyzed at least once every 8 hours.
Iodine Sampler Cartridge	1	At all times	Releases via this pathway may continue for up to 30 days provided within 8 hours samples are continuously collected with auxiliary sampling equipment as required by Table 4.8
Particulate Sampler Filter	1	At all times	Releases via this pathway may continue for up to 30 days provided within 8 hours samples are continuously collected with auxiliary sampling equipment as required by Table 4.8.4.
Stack Flow Monitor	1	At all times	Releases via the pathway may continue for up to 30 days provided the flow rate is estimated at least once every 4 hours.
Sample Flow Instrument	1	At all times	Releases via the pathway may continue for up to 30 days provided the flow rate is estimated at least once every 4 hours.

TABLE 3.8.2 - RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
(Page 2 of 2)

Instrument	Minimum Channels Operable	Applicability	Action if Minimum Channels not Operable
Reactor Building Vent (includes Turbine Building & Radwaste Building releases)			
Noble Gas Activity Monitor	1	At all times	Releases via this pathway may continue for up to 30 days provided grab samples are taken and analyzed at least once every 8 hours.
Iodine Sampler Cartridge	1	At all times	Releases via the pathway may continue for up to 30 days provided within 8 hours samples are continuously collected with auxiliary sampling equipment as required by Table 4.8.4.
Particulate Sampler Cartridge	1	At all times	Releases via the pathway may continue for up to 30 days provided within 8 hours samples are continuously collected with auxiliary sampling equipment as required by Table 4.8.4.
Duct Flow Monitors	1	At all times	Releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once every 4 hours.
Sample Flow Instruments	1	At all times	Releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once every 4 hours.

Notes:

- Indicates number of channels required per operating recombiner train.
* - Indicates noble gas effluent monitor having provision for automatic termination of gaseous release.

TABLE 4.8.1 - RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Sensor Check Frequency	Source Check Frequency	Functional Test Frequency	Calibration Frequency
Liquid Radwaste Effluent Line Gross Radioactivity Monitor	Daily during release	Immediately Prior to Each Release	Within 3 months prior to making a release	Within 12 months prior to making a release.*
Liquid Radwaste Effluent Line Flow Instrument	Daily during release	-	Within 3 months prior to making a release	Within 12 months prior to making a release.
Instruments used in Determination of Discharge Canal Flow	Daily during release	-	Within 3 months prior to making a release	Within 18 months prior to making a release.
Service Water Discharge Pipe Gross Radioactivity Monitor	Daily	Monthly	Quarterly	Each Operating Cycle *
Discharge Canal Gross Radioactivity Monitor	Daily	Monthly	Quarterly	Each Operating Cycle *
Turbine Building Normal Drain Sump Monitor	Daily	Monthly	Quarterly	Each Operating Cycle
Level Monitors for Temporary Outdoor Tanks Holding Radioactive Liquid	Daily when in use	-	Quarterly when in use	Each Operating Cycle when in use

* - The initial Instrument Calibration shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using sources traceable to NBS standards. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent calibration sources that have been related to the initial calibration shall be used.

TABLE 4.8.2 - RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS
(Page 1 of 2)

Instrument	Sensor Check Frequency	Source Check Frequency	Functional Test Frequency	Calibration Frequency
Main Condenser Air Ejector Noble Gas Activity Monitors	Daily during air ejector operation	-	Quarterly	Once each Operating Cycle
Main Condenser Offgas Treatment System Hydrogen Monitors	Daily during air ejector operation	-	Monthly	Quarterly #
Plant Stack Noble Gas Activity Monitors	Daily	Monthly	Quarterly	Once each Operating Cycle*
Plant Stack Iodine and Particulate Samplers	Weekly	-	-	-
Plant Stack Flow Monitor	Daily	-	-	Once each Operating Cycle
Plant Stack Sample Flow Instruments	Daily	-	-	Once each Operating Cycle

Amendment No. 15

TABLE 4.8.2 - RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS
(Page 2 of 2)

Instrument	Sensor Check Frequency	Source Check Frequency	Functional Test Frequency	Calibration Frequency
Reactor Building Vent Noble Gas Activity Monitors	Daily	Monthly	Quarterly	Once each Operating Cycle*
Reactor Building Vent Isolation Noble Gas Monitors	Daily	Monthly	Quarterly	Once each Operating Cycle
Reactor Building Vent Iodine and Particulate Samplers	Weekly	-	-	-
Reactor Building Vent Duct Flow Monitors	Daily	-	-	Once each Operating Cycle
Reactor Building Vent Sample Flow Instruments	Daily	-	-	Once each Operating Cycle

* - The initial Instrument Calibration shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using sources traceable to NBS standards. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent calibration sources that have been related to the initial calibration shall be used.

\# - The Calibration shall include the use of standard gas samples containing a nominal four volume percent hydrogen.

TABLE 4.8.3 - RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
(Page 1 of 3)

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)[a,e]
Batch Waste Release Tanks[b]	Each Batch	Each Batch	Principal Gamma Emitters[d]	5×10^{-7}
			I-131	1×10^{-6}
	One Batch Each Month	One Batch Each Month	Dissolved and Entrained Gases	1×10^{-5}
	Each Batch	Monthly Composite[c]	H-3	1×10^{-5}
			Gross alpha	1×10^{-7}
	Each Batch	Quarterly Composite[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.8.3 - RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
(Page 2 of 3)

Notes

a - The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

where:

LLD is the a priori lower limit of detection as defined above (as picocurie per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute). Typical values of E, V, Y and Δt shall be used in the calculations.

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting.

TABLE 4.8.3 - RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM (Continued)
(Page 3 of 3)

b. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated and then thoroughly mixed to assure representative sampling.

c. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

d. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

e. Nuclides which are below the LLD for the analyses shall be reported as "less than" the LLD of the nuclide and should not be reported as being present at the LLD level for that nuclide. The "less than" values shall not be used in the required dose calculations. When unusual circumstances result in LLD's higher than required, the reasons shall be documented in the Semiannual Radioactive Effluent Release Report.

TABLE 4.8.4 - RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM
(Page 1 of 2)

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)[a,e,f]
Containment Purge	Each Purge Grab Sample	Each Purge	Principal Gamma Emitters(f)	1×10^{-4}
			H-3[h]	1×10^{-6}
Plant Stack and Reactor Building Vent	Monthly[b] Grab Sample	Monthly	Principal Gamma Emitters(f)	1×10^{-4}
			H-3[i]	1×10^{-6}
	Continuous[g]	Weekly[c] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[g]	Weekly[c] Particulate Sample	Principal Gamma Emitters (I-131, Others)	1×10^{-11}
	Continuous[g]	Monthly[d] Composite Particulate Sample	Gross alpha	1×10^{-11}
	Continuous[g]	Quarterly[d] Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[g]	Continuous monitor	Gross gamma or gross beta noble gas activity	1×10^{-6}

Notes:

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal. Note (a) of Table 4.8.3 is applicable.

b. Grab samples taken at the discharge of the plant stack and reactor building vent are generally below minimum detectable levels for most nuclides with existing analytical equipment. For this reason, isotopic analysis data, corrected for holdup time, for samples taken at the steam jet air ejector may be used to calculate noble gas ratios.

c. Whenever the steady state radioiodine concentration is greater than 10 percent of the limit of Specification 3.6.C.1, daily sampling of reactor coolant for radioactive iodines of I-131 through I-135 is required. Whenever a change of 25% or more in calculated Dose Equivalent I-131 is detected under these conditions, the iodine and particulate collection devices for all release points shall be removed and analysed daily until it is shown that a pattern exists which can be used to predict the release rate. Sampling may then revert to weekly. When samples collected for one day are analyzed, the corresponding LLD's may be increased by a factor of 10. Samples shall be analyzed within 48 hours after removal.

d. To be representative of the average quantities and concentrations of radioactive materials in particulate form in gaseous effluents, samples should be collected in proportion to the rate of flow of the effluent streams.

e. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

f. Nuclides which are below the LLD for the analyses shall be reported as "less than" the LLD of the nuclide and should not be reported as being present at the LLD level for that nuclide. The "less than" values shall not be used in the required dose calculations. When unusual circumstances result in LLD's higher than reported, the reasons shall be documented in the semiannual effluent report.

g. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period sampled.

h. H^3 analysis shall not be required prior to venting if the limits of 3.8.B.1 are satisfied for other nuclides. The analysis shall be completed within 24 hours after sampling, however.

i. In lieu of grab samples, continuous monitoring with bi-weekly analysis using silica-jel samplers may be provided.

3.8 and 4.8 Bases:

A. Liquid Effluents

Specification 3.8.A.1 is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site will be less than the concentration levels specified in 10CFR Part 20, Appendix B, Table II. This limitation provides additional assurance that the levels of radioactive materials in the Mississippi River will not result in exposures exceeding (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to an individual and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

Specification 3.8.A.2.a is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. Action required by Specification 3.8.A.2.b provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable". Considering that the nearest drinking water supply using the receiving water is 33 river miles downstream, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I, Revision 1," April 1977. NUREG-0133, October, 1978, provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.113.

3.8 and 4.8 Bases: (continued)

Specification 3.8.A.3 provides assurance that the liquid radwaste treatment system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirements that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the guide set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

Restrictions on the quantity of radioactive liquid material contained in tanks are required only for temporary tanks. All exterior permanent tanks are diked to prevent release of their contents in the event of leakage. Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

Surveillance requirements for continuous liquid release points are not provided since all Monticello releases are "batch" type releases.

Radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm setpoints for these instruments will be calculated in accordance with NRC approved methods in the ODCM to ensure that the alarm will occur prior to exceeding the limits of 10 CFR Part 20. The operability requirements for instrumentation are consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

B. Gaseous Effluents

Specification 3.8.B.1.a is provided to ensure that the dose rate at anytime at the site boundary from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individuals who may at times be within the restricted area boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the restricted area boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the restricted area boundary to 500 mrem/year to the total body or to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to $\leq$1500 mrem/year at the site boundary.

Specification 3.8.B.2.a is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation implement the guides set forth in Section II.B of Appendix I. Action required by Specification 3.8.B.2.b provides the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at the restricted area boundary may be based upon the historical average atmospheric conditions. NUREG-0133, October, 1978 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111.

Specification 3.8.B.3.a is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The release rate specifications for I-131, tritium, and radioactive particulates with half-lives greater than eight days are dependent on the existing radionuclide pathways to man in the unrestricted area. The pathways which are examined in the development of these calculations are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

Specification 3.8.B.4 provides assurance that the offgas treatment system will be in operation whenever main condenser offgas is released to the environment. The requirement that the appropriate portions of this system be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section IID of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the guide set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

Radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments will be calculated in accordance with NRC approved methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The operability requirements for this instrumentation are consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

Specification 3.8.B.4.c is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas treatment system is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. Maintaining the concentration of hydrogen below the flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

Specification 3.8.B.4.e is provided to limit the radioactivity which can be stored in one decay tank. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tanks contents, the resulting total body exposure to an individual at the site restricted area boundary will not exceed 20 mrem. A flow restrictor in the discharge line of the decay tanks prevents a tank from being discharged at an uncontrolled rate. In addition, interlocks prevent the contents of a tank from being released with less than 12 hours of holdup.

Specification 3.8.B.5 establishes a maximum activity at the steam jet air ejector. Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the restricted area boundary will not exceed the limits of 10 CFR Part 20 in the event this effluent is inadvertently discharged directly to the environment with minimal treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

Specification 3.8.B.6 requires the containment to be purged and vented through the standby gas treatment system except during inerting and deinerting operations. This provides for iodine and particulate removal from the containment atmosphere. During outages when the containment is opened for maintenance, the containment ventilation exhaust is directed to the monitored reactor building vent. Use of the 2-inch flow path prevents damage to the standby gas treatment system in the event of a loss of coolant accident during purging or venting. Use of the reactor building plenum and vent flow path for inerting and deinerting operations permits the plenum monitors to automatically terminate releases in the event that release rate limits are exceeded.

C. Solid Radioactive Waste

Specification 3.8.C.1 provides assurance that the solid radwaste system will be used whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criteria 60 of Appendix A to 10 CFR Part 50.

D. Dose from All Uranium Fuel Cycle Sources

Specification 3.8.D is provided to meet the dose limitations of 40 CFR 190. The specification requires the preparation and submittal of a special report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. Submittal of the report is considered a timely request and a variance is granted until Staff action on the request is complete. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a real individual will exceed 40 CFR 190 if the individual reactors remain with the reporting requirement level. For the purpose of the special report it may be assumed that the dose commitment to the real individual from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

4.16 RADIATION ENVIRONMENTAL MONITORING PROGRAM

Applicability

Applies to the periodic monitoring and recording of radioactive effluents found in the plant environs.

Objective

To provide for measurement of radiation levels and radioactivity in the site environs on a continuing basis.

Specification

A. Sample Collection & Analysis

1. The Radiation Environmental Monitoring Program given in Table 4.16.1 shall be conducted. Radioanalysis shall be conducted meeting the requirements of Table 4.16.2.

   A map and a table identifying the locations of the sampling points shall be provided in the Offsite Dose Calculation Manual (ODCM).

2. Whenever the Radiation Environmental Monitoring Program is not being conducted as specified in Table 4.16.1, the Annual Radiation Environmental Monitoring Report shall include a description of the reasons for not conducting the program as required and plans for preventing a recurrence.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3. Deviations are permitted from the required sampling schedule if samples are unobtainable due to hazardous conditions, seasonable unavailability, or to malfunction of automatic sampling equipment. If the latter occurs, every effort shall be made to complete corrective action prior to the end of the next sampling period.

4. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of Table 4.16.3 when averaged over any calendar quarter, in lieu of any other report, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Report pursuant to Specification 6.7.C.2.a. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots > 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.8.A.2, 3.8.B.2, or 3.8.B.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiation Environmental Monitoring Report.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

5. Although deviations from the required sampling schedule are permitted under Item 3, above, whenever milk or leafy green vegetables can no longer be obtained from the designated sample locations required by Table 4.16.1, the Semi-annual Radioactive Effluent Release Report for this period shall explain why the samples can no longer be obtained and will identify the locations which will be added to and deleted from the monitoring program as soon as practicable.

B. Land Use Census

1. A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence, and the nearest garden of greater than 500 ft^2 producing fresh leafy vegetables, in each of the 16 meteorological sectors within a distance of five miles. The census shall also identify the locations of all milk animals and all 500 ft^2 or greater gardens producing broad leaf vegetation in each of the meteorological sectors within a distance of three miles. This census shall be conducted at least once per year between the dates of May 1 and October 31 by door to door survey, aerial survey, or by consulting local agricultural authorities associations.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	2. With a land use census identifying a location which yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 4.16.A.1, the Semiannual Radioactive Effluent Release Report for this period shall identify the new location. The new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.
	C. Interlaboratory Comparison Program
	1. Analyses shall be performed on radioactive materials supplied as part of an NRC approved interlaboratory comparison program as described in the ODCM.
	2. The results of analyses performed as a part of the above required program shall be included in the Annual Radiation Environmental Monitoring Report. When required analyses are not performed, corrective action shall be reported in the Annual Radiation Environmental Monitoring Report.

Table 4.16.1
(Page 1 of 5)

MONTICELLO NUCLEAR GENERATING PLANT
RADIATION ENVIRONMENTAL MONITORING PROGRAM
SAMPLE COLLECTION AND ANALYSIS

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
1. Airborne Radioiodine & Particulates	Samples from 5 locations: 3 samples from offsite locations (in different sectors) of the highest calculated annual average ground level D/Q, 1 sample from the vicinity of a community having the highest calculated annual average ground-level D/Q, and 1 sample from a control location 8-20 miles distance and in the least prevalent wind direction	Continuous Sampler operation with sampler collection weekly.	Radioiodine analysis Weekly for I-131 Particulate: Gross beta activity on each filter weekly*. Analyses shall be performed more than 24 hours following filter change. Perform gamma isotopic analysis on composite (by location) sample quarterly.
2. Direct Radiation	37 TLD stations established with duplicate dosimeters placed at the following locations:	Quarterly	Gamma Dose quarterly

* If gross beta activity in any indication sample exceeds 10 times the yearly average of the control sample, a gamma isotopic analysis is required.

** Sample locations are given on the figure and table in the ODCM.

Table 4.16.1
(Page 2 of 5)

MONTICELLO NUCLEAR GENERATING PLANT
RADIATION ENVIRONMENTAL MONITORING PROGRAM
SAMPLE COLLECTION AND ANALYSIS

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
2. Direct Radiation (Con't.)			
	1. Using the 16 meterological wind sectors as guidelines, an inner ring of stations in the general area of the site boundary is established and an outer ring of stations at 4 to 5 mile distance from the plant site is established. Because of inacessibility, two sectors in the inner and outer rings are not covered.		
	2. Seven dosimeters are established at special interest areas and a control station.		
3. Waterborne			
a. Surface	Upstream & downstream locations	Monthly composite of weekly samples (water & ice conditions permitting)	Gamma Isotopic analysis of each monthly composite Tritium analysis of quarterly composites of monthly composites

** Sample locations are given on the figure and table in the ODCM.

Table 4.16.1
(Page 3 of 5)

MONTICELLO NUCLEAR GENERATING PLANT
RADIATION ENVIRONMENTAL MONITORING PROGRAM
SAMPLE COLLECTION AND ANALYSIS

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
3. Waterbourne (con't.)			
b. Ground	Three samples from wells within 5 miles of the plant site and one sample from a well greater than 10 miles from the plant site.	Quarterly	Gamma Isotopic and tritium analyses of each sample
c. Drinking	One sample from the City of Minneapolis water supply	Monthly composite of weekly samples.	I-131 Analysis and Gross beta and Gamma isotopic analysis of each monthly composite Tritium analysis of quarterly composites of monthly composites
d. Sediment from Shoreline	One sample upstream of plant, one sample downstream of plant, and one sample from shoreline of recreational area	Semiannually	Gamma isotopic analysis of each sample

** Sample locations are given on the figure and table in the ODCM.

Table 4.16.1
(Page 4 of 5)

MONTICELLO NUCLEAR GENERATING PLANT
RADIATION ENVIRONMENTAL MONITORING PROGRAM
SAMPLE COLLECTION AND ANALYSIS

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
4. Ingestion			
a. Milk	One sample from dairy farm having highest D/Q, one sample from each of three dairy farms calculated to have doses from I-131 >1 mrem/yr, and one sample from 10-20 miles	Monthly or biweekly if animals are on pasture	Gamma isotopic and I-131 analysis of each sample
b. Fish and Invertebrates	One sample of one game specie of fish located upstream and downstream of the plant site. One sample of Invertebrates upstream and downstream of the plant site.	Samples collected semi-annually	Gamma isotopic analysis on each sample (edible portion only on fish).

** Sample locations are given on the figure and table in the ODCM.

Table 4.16.1
(Page 5 of 5)

MONTICELLO NUCLEAR GENERATING PLANT
RADIATION ENVIRONMENTAL MONITORING PROGRAM
SAMPLE COLLECTION AND ANALYSIS

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
c. Food Products	One sample of corn from highest D/Q farm and one sample from 10-20 miles	At time of harvest	Gamma isotopic analysis of edible portion of each sample
	One sample of potatoes from highest D/Q farm and one sample from 10-20 miles	At time of harvest	Gamma isotopic analysis of edible portion of each sample
	One sample of broad leaf vegetation from highest D/Q garden and one sample from 10-20 miles	At time of harvest	I-131 analysis of edible portion of each sample

** Sample locations are given on the figure and table in the ODCM.

Table 4.16.2
(Page 1 of 2)

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,e]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4[b]	1 x 10-2				
^{3}H	2000(1000[b])					
^{54}Mn	15		130			
^{59}Fe	30		120			
$^{58,60}Co$	15		130			
^{65}Zn	30		260			
$^{95}Zr-Nb$	15[c]					
^{131}I	1[b, d]	7×10^{-2}		1[d]	60	
$^{134,137}Cs$	15(10[b]), 18	1×10^{-2}	130	15	60	150
$^{140}Ba-La$	15[c]			15[c]		

TABLE 4.16.2
(Page 2 of 2)

TABLE NOTATION

a - The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66 \ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

where

LLD is the a priori lower limit of detection as defined above (as picocurie per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute). Typical values of E, V, Y and Δt shall be used in the calculations.

E is the counting efficiency (as counts per transformation)

V is the sample size (in units of mass or volume)

2.22 is the number of transformations per minute per picocurie

Y is the fraction radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting

b - LLD for drinking water.
c - Total for parent and daughter.
d - Applies to specific isotope analysis-not to gamma spectrum analyses.
e - Other peaks which are measurable and identifiable, together with the radionuclides in TAble 4.15.2, shall be identified and reported.

Table 4.16.3

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Vegetables (pCi/kg, wet)
H-3	2×10^4 (a)				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-Nb-95	4×10^2 (b)				
I-131	2	0.9		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
Cs-137	50	20	2×10^3	70	2×10^3
Ba-La-140	2×10^2 (b)			3×10^2 (b)	

a - For drinking water samples
b - Total for parent and daughter

Amendment No. 15

3.16 and 4.16 BASES

A. Sample Collection & Analysis

The Radiation Environmental Monitoring Program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the plant operation. This program thereby supplements the radiological effluent monitoring by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. After a specific program has been in effect for at least three years of operation, program changes may be initiated based on this experience.

The detection capabilities required by Table 4.15.2 are state-of-the-art for routine environmental measurements in industrial laboratories. The LLD's for drinking water meet the requirement of 40CFR 141.

B. Land Use Census

This specification is provided to ensure that changes in the use of off site areas are identified and that modifications to the monitoring program are made if required by the results of this census. The best survey information from door-to-door, aerial or consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used, 1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/square meter.

C. Interlaboratory Comparison Program

The requirement for participation in an interlaboratory comparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as a part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonable valid.

f. Investigation of all events which are required by regulation or technical specifications to be reported to NRC in writing within 24 hours.

g. Revisions to the Facility Emergency Plan, the Facility Security Plan, and the Fire Protection Program.

h. Operations Committee minutes to determine if matters considered by that Committee involve unreviewed or unresolved safety questions.

i. Other nuclear safety matters referred to the SAC by the Operations Committee, plant management or company management.

j. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components.

k. Reports of special inspections and audits conducted in accordance with specification 6.3.

l. Changes to the Offsite Dose Calculation Manual (ODCM).

m. Review of investigative reports of unplanned releases of radioactive material to the environs.

6. Audit - The operation of the nuclear power plant shall be audited formally under the cognizance of the SAC to assure safe facility operation.

a. Audits of selected aspects of plant operation, as delineated in Paragraph 4.4 of ANSI N18.7-1972, shall be performed with a frequency commensurate with their nuclear safety significance and in a manner to assure that an audit of all nuclear safety-related activities is completed within a period of two years. The audits shall be performed in accordance with appropriate written instructions and procedures.

b. Audits of aspects of plant radioactive effluent treatment and radiological environmental monitoring shall be performed as follows:

1. Implementation of the Offsite Dose Calculation Manual and quality controls for effluent monitoring at least once every two years.
2. Implementation of the Process Control Program for solidification of radioactive waste at least once every two years.
3. The Radiological Environmental Monitoring Program and the results thereof, including quality controls, at least once every year.

c. Periodic review of the audit program should be performed by the SAC at least twice a year to assure its adequacy.

d. Written reports of the audits shall be reviewed by the Vice President - Power Production, by the SAC at a scheduled meeting, and by members of management having responsibility in the areas audited.

f. All events which are required by regulation or technical specifications to be reported to NRC in writing within 24 hours.

g. Drills on emergency procedures (including plant evacuation) and adequacy of communication with off-site support groups.

h. All procedures required by these Technical Specifications, including implementing procedures of the Emergnecy Plan and the Security Plan shall be reviewed with a frequency commensurate with their safety significance but at an interval of not more than two years.

i. Perform special reviews and investigations, as requested by the Safety Audit Committee.

j. Review of investigative reports of unplanned releases of radioactive material to the environs.

k. All changes to the Process Control Program (PCP) and the Offsite Dose Calculation Manual (ODCM).

5. Authority

The OC shall be advisory to the Plant Manager. In the event of disagreement between the recommendations of the OC and the Plant Manager, the course determined by the Plant Manager to be the more conservative will be followed. A written summary of the disagreement will be sent to the General Manager Nuclear Plants and the Chairman of the SAC for review.

6. Records

Minutes shall be recorded for all meetings of the OC and shall identify all documentary material reviewed. The minutes shall be distributed to each member of the OC, the Chairman and each member of the Safety Audit Committee, the General Manager Nuclear Plants and others designated by OC Chairman or Vice Chairman.

7. Procedures

A written charter for the OC shall be prepared that contains:

a. Responsibility and authority of the group.

b. Content and method of submission of presentations to the Operations Committee.

6.5 Plant Operating Procedures

Detailed written procedures, including the applicable check-off lists and instructions, covering areas listed below shall be prepared and followed. These procedures and changes thereto, except as specified in 6.5.F shall be reviewed by the Operation Committee and approved by a member of plant management designated by the Plant Manager.

A. Plant Operations

1. Integrated and system procedures for normal startup, operation and shutdown of the reactor and all systems and components involving nuclear safety of the facility.

2. Fuel handling operations.

3. Actions to be taken to correct specific and foreseen potential or actual malfunction of systems or components including responses to alarms, primary system leaks and abnormal reactivity changes and including follow-up actions required after plant protective system actions have initiated.

4. Surveillance and testing requirements that could have an effect on nuclear safety.

5. Implementing procedures of the security plan.

6. Implementing procedures of the emergency plan, including procedures for coping with emergency conditions involving potential or actual releases of radioactivity.

7. Implementing procedures of the fire protection program.

8. Implementing procedures for the Process Control Program and Offsite Dose Calculation Manual including quality control measures.

Drills on the procedures specified in A.3 above shall be conducted as a part of the retraining program. Drills on the procedures specified in A.6 above shall be conducted at least semi-annually, including a check of communications with offsite support groups.

B. Radiological

Radiation control procedures shall be maintained and made available to all plant personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20.

C. Maintenance and Test

The following maintenance and test procedures will be developed to satisfy routine inspection, preventive maintenance programs, and operating license requirements:

1. Routine testing of Engineered Safeguards and equipment as required by the facility license and the Technical Specifications.

2. Routine testing of standby and redundant equipment.

3. Preventive or corrective maintenance of plant equipment and systems that could have an effect on nuclear safety.

4. Calibration and preventive maintenance of instrumentation that could affect the nuclear safety of the plant.

5. Special testing of equipment for proposed changes to operational procedures or proposed system design changes.

D. Process Control Program (PCP)

The PCP shall be approved by the Commission prior to initial implementation. Changes to the PCP shall satisfy the following requirements:

1. A description of changes shall be submitted to the Commission with the Semi-Annual Radioactive Effluent Release Report for the period in which the changes(s) were made. This submittal shall contain:
   a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. a determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. documentation of the fact that the change has been reviewed and found acceptable by the Operations Committee.

2. Shall become effective upon review and acceptance by the Operations Committee.

E. Offsite Dose Calculation Manual (ODCM)

The ODCM shall be approved by the Commission prior to initial implementation. Changes to the ODCM shall satisfy the following requirements:

1. Shall be submitted to the Commission with the Semi-Annual Radioactive Effluent release report for the period in which the change(s) were made effective. This submittal shall contain:

   a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with a revision date, together with appropriate analyses or evaluations justifying the changes(s).

   b. a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. documentation of the fact that the change has been reviewed and found acceptable by the Operations Committee.

2. Shall become effective upon review and acceptance by the Operations Committee.

F. Temporary Changes to Procedures

Temporary changes to procedures described in A, B, C, D, and E above, which do not change the intent of the original procedures may be made with the concurrence of two individuals holding senior operator licenses. Such changes shall be documented, reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager within one month.

6.6 Plant Operating Records

A. Records Retained for Five Years

Records and logs relative to the following items shall be retained for a minimum of five years:

1. Normal plant operation including such items as power level, periods of operation at each level, fuel exposure and shutdowns.

2. Written shift supervisory and reactor logs.

3. Periodic checks, inspections, tests and calibrations of components and systems, as related to these Technical Specifications.

4. Reviews of changes made to procedures or equipment and reviews of tests and experiments.

2. Occupational Exposure Report.(1) An annual report of occupational exposure covering the previous calendar year shall be submitted prior to March 1 of each year.

The report should tabulate on an annual basis the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

3. Monthly Operating Report. A monthly report of operating statistics and shutdown experience covering the previous month shall be submitted by the 15th of the following month to the Office of Management Information and Program Control, U S Nuclear Regulatory Commission, Washington, DC 20555.

4. Semiannual Radioactive Effluent Release Report. Routine radioactive effluent release reports covering the operation of the unit during the previous six months of operation shall be submitted within 60 days after January 1st and July 1st of each year.

The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents as outlined in Appendix B of Regulatory Guide 1.21, Revision 1, June, 1974, with data summarized on a quarterly basis.

The report to be submitted 60 days after January 1st of each year shall include an assessment of the radiation doses from radioactive effluents released from the plant during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to individuals due to their activities inside the site boundary (Figures 3.8.1 and 3.8.2) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual (ODCM) or standard NRC computer codes.

1/ This report supplements the requirements of 10CFR20, Section 20.407. If 10CFR20, Section 20.407 is revised to include such information, this Specification is unnecessary.

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the general public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation.

The radioactive effluent release reports shall include the following information for solid waste shipped offsite during the report period.

a. container volume,

b. total curie quantity (specify whether determined by measurement or estimate).

c. principal radionuclides (specify whether determined by measurement or estimate),

d. type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

e. type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site of radioactive materials in gaseous and liquid effluents on a quarterly basis, changes to the ODCM, a description of changes to the PCP, a report of when milk or vegetable samples can not be obtained as required by Table 4.16.1, and changes in land use resulting in significant increases in calculated doses.

5. Annual Summaries of Meteorological Data

An annual summary of meteorological data shall be submitted for the previous calendar year in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability at the request of the Commission.

B. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date. Unless explicitly stated, the requirements of this section do not apply to the fire protection systems and measures contained in Section 3.13/4.13, the radiological effluent limitations and measures in Section 3.8/4.8, or the radiological environmental monitoring program in sections 3.16/4.16. Reporting requirements have been separately specified in those sections.

1. Prompt Notification With Written Followup

The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the appropriate NRC Regional Administrator or designate no later than the first working day following the event, with a written followup report within two weeks. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

   Note: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under items 6.7.B.1.e, 6.7.B.1.f, 6.7.B.2.a below.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

   Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the technical specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under item 6.7.B.2.b below.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment

Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this term.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation, greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if sub-critical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$ or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fullfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

Note: For items 6.7.B.1.e and 6.7.B.1.f reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under items 6.7.B.2.b. and 6.7.B.2.c. below.

g. Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

Note: This item is intended to provide for reporting of potentially generic problems.

j. Release of radioactive material in liquids from the site to the unrestricted areas in excess of the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases or in excess of 2×10^{-4} uCi/ml for total dissolved and entrained noble gases.

k. Release of radioactive material in gases from the site to unrestricted areas at a rate which exceeds the following dose rates:

For noble gases	-	500 mrem/year to the total body or 3000 mrem/year to the skin
For radioiodines and particulates with half-lives greater than eight days	-	1500 mrem/year to any organ

l. Exceeding the limits for the storage of radioactive materials in outside temporary tanks. The written follow-up report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

2. Thirty Day Written Reports

The reportable occurrences discussed below shall be the subject of written reports to the appropriate NRC Regional Administrator or designate within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items 6.7.B.2.a and 6.7.B.2.b need not be reported except where test results themselves reveal a degraded mode as described above.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in item 6.7.B.1.c above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

e. An unplanned offsite release of 1) more than one curie of radioactive material in liquid effluents, 2) more than 150 curies of noble gas in gaseous effluents, or 3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

1. A description of the event and equipment involved.
2. Cause(s) for the unplanned release.
3. Actions taken to prevent recurrence.
4. Consequences of the unplanned release.

C. Environmental Reports

The reports listed below shall be submitted to the Administrator of the appropriate Regional Office or designate:

1. Annual Radiation Environmental Monitoring Report

a. Annual Radiation Environmental Monitoring Reports covering the operation of the program during the previous calendar year shall be submitted prior to May 1 of each year.

b. The Annual Radiation Environmental Monitoring Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use consensus required by Specification 4.16.B.1. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

c. The Annual Radiation Environmental Monitoring Reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

d. The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from the reactor; and the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 4.16.C.1.

2. An Annual Environmental Monitoring and Ecological Studies Program Report shall be submitted by August 1 of the subsequent year. The report will include summaries, interpretations, and statistical evaluation of the results of the non-radiological environmental surveillance activities.

2. Special Reports

a. When radioactivity levels in samples exceed limits specified in Table 4.16.3 a Special Report shall be submitted within 30 days from the end of the affected calendar quarter. For certain cases involving long analysis time, determination of quarterly averages may extend beyond the 30 day period. In these cases the potential for exceeding the quarterly limits will be reported within the 30 day period to be followed by the Special Report as soon as practicable.

6.8 ENVIRONMENTAL QUALIFICATION

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-22 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 15 TO FACILITY OPERATING

LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 Introduction

To comply with Section V of Appendix I of 10 CFR Part 50, Northern States Power Company (the licensee) has filed with the Commission plans and proposed Technical Specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal operations, including expected operational occurrences, as low as is reasonably achievable. The licensee submitted this information to the Commission by letter dated May 1, 1979, revised July 23, 1982 (References 1 and 2) which requested changes to the Technical Specifications appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The proposed Technical Specifications update those portions of the Technical Specifications addressing radioactive waste management and make them consistent with the current staff positions as expressed in NUREG-0473. These revised Technical Specifications would reasonably assure compliance, in radioactive waste management, with the provisions of 10 CFR Part 50.36a, as supplemented by Appendix I to 10 CFR Part 50, with 10 CFR Part 20.105(c), 106(g), and 405(c); with 10 CFR Part 50, Appendix A, General Design Criteria 60, 63, and 64; and with 10 CFR Part 50, Appendix B.

2.0 Background and Discussion

2.1 Regulations

Section 50.36a of 10 CFR Part 50, "Technical Specifications on Effluents from Nuclear Power Reactors," provides that each license authorizing operation of a nuclear power reactor will include Technical Specifications that (1) require compliance with applicable provisions of Section 20.106 of 10 CFR Part 20, "Radioactivity in Effluents to Unrestricted Areas;" (2) require that operating procedures developed for the control of effluents be established and followed and that equipment installed in the radioactive waste system be maintained and used; and (3) require the periodic submission of reports to the NRC specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and gaseous effluents, any quantities of radioactive materials released that are significantly above design objectives, and such other information

8301040072 821217
PDR ADOCK 05000263
P PDR

as may be required by the Commission to estimate maximum potential radiation dose to the public resulting from the effluent releases.

Sections 20.105(c), 20.106(g), and 20.405(c) of 10 CFR Part 20, require that nuclear power plant and other licensees comply with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," and submit reports to the NRC when the 40 CFR Part 190 limits have been or may be exceeded.

Appendix A of 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," contains Criterion 60, Control of releases of radioactive materials to the environment; Criterion 63, Monitoring fuel and waste storage; and Criterion 64, Monitoring radioactivity releases. Criterion 60 requires that the nuclear power unit design includes means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Criterion 63 requires that appropriate systems be provided in radioactive waste systems and associated handling areas to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions. Criterion 64 requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents. Appendix B of 10 CFR Part 50 establishes quality assurance requirements for nuclear power plants.

Appendix I of 10 CFR Part 50, Section IV, provides guides on Technical Specifications for limiting conditions for operation for operations for light-water-cooled nuclear power reactors licensed under 10 CFR Part 50.

2.2 Standard Radiological Effluent Technical Specifications

NUREG-0473 (Reference 3) provides radiological effluent Technical Specifications for boiling water reactors which the staff finds to be an acceptable standard for licensing actions. Further clarification of these acceptable methods is provided in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants" (Reference 4). NUREG-0133 describes methods found acceptable by the staff to calculate certain key values required in the preparation of proposed radiological effluent Technical Specifications for light-water-cooled nuclear power plants. NUREG-0133 also provides guidance to licensees in preparing requests for changes to existing radiological effluent Technical Specifications for operating reactors. It also describes current staff positions on the methodology for estimating radiation exposure due to the release of radioactive materials in effluents and on the administrative control of radioactive waste treatment systems.

The above NUREG documents address all of the radiological effluent Technical Specifications needed to assure compliance with the requirements and also

provide guidance to meet the intent of the regulations. However, alternative approaches to the preparation of radiological effluent technical specifications and alternative radiological effluent Technical Specifications may be acceptable, if the staff determines that the alternatives are in compliance with and meet the intent of the regulations.

The standard radiological effluent Technical Specifications can be grouped under the following categories:

(1) Instrumentation
(2) Radioactive effluents
(3) Radiological environmental monitoring
(4) Design features
(5) Administrative controls

Each of the specifications under the first three categories are comprised of two parts: the limiting condition for operation and the surveillance requirements. The limiting condition for operation provides a statement of the limiting condition, the times when it is applicable, and the actions to be taken in the event that the limiting condition is not met.

In general, the specifications that assure compliance with 10 CFR Part 20 standards require that when the limiting conditions of operation (LCO) are exceeded, conditions are restored (within LCO), without delay. In general, the specifications that assure compliance with 10 CFR Part 50 require that when the limiting conditions for operation are exceeded, corrective actions are to be taken within specified times; alternative means of operation are to be employed, and certain reports are to be submitted to the NRC describing these conditions and actions.

The specifications concerning design features and administrative controls contain no limiting conditions foroperation or surveillance requirements.

Table 1 indicates the standard radiological effluent Technical Specifications that are needed to assure compliance with the particular provisions of the regulations described in Section 1.0.

3.0 Evaluation

3.1 General Description of Radiological Effluent System

This section briefly describes the radwaste liquid and gaseous effluent treatment and control systems installed at the Monticello Nuclear Generating Plant.

Table 1. Relation Between Provisions of the Regulations and the Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors and Boiling Water Reactors

● Indicate the specifications that are needed to assure compliance with the identified provision of the regulations.

Provisions of Title 10 Code of Federal Regulations	Standard Radiological Effluent Technical Specifications: Instrumentation		Radioactive Effluents: Liquid			Radioactive Effluents: Gaseous: PWR/BWR			Radioactive Effluents: Gaseous: PWR		Radioactive Effluents: Gaseous: BWR					Rad. Envir. Monitoring			Design Features	Administrative Control						
	Rad. Liquid Effl. Monitoring	Rad. Gas. Effl. Monitoring	Effluent Concentration	Dose	Liquid Radwaste Treatment	Liquid Holdup Tanks	Dose Rate	Dose Noble Gases	Dose I-131, Trit. and Part.	Explosive Gas Mixture	Gaseous Radwaste Treatment	Gas Storage Tanks	Gaseous Radwaste Treatment	Ventilation Exhaust Treatment	Main Condenser	Mark I or II Containment	Solid Radioactive Waste	Total Dose	Rad. Env. Monitoring Program	Land Use Census	Interlab. Comparison Program	Site Boundaries*	Review and Audits	Procedures	Reports	Record Retention	Process Control Program	Offsite Dose Calc. Manual	Major Changes to Rad. Systems
§ 50.36a Technical specifications on effluents from nuclear power reactors																													
Remain within limits of § 20.106			●			●	●					●				●												●	
Establish and follow procedures to control effluents			●	●	●	●	●	●	●	●	●	●	●	●	●	●	●							●				●	
Maintain and use radioactive waste system equipment				●	●			●	●		●		●	●														●	
Submit reports, semi-annual and other	●	●	●	●	●	●	●	●	●	●	●	●	●	●			●	●									●	●	●
§§ 20.105(c), 20.106(g), 20.405(c) Compliance with 40 CFR 190																		●	●	●	●							●	
Part 50 Appendix A - General Design Criteria																													
Criterion 60 - Control of releases of radioactive materials to the environment	●	●			●	●				●	●	●	●	●	●	●	●							●			●	●	
Criterion 61 - Fuel storage and handling and radioactivity control	●	●																											
Criterion 63 - Monitoring fuel and waste storage	●	●																											
Criterion 64 - Monitoring radioactivity releases	●	●													●				●									●	
Part 50 Appendix B - Quality Assurance Criteria	●	●																			●		●	●			●		
Part 50 Appendix I - Guides to Meet "As Low As Is Reasonably Achievable (ALARA)"																													
Maintain releases within design objectives				●	●			●	●		●			●														●	
Establish surveillance & monitoring program to provide data on:																													
(1) quantities of rad. matls. in effluents	●	●																											
(2) radiation & rad. matls. in the environment																			●						●	●			
(3) changes in use of unrestricted areas																				●									
Exert best efforts to keep releases "ALARA"				●	●			●	●		●		●	●															●
Submit report if calculated doses exceed the design objective				●				●	●		●			●											●				
Demonstrate conform. to des. obj. by calc. proced.				●				●	●		●			●														●	
Part 100															●														

*Note: Needed to fully implement other specifications.

3.1.1 Liquid Effluents

Most of the water required for the operation of the Monticello Nuclear Generating Plant is taken from the Mississippi River and returned to the river, or evaporated. The liquid and gaseous discharge pathways are shown in Figure 1. The discharge point for the service water system is the discharge canal. Any potential component cooling water releases would be via the service water system. The schematic diagram and flow path of the radioactive liquid waste system is shown in Figure 2, but it should be noted that no liquid radwaste has been discharged for 6 years and no discharges are anticipated. Special permission is required from the State of Minnesota before a release may be made.

3.1.2 Gaseous Effluents

The gaseous effluent discharge points, shown in Figure 1, are the plant stack and the reactor building vent. The turbine building exhaust has been modified to vent to the reactor building exhaust plenum. These release points are continuously monitored. Radiation monitors automatically isolate the plant stack and/or the reactor building vent if a high level of radioactivity is detected. Figure 3 shows the gaseous radwaste processing system at Monticello.

3.2 Radiological Effluent Technical Specifications (RETS)

Our consultant, EG&G, used as guidance NUREG-0473 and NUREG-0133 (References 3 and 4), to evaluate the licensee's proposed changes to the Technical Specifications against the requirements of (1) 10 CFR 50.36a, as supplemented by Appendix I to 10 CFR Part 50, (2) 10 CFR 20.105(c), 106(g) and 405(c), (3) 10 CFR Part 50, Appendix A: General Design Criteria 60, 63 and 64, and (4) 10 CFR Part 50, Appendix B. EG&G evaluated the following: (1) a review of information provided in the licensee's May 1, 1979 (revised January 24, 1980) (RETS) submittal (References 1 and 5), (2) a review of the licensee's November 25, 1981 submittal (Reference 6), (3) the resolution of problem areas in that submittal by means of a site visit (Reference 7), (4) a review of the licensee's April 2, 1982 submittal (Reference 8), (5) telephone conferences on May 19 and May 26, 1982 (References 9 and 10), to discuss the review, and (6) review of the licensee's July 23, 1982 revision (Reference 2). A conference call then was held on August 6, 1982 with the EG&G review team to discuss deviations from the model requirements. All open questions were resolved at this time and a Technical Evaluation Report (TER) was finalized for transmittal (Reference 11).

EG&G concluded that the proposed changes to the Monticello Technical Specifications meet the provisions of 10 CFR Part 50, Appendix I (subject to subquent discussion between the NRC and the licensee's staff). To account for plant unique differences, the licensee and the staff agreed to minor modifications which still meet the intent of NUREG-0473.

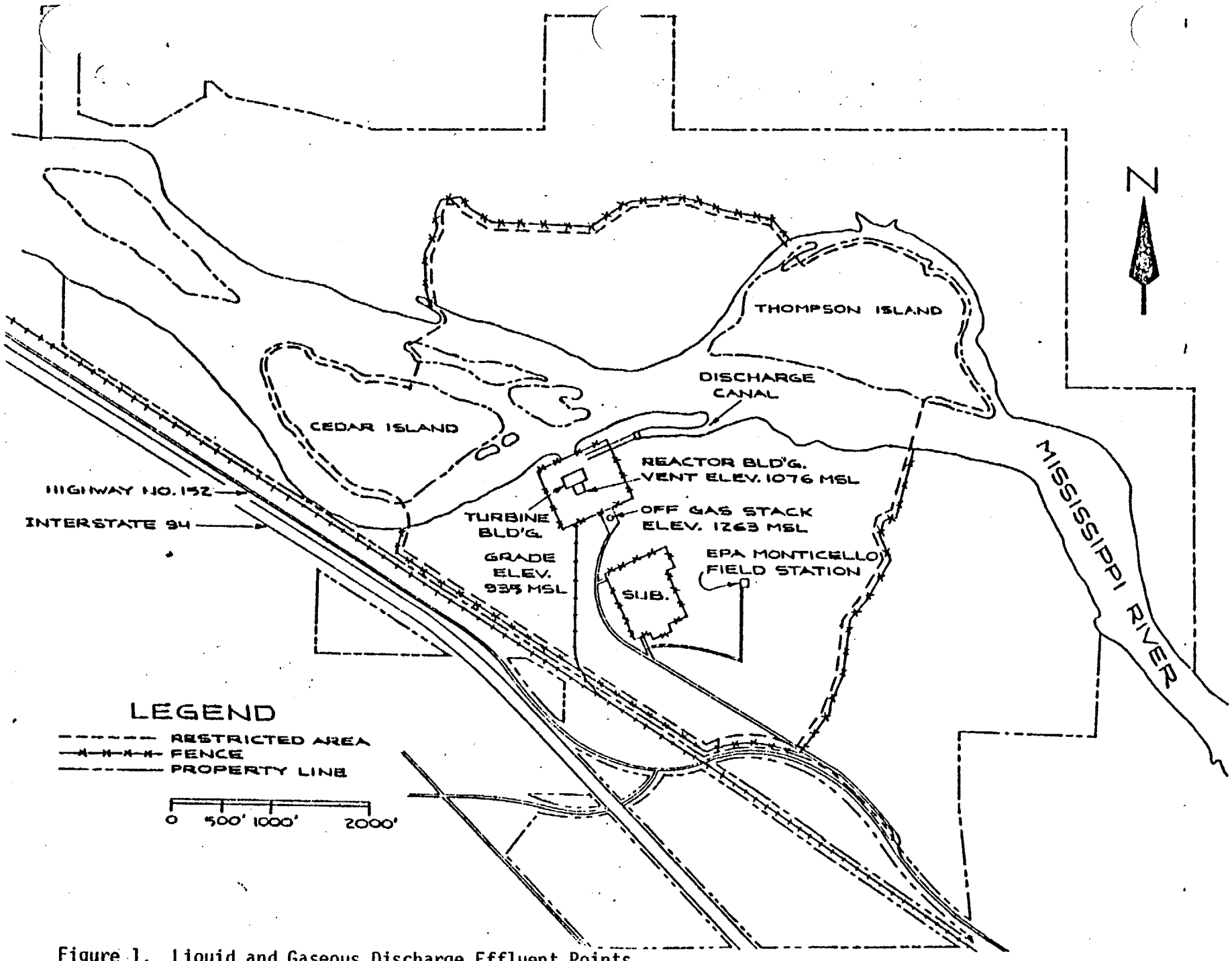

Figure 1. Liquid and Gaseous Discharge Effluent Points.

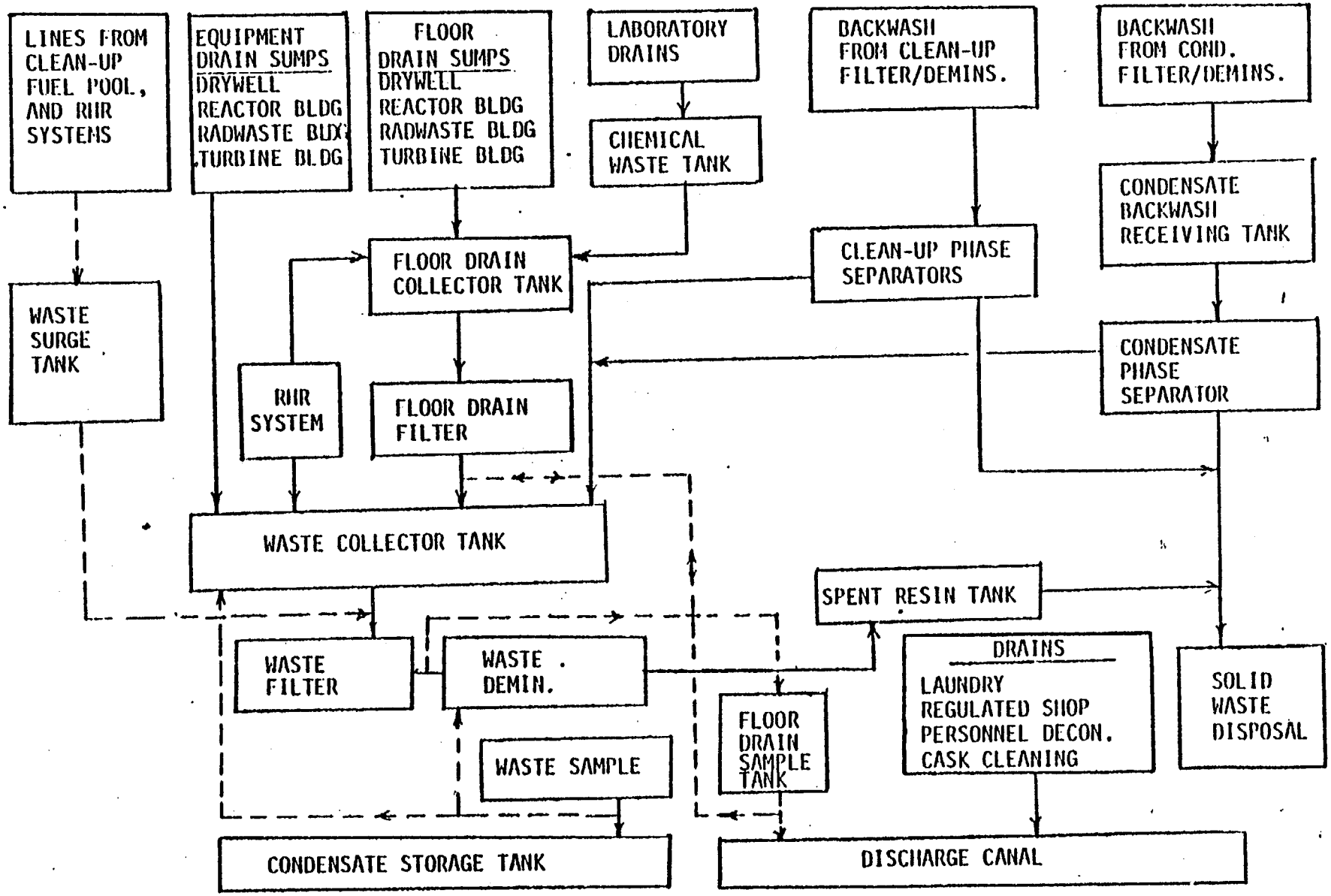

Figure 2. Monticello Liquid Radwaste System.

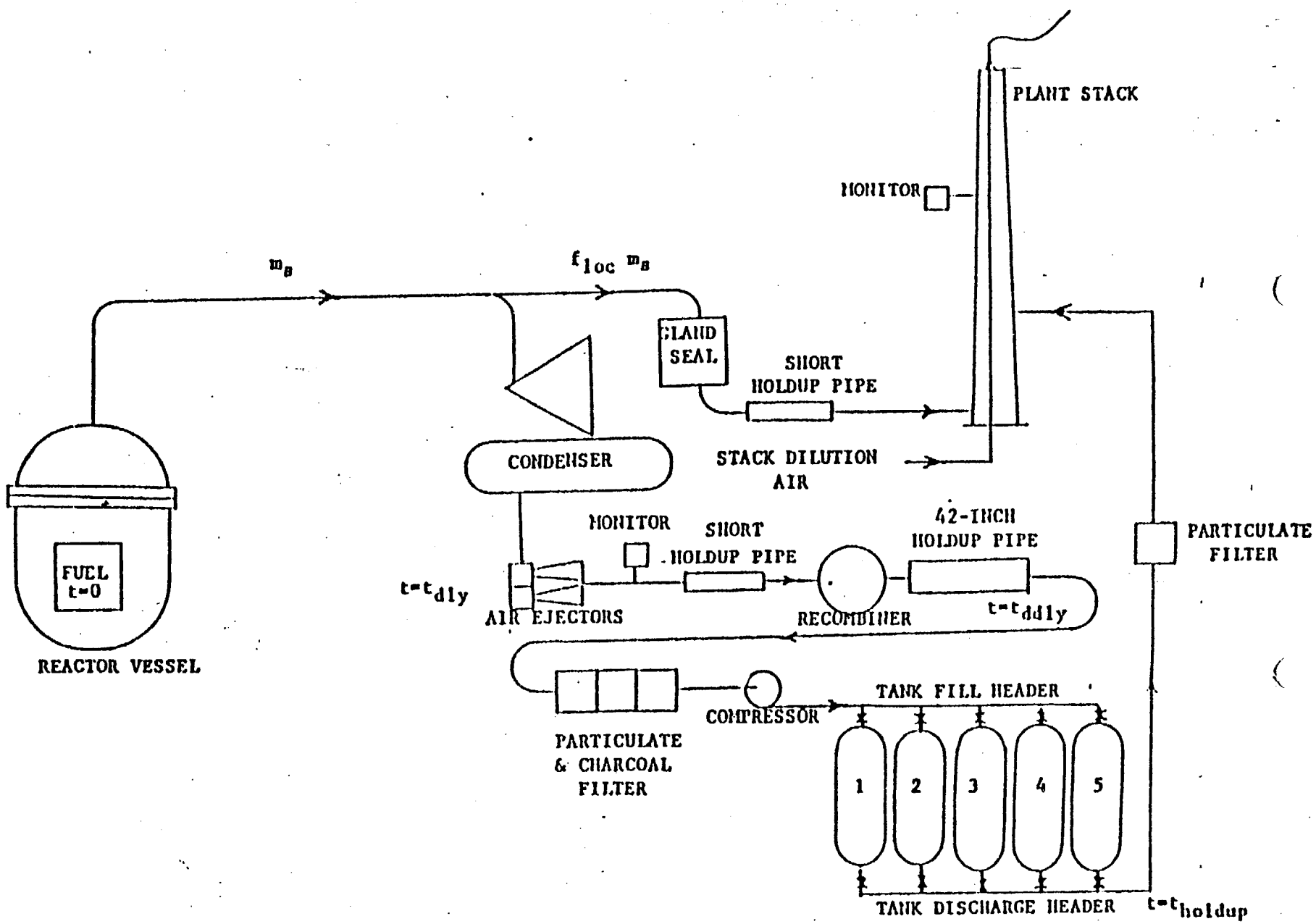

Figure 3. Monticello Gaseous Radwaste Treatment System.

3.2.1 Effluent Instrumentation

The primary objective of the RETS with regard to effluent instrumentation is to ensure that all significant releases of radioactivity are monitored. The licensee's information provides assurance that all effluent release points are monitored. In the past 6 years, no liquid radwaste discharges have been made from the plant because Minnesota is concerned about releases to the Mississippi River. Therefore, the provision for automatic termination of liquid radwaste releases is not necessary since releases may be made only under very special conditions. Before any releases are made, functional surveillance is required. Furthermore, because the component cooling water system at Monticello is a closed system, monitoring is not required; any potential releases would be via monitored systems (i.e., the service water system and the discharge canal).

Gaseous effluent releases from Monticello are monitored, including the turbine building ventilation. A recent design change has been made to route the turbine building gaseous effluents to the reactor building plenum, a monitored release point. The reactor building plenum has three noble gas monitors; one for monitoring purposes and two for building isolation and initiation of the standby gas treatment system. The plant stack noble gas monitoring system initiates automatic isolation and monitors the main condenser effluent off-gas. Some instrumentation surveillance will be done less frequently than required by the model (i.e., the monthly source check for the air ejector monitors and the stack flow rate monitors), but adequately to assure proper quantification of releases. Air ejector radionuclide releases are measured at the stack and conservative flow rate assumptions are used to assure that stack flows are quantified. Therefore, we find this acceptable.

3.2.2 Concentration and Dose Rates of Effluents

The objective of the RETS with regard to concentration and dose rates of effluents is to ensure that offsite effluent concentrations do not exceed the maximum permissible concentration (MPC) established by 10 CFR Part 20, Appendix B Table II, Columns 1 and 2. The licensee has stated that the concentration of radioactive material will be monitored "at all times," or "during releases" for batch releases. The setpoints at each release point are pre-established to prevent exceeding the release concentrations or corresponding dose rates of 10 CFR Part 20 in unrestricted areas. The concentration of liquid effluents and the dose rate due to gaseous effluents will be determined in accordance with the Offsite Dose Calculational Manual (ODCM). We have reviewed this and find it acceptable.

The gaseous monitoring systems are equipped with automatic termination of effluents. Should concentrations exceed the MPC specified in 10 CFR Part 20, based on monitoring setpoint values, release rates will immediately be decreased.

The concentration of radioactive materials in releases from Monticello will be determined as required by the RETS. The ventilation exhaust systems will be monitored as required except for noble gas grab samples. The noble gas concentration in the plant exhaust is generally below minimum detection limits. Therefore, it is considered acceptable to take samples of the air ejector exhaust and with appropriate decay periods to calculate the release rates from the ventilation exhausts. Also, the sampling requirements for startups, shutdowns, and 15% power changes is considered too severe for normal failed fuel levels. Therefore, a daily sampling requirement is proposed when the activity in the primary coolant is $\geq$ 10% of the Technical Specification level and/or when a 25% change in the reactor coolant dose equivalent I-131 concentration occurs. We have reviewed this and find it acceptable because it meets the intent of the model RETS.

3.2.3 Offsite Doses From Effluents

The objective of the RETS with regard to offsite doses from effluents is to ensure that offsite doses are kept As Low As Reasonable Achievable (ALARA), and are kept to a small fraction of the 10 CFR Part 20 limits, and are in accordance with 10 CFR Part 50, Appendix I. The licensee has proposed changes to the Technical Specifications to meet the quarterly and yearly dose criteria for liquid effluents and to use the ODCM methodology for determining the cumulative dose to individuals, thus meeting the intent of NUREG-0473. The licensee has proposed changes to the Technical Specifications to maintain the air doses in unrestricted areas, for noble gases, to those specified in Section 3.11.2.2 of the model RETS. The licensee has also proposed changes to the Technical Specifications to maintain the dose level to an individual from release of Iodine-131, tritium and radioactive particulates with half-lives greater than 8 days to meet the design objectives of Section 3.11.2.3, of the model RETS, thus satisfying the intent of NUREG-0473. We have reviewed this and find it acceptable.

3.2.4 Effluent Treatment

The objectives of the RETS with regard to effluent treatment are to ensure that wastes are treated to keep releases ALARA and to satisfy the requirement for Technical Specifications governing the maintenance and use of radwaste treatment equipment. The licensee has proposed changes to the Technical Specifications to use the liquid and gaseous radwaste treatment system when the projected doses averaged over 31 days exceed 25% of the annual dose design objectives prorated monthly. The gaseous waste treatment must be in operation at all times, except for maintenance periods, to meet the dose design objectives. This meets the intent of 10 CFR Part 50, Appendix I, Section II.D. The licensee has proposed changes to the Technical Specifica- least to make necessary dose projections in accordance with the ODCM, at least once per month. This indicates that the licensee has met the intent of NUREG-0473. Therefore, we find the licensee's response acceptable.

3.25 Tank Inventory Limits

The objective of the RETS with regard to tank inventory limits is to ensure that the rupture of a radwaste tank would not cause offsite doses greater than the limits set in 10 CFR Part 20 for nonoccupational exposure. The licensee has put a curie limit on all temporary, outside liquid tanks that are not diked and has committed to surveillance in accordance with NUREG-0473. For liquid holdup tanks, this limit (i.e., $\leq$10 curies) excludes tritium and dissolved or entrained noble gases. For waste gas storage tanks which are in constant use, a limit of $\leq$ 22000 curies for noble gases has been set. Surveillance to determine gas storage tank inventory will be done by calculation from the steam jet air ejector off-gas analysis. This is considered an acceptable method of determining that an unplanned release from a waste gas decay tank could not exceed effluent release limits.

3.2.6 Explosive Gas Mixtures

The objective of the RETS with regard to explosive gas mixtures is to prevent hydrogen explosions in the main condenser off-gas treatment system. The licensee has proposed changes to the Technical Specifications that maintain a safe concentration of hydrogen in this system. The hydrogen concentration will be maintained at $\leq$2%. If the concentration increases above this limit, it will be reduced to the acceptable limit within 48 hours. The licensee will maintain a constant monitoring of H_2 by using a minimum of three channels and two independent shutdown signals. This monitoring system is considered acceptable as was discussed with the licensee at the site visit on February 2-3, 1982. The system will be in use only during power operation, which is considered adequate, and the requirements for an explosion proof system are being met, although the system is of an explosion proof design. Therefore, we find this acceptable.

3.2.7 Solid Radwaste System

The objective of the RETS with regard to the solid radwaste system is to ensure that radwaste will be properly processed and packaged before shipped to the burial site. The licensee has proposed changes to the Technical Specifications to use the methods prescribed in the process control program (PCP) to ensure that the requirements of 10 CFR Part 20 and 10 CFR Part 71 are met prior to shipment of radwaste from the site.

3.2.8 Environmental Monitoring

The objectives of the RETS with regard to environmental monitoring are to ensure that an adequate and full-area-coverage environmental monitoring program exists and that the 10 CFR Part 50, Appendix I requirements for Technical Specifications on environmental monitoring are satisfied. The licensee has explicitly followed NUREG-0473, including the Branch Position dated November 1979. The licensee's methods of analysis and maintaining yearly records

satisfy the requirements and meet the intent of 10 CFR Part 50, Appendix I. The specification for the land-use census satisfies the requirements of Section 3.12.2 of NUREG-0473 by providing for the census once a year in the areas specified. The specification for interlaboratory comparison satisfies the requirements of Section 3.12.3 of NUREG-0473 by requiring the licensee to participate in the NRC approved program. Therefore, we find this acceptable.

3.29 Audits and Reviews

The objective of the RETS with regard to audits and reviews is to ensure that audits and reviews of the radwaste and environmental monitoring programs are properly conducted. The licensee's administrative structure designates the Operations Committee (OC) and the Safety Audit Committee (SAC) as the two groups responsible for the review and audit of the radiological environmental monitoring program, the ODCM, the PCP, and a quality assurance (QA) program meeting the requirements of ANSI N18.7-1972 rather than Regulatory Guide 4.15. The SAC is responsible for auditing these four programs, with the frequency of review to be equal to or greater than that required by NUREG-0473. The OC is responsible for reviewing every unplanned release of radioactive material. The review will include an event description, remedial action to prevent recurrence, and corrective action. The OC also reviews any changes in the ODCM and the PCP. Therefore, we find this acceptable.

3.2.10 Procedures

The objective of the RETS with regard to procedures is to establish a requirement for implementing the ODCM, the PCP, and the QA program. The licensee has proposed changes to the Technical Specifications that establish, implement, and maintain written procedures for the PCP, ODCM, and QA program. Therefore, we find the proposed changes to the Technical Specifications to be acceptable.

3.2.11 Reports

The objective of the RETS with regard to reports is to ensure that appropriate periodic and special reports are submitted to the NRC, and that these reports meet the requirements of 10 CFR Part 50.36a. The licensee has proposed changes to the Technical Specifications that require issuance of annual and semiannual reports (meeting the requirements of Sections 6.9.1.12, and 6.9.1.9, respectively, of NUREG-0473). The licensee, also proposed changes to the Technical Specifications to require the issuance of special reports (Prompt Notification) when (1) offsite releases exceed the limits specified under paragraphs 3.11.1.1 and 3.11.2.1 of NUREG-0473 and (2) inventory limits for the storage tanks have exceeded those listed under Section 3.11.1.4 and 3.11.2.6 of NUREG-0473. We have reviewed the proposed changes to the Technical Specifications and conclude that they meet the requirements stated in NUREG-0473 and thererfore, are acceptable.

3.3 Offsite Dose Calculation Manual (ODCM)

A brief discussion of the methodology and approach used by the licensee to calculate offsite dose and to maintain the operability of the effluent system is provided in this section. The methodology used by the licensee

is evaluated for consistency against the methodology and guidelines set by the NRC staff. As a minimum, it is required that the ODCM provide equations and methodology for the following topics:

- o alarm and trip setpoint on effluent instrumentation
- o liquid effluent concentration in unrestricted areas
- o gaseous effluent dose rate at or beyond the site boundary
- o liquid and gaseous effluent dose contributions
- o liquid and gaseous effluent dose projections
- o description and location of samples for the environmental monitoring program

The licensee meets these requirements. In addition, it has been suggested (but not required) that flow diagrams defining the treatment paths and the components of the radioactive liquid, gaseous, and solid waste management systems be included and reviewed for consistency against the system being used at the station. The licensee has provided, in the ODCM, a diagram of the gaseous radwaste treatment system and a description and location of samples in support of the environmental monitoring program.

3.3.1 Evaluation

The licensee has followed the methodology of NUREG-0133 and Regulatory Guide 1.109 to determine the alarm and trip setpoints for the liquid and gaseous effluent monitors. A conservative factor is used for the setpoints, which ensures that maximum permissible concentrations (MPC's) will not be exceeded. Continuous releases of liquid effluents do not occur at Monticello but if they do occur, the equations for calculating concentrations have been included.

The dose rate at or beyond the site boundary due to gaseous effluent release is in compliance with 10 CFR Part 20. Gaseous effluents are released from two release points for which conservative values of relative concentration and relative deposition for the average atmospheric dispersion conditions are used by the licensee.

The dose evaluation of pathways associated with the release of radioactive material in liquid effluents is stated to be in compliance with 10 CFR Part 50. The dose contributions are calculated once per 31 days.

Evaluation of noble gases released to the atomsphere includes both beta and gamma doses at the off-site locations with the highest calculated annual average relative concentrations of effluents from the reactor building vent (X/Q).

For radioiodine, tritium, and particulates, the licensee has shown that the method used in the ODCM for calculating releases to unrestricted areas will meet the design objective values of maintaining an annual dose or dose commitment not to exceed 15 mrem to any organ of the maximum exposed individual. The licensee has shown acceptable methods of calculating the dose

using the highest calculated annual average relative concentration of effluents released via plant vents for any area at or beyond the site boundary including the effects due to relative deposition for long-term release at the critical locations (X/Q and D/Q values).

The licensee has proposed changes to the Technical Specifications to perform dose projections for liquid effluent releases once every 31 days to determine the use of appropriate portions of the radwaste system except where systems are in operation at all times. Also, the licensee complies with 40 CFR Part 141 and 40 CFR Part 190 including dose from shine.

The licensee provides a complete description of sample locations in the ODCM Figures 5.1-1 through 5.1-3 and Table 5.1-1. This description is consistent with the sampling locations specified in the licensee's RETS. Table 5.1.-1 in the licensee's ODCM tabulates the site number identification, sector location, distance from station center, and sample point description. Table 5.1-1 covers all of the licensee's committed sampling exposure pathways in accordance with Table 3.12-1, RETS Environmental Monitoring program.

3.4 Summary of Technical Evaluation

Table 2 contains a correspondence between NUREG-0473 the current Technical Specifications and the licensee's proposal. The licensee's proposal was evaluated and the following conclusions were reached:

1. The licensee's proposed RETS meets the intent of the NRC staff's current standard, "Radiological Effluent Technical Specifications," NUREG-0473, Revision 2, February 1980.

2. The licensee's Offsite Dose Calculation Manual (ODCM) uses documented and approved methods that are consistent with our methodology in NUREG-0133. The ODCM is also consistent with the proposed Technical Specifications. On this basis, we find the licensee's ODCM acceptable.

3. The licensee submitted a Process Control Program (PCP) which was given a preliminary review. It consists of portions of the topical reports on the ATCOR and Chem Nuclear Services, Inc., waste solidification systems. The licensee requested a later inspection by the Office of Inspection and Enforcement.

3.5 Evaluation Conclusions

The proposed radiological effluent Technical Specifications for the Monticello Nuclear Generating Plant have been evaluated, reviewed and found to be in compliance with the express requirements of the NRC regulations and with the intent of NUREG-0133 and NUREG-0473 (Monticello is comprised of one boiling water reactor) and thereby, all the requirements of the regulations related to radiological effluent Technical Specifications have been fulfilled.

TABLE 2. CORRESPONDENCE OF PROVISIONS OF NUREG-0473, THE CURRENT TECHNICAL SPECIFICATIONS AND THE LICENSEE'S PROPOSAL

RETS Requirement	NUREG-0473 (Section)	Current Technical 1 Specification* (Section)	Licensee Proposal (Section)
Effluent Instrumentation	3.3.7.11 3.3.7.12	2.4.2.d, e, f 2.4.3.d, e, h, 2.4.4.d	4.8.A.1, a, b, 4.8.B.1, a, b, d
Concentrations	3.11.1.1 3.11.2.1	2.4.4.e 2.4.2.c	3.8.A.1.a 3.8.B.1
Off-site Doses	3.11.1.2 3.11.2.2 3.11.2.3 3.11.4	2.4.a, b 2.4.c, d, e	3.8.A.2.a, 3.8.B.2.a 3.8.B.3, 3.8.D
Effluent Treatment	3.11.1.3 3.11.2.4 3.11.2.5 3.11.2.8	2.4.1 2.4.3.e, f, g	3.8.A.3.a 3.8.B.4.a
Tank Inventory Limits	3.11.1.4	2.4.1.e 2.4.3.j	3.8.A.4.a 3.8.B.4.c
Condenser Air Ejector Release Rate	3.11.2.7	2.4.3.f	
Explosive Gas Mixtures	3.11.2.6		3.8.B.4.c
Solid Radwaste	3.11.3	2.4.5	3.8.C.1
Environmental Monitoring	3.12.1	4.8	4.16.A.1
Audit and Review	6.5.1 6.5.2	6.2.A	6.2.A
Procedures	6.8	6.5.A	6.5.A
Reports	6.9.1.6 6.9.1.10	6.7.C.1.a 6.7.C.1.b	6.7.C.1.b, c, d 6.7.A.3

* Existing Sections Affected by the Proposed RETS

The proposed changes will not remove nor relax any existing requirement related to the probability or consequences of accidents previously considered and do not involve a significant hazards consideration.

The proposed changes will not remove nor relax any existing requirement needed to provide reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

4.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 17, 1982

Principal Contributors:

F. Congel

6.0 References

1. Letter, L. O. Mayer (NSP) to Director, Office of Nuclear Reactor Regulation (USNRC), May 1, 1979 and revised January 24, 1980 (with attachments).

2. Letter, D. M. Musolf (NSP) to Director, Office of Nuclear Reactor Regulation (US NRC), July 23, 1982, (with attachments).

3. NUREG-0473 "Radiological Effluent Technical Specifications," Revision 2, February 1980.

4. NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants."

5. F. B. Simpson, letter of transmittal, Transmittal of Questions for the Monticello RETS Review - SIM-33-81, December 16, 1981.

6. Northern States Power Company, letter of transmittal, Revised Monticello RETS Submittal, November 25, 1981.

7. Monticello Plant visit, Review of Monticello Radiological Effluent Technical Specifications, February 2-3, 1982.

8. Northern States Power Company, letter of transmittal, Revised Monticello Nuclear Generating Station RETS Submittal, April 2, 1982.

9. H. Nicolaras (NRC), D. Musolf (NSP), and D. W. Akers (EG&G), telephone conference, May 19, 1982.

10. H. Nicolaras (NRC), F. Congel (NRC), D. Musolf (NSP), J. W. Mandler (EG&G), and D. W. Akers (EG&G), telephone conference, May 26, 1982.

11. Akers, D. W. and et.al., "Radiological Effluent Technical Specifications (RETS) Implementation-Monticello Nuclear Generating Station", Technical Evaluation Report, EGG-PHYS-5972, September 1982.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-263

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 15 to Facility Operating License No. DPR-22, issued to Northern States Power Company, which revised the Technical Specifications for operation of the Monticello Nuclear Generating Plant (the facility) located in Wright County, Minnesota. The amendment is effective on January 1, 1983.

The amendment authorizes changes to the Technical Specifications 1) to implement the requirements of Appendix I to 10 CFR Part 50, 2) to establish new limiting conditions for operation for the quarterly and annual average release rates, and 3) to revise environmental monitoring programs to assure conformance with the Commission's regulations.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License was published in the FEDERAL REGISTER on July 9, 1979 (44 FR 40163). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement or negative declaration

8301040075 821217
PDR ADOCK 05000263
P PDR

and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

For further details with respect to this action, see (1) the application for amendment dated May 1, 1979, as revised by letter dated July 23, 1982 (2) Amendment No.15 to License No. DPR-22, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W. Washington, D. C. and at the Environmental Conservation Library, Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 17th day of December, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing