November 2, 1984

Docket No. 50-220

Mr. B. G. Hooten
Executive Director, Nuclear Operations
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Dear Mr. Hooten:

The Commission has issued the enclosed Amendment No. 66 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your request dated February 1, 1984 and is effective January 1, 1985.

The revision to the Technical Specifications adds Limiting Conditions for Operation and surveillance requirements for Radiological Effluents. We conclude that your proposed Radiological Effluent Technical Specifications (RETS) meet the intent of our model RETS for BWRs, NUREG-0473, Revision 2, February 1, 1980, satisfies the requirements of Appendix I, 10 CFR 50 and, therefore, is acceptable.

A copy of the Safety Evaluation is also enclosed.

Further, we have reviewed your "Offsite Dose Calculation Manual" (ODCM) submitted March 7, 1984 and find it generally uses documented and approved methods that are consistent with the methodology and guidelines in NUREG-0133 and, therefore, is an acceptable reference.

Sincerely,

Original signed by/

Robert A. Hermann, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 66 to License No. DPR-63
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. B. G. Hooten
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station, Unit No. 1

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 66
License No. DPR-63

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

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated February 1, 1984, 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-63 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 66, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of January 1, 1985.

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: November 2, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 66

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

A. Revise the Appendix A Technical Specifications by removing and inserting the following pages:

Existing Page	Revised Page
iiia	iiia
iv	iv
v	v
4a	4a
-	4b
-	4c
186	186
224	224
-	241jj thru 241zz
-	241aaa thru 241zzz
-	241aaaa thru 241pppp
242	242
-	242a
-	242b
255	255
257 - 266	257 - 271

The revised areas are indicated by marginal lines.

B. Delete Appendix B in its entirety.

* to be submitted at a future date

Amendment No. 66

Amendment No. 66

1.16 Fire Suppression Water System

A Fire Suppression Water System shall consist of: a water supply system, fixed extinguishing systems of both automatic sprinklers and sprays, and manual fire fighting equipment consisting of standpipe risers with hose connections and hose reels.

1.17 Fire Watch Patrol

At least each hour, an area with inoperable Fire Protection Equipment shall be inspected for abnormal conditions.

1.18 Gaseous Radwaste Treatment System

A gaseous radwaste treatment system is any system designed and installed to reduce radioactive gaseous effluents by collecting main condenser offgas and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

1.19 Member(s) of the Public

Member(s) of the public shall include persons who are not occupationally associated with the Nine Mile Point Nuclear Station. This category does not include employees of Niagara Mohawk Power Corporation, the New York State Power Authority, its contractors or vendors who are occupationally associated with Nine Mile Point Unit 1. 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 Nine Mile Point Unit 1.

1.20 Milk Sampling Location

A milk sampling location is that location where 10 or more head of milk animals are available for the collection of milk samples.

1.21 Offsite Dose Calculation Manual (ODCM)

The Offsite Dose Calculational Manual shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

1.22 Process Control Program (PCP)

The process control program shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of radioactive wastes, based on demonstrated processing of actual or simulated wet or liquid wastes, will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 61, 10 CFR Part 71, and Federal and State regulations and other requirements governing the transport and disposal of radioactive waste.

1.23 Purge - Purging

Purge or 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. The purge is completed when the oxygen concentration exceeds 19.5 percent.

1.24 Site Boundary

The site boundary shall be that line around the Nine Mile Point Nuclear Station beyond which the land is neither owned, leased, nor otherwise controlled by Niagara Mohawk Power Corporation or the New York Power Authority.

1.25 Solidification

Solidification shall be the conversion of wet or liquid waste into a form that meets shipping and burial ground requirements.

1.26 Source Check

A source check shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

1.27 Unrestricted Area

The unrestricted area shall be any area at or beyond the site boundary access that is not controlled by Niagara Mohawk Power Corporation or the New York Power Authority 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 for industrial, commercial, institutional, and/or recreational purposes. That area outside the restricted area (10 CFR 20.3(a)(14)) but within the site boundary will be controlled by the owner as required.

1.28 Ventilation Exhaust Treatment System

A ventilation exhaust treatment system is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be ventilation exhaust treatment system components.

1.29 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 during venting. Vent, used in system names, does not imply a venting process.

LIMITING CONDITION FOR OPERATION

3.6.1 STATION PROCESS EFFLUENTS

a. Effluent release limits are described in Specification 3.6.15.

b. The mechanical vacuum pump line shall be capable of automatic isolation by closure of the air-operated valve upstream of the pumps. The signal to initiate isolation shall be from high radioactivity (five times normal) in the mainsteam line.

SURVEILLANCE REQUIREMENT

4.6.1 STATION PROCESS EFFLUENTS

a. Monitoring the radioative discharges from Nine Mile Point Unit 1 is described in Specification 4.6.15.

b. At least once during each operating cycle (prior to startup), verify automatic securing and isolation of the mechanical vacuum pump.

Table 3.6.2h

VACUUM PUMP ISOLATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels Per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
VACUUM PUMP							
High Radiation Main Steam Line	2	2	$\leq$ 5 times normal background		X	X	X

Amendment No. 66

3.6.14 RADIOACTIVE EFFLUENT INSTRUMENTATION

Applicability:

Applies to the operability of plant instrumentation that monitors plant effluents.

Objective:

To assure the operability of instrumentation to monitor the release of radiaoctive plant effluents.

Specification:

a. Liquid Effluent

The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.6.14-1 shall be operable with their alarm setpoints set to ensure that the limits of Specification 3.6.15.a.1 are not exceeded. The alarm setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

With a radioactive liquid effluent monitoring instrumentation channel alarm setpoint less conservative than a value which will ensure that the limits of 3.6.15.a.1 are met, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

4.6.14 RADIOACTIVE EFFLUENT INSTRUMENTATION

Applicability:

Applies to the surveillance of instrumentation that monitors plant effluents.

Objective:

To verify operation of monitoring instrumentation.

Specification:

a. Liquid Effluent

Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated operable by performance of the sensor check, source check instrument channel calibration and channel test operations at the frequencies shown in Table 4.6.14-1.

Records - Auditable records shall be maintained, in accordance with procedures in the Offsite Dose Calculation Manual, of all radioactive liquid effluent monitoring instrumentation alarm setpoints. Setpoints and setpoint calculations shall be available for review to ensure that the limits of Specification 3.6.15.a.1 are met.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.14 RADIOACTIVE EFFLUENT INSTRUMENTATION (Cont) a. Liquid Effluent (Continued) With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels operable, take the action shown in Table 3.6.14-1. Restore the instruments to OPERABLE status within 30 days, or outline in the next Semi-Annual Radioactive Effluent Release Report the cause of the inoperability and how the instruments were or will be restored to operable status.	

Table 3.6.14-1
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Limiting Condition for Operation

Instrument	Minimum Channels Operable	Applicability
1. Gross Radioactivity Monitors [(a)]		
A. Liquid Radwaste Effluent Line	1[(c)]	At all times [(b)]
B. Service Water System Effluent Line	1[(d)]	At all times
2. Flow Rate Measurement Devices		
A. Liquid Radwaste Effluent Line	1[(e)]	At all times
B. Discharge Canal	**	**
3. Tank Level Indicating Devices [(g)]		
A. Outside Liquid Radwaste Storage Tanks	1[(f)]	At all times

**Pumps curves or rated capacity will be utilized to estimate flow.

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NOTES FOR TABLE 3.6.14-1

(a) Provide alarm, but do not provide automatic termination of release.

(b) An operator shall be present in the Radwaste Control Room at all times during a release.

(c) With the number of channels operable less than required by the minimum channels operable requirement, effluent releases may continue provided that prior to initiating a release:

1. At least two independent samples are analyzed in accordance with Specification 4.6.15.a, and

2. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving.

Otherwise suspend release of radioactive effluents via this pathway.

(d) With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for gamma radioactivity at a lower limit of detection of at least 5×10^{-7} microcurie/ml.

(e) During discharge, with the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases.

(f) With the number of channels operable less than required by the minimum channels operable requirement, liquid additions to this tank may continue provided the tank liquid level is estimated during liquid additions to the tank.

(g) Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes or walls capable of holding the tank contents.

(h) With the number of channels operable less than required by the mimimum channels operable requirement, steam release via this pathway may commence or continue provided vent pipe radiation dose rates are monitored once per four hours.

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Table 4.6.14-1
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Surveillance Requirement

Instrument	Sensor Check	Source Check(f)	Channel Test	Channel Calibration
1. Gross Beta or Gamma Radioactivity Monitors				
a. Liquid Radwaste Effluent Line	Once/day*	Once/ discharge*	Once/3 months(a)*	Once/year (b)*
b. Service Water Effluent Line	Once/day	Once/month	Once/3 months(a)	Once/year(b)
2. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line	Once/day(c)	None	None	Once/year
b. Discharge Canal (d)	None	None	None	Once/year
3. Tank Level Indicating Devices (e)				
a. Outside Liquid Radwaste Storage Tanks	Once/day**	None	Once/3 months	Once/18 months

* Required prior to removal of blank flange in discharge line and until blank flange is replaced.
** During liquid addition to the tank.

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NOTES FOR TABLE 4.6.14-1

(a) The channel test shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

1. Instrumentation indicates measured levels above the alarm setpoint.

2. Instrument indicates a downscale failure.

3. Instrument controls not set in operate mode.

(b) The channel calibration shall be performed using one or more reference standards certified by the National Bureau of Standards or using standards that are traceable to the National Bureau of Standards or using actual samples of liquid waste that have been analyzed on a system that has been calibrated with National Bureau of Standard traceable sources. These standards shall permit calibrating the system over its intended range of energy and measurement.

(c) Sensor check shall consist of verifying indication of flow during periods of release. Sensor check shall be made at least once per 24 hours on days on which continuous, periodic or batch releases are made.

(d) Pump performance curves or rated data may be used to estimate flow.

(e) Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes or walls capable of holding the tank contents.

(f) Source check may consist of an installed check source, response to an external source, or (for liquid radwaste monitors) verification within 30 minutes of commencing discharge of monitor response to effluent.

3.6.14 RADIOACTIVE EFFLUENT INSTRUMENTATION (Cont)

b. Gaseous Process and Effluent

The radioactive gaseous process and effluent monitoring instrumentation channels shown in Table 3.6.14-2 shall be operable with their alarm setpoints set to ensure that the limits of Specification 3.6.15.b.1 are not exceeded. The alarm setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

With a radioactive gaseous process and effluent monitoring instrumentation channel alarm setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

With less than the minimum number of radioactive gaseous process and effluent monitoring instrumentation channels operable, take the action shown in Table 3.6.14-2. Restore the instruments to OPERABLE status within 30 days or outline in the next Semi-Annual Radioactive Effluent Release Report the cause of the inoperability and how the instruments were or will be restored to operable status.

4.6.14 RADIOACTIVE EFFLUENT INSTRUMENTATION (Cont)

b. Gaseous Process and Effluent

Each radioactive gaseous process and effluent monitoring instrumentation channel shall be demonstrated operable by performance of the sensor check, source check, instrument channel calibration and instrument channel test operations at the frequencies shown in Table 4.6.14-2.

Auditable records shall be maintained of the calculations made, in accordance with procedures in the Offsite Dose Calculation Manual, of radioactive gaseous process and effluent monitoring instrumentation alarm setpoints. Setpoints and setpoint calculations shall be available for review to ensure that the limits of Specification 3.6.15.b.1 are met.

Table 3.6.14-2

RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION

Limiting Condition for Operation

Instrument	Minimum Channels Operable	Applicability	Parameter
1. Stack Effluent Monitoring			
a. Noble Gas Activity Monitor	1(a)	*	Radioactivity Rate Measurement
b. Iodine Sampler Cartridge	1(b)	*	Verify presence of cartridge
c. Particulate Sampler Filter	1(b)	*	Verify presence of filter
d. Sample Flow Rate Measuring Device	1(c)	*	Sampler flow rate measurement
e. Stack Gas Flow Rate Measuring Device	1(d)	*	Effluent flow rate measurement
2. Main Condenser Offgas Treatment Explosive Gas Monitoring System			
a. Hydrogen Monitor (f)	1(e)	**	Hydrogen

* Required prior to removal of blank flange in discharge line and until blank flange is replaced.
** During liquid addition to the tank.

Table 3.6.14-2
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION (Continued)

Instrument	Minimum Channels Operable	Applicability	Parameter
3. Condenser Air Ejector Radioactivity Monitor (Recombiner discharge or air ejector discharge)			
a. Noble Gas Activity	1(g)	***	Noble gas radioactivity rate measurement
b. Offgas System Flow Rate Measuring Devices	1(c)	***	System flow rate measurement
c. Sampler Flow Rate Measuring Devices	1(c)	***	Sampler flow rate measurement
4. Emergency Condenser System			
a. Noble Gas Activity Monitor	1 per vent(h)	****	Noble gas radioactivity rate measurement

*** During operation of the main condenser air ejector
**** During reactor power operationg condition

NOTES FOR TABLE 3.6.14-2

(a) With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided grab samples are taken once per 12 hours and these samples are analyzed for gross activity within 24 hours.

(b) With the number of channels operable less than required by the minimum channels operable requirements, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment in accordance with the requirements of Table 4.6.15-2.

(c) With the number of channels operable less than required by the minimum channels operable requirements, effluent releases via this pathway may continue provided the flow rate is estimated once per 8 hours.

(d) Stack gas flow rate may be estimated by exhaust fan operating configuration.

(e) With the number of channels operable less than required by the minimum channels operable requirement, operation of the main condenser offgas treatment system may continue provided gas samples are collected and analyzed once per 8 hours.

(f) One monitor on each recombiner. The system is designed to withstand the effects of a hydrogen explosion.

(g) With the number of channels operable less than required by the minimum channels operable requirement, gases from the main condenser offgas treatment system may be released provided:

1. Offgas grab samples are collected and analyzed once per 12 hours.

2. The stack monitor is operable.

3. Otherwise, be in at least hot shutdown within 12 hours.

(h) With the number of channels operable less than required by the minimum channels operable requirements, steam release via this pathway may commence or continue provided vent pipe radiation dose rates are monitored once per four hours.

Table 4.6.14-2
RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENT

Surveillance Requirements

Instrument	Sensor Check	Source Check	Channel Test	Channel Calibration
1. Stack Effluent Monitoring System				
a. Noble Gas Activity Monitor	Once/day (a)	Once/month	Once/3 months(g)	Once/year(b)
b. Iodine Sampler Cartridge	None	None	None	None
c. Particulate Sampler Filter	None	None	None	None
d. Sampler Flow Rate Measuring Device	Once/day (a)	None	None	Once/year
e. Stack Gas Flow Rate Measuring Device	Once/day	None	None	Once/year
2. Main Condenser Offgas Treatment System Explosive Gas Monitoring system (for system designed to withstand the effects of a hydrogen explosion)				
a. Hydrogen Monitor	Once/day (d)	None	Once/month	Once/3 months(e)
3. Condenser Air Ejector Radioactivity Monitor (Recombiner Discharge or Air Ejector Discharge)				
a. Noble Gas Activity Monitor	Once/day (f)	Once/month	Once/operating cycle(c)	Once/year(b)
b. Flow Rate Monitor	Once/day (f)	None	None	Once/year
c. Sampler Flow Rate Monitor	Once/day (f)	None	None	Once/year
4. Emergency Condenser System				
a. Noble Gas Activity Monitor	Once/day(h)	Once/month	Once/3 months(g)	Once/operating cycle(b)

NOTES FOR TABLE 4.6.14-2

(a) At all times.

(b) The channel calibration shall be performed using one or more of the reference standards certified by the National Bureau of Standards, standards that are traceable to the National Bureau of Standards or using actual samples of gaseous effluent that have been analyzed on a system that has been calibrated with National Bureau of Standards traceable sources. These standards shall permit calibrating the system over its intended range of energy and measurement.

(c) The channel function test shall demonstrate that control room alarm annunciation occurs if either of the following conditions exist:

1) Instrument indicates measured levels above the Hi or Hi Hi alarm setpoint.

2) Instrument indicates a downscale failure.

The channel function test shall also demonstrate that automatic isolation of this pathway occurs if either of the following conditions exist:

1) Instruments indicate two channels above Hi Hi alarm setpoint.

2) Instruments indicate one channel above Hi Hi alarm setpoint and one channel downscale.

(d) During main condenser offgas treatment system operation.

(e) The channel calibration shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen.

2. Four volume percent hydrogen, balance nitrogen.

(f) During operation of the main condenser air ejector.

(g) The channel test shall produce upscale and downscale annunciation.

(h) During reactor power operating condition

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BASES FOR RADIOACTIVE EFFLUENT INSTRUMENTATION 3.6.14 and 4.6.14

The radioactive liquid and gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid and gaseous effluents during actual or potential releases of liquid and gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual to ensure that the alarm/trip will occur prior to exceeding the limits of 10CFR Part 20. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the main condenser offgas treatment system. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10CFR Part 50. The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to unrestricted areas.

LIMITING CONDITION FOR OPERATION

3.6.15 RADIOACTIVE EFFLUENTS

Applicability:

Applies to the radioactive effluents from the station.

Objective:

To assure that radioactive material is not released to the environment in any uncontrolled manner and is within the limits of 10CFR20 and 10CFR50 Appendix I.

Specification:

a. Liquid

(1) Concentration

The concentration of radioactive material released in liquid effluents to unrestricted areas shall be limited to the concentrations specified in 10CFR 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} microcuries/ml total activity.

Should the concentration of radioactive material released in liquid effluents to unrestricted areas exceed the above limits, restore the concentration to within the above limits immediately.

SURVEILLANCE REQUIREMENT

4.6.15 RADIOACTIVE EFFLUENTS

Applicability:

Applies to the periodic test and recording requirements of the station process effluents.

Objective:

To ascertain that radioactive effluents from the station are within allowable values of 10CFR20, Appendix B and 10CFR50, Appendix I.

Specification:

a. Liquid

(1) Concentration

Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.6.15-1.

The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the Offsite Dose Calculation Manual to assure that the concentrations at the point of release are maintained within the limits of Specification 3.6.15.a.(1)

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3.6.15 RADIOACTIVE EFFLUENTS (Continued)

a. Liquid (Continued)

(2) Dose

The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released, from each reactor unit, to unrestricted areas (see Figures 5.1-1) shall be limited:

(a) During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

(b) During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3 a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

4.6.15 RADIOACTIVE EFFLUENTS (Continued)

a. Liquid (Continued)

(2) Dose

Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the Offsite Dose Calculation Manual, prior to each release of a batch of liquid waste.

Table 4.6.15-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Surveillance Requirement

Liquid Release Type	Minimum Sampling Frequency	Analysis Frequency	Type of Activity Analysis	Lower Limit[a] of Detection (LLD) (uCi/ml)
A. Batch Waste [b] Tanks	* Each Batch	* Each Batch	Principal Gamma [c] Emitters	5×10^{-7}
			I-131	1×10^{-6}
	* Each Batch[d]	* Each Batch [d]	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	* Each Batch	Monthly Composite[e]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	* Each Batch	Quarterly Composite[e]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Service Water System Effluent	Once/month[f]	Once/month[f]	Principal Gamma [c] Emitters	5×10^{-7}
			I-131	1×10^{-6}
			Dissolved and Entrained Gases	1×10^{-5}
			H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Once/quarter[f]	Once/quarter[f]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

* Completed prior to each release.

NOTES FOR TABLE 4.6.15-1

(a) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count above system background that will be detected with 95 percent probability with only 5 percent probability of falsely concluding that a blank observation represents a "real" signal. For a particular measurement system which may include radiochemical separation:

$$LLD = \frac{4.66\ S_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries 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,

E is the counting efficiency, as counts per disintegration,

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

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

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

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

Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is defined as a before the fact limit representing the capability of a measurement system and not as an after the fact for a particular measurement.

NOTES FOR TABLE 4.6.15-1 (Continued)

(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) The principal gamma emitters for which the LLD specification applies exclusively are 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 considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semi-Annual Radioactive Effluent Release Report

(d) If more than one batch is released in a calendar month, only one batch need be sampled and analyzed during that month.

(e) 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 that is representative of the liquids released.

(f) If the alarm setpoint of the service water effluent monitor, as determined by the method presented in the Offsite Dose Calculation Manual, is exceeded, the frequency of sampling shall be increased to daily until the condition no longer exists. Frequency of analysis shall be increased to daily for principal gamma emitters (including dissolved and entrained gases) and an incident composite for H-3, gross alpha, Sr-89, Sr-90 and Fe-55.

3.6.15 RADIOACTIVE EFFLUENTS (Continued)

b. Gaseous

(1) Dose Rate

The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the site boundary shall be limited to the following:

(a) For noble gases: Less than or equal to 500 mrems/year to the total body and less than or equal to 3000 mrems/year to the skin, and

(b) For iodine-131, iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/year to any organ.

With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

4.6.15 RADIOACTIVE EFFLUENTS (Continued)

b. Gaseous

(1) Dose Rate

The dose rate due to noble gases in gaseous effluents shall be determined to be within the limits of Specification 3.6.15 in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

The dose rate due to iodine-131, iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the limits of Specification 3.6.15 in accordance with methodology and parameters in the Offsite Dose Calculation Manual by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.6.15-2.

LIMITING CONDITION FOR OPERATION

3.6.15 RADIOACTIVE EFFLUENTS (Continued)

b. Gaseous (Continued)

(2) Air Dose

The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas at and beyond the site boundary shall be limited to the following:

(a) During any calendar quarter: Less than or equal to 5 milliroentgen for gamma radiation and less than or equal to 10 mrads for beta radiation and,

(b) During any calendar year: Less than or equal to 10 milliroentgen for gamma radiation and less than or equal to 20 mrads for beta radiation.

With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

SURVEILLANCE REQUIREMENT

4.6.15 RADIOACTIVE EFFLUENTS (Continued)

b. Gaseous (Continued)

(2) Air Dose

Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined monthly in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

3.6.15 RADIOACTIVE EFFLUENTS (Continued)

b. Gaseous (Continued)

(3) Tritium, Iodines and Particulates

The dose to a member of the public from iodine-131, iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released, from each reactor unit, to areas at and beyond the site boundary shall be limited to the following:

(a) During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

(b) During any calendar year: Less than or equal to 15 mrems to any organ.

4.6.15 RADIOACTIVE EFFLUENTS (Continued)

b. Gaseous (Continued)

(3) Tritium, Iodines and Particulates

Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days shall be determined monthly in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.15 RADIOACTIVE EFFLUENTS (Continued) b. Gaseous (Continued) With the calculated dose from the release of iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.	

Table 4.6.15-2
RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Surveillance Requirements

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit[a] of Detection (LLD) (uCi/ml)
A. Containment Purge[b]	Each Purge	Prior to each release	Principal Gamma Emmiters[c]	1×10^{-4}
	Grab Sample	Each Purge	Principal Gamma Emitters[c]	1×10^{-4}
			H-3	1×10^{-6}
B. Stack	Once/Month[d]	Once/Month[d]	Principal Gamma Emitters[c]	1×10^{-4}
	Once/Month[h]	Once/Month	H-3	1×10^{-6}
C. Stack	Continuous[e]	Once/Week[f] Charcoal Sample	I-131	1×10^{-12}
	Continuous[e]	Once/Week[f] Particulate Sample	Principal Gamma Emitters[c]	1×10^{-11}
	Continuous[e]	Once/Month Composite Particulate Sample	Gross alpha Sr-89, Sr-90	1×10^{-11}
	Continuous[e]	Noble gas monitor	Noble Gases, Gross Gamma or Principal Gamma Emitters[c]	1×10^{-6}[g]

NOTES FOR TABLE 4.6.15-2

(a) The LLD is defined in notation (a) of Table 4.6.15-1.

(b) Purge is defined in Section 1.23.

(c) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, 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, I-131 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semi-Annual Radioactive Effluent Release Report pursuant to Specification 6.9.1.

(d) Sampling and analysis shall also be performed following shutdown, startup or an increase on the recombiner discharge monitor of greater than 50 percent, factoring out increases due to changes in thermal power level or dilution flow; or when the stack release rate is in excess of 1000 uCi/second and steady-state gaseous release rate increases by 50 percent.

(e) The sample flow rate and the stack flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.6.15.b.(1).(b) and 3.6.15.b.(3).

(f) When the release rate is in excess of 1000 uCi/sec and steady state gaseous release rate increases by 50 percent. The iodine and particulate collection device shall be removed and analyzed to determine the changes in iodine-131 and particulate release rate. The analysis shall be done daily following each change until it is shown that a pattern exists which can be used to predict the release rate; after which it may revert to weekly sampling frequency. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

(g) When RAGEMS is inoperable the LLD for noble gas gross gamma analysis shall be 1×10^{-4}.

(h) Tritium grab samples shall be taken weekly from the station ventilation exhaust (stack) when fuel is offloaded until stable tritium release levels can be demonstrated.

LIMITING CONDITION FOR OPERATION

3.6.15 RADIOACTIVE EFFLUENTS (Continued)

c. Main Condenser

The gross radioactivity (beta and/or gamma) rate of noble gases measured at the recombiner discharge shall be limited to less than or equal to 500,000 uCi/sec. This limit can be raised to 1 Ci/sec. for a period not to exceed 60 days provided the offgas treatment system is in operation.

With the gross radioactivity (beta and/or gamma) rate of noble gases at the recombiner discharge exceeding the above limits, restore the gross radioactivity rate to within its limit within 72 hours or be in at least Hot Shutdown within the next 12 hours.

SURVEILLANCE REQUIREMENT

4.6.15 RADIOACTIVE EFFLUENTS (Continued)

c. Main Condenser

The radioactivity rate of noble gases at the recombiner discharge shall be continuously monitored in accordance with Table 3.6.14-2.

The gross radioactivity (beta and/or gamma) rate of noble gases from the recombiner discharge shall be determined to be within the limits of Specification 3.6.15 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the recombiner discharge:

Monthly.

Within 4 hours following an increase on the recombiner discharge monitor of greater than 50%, factoring out increases due to changes in thermal power level and dilution flow changes.

3.6.15 RADIOACTIVE EFFLUENTS (Continued)

d. Uranium Fuel Cycle

The annual (calendar year) dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources from Nine Mile Point Unit 1 shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

With the calculated doses from the release of radiactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.6.15.a.2(b), 3.6.15.b.2(b) and 3.6.15.b.3(b), calculations shall be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above listed 40CFR190 limits have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.3, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the

4.6.15 RADIOACTIVE EFFLUENTS (Continued)

d. Uranium Fuel Cycle

Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.6.15.a.(2), 4.6.15.b.(2) and 4.6.15.b.(3) and in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the Offsite Dose Calculation Manual. This requirement is applicable only under conditions set forth in Specification 3.6.15.d.

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

3.6.15 RADIOACTIVE EFFLUENTS (Continued)

d. Uranium Fuel Cycle (Continued)

calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40CFR 190. Submittal of the report is considered a timely request and a variance is granted until staff action on the request is complete.

LIQUID CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to unrestricted areas will be less than the concentration levels specified in 10CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in unrestricted areas will result in exposures within (1) the Section II.A design objectives of Appendix I, 10CFR Part 50, to a member of the public and (2) the limits of 10CFR Part 20.106 (e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its maximum permissable concentration in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40. 586-93 (1968), and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

Liquid Dose

This specification is provided to implement the requirements of Section II.A, III.A and IV.A of Appendix I, 10CFR Part 50. The Limiting Conditions for Operation expressed as quarter and annual limits are set at those values found in Section II.A. of Appendix I, in accordance with Section IV.A. The Limiting Conditions for Operation 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 to unrestricted areas will be kept "as low as is reasonably achievable." There are no drinking water supplies that can be potentially affected by plant operations. The dose calculation methodology and parameters in the Offsite Dose Calculation Manual implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculation procedures based on models and data, such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the Offsite Dose Calculation Manual for calculating the doses due to the actual release rates of radioactiave materials in liquid effluents are consistent with the methodology provided in Reguatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR 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," April 1977.

Gaseous Dose Rate

This specification is provided to ensure that the dose at any time at and beyond the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10CFR Part 20 to unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a member of the public in an unrestricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10CFR Part 20 (10CFR Part 20.106(b)). For members of the public who may at times be within the site boundary, the occupancy of that member of the public will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a member of the public at or beyond the site boundary to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

Dose - Noble Gases

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10CFR Part 50. The Limiting Condition for Operation expressed as quarter and annual limits are set at those values found in Section II.B of Appendix I in accordance with the guidance of Section IV.A. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV-A of Appendix I to assure that the releases of radioactive material in gaseous effluents to unrestricted areas will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conform with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the Offsite Dose Calculation Manual for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are 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 10CFR 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 Offsite Dose Calculation Manual equations provided to determine the air doses at and beyond the site boundary are based upon the historical average atmospheric conditions.

Dose - Iodine-131, Iodine-133, Tritium and Radionuclides in Particulate Form

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10CFR Part 50. The Limiting Conditions for Operation expressed as quarter and annual limits are set at those valves found in Section II.C of Appendix I in accordance with the guidance of Section IV.A. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to unrestricted areas will be kept "as low as is reasonably achievable." The Offsite Dose Calculation Manual calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conform with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The Offsite Dose Calculation Manual calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are 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 10CFR 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. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the site boundary. The pathways that were examined in the development of these calculations were: 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.

Main Condenser

Restricting the gross radioactivity rate of noble gases from the main condenser provides assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a very small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50. The primary purpose of providing this specification is to limit buildup of fission product activity within the station systems which would result if high fuel leakage were to be permitted over extended periods.

Total Dose - Uranium Fuel Cycle

This specification is provided to meet the dose limitations of 40CFR Part 190 that have been incorporated into 10CFR Part 20 by 46FR 182525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to a member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contribution from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40CFR Part 190, the Special Report with a request for variance (provided the release conditions resulting in violation of 40CFR Part 190 have not already been corrected), in accordance with the provisions of 40CFR Part 190.11 and 10CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40CFR Part 190 and does not apply in any way to the other requirements for dose limitation of 10CFR Part 20, as addressed in Specification 3.6.15.a.(1) and 3.6.15.b.(1). An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

LIMITING CONDITION FOR OPERATION

3.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS

Applicability:

Applies to the operating status of the liquid, gaseous and solid effluent treatment systems.

Objective:

To assure operability of the liquid, gaseous and solid effluent treatment system.

Specification:

a. Liquid

The liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge as necessary to meet the requirements of Specification 3.6.15.

b. Gaseous

The gaseous radwaste treatment system shall be operable. The gaseous radwaste treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge as necessary to meet the requirements of Specification 3.6.15.

With gaseous radwaste from the main condenser air ejector system being discharged without treatment for more than 7 days, prepare and submit to the Commission within 30 days, pursuant to

SURVEILLANCE REQUIREMENT

4.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS

Applicability:

Applies to the surveillance requirements for the liquid, gaseous and solid effluent treatment systems.

Objective:

To verify operability of the liquid, gaseous and solid effluent treatment system.

Specification:

a. Liquid

Doses due to liquid releases to unrestricted areas shall be projected prior to the release of each batch of liquid radioactive waste in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

b. Gaseous

Doses due to gaseous releases to areas at and beyond the site boundary shall be calculated monthly in accordance with the methodology and parameters in the Offsite Dose Calculation Manual.

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3.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS (Cont)

b. Gaseous (Continued)

Specification 6.9.3, Special Report that identifies the inoperable equipment and the reason for its inoperability, actions taken to restore the inoperable equipment to OPERABLE status, and a summary description of those actions taken to prevent a recurrence.

c. Solid

The solid radwaste system shall be used in accordance with a Process Control Program to process wet radioactive wastes to meet shipping and burial ground requirements.

With the provisions of the process control program not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

4.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS (Cont)

c. Solid

The process control program 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 and evaporator bottoms).

(1) If any test specimen fails to verify solidification, the solidification of the batch may then be resumed using the alternative solidification parameters determined by the process control program.

(2) If the initial test specimen from a batch of waste fails to verify solidification, the process control program 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 3 consecutive initial test specimens demonstrate solidification.

BASES FOR 3.6.16 AND 4.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS

Liquid Radwaste Treatment System

The requirement that the appropriate portions of this system be used 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 10CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10CFR Part 50 and the design objective given in Section II.D of Appendix I to 10CFR Part 50.

Gaseous Radwaste Treatment System

The requirement that this system be used 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 10CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10CFR Part 50 and the design objectives given in Section II.D of Appendix I to 10CFR Part 50. Since the capability exists to operate within specification without use of the system, it is conceivable that due to unforeseen circumstances, limited operation without the system may be made sometime during the life of the plant.

Solid Radioactive Waste

This specification implements the requirements of 10CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10CFR part 50. The process parameters included in establishing the process control program may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constitutents and mixing and curing times.

3.6.17 EXPLOSIVE GAS MIXTURE

Applicability:

Applies to the operability of instrumentation to monitor hydrogen concentration in the main condenser off-gas treatment system.

Objective:

To assure the operability of the hydrogen monitoring instrumentation in the main condenser off-gas treatment system.

Specification:

The concentration of hydrogen in the main condenser off-gas treatment system shall be limited to 4 percent by volume.

If the concentration of hydrogen in the main condenser off-gas treatment system exceeds this limit, restore the concentration to within the limit within 48 hours.

4.6.17 EXPLOSIVE GAS MIXTURE

Applicability:

Applies to the surveillance of instrumentation that monitors hydrogen concentration in the main condenser off-gas treatment system.

Objective:

To verify operation of monitoring instrumentation.

Specification:

The concentration of hydrogen in the main condenser off-gas treatment system shall be determined to be within the above limits by continuously monitoring the waste gases in the main condenser off-gas treatment system in accordance with Table 3.6.14-2 of Specification 3.6.14.

BASES FOR 3.6.17 AND 4.6.17 EXPLOSIVE GAS MIXTURE

This specification 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. Automatic control features are included in the system to prevent the hydrogen concentration from reaching these flammability limits. Maintaining the concentration of hydrogen below flammability limits 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 10CFR Part 50.

3.6.18 MARK I CONTAINMENT

Applicability:

Applies to the venting/purging of the Mark I Containment.

Objective:

To assure that the Mark I Containment is vented/purged so that the limits of specifications 3.15.b.1 and 3.6.15.b.3 are met.

Specification:

The Mark I Containment drywell shall be vented/purged through the Emergency Ventilation System unless Specification 3.6.15.b.1 and 3.6.15.b.3 can be met without use of the Emergency Ventilation System.

If these requirements are not satisfied, suspend all venting/purging of the drywell.

4.6.18 MARK I CONTAINMENT

Applicability:

Applies to the surveillance requirement for venting and purging of the Mark I Containment when required to be vented/purged through the Emergency Ventilation System.

Objective:

To verify that the Mark I Containment is vented through the Emergency Ventilation System when required.

Specification:

The containment drywell shall be determined to be aligned for venting/purging through the Emergency Ventilation System within four hours prior to start of and at least once per 12 hours during venting/purging of the drywell.

BASES FOR 3.6.18 AND 4.6.18 MARK I CONTAINMENT

This specification provides reasonable assurance that releases from drywell purging operations will not exceed the annual dose limits of 10CFR Part 20 for unrestricted areas.

LIMITING CONDITION FOR OPERATION

3.6.19 LIQUID WASTE HOLDUP TANKS*

Applicability:

Applies to the quantity of radioactive material that may be stored in an outdoor liquid waste holdup tank.

Objective:

To assure that the quantity of radioactive material stored in outdoor holdup tanks does not exceed a specified level.

Specification:

The quantity of radioactive material contained in an outdoor liquid waste tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

With the quantity of radioactive material in any such tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank. Within 48 hours reduce the tank contents to within the limit and describe the events leading to this condition in the next Semi-Annual Radioactive Effluent Release Report.

*Tanks included in this Specification are those outdoor tanks that are not surrounded by liners, dikes or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

SURVEILLANCE REQUIREMENT

4.6.19 LIQUID WASTE HOLDUP TANKS

Applicability:

Applies to the surveillance requirements for outdoor liquid waste holdup tanks.

Objective:

To verify the quantity of radioactive material stored in an outdoor liquid waste holdup tank.

Specification:

The quantity of radioactive material contained in each of the tanks listed in Specification 3.6.19 shall be determined to be within the limit of Specification 3.6.19 by analyzing a representative sample of the tank's contents at least weekly when radioactive materials are being added to the tank.

BASES FOR 3.6.19 AND 4.6.19 LIQUID HOLDUP TANKS

This specification applies to any outdoor tank that is not surrounded by liners, dikes or walls capable of holding the tank contents and that does not have tank overflows and surrounding areas drains connected to the liquid radwaste treatment system.

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 10CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

LIMITING CONDITION FOR OPERATION

3.6.20 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Applicability:

Applies to radiological samples of station environs.

Objective:

To evaluate the effects of station operations and radioactive effluent releases on the environs and to verify the effectiveness of the controls on radioactive material sources.

Specification:

The radiological environmental monitoring program shall be conducted as specified in Table 3.6.20-1.

With the radiological environmental monitoring program not being conducted as specified in Table 3.6.20-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

Deviations are permitted from the required sample schedule if samples are unobtainable due to hazardous conditions, seasonal unavailability, theft, uncooperative residents or to malfunction of automatic sampling equipment. In the event of the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.

SURVEILLANCE REQUIREMENT

4.6.20 RADIOLOGICAL ENVIRONMENTAL MONITIORING PROGRAM

Applicability:

Applies to the periodic sampling and monitoring requirements of the radiological environmental monitoring program.

Objective:

To ascertain what effect station operations and radioactive effluent releases have had upon the environment.

Specification:

The radiological environmental monitoring samples shall be collected pursuant to Table 3.6.20-1 from the specific locations given in the table and figure(s) in the Offsite Dose Calculation Manual and shall be analyzed pursuant to the requirements of Table 3.6.20-1 and the detection capabilities required by Table 4.6.20-1.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.20 <u>RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (Continued)</u>

<u>Specification</u>: (Continued)

With the level of radioactivity (as the result of plant effluents), in an environmental sampling medium exceeding the reporting levels of Table 6.9.3-1 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Special Report pursuant to Specification 6.9.3. The Special Report shall identify the cause(s) for exceeding the limit(s) and define the corrective action(s) to be taken to reduce radioactive effluents so that the potential annual dose to a member of the public is less than the calendar year limits of Specifications 3.6.15.a.(2), 3.6.15.b.(2) and 3.6.15.b.(3). When more than one of the radionuclides in Table 6.9.3-1 are detected in the sampling medium, this report shall be submitted if:

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

When radionuclides other than those in Table 6.9.3-1 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 Specification 3.6.15.a.(2), 3.6.15.b.(2) and 3.6.15.b.(3).

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LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.20 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (Continued) Specification: (Continued) 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 Radiological Environmental Operating Report. With milk or fruit and/or vegetables no longer available at one or more of the sample locations specified in Table 3.6.20-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semi-Annual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the Offsite Dose Calculation Manual reflecting the new location(s).	

Table 3.6.20-1

OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Limiting Condition for Operation

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency (a)	Type of Analysis and Frequency
Radioiodine & Particulates	Samples from 5 locations: 1) 3 samples from off-site locations in different sectors of the highest calculated site average D/Q (based on all site licensed reactors) 2) 1 sample from the vicinity of an established year round community having the highest calculated site average D/Q (based on all site licensed reactors) 3) 1 sample from a control location 10-17 miles distant and in a least prevalent wind direction [d]	Continuous sampler operation with sample collection weekly or as required by dust loading, whichever is more frequent	Radioiodine Canisters analyze once/week for I-131. Particulate Samplers Gross beta radioactivity following filter change,[b] composite (by location) for gamma isotopic analysis [c] once per 3 months, (as a minimum)

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency(a)	Type of Analysis and Frequency
Direct Radiation [e]	32 stations with two or more dosimeters to be placed as follows: an inner ring of stations in the general area of the site boundary and an outer ring in the 4 to 5 mile range from the site with a station in each land based sector.* The balance of the stations should be placed in special interest areas such as population centers, nearby residences, schools and in 2 or 3 areas to serve as control stations.	Once per 3 months	Gamma dose once per 3 months
WATERBORNE			
Surface [f]	1) 1 sample upstream 2) 1 sample from the site's downstream cooling water intake	Composite sample over 1 month period [g]	Gamma isotopic analysis[c] once/month. Composite for once per 3 months tritium analysis.
Sediment from Shoreline	1 sample from a downstream area with existing or potential recreational value	Twice per year	Gamma isotopic analysis[c]

* At this distance, 8 wind rose sectors are over Lake Ontario.

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Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency(a)	Type of Analysis and Frequency
INGESTION			
Milk	1) Samples from milk sampling locations in 3 locations within 3.5 miles distance having the highest calculated site average D/Q. If there are none, then 1 sample from milking animals in each of 3 areas 3.5-5.0 miles distant having the highest calculated site average D/Q (based on all site licensed reactors)	Twice per month, April-December (samples will be collected in January-March if I-131 is detected in November and December of the preceding year)	Gamma isotopic[c] and I-131 analysis twice per month when animals are on pasture (April-December); once/month at other times (January-March) if required
	2) 1 sample from a milk sampling location at a control location (9-20 miles distant and in a least prevalent wind direction)[d]		
Fish	1) 2 samples of commercially or recreationally important species in the vicinity of a site discharge point[h]	Twice per year	Gamma isotopic analysis[c] on edible portions twice per year
	2) 1 sample each of the same species (or of a species with similar feeding habits) from an area at least 5 miles distant from the site.[d]		

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency[a]	Type of Analysis and Frequency
Food Products	1) 6 samples total (utilizing at least 2 sectors) of fruits and/or vegetables will be collected from available off-site locations of highest calculated site average D/Q (based on all licensed site reactors) 2) 1 sample of each of similar vegetation grown 9-20 miles distant in a less prevalent wind direction	Once per year during harvest season	Gamma isotopic analysis of edible portions (isotopic to include I-131) Once during the harvest season

NOTES FOR TABLE 3.6.20-1

(a) It is recognized that, at times, it may not be possible or practical to obtain samples of the media of choice at the most desired location or time. In these instances, suitable alternative media and locations may be chosen for the particular pathway in question and may be substituted. Actual locations (distance and directions) from the site shall be provided in the Annual Radiological Environmental Operating Report. Highest D/Q locations are based on historical meteorological data for all site licensed reactors.

(b) Particulate sample filters should be analyzed for gross beta 24 hours or more after sampling to allow for radon and thoron daughter decay. If the gross beta activity in air is greater than 10 times a historical yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

(c) Gamma isotopic analysis means the identification and quantification of gamma emitting radionuclides that may be attributable to the effluents from the facility.

(d) The purpose of these samples is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites, such as historical control locations which provide valid background data may be substituted.

(e) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purpose of this table, a thermoluminescent dosimeter may be considered to be one phosphor and two or more phosphors in a packet may be considered as two or more dosimeters. Film badges shall not be used for measuring direct radiation.

(f) The "upstream sample" should be taken at a distance beyond significant influence of the dischage. The "downstream sample" should be taken in an area beyond but near the mixing zone, if possible.

(g) Composite samples should be collected with equipment (or equivalent) which is capable of collecting an aliquot at time intervals which are very short (e.g. hourly) relative to the compositing period (e.g. monthly) in order to assure obtaining a representative sample.

(h) In the event commercial or recreational important species are not available as a result of three attempts, then other species may be utilized as available.

TABLE 4.6.20-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a, b]

LOWER LIMIT OF DETECTION LLD (c)

Surveillance Requirement

Analysis	Water (c) (pCi/l)	Airborne Particulate or Gases (pCi/m)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	0.01				
H-3	3000					
Mn-54	15		130			
Fe-59	30		260			
Co-58, Co-60	15		130			
Zn-65	30		260			
Zr-95, Nb-95	15					
I-131	(d)	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba/La-140	15			15		

NOTES FOR TABLE 4.6.20-1

(a) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.d.

(b) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in ANSI N.545 (1975), Section 4.3. Allowable exceptions to ANSI N.545 (1975), Section 4.3 are contained in the Nine Mile Point Unit 1 Offsite Dose Calculation Manual (ODCM).

(c) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95 percent probability with only 5 percent 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 picocuries 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.

E is the counting efficiency, as counts per disintegration,

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

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, where applicable.

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

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period and time of counting

Typical values of E, V, Y and Δt should be used in the calculation.

NOTES FOR TABLE 4.6.20-1

It should be recognized that the LLD is defined as a before the fact limit representing the capability of a measurement system and not as an after the fact limit for the particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally, background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.d.

(d) LLD for drinking water samples would be 1 pCi/l. No drinking water pathway exists at the Nine Mile Point Site under normal operating conditions due to the direction and distance of the nearest drinking water intake. Therefore, the LLD of the gamma isotope analysis may be used.

BASES 3.6.20 AND 4.6.20 RADIOLOGICAL ENVIRONMENTAL MONITORING PLAN

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of members of the public resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10CFR Part 50 and thereby supplements the radiological effluent monitoring program 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 the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.6.20-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as a before the fact limit representing the capability of a measurement system and not as an after the fact limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem 40, 586-93 (1968) and Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3.6.21 INTERLABORATORY COMPARISON PROGRAM

Applicability:

Applies to participation in an interlaboratory comparison program on environmental sample analysis.

Objective:

To ensure the accuracy of measurements of radioactive material in environmental samples.

Specification:

Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission. Participation in this program shall include media for which environmental samples are routinely collected and for which intercomparison samples are available.

With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

4.6.21 INTERLABORATORY COMPARISON PROGRAM

Applicability:

Applies to testing the validity of measurements on environmental samples.

Objective:

To verify the accuracy of measurements on radioactive material in environmental samples.

Specification:

The Interlaboratory Comparison Program shall be described in the Offsite Dose Calculation Manual. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report. Participants in the EPA Cross Check Program may provide the EPA program code designation in lieu of providing results.

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BASES FOR 3.6.21 AND 4.6.21 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved 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 part of the quality assurance program for environmental monitoring for the purposes of Section IV.B.2 of Appendix I to 10CFR Part 50.

LIMITING CONDITION FOR OPERATION

3.6.22 LAND USE CENSUS

Applicability:

Applies to the performance of a land use census in the vicinity of the Nine Mile Point Nuclear Facility.

Objective:

To determine the utilization of land within a distance of three miles from the Facility.

Specification:

A land use census shall be conducted and shall identify within a distance of three miles the location in each of the 16 meteorological sectors the nearest residence and within a distance of three miles the location in each of the 16 meteorological sectors of all milk animals. In lieu of a garden census, specifications for vegetation sampling in Table 3.6.20-1 shall be followed, including analysis of appropriate controls.

With a land use census identifying a milk animal location(s) that represents a calculated D/Q value greater than the D/Q value currently being used in specification 4.6.13.b.3, identify the new location(s) in the next Semi-Annual Radioactive Effluent Release Report.

SURVEILLANCE REQUIREMENT

4.6.22 LAND USE CENSUS

Applicability:

Applies to assuring that current land use is known.

Objective:

To verify the appropriateness of the environmental surveillance program.

Specification:

The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as conducting a door-to-door survey, aerial survey or consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report.

3.6.22 LAND USE CENSUS (Continued)

Specification: (Continued)

If the D/Q value at a new milk sampling location is significantly greater (50%) than the D/Q value at an existing milk sampling location, add the new location to the radiological environmental monitoring program within 30 days. The sampling location(s) excluding the control station location, having the lowest calculated D/Q may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.e identify the new location(s) in the next Semi-Annual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the Offsite Dose Calculation Manual reflecting the new location(s).

BASES FOR 3.6.20 AND 4.6.20 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best survey information such as from a door-to-door survey(s), from an aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10CFR Part 50.

In lieu of a garden census, the significance of the exposure via the garden pathway can be evaluated by the sampling of vegetation as specified in Table 3.6.20-1.

A milk sampling location, as defined in Section 1, requires that at least 10 milking cows are present at a designated milk sample location. It has been found from past experience, and as a result of conferring with local farmers, that a minimum of 10 milking cows is necessary to guarantee an adequate supply of milk twice per month for analytical purposes. Locations with less than 10 milking cows are usually utilized for breeding purposes which eliminates a stable supply of milk for samples as a result of suckling calves and periods when the adult animals are dry.

5.0 DESIGN FEATURES

5.1 Site

The Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant site comprising approximately 1500 acres, is located on the shores of Lake Ontario, about seven miles northeast of Oswego, New York. An exclusion distance of nearly 4000 feet is provided between the Station and the nearest site boundary to the west, a mile to the boundary on the east, and a mile and a half to the southern site boundary (as described in the Sixth Supplement of the FSAR).

Figure 5.1-1 is a Site Boundary Map of Nine Mile Point which allows the identification of gaseous and liquid waste release points. Figure 5.1-1 also defines the unrestricted area within the site boundary that is accessible (except for fenced areas) to member of the public.

5.2 Reactor

The reactor core consists of no more than 532 fuel assemblies containing enriched uranium dioxide pellets clad in Zircaloy-2. The core excess reactivity will be controlled by movable control rods and burnable poisons. The core will be cooled by circulation of water internally and external to the pressure vessel through recirculation loops.

5.3 Reactor Vessel

The pertinent features of the reactor vessel other than those referred to in the technical specifications are as follows:

Internal Height	63'-10"
Internal Diameter	17'- 9"
Vessel Design Lifetime	40 years
Materials of Construction Base Metal	SA302B
Clad	Weld Deposited 308L Electrode

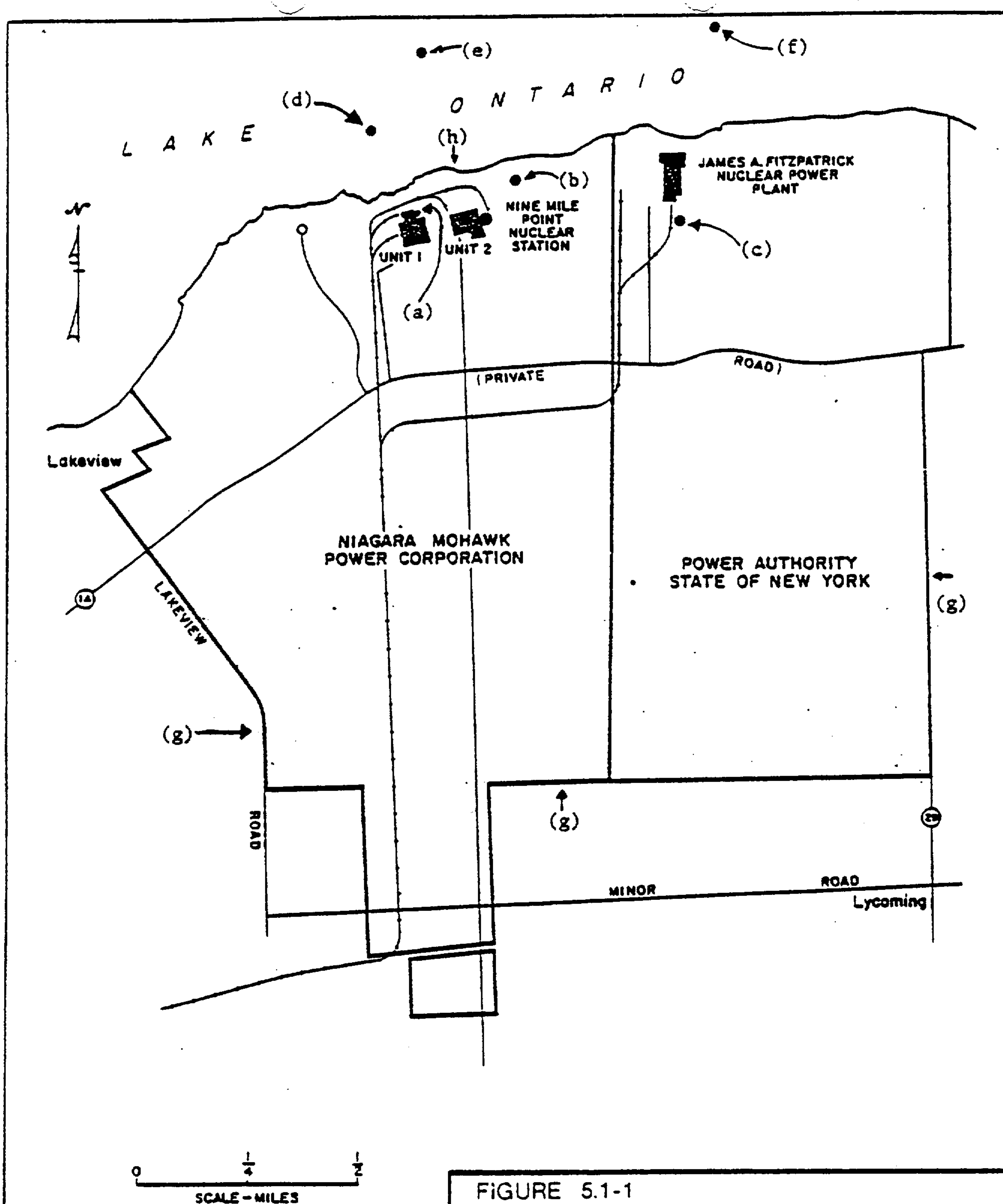

FIGURE 5.1-1

SITE BOUNDARIES

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 1

NOTES TO FIGURE 5.1-1

(a) NMP1 Stack (height is 350')

(b) NMP2 Stack (height is 430')

(c) JAFNPP Stack (height is 385')

(d) NMP1 Radioactive Liquid Discharge (Lake Ontario, bottom)

(e) NMP2 Radioactive Liquid Discharge (Lake Ontario, bottom)

(f) JAFNPP Radioactive Liquid Discharge (Lake Ontario, bottom)

(g) Site Boundary

(h) Lake Ontario Shoreline

Additional Information:

- NMP2 Reactor Building Vent is located 187 feet above ground level
- JAFNPP Reactor and Turbine Building Vents are located 173 feet above ground level
- JAFNPP Radwaste Building Vent is 112 feet above ground level

Activities (Cont'd)

6.5.2.9 The General Superintendent-Nuclear Generation shall assure the performance of a review by a qualified individual/organization of changes to the Radiological Waste Treatment systems.

6.5.2.10 Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President - Nuclear Generation and to the Safety Review and Audit Board.

6.5.2.11 Review of changes to the Process Control Program and the Offsite Dose Calculation Manual. Approval of any changes shall be made by the General Superintendent - Nuclear Generation or his designee before implementation of such changes.

6.5.2.12 Reports documenting each of the activities performed under Specifications 6.5.2.1 through 6.5.2.9 shall be maintained. Copies shall be provided to the Vice President-Nuclear Generation and the Safety Review and Audit Board.

6.5.3 Safety Review and Audit Board (SRAB)

Function

6.5.3.1 The Safety Review and Audit Board shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. quality assurance practices
i. (other appropriate fields associated with the unique characteristics of the nuclear power plant)

Review

6.5.3.7 The SRAB shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or operating license.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meeting minutes of the Site Operations Review Committee.

Audits

6.5.3.8 Audits of facility activities shall be performed under the cognizance of the SRAB. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b. The performance, training and qualifications of the entire facility staff at least once per year.

c. The results of all actions taken to correct deficiencies occuring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per two years.

e. The Facility Emergency Plan and implementing procedures at least once every 12 months.

f. The Facility Security Plan and implementing procedures at least once every 12 months.

g. The Facility Fire Protection Program and implementing procedures at least once per two years.

h. Any other area of facility operation considered appropriate by the SRAB, the Vice President - Nuclear Generation or the Vice President - Nuclear Engineering and Licensing.

i. The radiological environmental monitoring program and the results thereof at least once per 12 months.

j. The Offsite Dose Calculation Manual and implementing procedures at least once per 24 months.

k. The Process Control Program and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

Authority

6.5.3.9 The SRAB shall report to and advise the Vice President - Nuclear Generation and Vice President - Nuclear Engineering and Licensing on those areas of responsibility specified in Section 6.5.3.7 and 6.5.3.8.

Records

6.5.3.10 Records of SRAB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each SRAB meeting shall be prepared, approved and forwarded to the Vice President - Nuclear Generation and Vice President - Nuclear Engineering and Licensing within 30 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.3.7 e,f,g and h above, shall be prepared, approved and forwarded to the Vice President - Nuclear Generation and Vice President - Nuclear Engineering and Licensing within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.3.8 above, shall be forwarded to the Vice President - Nuclear Generation and Vice President - Nuclear Engineering and Licensing within 90 days following completion of the review.

6.6 Reportable Event Action

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.72 and 50.73 to 10CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SORC and the results of this review submitted to the SRAB and the Vice President - Nuclear Generation.

6.7 Safety Limit Violation

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with immediately.

b. The Safety Limit violation shall be reported to the Commission, the Vice President - Nuclear Generation and to the SRAB immediately.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the SRAB and the Vice President - Nuclear Generation within 10 days of the violation.

6.8 Procedures

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USAEC Regulatory Guide 1.33 except as provided in 6.8.2 and 6.8.3 below.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed and approved by the General Superintendent-Nuclear Generation or designee prior to implementation and periodically as set forth in each document.

6.8 Procedures (Continued)

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed and approved by the General Superintendent-Nuclear Generation or designee within 14 days of implementation.

6.9 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of Inspection and Enforcement Regional Office I, King of Prussia, Pennsylvania 19406, unless otherwise noted.

6.9.1 Routine Reports

a. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1 Routine Reports (Continued)

d. Annual Radiological Environmental Operating Report*.

Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1, 1985.

The Annual Radiological Environmental Operating 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 operational controls as appropriate, and with environmental surveillance reports from the previous 5 years, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.6.22.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the Offsite Dose Calculation Manual, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual 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.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.6.21; discussion of all deviations from the sampling schedule of Table 3.6.20-1; and discussion of all analyses in which the LLD required in Table 4.6.20-1 was not achievable.

* A single submittal may be made for a multiple unit station.

** One map shall cover stations near the site boundary; a second shall include the more distant stations.

6.9.1 Routine Reports (cont'd)

e. Semiannual Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin on January 1, 1985.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figure 5.1-1) 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 methodology and parameters in the Offsite Dose Calculation Manual.

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 public from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in the Offsite Dose Calculation Manual.

6.9.1 Routine Reports (cont'd)

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) 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. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

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

f. Solidifcation agent or absorbent (e.g., cement)

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the Process Control Program (PCP) and to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.6.20.

Changes to the Process Control Program (PCP) shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was 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.

6.9.1 Routine Reports (cont'd)

Changes to the Offsite Dose Calculation Manual (ODCM): Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was 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 Offsite Dose Calculation Manual to be changed, together with appropriate analyses or evaluations justifying the change(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.

6.9.2 Fire Protection Program Reports

a. Submit a special report to the appropriate Regional Office as follows:

- Notify the Director of the appropriate Regional Office by telephone within 24 hours.
- Confirm by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and
- Follow-up in writing within 14 days after the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to an operable status.

b. Submit a special report to the Director of the appropriate Regional Office within 30 days following the event outlining the plans and procedures to be used to restore the inoperable equipment to an operable status.

6.9.3 Special Reports

Special reports shall be submitted to the Director of Regulatory Operations Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Reactor Vessel Material Surveillance Specimen Examination, Specification 4.2.2(c) (12 months)

b. Safety Class 1 Inservice Inspection, Specification (See Table 4.2.6(a)) (Three months)

c. Safety Class 2 Inservice Inspections, Specification (See Table 4.2.6(b)) (Three months)

d. Safety Class 3 Inservice Inspections; Specification (See Table 4.2.6(c)) (Three months)

e. Primary Containment Leakage Testing, Specification 3.3.3 (Three months)

f. Secondary Containment Leakage Testing, Specification 3.4.1 (Three months)

g. Sealed Source Leakage In Excess Of Limits, Specification 3.6.5.2 (Three months)

h. Calculate Dose from Liquid Effluent in Excess of Limits, Specification 3.6.15.a(2)(b) (30 days from the end of the affected calendar quarter).

i. Calculate Air Dose from Noble Gases Effluent in Excess of Limits, Specification 3.6.15.b(2)(b), (30 days from the end of the affected calendar quarter).

j. Calculate Dose from I-131, I-133, H-3 and Radioactive Particulates with half lives greater than eight days in Excess of Limits, Specification 3.6.15.b(3)(b), (30 days from the end of the affected calendar quarter).

k. Calculated Doses from Uranium Fuel Cycle Source in Excess of Limits Specification 3.6.15.d (30 days from the end of the affected calendar year).

l. Inoperable Gaseous Radwaste Treatment System, Specification 3.6.16.b (30 days from the event).

m. Environmental Radiological Reports. With the level of radioactivity (as the result of plant effluents), in an environmental sampling medium exceeding the reporting level of Table 6.9.3-1, when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within thirty (30) days from the end of the calendar quarter a special report identifying the cause(s) for exceeding the limits, and define the corrective action to be taken.

Table 6.9.3-1
REPORTING LEVEL FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/kg,wet)	Milk (pCi/l)	Food Products (pCi/kg,wet)
H-3	30,000				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr/Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10.0	1,000	60	1,000
Cs-137	50	20.0	2,000	70	2,000
Ba/La-140	200			300	

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6.10 Record Retention

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. REPORTABLE EVENT REPORTS.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

6.10 Record Retention (Continued)

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the SORC and the SRAB.

l. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and Quality Assurance records showing that these procedures were followed.

6.11 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 High Radiation Area

6.12.1 In lieu of the "control device" or "alarm signal" required by Paragraph 20.203(c)(2) of 10CFR20, each high radiation area normally accessible* by personnel in which the intensity of radiation is greater than 100 mrem/hr** but less than 1000 mrem/hr** shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit in accordance with site approved procedures. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by oneor more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection, with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Protection Supervisor or designate in the Radiation Work Permit.

6.12.2 In addition to the requirements of 6.12.1 areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem** shall be provided with locked doors to prevent unauthorized entry, and the hard keys or access provided by magnetic keycard shall be maintained under the administrative control of the Station Shift Supervisor or designate on duty and/or the Radiation Protection Supervisor or designate. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify in accordance with site approved procedures accordingly, the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, continuous surveillance, direct or remote, such as use of closed circuit TV cameras, may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as the drywell, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device.

* by accessible passage and permanently fixed ladders
** measurement made at 18" from source of radioactivity

6.13 Fire Protection Inspection

6.13.1 An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm.

6.13.2 An inspection and audit by an outside qualified fire consultant shall be performed at intervals no greater than 3 years.

6.14 Systems Integrity

Procedure shall be established, implemented and maintained to meet or exceed the requirements and recommendations of Section 2.1.6.a of NUREG 0578.

6.15 Iodine Monitoring

Procedures shall be established, implemented and maintained to meet or exceed the requirements and recommendations of Section 2.1.8.c of NUREG 0578.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 INTRODUCTION

To comply with Section V of Appendix I of 10 CFR Part 50, Niagra Mohawk Power Corporation 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. Niagra Mohawk filed this information with the Commission by letter dated February 1, 1984, which requested changes to the Technical Specifications appended to Facility Operating License No. DPR-63 for Nine Mile Point Unit 1. 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 Parts 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

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", Section 50.36a, "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 Part 20.106, "Radioactivity in Effluents to Unrestricted Areas"; (2) require that operating procedures developed for the control of effluents be established and followed; (3) require that equipment installed in the radioactive waste system be maintained and used; and (4) 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 as may be required by the Commission to estimate maximum potential radiation dose to the public resulting from the effluent releases.

10 CFR Part 20, "Standards for Protection Against Radiation," paragraphs 20.105(c), 20.106(g), and 20.405(c), 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.

10 CFR Part 50, Appendix A - General Design Criteria for Nuclear Power Plants, contains Criterion 60, Control of releases for 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 include 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.

10 CFR Part 50, Appendix B, establishes quality assurance requirements for nuclear power plants.

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

2.2 Standard Radiological Effluent Technical Specifications

NUREG-0473 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." NUREG-0133 describes methods found acceptable to the staff of the NRC for the calculation of 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 sytems.

The above NUREG documents address all of the radiological effluent technical specifications needed to assure compliance with the guidance and requirements provided by the regulations previously cited. 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 the regulations and with the intent of the regulatory guidance.

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 is 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 established to assure compliance with 10 CFR Part 20 standards provide, in the event the limiting conditions of operation are exceeded, that without delay conditions are restored to within the limiting conditions. Otherwise, the facility is required to effect approved shutdown procedures. In general, the specifications established to assure compliance with 10 CFR Part 50 provide, in the event the limiting conditions of operation are exceeded, that within specified times corrective actions are to be taken, 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 of operation 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

The enclosed report (TER-C5506-100) was prepared for us by Franklin Reasearch Center (FRC) as part of our technical assistance contract program. Their report provides their technical evaluation of the compliance of the Licensee's submittal with NRC provided criteria. The staff has reviewed this TER and agrees with the evaluation.

3.1 SUMMARY

The proposed changes to the radiological effluent technical specifications for Nine Mile Point Unit 1 have been reviewed, evaluated, and found to be in compliance with the requirements of the NRC regulations and with the intent of NUREG-0133 and NUREG-0473 (Nine Mile Point Unit 1 is a boiling water reactor) and thereby fulfill all the requirements of the regulations related to radiological effluent technical specifications.

The proposed changes will not remove or relax any existing requirement related to the probability or consequences of accidents previously considered and do not involve a significant hazards consideration.

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.

	Standard Radiological Effluent Technical Specifications																												
	Instrumentation		Radioactive Effluents																Rad. Envir. Monitoring			Design Features	Administrative Control						
			Liquid				Gaseous																						
							PWR/BWR				PWR		BWR																
Provisions of Title 10 Code of Federal Regulations	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.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in inspection and surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this 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.

Attachment:
Technical Evaluation
Report

Principal Contributors: W. Meinke
C. Willis

Dated: November 2, 1984