

# REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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 FACIL: 50-220 Nine Mile Point Nuclear Station, Unit 1, Niagara Powe 05000220  
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 LEMPGES, T.E. Niagara Mohawk Power Corp.  
 RECIP. NAME RECIPIENT AFFILIATION  
 VASSALLO, D.B. Operating Reactors Branch 2

SUBJECT: Forwards application for amend to License DPR-63, changing  
 radiological effluent Tech Specs re radiological environ  
 monitoring program.

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1. The first step is to identify the problem or question that needs to be answered. This involves understanding the context and the specific requirements of the task.

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February 1, 1984

Director of Nuclear Reactor Regulation  
Attention: Mr. Domenic B. Vassallo, Chief  
Operating Reactors Branch No. 2  
Division of Licensing  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

Re: Nine Mile Point Unit 1  
Docket No. 50-220  
DPR-63

Dear Mr. Vassallo:

Niagara Mohawk Power Corporation hereby transmits three (3) originals and nineteen (19) copies of an application for an amendment to Nine Mile Point Unit 1's Operating License DPR-63. Also enclosed are forty (40) copies of the proposed change to the Technical Specifications in Appendix A of the above mentioned license.

The proposed Technical Specifications contained herein pertain to the Radiological Effluent Technical Specifications requested in your letter of July 11, 1978 and subsequent discussions and submittals. Due to the integrated Radiological Environmental Monitoring Program between Nine Mile Point Unit 1 and the James A. Fitzpatrick Nuclear Power Plant, we ask that this section of the proposed Technical Specification be approved at the same time as it is approved in the proposed James A. Fitzpatrick Nuclear Power Plant Radiological Effluent Technical Specifications.

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1. The first step in the process of identifying a problem is to determine the nature of the problem. This involves a thorough understanding of the situation and the factors that are contributing to the problem.

11/16/68, 21, 22, 23, 24

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Pursuant to 10CFR50.91(b)(1), we have provided a copy of this license amendment application and the associated analysis regarding no significant hazard considerations to the appropriate state representative as outlined in your May 2, 1983 Generic Letter 83-19.

Very truly yours,

  
T. E. Lempges  
Vice President  
Nuclear Generation

TEL/AMS:djm

Attachments

xc: Mr. Jay Dunkleberger  
Division of Policy Analysis and Planning  
New York State Energy Office  
Agency Building 2  
Empire State Plaza  
Albany, NY 12223

1. The first part of the document is a list of names and addresses. The names are: John Doe, Jane Smith, and Bob Johnson. The addresses are: 123 Main St, 456 Elm St, and 789 Oak St.

2. The second part of the document is a list of dates and times. The dates are: 1/1/2020, 2/1/2020, and 3/1/2020. The times are: 10:00 AM, 2:00 PM, and 5:00 PM.

3. The third part of the document is a list of locations. The locations are: New York, Los Angeles, and Chicago.

4. The fourth part of the document is a list of items. The items are: a book, a pen, and a paper.

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\* to be submitted at a future date

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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 Calculation 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 mainstream line.

## SURVEILLANCE REQUIREMENT

### 4.6.1 STATION PROCESS EFFLUENTS

- a. Monitoring the radioactive 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.
- 





VACUUM PUMP ISOLATION

Limiting Condition for Operation

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



## LIMITING CONDITION FOR OPERATION

### 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 radioactive 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.

## SURVEILLANCE REQUIREMENT

### 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

| <u>Instrument</u>                        | <u>Minimum<br/>Channels<br/>Operable</u> | <u>Applicability</u> |
|--|--|----------------------|
| 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.





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  $5 \times 10^{-7}$  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 minimum channels operable requirement, steam release via this pathway may commence or continue provided vent pipe radiation dose rates are monitored once per four hours.



Table 4.6.14-1  
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Surveillance Requirement

| <u>Instrument</u>                             | <u>Sensor<br/>Check</u> | <u>Source<br/>Check (f)</u> | <u>Channel<br/>Test</u> | <u>Channel<br/>Calibration</u> |
|---|-------------------------|-----------------------------|-------------------------|--------------------------------|
| 1. Gross Beta or Gamma Radioactivity Monitors |                         |                             |                         |                                |
| a. Liquid Radwaste Effluent Line              | Once/day*               | Once/<br>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.



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.



## LIMITING CONDITION FOR OPERATION

### 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.

## SURVEILLANCE REQUIREMENT

### 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

| <u>Instrument</u>  | <u>Minimum<br/>Channels<br/>Operable</u> | <u>Applicability</u> | <u>Parameter</u>               |
|--|--|----------------------|--------------------------------|
| 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)

| <u>Instrument</u>  | <u>Minimum<br/>Channels<br/>Operable</u> | <u>Applicability</u> | <u>Parameter</u>                         |
|--|--|----------------------|--|
| 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

| <u>Instrument</u>   | <u>Surveillance Requirements</u> |                         |                            |                                |
|---|----------------------------------|-------------------------|----------------------------|--------------------------------|
|   | <u>Sensor<br/>Check</u>          | <u>Source<br/>Check</u> | <u>Channel<br/>Test</u>    | <u>Channel<br/>Calibration</u> |
| 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<br>Explosive Gas Monitoring system (for<br>system designed to withstand the effects<br>of a hydrogen explosion) |                                  |                         |                            |                                |
| a. Hydrogen Monitor   | Once/day (d)                     | None                    | Once/month                 | Once/3 months(e)               |
| 3. Condenser Air Ejector Radioactivity Monitor<br>(Recombiner Discharge or Air Ejector<br>Discharge)  |                                  |                         |                            |                                |
| a. Noble Gas Activity Monitor   | Once/day (f)                     | Once/month              | Once/operating<br>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<br>cycle(b)     |



A



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



## 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  $2 \times 10^{-4}$  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)



## LIMITING CONDITION FOR OPERATION

### 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 mrem to the total body and to less than or equal to 5 mrem to any organ, and
- (b) During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem 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.

## SURVEILLANCE REQUIREMENT

### 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

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

\* Completed prior to each release.

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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,

$2.22 \times 10^6$  is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

$\lambda$  is the radioactive decay constant for the particular radionuclide, and

$\Delta t$  for plant effluents is the elapsed time between the midpoint of sample collection and time of counting. Typical values of E, V, Y and  $\Delta t$  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 mrem/year to the total body and less than or equal to 3000 mrem/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 mrem/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 mrad 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 mrad 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.



## LIMITING CONDITION FOR OPERATION

### 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 mrem to any organ and,
- (b) During any calendar year: Less than or equal to 15 mrem to any organ.

## SURVEILLANCE REQUIREMENT

### 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

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





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  $1 \times 10^{-4}$ .
- (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 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

With the calculated doses from the release of radioactive 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.



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.





## BASES FOR 3.6.15 AND 4.6.15 RADIOACTIVE EFFLUENTS

### 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 permissible 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).



## BASES FOR 3.6.15 AND 4.6.15 RADIOACTIVE EFFLUENTS

### 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 radioactive materials in liquid 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 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.



## BASES FOR 3.6.15 AND 4.6.15 RADIOACTIVE EFFLUENTS

### 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 mrem/year to the total body or to less than or equal to 3000 mrem/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 mrem/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).



## BASES FOR 3.6.15 AND 4.6.15 RADIOACTIVE EFFLUENTS

### 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.





## BASES FOR 3.6.15 AND 4.6.15 RADIOACTIVE EFFLUENTS

### 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 values 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.



## BASES FOR 3.6.15 AND 4.6.15 RADIOACTIVE EFFLUENTS

### 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 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem. 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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## LIMITING CONDITION FOR OPERATION

### 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.

## SURVEILLANCE REQUIREMENT

### 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 constituents and mixing and curing times.



## LIMITING CONDITION FOR OPERATION

### 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.

## SURVEILLANCE REQUIREMENT

### 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.



## LIMITING CONDITION FOR OPERATION

### 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.

## SURVEILLANCE REQUIREMENT

### 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 MONITORING 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 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (Continued)

### Specification: (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:

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

$$\dots \leq 1.0$$

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).

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Table 3.6.20-1  
OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Limiting Condition for Operation

| <u>Exposure Pathway<br/>and/or Sample</u> | <u>Number of Samples(a)<br/>and Locations</u>   | <u>Sampling and Collection<br/>Frequency (a)</u>   | <u>Type of Analysis<br/>and Frequency</u>  |
|---|---|--|--|
| Radioiodine &<br>Particulates             | <p>Samples from 5 locations:</p> <ol style="list-style-type: none"> <li>1) 3 samples from off-site locations in different sectors of the highest calculated site average D/Q (based on all site licensed reactors)</li> <li>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)</li> <li>3) 1 sample from a control location 10-17 miles distant and in a least prevalent wind direction (d)</li> </ol> | <p>Continuous sampler operation with sample collection weekly or as required by dust loading, whichever is more frequent</p> | <p><u>Radioiodine Canisters</u><br/>analyze once/week for I-131.</p> <p><u>Particulate Samplers</u><br/>Gross beta radio-activity following filter change, (b) composite (by location) for gamma isotopic analysis (c) once per 3 months, (as a minimum)</p> |



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.





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)
- 2) 1 sample from a milk sampling location at a control location (9-20 miles distant and in a least prevalent wind direction)(d)

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

Fish

- 1) 2 samples of commercially or recreationally important species in the vicinity of a site discharge point(h)
- 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)

Twice per year

Gamma isotopic analysis(c) on edible portions twice per year



Exposure Pathway  
and/or Sample

Food Products

Number of Samples(a)  
and Locations

- 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

Sampling and Collection  
Frequency(a)

Once per year during  
harvest season

Type of Analysis  
and Frequency

Gamma isotopic  
analysis of edible  
portions (isotopic  
to include I-131)  
Once during the  
harvest season

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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 discharge. 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

| <u>Analysis</u> | <u>Water (c)</u><br><u>(pCi/l)</u> | <u>Airborne Particulate</u><br><u>or Gases (pCi/m )</u> | <u>Fish</u><br><u>(pCi/kg, wet)</u> | <u>Milk</u><br><u>(pCi/l)</u> | <u>Food Products</u><br><u>(pCi/kg, wet)</u> | <u>Sediment</u><br><u>(pCi/kg, dry)</u> |
|-----------------|------------------------------------|---|-------------------------------------|-------------------------------|--|---|
| gross beta      | 4                                  | 0.01  |                                     |                               |  |   |
| H-3             | 3000                               |   |                                     |                               |  |   |
| Mn-54           | 15                                 |   | 130                                 |                               |  |   |
| Fe-59           | 30                                 |   | 260                                 |                               |  |   |
| Co-58-60        | 15                                 |   | 130                                 |                               |  |   |
| Zn-65           | 30                                 |   | 260                                 |                               |  |   |
| Zr/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.

$\lambda$  is the radioactive decay constant for the particular radionuclide, and

$\Delta t$  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  $\Delta t$  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).



## LIMITING CONDITION FOR OPERATION

### 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.

## SURVEILLANCE REQUIREMENT

### 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.





#### 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.



# LIMITING CONDITION FOR OPERATION

# SURVEILLANCE REQUIREMENT

## 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<br>308L Electrode |



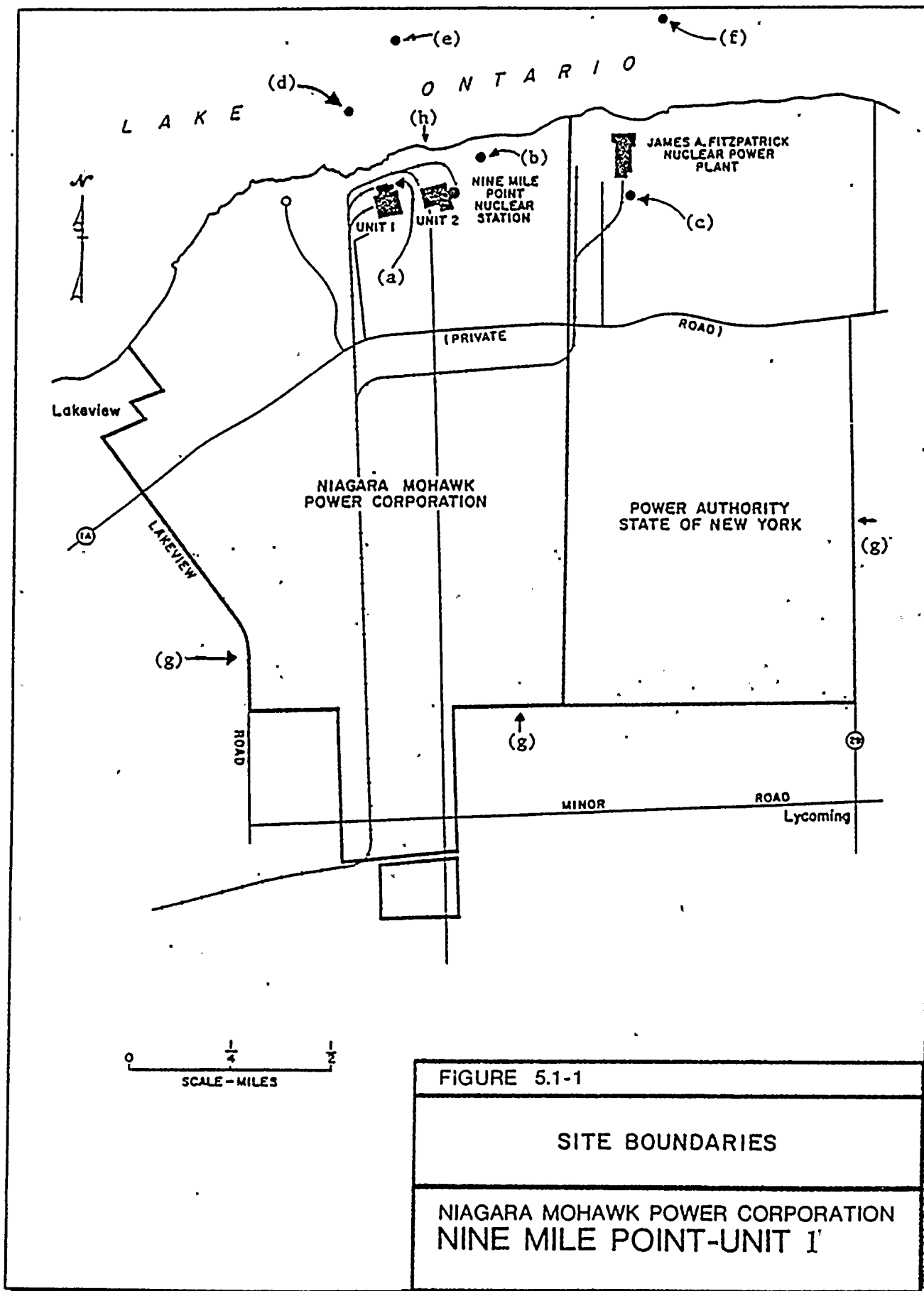


FIGURE 5.1-1

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. 3
- 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. 4
- 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. 3

### 6.5.3 Safety Review and Audit Board (SRAB) 3

#### 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: 3
  - 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 occurring 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. Solidification 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

| <u>Analysis</u> | <u>Water<br/>(pCi/l)</u> | <u>Airborne Particulate<br/>or Gases (pCi/m3)</u> | <u>Fish<br/>(pCi/kg,wet)</u> | <u>Milk<br/>(pCi/l)</u> | <u>Food Products<br/>(pCi/kg,wet)</u> |
|-----------------|--------------------------|---|------------------------------|-------------------------|---------------------------------------|
| 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                     |                                       |



#### 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 one or more of the following:





ATTACHMENT B

NIAGARA MOHAWK POWER CORPORATION

LICENSE DPR-63

DOCKET NO. 50-220

Supporting Information

This revision affects those sections of the Technical Specifications pertaining to radiological effluents and environmental monitoring. Commonly referred to as the Radiological Effluent Technical Specifications, these changes are brought about by the request of the Nuclear Regulatory Commission in its July 11, 1978 letter. Niagara Mohawk Power Corporation's submittal of a Radiological Effluent Technical Specification amendment request to the Nuclear Regulatory Commission on February 28, 1979 provided the requested changes to Nine Mile Point Unit 1's Technical Specifications. The Nuclear Regulatory Commission reviewed this submittal and requested Niagara Mohawk incorporate additional information. A meeting between the Nuclear Regulatory Commission and Niagara Mohawk on May 25-26, 1982 resulted in submittal of a draft Radiological Effluent Technical Specification on March 1, 1983. Nuclear Regulatory Commission's comments on the draft were subsequently discussed in a telephone conference call on November 10, 1983. It was mutually agreed that another formal Radiological Effluent Technical Specification amendment request for Nine Mile Point Unit 1 would be submitted by February 1, 1984.

The model Radiological Effluent Technical Specification referenced in this Attachment include NUREG-0473, Revision 2, "Radiological Effluent Technical Specifications for Boiling Water Reactors" dated February 1, 1980 and NUREG-0472, Revision 3, Draft 7", "Radiological Effluent Technical Specifications for Pressurized Water Reactors" dated September 3, 1982. During the November 10, 1983 conference call, the Nuclear Regulatory Commission requested Niagara Mohawk to provide justification on selected items in Niagara Mohawk's Radiological Effluent Technical Specification amendment. The requested information is presented below.

1. A "Special Reports" section has been added to the Radiological Effluent Technical Specifications as Section 6.9.3 replacing the "Unique Reporting Requirements" section appearing in the previous draft Radiological Effluent Technical Specifications. This section is not specifically indicated in the table of contents because it is a sub-part of "Reporting Requirements," Section 6.9 of the existing Technical Specifications.
2. A definition of source check, identical to definition provided in the model Radiological Effluent Technical Specifications, has been included in the submittal. This definition can be found in Paragraph 1.25.
3. The term rated flux is used at Nine Mile Point Unit 1 to denote the neutron flux that corresponds to a steady-state power level of 1850 megawatts thermal. This term is similar to the model Radiological Effluent Technical Specifications definition of rated thermal power. The definition of rated flux can be found as Paragraph 1.14 of the Technical Specifications.



ATTACHMENT B (Continued)

4. Niagara Mohawk's definition of purging (Paragraph 1.23) includes a numerical value of 19.5 percent oxygen concentration to define purge completion not found in the model Radiological Effluent Technical Specifications. Niagara Mohawk proposes to maintain the use of this value to quantitatively determine when the purge can be terminated and when the normal ventilation system can be initiated. The 19.5 percent oxygen concentration value is used to approximately correspond to an environment which is not considered to be an oxygen deficient atmosphere, as defined by the Mine Safety and Health Administration in 30CFR11.3.
5. Source check surveillance requirements have been included for Radioactive Liquid Effluent Monitoring Instrumentation indicated in Table 4.6.14-1 of the Radiological Effluent Technical Specifications submittal.
6. Footnote (b) to Table 3.6.14-1 of the submittal indicates that an operator will be in the Radwaste Control Room at all times during a release of liquid radwaste from the plant. The Liquid Radwaste Effluent Monitor at Nine Mile Point Unit 1 does not have an automatic isolation capability as the model Radiological Effluent Technical Specifications suggests. This footnote serves the same function as automatic isolation capability, namely to ensure the termination of the liquid radwaste effluent release should the monitor alarm setpoint be exceeded during the release.
7. Footnote (a) to Table 4.6.14-1 indicates that the channel test for the liquid radwaste and service water effluent monitors shall include a demonstration that control room alarm annunciation will occur if the instrument indicates a down-scale failure. This down-scale failure is indicative of an inoperable instrument, possibly a circuit failure. The circuitry for these instruments is such that a circuit failure will cause a down-scale failure, which will in turn trip an output signal and finally cause an annunciation. Although the model Radiological Effluent Technical Specification suggests a circuit failure test specifically as a part of the channel test, these monitors test for a circuit failure in an alternate manner.
8. Footnote (g) to Table 3.6.14-2 indicates that should the condenser air ejector noble gas activity monitor become inoperable, gases from the main condenser off-gas treatment system may continue to be released if the two conditions specified are met. The purpose of the off-gas monitor is to monitor fuel performance, not evaluate actual radioactive gaseous effluents. The noble gas activity monitor of the stack effluent monitoring system is used to evaluate actual radioactive gaseous effluents. The later system is used because it is at the stack effluent release point where the gases are released after treatment (i.e. hold-up and charcoal columns) and from where the actual effluent can best be evaluated. Based on this information, termination of gaseous effluent after 72 hours as the model suggests cannot be justified. In addition, Niagara Mohawk would require sampling and analysis of gases from the main condenser off-gas treatment system as a condition for continued gaseous effluent. This condition is one not suggested by the model Radiological Effluent Technical Specifications, but one which is felt to be prudent.



ATTACHMENT B (Continued)

9. Many Action Statements such as flow rate estimates, are performed at frequencies of once per eight hours (once per shift). This frequency has been preserved in footnote (c) of Table 3.6.14.-2 as the Action Statement should flow rate measurement devices become inoperable. Although the Nuclear Regulatory Commission's staff suggested frequencies of six hours, past operating experience indicates that frequent changes in flow do not occur. Niagara Mohawk cannot justify changing the flow rate estimate frequencies.
10. As mutually agreed between Niagara Mohawk and the Nuclear Regulatory Commission during the November 10, 1983 conference call, footnote (a) to Table 3.6.14-2 is acceptable as it relates to the noble gas activity monitor.
11. The hydrogen monitoring capability of the Main Condenser Off-Gas Treatment Explosive Gas Monitoring System requires one channel to be operable as a minimum. Footnote (e) to Table 3.6.14-2 would allow an alternative should this capability be lost, namely, the release of radioactive gaseous effluent via this pathway may continue contingent upon grab samples being collected and analyzed once per eight hours. Niagara Mohawk feels there is insufficient justification to require releases to be terminated after 30 days of inoperable instrumentation and suggests that obtaining and analyzing samples once per eight hours is equivalent to requiring grab samples to be obtained every four hours and analyzed within the following four hours as the model suggests. The model suggested frequencies are inconsistent with standard sampling frequencies.
12. The model Technical Specifications suggest that channel checks (sensor checks) be performed on the iodine and particulate samplers of the Main Stack Monitoring System on a weekly basis. Table 4.6.14-2 of the Radiological Effluent Technical Specifications submittal would require no sensor check of the plant's Stack Effluent Monitoring System iodine sampler cartridges and particulate sampler filters. No sensor check is required because the cartridges and filters are changed out as frequently or more frequently than the model suggests for the check.
13. Table 4.6.14-2 includes a source check frequency of once per month for the noble gas activity monitor of the Stack Effluent Monitoring System. This item is in accordance with the model Radiological Effluent Technical Specifications and Nuclear Regulatory Commission's staff recommendation.
14. As discussed in Number 7 of this attachment, a circuit failure will cause a down-scale failure in the liquid radwaste and service water effluent line monitors. This sequence of events would also hold true for the noble gas activity monitor of the condenser air ejector radioactivity monitor as noted in footnote (c) of Table 4.6.14-2. These events include a circuit failure causing a down-scale failure which in turn causes a trip output signal, finally causing an annunciation in the control room.



ATTACHMENT B (Continued)

15. Niagara Mohawk has agreed to provide for quarterly and monthly batch sampling and analysis in lieu of continuous composite sampling of the service water effluent system in Table 4.6.15-1. Footnote (f) to this table provides for daily and composite sampling of this effluent should the alarm setpoint of the service water effluent monitor be exceeded. As suggested by the Nuclear Regulatory Commission staff, the Quad-Cities Technical Specification was the basis for this item submittal.
16. The model Radiological Effluent Technical Specifications suggest that a Limiting Condition for Operation be provided for both calendar quarter and calendar year dose or dose commitment to a member of the public from liquid radioactive effluents from an operating plant. However, it is noted that 10CFR50 does not address annual Limiting Condition for Operation, only quarterly Limiting Condition for Operation. This omission in 10CFR50 is interpreted by Niagara Mohawk to mean that an annual Limiting Condition for Operation is not required for this item. However, in keeping with Niagara Mohawk's policy of maintaining radiation exposure "as low as is reasonably achievable," we have provided both annual and quarterly Limiting Condition for Operations in Section 3.6.15 a(2).
17. Niagara Mohawk has provided for alternative means of analyzing noble gases, gross gamma or principal gamma emitters from stack gaseous waste as identified as footnote (g) to Table 4.6.15-2. Should the primary means of the specified analysis instrumentation, the Radiological Gaseous Effluent Monitoring System (RAGEMS) be inoperable, Niagara Mohawk proposes to use its backup instrumentation which has a higher Lower Limit of Detection than the RAGEMS. It is Niagara Mohawk's opinion that the back-up instrumentation will provide for a better monitoring system overall than other alternatives presently available. The primary instrumentation has a better detection capability than the backup. The need for different values is indicated in this Radiological Effluent Technical Specifications submittal.
18. Niagara Mohawk indicates in footnote (h) to Table 4.6.15-2 that tritium (H-3) grab samples shall be taken from the station ventilation exhaust (stack) weekly when fuel is off-loaded and continued until such time stable tritium release levels can be demonstrated. No grab samples from the spent fuel pool area for tritium evaluation are proposed. Niagara Mohawk notes that fuel has been in the Nine Mile Point Unit 1 Spent Fuel Pool for many years with moderate tritium release levels demonstrated. Footnote (h) will require weekly sampling from the stack when fuel is offloaded so other possible outage related tritium pathways, such as the radwaste system, can also be evaluated.





ATTACHMENT B (Continued)

19. Footnote (f) to Table 4.6.15-2 indicates that the stack particulate and charcoal samples will be analyzed to determine changes in Iodine-131 and particulate release rate from the stack when the noble gas release rate is in excess of 1000 uCi/second and the steady-state noble gas release rate increases by 50 percent. The model Radiological Effluent Technical Specifications suggests that samples should be analyzed if the reactor power level change exceeds 15 percent per hour. Nine Mile Point Unit 1 operating experience indicates that normally release rate changes do not occur even with 20-25 percent per hour reactor power level changes but may occur with no change in reactor power since effluent may originate from sources other than the main condenser (e.g. increased steam leakage). Based on this observation, Niagara Mohawk proposes to maintain the 50 percent release rate change trigger level for sampling and analysis.
20. The model Radiological Effluent Technical Specification suggests that a Limiting Condition for Operation be provided for both calendar quarter and annual air dose or air dose commitment to a member of the public from radioactive gaseous effluents from an operating plant. It is noted that 10CFR50 does not address annual Limiting Condition for Operation, only quarterly. This omission is interpreted by Niagara Mohawk to mean an annual Limiting Condition for Operation is not required for this item. Niagara Mohawk has provided this information in Section 3.6.15.b(2) as the model Radiological Effluent Technical Specification suggests in keeping with our policy of maintaining radiation exposure "as low as is reasonably achievable."
21. Footnote (d) to Table 4.6.15-2 indicates that stack release of principal gamma emitters will be analyzed following an increase of greater than 50 percent in the recombiner discharge monitor activity or when the stack release rate is in excess of 1,000 uCi/second and steady-state gaseous release rate increases by 50 percent. The model Radiological Effluent Technical Specifications suggests that samples should be analyzed if the reactor power level change exceeds 15 percent per hour. Nine Mile Point Unit 1 operating experience indicates that normally release rate changes do not occur even with 20-25 percent per hour reactor power level changes but may occur with no change in reactor power since effluent may originate from sources other than the main condenser (e.g. increased steam leakage). The proposed footnote would require calculation of possible effluent isotopic change whether accompanied by reactor power level change or not.
22. Section 3.6.16.b of this submittal indicates that Niagara Mohawk will normally use the gaseous effluent treatment system to limit effluents to those specified by Section 3.6.15. Should this system become inoperable for more than seven days, a Special Report will be submitted in accordance with the model Radiological Effluent Technical Specification. Niagara Mohawk feels this operating flexibility is justified due to the negligible impact short term operations without gaseous effluent treatment will have on doses to unrestricted areas from gaseous effluent releases.



ATTACHMENT B (Continued)

23. In Section 3.6.15.c of this submittal, Niagara Mohawk provides for exceeding the main condenser release specification provided the Off-Gas Treatment System is in service to allow for anticipated short duration operation with moderate fuel leakage. With the Off-Gas System in operation, release of radioactive gaseous effluent from the stack would continue to be negligible (typical DF is in the range of 50-250) and because the Off-Gas System is a negative pressure system, no leakage would occur.
24. The specification for Mark I Containment venting/purging (Section 3.6.18) was revised to make the objectives of the specification consistent with the model bases (NUREG 0473, 3/4 11.28).
25. Footnote (b) to Table 4.6.20-1 addresses the required detection capabilities for thermoluminescent dosimeters used for environmental measurements. As indicated, the specific capabilities required are included in the Offsite Dose Calculation Manual. ANSI N.545 (1975) was used to define the detection capabilities. However, several deviations exist between Niagara Mohawk's required detection capabilities and those identified in ANSI N.54<sup>c</sup> (1975). These deviations include those areas of reproducibility and dependance of exposure interpretation on the length of the field cycle and other minor deviations.
26. Pursuant to conversation with the Nuclear Regulatory Commission on November 10, 1983, Niagara Mohawk proposes to obtain six samples of fruits and/or vegetables from available offsite locations in lieu of conducting a garden census in the Radiological Environmental Monitoring Program. Sampling shall be conducted as indicated in Table 3.6.20-1.
27. Specification 3.6.22 indicates that if a D/Q value at a new milk sampling location is greater than 50 percent of the D/Q value at an existing milk sampling location, this new milk sampling location will be added to the Radiological Environmental Monitoring Program within 30 days pursuant to specifications in the Radiological Effluent Technical Specifications. This 50 percent value was mutually agreed to by the Nuclear Regulatory Commission and Niagara Mohawk during conversation on November 10, 1983.
28. The model Radiological Effluent Technical Specifications suggests that the semi-annual Radioactive Effluent Release Report include a discussion on major changes to radioactive liquid, gaseous and solid waste treatment systems that took place since the last report. In lieu of this method of reporting such changes, Niagara Mohawk will include this discussion as a part of the annual Final Safety Analysis Report update for Nine Mile Point Unit 1. This method of reporting changes to the mentioned systems is indicated as an acceptable alternative by the model Radiological Effluent Technical Specification.



ATTACHMENT C

NIAGARA MOHAWK POWER CORPORATION

LICENSE DPR-63

DOCKET NO. 50-220

Amendment Classification

This proposed amendment to the Operating License had previously been evaluated and was determined to fall within the definition of Class III of 10CFR170.22. Niagara Mohawk Power Corporation's submittal of February 28, 1979 included a check in the amount of four thousand dollars (\$4,000). Since this submittal is a result of a request by the U.S. Nuclear Regulatory Commission to supply additional information, no additional fee is required.



ATTACHMENT D

NIAGARA MOHAWK POWER CORPORATION

LICENSE DPR-63

DOCKET NO. 50-220

No Significant Hazards Considerations Analysis

The proposed amendment increases environmental surveillance requirements as it relates to doses to a member of the public from effluents as well as environmental monitoring. The Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48FR14870) of amendments considered not likely to involve significant hazards consideration. One of the examples relates to a change which constitutes an additional limitation, restriction or control not presently in the Technical Specifications. The proposed Radiological Effluent Technical Specifications change is similar to this example in that it imposes a greater restriction on effluent releases and dose associated with these releases from the plant. The proposed Radiological Effluent Technical Specifications change, therefore, involves no significant hazards consideration.

