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Constellation
Nuclear

Nine Mile Point
Nuclear Station

A Member of the
Constellation Energy Group

February 28, 2002
NMP2L 2051

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: Nine Mile Point Unit 2
Docket No. 50-410
NPF-69

Subject: ***January- December 2001 Radioactive Effluent Release Report***

Gentlemen:

In conformance with the Nine Mile Point Unit 2 (NMP2) Technical Specifications, enclosed is the Radioactive Effluent Report for the reporting period January through December 2001.

Included in this report is a summary of gaseous, liquid, and solid effluents released from the station during the reporting period (Attachments 1 - 6), a summary of revisions to the Offsite Dose Calculation Manual and the Radwaste Process Control Program during the reporting period (Attachments 7 and 8), and an explanation as to the cause and corrective actions regarding the inoperability of any station liquid and/or gaseous effluent monitoring instrumentation (Attachment 9). Attachments 10 and 11 provide a summary and assessment of radiation doses to members of the public within and outside the site boundary, respectively, from liquid and gaseous effluents as well as direct radiation in accordance with 40 CFR 190.

The format used for the effluent data is outlined in Appendix B of Regulatory Guide 1.21, Revision 1. Dose assessments were made in accordance with the NMP2 Offsite Dose Calculation Manual. Distribution is in accordance with 10 CFR 50.4(b)(1) and the Technical Specifications.

Attachment 12 to this report is an update of actual data for the third and fourth quarters 2000 used in the July through December 2000 Semi-Annual Radioactive Effluent Release Report.

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Attachment 13 is a Summary of Changes to the Environmental Monitoring and Dose Calculation Locations.

Attachment 14 is a copy of Revision 22 of the Offsite Dose Calculation Manual.

Attachment 15 is a copy of Revision 5 of the Radwaste Process Control Program.

During the reporting period from January through December 2001, NMP2 did not exceed any 10 CFR 20, 10 CFR 50, Technical Specification, or Offsite Dose Calculation Manual limits for gaseous or liquid effluents.

If you have any questions concerning the attached report, please contact Mr. Anthony Salvagno, (315) 349-1456, Engineering Services, Nine Mile Point.

Very truly yours,

Bruce S. Montgomery
General Manager Nuclear Engineering

BSM/CLW/cld
Enclosure

xc: Mr. H. J. Miller, Regional Administrator, Region I
Mr. G. K. Hunegs, NRC Senior Resident Inspector, Region I
Mr. P. S. Tam, Senior Project Manager, NRR (2 copies)
Records Management

NINE MILE POINT NUCLEAR STATION - UNIT 2

RADIOACTIVE EFFLUENT RELEASE REPORT

January – December 2001

Constellation Nuclear

Nine Mile Point Nuclear Station

A Member of the Constellation Energy Group

NINE MILE POINT NUCLEAR STATION - UNIT 2

RADIOACTIVE EFFLUENT RELEASE REPORT

JANUARY - DECEMBER 2001

SUPPLEMENTAL INFORMATION

Facility: **Nine Mile Point Unit #2** **Licensee:** **Nine Mile Point Nuclear Station, LLC**

1. **TECHNICAL SPECIFICATION PROGRAM** - (ODCM Limits – Radioactive Effluent Controls Program)

A) FISSION AND ACTIVATION GASES

1. The dose rate limit of noble gases released in gaseous effluents from the site to areas at or beyond the site boundary shall be less than or equal to 500 mrem/year to the whole body and less than or equal to 3000 mrem/year to the skin.

2. The air dose from noble gases released in gaseous effluents from Nine Mile Point Unit 2 to areas at or beyond the site boundary shall be limited during any calendar quarter to less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation, and during any calendar year to less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

B&C) TRITIUM, IODINES AND PARTICULATES, HALF LIVES > 8 DAYS

1. The dose rate limit of Iodine-131, Iodine-133, Tritium and all radionuclides in particulate form with half-lives greater than eight days, released in gaseous effluents from the site to areas at or beyond the site boundary shall be less than or equal to 1500 mrem/year to any organ.

2. 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 eight days in gaseous effluents released from Nine Mile Point Unit 2 to areas at or beyond the site boundary shall be limited during any calendar quarter to less than or equal to 7.5 mrem to any organ and, during any calendar year to less than or equal to 15 mrem to any organ.

D) LIQUID EFFLUENTS

1. The concentration of radioactive material released in the liquid effluents to unrestricted areas shall be limited to ten times the concentrations specified in 10CFR Part 20.1001-20.2402, Appendix B, Table 2, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2E-04 microcuries/ml total activity.

2. The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released from Nine Mile Point Unit 2 to unrestricted areas shall be limited during any calendar quarter to less than or equal to 1.5 mrem to the whole body and to less than or equal to 5 mrem to any organ, and during any calendar year to less than or equal to 3 mrem to the whole body and to less than or equal to 10 mrem to any organ.

2. MEASUREMENTS AND APPROXIMATIONS OF TOTAL RADIOACTIVITY

Described below are the methods used to measure or approximate the total radioactivity and radionuclide composition in effluents.

A) FISSION AND ACTIVATION GASES

Noble gas effluent activity is determined by on-line gamma spectroscopic monitoring (intrinsic germanium crystal) of an isokinetic sample stream.

B) IODINES

Iodine effluent activity is determined by gamma spectroscopic analysis (at least weekly) of charcoal cartridges sampled from an isokinetic sample stream.

C) PARTICULATES

Activity released from the main stack and the combined Radwaste/Reactor Building vent is determined by gamma spectroscopic analysis (at least weekly) of particulate filters sampled from an isokinetic sample stream and composite analysis of the filters for non-gamma emitters.

D) TRITIUM

Tritium effluent activity is measured by liquid scintillation or gas proportional counting of monthly samples taken with an air sparging/water trap apparatus.

E) LIQUID EFFLUENTS

Isotopic contents of liquid effluents are determined by isotopic analysis of a representative sample of each batch and composite analysis of non-gamma emitters.

F) SOLID EFFLUENTS

Isotopic contents of waste shipments are determined by gamma spectroscopy analyses of a representative sample of each batch. Scaling factors established from primary composite sample analyses conducted off-site are applied, where appropriate, to find estimated concentration of non-gamma emitters. For low activity trash shipments, curie content is estimated by dose rate measurement and application of appropriate scaling factors.

ATTACHMENT 1

Summary Data

Unit 1 ___ Unit 2 X Reporting Period January – December 2001

Liquid Effluents:

10CFR20.1001-20.2402, Appendix B, Table 2, Column 2[1]

Average MEC - µCi/ml (Qtr. 1)	=	8.55E-03	Average MEC - µCi/ml (Qtr. 3)	=	7.84E-03
Average MEC - µCi/ml (Qtr. 2)	=	7.49E-03	Average MEC - µCi/ml (Qtr. 4)	=	5.95E-03

Average Energy (Fission and Activation gases – Mev):

Qtr. 1 :	$\bar{E}\gamma$	= 3.22E-01	$\bar{E}_\beta$	= 4.26E-01
Qtr. 2 :	$\bar{E}\gamma$	= 2.80E-01	$\bar{E}_\beta$	= 3.33E-01
Qtr. 3 :	$\bar{E}\gamma$	= 6.66E-01	$\bar{E}_\beta$	= 1.04E+00
Qtr. 4 :	$\bar{E}\gamma$	= 7.86E-01	$\bar{E}_\beta$	= 8.91E-01

Liquid:

Number of batch releases	:	89			
Total time period for batch releases (hrs)	:	2.85E+02			
Maximum time period for a batch release (hrs)	:	3.32E+00			
Average time period for a batch release (hrs)	:	3.20E+00			
Minimum time period for a batch release (hrs)	:	1.88E-02			
Total volume of water used to dilute the liquid effluent during the release		1st	2nd	3rd	4th
period (L)	:	1.66E+08	3.85E+08	7.79E+08	6.58E+08
Total volume of water used to dilute the liquid effluent during reporting		1st	2nd	3rd	4th
Period (L)	:	1.23E+10	1.37E+10	1.46E+10	1.39E+10

Gaseous (Emergency Condenser Vent): "Not Applicable for Unit 2"

Number of batch releases	:	N/A
Total time period for batch releases (hrs)	:	N/A
Maximum time period for a batch release (hrs)	:	N/A
Average time period for a batch release (hrs)	:	N/A
Minimum time period for a batch release (hrs)	:	N/A

Gaseous (Primary Containment Purge):

Number of batch releases	:	14
Total time period for batch releases (hrs)	:	3.57E+02
Maximum time period for a batch release (hrs)	:	5.58E+01
Average time period for a batch release (hrs)	:	2.55E+01
Minimum time period for a batch release (hrs)		4.50E+00

[1] Improved Technical Specifications limit the concentration of radioactive material released in the liquid effluents to unrestricted areas to ten times the concentrations specified in 10CFR20.1001-20.2402, Appendix B, Table 2, Column 2. Maximum Effluent Concentrations (MEC) numerically equal to ten times the 10CFR20.1001-20.2402 concentrations were adopted to evaluate liquid effluents.

ATTACHMENT 1

Summary Data

Unit 1 ___ Unit 2 X — Reporting Period **January – December 2001**

Abnormal Releases: There were no abnormal releases during this report period.

A. Liquids:

Number of releases	0
Total activity released	N/A Ci

B. Gaseous:

Number of releases	0
Total activity released	N/A Ci

ATTACHMENT 2

Unit 1 ___ Unit 2 _X_ Reporting Period January – December 2001

GASEOUS EFFLUENTS – SUMMATION OF ALL RELEASES, ELEVATED AND GROUND LEVEL

			1st QUARTER	2nd QUARTER	3rd QUARTER	4th QUARTER	EST. TOTAL ERROR, %
A.	Fission & Activation gases						
	1. Total release	Ci	1.02E+00	1.35E+00	7.67E+00	1.62E+01	5.00E+01
	2. Average release rate	μCi/sec	1.31E-01	1.72E-01	9.65E-01	2.03E+00	
B.	Iodines						
	1. Total Iodine-131	Ci	1.47E-05	9.23E-05	1.93E-05	4.53E-04	3.00E+01
	2. Average release rate for period	μCi/sec	1.87E-06	1.17E-05	2.45E-06	5.75E-05	
C.	Particulates[1]						
	1. Particulates with half-lives >8 days	Ci	6.17E-04	3.52E-03	4.91E-04	1.74E-03	3.00E+01
	2. Average release rate for period	μCi/sec	7.85E-05	4.67E-04	6.25E-05	2.22E-04	
	3. Gross alpha radioactivity	Ci	1.18E-05	2.46E-05	3.76E-05	2.09E-05	2.50E+01
D.	Tritium[1]						
	1. Total release	Ci	4.92E+00	5.49E+00	9.00E+00	9.30E+00	5.00E+01
	2. Average release rate for period	μCi/sec	6.26E-01	7.28E-01	1.15E+00	1.18E+00	
E.	Percent of Tech. Spec. Limits						
	Fission and Activation Gases						
	Percent of Quarterly Gamma Air Dose Limit (5 mR)	%	7.28E-04	8.10E-04	1.18E-02	2.94E-02	
	Percent of Quarterly Beta Air Dose Limit (10 mrad)	%	4.80E-05	4.97E-05	8.84E-04	1.59E-03	
	Percent of Annual Gamma Air Dose Limit to Date (10 mR)	%	3.64E-04	7.69E-04	6.67E-03	2.14E-02	
	Percent of Annual Beta Air Dose Limit to Date (20 mrad)	%	2.40E-05	4.88E-05	4.91E-04	1.29E-03	
	Percent of Whole Body Dose Rate Limit (500 mrem/yr)	%	2.82E-05	3.13E-05	4.49E-04	1.13E-03	
	Percent of Skin Dose Rate Limit (3000 mrem/yr)	%	6.00E-06	6.57E-06	9.76E-05	2.35E-04	
	Tritium, Iodines, and Particulates[1] (with half-lives greater than 8 days)						
	Percent of Quarterly Dose Limit (7.5 mrem)	%	7.69E-03	4.03E-02	9.18E-03	1.19E-01	
	Percent of Annual Dose Limit (15 mrem)	%	3.87E-03	2.42E-02	2.88E-02	1.16E-01	
	Percent of Organ Dose Rate Limit (1500 mrem/yr)	%	1.54E-04	8.43E-04	1.84E-04	2.39E-03	

[1] Tritium, Iron-55, and Strontium results for the fourth quarter were not received from the off-site vendor at the time of this report. These values include estimates, and actual numbers will be provided in the next Radioactive Effluent Release Report.

ATTACHMENT 3

Unit 1 ___ Unit 2 X

Reporting Period January – December 2001

GASEOUS EFFLUENTS – ELEVATED RELEASE

Nuclides Released			CONTINUOUS MODE[3] 1st QUARTER	2nd QUARTER	3rd QUARTER	4th QUARTER
1.	Fission Gases[1]					
	Argon-41	Ci	1.79E-02	1.85E-02	1.16E-02	3.99E-01
	Krypton-85	Ci	**	**	**	**
	Krypton-85m	Ci	6.68E-01	8.68E-01	4.75E-01	1.51E+00
	Krypton-87	Ci	**	1.93E-02	2.18E-01	2.20E-01
	Krypton-88	Ci	**	2.74E-02	7.20E-01	3.08E+00
	Xenon-127	Ci	**	**	**	**
	Xenon-131m	Ci	**	**	**	**
	Xenon-133	Ci	**	**	**	8.70E-02
	Xenon-133m	Ci	**	**	**	**
	Xenon-135	Ci	4.57E-02	2.00E-01	5.36E-02	4.51E-01
	Xenon-135m	Ci	6.87E-02	1.30E-01	4.14E-01	8.31E-01
	Xenon-137	Ci	9.99E-02	5.66E-02	3.62E+00	6.11E+00
	Xenon-138	Ci	1.17E-01	2.53E-02	2.16E+00	3.48E+00
2.	Iodines[1]					
	Iodine-131	Ci	1.47E-05	9.23E-05	1.93E-05	4.53E-04
	Iodine-133	Ci	8.49E-05	1.93E-05	4.10E-05	4.28E-03
	Iodine-135	Ci	**	**	**	**
3.	Particulates[1,2]					
	Strontium-89	Ci	**	**	**	1.39E-05
	Strontium-90	Ci	**	**	**	4.54E-06
	Cesium-134	Ci	**	**	**	**
	Cesium-137	Ci	**	**	**	**
	Cobalt-60	Ci	1.42E-05	2.01E-05	3.07E-05	3.49E-05
	Cobalt-58	Ci	**	**	**	**
	Manganese-54	Ci	4.49E-06	7.52E-06	**	2.80E-05
	Barium-Lanthanum-140	Ci	**	**	**	**
	Antimony-125	Ci	**	**	**	**
	Niobium-95	Ci	**	**	**	**
	Cerium-141	Ci	**	**	**	**
	Cerium-144	Ci	**	**	**	**
	Iron-59	Ci	**	**	**	**
	Cesium-136	Ci	**	**	**	**
	Chromium-51	Ci	**	**	**	**
	Zinc-65	Ci	**	5.01E-06	**	**
	Iron-55	Ci	**	4.15E-05	**	4.16E-05
	Molybdenum-99	Ci	1.71E-06	**	**	**
	Silver-110m	Ci	2.04E-06	**	**	**
		Ci				
4.	Tritium[2]	Ci	3.49E+00	4.58E+00	7.60E+00	7.90E+00

[1] Concentrations less than the lower limit of detection of the counting system used are indicated with a double asterisk. A lower limit of detection of 1.00E-04 µCi/ml for required noble gases, 1.00E-11 µCi/ml for required particulates, 1.00E-12 µCi/ml for required Iodines, and 1.00E-06 µCi/ml for Tritium, as required by the Off-Site Dose Calculation Manual (ODCM), has been verified.

[2] Tritium, Iron-55, and Strontium results for the fourth quarter were not received from the off-site vendor at the time of this report. These values include estimates. Actual values will be included in the next Radioactive Effluent Release Report.

[3] Contributions from purges are included.

ATTACHMENT 4

Unit 1 ___ Unit 2 X

Reporting Period January – December 2001

GASEOUS EFFLUENTS – GROUND LEVEL RELEASES

			CONTINUOUS MODE			
			1st QUARTER	2nd QUARTER	3rd QUARTER	4th QUARTER
1.	Fission Gases[1]					
	Argon-41	Ci	**	**	**	**
	Krypton-85	Ci	**	**	**	**
	Krypton-85m	Ci	**	**	**	**
	Krypton-87	Ci	**	**	**	**
	Krypton-88	Ci	**	**	**	**
	Xenon-127	Ci	**	**	**	**
	Xenon-131m	Ci	**	**	**	**
	Xenon-133	Ci	**	**	**	**
	Xenon-133m	Ci	**	**	**	**
	Xenon-135	Ci	**	**	**	**
	Xenon-135m	Ci	**	**	**	**
	Xenon-137	Ci	**	**	**	**
	Xenon-138	Ci	**	**	**	**
2.	Iodines[1]					
	Iodine-131	Ci	**	**	**	**
	Iodine-133	Ci	**	**	**	**
	Iodine-135	Ci	**	**	**	**
3.	Particulates[1,2]					
	Strontium-89	Ci	**	**	**	2.67E-05
	Strontium-90	Ci	**	**	**	3.29E-05
	Cesium-134	Ci	**	**	**	**
	Cesium-137	Ci	4.47E-06	**	**	**
	Cobalt-60	Ci	1.08E-04	6.34E-04	1.18E-04	1.10E-04
	Cobalt-58	Ci	6.13E-06	7.21E-05	**	2.36E-05
	Manganese-54	Ci	5.92E-05	4.77E-04	2.75E-05	1.33E-04
	Barium-Lanthanum-140	Ci	**	**	**	**
	Antimony-125	Ci	**	**	**	**
	Niobium-95	Ci	**	**	**	**
	Cerium-141	Ci	**	**	**	**
	Cerium-144	Ci	**	**	**	**
	Iron-59	Ci	**	1.85E-04	**	5.28E-05
	Cesium-136	Ci	**	**	**	**
	Chromium-51	Ci	**	4.09E-04	**	7.73E-05
	Zinc-65	Ci	**	1.08E-04	**	**
	Iron-55	Ci	3.98E-04	1.57E-03	3.15E-04	1.16E-03
	Molybdenum-99	Ci	**	**	**	**
	Silver-110m	Ci	1.86E-05	**	**	**
		Ci				
4.	Tritium[2]	Ci	1.43E+00	9.16E-01	1.40E+00	1.40E+00

[1] Concentrations less than the lower limit of detection of the counting system used are indicated with a double asterisk. A lower limit of detection of 1.00E-04 μCi/ml for required noble gases, 1.00E-11 μCi/ml for required particulates, 1.00E-12 μCi/ml for required Iodines, and 1.00E-06 μCi/ml for Tritium, as required by the Off-Site Dose Calculation Manual (ODCM), has been verified.

[2] Tritium, Iron-55 and Strontium 89 and 90 results for the fourth quarter were not received from the off-site vendor at the time of this report. These values include estimates, and actual values will be included in the next Radioactive Effluent Release Report.

Unit 1 ___ Unit 2 X Reporting Period January - December 2001

GASEOUS EFFLUENTS - GROUND LEVEL RELEASES

BATCH MODE

There were no batch releases during the reporting period.

			1st QUARTER	2nd QUARTER	3rd QUARTER	4th QUARTER
1.	Fission Gases[1]					
	Argon-41	Ci				
	Krypton-85	Ci				
	Krypton-85m	Ci				
	Krypton-87	Ci				
	Krypton-88	Ci				
	Xenon-127	Ci				
	Xenon-131m	Ci				
	Xenon-133	Ci				
	Xenon-133m	Ci				
	Xenon-135	Ci				
	Xenon-135m	Ci				
	Xenon-137	Ci				
	Xenon-138	Ci				
2.	Iodines[1]					
	Iodine-131	Ci				
	Iodine-133	Ci				
	Iodine-135	Ci				
3.	Particulates[1,2]					
	Strontium-89	Ci				
	Strontium-90	Ci				
	Cesium-134	Ci				
	Cesium-137	Ci				
	Cobalt-60	Ci				
	Cobalt-58	Ci				
	Manganese-54	Ci				
	Barium-Lanthanum-140	Ci				
	Antimony-125	Ci				
	Niobium-95	Ci				
	Cerium-141	Ci				
	Cerium-144	Ci				
	Iron-59	Ci				
	Cesium-136	Ci				
	Chromium-51	Ci				
	Zinc-65	Ci				
	Iron-55	Ci				
	Molybdenum-99	Ci				
	Silver-110m	Ci				
4.	Tritium[2]	Ci				

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1 Concentrations less than the lower limit of detection of the counting system used are indicated with a double asterisk. A lower limit of detection of 1.00E-04 μCi/ml for required noble gases, 1.00E-11 μCi/ml for required particulates, 1.00E-12 μCi/ml for required Iodines, and 1.00E-06 μCi/ml for Tritium, as required by the Off-Site Dose Calculation Manual (ODCM), has been verified.

2 Tritium, Iron-55 and Strontium 89 and 90 results for the fourth quarter were not received from the off-site vendor at the time of this report. These values include estimates, and actual values will be included in the next Radioactive Effluent Release Report.

ATTACHMENT 5

Unit 1 ___ Unit 2 X

Reporting Period January – December 2001

LIQUID EFFLUENTS – SUMMATION OF ALL RELEASES

			1st QUARTER	2nd QUARTER	3rd QUARTER	4th QUARTER	EST. TOTAL ERROR, %
A.	Fission & Activation Products[1]						
	1. Total release (not including Tritium, gases, alpha)	Ci	4.07E-03	3.04E-02	3.48E-02	7.04E-02	5.00E+01
	2. Average diluted concentration during reporting period	μCi/ml	3.31E-10	2.22E-09	2.38E-09	5.07E-09	
B.	Tritium[1]						
	1. Total release	Ci	1.90E+00	6.81E+00	1.16E+01	1.08E+01	5.00E+01
	2. Average diluted concentration during reporting period	μCi/ml	1.54E-07	4.97E-07	7.90E-07	7.77E-07	
C.	Dissolved and Entrained Gases[3]						
	1. Total release	Ci	**	**	**	**	5.00E+01
	2. Average diluted concentration during reporting period	μCi/ml	**	**	**	**	
D.	Gross Alpha Radioactivity[3]						
	1. Total release	Ci	**	**	**	6.03E-05	5.00E+01
E.	Volumes						
	1. Prior to dilution	Liters	6.19E+05	1.50E+06	3.09E+06	2.56E+06	5.00E+01
	2. Volume of dilution water used during release period	Liters	1.66E+08	3.85E+08	7.79E+08	6.58E+08	5.00E+01
	3. Volume of dilution water available during reporting period:	Liters	1.23E+10	1.37E+10	1.46E+10	1.39E+10	5.00E+01
F.	Percent of Technical Specification Limits	%	2.00E-02	8.25E-02	1.52E-01	2.19E-01	
	Percent of Quarterly Whole Body Dose Limit (1.5 mrem)	%	2.78E-02	1.42E-01	1.60E-01	3.52E-01	
	Percent of Quarterly Organ Dose Limit (5 mrem)	%	1.00E-02	5.15E-02	1.28E-01	2.46E-01	
	Percent of Annual Whole Body Dose Limit to Date (3 mrem)	%	1.39E-02	8.46E-02	1.62E-01	3.39E-01	
	Percent of Annual Organ Dose Limit to Date (10 mrem)	%	1.81E-03	6.67E-03	1.01E-02	1.31E-02	
	Percent of 10CFR20 Concentration Limit[2,4]	%	**	**	**	**	
	Percent of Dissolved or Entrained Noble Gas Limit (2.00E-04 μCi/ml)[3,4]						

1 Iron-55, Strontium 89 and 90 and Tritium results for the fourth quarter were not received from the off-site vendor at the time of this report. These values include estimates, and actual values will be included in the next Radioactive Effluent Release Report.

2 The percent of 10CFR20 concentration limit is based on the average concentration during the quarter.

3 Concentrations less than the lower limit of detection of the counting system used are indicated with a double asterisk. A lower limit of detection of 5.00E-07 μCi/ml for required gamma emitting nuclides, 1.00E-05 μCi/ml for required dissolved and entrained noble gases and Tritium, 5.00E-08 μCi/ml for Sr-89/90, 1.00E-06 μCi/ml for Fe-55 and 1.00E-07 μCi/ml for gross alpha radioactivity, as required by the Off-Site Dose Calculation Manual (ODCM), has been verified.

4 Improved Technical Specifications limit the concentration of radioactive material released in the liquid effluents to unrestricted areas to ten times the concentrations specified in 10CFR20.1001-20.2402, Appendix B, Table 2, Column 2. Maximum Effluent Concentrations (MEC) numerically equal to ten times the 10CFR20.1001-20.2402 concentrations were adopted to evaluate liquid effluents.

ATTACHMENT 5

Unit 1 ___ Unit 2 X

Reporting Period January - December 2001

LIQUID EFFLUENTS RELEASED

Nuclides Released[1,2]		BATCH MODE[3] 1st QUARTER	2nd QUARTER	3rd QUARTER	4th QUARTER
Silver-110m	Ci	1.86E-05	6.47E-04	1.33E-03	4.11E-04
Arsenic-76	Ci	**	7.43E-05	5.46E-05	3.02E-05
Gold-199	Ci	2.07E-05	2.44E-04	6.78E-04	1.37E-04
Barium-140	Ci	**	**	**	**
Cerium-141	Ci	**	**	**	**
Cerium-144	Ci	**	**	**	**
Cobalt-58	Ci	2.45E-05	3.17E-04	2.31E-04	1.27E-03
Cobalt-60	Ci	5.20E-04	3.60E-03	5.75E-03	1.52E-02
Chromium-51	Ci	1.82E-04	2.46E-03	1.07E-03	3.17E-03
Cesium-134	Ci	**	**	**	**
Cesium-136	Ci	**	2.72E-05	**	**
Cesium-137	Ci	**	**	**	**
Copper-64	Ci	6.99E-06	6.22E-04	5.31E-04	1.70E-03
Iron-55	Ci	2.57E-04	2.82E-03	**	4.93E-03
Iron-59	Ci	1.31E-04	2.43E-03	2.05E-03	4.38E-03
Iodine-131	Ci	**	**	**	**
Iodine-132	Ci	**	**	**	**
Iodine-133	Ci	**	**	**	**
Lanthanum-140	Ci	**	**	**	**
Manganese-54	Ci	2.70E-03	1.64E-02	1.25E-02	3.67E-02
Manganese-56	Ci	**	**	**	**
Molybdenum-99	Ci	**	1.12E-05	**	**
Sodium-24	Ci	**	**	**	**
Niobium-95	Ci	**	**	**	1.43E-05
Nickel-65	Ci	**	**	**	**
Neptunium-239	Ci	**	**	**	**
Antimony-124	Ci	**	9.20E-05	3.15E-05	3.81E-04
Strontium-89	Ci	**	**	**	5.08E-05
Strontium-90	Ci	**	**	**	3.50E-05
Strontium-92	Ci	**	**	**	**
Technecium-99m	Ci	**	1.18E-05	**	**
Tellurium-132	Ci	**	**	**	**
Tungsten-187	Ci	**	**	**	**
Zinc-65	Ci	2.11E-04	6.94E-04	1.70E-03	1.99E-03
Zinc-69m	Ci	**	1.14E-05	2.65E-05	**
Zirconium-95	Ci	**	**	**	**
Zirconium-97	Ci	**	**	**	**
Dissolved or Entrained Gases[1]	Ci	**	**	**	**
Tritium[2]	Ci	1.90E+00	6.81E+00	1.16E+01	1.08E+01

1 Concentrations less than the lower limit of detection of the counting system used are indicated with a double asterisk. A lower limit of detection of 5.00E-07 μCi/ml for required gamma emitting nuclides, 1.00E-05 μCi/ml for required dissolved and entrained noble gases and Tritium, 5.00E-08 μCi/ml for Sr-89/90, 1.00E-06 μCi/ml for Fe-55 and 1.00E-07 μCi/ml for gross alpha radioactivity, as required by the Off-Site Dose Calculation Manual (ODCM), has been verified.

2 Iron-55, Strontium 89 and 90 and Tritium results for the fourth quarter were not received from the off-site vendor at the time of this report. These values include estimates, and actual values will be included in the next Radioactive Effluent Release Report.

3 No continuous mode releases occurred during the reporting period.

ATTACHMENT 6

Unit 1 ___ Unit 2 X

Reporting Period January – December 2001

SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

A. TYPE	Volume (m^3) Class			Activity[1] (Ci) Class		
	A	B	C	A	B	C
1. Spent Resins (Dewatered)	8.40E+01	0	0	1.46E+02	0	0
2. Dry Active Waste	0	0	0	0	0	0
3. Irradiated Components, Control Rods, etc.	0	3.41E+00	3.25E+00	0	2.17E+01	3.97E+04
4. Other: (to Vendor for Processing or Consolidation)						
a. Dry Active Waste (Compactible and Non-Compactible)	1.76E+02	0	0	4.06E+00	0	0
b. Spent Resins (Dewatered)	3.50E+01	0	0	8.32E+01	0	0
c. DAW – Contaminated Equipment (Non-compactible)	8.34E+01	0	0	4.30E-01	0	0

[1] The estimated total error is 5.00E+01%.

ATTACHMENT 6

Unit 1 ___ Unit 2 X

Reporting Period January – December 2001

SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

A.1 TYPE	Container	Package	Solidification Agent
1. Spent Resins (Dewatered)	HIC – Poly w/ steel shell	STP	None
	HIC – Poly	STP	None
2. Dry Active Waste (Compactible and Non-Compactible)	N/A	N/A	N/A
3. Irradiated Components, Control Rods, etc.	Steel Liner	Type B	None
	HIC - Poly	Type B	None
4. Other: (To Vendor for Processing or Consolidation)			
a. Dry Active Waste (Compactible and Non-Compactible)	Metal Box	STP	None
	HIC	STP	None
b. Spent Resins (Dewatered)	HIC	Type A	None
c. DAW – Contaminated Equipment (Non-Compactible)	Metal Box	STP	None

Unit 1 ___ Unit 2 X

Reporting Period January – December 2001

SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

A.2 ESTIMATE OF MAJOR NUCLIDE COMPOSITION (BY TYPE OF WASTE)

1. Spent Resins (Dewatered):

Nuclide	Percent
(1) Fe-55	6.63E+01
(2) Co-60	1.50E+01
(3) Zn-65	9.29E+00
(4) Mn-54	8.37E+00
(5) Other	1.04E+00

2. Dry Compressible Waste:

Nuclide	Percent

3. Irradiated Components, Control Rods, etc.:

Nuclide	Percent
(1) Co-60	5.87E+01
(2) Fe-55	3.73E+01
(3) Ni-63	3.38E+00
(4) Other	6.20E-01

4. Other: (to Vendor for Processing or Consolidation)

a. Dry Active Waste (Compactible and Non-Compactible)

Nuclide	Percent
(1) Fe-55	4.70E+01
(2) Zn-65	2.56E+01
(3) Co-60	2.13E+01
(4) Mn-54	4.40E+00
(5) Other	1.70E+00

b. Spent Resins (Dewatered)

Nuclide	Percent
(1) Fe-55	6.97E+01
(2) Co-60	1.59E+01
(3) Mn-54	7.06E+00
(4) Zn-65	6.05E+00
(5) Other	1.29E+00

c. DAW - Contaminated Equipment (Non-Compactible)

Nuclide	Percent
(1) Co-60	5.47E+01
(2) Fe-55	2.43E+01
(3) Zn-65	1.35E+01
(4) Ni-63	4.16E+00
(5) Mn-54	1.87E+00
(6) Other	1.47E+00

ATTACHMENT 6

Unit 1 ___ Unit 2 X

Reporting Period January - December 2001

SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

A.3. SOLID WASTE DISPOSITION

Number of Shipments	Mode of Transportation	Destination
9	Truck	GTS Duratek Oak Ridge, TN
1	Truck	ATG Richland Corporation Richland, WA
6	Truck	ATG Catalytics, LLC Kingston, TN
15	Truck	Barnwell Waste Management Facility Barnwell, SC
4	Truck	Chem Nuclear Consolidation Facility Barnwell, SC

B. IRRADIATED FUEL SHIPMENTS (DISPOSITION): There were no shipments.

Number of Shipments	Mode of Transportation	Destination
0	N/A	N/A

Unit 1 ___ Unit 2 X Reporting Period **January – December 2001**

SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

C. SOLID WASTE SHIPPED OFF-SITE TO VENDORS FOR PROCESSING AND SUBSEQUENT BURIAL
Below is a summary of NMP-2 radwaste buried by vendor facilities during **January – December 2001**. These totals were reported separately from "10CFR61 Solid Waste Shipped for Burial" since (a) waste classification and burial was performed by the vendors, and (b) Improved Technical Specification (ITS) Section 5.6.3 requires reporting of "information for each class of solid waste (as defined by 10CFR61) shipped off-site during the reporting period." The following data represents the actual shipments made from the off-site vendors of our radwaste (e.g., non-compacted trash, dry non-compressible waste, scrap metal, and resins) that was processed and commingled prior to burial.

C.1. TYPE OF WASTE – Non-compacted trash, dry non-compressible waste, scrap metals, and resins processed by vendor facilities prior to burial.

Burial Volume (m^3)	Activity (Ci)	Est. Total Error, %
1.78E+01	4.23E-01	5.00E+01

C.2 ESTIMATE OF MAJOR NUCLIDE COMPOSITION

Nuclide	Percent
(1) Co-60	4.03E+01
(2) Mn-54	1.90E+01
(3) Zn-65	1.61E+01
(4) Fe-55	1.39E+01
(5) Cr-51	7.52E+00
(6) Co-58	1.43E+00
(7) Other	1.75E+00

C.3 SOLID WASTE DISPOSITION

Number of Shipments	Mode of Transportation	Destination
43	Truck	Envirocare, UT

ATTACHMENT 6

Unit 1 ____ Unit 2 X

Reporting Period January - December 2001

SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

D. SEWAGE WASTES SHIPPED TO A TREATMENT FACILITY FOR PROCESSING AND BURIAL

Below is a summary of the sewage sludge, which was removed from the site sanitary treatment facility and transferred to a municipal sewage treatment facility, for subsequent drying and disposal to a landfill. This is a site release and therefore includes the results from both Unit 1 and Unit 2 activities.

D. 1 TYPE OF WASTE – Sewage Sludge	Burial Volume (m^3)	Activity (Ci)	Est. Total Error, %
	2.72E+01	1.72E-03	5.00E+01

D. 2 ESTIMATE OF MAJOR NUCLIDE COMPOSITION

Nuclide	Percent
(1) H-3	5.63E+01
(2) Ni-63	1.48E+01
(3) C-14	1.31E+01
(4) Tc-99	7.88E+00
(5) I-129	7.13E+00
(6) Other	7.90E-01

D. 3 SOLID WASTE DISPOSITION

Number of Shipments	Mode of Transportation	Destination
1	**Truck**	**Landfill**

ATTACHMENT 7

Unit 1 ___ Unit 2 X Reporting Period January - December 2001

SUMMARY OF CHANGES TO THE OFF-SITE DOSE CALCULATION MANUAL (ODCM)

The Unit 2 Off-Site Dose Calculation Manual (ODCM) Revision 22 was implemented in December 2001. Administrative changes were made to reflect a change in site ownership, editorial changes for clarification, and a new control location for milk sampling. The ODCM changes do not reduce the overall conformance of existing criteria in accordance with Technical Specifications. A copy of the ODCM, Revision 22 is attached and below is a summary of the changes accepted by the Station Operations Review Committee on December 11, 2001.

Old Page #	New Page #	New/Amended Section #	Description of Change	Reason for Change
I 1.0-1	I 1.0-1	1.0	In the definition for MEMBER(S) OF THE PUBLIC, replaced "the Niagara Mohawk Power Corporation, the Nine Mile Point Unit 2 co-tenants, the New York State Power Authority" with "the owners and operators of the Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant".	Administrative
I 1.0-2	I 1.0-2	1.0	In the definition for SITE BOUNDARY, replaced "the Niagara Mohawk Power Corporation or the New York Power Authority" with "the owners and operators of the Nine Mile Point Station and James A. Fitzpatrick Nuclear Power Plant".	Administrative
I 1.0-2	I 1.0-2	1.0	In the definition for UNRESTRICTED AREA, replaced "the Niagara Mohawk Power Corporation and the New York State Power Authority" with "the owners and operators of the Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant".	Administrative
I 1.0-4	I. 1.0-4	1.0	Replaced the words "Niagara Mohawk Power Corporation" and "Power Authority State of New York" with "Nine Mile Point Nuclear Station, LLC*" and "ENTERGY", respectively. Also added note: "*Niagara Mohawk Power Corporation retains ownership in certain transmission line and switchyard facilities within the exclusion area boundary. Access and usage are controlled by Nine Mile Point Nuclear Station, LLC by Agreement".	Administrative
IB 3.3-2	IB 3.3-2	IB 3.3-2	Added words to clarify that even though the Offgas System Noble Gas Activity Monitor is included in Table D 3.3.2-1, "Radioactive Gaseous Effluent Monitoring Instrumentation", it is not an effluent monitor. Its alert setpoint is based on ITS SR 3.7.4.1 and its trip setpoint is based on 10CFR100.	Administrative
I 4.1-1 N/A	I 4.1-1 and I 4.1-1a	D 4.1.2 and D 4.1.3	Added REPORTING REQUIREMENTS Section and D 4.1.2, "Annual Radiological Environmental Operating Reports", and Section D 4.1.3, "Radiological Effluent Release Report" which were omitted in the ITS implementation.	Administrative
II 11	II 11	2.1.1	Added "in accordance with Technical Specification 5.5.4.g" at the end of the 1st sentence of Section 2.1.1.	Administrative
II 11	II 11	2.1.1	Added "in accordance with Technical Specification 3.7.4" at the end of the last sentence of Section 2.1.1.	Administrative
II 15	II 15	2.2	Replaced "per 10CFR20" with "per Technical Specification 5.5.4.g" at the end of the 2nd paragraph of Section 2.2.	Administrative
II 29	II 29	4.1	Section 4.1, combined the 2nd and 3rd sentences in the 1st paragraph for clarification.	Editorial

ATTACHMENT 7

Unit 1 ___ Unit 2 X Reporting Period January – December 2001

SUMMARY OF CHANGES TO THE OFF-SITE DOSE CALCULATION MANUAL (ODCM)

Old Page #	New Page #	New/Amended Section #	Description of Change	Reason for Change
II 29	II 29	4.1	Reworded the 2nd paragraph of Section 4.1 to clarify that the calculated dispersion and deposition parameters are averages, and to refer to Controls 3.5.1 and 3.5.2 and the Radiological Environmental Monitoring Program.	Administrative
II 33 through II 36	II 33 through II 36	Table D 2-2 through Table D 2-5	Table D 2-2 through D 2-5; Added adult, teen, child and infant liquid dose factors for Co-57, I-135 and Nb-95m, per DER 2000-4230, using the methodology and equations given in Appendix A to the ODCM.	Administrative
II 63	II 63	Table D 5.1	Map Location 45, Milk Location #7 was deleted.	Administrative
II 63	II 63	Table D 5.1	"★" and the corresponding footnote have been deleted as they are no longer necessary.	Administrative
II 63	II 63	Table D 5.1	Map Location 47, Milk Location #65 was deleted.	Administrative
II 63	II 63	Table D.5.1	Map Location 73, Milk Control Location (Woodworth), 13.9 mi @ 234° SW, was replaced by Map Location 77, Milk Location (Summerville), 13.9 mi @ 191° SSW, as Mr. Woodworth retired and sold his herd.	Administrative ODCM Table D3.5.1-1
II 107	II 107	Figure 5.1-2	Figure 5.1-2 (page 1 of 2) was revised to delete Map Location 45.	Administrative
II 108	II 108	Figure 5.1-2a	Figure 5.1-2 (page 2 of 2) was revised to delete Map Locations 47 and 73, and add Map Location 77.	Administrative

ATTACHMENT 8

Unit 1 ___ Unit 2 _X_	Reporting Period January - December 2001

SUMMARY OF CHANGES TO THE PROCESS CONTROL PROGRAM (RPCP)

The Unit 2 Radwaste Process Control Program (RPCP) Revision 5 was implemented in February 2001. Administrative changes were made to reflect changes in training requirements and clarification of existing procedure requirements. The RPCP changes do not reduce the overall conformance in accordance with Technical Specifications. A copy of the RPCP, Revision 5 is attached and below is a summary of the changes accepted by the Station Operations Review Committee.

Old Page #	New Page #	New/Amended Section #	Description of Change	Reason for Change
All	All	N/A	Changed page format, font, and font size.	Editorial
6	7	3.2.9	Replaced "bi-annual recertification" with "recertification every two years".	Clarification
11	12	3.6.2.e	Replaced "Retraining of Radwaste Operator personnel on an annual basis" with "Continuing Training of Radwaste Operator personnel on a cyclic basis (i.e., every 4 years)".	Administrative

ATTACHMENT 9

Unit 1 ___ Unit 2 _X_	Reporting Period January – December 2001
SUMMARY OF INOPERABLE MONITORS	
There were no inoperable monitors for a period greater than 30 days during the reporting period.	

ATTACHMENT 10

Doses to Members of the Public Due To Their Activities Inside the Site Boundary

ATTACHMENT 10

RADIOACTIVE EFFLUENT RELEASE REPORT (2001) NINE MILE POINT NUCLEAR STATION UNIT 2 DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES INSIDE THE SITE BOUNDARY

JANUARY – DECEMBER 2001

Doses to members of the public (as defined by the Unit 2 Off-Site Dose Calculation Manual (ODCM)) from the operation of the Nine Mile Point Unit 2 (NMP2) facility as a result of activity inside the site boundary are based on activities at the Energy Center located approximately one quarter mile west of Nine Mile Point Unit 1 (NMP1). This facility was open to the public and offered educational information, summer picnicking activities and fishing; however, since the events of September 11, 2001, access has been restricted. Any possible doses received by a member of the public by utilizing the private road that transverses the east and west site boundaries prior to September 11, 2001 are not considered here since it takes a matter of minutes to travel the distance.

The activity at the Energy Center that is used for the dose analysis is fishing near the shoreline adjacent to the NMP site. Although access to this area has been restricted since September 11, 2001, the dose analysis will assume access granted for all of 2001. Dose pathways considered for this activity include direct radiation, inhalation and external ground (shoreline sediment or soil) doses. Other pathways, such as ingestion pathways, are not considered because they are either not applicable, insignificant, or are considered as part of the evaluation of the total dose to a member of the public located off-site. In addition, only releases from the NMP2 stack and vent were evaluated for the inhalation pathway.

The direct radiation pathway is evaluated in accordance with the methodology found in the Off-Site Dose Calculation Manual (ODCM). This pathway considers four components: direct radiation from the generating facilities, direct radiation from any possible overhead plume, direct radiation from ground deposition and direct radiation plume submersion. The direct radiation pathway is evaluated by the use of high sensitivity environmental Thermoluminescent Dosimeters (TLDs). Since any significant fishing activity near the Energy Center occurs between April through December, environmental TLD data for the approximate period of April 1 – December 31, 2001 were considered. Data from environmental TLDs from the approximate area where the fishing occurs were compared to control environmental TLD locations for the same time period. The average fishing area TLD dose rate was 7.02E-03 mRem per hour for the period. The average control TLD dose rate was 6.02E-03 mRem per hour for the period (approximate second, third and fourth calendar quarters of the year). The average increase in dose as a result of fishing in this area at a conservative frequency of eight hours per week for thirty-nine weeks is 3.10E-01 mRem from direct radiation for the period in question. The majority of the dose from this pathway is from the NMP1 facility because of its proximity to the fishing area. A small portion may be due to the NMP2 facility.

ATTACHMENT 10

RADIOACTIVE EFFLUENT RELEASE REPORT (2001)
NINE MILE POINT NUCLEAR STATION UNIT 2
DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES INSIDE THE SITE BOUNDARY

JANUARY – DECEMBER 2001

The inhalation dose pathway is evaluated by utilizing the inhalation equation in the ODCM, as adapted from Regulatory Guide 1.109. The equation basically gives a total inhalation dose in mRem for the time period in question (April – December). The total dose equals the sum, for all applicable radionuclides, of the NMP2 stack and vent release concentrations, times the average NMP2 stack and vent flow rate, times the applicable five-year average calculated X/Q, times the inhalation dose factors from Regulatory Guide 1.109, Table E-7, times the Regulatory Guide 1.109 annual air intake, times the fractional portion of the year in question. In order to be slightly conservative, no radiological decay is assumed.

The 2001 calculation utilized the following information:

NMP2 Stack:

- Unit 2 average stack flowrate = 4.45E+01 m^3/sec
- X/Q value = 9.60E-07 (annual NWN sector, historical average)
- Inhalation dose factor = Table E-7 of Regulatory Guide 1.109
- Annual air intake = 8000 m^3 per year (adult)
- Fractional portion of the year = 0.0356 (312 hours)
- H-3 = 1.90E+04 pCi/m^3
- Mn-54 = 3.52E-02 pCi/m^3
- Fe-55 = 8.24E-02 pCi/m^3
- Co-60 = 8.11E-02 pCi/m^3
- Zn-65 = 4.99E-03 pCi/m^3
- Sr-89 = 1.38E-02 pCi/m^3
- Sr-90 = 4.49E-03 pCi/m^3
- I-131 = 5.57E-01 pCi/m^3
- I-133 = 4.29E+00 pCi/m^3

ATTACHMENT 10

RADIOACTIVE EFFLUENT RELEASE REPORT (2001)
NINE MILE POINT NUCLEAR STATION UNIT 2
DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES INSIDE SITE BOUNDARY

JANUARY – DECEMBER 2001

NMP2 Vent:

- Unit 2 average vent flowrate = 9.82E+01 m^3 /sec
- X/Q value = 2.8E-06 (conservative ground level value)
- Inhalation dose factor = Table E-7 of Regulatory Guide 1.109
- Annual Air intake = 8000 m^3 per year (adult)
- Fractional portion of the year = 0.0356 (312 hours)
- H-3 = 1.59E+03 pCi/m^3
- Cr-51 = 2.20E-01 pCi/m^3
- Mn-54 = 2.86E-01 pCi/m^3
- Fe-55 = 1.34E+00 pCi/m^3
- Fe-59 = 1.07E-01 pCi/m^3
- Co-58 = 4.31E-02 pCi/m^3
- Co-60 = 3.83E-01 pCi/m^3
- Zn-65 = 4.92E-02 pCi/m^3
- Sr-89 = 1.17E-02 pCi/m^3
- Sr-90 = 1.45E-02 pCi/m^3

The inhalation dose to a member of the public from NMP2 as a result of activities inside the site boundary is 8.76E-05 mRem to the lung (maximum organ dose) and 5.71E-05 mRem to the whole body.

The dose from standing on the shoreline while fishing is based on the methodology in the ODCM, as adapted from Regulatory Guide 1.109. During 2001, it was noted that fishing was performed from the shoreline on many occasions although waders were also utilized. In order to be conservative, it is assumed that the maximum exposed individual fished from the shoreline at all times.

The ODCM equation gives the total dose to the whole body and skin from the sum of all plant-related radionuclides detected in shoreline sediment samples. The plant-related radionuclide concentration is adjusted for background sample results, as applicable. The equation, therefore, yields the whole body and skin dose by multiplying the radionuclide concentration adjusted for any background data (as applicable), times a usage factor, times the sediment or soil density in grams per square meter (to a depth of one centimeter), times the applicable shore width factor, times the regulatory guide dose factor, times the fractional portion of the year over which the dose is applicable. In order to be conservative and to simplify the equation, no radiological decay is assumed since the applicable radionuclides are usually long lived.

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RADIOACTIVE EFFLUENT RELEASE REPORT (2001)
NINE MILE POINT NUCLEAR STATION UNIT 2
DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES INSIDE THE SITE BOUNDARY

JANUARY – DECEMBER 2001

The calculation utilized the following information:

- Usage factor = 312 hours
- Density in grams per square meter = 40,000
- Shore width factor = 0.3
- Whole body and skin dose factor for each radionuclide = Regulatory Guide 1.109, Table E-6
- Fractional portion of the year = 1 (used average radionuclide concentration over total time period)
- Average Cs-137 concentration = 0.22 pCi/g

The total whole body and skin dose from standing on the shoreline to fish is 3.44E-03 mRem whole body and 4.02E-03 mRem skin dose for the period.

Doses to members of the public relative to activities inside the site boundary from aquatic pathways other than ground dose from shoreline sediment/soil are not applicable.

In summary, the total dose to a member of the public as a result of activities inside the site boundary from the direct radiation, inhalation and shoreline dose pathways is 3.10E-01 mRem to the whole body and 8.76E-05 mRem to the maximum exposed internal organ (lung). The dose to the skin of an adult is 4.02E-03 mRem. These doses are generally a result of the operation of NMP2. However, a portion of these doses for the direct radiation pathway may be attributable to the NMP1 facility.

ATTACHMENT 11

Doses to Members of the Public Due To Their Activities Outside the Site Boundary

ATTACHMENT 11

RADIOACTIVE EFFLUENT RELEASE REPORT (2001) NINE MILE POINT NUCLEAR STATION UNIT 2 DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES OUTSIDE THE SITE BOUNDARY

JANUARY – DECEMBER 2001

Radiation doses to the likely most exposed member of the public outside of the site boundary are evaluated relative to 40 CFR 190 requirements. The dose limits of 40 CFR 190 are 25 mRem (whole body or organ) per calendar year and 75 mRem (thyroid) per calendar year. The intent of 40 CFR 190 also requires that the effluents of Nine Mile Point Unit 2 (NMP2) , as well as other nearby uranium fuel cycle facilities, be considered. In this case, the effluents of Nine Mile Point Unit 1 (NMP1), NMP2 and the James A. FitzPatrick (JAF) facilities must be considered.

Doses to the likely most exposed member of the public as a result of effluents from the site can be evaluated by using calculated dose modeling based on the accepted methodologies of the facilities' Off-Site Dose Calculation Manuals (ODCMs) or may, in some cases, be calculated from the analysis results of actual environmental samples. Acceptable methods of calculating doses from environmental samples are also found in the facilities' ODCMs. These methods are based on Regulatory Guide 1.109 methodology.

Dose calculations from actual environmental samples are, at times, difficult to perform for some pathways. Some pathway doses should be estimated using calculational dose modeling. These pathways include noble gas air dose, inhalation dose, etc. Other pathway doses may be calculated directly from environmental sample concentrations using Regulatory Guide 1.109 methodology.

Since the effluents from the generating facilities are low, the resultant gaseous and liquid effluent doses are anticipated to be low. In view of this, doses can be based on calculated data. Doses are not based on actual environmental data for 2001 with the exception of doses from direct radiation, fish consumption and shoreline sediment. In addition, in order to be conservative and for the sake of simplicity, it is assumed in the dose calculations that the likely most exposed member of the public is positioned in the maximum receptor location for each pathway at the same time. This approach is utilized because the doses are very low and the computations are greatly simplified.

The following pathways are considered:

1. The inhalation dose is calculated at the critical residence because of the high occupancy factor. In order to be conservative, the maximum whole body and organ dose assumes no correction for residing inside a residence.

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RADIOACTIVE EFFLUENT RELEASE REPORT (2001)
NINE MILE POINT NUCLEAR STATION UNIT 2
DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES OUTSIDE THE SITE BOUNDARY

JANUARY – DECEMBER 2001

2. The milk ingestion dose is calculated utilizing the maximum milk cow location. As noted previously, in order to be conservative and for the sake of simplicity, the likely most exposed member of the public is assumed to be at all critical receptors at one time. In this case, the member of the public at the critical residence is assumed to consume milk from the critical milk location.

3. The maximum dose from the milk ingestion pathway as a result of consuming goat's milk is based on the same criteria established for item "2" above (ingestion of cow's milk).

4. The maximum dose associated from consuming meat is based on the critical meat animal. The likely most exposed member at the critical residence is assumed to consume meat from the critical meat animal location.

5. The maximum site dose associated with the consumption of vegetables is calculated from the critical vegetable garden location. As noted previously, the likely most exposed member of the public is assumed to be located at the critical residence and is assumed to consume vegetables from the critical garden location.

6. The dose, as a result of direct gamma radiation from the site, encompasses doses from direct "shine" from the generating facilities, direct radiation from any overhead gaseous plumes, plume submersion and from ground deposition. This total dose is measured by environmental TLDs. The critical location is based on the closest year-round residence from the generating facilities as well as the closest residence in the critical downwind sector in order to evaluate both direct radiation from the generating facilities and gaseous plumes as determined by the local meteorology. During 2001, the closest residence and the critical downwind residence are at the same location.

   The measured average dose for 2001 at the critical residence was 54.8 mRem. The average control dose was 52.8 mRem. The average dose at the critical residence is slightly greater than the average control location dose. The net increase in dose is due to the differences between doses from naturally occurring radionuclides in the soil and rock at the different locations and due to the standard deviation in TLD measurements. There is not a significant difference between the control and critical resident dose and is within expected normal statistical variation.

ATTACHMENT 11

RADIOACTIVE EFFLUENT RELEASE REPORT (2001)
NINE MILE POINT NUCLEAR STATION UNIT 2
DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES OUTSIDE THE SITE BOUNDARY

JANUARY – DECEMBER 2001

7. The dose, as a result of fish consumption, is considered as part of the aquatic pathway. The dose for 2001 is calculated from actual results of the analysis of environmental fish samples. For the sake of being conservative, the average plant-related radionuclide concentrations were utilized from fish samples taken near the site discharge points. No plant related radionuclides were detected in either the control or indicator samples. Therefore, no dose was calculated and was assumed to be zero for this pathway.

8. The shoreline sediment pathway is considered relative to recreational activities. The dose due to recreational activities from shoreline sediment is based on the methodology in the ODCM, as adapted from Regulatory Guide 1.109. The ODCM gives the total dose to the whole body and skin from the sum of plant-related radionuclides detected in actual shoreline sediment samples. The plant-related radionuclide concentration is adjusted for background sample results, as applicable. The total whole body and skin dose from shoreline recreational activities are 7.51E-04 mRem whole body and 8.77E-04 mRem skin dose for the period.

In summary, the maximum dose to the likely most exposed member of the public is 3.25E-01 mRem to the Thyroid (maximum organ dose) and 2.45E-01 mRem to the whole body. It should be noted that the maximum organ dose and maximum whole body doses are based on the sum of the maximum doses observed for all three facilities regardless of age group. This results in some conservatism. The maximum organ and whole body doses were a result of gaseous effluents. Doses as a result of liquid effluents are secondary. The total whole body and skin dose from shoreline recreational activities are 7.51E-04 mRem whole body and 8.77E-04 mRem skin dose for the period. The direct radiation dose to the critical residence from the generating facilities was insignificant or zero. The dose to an individual as a result of fish consumption was zero. These maximum total doses are a result of operations at the NMP1, NMP2 and the JAF facilities. The maximum organ dose and whole body dose are below the 40 CFR 190 criteria of 25 mRem per calendar year to the maximum exposed organ or the whole body, and below 75 mRem per calendar year to the thyroid.

ATTACHMENT 12

Update of Actual Data for the Third and Fourth Quarter 2000

ATTACHMENT 12

Unit 1 ___ Unit 2 X   Reporting Period July – December 2001

UPDATE OF RELEASE AND DOSE DATA FOR GASEOUS (ELEVATED AND GROUND LEVEL) AND LIQUID EFFLUENTS

Update of data using actual results from the off-site vendors for Strontium, Tritium, and Iron-55 for the third and fourth quarter 2000.

	GASEOUS 3rd QUARTER 2000	GASEOUS 4th QUARTER 2000	LIQUID 3rd QUARTER 2000	LIQUID 4th QUARTER 2000
Nuclide[1]	Activity (Ci)	Activity (Ci)	Activity (Ci)	Activity (Ci)
Sr-89	3.40E-04	**	**	**
Sr-90	**	**	**	**
H-3	3.74E+00	3.08E+00	8.85E+00	4.64E+00
Fe-55	2.50E-03	9.29E-05	6.48E-03	1.15E-03

Particulates		GASEOUS 3rd QUARTER	GASEOUS 4th QUARTER	LIQUID 3rd QUARTER	LIQUID 4th QUARTER
1. Particulates with half-lives >8 days	Ci	3.71E-03	6.61E-04	2.14E-02	5.20E-02
2. Average release rate (gaseous) or diluted concentration (liquid) for reporting period	μCi/sec (gaseous) μCi/ml (liquid)	4.38E-04	8.41E-05	1.58E-09	3.93E-09
Tritium					
1. Total release	Ci	3.74E+00	3.08E+00	8.85E+00	4.64E+00
2. Average release rate for period (gaseous) or diluted concentration (liquids) for the reporting period	μCi/sec (gaseous) μCi/ml (liquid)	4.41E-01	3.92E-01	6.51E-07	3.51E-07

Tritium, Iodines, and Particulates (with half-lives greater than 8 days)		GASEOUS 3rd QUARTER	GASEOUS 4th QUARTER	LIQUID 3rd QUARTER	LIQUID 4th QUARTER
1. Percent of Quarterly[2] Dose Limit (Gaseous – 7.5 mrem, Liquid – 1.5 mrem)	%	2.50E-02 (Quarterly	1.34E-02 (Quarterly	1.26E-01 (Quarterly)	1.98E-01 (Quarterly)
2. Percent of Annual[2] Dose Limit to Date (Gaseous – 15 mrem, Liquid – 3 mrem)	%	3.79E-02 (Annual)	4.47E-02 (Annual)	3.83E-01 (Annual)	4.78E-01 (Annual)
3. Percent of Organ -Dose Rate Limit (Gaseous – 1500 mrem/yr), Dose Limit (Liquid – 5 mrem Quarter, 10 mrem Annual)	%	4.66E-04 Quarterly	2.70E-04 Quarterly	1.16E-01 (Quarterly) 2.51E-01 (Annual)	3.17E-01 (Quarterly) 3.89E-01 (Annual)
4. Percent of 10CFR20[3] Concentration Limit (Liquid)	%			2.35E-02 (Quarterly)	1.67E-02 (Quarterly)
5. Percent of Dissolved or Entrained Noble Gas (Liquid – 2.00E-04 μCi/ml)	%			** (Quarterly)	1.41E-06 (Quarterly)

1 Concentrations less than the lower limit of detection, as required by the Off-Site Dose Calculation Manual (ODCM) following implementation of Improved Technical Specifications (ITS) are indicated with a double asterisk.

2 The dose is to the whole body for liquid effluents and to the maximally exposed organ for gaseous effluents.

3 The percent of the 10CFR20 concentration limit is based on the average concentration during the quarter.

ATTACHMENT 13

Summary of Changes to the Environmental Monitoring and Dose Calculation Locations

ATTACHMENT 13

Unit 1 ___ Unit 2 X Reporting Period January – December 2001

SUMMARY OF CHANGES TO THE ENVIRONMENTAL MONITORING AND CALCULATION LOCATIONS

Changes in Environmental Monitoring Locations

During the report period, the control location for milk, (Woodworth) map location 73, was replaced by a new control location for milk (Summerville), map location 77, as Mr. Woodworth has retired and sold his herd. Sample location selections are based on the annual land use census. Refer to Attachment 7, changes to the ODCM, for changes in distance and direction.

New Locations for Dose Calculations

During the report period, no changes in Dose Calculation Receptor Locations were required based on the results of the land use census.

ATTACHMENT 14

Off-Site Dose Calculation Manual (ODCM)

NINE MILE POINT NUCLEAR STATION

NINE MILE POINT UNIT 2

OFF-SITE DOSE CALCULATION MANUAL (ODCM)

APPROVALS	**SIGNATURES**	**DATE** **REVISION 22**
Prepared by:	G. R. Stinson G. R. Stinson Health Physicist	12/11/01
Reviewed by:	T. M. Kurtz T. M. Kurtz Health Physicist	12/11/01
Concurred by:	Ted Kulczycky T. G. Kulczycky Supervisor, Analysis Services	12/11/01
Concurred by:	Ted Kulczycky for WRY W. R. Yaeger Manager, Engineering Services	12/12/01

M. F. Peckham	MF Peckham **Plant General Manager Unit 2**	12/18/01
B. S. Montgomery	BS Montgomery **General Manager Nuclear Engineering**	12/13/01

SUMMARY OF REVISIONS

Revision 22 (Effective December 2001)

PAGE	***DATE***
I 3.3-13,14	August 2000
I 3.3-6	November 2000
I 4.0-1	November 2000
II 2-10,26,33-36,66,67,75,80	November 2000
ix, I 1.0-1, I 1.0-2, I B 3.3-2, I 4.1-1 & 1a, II 11, II 15, II 29, II 63, II 107, II 108	December 2001

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PART II – CALCULATIONAL METHODOLOGIES

LIST OF FIGURES

INTRODUCTION

The OFFSITE DOSE CALCULATION MANUAL (ODCM) is a supporting document of the Technical Specifications Section 5.5.1. The previous Limiting Conditions for Operation that were contained in the Radiological Effluent Technical Specifications are now transferred to the ODCM as Radiological Effluent Controls. The ODCM contains two parts: Radiological Effluent Controls, Part I; and Calculational Methodologies, Part II. Radiological Effluent Controls, Part 1, includes the following: (1) The Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification 5.5.1 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Technical Specifications 5.6.2 and 5.6.3. Calculational Methodologies, Part II, describes the methodology and parameters to be used in the calculation of liquid and gaseous effluent monitoring instrumentation alarm/trip setpoints and the calculation of offsite doses due to radioactive liquid and gaseous effluents. The ODCM also contains a list and graphical description of the specific sample locations for the radiological environmental monitoring program, and liquid and gaseous radwaste treatment system configurations.

The ODCM follows the methodology and models suggested by NUREG-0133 and Regulatory Guide 1.109, Revision 1. Simplifying assumptions have been applied in this manual where applicable to provide a more workable document for implementing the Radiological Effluent Control requirements; this simplified approach will result in a more conservative dose evaluation for determining compliance with regulatory requirements.

The ODCM will be maintained for use as a reference and training document of accepted methodologies and calculations. Changes to the calculation methods or parameters will be incorporated into the ODCM to assure that the ODCM represents the present methodology in all applicable areas. Any changes to the ODCM will be implemented in accordance with Section 5.5.1 of the Technical Specifications.

PART I - RADIOLOGICAL EFFLUENT CONTROLS

PART I - RADIOLOGICAL EFFLUENT CONTROLS

SECTION 1.0 DEFINITIONS

1.0 DEFINITIONS

--NOTE--
Technical Specifications defined terms and the following additional defined terms appear in capitalized type and are applicable throughout these specifications and bases.

TERM	DEFINITION
GASEOUS RADWASTE TREATMENT SYSTEM	A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting offgases from the main condenser evacuation system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.
MEMBER(S) OF THE PUBLIC	MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the Nine Mile Point Nuclear Station and James A. FitzPatrick Nuclear Power Plant. This category does not include employees of owners and operators of the Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant, their contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with Nine Mile Point Nuclear Station and James A. FitzPatrick Nuclear Power Plant.
MILK SAMPLING LOCATION	A MILK SAMPLING LOCATION is a location where 10 or more head of milk animals are available for collection of milk samples.
OFFSITE DOSE CALCULATION MANUAL	The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the current methodology and parameters used in the calculation of offsite doses that result from 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. The ODCM shall also contain: (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Program required by Specification 5.5.1 of Technical Specifications and, (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Technical Specifications 5.6.2 and 5.6.3.

(continued)

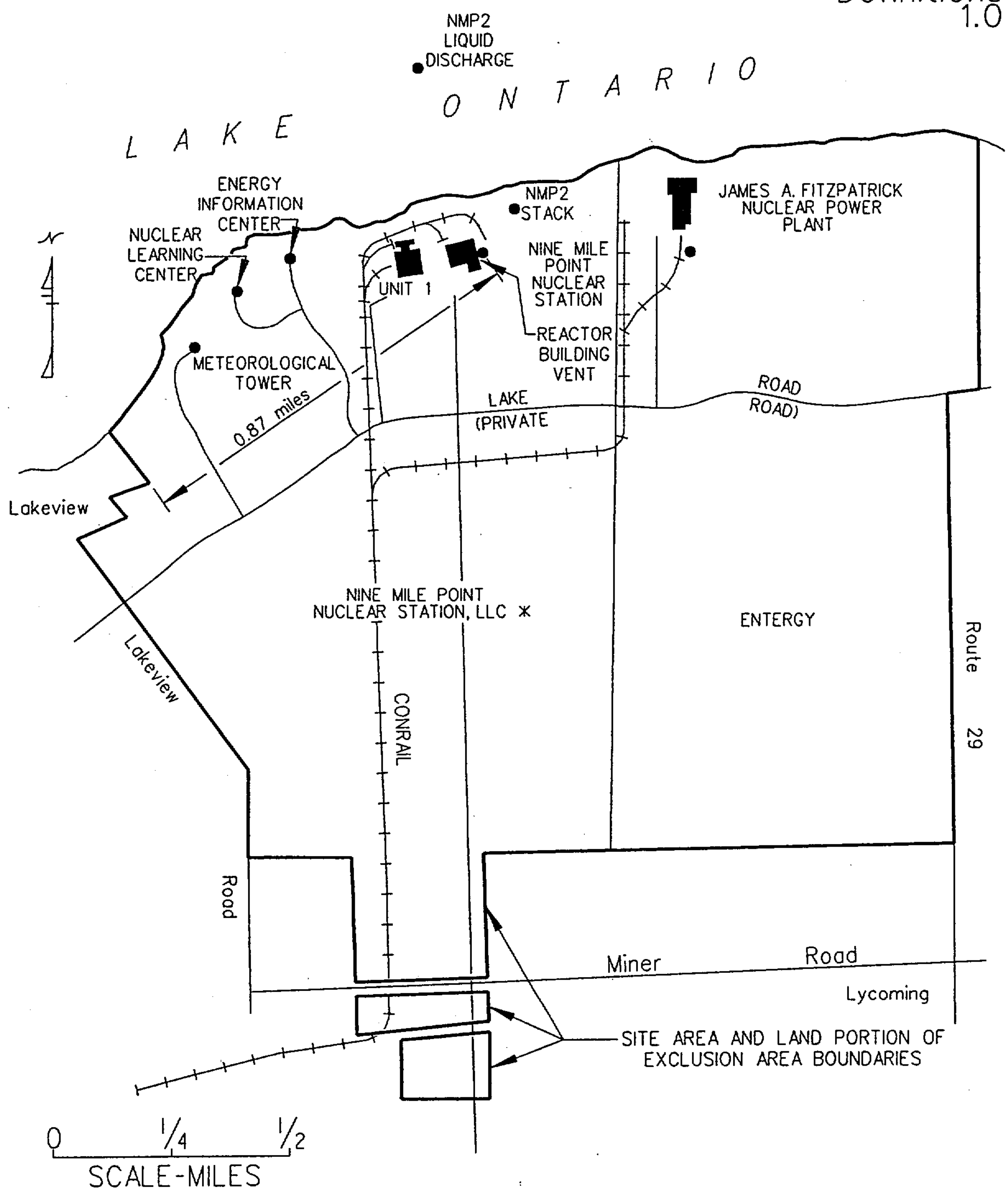
1.0 DEFINITIONS (continued)

TERM	DEFINITION
PURGE – PURGING	PURGE and PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.
REPORTABLE EVENT	A REPORTABLE EVENT shall be any of those conditions specified in 10 CFR 50.73.
SITE BOUNDARY	The SITE BOUNDARY shall be that line around the Nine Mile Point Nuclear Station beyond which the land is not owned, leased or otherwise controlled by the owners and operators of Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant. See Figure D 1.0-1.
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.
UNRESTRICTED AREA	An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY, access to which is not controlled by the owners and operators of Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant 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.
VENTILATION EXHAUST TREATMENT SYSTEM	A VENTILATION EXHAUST TREATMENT SYSTEM shall be 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 features (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

(continued)

1.0 DEFINITIONS (continued)

TERM	DEFINITION
VENTING	VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, 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.



* Niagara Mohawk Power Corporation retains ownership in certain transmission line and switchyard facilities within the exclusion area boundary. Access and usage are controlled by Nine Mile Point Nuclear Station, LLC by agreement.

Figure D 1.0-1 (Page 1 of 1)
Site Area and Land Portion of Exclusion Area Boundaries

PART I - RADIOLOGICAL EFFLUENT CONTROLS

SECTION 3.0 APPLICABILITY

3.0 APPLICABILITY

The Offsite Dose Calculation Manual (ODCM) Specifications are contained in Section 3.0 of Part I. They contain operational requirements, Surveillance Requirements, and reporting requirements. Additionally, the Required Actions and associated Completion Times for degraded Conditions are specified. The format is consistent with the Technical Specifications (Appendix A to the NMP2 Operating License).

The rules of usage for the ODCM Specification are the same as those for the Technical Specifications. These rules are found in Technical Specifications Sections 1.2, "Logical Connectors," 1.3, "Completion Times," and 1.4, "Frequency."

The ODCM Specifications are subject to Technical Specifications Section 3.0, "Limiting Condition for Operation (LCO) Applicability and Surveillance Requirement (SR) Applicability," with the following exceptions:

1. LCO 3.06, regarding support/supported system ACTIONS is not applicable to ODCM Specifications.
2. LCO 3.0.7, regarding allowances to change specified Technical Specifications is not applicable to ODCM Specifications.
3. Section 3.0 requirements are not required when so stated in notes within individual specifications.

D 3.1 RADIOACTIVE LIQUID EFFLUENTS

D 3.1.1 Liquid Effluents Concentration

DLCO 3.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (Figure D 1.0-1) shall be limited to:

a. Ten times the concentration specified in 10 CFR Part 20, Appendix B, Table 2, Column 2 for radionuclides other than dissolved or entrained noble gases; and

b. 2×10^{-4} µCi/ml total activity concentration for dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeds limits.	A.1 Initiate action to restore concentration to within limits.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.1.1.1	Perform radioactive liquid waste sampling and activity analysis.	In accordance with Table D 3.1.1-1
DSR 3.1.1.2	Verify the results of the DSR 3.1.1.1 analyses to assure that the concentrations at the point of release are maintained within the limits of DLCO 3.1.1.	In accordance with Table D 3.1.1-1

Table D 3.1.1-1 (Page 1 of 2)
Radioactive Liquid Waste Sampling and Analysis

LIQUID RELEASE TYPE	SAMPLE TYPE	SAMPLE FREQUENCY	ANALYSIS FREQUENCY	SAMPLE ANALYSIS	SAMPLE LOWER LIMIT OF DETECTION (LLD) (a)
1. Batch Waste Release Tanks (b) a. 2LWS-TK4A b. 2LWS-TK4B c. 2LWS-TK5A d. 2LWS-TK5B	Grab Sample	Each Batch (g)	Each Batch (g)	Principal Gamma Emitters (c)	5×10^{-7} μCi/ml
				I-131	1×10^{-6} μCi/ml
	Grab Sample	One batch/31 days (g)	31 days	Dissolved and Entrained Gases (gamma emitters)	1×10^{-5} μCi/ml
	Proportional Composite of grab samples (d)	Each batch (g)	31 days	H-3	1×10^{-5} μCi/ml
				Gross Alpha	1×10^{-7} μCi/ml
	Proportional Composite of grab samples (d)	Each batch (g)	92 days	Sr-89	5×10^{-8} μCi/ml
				Sr-90	5×10^{-8} μCi/ml
				Fe-55	1×10^{-6} μCi/ml
2. Continuous Releases a. Service Water Effluent A b. Service Water Effluent B c. Cooling Tower Blowdown	Grab Sample	31 days (e)	31 days (e)	Principal Gamma Emitters (c)	5×10^{-7} μCi/ml
	Grab Sample	31 days (e)	31 days (e)	I-131	1×10^{-6} μCi/ml
	Grab Sample	31 days (e)	31 days (e)	Dissolved and Entrained Gases (gamma emitters)	1×10^{-5} μCi/ml
	Grab Sample	31 days (e)	31 days (e)	H-3	1×10^{-5} μCi/ml
	Grab Sample	31 days (e)	31 days (e)	Gross Alpha	1×10^{-7} μCi/ml
	Grab Sample	92 days (e)	92 days (e)	Sr-89	5×10^{-8} μCi/ml
	Grab Sample	92 days (e)	92 days (e)	Sr-90	5×10^{-8} μCi/ml
	Grab Sample	92 days (e)	92 days (e)	Fe-55	1×10^{-6} μCi/ml
3. Continuous Release Auxiliary Boiler Pump Seal and Sample Cooling Discharge (Service Water)	Grab Sample	31 days (f)	31 days (f)	Principal Gamma Emitters (c)	5×10^{-7} μCi/ml
	Grab Sample	92 days (f)	92 days (f)	H-3	1×10^{-5} μCi/ml

Table D 3.1.1-1 (Page 2 of 2)
Radioactive Liquid Waste Sampling and Analysis

(a) The LLD is defined 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% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

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

where:

LLD	=	The before-the-fact lower limit of detection (μCi per unit mass or volume),
S_b	=	The standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),
E	=	The counting efficiency (counts per disintegration),
V	=	The sample size (units of mass or volume),
2.22×10^6	=	The number of disintegrations per minute per μCi,
Y	=	The fractional radiochemical yield, when applicable,
λ	=	The radioactive decay constant for the particular radionuclide (sec^{-1}), and
Δt	=	The elapsed time between the midpoint of sample collection and the time of counting (seconds).

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

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

(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 by the method described in Part II, Section 1.4 to assure representative sampling.

(c) The principal gamma emitters for which the LLD applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, and Ce-141. Ce-144 shall also be measured, but with an LLD of 5×10^{-6} μCi/ml. 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 Radioactive Effluent Release Report pursuant to Technical Specification 5.6.3 in the format outlined in RG 1.21, Appendix B, Revision 1, June 1974.

(d) 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.

(e) If the alarm setpoint of the effluent monitor 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 and an incident composite for H-3, gross alpha, Sr-89, Sr-90, and Fe-55.

(f) If the alarm setpoint of Service Water Effluent Monitor A and/or B 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 and an incident composite for H-3, gross alpha, Sr-89, Sr-90, and Fe-55.

(g) Complete prior to each release.

D 3.1 RADIOACTIVE LIQUID EFFLUENTS

D 3.1.2 Liquid Effluents Dose

DLCO 3.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials released in liquid effluents from each unit to UNRESTRICTED AREAS (Figure D 1.0-1) shall be limited to:

a. ≤ 1.5 mrem to the whole body and ≤ 5 mrem to any organ during any calendar quarter; and

b. ≤ 3 mrem to the whole body and ≤ 10 mrem to any organ during any calendar year.

APPLICABILITY: At all times.

ACTIONS

-----NOTES-----
1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Calculated dose to a MEMBER OF THE PUBLIC from the release of radioactive materials in liquid effluents to UNRESTRICTED AREAS exceeds limits.	A.1 Prepare and submit to the NRC, pursuant to D 4.1.1, a Special Report that (1) Identifies the cause(s) for exceeding the limit(s) and (2) 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 DLCO 3.1.2.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Calculated dose to a MEMBER OF THE PUBLIC from the release of radioactive materials in liquid effluents exceeds 2 times the limits.	B.1 Calculate the annual dose to a MEMBER OF THE PUBLIC which includes contributions from direct radiation from the units (including outside storage tanks, etc.).	Immediately
	<u>AND</u>	
	B.2 Verify that the limits of DLCO 3.4 have not been exceeded.	Immediately
C. Required Action B.2 and Associated Completion time not met.	C.1 Prepare and submit to the NRC, pursuant to D 4.1.1, a Special Report, as defined in 10 CFR 20.2203 (a)(4), of Required Action A.1 shall also include the following: (1) The corrective action(s) to be taken to prevent recurrence of exceeding the limits of DLCO 3.4 and the schedule for achieving conformance, (2) An analysis that estimates the dose to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s), and (3) Describes the levels of radiation and concentrations of radioactive material involved and the cause of the exposure levels or concentrations.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.1.2.1	Determine cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year.	31 days

D 3.1 RADIOACTIVE LIQUID EFFLUENTS

D 3.1.3 Liquid Radwaste Treatment System

DLCO 3.1.3 The liquid radwaste treatment system shall be OPERABLE.

APPLICABILITY: At all times.

ACTIONS

-----NOTES-----

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Radioactive liquid waste being discharged without treatment. AND Projected doses due to the liquid effluent, from the unit, to UNRESTRICTED AREAS would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31 day period. AND Any portion of the liquid radwaste treatment system not in operation.	A.1 Prepare and submit to the NRC, pursuant to D 4.1.1, a Special Report that includes: (1) An explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability, (2) Action(s) taken to restore the inoperable equipment to OPERABLE status, and (3) Summary description of action(s) taken to prevent a recurrence.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.1.3.1	---------------------------NOTE---------------------------- Only required to be met when liquid radwaste treatment systems are not being fully utilized. --- Project the doses due to liquid effluents from each unit to UNRESTRICTED AREAS.	31 days

D 3.2 RADIOACTIVE GASEOUS EFFLUENTS

D 3.2.1 Gaseous Effluents Dose Rate

DLCO 3.2.1 The dose rate from radioactive materials released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY (Figure D 1.0-1) shall be limited to:

a. For noble gases, ≤ 500 mrem/yr to the whole body and ≤ 3000 mrem/yr to the skin and

b. For I-131, I-133, H-3 and all radionuclides in particulate form with half-lives > 8 days, ≤ 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The dose rate(s) at or beyond the SITE BOUNDARY due to radioactive gaseous effluents exceeds limits.	A.1 Restore the release rate to within the limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.2.1.1	The dose rate from noble gases in gaseous effluents shall be determined to be within the limits of DLCO 3.2.1.a.	In accordance with Table D 3.2.1-1
DSR 3.2.1.2	The dose rate from I-131, I-133, H-3 and all radionuclides in particulate form with half-lives > 8 days in gaseous effluents shall be determined to be within the limits of DLCO 3.2.1.b.	In accordance with Table D 3.2.1-1

Table D 3.2.1-1 (Page 1 of 2)
Radioactive Gaseous Waste Sampling and Analysis

GASEOUS RELEASE TYPE	SAMPLE TYPE	SAMPLE FREQUENCY	ANALYSIS FREQUENCY	SAMPLE ANALYSIS	SAMPLE LOWER LIMIT OF DETECTION (LLD) (a)
1. Containment (b)	Grab Sample	Each Purge	(h)	Principal Gamma Emitters (c)	1 x 10^{-4} μCi/ml
			Each Purge	H-3 (oxide)	1 x 10^{-6} μCi/ml
			Each Purge	Principal Gamma Emitters (c)	1 x 10^{-4} μCi/ml
2. Main Stack, Radwaste/Reactor Building Vent	Grab Sample	31 days (d)	31 days (d)	Principal Gamma Emitters (c)	1 x 10^{-4}μCi/ml
	Grab Sample	31 days (e)	31 days (e)	H-3 (oxide)	1x 10^{-6}μCi/ml
	Charcoal Sample	Continuous (f)	7 days (g)	I-131	1x 10^{-12} μCi/ml
	Particulate Sample	Continuous (f)	7 days (g)	Principal Gamma Emitters (c)	1 x 10^{-11} μCi/ml
				Gross Alpha	1x 10^{-11} μCi/ml
	Composite Particulate Sample	Continuous (f)	92 days	Sr-89	1 x 10^{-11} μCi/ml
				Sr-90	1 x 10^{-11} μCi/ml

See the notes on the next page.

Table D 3.2.1-1 (Page 2 of 2)
Radioactive Gaseous Waste Sampling and Analysis

(a) The LLD is defined 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% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{(4.66)(S_b)}{(E)(V)(2.22\times10^6)(Y)e^{-\lambda\Delta t}}$$

where:

LLD = The before-the-fact lower limit of detection (µCi per unit mass or volume),

S_b = The standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = The counting efficiency (counts per disintegration),

V = The sample size (units of mass or volume),

2.22×10^6 = The number of disintegrations per minute per µCi,

Y = The fractional radiochemical yield, when applicable,

λ = The radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = The elapsed time between the midpoint of sample collection and the time of counting (seconds).

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

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

(b) Sample and analysis before PURGE is used to determine permissible PURGE rates. Sample and analysis during actual PURGE is used for offsite dose calculations.

(c) The principal gamma emitters for which the LLD applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141 and Ce-144 in iodine and particulate releases. 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 Radioactive Effluent Release Report pursuant to Technical Specification 5.6.3 in the format outlined in RG 1.21, Appendix B, Revision 1, June 1974.

(d) If the main stack or reactor/radwaste building isotopic monitor is not OPERABLE, sampling and analysis shall also be performed following shutdown, startup, or when there is an alarm on the offgas pretreatment monitor.

(e) H-3 grab samples shall be taken once every 7 days from the reactor/radwaste ventilation system when fuel is offloaded until stable H-3 release levels can be demonstrated.

(f) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with DLCO 3.2.1.b and DLCO 3.2.3.

(g) When the release rate of the main stack or reactor/radwaste building vent exceeds its alarm setpoint, the iodine and particulate device shall be removed and analyzed to determine the changes in iodine and particulate release rates. The analysis shall be done once per 24 hours until the release no longer exceeds the alarm setpoint. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

(h) Complete prior to each release.

D 3.2 RADIOACTIVE GASEOUS EFFLUENTS

D 3.2.2 Gaseous Effluents Noble Gas Dose

DLCO 3.2.2 The air dose from noble gases released in gaseous effluents from each unit to areas at or beyond the SITE BOUNDARY (Figure D 1.0-1) shall be limited to:

a. During any calendar quarter: ≤ 5 mrad for gamma radiation and ≤ 10 mrad for beta radiation and

b. During any calendar year: ≤ 10 mrad for gamma radiation and ≤ 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTIONS

---NOTES---

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. The air dose at or beyond the SITE BOUNDARY due to noble gases released in gaseous effluents exceeds limits.	A.1	Prepare and submit to the NRC, pursuant to D 4.1.1, a Special Report that (1) Identifies the cause(s) for exceeding the limit(s) and (2) 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 DLCO 3.2.2.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Calculated dose to a MEMBER OF THE PUBLIC from the release of radioactive materials in gaseous effluents due to noble gases exceeds 2 times the limits.	B.1	Calculate the annual dose to a MEMBER OF THE PUBLIC which includes contributions from direct radiation from the units (including outside storage tanks, etc.).	Immediately
	<u>AND</u>		
	B.2	Verify that the limits of DLCO 3.4 have not been exceeded.	Immediately
C. Required Action B.2 and Associated Completion time not met.	C.1	Special Report, as defined in 10 CFR 20.2203 (a)(4), of Required Action A.1 shall also include the following: (1) The corrective action(s) to be taken to prevent recurrence of exceeding the limits of DLCO 3.4 and the schedule for achieving conformance, (2) An analysis that estimates the dose to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s), and (3) Describes the levels of radiation and concentrations of radioactive material involved and the cause of the exposure levels or concentrations.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.2.2.1	Determine cumulative dose contributions for the current calendar quarter and current calendar year.	31 days

D 3.2 RADIOACTIVE GASEOUS EFFLUENTS

D 3.2.3 Gaseous Effluents Dose – I-131, I-133, H-3 and Radioactive Material in Particulate Form

DLCO 3.2.3 The dose to a MEMBER OF THE PUBLIC from I-131, I-133, H-3, and all radioactive material in particulate form with half-lives > 8 days in gaseous effluents released, from each unit, to areas at or beyond the SITE BOUNDARY (Figure D 1.0-1) shall be limited to:

a. During any calendar quarter: ≤ 7.5 mrem to any organ and

b. During any calendar year: ≤ 15 mrem to any organ.

APPLICABILITY: At all times.

ACTIONS

--NOTES--

1. LCO 3.0.3 is not applicable.

2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The dose from I-131, I-133, H-3 and radioactive material in particulate form with half-lives > 8 days released in gaseous effluents at or beyond the SITE BOUNDARY exceeds limits.	A.1 Prepare and submit to the NRC, pursuant to D 4.1.1, a Special Report that (1) Identifies the cause(s) for exceeding the limit(s) and (2) 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 DLCO 3.2.3.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Calculated dose to a MEMBER OF THE PUBLIC from the release of radioactive materials in gaseous effluents exceeds 2 times the limits.	B.1 Calculate the annual dose to a MEMBER OF THE PUBLIC which includes contributions from direct radiation from the units (including outside storage tanks, etc.).	Immediately
	<u>AND</u>	
	B.2 Verify that the limits of DLCO 3.4 have not been exceeded.	Immediately
C. Required Action B.2 and Associated Completion time not met.	C.1 Special Report, as defined in 10 CFR 20.2203 (a)(4), of Required Action A.1 shall also include the following: (1)The corrective action(s) to be taken to prevent recurrence of exceeding the limits of DLCO 3.4 and the schedule for achieving conformance, (2)An analysis that estimates the dose to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s), and (3)Describes the levels of radiation and concentrations of radioactive material involved and the cause of the exposure levels or concentrations.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.2.3.1	Determine cumulative dose contributions for the current calendar quarter and current calendar year for I-131, I-133, H-3 and radioactive material in particulate form with half-lives > 8 days.	31 days

D 3.2 RADIOACTIVE GASEOUS EFFLUENTS

D 3.2.4 Gaseous Radwaste Treatment System

DLCO 3.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM shall be in operation.

APPLICABILITY: Whenever the main condenser air ejector system is in operation.

ACTIONS

--NOTES --

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The gaseous radwaste from the main condenser air ejector system is being discharged without treatment.	A.1 Restore treatment of gaseous radwaste effluent.	7 days
B. Required Action and associated Completion Time not met.	B.1 Prepare and submit to the NRC, pursuant to D 4.1.1, a Special Report that includes the following: (1) Identification of any inoperable equipment or subsystems and the reason for the inoperability, (2) Action(s) taken to restore the inoperable equipment to OPERABLE status, and (3) Summary description of action(s) taken to prevent a recurrence.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.2.4.1	Check the readings of the relevant instruments to ensure that the GASEOUS RADWASTE TREATMENT SYSTEM is functioning.	12 hours

D 3.2 RADIOACTIVE GASEOUS EFFLUENTS

D 3.2.5 Ventilation Exhaust Treatment System

DLCO 3.2.5 The VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE.

APPLICABILITY: At all times.

ACTIONS

--NOTES--

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The radioactive gaseous waste is being discharged without treatment. <u>AND</u> Projected doses in 31 days from iodine and particulate releases, from each unit, to areas at or beyond the SITE BOUNDARY (see Figure D 1.0-1) would exceed 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.	A.1 Prepare and submit to the NRC, pursuant to D 4.1.1, a Special Report that includes the following: (1) Identification of any inoperable equipment or subsystems and the reason for the inoperability, (2) Action(s) taken to restore the inoperable equipment to OPERABLE status, and (3) Summary description of action(s) taken to prevent a recurrence.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.2.5.1	----------------------------NOTE---------------------------- Only required to be met when the VENTILATION EXHAUST TREATMENT SYSTEM is not being fully utilized. --- Project the doses from iodine and particulate releases from each unit to areas at or beyond the SITE BOUNDARY.	31 days

D 3.2 RADIOACTIVE GASEOUS EFFLUENTS

D 3.2.6 Venting or Purging

DLCO 3.2.6 VENTING or PURGING of the drywell and/or suppression chamber shall be through the standby gas treatment system.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTES-----

1. LCO 3.0.3 is not applicable.

2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. VENTING or PURGING of the drywell and/or suppression chamber not through the standby gas treatment system.	A.1 Suspend all VENTING and PURGING of the drywell and/or suppression chamber.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.2.6.1	The drywell and/or suppression chamber shall be determined to be aligned for VENTING or PURGING through the standby gas treatment system.	Within 4 hours before start of VENTING or PURGING <u>AND</u> 12 hours thereafter during VENTING or PURGING

D 3.3 INSTRUMENTATION

D 3.3.1 Radioactive Liquid Effluent Monitoring Instrumentation

DLCO 3.3.1 The radioactive liquid effluent monitoring instrumentation channels shown in Table D 3.3.1-1 shall be OPERABLE with:

a. The minimum OPERABLE channel(s) in service.

b. The alarm/trip setpoints set to ensure that the limits of DLCO 3.1.1 are not exceeded.

APPLICABILITY: According to Table D 3.3.1-1.

ACTIONS

-----NOTES-----

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.
3. Separate condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required.	A.1 Suspend the release of radioactive liquid effluents monitored by the affected channel.	Immediately
	<u>OR</u>	
	A.2 Declare the channel inoperable.	Immediately
	<u>OR</u>	
	A.3 Change the setpoint so it is acceptably conservative.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more required channels inoperable.	B.1 Enter the Condition referenced in Table D 3.3.1-1 for the channel.	Immediately
	<u>AND</u>	
	B.2 Restore inoperable channel(s) to OPERABLE status.	30 days
C. As required by Required Action B.1 and referenced in Table D 3.3.1-1.	C.1 Analyze at least 2 independent samples in accordance with Table D 3.1.1-1.	Prior to initiating a release
	<u>AND</u>	
	C.2 -----------NOTE--------------- Verification Action will be performed by at least 2 separate technically qualified members of the facility staff. ----------------------------------	
	Independently verify the release rate calculations and discharge line valving.	Prior to initiating a release
D. As required by Required Action B.1 and referenced in Table D 3.3.1-1.	D.1 Collect and analyze grab samples for radioactivity at a limit of detection of at least 5×10^{-7} µCi/ml.	12 hours <u>AND</u> Once per 12 hours thereafter

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action B.1 and referenced in Table D 3.3.1-1.	E.1 ---------------NOTE------------ Pump performance curves generated in place may be used to estimate flow. ---------------------------------- Estimate the flow rate during actual releases.	 4 hours AND Once per 4 hours thereafter
F. As required by Required Action B.1 and referenced in Table D 3.3.1-1.	F.1 Estimate tank liquid level.	Immediately AND During liquid additions to the tank
G. Required Action B.2 and associated Completion Time not met.	G.1 Explain in the next Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.	In accordance with Radioactive Effluent Release Report
H. Required Action and associated Completion Time for Condition C, D, or E not met.	H.1 Suspend liquid effluent releases monitored by the inoperable channel(s).	Immediately
I. Required Action and associated Completion Time for Condition F not met.	I.1 Suspend liquid additions to the tank monitored by the inoperable channel(s).	Immediately

SURVEILLANCE REQUIREMENTS

---NOTE---
Refer to Table D 3.3.1-1 to determine which DSRs apply for each function.

	SURVEILLANCE	FREQUENCY
DSR 3.3.1.1	Perform CHANNEL CHECK.	24 hours
DSR 3.3.1.2	Perform CHANNEL CHECK by verifying indication of flow during periods of release.	24 hours on any day on which continuous, periodic, or batch releases are made
DSR 3.3.1.3	Perform SOURCE CHECK.	Prior to release
DSR 3.3.1.4	Perform SOURCE CHECK.	31 days
DSR 3.3.1.5	Perform CHANNEL FUNCTIONAL TEST. The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measured levels above the alarm/trip setpoint; and control room alarm annunciation occurs for instrument indication levels measured above the alarm setpoint, circuit failure, instrument indicating a downscale failure, or instrument controls not set in operate mode.	31 days
DSR 3.3.1.6	Perform CHANNEL FUNCTIONAL TEST.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
DSR 3.3.1.7	Perform CHANNEL FUNCTIONAL TEST. The CHANNEL FUNCTIONAL TEST shall also demonstrate control room alarm annunciation occurs for instrument indication levels measured above the alarm setpoint, circuit failure, instrument indicating a downscale failure, or instrument controls not set in operate mode.	184 days
DSR 3.3.1.8	Perform CHANNEL CALIBRATION. The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST), standards that are traceable to NIST standards, or using actual samples of liquid effluents that have been analyzed on a system that has been calibrated with NIST traceable sources. These standards shall permit calibrating the system over its intended range of energy and measurement. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration may be used.	18 months
DSR 3.3.1.9	Perform CHANNEL CALIBRATION.	18 months

Table D 3.3.1-1 (page 1 of 1)
Radioactive Liquid Effluent Monitoring Instrumentation

INSTRUMENT	APPLICABILITY OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER INSTRUMENT	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
Liquid Radwaste Effluent Line	(a)	1	C	DSR 3.3.1.1 DSR 3.3.1.3 DSR 3.3.1.5 DSR 3.3.1.8
2. Radioactivity Monitors Providing Alarm but not Providing Automatic Termination of Release				
a. Service Water Effluent Line A	(a)	1	D	DSR 3.3.1.1 DSR 3.3.1.4 DSR 3.3.1.7 DSR 3.3.1.8
b. Service Water Effluent Line B	(a)	1	D	DSR 3.3.1.1 DSR 3.3.1.4 DSR 3.3.1.7 DSR 3.3.1.8
c. Cooling Tower Blowdown Line	(a)	1	D	DSR 3.3.1.1 DSR 3.3.1.4 DSR 3.3.1.7 DSR 3.3.1.8
3. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line	(a)	1	E	DSR 3.3.1.2 DSR 3.3.1.6 DSR 3.3.1.9
b. Service Water Effluent Line A	(a)	1	E	DSR 3.3.1.2 DSR 3.3.1.6 DSR 3.3.1.9
c. Service Water Effluent Line B	(a)	1	E	DSR 3.3.1.2 DSR 3.3.1.6 DSR 3.3.1.9
d. Cooling Tower Blowdown Line	(a)	1	E	DSR 3.3.1.2 DSR 3.3.1.6 DSR 3.3.1.9
4. Tank Level Indicating Devices (c)	(b)	1	F	DSR 3.3.1.1 DSR 3.3.1.6 DSR 3.3.1.9

(a) During releases via this pathway.
(b) During liquid addition to the associated tank.
(c) Tanks included in this DLCO are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system, such as temporary tanks.

D 3.3 INSTRUMENTATION

D 3.3.2 Radioactive Gaseous Effluent Monitoring Instrumentation

DLCO 3.3.2 The radioactive gaseous effluent monitoring instrumentation channels shown in Table D 3.3.2-1 shall be OPERABLE with:

a. The minimum OPERABLE channel(s) in service.

b. The alarm/trip setpoints set to ensure that the limits of DLCO 3.2.1 are not exceeded.

APPLICABILITY: According to Table D 3.3.2-1.

ACTIONS

-----NOTES-----

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.
3. Separate condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required.	A.1 Suspend the release of radioactive gaseous effluents monitored by the affected channel.	Immediately
	<u>OR</u>	
	A.2 Declare the channel inoperable.	Immediately
	<u>OR</u>	
	A.3 Change the setpoint so it is acceptably conservative.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more channels inoperable.	B.1 Enter the Condition referenced in Table D 3.3.2-1 for the channel.	Immediately
	AND	
	B.2 Restore inoperable channel(s) to OPERABLE status.	30 days
C. As required by Required Action B.1 and referenced in Table D 3.3.2-1.	C.1 Place the inoperable channel in the tripped condition.	12 hours
	OR	
	C.2.1 Take grab samples.	12 hours AND Once per 12 hours thereafter
	AND	
	C.2.2 Analyze samples for gross activity.	24 hours from time of sampling completion

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. As required by Required Action B.1 and referenced in Table D 3.3.2-1.	D.1 Estimate the flow rate for the inoperable channel(s).	4 hours <u>AND</u> Once per 4 hours thereafter
E. As required by Required Action B.1 and referenced in Table D 3.3.2-1.	E.1 Continuously collect samples using auxiliary sampling equipment as required in Table D 3.2.1-1.	8 hours
F. As required by Required Action B.1 and referenced in Table D 3.3.2-1.	F.1.1 Take grab samples.	12 hours <u>AND</u> Once per 12 hours thereafter
	<u>AND</u>	
	F.1.2 Analyze samples for gross activity with a radioactivity limit of detection of at least 1 x 10^{-4} µCi/ml.	24 hours from time of sampling completion
	<u>AND</u>	
	F.2.1 Restore the inoperable channel(s) to OPERABLE status. <u>OR</u>	72 hours
	F.2.2 In lieu of another required report, prepare and submit to the NRC, pursuant to D 4.1.1, a special report that: (1) Identifies the cause(s) of the inoperability. (2) Outlines the action taken and the schedule for restoring the system to OPERABLE status	14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Required Action B.2 and associated Completion Time not met.	G.1 Explain in the next Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.	In accordance with Radioactive Effluent Release Report frequency
H Required Action and associated Completion Time for Condition C, D, E or F not met.	H.1 Suspend gaseous effluent releases monitored by the inoperable channel(s).	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.3.2.1	Perform CHANNEL CHECK.	24 hours
DSR 3.3.2.2	Perform CHANNEL CHECK.	7 days
DSR 3.3.2.3	Perform SOURCE CHECK.	31 days
DSR 3.3.2.4	Perform CHANNEL FUNCTIONAL TEST. The CHANNEL FUNCTIONAL TEST shall also demonstrate the automatic isolation capability of this pathway and that control room alarm annunciation occurs if the instrument indicates measured levels above the alarm/trip setpoint (each channel will be tested independently so as to not initiate isolation during operation); and control room alarm annunciation occurs for instrument indication levels measured above the alarm setpoint, circuit failure, instrument indicating a downscale failure, and instrument controls not set in operate mode.	31 days
DSR 3.3.2.5	Perform CHANNEL FUNCTIONAL TEST.	92 days
DSR 3.3.2.6	Perform CHANNEL FUNCTIONAL TEST. The CHANNEL FUNCTIONAL TEST shall also demonstrate control room alarm annunciation occurs for instrument indication levels measured above the alarm setpoint, circuit failure, instrument indicating a downscale failure, and instrument controls not set in operate mode.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
DSR 3.3.2.7	Perform CHANNEL CALIBRATION. The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST, or using actual samples of gaseous effluents that have been analyzed on a system that has been calibrated with NIST traceable sources. These standards shall permit calibrating the system over its intended range of energy and measurement. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration may be used. The CHANNEL CALIBRATION shall also demonstrate that automatic isolation of this pathway occurs when the instrument channels indicate measured levels above the Trip Setpoint.	18 months
DSR 3.3.2.8	Perform CHANNEL CALIBRATION.	18 months
DSR 3.3.2.9	Perform CHANNEL CALIBRATION. The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST, or using actual samples of gaseous effluents that have been analyzed on a system that has been calibrated with NIST traceable sources. These standards shall permit calibrating the system over its intended range of energy and measurement. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration may be used.	18 months

Table D 3.3.2-1 (page 1 of 2)
Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT	APPLICABILITY OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER INSTRUMENT	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS
1. Offgas System				
a. Noble Gas Activity Monitor – Providing Alarm and Automatic Termination of Release	(a)	2	C	DSR 3.3.2.1 DSR 3.3.2.4 DSR 3.3.2.7
b. System Flow-Rate Measuring Device	(a)	1	D	DSR 3.3.2.1 DSR 3.3.2.5 DSR 3.3.2.8
c. Sample Flow-Rate Measuring Device	(a)	2	D	DSR 3.3.2.1 DSR 3.3.2.5 DSR 3.3.2.8
2. Radwaste/Reactor Building Vent Effluent System				
a. Noble Gas Activity Monitor (c)	(b)	1	F	DSR 3.3.2.1 DSR 3.3.2.3 DSR 3.3.2.6 DSR 3.3.2.9
b. Iodine Sampler	(b)	1	E	DSR 3.3.2.2
c. Particulate Sampler	(b)	1	E	DSR 3.3.2.2
d. Flow-Rate Monitor	(b)	1	D	DSR 3.3.2.1 DSR 3.3.2.5 DSR 3.3.2.8
e. Sample Flow-Rate Monitor	(b)	1	D	DSR 3.3.2.1 DSR 3.3.2.5 DSR 3.3.2.8

(continued)

(a) During offgas system operation.

(b) At all times.

(c) Includes high range noble gas monitoring capability.

Table D 3.3.2-1 (page 2 of 2)
Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT	APPLICABILITY OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER INSTRUMENT	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS
3. Main Stack Effluent				
a. Noble Gas Activity Monitor (c)	(b)	1	F	DSR 3.3.2.1 DSR 3.3.2.3 DSR 3.3.2.6 DSR 3.3.2.9
b. Iodine Sampler	(b)	1	E	DSR 3.3.2.2
c. Particulate Sampler	(b)	1	E	DSR 3.3.2.2
d. Flow-Rate Monitor	(b)	1	D	DSR 3.3.2.1 DSR 3.3.2.5 DSR 3.3.2.8
e. Sample Flow-Rate Monitor	(b)	1	D	DSR 3.3.2.1 DSR 3.3.2.5 DSR 3.3.2.8

(b) At all times.

(c) Includes high range noble gas monitoring capability.

D 3.4 RADIOACTIVE EFFLUENTS TOTAL DOSE

D 3.4 Radioactive Effluents Total Dose

DLCO 3.4 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 shall be limited to ≤ 25 mrem to the whole body or any organ, except the thyroid, which shall be limited to ≤ 75 mrem.

APPLICABILITY: At all times.

ACTIONS

---NOTES---

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Estimated dose or dose commitment due to direct radiation and the release of radioactive materials in liquid or gaseous effluents exceeds the limits.	A.1 Verify the condition resulting in doses exceeding these limits has been corrected.	Immediately
B. Required Action and associated Completion Time not met.	B.1 --------------NOTE------------- This is the Special Report required by D 3.1.2, D 3.2.2, or D 3.2.3 supplemented with the following. ---------------------------------- Submit a Special Report, pursuant to D 4.1.1, including a request for a variance in accordance with the provisions of 40 CFR 190. This submission is considered a timely request, and a variance is granted until staff action on the request is complete.	 30 days

D 3.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

D 3.5.1 Monitoring Program

DLCO 3.5.1 The Radiological Environmental Monitoring Program shall be conducted as specified in Table D 3.5.1-1.

APPLICABILITY: At all times.

ACTIONS

------------------------------------NOTES------------------------------------

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Radiological Environmental Monitoring Program not conducted as specified in Table D 3.5.1-1.	A.1 Prepare and submit to the NRC 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.	In accordance with the Annual Radiological Environmental Operating Report frequency
B. Level of radioactivity in an environmental sampling medium at a specified location exceeds the reporting levels of Table D 3.5.1-2 when averaged over any calendar quarter. <u>OR</u>	B.1 ------------NOTES------------ 1. Only applicable if the radioactivity/radionuclides are the result of plant effluents. 2. For radionuclides other than those in Table D 3.5.1-2, this report shall indicate the methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC. -----------------------------------	

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
More than one of the radionuclides in Table D 3.5.1-2 are detected in the environmental sampling medium and $\frac{\text{Concentration 1}}{\text{reporting level 1}} + \frac{\text{concentration 2}}{\text{reporting level 2}} + \ldots \geq 1.0.$ OR Radionuclides other than those in Table D 3.5.1-2 are detected in an environmental sampling medium at a specified location which are the result of plant effluents and the potential annual dose to a MEMBER OF THE PUBLIC from all radionuclides is ≥ the calendar year limits of D 3.1.2, D 3.2.2 or D 3.2.3.	Prepare and submit to the NRC, pursuant to D 4.1.1, a Special Report that (1) Identifies the cause(s) for exceeding the limit(s) and (2) Defines the corrective actions 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 D 3.1.2, D 3.2.2, or D 3.2.3.	30 days
	OR	
	B.2 ------------NOTES------------- 1. Only applicable if the radioactivity/radionuclides are not the result of plant effluents. 2. For radionuclides other than those in Table D 3.5.1-2, this report shall indicate the methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC. ------------------------------------ Report and describe the condition in the Annual Radiological Environmental Operating Report.	In accordance with the Annual Radiological Environmental Operating Report frequency

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Milk or fresh leafy vegetation samples unavailable from one or more of the sample locations required by Table D 3.5.1-1.	C.1 Identify specific locations for obtaining replacement samples and add them to the Radiological Environmental Monitoring Program.	30 days
	AND	
	C.2 Delete the specific locations from which samples were unavailable from the Radiological Environmental Monitoring Program.	30 days
	AND	
	C.3 Pursuant to Technical Specification 5.6.3, submit in the next Radioactive Effluent Release Report documentation for a change in the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.	In accordance with the Radioactive Effluent Release Report
D. Environmental samples required in Table D 3.5.1-1 are unobtainable due to sampling equipment malfunctions.	D.1 Ensure all efforts are made to complete corrective action(s).	Prior to the end of the next sampling period
	AND	
	D.2 Report all deviations from the sampling schedule in the Annual Radiological Environmental Operating Report.	In accordance with the Annual Radiological Environmental Operating Report

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Samples required by Table D 3.5.1-1 not obtained in the media of choice, at the most desired location, or at the most desired time.	E.1	Choose suitable alternative media and locations for the pathway in question.	30 days
	AND		
	E.2	Make appropriate substitutions in the Radiological Environmental Monitoring Program.	30 days
	AND		
	E.3	Submit in the next Radioactive Effluent Release Report documentation for a change in the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples for that pathway and justifying the selection of the new location(s) for obtaining samples.	In accordance with the Radioactive Effluent Release Report

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.5.1.1	Collect and analyze radiological environmental monitoring samples pursuant to the requirements of Table D 3.5.1-1 and the detection capabilities required by Table D 3.5.1-3.	In accordance with Table D 3.5.1-1

Table D 3.5.1-1 (page 1 of 4)
Radiological Environmental Monitoring Program

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLES STATIONS	SAMPLE LOCATIONS (a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation	32 routine monitoring stations (b)	(1) An inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY (2) An outer ring of stations, one in each land base meteorological sector in the 4 to 5 mile (c) range from the site (3) The balance of the stations should be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations (d)	Once per 3 months	Gamma dose: once per 3 months
2. Airborne Radioiodine and Particulates	5 locations	(1) 3 samples from offsite locations close to the site boundary (within 1 mile) in different sectors (e) (2) 1 sample from the vicinity of an established year-round community (e) (3) 1 sample from a control location, at least 10 miles distant and in a least prevalent wind direction (d)	Continuous sampler operation with sample collection weekly or more frequently if required by dust loading	Radioiodine canister: Analyze weekly for I-131 Particulate sampler: (1) Analyze for gross beta radioactivity $\geq$ 24 hours following filter change (f). (2) Perform gamma isotopic analysis on each sample (g) in which gross beta activity is > 10 times the previous yearly mean of control samples. (3) Gamma isotopic analysis of composite sample (g) (by location) once per 3 months
3. Waterborne				
a. Surface	1 sample 1 sample	Upstream (d) (h) Site's downstream cooling water intake (h)	Composite sample over a one month period (i)	(1) Gamma isotopic analysis of each sample (g) once per month (2) H-3 analysis of each composite sample and once per 3 months
b. Ground	As required	From one or two sources if likely to be affected (j)	Grab sample once per 3 months	(1) Gamma isotopic analysis of each sample (g) once per 3 months (2) H-3 analysis of each sample once per 3 months

(continued)

Table D 3.5.1-1 (page 2 of 4)
Radiological Environmental Monitoring Program

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLES	SAMPLE LOCATIONS (a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. Waterborne (continued)				
c. Drinking	1 sample of each	One to three of the nearest water supplies that could be affected by its discharge (k)	When I-131 analysis is performed, a composite sample over a two week period (i); otherwise, a composite sample monthly	(1) I-131 analysis on each composite sample when the dose calculated for the consumption of the water is greater than 1 mrem/yr (l) (2) Gross beta and gamma isotopic analyses of each composite sample (g) monthly (3) H-3 analysis of each composite sample once per 3 months
d. Sediment from Shoreline	1 sample	From a downstream area with existing or potential recreational value	Twice per year	Gamma isotopic analysis of each sample (g)
4. Ingestion				
a. Milk	(1) 3 samples from MILK SAMPLING LOCATIONS	In 3 locations within 3.5 miles (e)	Twice per month, April through December (m)	(1) Gamma isotopic (g) and I-131 analysis of each sample twice per month April through December (2) Gamma isotopic (g) and I-131 analysis of each sample once per month January through March if required
	(2) If there are none, then 1 sample from MILK SAMPLING LOCATIONS	In each of 3 areas 3.5-5.0 miles distant (e)		
	(3) 1 sample from a MILK SAMPLING LOCATION	At a control location 9-20 miles distant and in a least prevalent wind direction (d)		
b. Fish	(1) 1 sample each of 2 commercially or recreationally important species (n)	In the vicinity of a plant discharge area	Twice per year	Gamma isotopic analysis of each sample (g) on edible portions twice per year
	(2) 1 sample of the same species	In areas not influenced by station discharge (d)		

(continued)

Table D 3.5.1-1 (page 3 of 4)
Radiological Environmental Monitoring Program

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLES	SAMPLE LOCATIONS (a)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion (continued)				
c. Food Products	(1) 1 sample of each principal class of food products	Any area that is irrigated by water in which liquid plant wastes have been discharged (o)	At time of harvest (p)	Gamma isotopic (g) and I-131 analysis of each sample of edible portions
	(2) Samples of 3 different kinds of broad leaf vegetation (such as vegetables)	Grown nearest to each of 2 different offsite locations (e)	Once per year during the harvest season-	
	(3) 1 sample of each of the similar broad leaf vegetation.	Grown at least 9.3 miles distant in a least prevalent wind direction	Once per year during the harvest season	

Table D 3.5.1-1 (page 3 of 4)
Radiological Environmental Monitoring Program

(a) Specific parameters of distance and direction sector from the centerline of one reactor, and additional descriptions where pertinent, shall be provided for each and every sample location in Table D 3.5.1-1. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable because of such circumstances as hazardous conditions, seasonal unavailability (which includes theft and uncooperative residents), or malfunction of automatic sampling equipment.

(b) 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. Each of the 32 routine monitoring stations shall be equipped with 2 or more dosimeters or with 1 instrument for measuring and recording dose rate continuously. For the purpose of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; 2 or more phosphors in a packet are considered as 2 or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation.

(c) At this distance, 8 windrose sectors (W, WNW, NW, NNW, N, NNE, NE, and ENE) are over Lake Ontario.

(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, which provide valid background data, may be substituted.

(e) Having the highest calculated annual site average ground-level D/Q based on all site licensed reactors.

(f) Airborne particulate sample filters shall be analyzed for gross beta activity 24 hours or more after sampling to allow for radon and thoron daughter decay.

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

(h) The upstream sample shall be taken at a distance beyond significant influence of the discharge. The downstream sample shall be taken in an area beyond but near the mixing zone.

(i) In this program, representative composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

(j) Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

(k) Drinking water samples shall be taken only when drinking water is a dose pathway.

(l) Analysis for I-131 may be accomplished by Ge-Li analysis provided that the lower limit of detection (LLD) for I-131 in water samples found on Table D 3.5.1-2 can be met. Doses shall be calculated for the maximum organ and age group.

(m) Samples will be collected January through March if I-131 is detected in November and December of the preceding year.

(n) In the event 2 commercially or recreationally important species are not available, after 3 attempts of collection, then 2 samples of one species or other species not necessarily commercially or recreationally important may be utilized.

(o) Applicable only to major irrigation projects within 9 miles of the site in the general downcurrent direction.

(p) If harvest occurs more than once/year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be taken monthly. Attention should be paid to including samples of tuberous and root food products.

Table D 3.5.1-2 (page 1 of 1)
Reporting Levels for Radioactivity in Environmental Samples

RADIONUCLIDE ANALYSIS	WATER (pCi/L)	AIRBORNE PARTIUCLATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/L)	FOOD PRODUCTS (pCi/kg, wet)
H-3	20,000 (a)				
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-95	400				
Nb-95	400				
I-131	2 (b)	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-140	200			300	
La-140	200			300	

(a) For drinking water samples. This is a 40 CFR 141 value. If no drinking water pathway exists, a value of 30,000 pCi/L may be used.

(b) If no drinking water pathway exists, a value of 20 pCi/L may be used.

Table D 3.5.1-3 (page 1 of 2)
Detection Capabilities for Environmental Sample Analysis [(a) (b)]

	LOWER LIMIT OF DETECTION (LLD) [(c)]					
RADIONUCLIDE ANALYSIS	WATER (pCi/L)	AIRBORNE PARTIUCLATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/L)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	2,000 [(d)]					
Mn-54	15		130			
Fe-59	30		260			
Co-58	15		130			
Co-60	15		130			
Zn-65	30		260			
Zr-95	15					
Nb-95	15					
I-131	1 [(e)]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-140	15			15		
La-140	15			15		

See the notes on the next page

Table 3.5.1-3 (page 2 of 2)
Detection Capabilities for Environmental Sample Analysis [(a)(b)]

(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.

(b) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in ANSI N-545, Section 4.3 1975. Allowable exceptions to ANSI N-545, Section 4.3 are contained in the ODCM.

(c) The LLD is defined 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% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{(4.66)(S_b)}{(E)(V)(2.22)(Y)e^{-\lambda \Delta t}}$$

where:

LLD	=	The before-the-fact lower limit of detection (pCi per unit mass or volume),
S_b	=	The standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),
E	=	The counting efficiency (counts per disintegration),
V	=	The sample size (units of mass or volume),
2.22	=	The number of disintegrations per minute per pCi,
Y	=	The fractional radiochemical yield, when applicable,
λ	=	The radioactive decay constant for the particular radionuclide (sec^{-1}), and
Δt	=	The elapsed time between environmental collection or end of the sample collection period, and the time of counting (seconds).

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

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

(d) If no drinking water pathway exists, a value of 3,000 pCi/L may be used.

(e) If no drinking water pathway exists, a value of 15 pCi/L may be used.

D 3.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

D 3.5.2 Land Use Census

DLCO 3.5.2 A land use census shall:

a. Be conducted,

b. Identify within a distance of 5 miles the location in each of the 16 meteorological sectors of the nearest milk animal and the nearest residence, and the nearest garden (broad leaf vegetation sampling controlled by Table D 3.5.1-1, part 5.c may be performed in lieu of the garden census) of > 500 ft^2 producing broad leaf vegetation, and

c. For elevated releases, identify within a distance of 3 miles the locations in each of the 16 meteorological sectors of all milk animals and all gardens (broad leaf vegetation sampling controlled by Table D 3.5.1-1, part 5.c may be performed in lieu of the garden census) > 500 ft^2 producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTIONS

-----NOTES-----

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Land use census identifies location(s) that yields a calculated dose, dose commitment, or D/Q value > than the values currently being calculated in DSR 3.2.3.1.	A.1 Identify the new location(s) in the next Radioactive Effluent Release Report.	In accordance with the Radioactive Effluent Release Report

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Land use census identifies location(s) that yields a calculated dose, dose commitment, or D/Q value (via the same exposure pathway) 50% > than at a location from which samples are currently being obtained in accordance with Table D 3.5.1-1.	B.1 Add the new location(s) to the Radiological Environmental Monitoring Program.	30 days
	<u>AND</u>	
	B.2 Delete the sampling location(s), excluding the control station location, having the lowest calculated dose, dose commitment(s) or D/Q value, via the same exposure pathway, from the Radiological Environmental Monitoring Program.	After October 31 of the year in which the land use census was conducted
	<u>AND</u>	
	B.3 Submit in the next Radioactive Effluent Release Report documentation for a change in the ODCM including revised figure(s) and table(s) for the ODCM reflecting the new location(s) with information supporting the change in sampling locations.	In accordance with the Radioactive Effluent Release Report

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.5.2.1	Conduct the land use census during the growing season using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities.	366 days
DSR 3.5.2.2	Report the results of the land use census in the Annual Radiological Environmental Operating Report.	In accordance with the Annual Radiological Environmental Operating Report

D 3.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

D 3.5.3 Interlaboratory Comparison Program

DLCO 3.5.3 The Interlaboratory Comparison Program shall be described in the ODCM.

AND

Analyses shall be performed on all radioactive materials, supplied as part of an Interlaboratory Comparison Program that has been approved by the NRC, that correspond to samples required by Table D 3.5.1-1. Participation in this program shall include media for which environmental samples are routinely collected and for which intercomparison samples are available.

APPLICABILITY: At all times.

ACTIONS

-- NOTES --

1. LCO 3.0.3 is not applicable.
2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Analyses not performed as required.	A.1 Report the corrective actions taken to prevent a recurrence to the NRC in the Annual Radiological Environmental Operating Report.	In accordance with the Annual Radiological Environmental Operating Report

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
DSR 3.5.3.1	Report a summary of the results obtained as part of the Interlaboratory Comparison Program in the Annual Radiological Environmental Operating Report.	In accordance with the Annual Radiological Environmental Operating Report

PART I - RADIOLOGICAL EFFLUENT CONTROLS

BASES

B 3.1 RADIOACTIVE LIQUID EFFLUENTS

B 3.1.1 Liquid Effluents Concentration

BASES

This is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than ten times the concentration levels specified in 10 CFR 20, Appendix B, Table 2, 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 to 10 CFR 50, to a MEMBER OF THE PUBLIC and (2) the levels required by 10 CFR 20.1301(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 effluent 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.

This applies to the release of radioactive materials in liquid effluents from all units at the site.

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 L. A. Currie, "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

B 3.1 RADIOACTIVE LIQUID EFFLUENTS

B 3.1.2 Liquid Effluents Dose

BASES

This is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I to 10 CFR 50. This implements the guides set forth in Section II.A of Appendix I. 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 liquid effluents to UNRESTRICTED AREAS will be kept as low as is reasonably achievable. Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the potable drinking water that are in excess of the requirements of 40 CFR 141. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the units including outside storage tanks, etc., 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 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBERS OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered. The dose calculation methodology and parameters implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by Calculational procedures based on models and data, so that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified for calculating the doses that result from actual release rates of radioactive material in liquid effluents are consistent with the methodology provided in RG 1.109, "Calculation of Annual Doses To Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and R.G. 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977. This applies to the release of radioactive materials in liquid effluents from each unit at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system.

B 3.1 RADIOACTIVE LIQUID EFFLUENTS

B 3.1.3 Liquid Radwaste Treatment System

BASES

The installed liquid radwaste treatment system shall be considered OPERABLE by meeting DLCO 3.1.1 and DLCO 3.1.2. The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment before release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept as low as is reasonably achievable. This implements the requirements of 10 CFR 50.36a, GDC 60 of Appendix A to 10 CFR 50 and the design objective given in Section II.D of Appendix I to 10 CFR 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I to 10 CFR 50 for liquid effluents. This applies to the release of radioactive materials in liquid effluents from each unit at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system.

B 3.2 RADIOACTIVE GASEOUS EFFLUENTS

B 3.2.1 Gaseous Effluents Dose Rate

BASES

This is provided to ensure that the dose rate 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 10 CFR 20 to UNRESTRICTED AREAS.

The annual dose limits are the doses associated with the concentrations of 10 CFR 20, Appendix B, Table 2, 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 2 of 10 CFR 20 or as governed by 10 CFR 20.1302(c). 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. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in Part II. 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 whole 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. This applies to the release of radioactive materials in gaseous effluents from all units at the site.

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 L. A. Currie, "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environments Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

B 3.2 RADIOACTIVE GASEOUS EFFLUENTS

B 3.2.2 Gaseous Effluents Noble Gas Dose

BASES

This is provided to implement the requirements of Section II.B, III.A, and IV.A of Appendix I to 10 CFR 50. The DLCO implements the guides set forth in Section II.B of Appendix I. The REQUIRED ACTIONS 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 conformance with the guidelines of Appendix I be shown by calculational procedures based on models and data so that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the units including outside storage tanks, etc., 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 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered. The dose calculation methodology and parameters for calculating the doses from the actual release rates of radioactive noble in gaseous effluents are consistent with the methodology provided in RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977, and RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1," July 1977. The ODCM equations provided for determining the air doses at or beyond the SITE BOUNDARY are based upon real-time meteorological conditions or the historical average atmospheric conditions. This applies to the release of radioactive material in gaseous effluents from each unit at the site.

B 3.2 RADIOACTIVE GASEOUS EFFLUENTS

B 3.2.3 Gaseous Effluents Dose – Iodine-131, Iodine-133, Tritium, and Radioactive Material In Particulate Form

BASES

This is provided to implement the requirements of Sections II.C, III.A, and IV.A of Appendix I to 10 CFR 50. The DLCO implements the guides set forth in Section II.C of Appendix I. The REQUIRED ACTIONS 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 calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, so that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. . For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the units including outside storage tanks, etc., 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 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered. The calculational methodology and parameters for calculating the doses from the actual release rates of the subject materials are consistent with the methodology provided in RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, " Revision 1, October 1977, and RG 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 DLCO for iodine-131, iodine-133, tritium, and radioactive material in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at or beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radioactive material, (2) deposition of radioactive material onto green leafy vegetation

B 3.2.3 Gaseous Effluents Dose – Iodine-131, Iodine-133, Tritium, and Radioactive Material In Particulate Form (continued)

with subsequent consumption by man, (3) deposition onto grassy areas where milk-producing animals and meat-producing animals graze (human consumption of the milk and meat is assumed), and (4) deposition on the ground with subsequent exposure to man. This applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

B 3.2 RADIOACTIVE GASEOUS EFFLUENTS

B 3.2.4 Gaseous Radwaste Treatment System

BASES

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM ensures that the system will be available for use whenever gaseous effluents require treatment before release to the environment. The requirement that the appropriate portions of this system be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept as low as is reasonably achievable. This implements the requirements of 10 CFR 50.36a, GDC 60 of Appendix A to 10 CFR 50, and the design objectives given in Section II.D of Appendix I to 10 CFR 50. Limits governing the use of appropriate portions of the system were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I to 10 CFR 50, for gaseous effluents. This applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportional among the units sharing that system.

B 3.2 RADIOACTIVE GASEOUS EFFLUENTS

B 3.2.5 Ventilation Exhaust Treatment System

BASES

The OPERABILITY of the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the system will be available for use whenever gaseous effluents require treatment before release to the environment. The requirement that the appropriate portions of this system be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept as low as is reasonably achievable. This implements the requirements of 10 CFR 50.36a, GDC 60 of Appendix A to 10 CFR 50, and the design objectives given in Section II.D of Appendix I to 10 CFR 50. Limits governing the use of appropriate portions of the system were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I to 10 CFR 50, for gaseous effluents. This applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportional among the units sharing that system.

The appropriate components, which affect iodine or particulate release, to be OPERABLE are:

1) HEPA Filter – Radwaste Decon Area
2) HEPA Filter – Radwaste Equipment Area
3) HEPA Filter – Radwaste General Area

Whenever one of these filters is not OPERABLE, iodine and particulate dose projections will be made for 31-day intervals starting with filter inoperability, and continuing as long as the filter remains inoperable, in accordance with DSR 3.2.5.1.

B 3.2 RADIOACTIVE GASEOUS EFFLUENTS

B 3.2.6 Venting or Purging

BASES

This provides reasonable assurance that releases from drywell and/or suppression chamber purging operations will not exceed the annual dose limits of 10 CFR 20 for unrestricted areas.

B 3.3 INSTRUMENTATION

B 3.3.1 Radioactive Liquid Effluent Monitoring Instrumentation

BASES

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in Part II to ensure that the alarm/trip will occur before exceeding ten times the limits of 10 CFR 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of GDC 60, 63, and 64 of Appendix A to 10 CFR 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.

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

B 3.3 INSTRUMENTATION

B 3.3.2 Radioactive Gaseous Effluent Monitoring Instrumentation

BASES

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in Part II to ensure that the alarm/trip will occur before exceeding the limits of 10 CFR 20. Although the Offgas System Noble Gas Activity Monitor is listed in Table D 3.3.2-1, "Radioactive Gaseous Effluent Monitoring Instrumentation", these monitors are actually located upstream of the Main Stack noble gas activity monitor and are not effluent monitors. They were included in Table D 3.3.2-1 in accordance with NUREG-0473. As such, Offgas System Noble Gas Activity Monitor alarm and trip setpoints are not based on 10CFR20. The offgas system noble gas monitor alert setpoint is set at 1.5 times nominal full power background to assure compliance with ITS SR 3.7.4.1 which requires offgas sampling be performed within four hours of a 50% increase in offgas monitoring readings, and to support MSLRM trip removal. The offgas system noble gas monitor trip setpoint is based on the 10CFR100 limits for the limiting design basis gaseous waste system accident which is the offgas system rupture. The range of the noble gas channels of the main stack and radwaste/reactor building vent effluent monitors is sufficiently large to envelope both normal and accident levels of noble gas activity. The capabilities of these instruments are consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of the TMI Action Plan Requirements," November 1980. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the offgas system. The OPERABILITY and use of this instrumentation is consistent with the requirements of GDC 60, 63, and 64 of Appendix A to 10 CFR 50.

B 3.4 RADIOACTIVE EFFLUENTS TOTAL DOSE

BASES

This is provided to meet the dose limitations of 40 CFR 190 that have been incorporated into 10 CFR 20 by 46 FR 18525. This requires the preparation and submittal of a Special Report whenever the calculated doses from releases of radioactivity and from radiation from uranium fuel cycle sources exceed 25 mrem to the whole body or any organ, except the thyroid (which shall be limited to less than or equal to 75 mrem). If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR 20, as addressed in 3.1.1 and 3.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which the individual is engaged in carrying out any operation that is part of the nuclear fuel cycle.

B 3.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

B 3.5.1 Monitoring Program

BASES

The Radiological Environmental Monitoring Program provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposure of MEMBERS OF THE PUBLIC resulting from the plant operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR 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, Revision 1, November 1979. 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 D 3.5.1-3 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 L. A. Currie, "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

B 3.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

B 3.5.2 Land Use Census

BASES

This is provided to ensure that changes in the use of areas at or 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 information, such as from a door-to-door survey, 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 10 CFR 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in RG 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage) and (2) the vegetation yield was 2 kg/m^2.

A MILK SAMPLING LOCATION, as defined in Section 1.0, 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 a month for analytical purposes. Locations with fewer than 10 milking cows are usually utilized for breeding purposes, eliminating a stable supply of milk for samples as a result of suckling calves and periods when the adult animals are dry. Elevated releases are defined in RG 1.111, Revision 1, July 1977.

B 3.5 RADIOLOGICAL ENVIRONMENTAL MONITORING

B 3.5.3 Interlaboratory Comparison Program

BASES

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 materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR 50.

PART I - RADIOLOGICAL EFFLUENT CONTROLS

SECTION 4.0 ADMINISTRATIVE CONTROLS

4.0 ADMINISTRATIVE CONTROLS

The ODCM Specifications are subject to Technical Specifications Section 5.5.4, "Radioactive Effluent Controls Program," Section 5.6.2, "Annual Radiological Environmental Operating Report," Section 5.6.3, "Radioactive Effluent Release Report," and Section 5.5.1, "Offsite Dose Calculation Manual."

D 4.1 REPORTING REQUIREMENTS

D 4.1.1 Special Reports

Special Reports shall be submitted in accordance with 10 CFR 50.4 within the time period specified for each report.

D 4.1.2 Annual Radiological Environmental Operating Reports

In addition to the requirements of Technical Specification 5.6.2 the report shall also include the following:

A summary description of the Radiological Environmental Monitoring Program; at least two legible maps, one shall cover stations near the SITE BOUNDARY and the second shall include the more distant stations, covering all sample locations keyed to a table giving distances and directions from the centerline of one reactor; the results of license participation in the Interlaboratory Comparison Program, required by Control D 3.5.3; discussion of all deviations from the Sampling Schedule of Table D 3.5.1-1; and discussion of all analysis in which the LLD required by Table D 3.5.1-3 was not achievable.

D 4.1.3 Radioactive Effluent Release Report

The Radiological Effluent Release Report described in Technical Specification section 5.6.3 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 distribution of wind speed, wind direction, and atmospheric stability. In lieu of submission with the Radiological Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.
- An assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during the previous year.

(Continued)

D 4.1.3 Radioactive Effluent Release Report (continued)

- As assessment of radiation doses from the radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC from their activities inside the SITE BOUNDARY 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 Part II.
- As assessment of 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 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Part II.
- A list of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.
- Any changes made during the reporting period to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL (ODCM).
- Any major changes to liquid, gaseous, or solid radwaste treatment systems pursuant to D 4.2.
- A listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Control D 3.5.2.
- An explanation of why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Controls D 3.3.1 and D 3.3.2.
- Description of events leading to liquid holdup tanks exceeding the limits of TRM 3.7.7.

D 4.2 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEM

-----NOTE-----

Licensees may choose to submit this information as part of the annual FSAR update.

Licensee-initiated major changes to the radwaste treatment systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Radioactive Effluent Release report for the period in which the evaluation was reviewed by the SORC. The discussion of each change shall contain:

   1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

   2. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

   4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

   5. An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

   6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period that precedes the time when the change is to be made;

   7. An estimate of the exposure to plant operating personnel as a result of the change; and

(Continued)

D 4.2 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEM (continued)

8. Documentation of the fact that the change was reviewed and found acceptable by the SORC.

b. Shall become effective upon review and acceptance by the SORC.

PART II - CALCULATIONAL METHODOLOGIES

1.0 LIQUID EFFLUENTS

Service Water A and B, Cooling Tower Blowdown and the Liquid Radioactive Waste Discharges comprise the Radioactive Liquid Effluents at Unit 2. Presently there are no temporary outdoor tanks containing radioactive water capable of affecting the nearest known or future water supply in an unrestricted area. NUREG 0133 and Regulatory Guide 1.109, Rev. 1 were followed in the development of this section.

1.1 Liquid Effluent Monitor Alarm Setpoints

1.1.1 Basis

The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure D 1.0-1) shall be limited to ten times the concentrations specified in 10 CFR 20, Appendix B, Table 2, Column 2, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained nobles gases, the concentration shall be limited to 2E-04 uCi/ml total activity.

1.1.2 Setpoint Determination Methodology

1.1.2.1 Liquid Radwaste Effluent Radiation Alarm Setpoint

The Liquid Radioactive Waste System Tanks are pumped to the discharge tunnel which in turn flows directly to Lake Ontario. At the end of the discharge tunnel in Lake Ontario, a diffuser structure has been installed. Its purpose is to maintain surface water temperatures low enough to meet thermal pollution limits. However, it also assists in the near field dilution of any activity released. Service Water and the Cooling Tower Blowdown are also pumped to the discharge tunnel and will provide dilution. If the Service Water or the Cooling Tower Blowdown is found to be contaminated, then its activity will be accounted for when calculating the permissible radwaste effluent flow for a Liquid Radwaste discharge. The Liquid Radwaste System Monitor provides alarm and automatic termination of release if radiation levels above its alarm setpoint are detected.

The radiation detector is a sodium iodide crystal. It is a scintillation device. The crystal is sensitive to gamma and beta radiation. However, because of the metal walls of the sample chamber and the absorption characteristics of water, the monitor is not particularly sensitive to beta radiation. Actual detector response $\sum_i (CG_i/CF_i)$, cpm, has been evaluated by placing a sample of typical radioactive waste into the monitor and recording the gross count rate, cpm. A calibration ratio was developed by dividing the noted detector response, $\sum_i (CG_i/CF_i)$ cpm, by total concentration of activity $\sum_i (CG_i)$, uCi/cc. The quantification of the gamma activity was completed with gamma spectrometry equipment whose calibration is traceable to NIST. This calibration ratio verified the manufacturer's prototype calibration, and any subsequent transfer calibrations performed. The current calibration factor (expressed as the reciprocal conversion factor, uCi/ml/cpm), will be used for subsequent setpoint calculations in the determination of detector response:

$$\sum_i (CG_i/CF_i) = \sum_i (CG_i)/CF$$

Where the factors are as defined above.

For the calculation of RDF = $\sum$ MEC fraction = $\sum_i (C_i/MEC_i)$ the contribution from non gamma emitting nuclides except tritium will be initially estimated based on the expected ratios to quantified nuclides as listed in the FSAR Table 11.2.5. Fe-55, Sr-89 and Sr-90 are 2.5, 0.25 and 0.02 times the concentration of Co-60. The contribution will be estimated using the results from the latest analysis of composite samples, when available.

Tritium concentration is assumed to equal the latest concentration detected in the monthly tritium analysis (performed offsite) of liquid radioactive waste tanks discharged.

Nominal flow rates of the Liquid Radioactive Waste System Tanks discharged is < 165 gpm while dilution flow from the Service Water Pumps, and Cooling Tower Blowdown cumulatively is typically over 10,200 gpm. Because of the large amount of dilution the alarm setpoint could be substantially greater than that which would correspond to the concentration actually in the tank. Potentially a discharge could continue even if the distribution of nuclides in the tank were substantially different from the grab sample obtained prior to discharge which was used to establish the detector alarm point. To avoid this possibility of "Non representative Sampling" resulting in erroneous assumptions about the discharge of a tank, the tank is recirculated for a minimum of 2.5 tank volumes prior to sampling.

This monitor's setpoint takes into account the dilution of Radwaste Effluents provided by the Service Water and Cooling Tower Blowdown flows. Detector response for the nuclides to be discharged (cpm) is multiplied by the Actual Dilution Factor (dilution flow/waste stream flow) and divided by the Required Dilution Factor (total fraction of the effluent concentration in the waste stream). A safety factor is used to ensure that the limit is never exceeded. Service Water and Cooling Tower Blowdown are normally non-radioactive. If they are found to be contaminated prior to a Liquid Radwaste discharge then an alternative equation is used to take into account the contamination. If they become contaminated during a Radwaste discharge, then the discharge will be immediately terminated and the situation fully assessed.

Normal Radwaste Effluent Alarm Setpoint Calculation:

Alarm Setpoint $\leq$ 0.8 * TDF/PEF * TGC/CF * 1/RDF + Background.

Where:

Alarm Setpoint	=	The Radiation Detector Alarm Setpoint, cpm
0.8	=	Safety Factor, unitless
TDF	=	Nonradioactive dilution flow rate, gpm. Service Water Flow (ranges from 30,000 to 58,000 gpm) + Blowdown flow (typically 10,200 gpm) - Tempering

C_i	=	Concentration of isotope i in Radwaste tank prior to dilution, μCi/ml (gamma + non-gamma emitters)
CF_i	=	Detector response for isotope i, net μCi/ml/cpm See Table D 2-1 for a list of nominal values
PEF	=	The permissible Radwaste Effluent Flow rate, gpm, 165 gpm is the maximum value used in this equation
MEC_i	=	Maximum Effluent Concentration, ten times the limiting effluent concentration for isotope i from 10 CFR 20 Appendix B, Table 2, Column 2, μCi/ml
Background	=	Detector response when sample chamber is filled with nonradioactive water, cpm
CF	=	Monitor Conversion Factor, μCi/ml/cpm, determined at each calibration of the effluent monitor
CG_i	=	Concentration of gamma emitting nuclide in Radwaste tank prior to dilution, μCi/ml
$TGC = \sum CG_i$	=	Summation of all gamma emitting nuclides (which monitor will respond to)
$\sum (CG_i/CF_i)$	=	The total detector response when exposed to the concentration of nuclides in the Radwaste tank, cpm
$RDF = \sum_i (C_i/MEC_i)$	=	The total fraction of ten times the 10 CFR 20, Appendix B, Table 2, Column 2 limit that is in the Radwaste tank, unitless. This is also known as the Required Dilution Factor (RDF), and includes non-gamma emitters
TGC/CF	=	An approximation to $\sum_i (CG_i/CF_i)$ using CF determined at each calibration of the effluent monitor
TDF/PEF	=	An approximation to (TDF + PEF)/PEF, the Actual Dilution Factor in effect during a discharge.
Tempering	=	A diversion of some fraction of discharge flow to the intake canal for the purpose of temperature control, gpm.

Permissible effluent flow, PEF, shall be calculated to determine that the maximum effluent concentration will not be exceeded in the discharge canal.

$$PEF = \frac{TDF}{(RDF)\ 1.5}$$

If Actual Dilution Factor is set equal to the Required Dilution Factor, then the alarm points required by the above equations correspond to a concentration of 80% of the Radwaste Tank concentration. No discharge could occur, since the monitor would be in alarm as soon as the discharge commenced. To avoid this situation, maximum allowable radwaste discharge flow is calculated using a multiple (usually 1.5 to 2) of the Required Dilution Factor, resulting in discharge canal concentration of 2/3 to 1/2 of the maximum effluent concentration prior to alarm and termination of release. In

performing the alarm calculation, the smaller of 165 gpm (the maximum possible flow) and PEF will be used.

To ensure the alarm setpoint is not exceeded, an alert alarm is provided. The alert alarm will be set in accordance with the equation above using a safety factor of 0.5 (or lower) instead of 0.8.

1.1.2.2 Contaminated Dilution Water Radwaste Effluent Monitor Alarm Setpoint Calculation:

The allowable discharge flow rate for a Radwaste tank, when one of the normal dilution streams (Service Water A, Service Water B, or Cooling Tower Blowdown) is contaminated, will be calculated by an iterative process. Using Radwaste tank concentrations with a total liquid effluent flow rate, the resulting fraction of the maximum effluent concentration in the discharge canal will be calculated.

$$FMEC = \sum_s [F_s / \sum_s (F_s) \sum_i (C_{is} \div MEC_i)]$$

Then the permissible radwaste effluent flow rate is given by:

$$PEF = \frac{\text{Total Radwaste Effluent Flow}}{FMEC}$$

The corresponding Alarm Setpoint will then be calculated using the following equation, with PEF limited as above.

$$\text{Alarm Setpoint} \leq 0.8 \quad \frac{TGC/CF}{FMEC} \quad + \text{Background}$$

Where:

Alarm Setpoint	=	The Radiation Detector Alarm Setpoint, cpm
0.8	=	Safety Factor, Unitless
F_s	=	An Effluent flow rate for stream s, gpm
C_i	=	Concentration of isotope i in Radwaste tank prior to dilution, μCi/ml
C_{is}	=	Concentration of isotope i in Effluent stream s including the Radwaste Effluent tank undiluted, μCi/ml
CF	=	Average detector response for all isotopes in the waste stream, net μCi/ml/cpm
MEC_i	=	Maximum Effluent Concentration, ten times the effluent concentration limit for isotope i from 10CFR20 Appendix B, Table 2, Column 2, μCi/ml
PEF	=	The permissible Radwaste Effluent Flow rate, gpm
Background	=	Detector response when sample chamber is filled with nonradioactive water, cpm

TGC/CF = $\sum_i (CG_i/CF)$	=	The total detector response when exposed to the concentration of nuclides in the Radwaste tank, cpm
$\sum_s [F_s C_{is}]$	=	The total activity of nuclide i in all Effluent streams, μCi-gpm/ml
$\sum_s [F_s]$	=	The total Liquid Effluent Flow rate, gpm (Service Water & CT Blowdown & Radwaste)

1.1.2.3 Service Water and Cooling Tower Blowdown Effluent Alarm Setpoint

These monitor setpoints do not take any credit for dilution of each respective effluent stream. Detector response for the distribution of nuclides potentially discharged is divided by the total MEC fraction of the radionuclides potentially in the respective stream. A safety factor is used to ensure that the limit is never exceeded.

Service Water and Cooling Tower Blowdown are normally non-radioactive. If they are found to be contaminated by statistically significant increase in detector response then grab samples will be obtained and analysis meeting the LLD requirements of Table D 3.1.1-1 completed so that an estimate of offsite dose can be made and the situation fully assessed.

Service Water A and B and the Cooling Tower Blowdown are pumped to the discharge tunnel which in turn flows directly to Lake Ontario. Normal flow rates for each Service Water Pump is 10,000 gpm while that for the Cooling Tower Blowdown may be as much as 10,200 gpm. Credit is not taken for any dilution of these individual effluent streams.

The radiation detector is a sodium iodide crystal. It is a scintillation device. The crystal is sensitive to gamma and beta radiation. However, because of the metal walls in its sample chamber and the absorption characteristics of water, the monitor is not particularly sensitive to beta radiation.

Detector response $\sum_i (C_i/CF_i)$ has been evaluated by placing a diluted sample of Reactor Coolant (after a two hour decay) in a representative monitor and noting its gross count rate. Reactor Coolant was chosen because it represents the most likely contaminant of Station Waters.

A two hour decay was chosen by judgement of the staff of Nine Mile Point. Reactor Coolant with no decay contains a considerable amount of very energetic nuclides which would bias the detector response term high. However assuming a longer than 2 hour decay is not realistic as the most likely release mechanism is a leak through the Residual Heat Removal Heat Exchangers which would contain Reactor Coolant during shutdowns.

Service Water and Cooling Tower Blowdown Alarm Setpoint Equation:

Alarm Setpoint $< 0.8\ 1/CF \sum_i C_i/[\sum_i (C_i/MEC_i)] +$ Background.

Where:

Alarm Setpoint	=	The Radiation Detector Alarm Setpoint, cpm
0.8	=	Safety Factor, unitless
C_i	=	Concentration of isotope i in potential contaminated stream, μCi/ml
CF_i	=	Detector response for isotope i, net μCi/ml/cpm See Table 2-1 for a list of nominal values
MEC_i	=	Maximum Effluent Concentration, ten times the effluent concentration limit for isotope i from 10 CFR 20 Appendix B, Table 2, Column 2, μCi/ml
Background	=	Detector response when sample chamber is filled with nonradioactive water, cpm
$\sum_i (C_i/CF_i)$	=	The total detector response when exposed to the concentration of nuclides in the potential contaminant, cpm
$\sum_i (C_i/MEC_i)$	=	The total fraction of ten times the 10CFR20, Appendix B, Table 2, Column 2 limit that is in the potential contaminated stream, unitless.
$(1/CF)\sum_i C_i$	=	An approximation to $\sum_i (C_i/CF_i)$, determined at each calibration of the effluent monitor
CF	=	Monitor Conversion Factor, μCi/ml/cpm

1.2 Liquid Effluent Concentration Calculation

This calculation documents compliance with Section D 3.1.1 of Part I:

As required by Technical Specification 5.5.4, "Radioactive Effluent Controls Program," the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure D 1.0-1) shall be limited to ten times the concentrations specified in 10 CFR 20, Appendix B, Table 2, Column 2, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2E-04 microcurie/ml total activity.

The concentration of radioactivity from Liquid Radwaste, Service Water A and B and the Cooling Tower Blowdown are included in the calculation. The calculation is performed for a specific period of time. No credit is taken for averaging. The limiting concentration is calculated as follows:

$$FMEC = \sum_s [F_s/\sum_s (F_s) \sum_i (C_{is} \div MEC_i)]$$

Where: FMEC = The Fraction of Maximum Effluent Concentration, the ratio at the point of discharge

of the actual concentration to ten times the limiting concentration of 10 CFR 20, Appendix B, Table 2, Column 2, for radionuclides other than dissolved or entrained noble gases, unitless

C_{is} = The concentration of nuclide i in a particular effluent stream s, μCi/ml

F_s = The flow rate of a particular effluent stream s, gpm

MEC_i = Maximum Effluent Concentration, ten times the limiting Effluent Concentration of a specific nuclide i from 10CFR20, Appendix B, Table 2, Column 2 (for noble gases, the concentration shall be limited to 2E-4 microcurie/ml), μCi/ml

$\sum_i (C_{is}/MEC_i)$ = The Maximum Effluent Concentration fraction of stream s prior to dilution by other streams

$\sum_s (F_s)$ = The total flow rate of all effluent streams s, gpm

A value of less than one for the MEC fraction is required for compliance.

1.3 Liquid Effluent Dose Calculation Methodology

The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Figure D 1.0-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the whole 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 whole body and to less than or equal to 10 mrem to any organ.

Doses due to Liquid Effluents are calculated monthly for the fish and drinking water ingestion pathways and the external sediment exposure pathways from all detected nuclides in liquid effluents released to the unrestricted areas using the following expression from NUREG 0133, Section 4.3.

$$D_t = \sum_i [A_{it} \sum_L (\Delta T_L C_{iL} F_L)]$$

Where:

D_t = The cumulative dose commitment to the total body or any organ, t from the liquid effluents for the total time period $\sum_L (\Delta T_L)$, mrem

ΔT_L = The length of the L th time period over which C_{iL} and F_L are averaged for all liquid releases, hours

C_{iL} = The average concentration of radionuclide, i, in undiluted liquid effluents during time period ΔT_L from any liquid release, μCi/ml

A_{it} = The site related ingestion dose commitment factor for the maximum individual to the total body or any or gan t for each identified principal gamma or beta emitter, mrem/hr per μCi/ml. Table D 2-2.

F_L = The near field average dilution factor for C_{it} during any liquid effluent release. Defined as the ratio of the maximum undiluted liquid waste flow during release to the product of the average flow from the site discharge structure to unrestricted receiving waters times 5.9. (5.9 is the site specific applicable factor for the mixing effect of the discharge structure.) See the Nine Mile Point Unit 2 Environmental Report - Operating License Stage, Table 5.4-2 footnote 1.

These factors can be related to batch release parameters as follows:

F_L = PEF / (TDF x 5.9) (Terms defined in Section 1.1.2.1 and above)

$\Delta T_L F_L$ = [PEF (gpm) x ΔT_L (min) x 1.67E-2 (hr/min)] / [TDF (gpm) x 5.9]

= [TV x 2.83E-3 (hours)] / TDF

For each batch, PEF (gpm) x ΔT_L (min) = Tank Volume. For each batch, a dose calculation common constant ($\Delta T_L F_L$) is calculated to be used with the concentration of each nuclide and dose factor, A_i, to calculate the dose to a receptor. Normally, the highest dose factor for any age group (adult, teen, child, infant) will be used for calculation, but specific age-group calculations to demonstrate compliance may be performed if required.

1.4 Liquid Effluent Sampling Representativeness

There are four tanks in the radwaste system designed to be discharged to the discharge canal. These tanks are labeled 4A, 4B, 5A, and 5B.

Liquid Radwaste Tank 5A and 5B at Nine Mile Point Unit 2 contain a sparger spray ring which assists the mixing of the tank contents while it is being recirculated prior to sampling. This sparger effectively mixes the tank four times faster than simple recirculation.

Liquid Radwaste Tank 4A and 4B contain a mixing ring but no sparger. No credit is taken for the mixing effects of the ring. Normal recirculation flow is 150 gpm for tank 5A and 5B, 110 gpm for tank 4A and 4B while each tank contains up to 25,000 gallons although the entire contents are not discharged. To assure that the tanks are adequately mixed prior to sampling, it is a plant requirement that the tank be recirculated for the time required to pass 2.5 times the volume of the tank:

$$\text{Recirculation Time} = 2.5T/RM$$

Where:

Recirculation Time	=	Is the minimum time to recirculate the Tank, min
2.5	=	Is the plant requirement, unitless
T	=	Is the tank volume, gal
R	=	Is the recirculation flow rate, gpm.
M	=	Is the factor that takes into account the mixing of the sparger, unitless, four for tank 5A and B, one for tank 4A and B.

Additionally, the Alert Alarm setpoint of the Liquid Radwaste Effluent monitor is set at approximately 60% of the High alarm setpoint. This alarm will give indication of incomplete mixing with adequate margin before exceeding ten times the effluent concentration.

Service Water A and B and the Cooling Tower Blowdown are sampled from the radiation monitor on each respective stream. These monitors continuously withdraw a sample and pump it back to the effluent stream. The length of tubing between the continuously flowing sample and the sample spigot contains less than 200 ml which is adequately purged by requiring a purge of at least 1 liter when grabbing a sample.

1.5 Liquid Radwaste System Operability

The Liquid Radwaste Treatment System shall be OPERABLE and used when projected doses due to liquid radwaste effluents would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31-day period. Cumulative doses will be determined at least once per 31 days (as indicated in Section 1.3) and doses will also be projected if the radwaste treatment systems are not being fully utilized.

The system collection tanks are processed as follows:

1) Low Conductivity (Waste Collector): Radwaste Filter and Radwaste Demineralizer or the Thermex System.

2) High Conductivity (Floor Drains): Regenerant Evaporator or the Thermex System.

3) Regenerant Waste: If resin regeneration is used at NMP-2; the waste will be processed through the regenerant evaporator or Thermex System.

The dose projection indicated above will be performed in accordance with the methodology of Section 1.3.

2.0 GASEOUS EFFLUENTS

The gaseous effluent release points are the stack and the combined Radwaste/Reactor Building vent. The stack effluent point includes Turbine Building ventilation, main condenser offgas (after charcoal bed holdup), and Standby Gas Treatment System exhaust. NUREG 0133 and Regulatory Guide 1.109, Rev. 1 were followed in the development of this section.

2.1 Gaseous Effluent Monitor Alarm Setpoints

2.1.1 Basis

The dose rate from radioactive materials released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY (see Figure D 1.0-1) shall be limited to the following in accordance with Technical Specification 5.5.4.g:

a. For noble gases: Less than or equal to 500 mrem/yr to the whole body and less than or equal to 3000 mrem/yr to the skin, and

b. For iodine-131, for iodine-133, for tritium, and for all radionuclides with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

The radioactivity rate of noble gases measured downstream of the recombiner shall be limited to less than or equal to 350,000 microcuries/second during offgas system operation in accordance with Technical Specification 3.7.4.

2.1.2 Setpoint Determination Methodology Discussion

Nine Mile Point Unit 1 and the James A FitzPatrick nuclear plants occupy the same site as Nine Mile Point Unit 2. Because of the independence of these plants' safety systems, control rooms and operating staffs it is assumed that simultaneous accidents are not likely to occur at the different units. However, there are two release points at Unit 2. It is assumed that if an accident were to occur at Unit 2 that both release points could be involved.

The alarm setpoint for Gaseous Effluent Noble Gas Monitors are based on a dose rate limit of 500 mRem/yr to the Whole Body. Since there are two release points at Unit 2, the dose rate limit of 500 mRem/yr is divided equally for each release point, but may be apportioned otherwise, if required. These monitors are sensitive to only noble gases. Because of this it is considered impractical to base their alarm setpoints on organ dose rates due to iodines or particulates. Additionally skin dose rate is never significantly greater than the whole body dose rate. Thus the factor R which is the basis for the alarm setpoint calculation is nominally taken as equal to 250 mRem/yr. If there are significant releases from any gaseous release point on the site (>25 mRem/yr) for an extended period of time then the setpoint will be recalculated with an appropriately smaller value for R.

The high alarm setpoint for the Offgas Noble Gas monitor is based on a limit of 350,000 uCi/sec. This is the release rate for which a FSAR accident analysis was

completed. At this rate the Offgas System charcoal beds will not contain enough activity so that their failure and subsequent release of activity will present a significant offsite dose assuming accident meteorology.

Initially, in accordance with Part I, Section D 3.3.2, the Germanium multichannel analysis systems of the stack and vent will be calibrated with gas standards (traceable to NIST) in accordance with DSR 3.3.2.9. Subsequent calibrations may be performed with gas standards, or with related solid sources. The quarterly Channel Functional Test will include operability of the 30cc chamber and the dilution stages to confirm monitor high range capability. (Appendix D, Gaseous Effluent Monitoring System).

2.1.2.1 Stack Noble Gas Detector Alarm Setpoint Equation:

The stack at Nine Mile Point Unit 2 receives the Offgas after charcoal bed delay, Turbine Building Ventilation and the Standby Gas Treatment system exhaust. The Standby Gas Treatment System Exhausts the primary containment during normal shutdowns and maintains a negative pressure on the Reactor Building to maintain secondary containment integrity. The Standby Gas Treatment will isolate on high radiation detected (by the SGTS monitor) during primary containment purges.

The stack noble gas detector is made of germanium. It is sensitive to only gamma radiation. However, because it is a computer based multichannel analysis system it is able to accurately quantify the activity released in terms of uCi of specific nuclides. Only pure alpha and beta emitters are not detectable, of which there are no common noble gases. A distribution of Noble Gases corresponding to offgas is chosen for the nominal alarm setpoint calculation. Offgas is chosen because it represents the most significant contaminant of gaseous activity in the plant. The release rate Q_i, corresponds to offgas concentration expected with the plant design limit for fuel failure. The alarm setpoint may be recalculated if a significant release is encountered. In that case the actual distribution of noble gases will be used in the calculation.

The following calculation will be used for the initial Alarm Setpoint.

$$\text{Alarm Setpoint, } \mu\text{Ci/sec} \leq \frac{0.8R \quad \sum_i (Q_i)}{\sum_i (Q_i V_i)}$$

0.8	=	Safety Factor, unitless
R	=	Allocation Factor. Normally, 250 mrem/yr; the value must be 500 mrem/yr or less depending upon the dose rate from other release points within the site such that the total dose rate corresponds to < 500 mrem/yr
Q_i	=	The release rate of nuclide i, μCi/sec
V_i	=	The constant for each identified noble gas nuclide accounting for the whole body dose from the elevated finite plume listed on Table D 3-2, mrem/yr per μCi/sec

$\sum_i (Q_i)$ = The total release rate of noble gas nuclides in the stack effluent, μCi/sec

$\sum_i (Q_i V_i)$ = The total of the product of each isotope release rate times its respective whole body plume constant, mrem/yr, μCi/sec

The alert alarm is normally set at less than 10% of the high alarm.

2.1.2.2 Vent Noble Gas Detector Alarm Setpoint Equation:

The vent contains the Reactor Building ventilation above and below the refuel floor and the Radwaste Building ventilation effluents. The Reactor Building Ventilation will isolate when radiation monitors detect high levels of radiation (these are separate monitors, not otherwise discussed in the ODCM). Nominal flow rate for the vent is 2.37E5 CFM.

This detector is made of germanium. It is sensitive to only gamma radiation. However, because it is a computer based multichannel analysis system it is able to accurately quantify the activity released in terms of μCi of specific nuclides. Only pure alpha and beta emitters are not detectable, of which there are no common noble gases. A distribution of Noble Gases corresponding to that expected with the design limit for fuel failure offgas is chosen for the nominal alarm setpoint calculation. Offgas is chosen because it represents the most significant contaminant of gaseous activity in the plant. The alarm setpoint may be recalculated if a significant release is encountered. In that case the actual distribution of noble gases will be used in the calculation.

$$\text{Alarm Setpoint, uCi/sec} < \frac{0.8R \sum_i (Q_i)}{(X/Q)_v \sum_i (Q_i K_i)}$$

Where:

0.8	=	Safety Factor, unitless
R	=	Allocation Factor. Normally, 250 mrem/yr; the value must be 500 mrem/yr or less depending upon the dose rate from other release points within the site such that the total rate corresponds to < 500 mrem/yr
Q_i	=	The release rate of nuclide i, μCi/sec
$(X/Q)_v$	=	The highest annual average atmospheric dispersion coefficient at the site boundary as listed in the Final Environmental Statement, NUREG 1085, Table D-2, 2.0E-6 sec/m^3
K_i	=	The constant for each identified noble gas nuclide accounting for the whole body dose from the semi-infinite cloud, listed on Table D 3-3, mrem/yr per μCi/m^3

$\sum_i(Q_i)$ = The total release rate of noble gas nuclides in the vent effluent, uCi/sec

$\sum_i(Q_iK_i)$ = The total of the product of the each isotope release rate times its respective whole body immersion constant, mrem/yr per sec/m^3

The alert alarm is normally set at less than 10% of the high alarm.

2.1.2.3 Offgas Pretreatment Noble Gas Detector Alarm Setpoint Equation:

The Offgas system has a radiation detector downstream of the recombiners and before the charcoal decay beds. The offgas, after decay, is exhausted to the main stack. The system will automatically isolate if its pretreatment radiation monitor detects levels of radiation above the high alarm setpoint.

The Radiation Detector contains a plastic scintillator disc. It is a beta scintillation detector. Detector response $\sum_i(C_i/CF_i)$ has been evaluated from isotopic analysis of offgas analyzed on a multichannel analyzer, traceable to NIST. A distribution of offgas corresponding to that expected with the design limit for fuel failure was used to establish the initial setpoint. However, the alarm setpoint may be recalculated using an updated nuclide distribution based on actual plant process conditions. The monitor nominal response values will be confirmed during periodic calibration using a Transfer Standard source traceable to the primary calibration performed by the vendor.

Particulates and Iodines are not included in this calculation because this is a noble gas monitor.

To provide an alarm in the event of failure of the offgas system flow instrumentation, the low flow alarm setpoint will be set at or above 10 scfm, (well below normal system flow) and the high flow alarm setpoint will be set at or below 110 scfm, which is well above expected steady-state flow rates with a tight condenser.

To provide an alarm for changing conditions, the alert alarm will normally be set at 1.5 times nominal full power background to ensure that the Specific Activity Action required by ITS SR 3.7.4.1, are implemented in a timely fashion.

$$\text{Alarm Setpoint, cpm} \leq 0.8 \; \frac{(3.50E{+}05)\,(2.12\ E{-}03)\ \sum_i(C_i/CF_i)}{F\ \sum_i(C_i)} + \text{Background}$$

Where:

Alarm Setpoint = The alarm setpoint for the offgas pretreatment Noble Gas Detector, cpm

0.8 = Safety Factor, unitless

350,000	=	The Technical Specification Limit for Offgas Pretreatment, µCi/sec
2.12E-03	=	Unit conversion Factor, 60 sec/min / 28317 cc/CF
C_i	=	The concentration of nuclide, i, in the Offgas, µCi/cc
CF_i	=	The Detector response to nuclide i, µCi/cc/cpm; See Table D 3-1 for a list of nominal values
F	=	The Offgas System Flow rate, CFM
Background	=	The detector response to non-fission gases and general area dose rates, cpm
$\sum_i (C_i/CF_i)$	=	The summation of the nuclide concentration divided by the corresponding detector response, net cpm
$\sum_i (C_i)$	=	The summation of the concentration of nuclides in offgas, µCi/cc

2.2 Gaseous Effluents Dose Rate Calculation

Dose rates will be calculated monthly at a minimum to demonstrate that the release of noble gases, tritium, iodines, and particulates with half lives greater than 8 days are within the dose rate limits specified in 10CFR20. These limits are as follows:

The dose rate from radioactive materials released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY (see Figure D 1.0-1) shall be limited per Technical Specification 5.5.4.g to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the whole body and less than or equal to 3000 mrem/yr to the skin, and

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

2.2.1 X/Q and W_v - Dispersion Parameters for Dose Rate, Table D 3-23

The dispersion parameters for the whole body and skin dose rate calculation correspond to the highest annual average dispersion parameters at or beyond the unrestricted area boundary. This is at the east site boundary. These values were obtained from the Nine Mile Point Unit 2 Final Environmental Statement, NUREG 1085 Table D-2 for the vent and stack. These were calculated using the methodology of Regulatory Guide 1.111, Rev. 1. The stack was modeled as an elevated release point because its height is more than 2.5 times any adjacent building height. The vent was modeled as a ground level release because even though it is higher than any adjacent building it is not more than 2.5 times the height.

The NRC Final Environmental Statement values for the site boundary X/Q and D/Q terms were selected for use in calculating Effluent Monitor Alarm Points and compliance with Site Boundary Dose Rate specifications because they are conservative

when compared with the corresponding Nine Mile Point Environmental Report values. In addition, the stack "intermittent release" X/Q was selected in lieu of the "continuous" value, since it is slightly larger, and also would allow not making a distinction between long term and short term releases.

The dispersion parameters for the organ dose calculations were obtained from the Environmental Report, Figures 7B-4 (stack) and 7B-8 (vent) by locating values corresponding to currently existing (1985) pathways. It should be noted that the most conservative pathways do not all exist at the same location. It is conservative to assume that a single individual would actually be at each of the receptor locations.

2.2.2 Whole Body Dose Rate Due to Noble Gases

The ground level gamma radiation dose from a noble gas stack release (elevated), referred to as plume shine, is calculated using the dose factors from Appendix B of this document. The ground level gamma radiation dose from a noble gas vent release accounts for the exposure from immersion in the semi-infinite cloud. The dispersion of the cloud from the point of release to the receptor at the east site boundary is factored into the plume shine dose factors for stack releases and through the use of X/Q in the equation for the immersion ground level dose rates for vent releases. The release rate is averaged over the period of concern. The factors are discussed in Appendix B.

Whole body dose rate $(DR)_\gamma$ due to noble gases:

$$(DR)_\gamma = 3.17E\text{-}08 \sum_i [V_i Q_{is} + K_i \ (X/Q)_v Q_{iv}]$$

Where:

DR_γ	=	Whole body dose rate (mrem/sec)
V_i	=	The constant accounting for the gamma whole body dose rate from the finite plume from the elevated stack releases for each identified noble gas nuclide, i. Listed on Table D 3-2, mrem/yr per μCi/sec
K_i	=	The constant accounting for the gamma whole body dose rate from immersion in the semi-infinite cloud for each identified noble gas nuclide, i. Listed in Table D 3-3, mrem/yr per uCi/m^3 (From Reg. Guide 1.109)
X/Q_v X/Q_s	=	The relative plume concentration at or beyond the land sector site boundary. Average meteorological data is used. Elevated X/Q values are used for the stack releases (s=stack); ground X/Q values are used for the vent releases (v=vent). Listed on Table D 3-23
Q_{is}, Q_{iv}	=	The release rate of each noble gas nuclide i, from the stack (s) or vent (v). Averaged over the time period of concern. (μCi/sec)

3.17E-08 = Conversion Factor; the inverse of the number of seconds in one year. (yr/sec)

2.2.3 Skin Dose Rate Due to Noble Gases

There are two types of radiation from noble gas releases that contribute to the skin dose rate: beta and gamma.

For stack releases this calculation takes into account the dose from beta radiation in a semi infinite cloud by using an immersion dose factor. Additionally, the dispersion of the released activity from the stack to the receptor is taken into account by use of the factor (X/Q). The gamma radiation dose from the elevated stack release is taken into account by the dose factors in Appendix B.

For vent releases the calculations also take into account the dose from the beta (β) and gamma (γ) radiation of the semi infinite cloud by using an immersion dose factor. Dispersion is taken into account by use of the factor (X/Q).

The release rate is averaged over the period of concern.

Skin dose rate $(DR)_{\gamma+\beta}$ due to noble gases:

$$(DR)_{\gamma+\beta} = 3.17E\text{-}8 \sum_i [(L_i (X/Q)_s + 1.11 B_i) Q_{is} + (L_i + 1.11 M_i)(X/Q)_v Q_{iv}]$$

Where:

$(DR)_{\gamma+\beta}$ = Skin dose rate (mrem/sec)

L_i = The constant to account for the gamma and beta skin dose rates for each noble gas nuclide, i, from immersion in the semi-infinite cloud, mrem/yr per $\mu Ci/m^3$, listed on Table D 3-3 (from R.G. 1.109)

M_i = The constant to account for the air gamma dose rate for each noble gas nuclide, i, from immersion in the semi-infinite cloud, mrad/yr per $\mu Ci/m^3$, listed on Table D 3-3 (from R.G. 1.109)

1.11 = Unit conversion constant, mrem/mrad

.7 = Structural shielding factor, unitless

B_i = The constant accounting for the air gamma dose rate from exposure to the overhead plume of elevated releases of each identified noble gas nuclide, i. Listed on Table D 3-2, mrad/yr per $\mu Ci/sec$.

$(X/Q)_s$ $(X/Q)_v$	=	The relative plume concentration at or beyond the land sector site boundary. Average meteorological data is used. Elevated X/Q values are used for the stack releases (s=stack); ground X/Q values are used for the vent releases (v=vent).
3.17E-8	=	Conversion Factor; the inverse of the number of seconds in a year; (yr/sec)
Q_{iv}, Q_{is}	=	The release rate of each noble gas nuclide i, from the stack(s) or vent (v) averaged over the time period of concern, μCi/sec.

2.2.4 Organ Dose Rate Due to I-131, I-133, Tritium, and Particulates with Half-lives greater than 8 days.

The organ dose rate is calculated using the dose factors (R_i) from Appendix C. The factor R_i takes into account the dose rate received from the ground plane, inhalation and ingestion pathways. W_s and W_v take into account the atmospheric dispersion from the release point to the location of the most conservative receptor for each of the respective pathways. The release rate is averaged over the period of concern.

Organ dose rates $(DR)_{at}$ due to iodine-131, iodine-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days:

$$(DR)_{at} = 3.17E\text{-}8 \sum_j [\sum_i R_{ijat} [W_s Q_{is} + W_v Q_{iv}]]$$

Where:

$(DR)_{at}$	=	Organ dose rate (mrem/sec)
R_{ijat}	=	The factor that takes into account the dose from nuclide i through pathway j to an age group a, and individual organ t. Units for inhalation pathway, mrem/yr per μCi/m^3. Units for ground and ingestion pathways, m^2-mrem/yr per uCi/sec. See Tables D 3-4 through D 3-22).
W_s, W_v	=	Dispersion parameter either X/Q (sec/m^3) or D/Q (1/m^2) depending on pathway and receptor location. Average meteorological data is used (Table D 3-23). Elevated W_s values are used for stack releases (s=stack); ground W_v values are used for vent releases (v=vent).
Q_{is}, Q_{iv}	=	The release rates for nuclide i, from the stack (s) and vent (v) respectively, μCi/sec.

When the release rate exceeds 0.75 uCi/sec from the stack or vent, the dose rate assessment shall, also, include JAF and NMP1 dose contributions. The use of the 0.75 μCi/sec release rate threshold is conservative because it is based on the dose conversion

factor (R_i) for the Sr-90 child bone which is significantly higher than the dose factors for the other isotopes present in the stack or vent release.

2.3 Gaseous Effluent Dose Calculation Methodology

Doses will be calculated monthly at a minimum to demonstrate that doses resulting from the release of noble gases, tritium, iodines, and particulates with half lives greater than 8 days are within the limits specified in 10 CFR 50. These limits are as follows:

The air dose from noble gases released in gaseous effluents, from each unit, to areas at or beyond the SITE BOUNDARY (see Figure D 1.0-1) shall be limited to the following.

a. During any calendar quarter: Less than or equal to 5 mrad 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 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radioactive material in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at or beyond the SITE BOUNDARY (see Figure D 1.0-1) 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.

The VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE and appropriate portions of this system shall be used to reduce releases of radioactivity when the projected doses in 31 days from iodine and particulate releases, from each unit, to areas at or beyond the SITE BOUNDARY (see Figure D 1.0-1) would exceed 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

2.3.1 W_v and W_s - Dispersion Parameters for Dose, Table D 3-23

The dispersion parameters for dose calculations were obtained chiefly from the Nine Mile Point Unit 2 Environmental Report Appendix 7B. These were calculated using the methodology of Regulatory Guide 1.111 and NUREG 0324. The stack was modeled as an elevated release point because height is more than 2.5 times the height of any adjacent building. The vent was modeled as a combined elevated/ground level release because the vent's height is not more than 2.5 times the height of any adjacent building. Average meteorology over the appropriate time period was used. Dispersion parameters not available from the ER were obtained from C.T. Main Data report dated November, 1985, or the FES.

2.3.2 Gamma Air Dose Due to Noble Gases

Gamma air dose from the stack or vent noble gas releases is calculated monthly. The gamma air dose equation is similar to the gamma dose rate equation except the receptor is air instead of the whole body or skin of whole body. Therefore, the stack noble gas releases use the finite plume air dose factors, and the vent noble gas releases use semi-infinite cloud immersion dose factors. The factor X/Q takes into account the dispersion of vent releases to the most conservative location. The release activity is totaled over the period of concern. The finite plume factor is discussed in Appendix B.

Gamma air dose due to noble gases:

D_γ = $3.17E-8 \; \sum_i [M_i (X/Q)_v \, Q_{iv} + B_i \, Q_{is}] \quad x \quad t$

D_γ = The gamma air dose for the period of concern, mrad

t = The duration of the dose period of concern, sec

Where all other parameters have been previously defined.

2.3.3 Beta Air Dose Due to Noble Gases

The beta air dose from the stack or vent noble gas releases is calculated using the semi-infinite cloud immersion dose factor in beta radiation. The factor X/Q takes into account the dispersion of releases to the most conservative location.

Beta air dose due to noble gases:

D_β = $3.17E-8 \; \sum_i N_i [(X/Q)_v \, Q_{iv} + (X/Q)_s \, Q_{is}] \quad x \quad t$

D_β = Beta air dose (mrad) for the period of concern

N_i = The constant accounting for the beta air dose from immersion in the semi-infinite cloud for each identified noble gas nuclide, i. Listed on Table D 3-3, mrad/yr per uCi/m^3. (From Reg. Guide 1.109).

t = The duration of the dose period of concern, sec

Where all other parameters have been previously defined.

2.3.4 Organ Dose Due to I-131, I-133, Tritium and Particulates with half-lives greater than 8 days.

The organ dose is based on the same equation as the dose rate equation except the dose is compared to the 10CFR50 dose limits. The factor R_i takes into account the dose received from the ground plane, inhalation, food (cow milk, cow meat and vegetation) pathways. W_s and W_v take into account the atmospheric dispersion from the release point to the location of the most conservative receptor for each of the respective pathways. The release is totaled over the period of concern. The R_i factors are discussed in Appendix C.

Organ dose D_{at} due to iodine-131, iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days.

$$D_{at} = 3.17E\text{-}8 \quad \sum_j \left[\sum_i R_{ijat} \left[W_s\, Q_{is} + W_v\, Q_{iv}\right]\right] \quad x \quad t$$

Where:

D_{at} = Dose to the critical organ t, for age group a, mrem

t = The duration of the dose period of concern, sec

Where all other parameters have been previously defined in Section 2.2.4.

2.4 I-133 and I-135 Estimation

Stack and vent effluent iodine cartridges are analyzed to a sensitivity of at least 1E-12 uCi/cc. If detected in excess of the LLD, the I-131 and I-133 analysis results will be reported directly from each cartridge analyzed. Periodically, (usually quarterly but on a monthly frequency if effluent iodines are routinely detected) a short-duration (12 to 24 hour) effluent sample is collected and analyzed to establish an I-135/I-131 ratio and an I-133/I-131 ratio, if each activity exceeds LLD. The short-duration ratio is used to confirm the routinely measured I-133 values. The short-duration I-135/I-131 ratio (if determined) is used with the I-131 release to estimate the I-135 release. The short-duration I-133/I-131 ratio may be used with the I-131 release to estimate the I-133 release if the directly measured I-133 release appears non-conservative.

2.5 Isokinetic Sampling

Sampling systems for the stack and vent effluent releases are designed to maintain isokinetic sample flow at normal ventilation flow rates. During periods of reduced ventilation flow, sample flow may be maintained at a minimum flow rate (above the calculated isokinetic rate) in order to minimize sample line losses due to particulate deposition at low velocity.

2.6 Use of Concurrent Meteorological Data vs. Historical Data

It is the intent to use dispersion parameters based on historical meteorological data to set alarm points and to determine or predict dose and dose rates in the environment due to gaseous effluents. If effluent levels approach limiting values, meteorological conditions concurrent with the time of release may be used to determine gaseous pathway doses.

2.7 Gaseous Radwaste Treatment System Operation

Part I, Section D 3.2.4 requires the GASEOUS RADWASTE TREATMENT SYSTEM to be in operation whenever the main condenser air ejector system is in operation. The system may be operated for short periods with the charcoal beds bypassed to facilitate

transients. The components of the system which normally should operate to treat offgas are the Preheater, Recombiner, Condenser, Dryer, Charcoal Adsorbers, HEPA Filter, and Vacuum Pump. (See Appendix D, Offgas System).

2.8 Ventilation Exhaust Treatment System Operation

Part I, Section D 3.2.5 requires the VENTILATION EXHAUST TREATMENT SYSTEM to be OPERABLE when projected doses in 31 days due to iodine and particulate releases would exceed 0.3 mrem to any organ of a member of the public. The appropriate components, which affect iodine or particulate release, to be OPERABLE are:

1) HEPA Filter - Radwaste Decon Area
2) HEPA Filter - Radwaste Equipment Area
3) HEPA Filter - Radwaste General Area

Whenever one of these filters is not OPERABLE, iodine and particulate dose projections will be made for 31-day intervals starting with filter inoperability, and continuing as long as the filter remains inoperable, in accordance with DSR 3.2.5.1. Predicted release rates will be used, along with the methodology of Section 2.3.4. (See Appendix D, Gaseous Radiation Monitoring.)

3.0 URANIUM FUEL CYCLE

The "Uranium Fuel Cycle" is defined in 40 CFR Part 190.02 (b) as follows:

"Uranium fuel cycle means the operations of milling of uranium ore chemical conversion of uranium, isotopic enrichment of uranium, fabrication of uranium fuel, generation of electricity by a light-water-cooled nuclear power plant using uranium fuel, and reprocessing of spent uranium fuel, to the extent that these directly support the production of electrical power for public use utilizing nuclear energy, but excludes mining operations, operations at waste disposal sites, transportation of any radioactive material in support of these operations, and the reuse of recovered non-uranium special nuclear and by-product materials from the cycle."

Sections D 3.1.2, D 3.2.2, and D 3.2.3 of Part I requires that when the calculated doses associated with the effluent releases exceed twice the applicable quarter or annual limits, the licensee shall evaluate the calendar year doses and, if required, submit a Special Report to the NRC and limit subsequent releases such that the dose commitment to a real individual from all uranium fuel cycle sources is limited to 25 mrem to the total body or any organ (except the thyroid, which is limited to 75 mrem). This report is to demonstrate that radiation exposures to all real individuals from all uranium fuel cycle sources (including all liquid and gaseous effluent pathways and direct radiation) are less than the limits in 40 CFR Part 190. If releases that result in doses exceeding the 40 CFR 190 limits have occurred, then a variance from the NRC to permit such releases will be requested and if possible, action will be taken to reduce subsequent releases.

The report to the NRC shall contain:

1) Identification of all uranium fuel cycle facilities or operations within 5 miles of the nuclear power reactor units at the site, that contribute to the annual dose of the maximum exposed member of the public.

2) Identification of the maximum exposed member of the public and a determination of the total annual dose to this person from all existing pathways and sources of radioactive effluents and direct radiation.

The total body and organ doses resulting from radioactive material in liquid effluents from Nine Mile Point Unit 2 will be summed with the doses resulting from the releases of noble gases, radioiodines, and particulates. The direct dose components will also be determined by either calculation or actual measurement. Actual measurements will utilize environmental TLD dosimetry. Calculated measurements will utilize engineering calculations to determine a projected direct dose component. In the event calculations are used, the methodology will be detailed as required by Technical Specification 5.6.3. The doses from Nine Mile Point Unit 2 will be added to the doses to the maximum exposed individual that are contributed from other uranium fuel cycle operations within 5 miles of the site.

For the purpose of calculating doses, the results of the Environmental Monitoring Program may be included to provide more refined estimates of doses to a real maximum exposed individual. Estimated doses, as calculated from station effluents, may be replaced by doses calculated from actual environmental sample results.

3.1 Evaluation of Doses From Liquid Effluents

For the evaluation of doses to real members of the public from liquid effluents, the fish consumption and shoreline sediment ground dose will be considered. Since the doses from other aquatic pathways are insignificant, fish consumption and shoreline sediment are the only two pathways that will be considered. The dose associated with fish consumption may be calculated using effluent data and Regulatory Guide 1.109 methodology or by calculating a dose to man based on actual fish sample analysis data. Because of the nature of the receptor location and the extensive fishing in the area, the critical individual may be a teenager or an adult. The dose associated with shoreline sediment is based on the assumption that the shoreline would be utilized as a recreational area. This dose may be derived from liquid effluent data and Regulatory Guide 1.109 methodology or from actual shoreline sediment sample analysis data.

Equations used to evaluate fish and shoreline sediment samples are based on Regulatory Guide 1.109 methodology. Because of the sample medium type and the half-lives of the radionuclides historically observed, the decay corrected portions of the equations are deleted. This does not reduce the conservatism of the calculated doses but increases the simplicity from an evaluation point of view. Table D 3-24 presents the parameters used for calculating doses from liquid effluents.

The dose from fish sample media is calculated as:

$$R_{apj} = \sum_i [C_{if} \, (U)(D_{aipj}) \, f] \, (1E+3)$$

Where:

R_{apj}	=	The total annual dose to organ j, of an individual of age group a, from nuclide i, via fish pathway p, in mrem per year; ex. if calculating to the adult whole body, then $R_{apj} = R_{wb}$ and $D_{aipj} = D_{iWB}$
C_{if}	=	The concentration of radionuclide i in fish samples in pCi/gram
U	=	The consumption rate of fish
1E+3	=	Grams per kilogram
(D_{aipj})	=	The ingestion dose factor for age group a, nuclide i, fish pathway p, and organ j, (Reg. Guide 1.109, Table E-11) (mrem/pCi). ex. when calculating to the adult whole body $D_{aipj} = D_{iWB}$
f	=	The fractional portion of the year over which the dose is applicable

The dose from shoreline sediment sample media is calculated as:

$$R_{apj} = \sum_i [C_{is} (U)(4E+4)(0.3)(D_{aipj}) f]$$

Where:

R_{apj} = The total annual dose to organ j, of an individual of age group a, from nuclide i, via the sediment pathway p, in mrem per year; ex. if calculating to the adult whole body, then $R_{apj} = R_{WB}$ and $D_{aipj} = D_{iWB}$

C_{is} = The concentration of radionuclide i in shoreline sediment in pCi/gram

U = The usage factor, (hr/yr) (Reg. Guide 1.109)

4E+4 = The product of the assumed density of shoreline sediment (40 kilogram per square meter to a depth of 2.5 cm) times the number of grams per kilogram

0.3 = The shore width factor for a lake

D_{aipj} = The dose factor for age group a, nuclide i, sediment pathway s, and organ j. (Reg. Guide 1.109, Table E-6) (mrem/hr per pCi/m^2); ex. when calculating to the adult whole body $D_{aipj} = D_{iWB}$

f = The fractional portion of the year over which the dose is applicable

NOTE: Because of the nature of the receptor location and the extensive fishing in the area, the critical individual may be a teenager or an adult.

3.2 Evaluation of Doses From Gaseous Effluents

For the evaluation of doses to real members of the public from gaseous effluents, the pathways contained in section 2 of the calculational methodologies section will be considered and include ground deposition, inhalation, cows milk, goats milk, meat, and food products (vegetation). However, any updated field data may be utilized that concerns locations of real individuals, real time meteorological data, location of critical receptors, etc. Data from the most recent census and sample location surveys should be utilized. Doses may also be calculated from actual environmental sample media, as available. Environmental sample media data such as TLD, air sample, milk sample and vegetable (food crop) sample data may be utilized in lieu of effluent calculational data.

Doses to members of the public from the pathways considered in section 2 as a result of gaseous effluents will be calculated using the methodology of Regulatory Guide 1.109 or the methodology of the ODCM, as applicable. Doses calculated from environmental sample media will be based on methodologies found in Regulatory Guide 1.109.

3.3 Evaluation of Doses From Direct Radiation

The dose contribution as a result of direct radiation shall be considered when evaluating whether the dose limitations of 40 CFR 190 have been exceeded. Direct radiation doses as a result of the reactor, turbine and radwaste buildings and outside radioactive storage tanks (as applicable) may be evaluated by engineering calculations or by evaluating environmental TLD results at critical receptor locations, site boundary or other special interest locations. For the evaluation of direct radiation doses utilizing environmental TLDs, the critical receptor in question, such as the critical residence, etc., will be compared to the control locations.

The comparison involves the difference in environmental TLD results between the receptor location and the average control location result.

3.4 Doses to Members of the Public Within the Site Boundary

The Radioactive Effluent Release 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 as defined by Figure D 1.0-1. A member of the public, would be represented by an individual who visits the sites' Energy Center for the purpose of observing the educational displays or for picnicking and associated activities.

Fishing is a major recreational activity in the area and on the Site as a result of the salmon and trout populations in Lake Ontario. Fishermen have been observed fishing at the shoreline near the Energy Center from April through December in all weather conditions. Thus, fishing is the major activity performed by members of the public within the site boundary. Based on the nature of the fishermen and undocumented observations, it is conservatively assumed that the maximum exposed individual spends an average of 8 hours per week fishing from the shoreline at a location between the

Energy Center and the Unit 1 facility. This estimate is considered conservative but not necessarily excessive and accounts for occasions where individuals may fish more on weekends or on a few days in March of the year.

The pathways considered for the evaluation include the inhalation pathway with the resultant lung dose, the ground dose pathway with the resultant whole body and skin dose and the direct radiation dose pathway with the associated total body dose. The direct radiation dose pathway, in actuality, includes several pathways. These include: the direct radiation gamma dose to an individual from an overhead plume, a gamma submersion plume dose, possible direct radiation dose from the facility and a ground plane dose (deposition). Because the location is in close proximity to the site, any beta plume submersion dose is felt to be insignificant.

Other pathways, such as the ingestion pathway, are not applicable. In addition, pathways associated with water related recreational activities, other than fishing, are not applicable here. These include swimming, boating and wading which are prohibited at the facility.

The inhalation pathway is evaluated by identifying the applicable radionuclides (radioiodine, tritium and particulates) in the effluent for the appropriate time period. The radionuclide concentrations are then multiplied by the appropriate X/Q value, inhalation dose factor, air intake rate, and the fractional portion of the year in question. Thus, the inhalation pathway is evaluated using the following equation adapted from Regulatory Guide 1.109. Table D 3-24 presents the reference for the parameters used in the following equation.

NOTE: The following equation is adapted from equations C-3 and C-4 of Regulatory Guide 1.109. Since many of the factors are in units of pCi/m^3, $m^3/sec.$, etc., and since the radionuclide decay expressions have been deleted because of the short distance to the receptor location, the equation presented here is not identical to the Regulatory Guide equations.

$$D_{ja} = \sum_i [(C_i) F (X/Q) (DFA)_{ija} (BR)_a t]$$

Where:

D_{ja} = The maximum dose from all nuclides to the organ j and age group (a) in mrem/yr; ex. if calculating to the adult lung, then $D_{ja} = D_L$ and $DFA_{ija} = DFA_{iL}$

C_i = The average concentration in the stack or vent release of nuclide i for the period in pCi/m^3.

F = Unit 2 average stack or vent flowrate in $m^3/sec.$

X/Q = The plume dispersion parameter for a location approximately 0.50 miles west of NMP-2 (The plume dispersion parameters are 9.6E-07 (stack) and 2.8E-06 (vent) and were obtained from the C.T. Main five year average annual X/Q tables. The vent X/Q (ground level) is ten times the listed 0.50 mile X/Q because the vent is approximately 0.3 miles from the receptor location. The stack (elevated) X/Q is conservative when based on 0.50 miles because of the close proximity of the stack and the receptor location.

$(DFA)_{ija}$ = the dose factor for nuclide i, organ j, and age group a in mrem per pCi (Reg. Guide 1.109, Table E-7); ex. if calculating to the adult lung the $DFA_{ija} = DFA_{iL}$

$(BR)_a$ = annual air intake for individuals in age group a in M^3 per year (obtained from Table E-5 of Regulatory Guide 1.109).

t = fractional portion of the year for which radionuclide i was detected and for which a dose is to be calculated (in years).

The ground dose pathway (deposition) will be evaluated by obtaining at least one soil or shoreline sediment sample in the area where fishing occurs. The dose will then be calculated using the sample results, the time period in question, and the methodology based on Regulatory Guide 1.109 as presented in Section 3.1. The resultant dose may be adjusted for a background dose by subtracting the applicable off-site control soil or shoreline sediment sample radionuclide activities. In the event it is noted that fishing is not performed from the shoreline but is instead performed in the water (i.e., the use of waders), then the ground dose pathway (deposition) will not be evaluated.

The direct radiation gamma dose pathway includes any gamma doses from an overhead plume, submersion in the plume, possible radiation from the facility and ground plane dose (deposition). This general pathway will be evaluated by average environmental TLD readings. At least two environmental TLDs will be used at one location in the approximate area where fishing occurs. The TLDs will be placed in the field on approximately the beginning of each calendar quarter and removed approximately at the end of each calendar quarter (quarter 2, 3, and 4).

The average TLD readings will be adjusted by the average control TLD readings. This is accomplished by subtracting the average quarterly control TLD value from the average fishing location TLD value. The applicable quarterly control TLD values will be used after adjusting for the appropriate time period (as applicable). In the event of loss or theft of the TLDs, results from a TLD or TLDs in a nearby area may be utilized.

4.0 ENVIRONMENTAL MONITORING PROGRAM

4.1 Sampling Stations

The current sampling locations are specified in Table D 5-1 and Figures D 5.1-1 and D 5.1-2. The meteorological tower location is shown on Figure D 5.1-1 and is located where TLD location #17 is identified. The Environmental Monitoring Program is a joint effort between the owners and operators of the Nine Mile Point Units 1 and 2 and the James A. FitzPatrick Nuclear Power Plants. Sampling locations are chosen on the basis of historical average dispersion or deposition parameters from both units. The environmental sampling location coordinates shown on Table D 5-1 are based on the NMP-2 reactor centerline.

The average dispersion and deposition parameters for the three units have been calculated for a 5 year period, 1978 through 1982. Average dispersion or deposition parameters for the site are calculated using the 1978 through 1982 data and are used to compare the results of the annual land use census. If it is determined that sample locations required by Control D 3.5.1 are unavailable or new locations are identified that yield a significantly higher (i.e., 50%) calculated D/Q value, actions will be taken as required by Controls D 3.5.1 and D 3.5.2 and the Radiological Environmental Monitoring Program updated accordingly.

4.2 Interlaboratory Comparison Program

Analyses shall be performed on samples containing known quantities of radioactive materials that are supplied as part of a Commission approved or sponsored Interlaboratory Comparison Program, such as the EPA Crosscheck Program. Participation shall be only for those media, e.g., air, milk, water, etc., that are included in the Nine Mile Point Environmental Monitoring Program and for which cross check samples are available. An attempt will be made to obtain a QC sample to program sample ratio of 5% or better. The Quality Control sample results shall be reported in the Annual Radiological Environmental Operating Report so that the Commission staff may evaluate the results.

Specific sample media for which EPA Cross Check Program samples are available include the following:

- gross beta in air particulate filters
- gamma emitters in air particulate filters
- gamma emitters in milk
- gamma emitters in water
- tritium in water
- I-131 in water

4.3 Capabilities for Thermoluminescent Dosimeters Used for Environmental Measurements

Required detection capabilities for thermoluminescent dosimeters used for environmental measurements required by the Technical Specifications are based on ANSI Standard N545, section 4.3. TLDs are defined as phosphors packaged for field use. In regard to the detection capabilities for thermoluminescent dosimeters, only one determination is required to evaluate the above capabilities per type of TLD. Furthermore, the above capabilities may be determined by the vendor who supplies the TLDs. Required detection capabilities are as follows.

4.3.1 Uniformity shall be determined by giving TLDs from the same batch an exposure equal to that resulting from an exposure rate of 10 uR/hr during the field cycle. The responses obtained shall have a relative standard deviation of less than 7.5%. A total of at least 5 TLDs shall be evaluated.

4.3.2 Reproducibility shall be determined by giving TLDs repeated exposures equal to that resulting from an exposure rate of 10 uR/hr during the field cycle. The average of the relative standard deviations of the responses shall be less than 3.0%. A total of at least 4 TLDs shall be evaluated.

4.3.3 Dependence of exposure interpretation on the length of a field cycle shall be examined by placing TLDs for a period equal to at least a field cycle and a period equal to half the same field cycle in an area where the exposure rate is known to be constant. This test shall be conducted under approximate average winter temperatures and approximate average summer temperatures. For these tests, the ratio of the response obtained in the field cycle to twice that obtained for half the field cycle shall not be less than 0.85. At least 6 TLDs shall be evaluated.

4.3.4 Energy dependence shall be evaluated by the response of TLDs to photons for several energies between approximately 30 keV and 3 MeV. The response shall not differ from that obtained with the calibration source by more than 25% for photons with energies greater than 80 keV and shall not be enhanced by more than a factor of two for photons with energies less than 80 keV. A total of at least 8 TLDs shall be evaluated.

4.3.5 The directional dependence of the TLD response shall be determined by comparing the response of the TLD exposed in the routine orientation with respect to the calibration source with the response obtained for different orientations. To accomplish this, the TLD shall be rotated through at least two perpendicular planes. The response averaged over all directions shall not differ from the response obtained in the standard calibration position by more than 10%. A total of at least 4 TLDs shall be evaluated.

4.3.6 Light dependence shall be determined by placing TLDs in the field for a period equal to the field cycle under the four conditions found in ANSI N545, section 4.3.6. The results obtained for the unwrapped TLDs shall not differ from those obtained for the TLDs wrapped in aluminum foil by more than 10%. A total of at least 4 TLDs shall be evaluated for each of the four conditions.

4.3.7 Moisture dependence shall be determined by placing TLDs (that is, the phosphors packaged for field use) for a period equal to the field cycle in an area where the exposure rate is known to be constant. The TLDs shall be exposed under two conditions: (1) packaged in a thin, sealed plastic bag, and (2) packaged in a thin, sealed plastic bag with sufficient water to yield observable moisture throughout the field cycle. The TLD or phosphor, as appropriate, shall be dried before readout. The response of the TLD exposed in the plastic bag containing water shall not differ from that exposed in the regular plastic bag by more than 10%. A total of at least 4 TLDs shall be evaluated for each condition.

4.3.8 Self irradiation shall be determined by placing TLDs for a period equal to the field cycle in an area where the exposure rate is less than 10 uR/hr and the exposure during the field cycle is known. If necessary, corrections shall be applied for the dependence of exposure interpretation on the length of the field cycle (ANSI N545, section 4.3.3). The average exposure inferred from the responses of the TLDs shall not differ from the known exposure by more than an exposure equal to that resulting from an exposure rate of 10 uR/hr during the field cycle. A total of at least 3 TLDs shall be evaluated.

TABLE D 2-1

LIQUID EFFLUENT DETECTORS RESPONSES*

NUCLIDE	(CPM/μCi/ml X 10^8)
Sr 89	0.78E-04
Sr 91	1.22
Sr 92	0.817
Y 91	2.47
Y 92	0.205
Zr 95	0.835
Nb 95	0.85
Mo 99	0.232
Tc 99m	0.232
Te 132	1.12
Ba 140	0.499
Ce 144	0.103
Br 84	1.12
I 131	1.01
I 132	2.63
I 133	0.967
I 134	2.32
I 135	1.17
Cs 134	1.97
Cs 136	2.89
Cs 137	0.732
Cs 138	1.45
Mn 54	0.842
Mn 56	1.2
Fe 59	0.863
Co 58	1.14
Co 60	1.65

* Values from SWEC purchase specification NMP2-P281F.

TABLE D 2-2
A_{iat} VALUES - LIQUID[1]
ADULT
$\frac{\text{mrem - ml}}{\text{hr - uCi}}$

NUCLIDE	T BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
H 3	3.67E-1	3.67E-1	--	3.67E-1	3.67E-1	3.67E-1	3.67E-1
Cr 51	1.26	3.13E2	1.18E-2	1.18E-2	2.86E-1	7.56E-1	1.66
Cu 64	1.28	2.33E2	--	2.73	6.89	--	--
Mn 54	8.38E2	1.34E4	3.98	4.38E3	1.31E3	3.98	3.98
Fe 55	1.07E2	2.62E2	6.62E2	4.57E2	--	--	2.55E2
Fe 59	9.28E2	8.06E3	1.03E3	2.42E3	7.53E-1	7.53E-1	6.76E2
Co 57	5.43E1	5.36E2	--	2.11E1	--	--	--
Co 58	2.01E2	1.81E3	1.07	9.04E1	1.07	1.07	1.07
Co 60	6.36E2	4.93E3	6.47E1	3.24E2	6.47E1	6.47E1	6.47E1
Zn 65	3.32E4	4.63E4	2.31E4	7.35E4	4.92E4	2.21	2.21
Sr 89	6.38E2	3.57E3	2.22E4	6.18E-5	6.18E-5	6.18E-5	6.18E-5
Sr 90	1.36E5	1.60E4	5.55E5	--	--	--	--
Sr 92	1.44E-2	6.61	3.34E-1	--	--	--	--
Zr 95	7.59E-1	2.83E2	9.77E-1	7.88E-1	8.39E-1	6.99E-1	6.99E-1
Mn 56	3.07E-2	5.52	--	1.73E-1	2.20E-1	--	--
Mo 99	1.60E1	1.95E2	1.97E-3	8.42E1	1.91E2	1.97E-3	1.97E-3
Na 24	1.34E2	1.34E2	1.34E2	1.34E2	1.34E2	1.34E2	1.34E2
I 131	1.16E2	5.36E1	1.42E2	2.03E2	3.48E2	6.65E4	2.77E-2
I 132	4.34E-3	2.33E-3	4.64E-3	1.24E-2	1.98E-2	4.34E-1	--
I 133	1.22E1	3.59E1	2.30E1	3.99E1	6.97E1	5.87E3	--
I 135	1.32E0	3.79E0	1.28E0	3.36E0	5.39E0	2.22E2	--
Ni 65	1.14E-2	6.35E-1	1.93E-1	2.50E-2	--	--	--
Cs 134	5.79E5	1.24E4	2.98E5	7.08E5	2.29E5	2.04E1	7.61E4
Cs 136	8.42E4	1.33E4	2.96E4	1.17E5	6.51E4	3.28E-1	8.92E3
Cs 137	3.42E5	1.01E4	3.82E5	5.22E5	1.77E5	3.10E1	5.89E4
Ba 140	1.37E1	4.30E2	2.09E2	3.04E-1	1.31E-1	4.17E-2	1.92E-1
Ce 141	3.79E-2	8.81E1	6.93E-2	5.83E-2	4.60E-2	3.53E-2	3.53E-2
Nb 95m	1.51E1	1.44E6	3.53E1	2.74E1	2.70E1	--	--
Nb 95	1.31E2	1.48E6	4.38E2	2.44E2	2.41E2	3.56E-1	3.56E-1
La 140	1.62E-2	3.72E3	1.03E-1	5.36E-2	2.83E-3	2.83E-3	2.83E-3
Ce 144	3.03E-1	6.15E2	2.02	9.66E-1	6.57E-1	2.06E-1	2.06E-1
Tc 99m	2.05E-2	9.54E-01	5.71E-4	1.61E-3	2.45E-2	--	7.90E-4
Np 239	1.8E-3	4.47E2	2.28E-2	2.78E-3	7.40E-3	5.95E-4	5.95E-4
Te 132	1.18E3	5.97E4	1.95E3	1.26E3	1.22E4	1.39E3	2.66E-3
Zr 97	5.08E-4	3.39E2	5.44E-3	1.10E-3	1.66E-3	7.11E-6	7.11E-6
W 187	4.31E1	4.04E4	1.48E2	1.23E2	4.43E-5	4.43E-5	4.43E-5
Ag 110m	1.09E1	3.94E2	1.14E1	1.13E1	1.22E1	1.04E1	1.04E1
Sb 124	4.72E1	3.36E2	1.07E3	4.33E1	4.31E1	4.31E1	5.12E1
Zn 69m	5.40E1	3.60E4	2.46E2	5.90E2	3.57E2	6.90E-2	6.90E-2
Au 199	3.95	7.33E2	1.26E-1	4.67	1.79E1	1.26E-1	1.26E-1
As 76	5.94	1.24E4	1.60E-1	6.19	1.16E1	1.60E-1	1.60E-1

1 Calculated in accordance with NUREG 0133, Section 4.3.1; and Regulatory Guide 1.109, Regulatory position C, Section 1.

TABLE D 2-3
A_{iat} VALUES - LIQUID[1]
TEEN
$\frac{\text{mrem - ml}}{\text{hr - uCi}}$

NUCLIDE	T BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
H 3	2.73E-1	2.73E-1	--	2.73E-1	2.73E-1	2.73E-1	2.73E-1
Cr 51	1.35	2.16E2	6.56E-2	6.56E-2	3.47E-1	7.79E-1	1.90
Cu 64	1.35	2.23E2	--	2.87	7.27	--	--
Mn 54	8.75E2	8.84E3	2.22E1	4.32E3	1.31E3	2.22E1	2.22E1
Fe 55	1.15E2	2.13E2	6.93E2	4.91E2	--	--	3.11E2
Fe 59	9.59E2	5.85E3	1.06E3	2.48E3	4.20	4.20	7.84E2
Co 57	1.44E2	4.08E2	--	2.19E1	--	--	--
Co 58	2.10E2	1.23E3	5.98	9.47E1	5.98	5.98	5.98
Co 60	9.44E2	3.73E3	3.61E2	6.20E2	3.61E2	3.61E2	3.61E2
Zn 65	3.40E4	3.08E4	2.10E4	7.28E4	4.66E4	1.24E1	1.24E1
Sr 89	6.92E2	2.88E3	2.42E4	3.45E-4	3.45E-4	3.45E-4	3.45E-4
Sr 90	1.14E5	1.30E4	4.62E5	--	--	--	--
Sr 92	1.54E-2	9.19E1	3.61E-1	--	--	--	--
Zr 95	3.96	2.10E2	4.19	3.99	4.03	3.90	3.90
Mn 56	3.22E-2	1.19E1	--	1.81E-1	2.29E-1	--	--
Mo 99	1.71E1	1.60E2	1.10E-2	8.95E1	2.05E2	1.10E-2	1.10E-2
Na 24	1.38E2	1.38E2	1.38E2	1.38E2	1.38E2	1.38E2	1.38E2
I 131	1.14E2	4.21E1	1.52E2	2.12E2	3.66E2	6.19E4	1.55E-1
I 132	4.56E-3	5.54E-3	4.86E-3	1.27E-2	2.00E-2	4.29E-1	--
I 133	1.28E1	3.17E1	2.47E1	4.19E1	7.35E1	5.85E3	1.02E-4
I 135	1.76E0	3.84E0	1.34E0	3.46E0	5.47E0	2.23E2	--
Ni 65	1.21E-2	1.44	2.08E-1	2.66E-2	--	--	--
Cs 134	3.33E5	9.05E3	3.05E5	7.18E5	2.28E5	1.14E2	8.72E4
Cs 136	7.87E4	9.44E3	2.98E4	1.17E5	6.38E4	1.83	1.01E4
Cs 137	1.90E5	7.91E3	4.09E5	5.44E5	1.85E5	1.73E2	7.21E4
Ba 140	1.44E1	3.40E2	2.21E2	5.03E-1	3.25E-1	2.33E-1	4.15E-1
Ce 141	2.00E-1	6.85E1	2.33E-1	2.21E-1	2.08E-1	1.97E-1	1.97E-1
Nb 95m	1.69E1	1.14E6	3.87E1	2.99E1	2.96E1	--	--
Nb 95	1.17E2	1.05E6	4.43E2	2.47E2	2.39E2	1.99	1.99
La 140	2.97E-2	3.01E3	1.22E-1	6.82E-2	1.58E-2	1.58E-2	1.58E-2
Ce 144	1.25	4.83E2	3.07	1.94	1.62	1.15	1.15
Tc 99m	2.11E-2	1.07	5.84E-4	1.63E-3	2.43E-2	--	9.04E-4
Np 239	4.63E-3	3.78E2	2.82E-2	5.67E-3	1.07E-2	3.32E-3	3.32E-3
Te 132	1.23E3	4.13E4	2.06E3	1.30E3	1.25E4	1.37E3	1.48E-2
Zr 97	5.68E-4	3.11E2	5.84E-3	1.19E-3	1.78E-3	3.97E-5	3.97E-5
W 187	4.55E1	3.52E4	1.59E2	1.30E2	2.47E-4	2.47E-4	2.47E-4
Ag 110m	5.85E1	3.17E2	5.89E1	5.88E1	5.97E1	5.79E1	5.79E1
Sb 124	2.45E2	4.53E2	2.51E2	2.41E2	2.41E2	2.41E2	2.50E2
Zn 69m	5.76E1	3.43E4	2.65E2	6.24E2	3.79E2	3.85E-1	3.85E-1
Au 199	4.85	5.78E2	7.04E-1	5.60	2.01E1	7.04E-1	7.04E-1
As 76	7.18	1.06E4	8.92E-1	7.40	1.33E1	8.92E-1	8.92E-1

[1]Calculated in accordance with NUREG 0133, Section 4.3.1; and Regulatory Guide 1.109, Regulatory position C, Section 1.

TABLE D 2-4
A_{iat} VALUES - LIQUID[1]
CHILD
$\frac{\text{mrem - ml}}{\text{hr - uCi}}$

NUCLIDE	T BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
H 3	3.34E-1	3.34E-1	--	3.34E-1	3.34E-1	3.34E-1	3.34E-1
Cr 51	1.39	7.29E1	1.37E-2	1.37E-2	2.22E-1	7.76E-1	1.41
Cu 64	1.60	1.25E2	--	2.65	6.41	--	--
Mn 54	9.02E2	2.83E3	4.65	3.37E3	9.49E2	4.65	4.65
Fe 55	1.50E2	8.99E1	9.15E2	4.85E2	--	--	2.74E2
Fe 59	1.04E3	2.18E3	1.29E3	2.09E3	8.78E-1	8.78E-1	6.08E2
Co 57	6.24E1	1.62E2	--	2.00E1	--	--	--
Co 58	2.21E2	4.20E2	1.25	7.30E1	1.25	1.25	1.25
Co 60	7.03E2	1.25E3	7.55E1	2.88E2	7.55E1	7.55E1	7.55E1
Zn 65	3.56E4	1.01E4	2.15E4	5.73E4	3.61E4	2.58	2.58
Sr 89	9.13E2	1.24E3	3.20E4	--	--	--	--
Sr 90	1.06E5	5.62E3	4.17E5	--	--	--	--
Sr 92	1.85E-2	8.73	4.61E-1	--	--	--	--
Zr 95	8.95E-1	9.36E1	1.22	9.04E-1	9.43E-1	8.15E-1	8.15E-1
Mn 56	3.73E-2	2.39E1	--	1.65E-1	2.00E-1	--	--
Mo 99	2.22E1	7.42E1	2.30E-3	8.98E1	1.92E2	2.30E-3	2.30E-3
Na 24	1.51E2	1.51E2	1.51E2	1.51E2	1.51E2	1.51E2	1.51E2
I 131	1.14E2	1.80E1	2.00E2	2.01E2	3.31E2	6.66E4	3.23E-2
I 132	5.08E-3	1.30E-2	6.01E-3	1.10E-2	1.69E-2	5.13E-1	--
I 133	1.51E1	1.60E1	3.22E1	3.98E1	6.64E1	7.40E3	--
I 135	1.53E0	2.30E0	1.68E0	3.02E0	4.63E0	2.67E2	--
Ni 65	1.46E-2	3.07	2.66E-1	2.51E-2	--	--	--
Cs 134	1.27E5	3.28E3	3.68E5	6.04E5	1.87E5	2.38E1	6.72E4
Cs 136	6.26E4	3.40E3	3.52E4	9.67E4	5.15E4	3.82E-1	7.68E3
Cs 137	7.28E4	3.12E3	5.15E5	4.93E5	1.61E5	3.62E1	5.78E4
Ba 140	1.87E1	1.62E2	3.19E2	3.28E-1	1.40E-1	4.87E-2	2.15E-1
Ce 141	4.61E-2	4.14E1	1.08E-1	7.43E-2	5.57E-2	4.12E-2	4.12E-2
Nb 95m	2.14E1	5.28E5	4.99E1	2.92E1	2.68E1	--	--
Nb 95	1.45E2	3.75E5	5.21E2	2.03E2	1.91E2	4.16E-1	4.16E-1
La 140	1.93E-2	1.33E3	1.39E-1	5.09E-2	3.30E-3	3.30E-3	3.30E-3
Ce 144	4.31E-1	2.92E2	3.81	1.36	8.61E-1	2.40E-1	2.40E-1
Tc 99m	2.29E-2	7.87E-1	7.05E-4	1.38E-3	2.01E-2	--	7.02E-4
Np 239	2.40E-3	1.79E2	3.44E-2	3.12E-3	7.70E-3	6.94E-4	6.94E-4
Te 132	1.38E3	1.15E4	2.57E3	1.14E3	1.06E4	1.66E3	3.10E-3
Zr 97	6.99E-4	1.77E2	8.11E-3	1.18E-3	1.69E-3	8.29E-6	8.29E-6
W 187	5.37E1	1.68E4	2.02E2	1.20E2	5.16E-5	5.16E-5	5.16E-5
Ag 110m	1.29E1	1.24E2	1.35E1	1.30E1	1.39E1	1.21E1	1.21E1
Sb 124	5.69E1	1.68E2	6.92E1	5.06E1	5.03E1	5.04E1	6.08E1
Zn 69m	6.80E1	1.87E4	3.37E2	5.75E2	3.34E2	8.05E-2	8.05E-2
Au 199	5.58	2.75E2	1.47E-1	5.02	1.80E1	1.47E-1	1.47E-1
As 76	8.31	5.47E3	1.86E-1	6.58	1.15E1	1.86E-1	1.86E-1

[1]Calculated in accordance with NUREG 0133, Section 4.3.1; and Regulatory Guide 1.109, Regulatory position C, Section 1.

TABLE D 2-5
A_{iat} VALUES - LIQUID[1]
INFANT
$\frac{\text{mrem - ml}}{\text{hr - uCi}}$

NUCLIDE	T BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG
H 3	1.87E-1	1.87E-1	--	1.87E-1	1.87E-1	1.87E-1	1.87E-1
Cr 51	8.21E-3	2.39E-1	--	--	1.17E-3	5.36E-3	1.04E-2
Cu 64	1.96E-2	8.70E-1	--	4.24E-2	7.17E-2	--	--
Mn 54	2.73	4.42	--	1.20E1	2.67	--	--
Fe 55	1.45	6.91E-1	8.42	5.44	--	--	2.66
Fe 59	1.25E1	1.52E1	1.82E1	3.18E1	--	--	9.41
Co 57	1.13E0	2.37E0	--	6.95E1	--	--	--
Co 58	5.36	5.36	--	2.15	--	--	--
Co 60	1.55E1	1.56E1	--	6.55	--	--	--
Zn 65	1.76E1	3.22E1	1.11E1	3.81E1	1.85E1	--	--
Sr 89	4.27E1	3.06E1	1.49E3	--	--	--	--
Sr 90	2.86E3	1.40E2	1.12E4	--	--	--	--
Sr 92	1.56E-5	4.54E-3	4.21E-4	--	--	--	--
Zr 95	2.12E-2	1.49E1	1.23E-1	2.99E-2	3.23E-2	--	--
Mn 56	1.81E-6	9.56E-4	--	1.05E-5	9.05E-6	--	--
Mo 99	2.65	4.48	--	1.36E1	2.03E1	--	--
Na 24	9.61E-1	9.61E-1	9.61E-1	9.61E-1	9.61E-1	9.61E-1	9.61E-1
I 131	9.78	7.94E-1	1.89E1	2.22E1	2.60E1	7.31E3	--
I 132	3.43E-6	7.80E-6	4.75E-6	9.63E-6	1.07E-5	4.52E-4	--
I 133	8.26E-1	4.77E-1	1.94	2.82	3.31	5.13E2	--
I 135	2.38E2	2.36E2	3.29E2	6.54E2	7.28E2	5.86E0	--
Ni 65	2.96E-6	4.96E-4	5.75E-5	6.51E-6	--	--	--
Cs 134	4.30E1	1.16	2.28E2	4.26E2	1.10E2	--	4.50E1
Cs 136	2.81E1	1.14	2.56E1	7.53E1	3.00E1	--	6.13
Cs 137	2.63E1	1.16	3.17E2	3.71E2	9.95E1	--	4.03E1
Ba 140	4.88	2.33E1	9.48E1	9.48E-2	2.25E-2	--	5.82E-2
Ce 141	3.31E-3	1.45E1	4.61E-2	2.81E-2	8.67E-3	--	--
Nb 95m	1.02E3	1.20E1	2.39E3	1.73E3	1.10E3	--	--
Nb 95	5.87E-3	8.57	2.47E-2	1.02E-2	7.28E-3	--	--
La 140	6.52E-4	2.98E1	6.43E-3	2.53E-3	--	--	--
Ce 144	1.01E-1	1.03E2	1.80	7.37E-1	2.98E-1	--	--
Tc 99m	3.17E-4	7.14E-3	1.19E-5	2.46E-5	2.64E-4	--	1.28E-5
Np 239	2.08E-4	1.06E1	4.12E-3	3.68E-4	7.34E-4	--	--
Te 132	4.08	1.62E1	8.83	4.37	2.74E1	6.46	--
Zr 97	1.38E-4	1.92E1	1.76E-3	3.02E-4	3.04E-4	--	--
W 187	4.13E-2	7.02	1.72E-1	1.19E-1	--	--	--
Ag 110m	2.91E-1	2.28E1	6.02E-1	4.39E-1	6.28E-1	--	--
Sb 124	3.95	3.93E1	1.27E1	1.87E-1	--	3.38E-2	7.98
Zn 69m	2.30E-2	3.50	1.24E-1	2.52E-1	1.02E-1	--	--
Au 199	2.23E-1	5.38	--	2.48E-1	6.26E-1	--	--
As 76	8.67E-2	2.85E1	--	8.46E-2	1.03E-1	--	--

[1]Calculated in accordance with NUREG 0133, Section 4.3.1; and Regulatory Guide 1.109, Regulatory position C, Section 1.

TABLE D 3-1
OFFGAS PRETREATMENT*
DETECTOR RESPONSE

NUCLIDE	NET CPM/μCi/cc
Kr 83m	--
Kr 85	4.28E+03
Kr 85m	3.85E+03
Kr 87	6.68E+03
Kr 88	3.97E+03
Kr 89	6.48E+03
Xe 131m	--
Xe 133	1.69E+03
Xe 133m	--
Xe 135	4.91E+03
Xe 135m	--
Xe 137	6.89E+03
Xe 138	5.51E+03

* Values from calculation H21C-070

TABLE D 3-2
PLUME SHINE PARAMETERS[1]

NUCLIDE	B_i mrad/yr / uCi/sec	V_i mrem/yr / uCi/sec
Kr 83m	9.01E-7	-------
Kr 85	6.92E-7	-------
Kr 85m	5.09E-4	4.91E-4
Kr 87	2.72E-3	2.57E-3
Kr 88	7.23E-3	7.04E-3
Kr 89	1.15E-2	1.13E-2
Kr 90	6.57E-3	4.49E-3
Xe 131m	7.76E-6	-------
Xe 133	7.46E-5	6.42E-5
Xe 133m	4.79E-5	3.95E-5
Xe 135	7.82E-4	7.44E-4
Xe 135m	1.45E-3	1.37E-3
Xe 137	6.25E-4	5.98E-4
Xe 138	4.46E-3	4.26E-3
Xe-127	1.96E-3	1.31E-3
Ar 41	5.00E-3	4.79E-3

1 B_i and V_i are calculated for critical site boundary location; 1.6km in the easterly direction. See Appendix B. Those values that show a dotted line were negligible because of high energy absorption coefficients.

TABLE D 3-3
IMMERSION DOSE FACTORS[1]

Nuclide	K_i (γ-Body)[2]	L_i (β-Skin)[2]	M_i (γ-Air)[3]	N_i (β-Air)[3]
Kr 83m	7.56E-02	---	1.93E1	2.88E2
Kr 85m	1.17E3	1.46E3	1.23E3	1.97E3
Kr 85	1.61E1	1.34E3	1.72E1	1.95E3
Kr 87	5.92E3	9.73E3	6.17E3	1.03E4
Kr 88	1.47E4	2.37E3	1.52E4	2.93E3
Kr 89	1.66E4	1.01E4	1.73E4	1.06E4
Kr 90	1.56E4	7.29E3	1.63E4	7.83E3
Xe 131m	9.15E1	4.76E2	1.56E2	1.11E3
Xe 133m	2.51E2	9.94E2	3.27E2	1.48E3
Xe 133	2.94E2	3.06E2	3.53E2	1.05E3
Xe 135m	3.12E3	7.11E2	3.36E3	7.39E2
Xe 135	1.81E3	1.86E3	1.92E3	2.46E3
Xe 137	1.42E3	1.22E4	1.51E3	1.27E4
Xe 138	8.83E3	4.13E3	9.21E3	4.75E3
Ar 41	8.84E3	2.69E3	9.30E3	3.28E3

[1]From, Table B-1.Regulatory Guide 1.109 Rev. 1
[2]mrem/yr per uCi/m^3.
[3]mrad/yr per uCi/m^3.

TABLE D 3-4
DOSE AND DOSE RATE
R_i VALUES - INHALATION - INFANT[1]

$$\frac{\text{mrem/yr}}{\text{uCi/m}^3}$$

NUCLIDE	BONE	LIVER	T. BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3*	--	6.47E2	6.47E2	6.47E2	6.47E2	6.47E2	6.47E2
C 14*	2.65E4	5.31E3	5.31E3	5.31E3	5.31E3	5.31E3	5.31E3
Cr 51	--	--	8.95E1	5.75E1	1.32E1	1.28E4	3.57E2
Mn 54	--	2.53E4	4.98E3	--	4.98E3	1.00E6	7.06E3
Fe 55	1.97E4	1.17E4	3.33E3	--	--	8.69E4	1.09E3
Fe 59	1.36E4	2.35E4	9.48E3	--	--	1.02E6	2.48E4
Co 58	--	1.22E3	1.82E3	--	--	7.77E5	1.11E4
Co 60	--	8.02E3	1.18E4	--	--	4.51E6	3.19E4
Zn 65	1.93E4	6.26E4	3.11E4	--	3.25E4	6.47E5	5.14E4
Sr 89	3.98E5	--	1.14E4	--	--	2.03E6	6.40E4
Sr 90	4.09E7	--	2.59E6	--	--	1.12E7	1.31E5
Zr 95	1.15E5	2.79E4	2.03E4	--	3.11E4	1.75E6	2.17E4
Nb 95	1.57E4	6.43E3	3.78E3	--	4.72E3	4.79E5	1.27E4
Mo 99	--	1.65E2	3.23E1	--	2.65E2	1.35E5	4.87E4
I-131	3.79E4	4.44E4	1.96E4	1.48E7	5.18E4	--	1.06E3
I 133	1.32E4	1.92E4	5.60E3	3.56E6	2.24E4	--	2.16E3
Cs 134	3.96E5	7.03E5	7.45E4	--	1.90E5	7.97E4	1.33E3
Cs 137	5.49E5	6.12E5	4.55E4	--	1.72E5	7.13E4	1.33E3
Ba 140	5.60E4	5.60E1	2.90E3	--	1.34E1	1.60E6	3.84E4
La 140	5.05E2	2.00E2	5.15E1	--	--	1.68E5	8.48E4
Ce 141	2.77E4	1.67E4	1.99E3	--	5.25E3	5.17E5	2.16E4
Ce 144	3.19E6	1.21E6	1.76E5	--	5.38E5	9.84E6	1.48E5
Nd 147	7.94E3	8.13E3	5.00E2	--	3.15E3	3.22E5	3.12E4
Ag 110m	9.99E3	7.22E3	5.00E3	--	1.09E4	3.67E6	3.30E4

* mrem/yr per μCi/m^3

[1]This and following R_i Tables Calculated in accordance with NUREG 0133, Section 5.3.1, except C 14 values in accordance with Regulatory Guide 1.109 Equation C-8.

TABLE D 3-5
DOSE AND DOSE RATE
R_i VALUES - INHALATION - CHILD
$\frac{\text{mrem/yr}}{\text{uCi/m}^3}$

NUCLIDE	BONE	LIVER	T. BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3*	--	1.12E3	1.12E3	1.12E3	1.12E3	1.12E3	1.12E3
C 14*	3.59E4	6.73E3	6.73E3	6.73E3	6.73E3	6.73E3	6.73E3
Cr 51	--	--	1.54E2	8.55E1	2.43E1	1.70E4	1.08E3
Mn 54	--	4.29E4	9.51E3	--	1.00E4	1.58E6	2.29E4
Fe 55	4.74E4	2.52E4	7.77E3	--	--	1.11E5	2.87E3
Fe 59	2.07E4	3.34E4	1.67E4	--	--	1.27E6	7.07E4
Co 58	--	1.77E3	3.16E3	--	--	1.11E6	3.44E4
Co 60	--	1.31E4	2.26E4	--	--	7.07E6	9.62E4
Zn 65	4.26E4	1.13E5	7.03E4	--	7.14E4	9.95E5	1.63E4
Sr 89	5.99E5	--	1.72E4	--	--	2.16E6	1.67E5
Sr 90	1.01E8	--	6.44E6	--	--	1.48E7	3.43E5
Zr 95	1.90E5	4.18E4	3.70E4	--	5.96E4	2.23E6	6.11E4
Nb 95	2.35E4	9.18E3	6.55E3	--	8.62E3	6.14E5	3.70E4
Mo 99	--	1.72E2	4.26E1	--	3.92E2	1.35E5	1.27E5
I 131	4.81E4	4.81E4	2.73E4	1.62E7	7.88E4	--	2.84E3
I 133	1.66E4	2.03E4	7.70E3	3.85E6	3.38E4	--	5.48E3
Cs 134	6.51E5	1.01E6	2.25E5	--	3.30E5	1.21E5	3.85E3
Cs 137	9.07E5	8.25E5	1.28E5	--	2.82E5	1.04E5	3.62E3
Ba 140	7.40E4	6.48E1	4.33E3	--	2.11E1	1.74E6	1.02E5
La 140	6.44E2	2.25E2	7.55E1	--	--	1.83E5	2.26E5
Ce 141	3.92E4	1.95E4	2.90E3	--	8.55E3	5.44E5	5.66E4
Ce 144	6.77E6	2.12E6	3.61E5	--	1.17E6	1.20E7	3.89E5
Nd 147	1.08E4	8.73E3	6.81E2	--	4.81E3	3.28E5	8.21E4
Ag 110m	1.69E4	1.14E4	9.14E3	--	2.12E4	5.48E6	1.00E5

* mrem/yr per μCi/m^3