NINE MILE POINT NUCLEAR STATION - UNIT 1

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

January – December 2002

Constellation
Energy Group

Nine Mile Point
Nuclear Station

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JANUARY - DECEMBER 2002

SUPPLEMENTAL INFORMATION

Facility: Nine Mile Point Unit 1 **Licensee: Nine Mile Point Nuclear Station, LLC**

1. TECHNICAL SPECIFICATION ADMINISTRATIVE CONTROLS - (Off-Site Dose Calculation Manual (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 total body and less than or equal to 3000 mrem/year to the skin.

      2. The air dose due to noble gases released in gaseous effluents from Nine Mile Point Unit 1 to areas beyond the site boundary shall be limited during any calendar quarter to less than or equal to 5 milliroentgen 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 milliroentgen 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 1 to areas 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 liquid effluents to unrestricted areas shall be limited to ten times the concentrations specified in 10 CFR Part 20.1001-20.2402, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 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 1 to unrestricted areas shall be limited during any calendar quarter to less than or equal to 1.5 mrem to the total body and to less than or equal to 5 mrem to any organ, and during any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

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 grab sample gamma spectroscopic analysis (intrinsic germanium crystal) or gross activity monitoring (calibrated against gamma isotopic analysis of a 4.0L Marinelli grab sample) of an isokinetic stack sample stream.

B) IODINES

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

C) PARTICULATES

Activity released from the main stack 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. Tritium effluent activity is measured during purge, monthly, or weekly when fuel is offloaded until stable tritium release rates are demonstrated.

E) EMERGENCY CONDENSER VENT EFFLUENTS

The effluent curie quantities are estimated based on the isotopic distribution in the Condensate Storage Tank water and the Emergency Condenser shell water. Actual isotopic concentrations are found via gamma spectroscopy. Initial release rates of Sr-89, Sr-90 and Fe-55 are estimated by applying scaling factors delineated in the Unit 1 Off-Site Dose Calculation Manual and actual release rates are determined from post offsite analysis results. The activity of fission and activation gases released due to tube leaks is based on reactor steam leak rates using offgas isotopic analyses.

F) 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. Tritium activity is estimated on the most recent analysis of the Condensate Storage Tank water. Initial release rates of Sr-89, Sr-90, and Fe-55 are estimated by applying scaling factors delineated in the Unit 1 Off-Site Dose Calculation Manual and actual release rates are determined from post offsite analysis results.

G) SOLID EFFLUENTS

Isotopic contents of waste shipments are determined by gamma spectroscopy analysis 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 X Unit 2 __ Reporting Period January – December 2002

Liquid Effluents:

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

Average MPC - µCi/ml (Qtr. 1) = 3.00E-03 Average MEC - µCi/ml (Qtr. 3) = 1.00E-02
Average MPC - µCi/ml (Qtr. 2) = 2.58E-03 Average MEC - µCi/ml (Qtr. 4) = N/A

Average Energy (Fission and Activation gases – Mev):

Qtr. 1 :	E(gamma)	= 2.47E-01	E(beta) =	3.17E-01
Qtr. 2 :	E(gamma)	= 1.18E+00	E(beta) =	6.11E-01
Qtr. 3 :	E(gamma)	= 1.18E+00	E(beta) =	6.11E-01
Qtr. 4 :	E(gamma)	= 4.54E-02	E(beta) =	1.35E-01

Liquid:

		1st	2nd	3rd	4th
Number of batch releases	:	0			
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			
Total volume of water used to dilute the liquid effluent during release period (L)	:	4.49E+10	9.45E+10	9.02E+10	N/A
Total volume of water used to dilute the liquid effluent during reporting period (L)	:	1.30E+11	1.28E+11	1.37E+11	1.33E+11

Gaseous – (There were no releases from the operation of the Emergency Condenser Vent):

Number of batch releases	:	0
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	:	2
Total time period for batch releases (hrs)	:	2.35E+01
Maximum time period for a batch release (hrs)	:	1.38E+01
Average time period for a batch release (hrs)	:	1.176E+01
Minimum time period for a batch release (hrs)	:	9.72E+00

[1]ODCM Radioactive Effluent Controls 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 II. 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 X Unit 2 __ Reporting Period January – December 2002

Abnormal Releases:

A. Liquids:

Number of releases 2

Total activity released 2.78E+01 Ci

B. Gaseous:

Number of releases 0

Total activity released N/A Ci

Unit 1 Abnormal Release

BACKGROUND

Cooling for several vital components is provided by the Reactor Building Closed Loop Cooling (RBCLC) System and the Turbine Building Closed Loop Cooling (TBCLC) System. Both systems have the residual heat removed via the Service Water System.

RELEASE 1

The analysis results for the monthly Service Water West Discharge sample collected in March 2002 showed positive detection of tritium (H-3). In addition to positive detection of tritium in the service water, elevated make-up flow rate to the TBCLC was observed. Action Request (ACR) 02-01355 was written to troubleshoot the suspected TBCLC heat exchanger (HTX-71-12R) for tube leakage. The heat exchanger was isolated on 6/7/02 to perform inspections and required maintenance. Three leaking tubes were found during the initial entry into the heat exchanger and further testing by Eddy Current identified 60 additional tubes requiring plugs due to excessive thinning of tube walls. This condition was entered into the corrective action program and Deviation Event Report (DER) 1-2002-1910 was initiated. Repairs were completed and the heat exchanger returned to service on 6/23/02. Analysis results of Service Water samples collected on 7/17/02 showed no positive detection of tritium, with all results being below the lowest level of detection (LLD). The extent of condition from DER 1-2002-1910 did full eddy current testing and found no additional leak in the closed loop cooling heat exchangers.

RELEASE 2

Monthly analyses test of service water discharge showed a positive detection for tritium for August and September. An analysis of vendor results for these months was reviewed showing the concentrations levels of tritium just exceeded the minimum detectable concentrations. The most likely source of tritium in the service water is from interface with closed loop cooling heat exchangers. However, the makeup flow rate to the closed loop cooling systems does not support a leak sufficient to achieve the levels of tritium reported in the service water. Other service water interfaces do not present a pathway where tritium may be introduced to the service water system. Based on this information, it would appear that the vendor data may be a false positive result for tritium analyses. To be conservative, the tritium activity will be treated as an effluent release The months following this release (through March 2003) do not show positive results for any activity in the service water system.

ANALYSIS

An offsite vendor analysis of the Service Water samples representing the period 03/01/02 through 09/30/02 yielded a calculated release of 27.771 Curies of activity. The nuclides identified are: H-3 (27.15 Ci) and Fe-55 (0.621 Ci). Analysis of both discharges from 03/01/02 to 09/30/02 resulted in a calculated whole body dose of 1.16E-04 mrem, and a calculated maximum organ dose (Bone) of 5.98E-04 mrem. This represents 3.86E-03 percent of the annual whole body dose limit (3 mrem) and 5.98E-03 percent of the annual organ dose limit (10 mrem). The activity and resulting dose contribution is reflected in Attachment 5.

Unit 1 X Unit 2 Reporting Period January – December 2002

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	2.11E-04	1.24E+00	1.66E+00	1.05E-04	5.00E+01
	2. Average release rate	µCi/sec	2.72E-05	1.58E-01	2.09E-01	1.32E-05	
B.	Iodines						
	1. Total Iodine-131	Ci	7.15E-04	5.31E-04	2.06E-04	1.67E-04	3.00E+01
	2. Average release rate for period	µCi/sec	9.19E-05	6.76E-05	2.60E-05	2.09E-05	
C.	Particulates						
	1. Particulates with half-lives >8 days	Ci	1.64E-03	7.86E-04	3.28E-04	8.84E-04	3.00E+01
	2. Average release rate for period	µCi/sec	2.10E-04	1.00E-04	4.12E-05	1.11E-04	2.50E+01
	3. Gross alpha radioactivity	Ci	3.53E-05	4.70E-05	8.76E-05	2.30E-05	
D.	Tritium						
	1. Total release	Ci	1.10E+01	6.60E+00	5.96E+00	7.79E+00	5.00E+01
	2. Average release rate for period	µCi/sec	1.41E+00	8.39E-01	7.50E-01	9.76E-01	
E.	Percent of ODCM Limits						
	Fission and Activation Gases						
	Percent of Quarterly Gamma Air Dose Limit (5 mR)	%	1.71E-06	2.78E-02	3.72E-02	1.56E-07	
	Percent of Quarterly Beta Air Dose Limit(10 mrad)	%	1.09E-06	4.48E-03	6.00E-03	2.33E-07	
	Percent of Annual Gamma Air Dose Limit to Date (10 mR)	%	8.56E-07	1.39E-02	3.25E-02	3.25E-02	
	Percent of Annual Beta Air Dose Limit to Date (20 mrad)	%	5.45E-07	2.24E-03	5.25E-03	5.25E-03	
	Percent of Whole Body Dose Rate Limit (500 mrem/yr)	%	4.55E-08	7.45E-04	9.91E-04	3.60E-09	
	Percent of Skin Dose Rate Limit (3000 mrem/yr)	%	2.01E-08	1.95E-04	2.59E-04	1.69E-09	
	Tritium, Iodines, and Particulates (with half-lives greater than 8 days)						
	Percent of Quarterly Dose Limit (7.5 mrem)	%	4.47E-01	3.18E-01	8.02E-02	7.83E-02	
	Percent of Annual Dose Limit (15 mrem)	%	2.25E-01	3.85E-01	2.85E-01	3.25E-01	
	Percent of Organ Dose Rate Limit (1500 mrem/yr)	%	9.07E-03	6.39E-03	1.59E-03	1.55E-03	

Unit 1 X Unit 2 __ Reporting Period January – December 2002

GASEOUS EFFLUENTS – ELEVATED RELEASE

Nuclides Released		CONTINUOUS MODE[2] 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	**	1.24E+00	1.66E+00	**
2. Iodines[1]					
Iodine-131	Ci	7.15E-04	5.31E-04	2.06E-04	1.67E-04
Iodine-133	Ci	4.40E-03	2.07E-03	5.59E-04	2.66E-04
Iodine-135	Ci	**	**	**	**
3. Particulates[1]					
Strontium-89	Ci	1.95E-04	1.23E-04	2.10E-04	1.27E-04
Strontium-90	Ci	**	6.99E-07	8.57E-07	**
Cesium-134	Ci	**	**	**	**
Cesium-137	Ci	2.04E-05	**	3.47E-06	2.74E-06
Cobalt-60	Ci	3.32E-04	2.36E-04	9.38E-05	3.42E-04
Cobalt-58	Ci	6.85E-06	2.21E-05	**	3.80E-05
Manganese-54	Ci	7.29E-05	1.78E-04	**	1.17E-04
Barium-Lanthanum-140	Ci	4.88E-05	8.76E-05	1.91E-05	1.06E-05
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	9.61E-04	1.39E-04	**	2.47E-04
Molybdenum-99	Ci	**	**	**	**
Neodymium-147	Ci	**	**	**	**
4. Tritium	Ci	8.20E+0 0	4.52E+00	4.11E+00	6.45E+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 ODCM, has been verified.

2 Contributions from purges are included.

Unit 1 X Unit 2 __ Reporting Period January – December 2002

GASEOUS EFFLUENTS – ELEVATED RELEASE

BATCH MODE

There were no batch releases during the reporting period.

Nuclides Released			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]					
	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				
	Neodymium-147	Ci				
4.	Tritium	Ci				

[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 ODCM, has been verified.

Unit 1 X Unit 2 __ Reporting Period January – December 2002

GASEOUS EFFLUENTS – GROUND LEVEL RELEASES

Ground level releases are determined in accordance with the Off-Site Dose Calculation Manual and Chemistry procedures.

			CONTINUOUS MODE			
			1st QUARTER	2nd QUARTER	3rd QUARTER	4th QUARTER
1.	Fission Gases[1]	Ci				
	Argon-41	Ci	**	**	**	**
	Krypton-85	Ci	**	**	**	**
	Krypton-85m	Ci	**	**	**	**
	Krypton-87	Ci	**	**	**	**
	Krypton-88	Ci	**	**	**	**
	Xenon-127	Ci	**	**	**	**
	Xenon-131m	Ci	**	**	**	**
	Xenon-133	Ci	**	**	**	1.05E-04
	Xenon-133m	Ci	**	**	**	**
	Xenon-135	Ci	2.11E-04	1.56E-05	3.64E-05	**
	Xenon-135m	Ci	**	**	**	**
	Xenon-137	Ci	**	**	**	**
	Xenon-138	Ci	**	**	**	**
2.	Iodines[1]					
	Iodine-131	Ci	**	**	**	**
	Iodine-133	Ci	**	**	**	**
	Iodine-135	Ci	**	**	**	**
3.	Particulates[1]					
	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	**	**	**	**
	Neodymium-147	Ci	**	**	**	**
4.	Tritium	Ci	2.78E+00	2.07E+00	1.85E+00	1.34E+00

[1] Concentrations less than the lower limit of detection of the counting system used are indicated with a double asterisk.

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GASEOUS EFFLUENTS – GROUND LEVEL RELEASES

Ground level releases are determined in accordance with the Off-Site Dose Calculation Manual and Chemistry procedures.

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]					
	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				
	Neodymium-147	Ci				
4.	Tritium	Ci				

[1] Concentrations less than the lower limit of detection of the counting system used are indicated with a double asterisk.

ATTACHMENT 5

Unit 1 X Unit 2 ___ Reporting Period January – December 2002

LIQUID EFFLUENTS – SUMMATION OF ALL RELEASES

			1st QUARTER	2nd QUARTER	3rd QUARTER	4th QUARTER	EST. TOTAL ERROR, %
A.	Fission & Activation Products						
	1. Total release (not including Tritium, gases, alpha)	Ci	**	6.21E-01	**	No Releases	5 00E+01
	2. Average diluted concentration during reporting period	µCi/ml	**	4.85E-09	**	No Releases	
B.	Tritium						
	1. Total release	Ci	2.86E+00	9.99E+00	1.43E+01	No Releases	5.00E+01
	2. Average diluted concentration during reporting period	µCi/ml	2.20E-08	7.80E-08	1.04E-07	No Releases	
C.	Dissolved and Entrained Gases						
	1. Total release	Ci	**	**	**	**	5.00E+01
	2. Average diluted concentration during reporting period	µCi/ml	**	**	**	**	
D.	Gross Alpha Radioactivity						
	1. Total release	Ci	**	**	**	No Releases	5.00E+01
E.	Volumes						
	1. Prior to dilution	Liters	N/A	N/A	N/A	No Releases	5.00E+01
	2. Volume of dilution water used during release period	Liters	4.49E+10	9.45E+10	9.02E+10	No Releases	5.00E+01
	3. Volume of dilution water available during reporting period:	Liters	1.30E+11	1.28E+11	1.37E+11	1.33E+11	5.00E+01
F.	Percent of ODCM Limits						
	Percent of Quarterly Whole Body Dose Limit (1.5 mrem)	%	8.86E-05	6.86E-03	8.01E-04	No Releases	
	Percent of Quarterly Organ Dose Limit (5 mrem)	%	8.86E-05	1.20E-02	2.40E-04	No Releases	
	Percent of Annual Whole Body Dose Limit to Date (3 mrem)	%	4.42E-05	3.49E-03	3.86E-03	3.86E-03	
	Percent of Annual Organ Dose Limit to Date (10 mrem)	%	4.42E-05	5.98E-03	5.98E-03	5.98E-03	
	Percent of 10CFR20 Concentration Limit	%	7.33E-04	3.21E-03	1.04E-03	No Releases	
	Percent of Dissolved or Entrained Noble Gas Limit (2.00E-04 µCi/ml)	%	**	**	**	No Releases	

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LIQUID EFFLUENTS RELEASED

Nuclides Released		CONTINUOUS MODE[1, 2] 1st QUARTER	2nd QUARTER	3rd QUARTER	4th QUARTER
Strontium-89	Ci	**	**	**	No Releases
Strontium-90	Ci	**	**	**	No Releases
Cesium-134	Ci	**	**	**	No Releases
Cesium-137	Ci	**	**	**	No Releases
Iodine-131	Ci	**	**	**	No Releases
Cobalt-58	Ci	**	**	**	No Releases
Cobalt-60	Ci	**	**	**	No Releases
Iron-59	Ci	**	**	**	No Releases
Zinc-65	Ci	**	**	**	No Releases
Manganese-54	Ci	**	**	**	No Releases
Chromium-51	Ci	**	**	**	No Releases
Zirconium-Niobium-95	Ci	**	**	**	No Releases
Molybdenum-99	Ci	**	**	**	No Releases
Technetium-99m	Ci	**	**	**	No Releases
Barium-Lanthanum-140	Ci	**	**	**	No Releases
Cerium-141	Ci	**	**	**	No Releases
Tungsten-187	Ci	**	**	**	No Releases
Iodine-133	Ci	**	**	**	No Releases
Iron-55	Ci	**	6.21E-01	**	No Releases
Neptunium-239	Ci	**	**	**	No Releases
Iodine-135	Ci	**	**	**	No Releases
Dissolved or Entrained Gases	Ci	**	**	**	No Releases
Tritium	Ci	2.86E+00	9.99E+00	1.43E+01	No Releases

1 No batch mode release occurred during the report period.

2 Concentrations less than lower limit of detection of the counting system used are indicated with a double asterisk.

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SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

A.1 TYPE	Volume (m^3) Class			Activity[1] (Ci) Class		
	A	B	C	A	B	C
1. Spent Resins	0	0	0	0	0	0
2. Dry Active Waste	0	5.55E+00	0	0	2.43E+01	0
3. Other: (to vendor for processing or consolidation)						
a. Dry Active Waste	1.45E+02	0	0	5.03E-01	0	0
b. Scraps and Debris	1.15E+02	0	0	1.05E+00	0	0
c. Spent Resins and Sludge	3.89E+01	5.55E+00	0	8.76E+01	7.23E+01	0
d. Other (Contaminated Oil)	1.63E+01	0	0	1.99E-04	0	0

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

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SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

A.1 TYPE	Container[1]	Package[2]	Solidification Agent
1. Spent Resins, Mechanical Filters (Dewatered)			
2. Dry Active Waste	Poly HIC w/steel shell	Type A	None
3. Other: (To Vendor for Processing or Consolidation)			
a. Dry Active Waste	Metal Box (sealand)	STP	None
b Scraps and Debris	Metal Tanks	STP	None
	Metal Box	STP	None
c. Spent Resins and Sludges	HIC	STP	None
	HIC	Type A	None
d. Other (Contaminated Oil)	Metal Box	STP	None

[1]HIC: High Integrity Container

[2]STP: Strong Type Package

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SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

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

1. Spent Resins, Mechanical Filters (Dewatered)

Nuclide	Percent

2. Dry Active Waste

Nuclide	Percent
(1) Fe-55	7.26E+01
(2) Co-60	1.59E+01
(3) Mn-54	4.65E+00
(4) Cs-137	4.64E+00
(5) Other	2.21E+00

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

a. Dry Active Waste

Nuclide	Percent
(1) Fe-55	7.17E+01
(2) Co-60	1.87E+01
(3) Cs-137	3.78E+00
(4) Mn-54	3.10E+00
(5) Ce-144	1.54E+00
(6) Ni-63	1.11E+00
(7) Other	7.00E-02

b. Scraps and Debris

Nuclide	Percent
(1) Co-60	6.30E+01
(2) Cs-137	1.31E+01
(3) Mn-54	1.15E+01
(4) Fe-55	5.11E+00
(5) C-14	2.95E+00
(6) Ce-144	1.60E+00
(7) Other	2.74E+00

c. Spent Resins and Sludges

Nuclide	Percent
(1) Fe-55	4.36E+01
(2) Co-60	3.44E+01
(3) Mn-54	1.50E+01
(4) Cs-137	5.48E+00
(5) Other	1.52E+00

d. Other (Contaminated Oil)

Nuclide	Percent
(1) Co-60	8.11E+01
(2) Cs-137	1.80E+01
(3) Mn-54	9 00E-01

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SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

A.3. SOLID WASTE DISPOSITION:

Number of Shipments	Mode of Transportation	Destination
8	Truck	Studsvik Processing Facility, LLC Erwin, TN
7	Truck	GTS Duratek Oak Ridge, TN
1	Truck	Barnwell Waste Management Facility Barnwell, SC

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

Number of Shipments	Mode of Transportation	Destination
0	NA	NA

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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-1 radwaste buried by vendor facilities during January – December 2002. These totals were reported separately from "10CFR61 Solid Waste Shipped for Burial" because waste classification and burial was performed by the vendors. The following data represents the actual shipments made from the off-site vendors of our radwaste (e.g., compacted and non-compacted trash, dry non-compressible waste, asbestos, scrap metal, and resins) that was processed and commingled prior to burial.

C.1. TYPE OF WASTE – Compacted and noncompacted trash, dry non-compressible waste, asbestos, scrap metal, and resins processed by vendor facilities prior to burial.

Burial Volume (m^3)	Activity (Ci)	Est. Total Error, %
1.48E+01	1.57E+02	5.00E+01

C.2. SOLID WASTE DISPOSITION

Number of Shipments	Mode of Transportation	Destination
24	Truck	Clive, UT
12	Truck	Barnwell, SC

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SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

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

There were no shipments of sewage sludge with detectable quantities of plant-related nuclides from NMP to the treatment facility during the reporting period.

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

The Unit 1 Off-Site Dose Calculation manual (ODCM) Revision 22 was implemented on November 7, 2002. As a result of implementing Technical Specification 176, the procedural details for Radiological Effluent Technical Specifications (RETS) are relocated to the ODCM Part 1. The calculation Methodology section (Part II) has been revised to reflect the changes as a result of Technical Specification Amendment 176. Because the implementation date for Revision 22 and Revision 23 of the ODCM is the same (see below), only a copy of the ODCM, Revision 23 is attached. Below is a summary of the changes approved by the Plant General Manager for Revision 22 of the ODCM.

Page Number	Old Section Number	New Section Number	Change	Reason for Change
ii	Table of Contents	Part I and Part II	Revised table of contents to include Radiological Effluent Controls. Renumbered accordingly.	Editorial.
ix	1.0	None	Added a discussion relating to the relocation of the procedural details of Radiological Effluent Technical Specifications to RETS.	Provide a description of the ODCM format with RETS relocation. This is an administrative change.
I through I 6.0-5	Technical Specification Pages 6, 7, 8, 282 to 293, 295 to 307, 314, 315, 319, 321, 323, 324, 325, 326, 327, 328, 329 to 332, 334, 336, 337, BASES 294, 306, 309, 310, 311, 312, 313, 316, 320, 322, 333, 335, 338, Admin. sections 363, 364, 365, 366, 368, 369.	Part I, Sections 1.0, 3.0/4.0, and 6.0	Relocated Radiological Effluent Controls pages containing the procedural details of RETS from Technical Specifications to Part I ODCM.	Implement Technical Specification Amendment 176. Administrative Change.
I 1.0-1 and I 1.0-2	Technical Specification page 6 and 7	1.0	Revised the Member(s) of the Public, Site Boundary, and Unrestricted Area Definitions to delete the reference to Niagara Mohawk Power Corporation and the New York State Power Authority.	To reflect the change of ownership of Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant. This is an editorial change.
I 3.0-I	N/A	3.0/4.0	Added a statement the ODCM Part I, Radiological Effluent Controls, is subject to Technical Specifications 3.0 and 4.0 requirements.	Implement Technical Specification Amendment 176. Administrative Change.
I B 3.1-11	Tech. Spec 338	B 3/4.6.22	Corrected the heading numbers from 3.6.20 and 4.6.20 to 3.6.22 and 4.6.22.	Editorial.
I 6.0-1	N/A	6.0	Added a description of the Technical Specification sections that the ODCM specifications are subject to.	Editorial.
Part II	N/A	Throughout	Renumbered pages with a II designator to differentiate from Part I pages.	Editorial.
II 2	2.1.1	1.1.1	Revised to reflect that the liquid concentrations are limited to ten times the "new" 10 CFR 20 Effluent Concentrations. Also changed references from Tech Specs to part I of the ODCM and changed MPC to MEC.	In accordance with Technical Specifications section 6.18 and the relocation of the procedural requirements of RETS to Part I of the ODCM. Administrative change.

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

Page Number	Old Section Number	New Section Number	Change	Reason for Change
II 2, II 3	2.1.2	1.1.2	Changed Maximum Permissible Concentration to Maximum Effluent Concentration to reflect that the "new" 10CFR20 is the regulation that applies.	In accordance with Technical Specifications 6.18. Administrative change.
II 3, II 4	2.1.3	1.1.3	See description of change and reason for change to section 1.1.2 above.	Administrative change.
II 4, II 5	2.1.4.2 2.1.4.1	1.1.4.2 1.1.4.1	See description of change and reason for change to section 1.1.1 above.	Administrative change.
II 5	2.1.4.3	1.1.4.3	Changed Technical Specifications to Part I and MPC to MEC.	Relocated specifications to Part I and incorporated the terminology associated with "new" 10 CFR 20. Editorial.
II 5	2.1.4.4	1.1.4.4	Changed Technical Specification reference to Part I and added a II designator for references to sections in the Calculational Methodology Part II.	Relocated specifications to Part I and incorporated the terminology associated with "new" 10 CFR 20. Editorial.
II 6	2.1.4.5	1.1.4.5	Changed reference from Technical Specifications to Part I.	Relocated procedural requirements of RETS to Part I of the ODCM. Editorial.
II 6, II 7	2.2	1.2	Revised to reflect that the liquid concentrations are limited to ten times the "new" 10 CFR 20 Effluent Concentrations. Also changed references from Tech Specs to part I of the ODCM and changed MPC to MEC.	In accordance with Technical Specifications section 6.18 and the relocation of the procedural requirements of RETS to Part I of the ODCM. Administrative change.
II 8, II 9	2.3.1	1.3.1	Changed the word Technical to Part I.	Relocated the requirements to Part I of the ODCM. Editorial.
II 10, II 11	3.1.1	2.1.1	Replaced 10CFR20 with Technical Specifications were the gaseous dose rate limits are found. Moved paragraphs for logical order.	Editorial changes.
II 12	3.1.3	2.1.3	Added "II" designator for reference. Replaced Technical Specifications with Part I.	Relocated requirements to Part I. Editorial.
II 13	3.1.5.1	2.1.5.1	Replaced Technical Specifications with Part I.	Relocated requirements to Part I. Editorial.
II 14	3.1.5.3	2.1.5.3	Replaced Technical Specifications with Part I.	Relocated requirements to Part I. Editorial.
II 14	3.1.5.4	2.1.5.4	Replaced Technical Specifications with Part I.	Relocated requirements to Part I. Editorial.
II 14	3.1.5.5	2.1.5.5	Added I-135 to the section heading.	Editorial change.

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

Page Number	Old Section Number	New Section Number	Change	Reason for Change
II 15	3.1.5.6	2.1.5.6	Replaced Technical Specifications with Part I.	Relocated requirements to Part I. Editorial.
II 15, II 16	3.2	2.2	Replaced Technical Specifications with Part I.	Relocated requirements to Part I. Editorial.
II 16, II 20	3.2.1.1 3.2.2.1	2.2.1.1 2.2.2.1	Changed Part II section reference.	Sections in Part II renumbered due to RETS relocation. Editorial change.
II 21	3.2.2.3	2.2.2.3	Replaced Technical Specifications with Part I.	Relocated requirements to Part I. Editorial.
II 21	3.3	2.3	Replaced Technical Specifications with Part I.	Relocated requirements to Part I. Editorial.
II 22	3.4	2.4	Changed Part II section reference.	Sections in Part II renumbered due to RETS relocation. Editorial change.
II 23	4.0	3.0	Replaced Technical Specifications with Part I.	Relocated requirements to Part I. Editorial.
II 25	4.2	3.2	Changed Part II section reference.	Sections in Part II renumbered due to RETS relocation. Editorial change.
II 26	4.4	3.4	Replaced Technical Specifications with Part I.	Relocated requirements to Part I. Editorial.
II 29	5.1	4.1	Replaced Technical Specifications with Part I.	Relocated requirements to Part I. Editorial.
II 30	5.3	4.3	Replaced Technical Specifications with Part I.	Relocated requirements to Part I. Editorial.
II 79	Appendix C	Appendix C	Replaced Technical Specifications with Part I.	Relocated requirements to Part I. Editorial.

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

The Unit 1 Off-Site Dose Calculation Manual (ODCM) Revision 23 was implemented on November 7, 2002. This change to the ODCM implements the requirements of Technical Specification 6.18 and NRC acceptance of elimination of RAGEMS. The nature of the changes are primarily administrative, with appropriate wording conversions and frequencies. The ODCM changes do not reduce the overall conformance of existing criteria in accordance with Technical Specifications. A copy of the ODCM, Revision 23 is attached and below is a summary of the changes approved by the Plant General Manager.

Page Number	Old Section Number	New Section Number	Change	Reason for Change
iv, v, vii, and viii	1.3.2, 2.1.5.2, 2.2.3, Table 5.1, and Figures	N/A, 2.1.5.2, N/A, Table 5.1, and Figures	Added Liquid and Gaseous Effluent Dose Projections sections and revised RAGEMS to reflect that the RAGEMS sample line will be used as an auxiliary sample point. Corrected a typo and deleted Figures relating to RAGEMS.	New sections are added to conform to Technical Specification 6.18 requirements. RAGEMS revision in accordance with "Second Supplemental Safety Evaluation by the Office of Nuclear Reactor Regulation Conformance with Regulatory Guide 1.97 and NUREG-0737 Nine Mile Point Nuclear Station, Unit No. 1" dated September 11, 2002. Administrative Change.
I 1.0-1 and I 1.0-2	1.0	1.0	Corrected typos.	Editorial.
I 3.1-1	DSR 4.6.14	DSR 4.6.14	Corrected typo in heading.	Editorial.
I 3.1-7	D 3/4.6.14.b	D 3/4.6.14.b	Replaced "specification" with "control."	Consistent with Terminology in NUREG-1302, "Offsite Dose Calculation Manual Guidance: Standard Effluent Controls for Boiling Water Reactors." Editorial.
I 3.1-8 and 9	Table D 3.6.14-2	Table D 3.6.14-2	Corrected instrument description and made additional editorial changes. Added action (c) for Stack Gas Flow Rate Measuring Device to provide direction on how frequently to perform the action when the device is out of service. to be consistent with regulatory guidance.	The Action to evaluate flow every 8 hours with the operable channels less than required is consistent with the Action for the Sampler Flow Rate Measuring Device and regulatory guidance. This is an Administrative Change.
I 3.1-11	Table D 4.6.14-2.3	Table D 4.6.14-2.3	Corrected Instrument descriptions.	To be consistent with Table D 3.6.14-2 descriptions. Editorial.
I 3.1-14	DLCO 3.6.15.a (2) (b)	DLCO 3.6.15.a (2) (b)	Moved margin in the second paragraph to clearly show that the requirement applies to both quarterly and annual limits.	Editorial change.
I 3.1-14	DSR 4.6.15.b(2)	DSR 4.6.15.b(2)	Changed frequency from "prior to release" to monthly.	To meet Tech. Spec. 6.18 requirements.
I 3.1-18	DLCO 3.6.15.b(1)	DLCO 3.6.15.b(1)	Revised wording to read "...from the site to areas at or beyond..."	To be consistent with the wording in Technical Specification Section 6.18. Administrative Change.

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

Page Number	Old Section Number	New Section Number	Change	Reason for Change
I 3.1-18	DLCO 3.6.15.b(1)(a)	DLCO 3.6.15.b(1)(a)	Replaced total body with whole body.	To be consistent with terminology in Technical Specification 6.18. Administrative Change.
I 3.1-19	DLCO 3.6.15.b(2)	DLCO 3.6.15.b(2)	Revised wording to read "...to areas beyond the site boundary..."	To be consistent with the wording in Technical Specification Section 6.18. Administrative Change.
I 3.1-20	DLCO 3.6.15.b(3)	DLCO 3.6.15.b(3)	Revised wording to read "...to areas beyond the site boundary..."	To be consistent with the wording in Technical Specification Section 6.18. Administrative Change.
I 3.1-21	Table D 4.6.15-2 C	Table D 4.6.15-2 C	Revised LLD for Continuous noble gas monitor to 1E-5 μCi/ml.	In accordance with the "Second Supplemental Safety Evaluation by the Office of Nuclear Reactor Regulation Conformance with Regulatory Guide 1.97 and NUREG-0737 Nine Mile Point Nuclear Station, Unit No. 1" dated September 11, 2002. Administrative Change.
I 3.1-22	Notes for Table D 4.6.15-2 (g)	Notes for Table D 4.6.15-2 (g)	Replaced RAGEMS with "the continuous Noble Gas Monitor. Corrected a grammatical error.	In accordance with the "Second Supplemental Safety Evaluation by the Office of Nuclear Reactor Regulation Conformance with Regulatory Guide 1.97 and NUREG-0737 Nine Mile Point Nuclear Station, Unit No. 1" dated September 11, 2002. Administrative Change.
I 3.1-23	DLCO 3.6.15.d	DLCO 3.6.15.d	Replaced total body with whole body.	For consistency in terminology. Administrative Change.
I 3.1-24	D 3/4.6.16	D 3/4.6.16	Deleted reference to section (b).	The requirement applies to both (a) and (b) and is consistent with Regulatory Guidance. This is an administrative change.
I 3.1-26	DLCO 3.6.16.a	DLCO 3.6.16.a	Revised to require the appropriate portions of the liquid radwaste treatment system be used to reduce releases of radioactivity when the projected doses due to liquid effluent to unrestricted areas would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ per batch.	In accordance with Technical Specifications 6.18.f. "Per Batch" is used instead of monthly because NMP1 does not perform liquid discharges frequently. Administrative Change.

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

Page Number	Old Section Number	New Section Number	Change	Reason for Change
I 3.1-26	DSR 4.6.16.b	DSR 4.6.16.b(1)	Revised wording to "...at or beyond the site boundary".	To be consistent with the terminology in Technical Specification 6.18. Administrative Change.
I 3.1-27	N/A	DLCO 3.6.16.b(2)	Added a control for Ventilation Exhaust Treatment System.	To meet the requirements in Technical Specification 6.18.f
I 3.1-27	N/A	DSR 4.6.16.b(2)	Added Surveillance Requirement for Ventilation Exhaust Treatment System.	To meet requirements of Technical Specification 6.18.f.
I B 3.1-1	B 3/4.6.14 and 15	B 3/4.6.14 and 15	Replaced 10CFR20 with Tech. Spec 6.18. Changed heading for Liquid Concentration to lower case.	Since the BASES for Rad. Eff. Inst. addresses both liquid and gaseous reference Tech. Spec. 6.18 to minimize extent of mark-up. Administrative change.
I B 3.1.3	B 3/4.6.15	B 3/4.6.15	Revised wording to read "...from the site to areas at or beyond..."	To be consistent with the wording in Technical Specification Section 6.18.g. Administrative Change.
I B 3.1-4	B 3/4.6.15	B 3/4.6.15	Deleted "...at and .."	To be consistent with the wording in Technical Specification Section 6.18.g. Administrative Change.
I B 3.1-5	B 3/4.6.15	B 3/4.6.15	Correct grammatical error.	Editorial.
I B 3.1-6	B 3/4.6.15	B 3/4.6.15	Replaced total body with whole body.	For consistency in terminology. Administrative Change.
I B 3.1-7	B 3/4.6.15	B 3/4.6.15	Revised the Bases for Liquid and Gaseous Radwaste Treatment Systems. The Liquid Bases is revised to explain that the dose projection is done on a per batch basis because liquid discharges are very infrequent at Unit 1. The Gaseous Radwaste Treatment System Bases is revised to add the Ventilation Exhaust Treatment System.	To meet the requirements of 6.18.f. The Ventilation Exhaust Treatment System wording was consistent with NUREG-1302 wording. Administrative Change.
I 6.0-2, 4, 5	D 6.9.1.e, D 6.9.3, and Table D 6.9.3-1	D 6.9.1.e, D 6.9.3, and Table D 6.9.3-1	Changes made to reference the correct figure in Part II, to reflect that special reports apply to (a) and (b), and to correct an editorial.	Administrative Changes to special report section consistent with regulatory requirements, other changes are editorial.
II 2, II 3	1.1.2	1.1.2	Added General Setpoint Equation and terminology for clarification.	The change provides clarification and is an administrative change.

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

Page Number	Old Section Number	New Section Number	Change	Reason for Change
II 3	1.1.3	1.1.3	Added General Setpoint Equation and terminology for clarification.	The change provides clarification and is an administrative change.
II 3	1.1.3	1.1.3	deleted " maximum permissible" in footnote.	The footnote relates to actual sample concentration in the RBCLC system not the maximum permissible concentration. Administrative Change.
II 4	1.1.3	1.1.3	Added/revised terminology for clarification.	The change provides clarification and is an administrative change.
II 7	1.1.4.5	1.1.4.5	Added sentence to provide guidance when non-gamma emitters are identified in service water, e.g. tritium.	Clarification. Administrative change.
II 7	1.2	1.2	Added "... at the point of discharge. For noble gases, this limit may also be satisfied by using 2E-4 μCi/ml as the MEC for each noble gas."	For clarification. Administrative Change.
II 11	N/A	1.3.2	Added section on how to perform liquid dose projections.	In accordance with Technical Specification 6.18.f. Administrative.
II 12, II 13	2.1.1 and 2.1.2	2.1.1 and 2.1.2	Replace "total" with "whole" and correct typo.	Administrative and editorial.
II 15, II 16	2.1.5.1, 2.1.5.2, 2.1.5.3	2.1.5.1, 2.1.5.2, 2.1.5.3	Rewritten to delete the use of RAGEMS as an effluent monitor and to describe the use of RAGEMS sample line as an auxiliary sample point. Added "whole body" to "dose rate"	Deletion of RAGEMS is in accordance with the "Second Supplemental Safety Evaluation by the Office of Nuclear Reactor Regulation Conformance with Regulatory Guide 1.97 and NUREG-0737 Nine Mile Point Nuclear Station, Unit No. 1" dated September 11, 2002. RAGEMS as an additional auxiliary sample point to obtain grab samples is acceptable since the heat trace will be maintained. Therefore representative samples can be obtained. Technical Change. Added the words "whole body" in conformance with T.S. 6.18 terminology.

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

Page Number	Old Section Number	New Section Number	Change	Reason for Change
II 16	2.1.5.4	2.1.5.4	Deleted reference to two stack monitoring systems and edited the sampling frequency and sample analysis to more clearly reflect sample frequencies and analysis	In accordance with above NRC correspondence and to more clearly reflect sample frequencies and analysis. Administrative and Editorial changes.
II 17	2.1.5.5	2.1.5.5	Added a sentence describing that I-135 will be included in the Radioactive Effluent Release Report for totaling curies but not totaling dose rate or dose.	In accordance with Regulatory Guide 1.21. This change is for clarification.
II 17	2.2	2.2	Revised the section to be consistent with Part I requirements.	In accordance with Technical Specification 6.18g and h.
II 18	2.2	2.2	Changed paragraph order and changed wording to reflect Technical Specification 6.18 requirements.	Paragraph order is an editorial change. Wording changes in accordance with Technical Specification 6.18.
II 20	2.2.1.1	2.2.1.1	Moved actions to take if offsite dose rates exceed 10% to the end of the section.	In order to describe how to calculate the dose rate first before describing what actions to take. Editorial change.
II 20	2.2.1.1	2.2.1.1	Changed total to whole body, added emergency condenser, deleted "...", and "the". Revised actions to request dose rate information instead of dose information to determine if the control is met. Added perform dose rate calculations as necessary. Stated that the setpoints are set at 90% of limit not 50% of limit.	Editorial changes.
II 21	2.2.1.2	N/A	Omitted statement that dose assessment also includes JAF and NMP-2.	Redundant. Editorial.
II 21	2.2.1.2	2.2.1.2	4.34E3. Added that the value is for the vegetation pathway, elevated release.	Technical change but does no change the final results (release rate action level). The pathway and release point is to allow future users to determine the value easily. Editorial.
II 21	2.2.1.2	2.2.1.2	Wording changes. Revised actions to request dose rate information instead of dose information to determine if the control is met. Added perform dose rate calculations as necessary.	Editorial and Administrative changes.

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

Page Number	Old Section Number	New Section Number	Change	Reason for Change
II 22	2.2.2.1	2.2.2.1	Added "II".	Editorial.
II 24	N/A	2.2.3	Added method to project gaseous doses.	In accordance with Technical Specification 6.18. Technical Change.
II 25	2.3	2.3	Rewording. Deleted discussion relating to Technical Specification wording.	Technical Specification wording changed in section 6.18. Administrative Change.
II 46	Table 3-2	Table 3-2	Revised B_i and V_i values for Xe-133 and Xe-138.	Math error in original calculation initiated change. Technical Change.
N/A	Figures D-7, D-9, D-10	N/A	Deleted RAGEMS figures.	Deletion of RAGEMS is in accordance with the "Second Supplemental Safety Evaluation by the Office of Nuclear Reactor Regulation Conformance with Regulatory Guide 1.97 and NUREG-0737 Nine Mile Point Nuclear Station, Unit No. 1" dated September 11, 2002. RAGEMS as an additional auxiliary sample point to obtain grab samples is acceptable since the heat trace will be maintained. Therefore representative samples can be obtained. Technical Change. Added the words "whole body" in conformance with T.S. 6.18 terminology.
D-8	Figure D-8	Figure D-8	Revised to reflect RAGEMS auxiliary sample point.	Administrative Change.

ATTACHMENT 8

Unit 1 X Unit 2 ___ Reporting Period January – December 2002

SUMMARY OF CHANGES TO THE PROCESS CONTROL PROGRAM (PCP)

The Unit 1 Radwaste Process Control Program (PCP) Revision 6 was implemented in November 2002. Information regarding solid radioactive waste was relocated from the Technical Specifications to the PCP. The PCP changes do not reduce the overall conformance of the solidified waste product to existing criteria for solid waste. A copy of the PCP, Revision 6 is attached and below is a summary of the changes accepted by the Station Operations Review Committee.

Old Page #	New Page #	New/Amended Section #	Change	Reason for Change
N/A	5	4.4.1.a	Changes to the Process Control Program (PCP) shall be reported to the Commission in the Radioactive Effluent Release Report for the period in which the change(s) was made. The submittal shall contain information as described in the Offsite Dose Calculation Manual (ODCM) section D 6.9.1.e, "Reporting Requirements."	Administrative Change
N/A	6 & 7	4.4.5	Solid Radioactive Waste Specification a. This Specification implements the requirements of 10CFR Part 50.36a and General Design Criteria 60 Of Appendix A to 10CFR part 50. The process parameters included in establishing the process control program may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents and mixing and curing times. b. The solid radwaste system shall be used in accordance with the Process Control Program to process wet radioactive wastes to meet shipping and burial ground requirements. c. With the provisions of the process control program not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site. d. The process control program shall be used to verify the solidification of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges and evaporator bottoms). 1. If any test specimen fails to verify solidification, the solidification of the batch may then be resumed using the alternative solidification parameters determined by the process control program. 2. If the initial test specimen from a batch of waste fails to verify solidification, the process control program shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate solidification.	Administrative Change

ATTACHMENT 8

Unit 1 X Unit 2 ___ Reporting Period January – December 2002

SUMMARY OF CHANGES TO THE PROCESS CONTROL PROGRAM (PCP)

Old Page #	New Page #	New/Amended Section #	Change	Reason for Change
N/A	7	5.2	5.2 Solidification Solidification shall be the conversion of wet or liquid waste into a form that meets shipping and burial ground requirements.	Clarification
7	7	6.1.1	Deleted subsections a, b, and e from 6.1.1. Relabled 6.1.1.c to 6.1.1.a and 6.1.1.d to 6.1.1.b	Administration Change
N/A	8	N/A	Added reference to ODCM section D6.9.1.e, Reporting Requirements	Administration Change

ATTACHMENT 9

Unit 1 X Unit 2 ___ Reporting Period January - December 2002

SUMMARY OF INOPERABLE MONITORS

There were no monitors inoperable for more than 30 days during the report period.

Unit 1 _X_ Unit 2 ___ Reporting Period January - December 2002

DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES INSIDE THE SITE BOUNDARY

Introduction

An assessment of the radiation dose received by a Member of the Public due to their activities inside the site boundary from Nine Mile Point Unit 1 (NMP1) liquid and gaseous effluents has been conducted for the period January through December 2002.

This assessment considers the likely maximum exposed individual and the various exposure pathways resulting from liquid and gaseous effluents to identify the maximum dose received by a Member of the Public during their activities within the site boundary.

Prior to September 11, 2001, fishing near the shoreline adjacent to the NMP Site was the onsite activity that resulted in the maximum dose received by a Member of the Public. Following September 11, 2001 access has been restricted and fishing by Members of the Public at locations on site is prohibited. Although fishing was not conducted during 2002 the annual dose to a fisherman was still evaluated

In addition to the dose received by a fisherman, dose received by a member of the National Guard was also evaluated For this reporting period the National Guard was conservatively assumed to be a Member of the Public.

Dose Pathways

Dose pathways considered for this evaluation included direct radiation, inhalation and external ground (shoreline sediment or soil doses) in accordance with the NMP1 Off-site Dose Calculation Manual (ODCM). 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 NMP1 Stack and Emergency Condenser Vent were evaluated for the inhalation pathway. Dose from liquid effluent pathways due to aquatic activities is not applicable since swimming is prohibited and lake water is not utilized as a source for drinking water at NMPNS

Dose to a fisherman is received through the following pathways while standing on the shoreline fishing:

- External ground pathway, received from plant related radionuclides detected in the shoreline sediment,
- Inhalation pathway, received through inhalation of gaseous effluents released from NMP1 Stack and Emergency Condenser Vent, and
- Direct radiation pathway, resulting from the operation of NMP1, Nine Mile Point Unit 2 (NMP2) and James A. Fitzpatrick (JAF).

Similarly, dose to a member of the National Guard is received through the following pathways while performing security functions at the site.

- Inhalation pathway, and
- Direct radiation pathway

Any dose received by a member of the National Guard through the external ground pathway is considered insignificant as compared to the dose received through inhalation and direct radiation.

Methodologies for Determining Dose for Applicable Pathways

External Ground (Shoreline Sediment) Pathway

Dose from the external ground (shoreline sediment) is based on the methodology in the NMP1 ODCM as adapted from Regulatory Guide 1.109. For this evaluation it is assumed that the maximum exposed individual fished from the shoreline at all times

The total dose received by the whole body and skin of the maximum exposed individual during 2002 was calculated using the following input parameters:

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DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES INSIDE THE SITE BOUNDARY

- Usage Factor = 312 hours (fishing 8 hours per week, 39 weeks per year)
- 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 20 pCi/g

The total whole body and skin doses received by the maximum exposed fisherman from the external ground pathway is presented in Table 1, Exposure Pathway Dose.

Inhalation Pathway

The inhalation dose pathway is evaluated by utilizing the inhalation equation in the NMP1 ODCM, as adapted from Regulatory Guide 1.109. The total whole body dose and organ dose received by the maximum exposed fisherman and member of the National Guard during 2002 is calculated using the following input parameters for gaseous effluents released from both the NMP1 Stack and Emergency Condenser Vent for the time period exposure is received.

NMP 1 Stack:

Variable	Fisherman *	National Guard *
Average Stack flow rate (m^3/sec)	1.14E+02	1.15E+02
X/Q (s/m^3)	8.9E-06	3.7E-07
Inhalation dose factor	Regulatory Guide 1.109 Table E-7	
Annual air intake (m^3/year) (adult)	8000	8000
Fractional portion of the year (hours)	0.0356	0.2630
H-3 (pCi/m^3)	5 550E+03	6 425E+03
Mn-54 (pCi/m^3)	1 084E-01	1.014E-01
Fe-55 (pCi/m^3)	1.421E-01	3.719E-01
Co-58 (pCi/m^3)	2 213E-02	1.849E-02
Co-60 (pCi/m^3)	2.473E-01	2.771E-01
Sr-89 (pCi/m^3)	1.692E-01	1.807E-01
Sr-90 (pCi/m^3)	5 711E-04	4 283E-04
Cs-137 (pCi/m^3)	2.289E-03	7.347E-03
Ba/La-140 (pCi/m^3)	4.300E-02	4 571E-02
I-131 (pCi/m^3)	3.318E-01	4 462E-01
I-133 (pCi/m^3)	1.061E+00	2.010E+00

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DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES INSIDE THE SITE BOUNDARY

NMP1 Emergency Condenser Vent:

Variable	Fisherman *	National Guard *
Average Vent flow rate (m^3/sec)	2.754E-05	2 854E-05
X/Q (s/m^3)	6.63E-06	6 63E-06
Inhalation dose factor	Regulatory Guide 1.109 Table E-7	
Annual air intake (m^3/year) (adult)	8000	8000
Fractional portion of the year (hours)	0.0356	0.2630
H-3 (pCi/m^3)	8.115+09	8.92E+09

- The maximum exposed fisherman is assumed to be present on site during the period of April through December at a rate of 8 hours per week for 39 weeks per year equivalent to 312 hours for the year (fractional portion of the year = 0 0356). Therefore, the Average Stack and Emergency Condenser Vent flow rates and radionuclide concentrations used to determine the dose are represented by second, third and fourth quarter gaseous effluent flow and concentration values The maximum exposed member of the National Guard is assumed to be on site 48 hours per week for 48 weeks per year equivalent to 2304 hours per year (fractional portion of the year = 0.2630) Therefore average Stack and Vent flow rates and average radionuclide concentrations used to determine the dose are represented by first, second, third and fourth quarter gaseous effluent flow and concentration data.

The total whole body dose and maximum organ dose received by the maximum exposed fisherman and member of the National Guard from the inhalation pathway is presented in Table 1, Exposure Pathway Dose.

Direct Radiation Pathway

The direct radiation pathway is evaluated in accordance with the methodology found in the NMP1 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 from plume submersion. The direct radiation pathway is evaluated by the use of high sensitivity environmental Thermoluminescent Dosimeters (TLDs). Since fishing activities occur between April 1 - December 31, TLD data for the second, third, and fourth quarters of 2002 from TLDs placed in the general area where fishing occurs were used to determine an average dose to the maximum exposed fisherman from direct radiation. The average dose to the maximum exposed member of the National Guard from direct radiation was determined from TLD data for the first, second, third and fourth quarters 2002. All four quarters were used as a result of the assumption that the maximum exposed member of the National Guard spends 48 weeks per year onsite. The following is a summary of the average dose rates and assumed time spent on site used to determine the total dose received by both a maximum exposed fisherman and member of the National Guard

Variable	Fisherman	National Guard
Average Dose Rate (mRem/hour)	1.32E-03	2.11E-04
Exposure time (hours)	312	2304

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DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES INSIDE THE SITE BOUNDARY

Total Doses received by the maximum exposed fisherman and member of the National Guard from direct radiation is presented in Table 1, Exposure Pathway Dose

Dose Received By The Maximum Exposed Member Of The Public During 2002

The following is a summary of the dose received by both the likely maximum exposed fisherman and likely maximum exposed member of the National Guard from Liquid and Gaseous effluents released from NMP1 during 2002:

Table 1

Exposure Pathway Dose

Exposure Pathway	Dose Type	Fisherman (mRem/year)	National Guard (mRem/year)
External Ground	Whole Body	3.11E-03	N/A
	Skin of Whole Body	3.62E-03	N/A
Inhalation	Whole Body	3.22E-04	6.52E-04
	Maximum Organ	Thyroid 5.47E-04	Thyroid 7.60E-04
Direct Radiation	Whole Body	0.411	0 486

These doses are generally a result of the operation of NMP1; however, a portion of these doses for the direct radiation pathway may be attributable to the Nine Mile Point Unit 2 and James A. Fitzpatrick Facilities. Based on the above doses the total annual dose received by the likely maximum exposed Member of the Public as a result of their activities inside the site boundary during 2002 are summarized below:

TABLE 2

Annual Dose Summary

Total Annual Dose for 2002	Fisherman (mRem/year)	National Guard (mRem/year)
Total Whole Body	0 414	0 487
Skin of Whole Body	3.62E-03	N/A
Maximum Organ	Thyroid 5 47E-04	Thyroid 7.60E-04

Unit 1 _X_ Unit 2 ___ Reporting Period January - December 2002

DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES OUTSIDE THE SITE BOUNDARY

Introduction

An assessment of radiation doses received by the likely most exposed Member of the Public located beyond the site boundary was conducted for the period January through December 2002 for comparison against the 40 CFR 190 annual dose limits

The intent of 40 CFR 190 requires that the effluents of Nine Mile Point Unit 1 (NMP1), as well as other nearby uranium fuel cycle facilities, be considered. In this case, the effluents of NMP1, Nine Mile Point Unit 2 (NMP2) and the James A. FitzPatrick (JAF) facilities must be considered

40 CFR 190 requires the annual radiation dose received by a Member of the Public in the general environment, as a result of plant operations be limited to:

- < 25 mRem wholebody
- < 25 mRem any organ (except thyroid)
- < 75 mRem thyroid

This evaluation compares doses resulting from Liquid and Gaseous effluents and direct radiation, originating from the site as a result of the operation of the NMP1, NMP2 and JAF nuclear facilities to the limits of 40 CFR 190

Dose Pathways

Dose pathways considered for this evaluation included doses resulting from liquid effluents, gaseous effluents and direct radiation from all nuclear operating facilities located on the Nine Mile Point Site.

Dose to the likely most exposed Member of the Public, outside the site boundary, is received through the following pathways:

- Fish consumption pathway, received from plant radionuclides that have concentrated in fish that is consumed by a Member of the Public,
- Shoreline Sediment, received as a result of an individual's exposure to plant radionuclides deposited in the shoreline sediment, which is used as a recreational area,
- Deposition, Inhalation and Ingestion pathways resulting from gaseous effluents, this dose is received through exposure of gaseous effluents released from NMP1, NMP2 and JAF operating facilities, and
- Direct Radiation pathway, resulting from the operation of NMP1, NMP2 and JAF facilities

Methodologies for Determining Dose for Applicable Pathways

Fish Consumption

Dose received as a result of fish consumption is based on the methodology specified in the NMP1 Off-site Dose Calculation Manual (ODCM) as adapted from Regulatory Guide 1.109. The dose for 2002 is calculated from actual analysis results of environmental fish samples taken near the site discharge points. For this evaluation it is assumed that the likely most exposed Member of the Public consumes fish taken near the site discharge points.

The total dose received by the whole body and organs of the likely most exposed Member of the Public during 2002 was calculated using the following input parameters:

- Average Cs-137 concentration = 1.60E-02 pCi/g (wet)
- Consumption Rate = 21 kg/yr
- Ingestion dose factor for Cs-137 = Regulatory Guide 1.109 Table E-11
- Fractional portion of the year = 1

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DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES OUTSIDE THE SITE BOUNDARY

Shoreline Sediment

Dose received from shoreline sediment is based on the methodology in the NMP1 ODCM as adapted from Regulatory Guide 1.109. For this evaluation it is assumed that the likely most exposed Member of the Public spends 67 hours/year along the shoreline for recreational purposes.

The total dose received by the whole body and skin of the likely most exposed individual during 2002 is calculated using the following input parameters:

- Usage Factor = 67 hours per year
- 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
- Average Cs-137 Concentration = 0 049 pCi/g

Dose Pathways Resulting From Gaseous Effluents

Dose received by the likely most exposed Member of the Public due to gaseous effluents is calculated in accordance with the methodology provided in the NMP1 ODCM, NMP2 ODCM, and the JAF ODCM. These calculations consider deposition, inhalation and ingestion pathways. The total sum of doses resulting from gaseous effluents from NMP1, NMP2 and JAF during 2002 provide a total dose to the whole body and maximum organ dose for this pathway.

Direct Radiation Pathway

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 Thermoluminescent Dosimeters (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 2002, the closest residence and the critical downwind residence are at the same location.

Dose Received by a Member of the Public Outside the Site Boundary During 2002

The following is a summary of doses received by the likely most exposed Member of the Public due to their activities outside the site boundary during 2002:

Exposure Pathway	Dose Type	Dose (mRem/year)
Fish Consumption	Total Whole Body	0.0240
	Total Maximum Organ	Liver: 0.0366
Shoreline Sediment	Total Whole Body	1.65E-04
	Total Skin of Whole Body	1.93E-04
Gaseous Effluents	Total Whole Body	1.18E-02
	Total Maximum Organ	Thyroid: 6 06E-02
Direct Radiation	Total Whole Body	No Dose

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DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES OUTSIDE THE SITE BOUNDARY

Based on these values the maximum total annual dose received by the likely most exposed Member of the Public during 2002 is as follows:

- Total Whole Body: 0 036 mRem / year
- Total Skin of Whole Body: 1.93E-04 mRem / year
- Maximum Organ: Thyroid: 0 061 mRem / year

40 CFR 190 Evaluation

The maximum total doses presented in this attachment are the result of operations at the NMP1, NMP2 and the JAF facilities The maximum organ dose (Thyroid 0 061 mRem) and the maximum whole body dose (0.036 mRem) 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

Radwaste Process Control Program (RPCP) Rev. 6

NINE MILE POINT NUCLEAR STATION UNIT 1

RPCP

REVISION 06

UNIT 1 RADWASTE PROCESS CONTROL PROGRAM

TECHNICAL SPECIFICATION REQUIRED

Approved by:
L. A. Hopkins

Plant General Manager

10/30/02
Date

Effective Date: 11/7/2002

TABLE OF CONTENTS

1.0 PURPOSE

To describe the methods for processing, packaging, transporting, and storing low-level radioactive waste and provide assurance of complete stabilization of various radioactive wastes in accordance with applicable NRC & DOT regulations and guidelines.

2.0 RESPONSIBILITIES

2.1 The Plant Manager is responsible for:

2.1.1 Ensuring the Unit 1 Radwaste Process Control Program provides for the health and safety of the general public as it applies to Radwaste Management.

2.1.2 Reviewing and approving changes to the Unit 1 Radwaste Process Control Program in accordance with the applicable Technical Specification.

2.2 The Radiation Protection Manager is responsible for the content and maintenance of this program.

2.3 The Supervisor Radwaste is responsible for overall implementation of the Radwaste Process Control Program.

3.0 PROGRAM

3.1 System Description

3.1.1 General

a. The Solid Waste Management System (SWMS) implemented by the procedures identified in the Unit 1 Radwaste Process Control Program Implementing Procedures (Attachment 1) collects, reduces the volume, dewaters and packages wet and dry types of radioactive waste in preparation for shipment off-site for further processing or disposal at a licensed burial site. The processing and storage methods used for interim storage are consistent with the present waste form stability requirements.

b. Types of solid waste sources are identified in Solid Waste Sources (Attachment 2).

3.1.1 (Cont)

c. Bead resins, powdered resins and charcoal are dewatered using approved vendor equipment in:

1. Vendor certified polyethylene containers, or

2. Carbon steel liners, or a

3. High Integrity Container (HIC)

d. Concentrated wastes are processed off-site to dryness by an approved vendor.

e. Evaporator bottoms are transferred to a liner in the Radwaste Truck Bay for off-site processing by an approved vendor.

f. Dry solid trash is collected in the Radwaste Facility, sorted, and sent off-site for further separation and processing.

3.1.2 Ventilation Systems

a. The Radwaste Building Ventilation System provides filtered, conditioned outside air to various areas of the Radwaste Building and exhausts the air to the atmosphere through the Turbine Building stack. (The system maintains the building at a pressure below atmospheric to help prevent any unmonitored air leakage to the environment.)

b. The Radwaste Solidification and Storage Building (RSSB) Ventilation System provides filtered, conditioned outside air to selected areas in the RSSB. Recirculation fans continuously filter and condition the air, and exhaust fans, taking a suction on the truck bays, exhaust the air to the Turbine Building stack. (The system maintains the building at a pressure below atmospheric to help prevent any unmonitored air leakage to the environment.)

3.1.3 Crane

a. All liner movements are completed using a remote controlled/operated crane. The movements are facilitated by the use of remote controlled cameras and monitors.

3.1.3 (Cont)

b. Liners are moved when required using a ceiling grid coordinated system for placement of the liner.

c. When liners stored in the RSSB storage area are to be shipped, the liners scheduled for shipment are moved to the East-West Truck Bay and then loaded for transportation.

4.0 RADIOACTIVE WASTES

4.1 Waste Processing System

The Supervisor Radwaste shall ensure:

4.1.1 Radioactive waste is processed using approved equipment with approved procedures.

4.1.2 Radioactive waste may be processed using approved vendor equipment and procedures.

4.1.3 Radioactive wastes are disposed of in the applicable approved containers.

4.1.4 Radioactive waste is transferred into shipping casks in accordance with approved procedures.

4.1.5 Waste is transferred between units and placed in interim storage in accordance with approved procedures.

4.2 Solid Dry Radioactive Wastes (SDRW)

The Supervisor Radwaste shall ensure:

4.2.1 Low Specific Activity (LSA) Solid Dry Radioactive Waste (SDRW) is collected and prepared in accordance with the applicable procedure, meeting 10CFR61, Sub Part D, Technical Requirements for Land Disposal Facilities and Final Waste Classification and Waste Form Technical Position Papers requirements.

4.2.2 SDRW is examined for liquids or items that could compromise the integrity of the package or violate the burial site license and/or criteria. These items are removed or separated.

4.2.3 SDRW is shipped in containers meeting the transport requirements of 49CFR173.427, Transport Requirements for Low Specific Activity (LSA) Radioactive Materials.

4.2.4 Waste precluded from disposal in LSA boxes or drums, due to radiation limits, is disposed of in the applicable containers.

4.2.5 Waste segregation and volume reduction processing techniques are used for waste generated during operation, maintenance, and modifications.

4.2.6 Scrap metal is separated from waste, when possible, for on-site or off-site decontamination.

NOTE: Vendor services may be used for waste segregation and further volume reduction processes.

4.2.7 Waste is placed in interim storage in accordance with approved procedures.

4.3 Waste Classification/Characterization

4.3.1 The Supervisor Radwaste shall ensure:

a. The minimum waste classification/characteristic requirements identified in 10CFR61.56, Waste Characteristics, are satisfied.

b. The radionuclide concentration determination methods and frequency are conducted in accordance with approved procedures.

4.3.2 The Manager Chemistry shall ensure the chemical and radionuclide content of waste is determined in accordance with the applicable Chemistry procedures.

4.3.3 The Manager Radiation Protection shall ensure classification of waste is performed in accordance with approved procedures.

4.4. Administrative Controls

4.4.1 The Supervisor Radwaste is responsible for overall administrative control of the Radwaste Process Control Program, ensuring:

4.4.1 (Cont)

a. Changes to the Process Control Program (PCP:) shall be reported to the Commission in the Radioactive Effluent Release Report for the period in which the change(s) was made. The submittal shall contain information as described in the Offsite Dose Calculation Manual (ODCM) section D 6.9.1.e, "Reporting Requirements";

b. Shipping manifests are completed and tracked to satisfy the requirements of 10CFR20.2006, Transfer for Disposal and Manifests, in accordance with Waste Handling Procedures.

c. Temporary storage of solid radioactive material awaiting shipment in an area other than a designated storage area is done in accordance with the applicable radioactive material storage procedures.

d. Interim storage of low level waste is performed in accordance with approved procedures.

4.4.2 The Nuclear Division Quality Assurance Program assures effective implementation of the Process Control Program, as follows:

NOTE: The Manager, Nuclear QA, Operations has the authority to stop work when significant conditions adverse to quality exist and require corrective action.

a. Under the cognizance of the Safety Review and Audit Board (SRAB), the Process Control Program and implementing procedures for processing and packaging of radioactive waste are audited at least once every 24 months as required by the applicable Unit 1 Technical Specification.

b. QA audits waste classification records to ensure compliance with 10CFR20.2006, Transfer for Disposal and Manifests.

c. QA Inspectors performing Radwaste inspections receive training in Department of Transportation and NRC Radwaste Regulatory requirements.

d. Management reviews results of QA audits.

4.4.3 The Nuclear Division Training Program assures personnel responsible for implementation of the Process Control Program are effectively trained in accordance with the applicable training procedures as follows:

a. Qualification as a Radwaste Operator requires satisfactory completion of the Radwaste Operations Unit 1 Initial Training Program and participation in continued training. This includes:

1. Demonstrating an acceptable level of skill and familiarity associated with Radwaste operations by achieving an average grade of 80 percent or above on written examinations.

2. Receiving on-the-job training in accordance with applicable training procedures.

3. Continued training conducted on a cyclical basis and includes a fundamental review of system modifications, revisions or changes to procedures, and changes or experiences in the nuclear industry.

4. Individuals that demonstrate a significant deficiency in a given area of knowledge and/or proficiency (as identified during continued training) are placed in a remedial training program as directed by approved training procedures.

4.4.4 Training records and Waste Management records are maintained in accordance with applicable Quality Assurance procedures.

4.4.5 Solid Radioactive Waste Specification

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

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

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

4.4.5 (Cont)

d. The process control program shall be used to verify the solidification of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges and evaporator bottoms).

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

2. If the initial test specimen from a batch of waste fails to verify solidification, the process control program shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate solidification.

5.0 DEFINITIONS

5.1 The applicable Radwaste packaging, processing, and transportation definitions will be used in accordance with 49CFR171 and 49CFR Sub Part I.

5.2 Solidification

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

6.0 REFERENCES

6.1 Licensee Documentation

6.1.1 Unit 1 Technical Specifications

a. Section 6.5.2.11, Technical Review and Control

b. Section 6.5.3.8.k, Audits of Facility Activities

6.1.2 Unit 1 Radiological Effluent Technical Specifications, Amendment No. 66

6.1.3 Nine Mile Point Unit 1 Operating License No. DPR-63 (Docket No. 50-220)

6.1.4 QATR-1, Quality Assurance Program Topical Report for Nine Mile Point Nuclear Station Operations, Section 17.0, Quality Assurance Records

6.1.5 UFSAR, Section XII.A, Radioactive Wastes

6.1.6 UFSAR, Section III.I, RSSB

6.1.7 Safety Evaluation 92-049, Rev. 04, Interim Storage

6.1.8 Offsite Dose Calculation Manual (ODCM) section D 6.9.1.e, Reporting Requirements

6.2 Standards, Regulations, and Codes

6.2.1 10CFR20, Standards for Protection Against Radiation

6.2.2 10CFR61, Sub Part D, Technical Requirements for Land Disposal Facilities and Final Waste Classification and Waste Form Technical Position Papers

6.2.3 10CFR61.55, Waste Classification

6.2.4 10CFR61.56, Waste Characteristics

6.2.5 10CFR71, Packaging and Transportation of Radioactive Material, (Refer to applicable S-RPIPs for the packaging and transportation of radioactive material)

6.2.6 49CFR173, Shippers - General Requirements for Shipment and Packagings, (Refer to applicable S-RPIPs for the packaging and transportation of radioactive material)

6.2.7 49CFR173.427, Transport Requirements for Low Specific Activity (LSA) Radioactive Materials

6.2.8 NUREG-0133, Section 3.5, Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants

6.2.9 NUREG-0473, Sections 3.11.3 and 6.14, Draft Radiological Effluent Technical Specifications for Boiling Water Reactors

6.2.10 NUREG-0800, Section 11.4, Standard Review Plan for Solid Waste Management Systems

6.3 Policies, Programs, and Procedures

6.3.1 NDD-LPP, Licenses, Plans, and Programs

6.3.2 NDD-OPS, Operations

6.3.3 NDD-RMP, Radioactive Material Processing, Transport, and Disposal

6.3.4 NIP-ECA-01, Deviation/Event Report

6.3.5 NIP-PRO-03, Preparation and Review of Technical Procedures

6.3.6 NIP-RMG-01, Records Management

6.3.7 NIP-TQS-01, Qualification and Certification

6.3.8 GAP-ALA-01, Site ALARA Program

6.3.9 GAP-INV-02, Control of Material Storage Areas

6.3.10 GAP-OPS-01, Administration of Operations

6.3.11 GAP-RPP-01, Radiation Protection Program

6.3.12 GAP-RPP-02, Radiation Work Permit

6.3.13 GAP-RMP-01, Interim Storage of Low-Level Radioactive Waste

6.4 Supplemental References

6.4.1 Vendor Training and Requalification Procedure

6.4.2 Nuclear Regulatory Commission's Branch Technical Position of Waste Classification and Waste Form, May 1983

6.4.3 DER 1-94-0549

6.4.4 Structural Calculation S.2.3-R5252-Tank 01

6.4.5 Modification N1-91-033

6.4.6 Procedure N1-MFT-30

ATTACHMENT 1: UNIT 1 RADWASTE PROCESS CONTROL PROGRAM IMPLEMENTING PROCEDURES

Waste Handling Procedures (N1-WHPs and S-WHPs)

Liquid Waste Processing Procedures (N1-LWPPs)

Radiation Protection Procedures (S-RPIPs)

Chemistry Technical Procedures (N1-CTPs)

Quality Assurance Audit and Surveillance Procedures (QAPs)

Nuclear Training Procedures (NTPs)

Generation Administrative Procedures (GAPs)

ATTACHMENT 2: SOLID WASTE SOURCES

1.0 RADWASTE FILTERS

1.1 Mechanical Radwaste filters filter resin and crud (backwash material) from the Waste Collector Sub-System.

1.2 When a filter reaches a pre-determined differential pressure, the filter is backwashed into the filter sludge tank, which is then processed via the clarifier to the thickener tanks.

2.0 RADWASTE DEMINERALIZER

2.1 The Radwaste Demineralizer is used as anionic exchange media for processing high quality water from the Waste Collector Tanks.

2.2 When determined the resin can NO longer be used, the depleted resin is transferred to the Spent Resin Tank.

3.0 CONDENSATE DEMINERALIZERS

3.1 The Condensate Demineralizers remove soluble and insoluble impurities from the condensate water to maintain reactor feedwater purity.

3.2 After it is determined these resins can NO longer be used, the depleted resin are transferred to the Radwaste Demineralizer or Spent Resin Tank.

4.0 THERMEX SYSTEM

4.1 Concentrate will be pumped to the Spent Resin Tank and dewatered or stored in a liner and eventually pumped to a transport liner in the Radwaste Truck Bay for off-site processing.

4.2 Exhausted resin and charcoal are transferred to the Spent Resin Tank, mixed to a homogenous mixture and then transferred to a liner in the truck bay for dewatering.

4.3 Exhausted Reverse Osmosis membranes will be processed as DAW.

5.0 FUEL POOL FILTER SLUDGE TANK

This tank receives the exhausted powdered filter media (resins) from the Fuel Pool Cleanup System, which is subsequently pumped to the Filter Sludge Tank for processing.

6.0 CLEANUP FILTER SLUDGE TANK

This tank receives the exhausted powdered filter media (resins) from the Reactor Cleanup System, which is subsequently pumped to the Filter Sludge Tank, Clarifier, or directly to a liner in the Radwaste Truck Bay for processing.

7.0 FILTER SLUDGE STORAGE TANK

This tank receives waste from the Radwaste filters, Fuel Pool and Cleanup Sludge Tanks, Clarifier and Thickener Tank overflows, and Radwaste Floor Drain Sump #11. Tank discharge is to the Clarifier (Filter Sludge Thickener System) or directly to a liner in the Radwaste Truck Bay for processing.

8.0 FILTER SLUDGE THICKENER TANKS (CLARIFIER)

Waste from the Filter Sludge Storage Tank or the Cleanup Filter Sludge Tank is pumped to the Clarifier, mixed with a flocculent and drained in the Thickener Tanks. The Thickener Tanks are pumped to a liner in the Radwaste Truck Bay for processing.

9.0 SPENT RESIN STORAGE TANK

Exhausted resin from the Condensate Demineralizers, Radwaste Demineralizer, and THERMEX System are transferred to the Spent Resin Tank. The tank is subsequently pumped to a liner in the Radwaste Truck Bay for dewatering and further processing.

10.0 CONTAMINATED OIL

Oil from sources within Unit 1 that becomes contaminated is stored in containers to be shipped off-site for incineration.

11.0 COMPACTIBLE SOLIDS

11.1 Compactible low level trash is shipped off-site for vendor separation and processing.

11.2 Shoe covers, trash, contaminated paper from the Chemistry Lab, and similar materials are included in this category.

12.0 FILTERS AND MISCELLANEOUS ITEMS

Solid items with high dose rates are handled on a case-by-case basis, being disposed of by methods acceptable to the burial site or shipped off-site for vendor recovery or disposal.

13.0 WASTE EVAPORATOR

13.1 The Waste Evaporator processes low quality waste from the Floor Drain Collector System.

13.2 The Waste Evaporator is designed to concentrate waste to a 25% solid concentration, which may then be discharged to the Evaporator Bottoms Tank for transfer to the Radwaste Truck Bay for vendor processing.

ATTACHMENT 13

Off-Site Dose Calculation Manual (ODCM) Rev. 23

ORIGINAL

Constellation Energy Group

Nine Mile Point
Nuclear Station

NINE MILE POINT NUCLEAR STATION

NINE MILE POINT UNIT 1

OFF-SITE DOSE CALCULATION MANUAL (ODCM)

APPROVALS	SIGNATURES	DATE REVISION 23
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Checked by:	G. R. Stinson Health Physicist	10/30/02
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CONTROLLED

L. A. Hopkins	LA Hopkins Plant General Manager	10/30/02
B.S. Montgomery	B Montgomery Manager Engineering Services	10/30/02

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INTRODUCTION

The Offsite Dose Calculation Manual (ODCM) provides the methodology to be used for demonstrating compliance with 10 CFR 20, 10 CFR 50, and 40 CFR 190. The contents of the ODCM are based on Draft NUREG-0472, Revision 3, "Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors," September 1982; Draft NUREG-0473, Revision 2, "Radiological Effluent Technical Specifications for BWR's", July 1979; NUREG 0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978; the several Regulatory Guides referenced in these documents; and, communication with the NRC staff.

Should it be necessary to revise the ODCM, these revisions will be made in accordance with Technical Specifications.

The Offsite Dose Calculation Manual (ODCM) is a supporting document of the Technical Specifications Section 6.11. 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 I, includes the following: (1) The Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specifications 6.18 and 6.11 respectively, 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 6.9.1.d and 6.9.1.e. Calculational Methodologies, Part II, describes 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.

PART I – RADIOLOGICAL EFFLUENT CONTROLS

PART I – RADIOLOGICAL EFFLUENT CONTROLS

Section 1.0 Definitions

***NOTE*:**

Technical Specifications defined terms and the following additional defined terms are applicable throughout these controls and bases.

Gaseous Radwaste Treatment System

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

Member(s) of the Public

Member(s) of the public shall include persons who are not occupationally associated with the Nine Mile Point Nuclear Station. This category does not include employees of owners and operators of Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant, their contractors or vendors who are occupationally associated with Nine Mile Point Unit 1. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with Nine Mile Point Unit 1.

Milk Sampling Location

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

Offsite Dose Calculation Manual (ODCM)

The Offsite Dose Calculational Manual shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Technical Specifications 6.9.1.d and 6.9.1.e and Controls D 6.9.1.d and D 6.9.1.e.

Purge – Purging

Purge or purging is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement. The purge is completed when the oxygen concentration exceeds 19.5 percent.

Site Boundary

The site boundary shall be that line around the Nine Mile Point Nuclear Station beyond which the land is neither owned, leased, nor otherwise controlled by the owners and operators of Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant.

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

The 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. That area outside the restricted area (10 CFR 20.1003) but within the site boundary will be controlled by the owner as required.

Ventilation Exhaust Treatment System

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

Venting

Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during venting. Vent, used in system names, does not imply a venting process.

PART I – RADIOLOGICAL EFFLUENT CONTROLS

Sections 3.0/4.0 Applicability

3.0 CONTROLS

The Offsite Dose Calculation Manual (ODCM) Part I, Radiological Effluent Controls, is subject to Technical Specifications Section 3.0 requirements, as applicable.

4.0 SURVEILLANCE REQUIREMENTS

The ODCM Part I, Radiological Effluent Controls , is subject to Technical Specifications Section 4.0 requirements, as applicable.

CONTROLS	SURVEILLANCE REQUIREMENT
DLCO 3.6.14 RADIOACTIVE EFFLUENT INSTRUMENTATION	DSR 4.6.14 RADIOACTIVE EFFLUENT INSTRUMENTATION
Applicability:	Applicability:
Applies to the operability of plant instrumentation that monitors plant effluents.	Applies to the surveillance of instrumentation that monitors plant effluents.
Objective:	Objective:
To assure the operability of instrumentation to monitor the release of radioactive plant effluents.	To verify operation of monitoring instrumentation.
Specification:	Specification:
a. Liquid Effluent	a. Liquid Effluent
The radioactive liquid effluent monitoring instrumentation channels shown in Table D 3.6.14-1 shall be operable with their alarm setpoints set to ensure that the limits of Control DLCO 3.6.15.a.1 are not exceeded. The alarm setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in Part II.	Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated operable by performance of the sensor check, source check, instrument channel calibration and channel test operations at the frequencies shown in Table D 4.6.14-1.
With a radioactive liquid effluent monitoring instrumentation channel alarm setpoint less conservative than a value which will ensure that the limits of DLCO 3.6.15.a.1 are met, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.	Records – Auditable records shall be maintained, in accordance with procedures in Part II, of all radioactive liquid effluent monitoring instrumentation alarm setpoints. Setpoints and setpoint calculations shall be available for review to ensure that the limits of Control DLCO 3.6.15.a.1 are met.

CONTROLS	SURVEILLANCE REQUIREMENT
With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels operable, take the action shown in Table D 3.6.14-1. Restore the instruments to OPERABLE status within 30 days, or outline in the next Radioactive Effluent Release Report the cause of the inoperability and how the instruments were or will be restored to operable status.	

TABLE D 3.6.14-1
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

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

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

NOTES FOR TABLE D 3.6.14-1

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

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

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

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

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

Otherwise suspend release of radioactive effluents via this pathway.

(d) With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for gamma radioactivity at a lower limit of detection of at least $5x10^{-7}$ microcurie/ml.

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

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

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

(h) Deleted.

(i) Monitoring will be conducted continuously by alternately sampling the reactor building and turbine building service water return lines for approximately 15-minute intervals.

TABLE D 4.6.14-1
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION
Surveillance Requirement

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

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

NOTES FOR TABLE D 4.6.14-1

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

1. Instrumentation indicates measured levels above the alarm setpoint.
2. Instrument indicates a downscale failure.
3. Instrument controls not set in operate mode.

(b) The channel calibration shall be performed using one or more reference standards certified by the National Institute of Standards and Technology (NIST), or using standards that are traceable to the NIST or using actual samples of liquid waste 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.

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

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

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

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

CONTROLS	SURVEILLANCE REQUIREMENT
b. Gaseous Process and Effluent The radioactive gaseous process and effluent monitoring instrumentation channels shown in Table D 3.6.14-2 shall be operable with their alarm setpoints set to ensure that the limits of Control DLCO 3.6.15.b.1 are not exceeded. The alarm setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in Part II. With a radioactive gaseous process and effluent monitoring instrumentation channel alarm setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative. With less than the minimum number of radioactive gaseous process and effluent monitoring instrumentation channels operable, take the action shown in Table D 3.6.14-2. Restore the instruments to OPERABLE status within 30 days or outline in the next Radioactive Effluent Release Report the cause of the inoperability and how the instruments were or will be restored to operable status.	b. Gaseous Process and Effluent Each radioactive gaseous process and effluent monitoring instrumentation channel shall be demonstrated operable by performance of the sensor check, source check, instrument channel calibration and instrument channel test operations at the frequencies shown in Table D 4.6.14-2. Auditable records shall be maintained of the calculations made, in accordance with procedures in Part II, of radioactive gaseous process and effluent monitoring instrumentation alarm setpoints. Setpoints and setpoint calculations shall be available for review to ensure that the limits of Control DLCO 3.6.15.b.1 are met.

TABLE D 3.6.14-2
RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Applicability	Action
1. Stack Effluent Monitoring			
a. Noble Gas Activity Monitor	1	*	(a)
b. Iodine Sampler Cartridge	1	*	(b)
c. Particulate Sampler Filter	1	*	(b)
d. Sampler Flow Rate Measuring Device	1	*	(c)
e. Stack Gas Flow Rate Measuring Device	1	*	(c), (d)
2. Deleted			

* At all times.
** Note Deleted.

TABLE D 3.6.14-2 (cont'd)
RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Applicability	Action
3. Condenser Air Ejector Radioactivity Monitor (Recombiner Discharge or Air Ejector Discharge)			
a. Noble Gas Activity Monitor	1	***	(g)
b. Offgas System Flow Rate Measuring Device	1	***	(c)
c. Sampler Flow Rate Measuring Device	1	***	(c)
4. Emergency Condenser System			
a. Noble Gas Activity Monitor	1 per vent	****	(h)

*** During operation of the main condenser air ejector

**** During power operating conditions and whenever the reactor coolant temperature is greater than 212°F except for hydrostatic testing with the reactor not critical.

NOTES FOR TABLE D 3.6.14-2

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

(b) With the number of channels operable less than required by the minimum channels operable requirements, effluent releases via this pathway may continue provided that samples are continuously collected with auxiliary sampling equipment starting within 8 hours of discovery in accordance with the requirements of Table D 4.6.15-2.

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

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

(e) Deleted

(f) Deleted

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

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

2. The stack monitor is operable.

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

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

TABLE D 4.6.14-2
RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION

Surveillance Requirements

Instrument	Sensor Check	Source Check	Channel Test	Channel Calibration
1. Stack Effluent Monitoring System				
a. Noble Gas Activity Monitor	Once/day[a]	Once/month	Once/3 months[g]	Once/year[b]
b. Iodine Sampler Cartridge	None	None	None	None
c. Particulate Sampler Filter	None	None	None	None
d. Sampler Flow Rate Measuring Device	Once/day[a]	None	None	Once/year
e. Stack Gas Flow Rate Measuring Device	Once/day	None	None	Once/year
2. Deleted				
3. Condenser Air Ejector Radioactivity Monitor (Recombiner Discharge or Air Ejector Discharge)				
a. Noble Gas Activity Monitor	Once/day[f]	Once/month	Once/operating cycle[c]	Once/year[b]
b. Offgas System Flow Rate Measuring Device	Once/day[f]	None	None	Once/year
c. Sampler Flow Rate Measuring Device	Once/day[f]	None	None	Once/year
4. Emergency Condenser System	Once/day[h]	Once/month	Once/3 months[g]	Once/operating cycle[b]
a. Noble Gas Activity Monitor				

NOTES FOR TABLE D 4.6.14-2

(a) At all times.

(b) The channel calibration shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST), standards that are traceable to the NIST or using actual samples of gaseous effluent 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.

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

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

2) Instrument indicates a downscale failure.

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

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

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

(d) Deleted

(e) Deleted

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

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

(h) During power operating conditions and whenever the reactor coolant temperature is greater than 212°F except for hydrostatic testing with the reactor not critical.

CONTROLS	SURVEILLANCE REQUIREMENT
DLCO 3.6.15 RADIOACTIVE EFFLUENTS	DSR 4.6.15 RADIOACTIVE EFFLUENTS
Applicability: Applies to the radioactive effluents from the station.	Applicability: Applies to the periodic test and recording requirements of the station process effluents.
Objective: To assure that radioactive material is not released to the environment in any uncontrolled manner and is within the limits of 10CFR20 and 10CFR50 Appendix I.	Objective: To ascertain that radioactive effluents from the station are within the allowable values of 10CFR20, Appendix B and 10CFR50, Appendix I.
Specification: a. Liquid (1) Concentration The concentration of radioactive material released in liquid effluents to unrestricted areas shall be limited to ten times the concentrations specified in 10CFR Part 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 2×10^{-4} microcuries/ml total activity. Should the concentration of radioactive material released in liquid effluents to unrestricted areas exceed the above limits, restore the concentration to within the above limits immediately.	Specification: a. Liquid (1) Concentration Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table D 4.6.15-1. The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in Part II to assure that the concentrations at the point of release are maintained within the limits of Control DLCO 3.6.15.a.(1).

CONTROLS	SURVEILLANCE REQUIREMENT
(2) Dose The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released, from each reactor unit, to unrestricted areas (see Figures 5.1-1) shall be limited: (a) During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and (b) During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Control D 6.9.3, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.	(2) Dose Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in Part II monthly.

TABLE D 4.6.15-1
RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
Surveillance Requirement

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

* Completed prior to each release.

NOTES FOR TABLE D 4.6.15-1

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

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

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

S_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

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

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

Y is the fractional radiochemical yield, when applicable,

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

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

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

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

NOTES FOR TABLE D 4.6.15-1

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

(c) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Radioactive Effluent Release Report.

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

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

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

CONTROLS	SURVEILLANCE REQUIREMENT
b. Gaseous (1) Dose Rate The dose rate due to radioactive materials released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following: (a) For noble gases: Less than or equal to 500 mrems/year to the whole body and less than or equal to 3000 mrems/year to the skin, and (b) For iodine-131, iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/year to any organ. With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limits(s).	b. Gaseous (1) Dose Rate The dose rate due to noble gases in gaseous effluents shall be determined to be within the limits of Control DLCO 3.6.15 in accordance with the methodology and parameters in Part II. The dose rate due to iodine-131, iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the limits of Control DLCO 3.6.15 in accordance with methodology and parameters in Part II by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table D 4.6.15-2.

CONTROLS	SURVEILLANCE REQUIREMENT
(2) Air Dose The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas beyond the site boundary shall be limited to the following: (a) During any calendar quarter: Less than or equal to 5 milliroentgen for gamma radiation and less than or equal to 10 mrads for beta radiation and, (b) During any calendar year: Less than or equal to 10 milliroentgen for gamma radiation and less than or equal to 20 mrads for beta radiation. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Control D 6.9.3, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.	(2) Air Dose Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined monthly in accordance with the methodology and parameters in Part II.

CONTROLS	SURVEILLANCE REQUIREMENT
(3) Tritium, Iodines and Particulates The dose to a member of the public from iodine-131, iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released, from each reactor unit, to areas beyond the site boundary shall be limited to the following: (a) During any calendar quarter: Less than or equal to 7.5 mrems to any organ and, (b) During any calendar year: Less than or equal to 15 mrems to any organ. With the calculated dose from the release of iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Control D 6.9.3, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.	(3) Tritium, Iodines and Particulates Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days shall be determined monthly in accordance with the methodology and parameters in Part II.

TABLE D 4.6.15-2
RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Surveillance Requirements

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit[a] of Detection (LLD) (μCi/ml)
A. Containment Purge[b]	Each Purge	Prior to each release	Principal Gamma Emitters[c]	1×10^{-4}
	Grab Sample	Each Purge	Principal Gamma Emitters[c]	1×10^{-4}
			H-3	1×10^{-6}
B. Stack	Once/Month[d]	Once/Month[d]	Principal Gamma Emitters[c]	1×10^{-4}
	Once/Month[h]	Once/Month	H-3	1×10^{-6}
C. Stack	Continuous[e]	Once/Week[f] Charcoal Sample	I-131	1×10^{-12}
	Continuous[e]	Once/Week[f] Particulate Sample	Principal Gamma Emitters[c]	1×10^{-11}
	Continuous[e]	Once/Month Composite Particulate Sample	Gross alpha, Sr-89, Sr-90	1×10^{-11}
	Continuous[e]	Noble Gas Monitor	Noble Gases, Gross Gamma or Principal Gamma Emitters[c]	1×10^{-5}[g]

NOTES FOR TABLE D 4.6.15-2

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

(b) Purge is defined in Section 1.0.

(c) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-135 and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, I-131 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Radioactive Effluent Release Report pursuant to Technical Specification 6.9.1.e and Control D 6.9.1.

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

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

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

(g) When the continuous Noble Gas Monitor is inoperable the LLD for noble gas gamma analysis shall be 1×10^{-4} μCi/cc.

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

CONTROLS	SURVEILLANCE REQUIREMENT
c. Deleted	c. Main Condenser The radioactivity rate of noble gases at the recombiner discharge shall be continuously monitored in accordance with Table D 3.6.14-2.
d. Uranium Fuel Cycle The annual (calendar year) dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.	d. Uranium Fuel Cycle Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Controls DSR 4.6.15.a.(2), DSR 4.6.15.b.(2) and DSR 4.6.15.b.(3) and in accordance with the methodology and parameters in Part II.

CONTROLS	SURVEILLANCE REQUIREMENT
With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Controls DLCO 3.6.15.a(2), DLCO 3.6.15.b(2) and DLCO 3.6.15.b(3), calculations shall be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above listed 40CFR190 limits have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Control D 6.9.3, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.2203(a)(4), shall include an analysis that estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report.	Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in Part II. This requirement is applicable only under conditions set forth in Control DLCO 3.6.15.d.

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

CONTROLS	SURVEILLANCE REQUIREMENT

DLCO 3.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS

Applicability:

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

Objective:

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

Specification:

a. Liquid

The liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected dose due to the liquid effluent, from each unit, to the Unrestricted Areas would exceed 0.06 mrem to the total body or 0.2 mrem to any organ for any batch.

b. Gaseous

(1) The Gaseous Radwaste Treatment System shall be operable. The Gaseous Radwaste Treatment System shall be used to reduce radioactive materials in gaseous waste prior to their discharge as necessary to meet the requirements of Control DLCO 3.6.15.

DSR 4.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS

Applicability:

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

Objective:

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

Specification:

a. Liquid

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

b. Gaseous

(1) Doses due to gaseous releases to areas at or beyond the site boundary shall be calculated in accordance with the methodology and parameters in Part II.

CONTROLS	
With gaseous radwaste from the main condenser air ejector system being discharged without treatment for more than 7 days, prepare and submit to the Commission within 30 days, pursuant to Control D 6.9.3, a Special Report that identifies the inoperable equipment and the reason for its inoperability, actions taken to restore the inoperable equipment to OPERABLE status, and a summary description of those actions taken to prevent a recurrence.	
(2) 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 due to gaseous effluent releases, from each unit, to areas beyond the site boundary would exceed 0.3 mrem to any organ of a member of the public. With radioactive gaseous waste being discharged without treatment and in excess of the above limit, complete a DER evaluation of the degraded condition within 30 days that identifies the inoperable equipment, the reason for the inoperability, and plans and schedule to restore the equipment to operable status.	(2) **NOTE:** Only required to be met when the Ventilation Exhaust Treatment System is not being fully utilized. Project the doses from the iodine and particulate releases from each unit to areas beyond the Site Boundary at least every 31 days.

CONTROLS	SURVEILLANCE REQUIREMENT
DLCO 3.6.18 MARK I CONTAINMENT	DSR 4.6.18 MARK I CONTAINMENT
Applicability: Applies to the venting/purging of the Mark I Containment.	Applicability: Applies to the surveillance requirement for venting and purging of the Mark I Containment when required to be vented/purged through the Emergency Ventilation System.
Objective: To assure that the Mark I Containment is vented/purged so that the limits of Controls DLCO 3.6.15.b(1) and DLCO 3.6.15.b(3) are met.	Objective: To verify that the Mark I Containment is vented through the Emergency Ventilation System when required.
Specification: The Mark I Containment drywell shall be vented/ purged through the Emergency Ventilation System unless Controls DLCO 3.6.15.b.(1) and DLCO 3.6.15.b.(3) can be met without use of the Emergency Ventilation System. If these requirements are not satisfied, suspend all venting/purging of the drywell.	Specification: The containment drywell shall be determined to be aligned for venting/purging through the Emergency Ventilation System within four hours prior to start of and at least once per 12 hours during venting/purging of the drywell.

CONTROLS	SURVEILLANCE REQUIREMENT
DLCO 3.6.19 LIQUID WASTE HOLDUP TANKS*	DSR 4.6.19 LIQUID WASTE HOLDUP TANKS
Applicability: Applies to the quantity of radioactive material that may be stored in an outdoor liquid waste holdup tank.	Applicability: Applies to the surveillance requirements for outdoor liquid waste holdup tanks.
Objective: To assure that the quantity of radioactive material stored in outdoor holdup tanks does not exceed a specified level.	Objective: To verify the quantity of radioactive material stored in an outdoor liquid waste holdup tank.
Specification: The quantity of radioactive material contained in an outdoor liquid waste tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases. With the quantity of radioactive material in any such tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank. Within 48 hours reduce the tank contents to within the limit and describe the events leading to this condition in the next Radioactive Effluent Release Report. *Tanks included in this Control are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.	Specification: The quantity of radioactive material contained in each of the tanks listed in Control DLCO 3.6.19 shall be determined to be within the limit of Control DLCO 3.6.19 by analyzing a representative sample of the tank's contents at least weekly when radioactive materials are being added to the tank.

CONTROLS	SURVEILLANCE REQUIREMENT
DLCO 3.6.20 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM	DSR 4.6.20 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM
Applicability: Applies to radiological samples of station environs.	Applicability: Applies to the periodic sampling and monitoring requirements of the radiological environmental monitoring program.
Objective: To evaluate the effects of station operations and radioactive effluent releases on the environs and to verify the effectiveness of the controls on radioactive material sources.	Objective: To ascertain what effect station operations and radioactive effluent releases have had upon the environment.
Specification: The radiological environmental monitoring program shall be conducted as specified in Table D 3.6.20-1. With the radiological environmental monitoring program not being conducted as specified in Table D 3.6.20-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. Deviations are permitted from the required sample schedule if samples are unobtainable due to hazardous conditions, seasonal unavailability, theft, uncooperative residents or to malfunction of automatic sampling equipment. In the event of the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.	Specification: The radiological environmental monitoring samples shall be collected pursuant to Table D 3.6.20-1 from the specific locations given in the table and figure(s) in Part II and shall be analyzed pursuant to the requirements of Table D 3.6.20-1 and the detection capabilities required by Table D 4.6.20-1.

CONTROLS	SURVEILLANCE REQUIREMENT

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

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

When radionuclides other than those in Table D 6.9.3-1 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Controls DLCO 3.6.15.a.(2), DLCO 3.6.15.b.(2) and DLCO 3.6.15.b.(3).

CONTROLS	SURVEILLANCE REQUIREMENT
This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report. With milk or fruit and/or vegetables no longer available at one or more of the sample locations specified in Table D 3.6.20-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for Part II reflecting the new location(s).	

TABLE D 3.6.20-1
OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency [a]	Type of Analysis and Frequency
Radioiodine & Particulates	Samples from 5 locations: 1) 3 Samples from off-site locations in different sectors of the highest calculated site average D/Q (based on all site licensed reactors) 2) 1 sample from the vicinity of an established year round community having the highest calculated site average D/Q (based on all site licensed reactors) 3) 1 sample from a control location 10-17 miles distant and in a least prevalent wind direction[d]	Continuous sampler operation with sample collection weekly or as required by dust loading, whichever is more frequent	Radioiodine Canisters analyze once/week for I-131. Particulate Samplers Gross beta radioactivity following filter change, [b] composite (by location) for gamma isotopic analysis[c] once per 3 months, (as a minimum)
Direct Radiation[e]	32 stations with two or more dosimeters to be placed as follows: an inner ring of stations in the general area of the site boundary and an outer ring in the 4 to 5 mile range from the site with a station in each land based sector.* The balance of the stations should be placed in special interest areas such as population centers, nearby residences, schools and in 2 or 3 areas to serve as control stations.	Once per 3 months	Gamma dose once per 3 months

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

TABLE D 3.6.20-1 (Cont)
OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples[a] and Locations	Sampling and Collection Frequency [a]	Type of Analysis and Frequency
WATERBORNE			
Surface[f]	1) 1 sample upstream	Composite sample over 1 month period[g]	Gamma isotopic analysis[c] once/month. Composite for once per 3 months tritium analysis.
	2) 1 sample from the site's downstream cooling water intake		
Sediment from Shoreline	1 sample from a downstream area with existing or potential recreational value	Twice per year	Gamma isotopic analysis[c]
INGESTION			
Milk	1) Samples from milk sampling locations in 3 locations within 3.5 miles distance having the highest calculated site average D/Q. If there are none, then 1 sample from milking animals in each of 3 areas 3.5-5.0 miles distant having the highest calculated site average D/Q (based on all site licensed reactors)	Twice per month, April-December (samples will be collected in January-March if I-131 is detected in November and December of the preceding year)	Gamma isotopic[c] and I-131 analysis twice per month when animals are on pasture (April-December); once/month at other times (January-March) if required
	2) 1 sample from a milk sampling location at a control location (9-20 miles distant and in a least prevalent wind direction)[d]		

TABLE D 3.6.20-1 (Cont)
OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample		Number of Samples[a] and Locations	Sampling and Collection Frequency [a]	Type of Analysis and Frequency
Fish	1)	1 sample each of two commercially or recreationally important species in the vicinity of a plant discharge area.[h]	Twice per year	Gamma isotopic analysis[c] on edible portions twice per year
	2)	1 sample each of the same species from an area at least 5 miles distant from the site.[d]		
Food Products	1)	Samples of three different kinds of broad leaf vegetation (such as vegetables) grown nearest to each of two different off-site locations of highest calculated site average D/Q (based on all licensed site reactors).	Once per year during harvest season	Gamma isotopic[c] analysis of edible portions (isotopic to include I-131 or a separate I-131 analysis may be performed) once during the harvest season
	2)	Once sample of each of the similar broad leaf vegetation grown at least 9.3-20 miles distant in a least prevalent wind direction.		

NOTES FOR TABLE D 3.6.20-1

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

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

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

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

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

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

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

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

TABLE D 4.6.20-1
DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a,b]
LOWER LIMIT OF DETECTION LLD[c]
Surveillance Requirement

Analysis	Water[c] (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58, Co-60	15		130			
Zn-65	30		260			
Zr-95, Nb-95	15					
I-131	1**	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba/La-140	15			15		

* If no drinking water pathway exists, a value of 3000 pCi/liter may be used.

** If no drinking water pathway exists, a value of 15 pCi/liter may be used.

NOTES FOR TABLE D 4.6.20-1

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

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

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

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

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

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

S_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

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

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, where applicable,

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

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

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

NOTES FOR TABLE D 4.6.20-1

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

CONTROLS	SURVEILLANCE REQUIREMENT
DLCO 3.6.21 INTERLABORATORY COMPARISON PROGRAM	DSR 4.6.21 INTERLABORATORY COMPARISON PROGRAM
Applicability: Applies to participation in an interlaboratory comparison program on environmental sample analysis.	Applicability: Applies to testing the validity of measurements on environmental samples.
Objective: To ensure the accuracy of measurements of radioactive material in environmental samples.	Objective: To verify the accuracy of measurements on radioactive material in environmental samples.
Specification: Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission. Participation in this program shall include media for which environmental samples are routinely collected and for which intercomparison samples are available. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.	Specification: The Interlaboratory Comparison Program shall be described in Part II. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report. Participants in the EPA Cross Check Program may provide the EPA program code designation in lieu of providing results.

CONTROLS	SURVEILLANCE REQUIREMENT
DLCO 3.6.22 LAND USE CENSUS	DSR 4.6.22 LAND USE CENSUS
Applicability:	Applicability:
Applies to the performance of a land use census in the vicinity of the Nine Mile Point Nuclear Facility.	Applies to assuring that current land use is known.
Objective:	Objective:
To determine the utilization of land within a distance of three miles from the Facility.	To verify the appropriateness of the environmental surveillance program.
Specification:	Specification:
A land use census shall be conducted and shall identify within a distance of three miles the location in each of the 16 meteorological sectors the nearest residence and within a distance of three miles the location in each of the 16 meteorological sectors of all milk animals. In lieu of a garden census, specifications for vegetation sampling in Table D 3.6.20-1 shall be followed, including analysis of appropriate controls.	The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as conducting a door-to-door survey, aerial survey or consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report.
With a land use census identifying a milk animal location(s) that represents a calculated D/Q value greater than the D/Q value currently being used in Control DSR 4.6.15.b.(3), identify the new location(s) in the next Radioactive Effluent Release Report.	

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

PART I – RADIOLOGICAL EFFLUENT CONTROLS

Bases

BASES FOR DLCO 3.6.14 and DSR 4.6.14 RADIOACTIVE EFFLUENT INSTRUMENTATION

The radioactive liquid and gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid and gaseous effluents during actual or potential releases of liquid and gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in Part II to ensure that the alarm/trip will occur prior to exceeding the limits as described in Technical Specification 6.18. Historically, the maximum allowable deviation of ±50% of setpoint for High Radiation-Offgas Line (Reference FSAR, Appendix D) and +100% and –50% of setpoint for High Radiation-Emergency Cooling System Vent had negligible effect on the initiations of these systems. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10CFR Part 50. The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to unrestricted areas.

BASES FOR DLCO 3.6.15 AND DSR 4.6.15 RADIOACTIVE EFFLUENTS

Liquid Concentration

This control 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 10CFR Part 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, 10CFR Part 50, to a member of the public and (2) the limits of 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.

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

BASES FOR DLCO 3.6.15 AND DSR 4.6.15 RADIOACTIVE EFFLUENTS

Liquid Dose

This control is provided to implement the requirements of Section II.A, III.A and IV.A of Appendix I, 10CFR Part 50. The controls expressed as quarter and annual limits are set at those values found in Section II.A. of Appendix I, in accordance with Section IV.A. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to unrestricted areas will be kept "as low as is reasonably achievable." There are no drinking water supplies that can be potentially affected by plant operations. The dose calculation methodology and parameters in Part II implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculation procedures based on models and data, such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The equations specified in Part II for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

BASES FOR DLCO 3.6.15 AND DSR 4.6.15 RADIOACTIVE EFFLUENTS

Gaseous Dose Rate

This control is provided to ensure that the dose at any time at or beyond the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10CFR Part 20 to unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10CFR Part 20, Appendix B, Table 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 10CFR Part 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. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a member of the public at or beyond the site boundary to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

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

BASES FOR DLCO 3.6.15 AND DSR 4.6.15 RADIOACTIVE EFFLUENTS

Dose - Noble Gases

This control is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10CFR Part 50. The controls expressed as quarter and annual limits are set at those values found in Section II.B of Appendix I in accordance with the guidance of Section IV.A. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV-A of Appendix I to assure that the releases of radioactive material in gaseous effluents to unrestricted areas will be kept "as low as is reasonably achievable." The Surveillance Requirement implements the requirements in Section III.A of Appendix I that conform with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in Part II for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I, "Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977.

The Offsite Dose Calculation Manual Part II equations provided to determine the air doses beyond the site boundary are based upon the historical average atmospheric conditions.

BASES FOR DLCO 3.6.15 AND DSR 4.6.15 RADIOACTIVE EFFLUENTS

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

This control is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10CFR Part 50. The controls expressed as quarter and annual limits are set at those values found in Section II.C of Appendix I in accordance with the guidance of Section IV.A. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to unrestricted areas will be kept "as low as is reasonably achievable." The Part II calculational methods specified in the Surveillance Requirement implements the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The Part II calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas beyond the site boundary. The pathways that were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man and 4) deposition on the ground with subsequent exposure of man.

BASES FOR DLCO 3.6.15 AND DSR 4.6.15 RADIOACTIVE EFFLUENTS

Total Dose - Uranium Fuel Cycle

This control is provided to meet the dose limitations of 40CFR Part 190 that have been incorporated into 10CFR Part 20 by 46FR 18525. The control requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to a member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contribution from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40CFR Part 190, the Special Report with a request for variance (provided the release conditions resulting in violation of 40CFR Part 190 have not already been corrected), in accordance with the provisions of 40CFR Part 190.11 and 10 CFR Part 20.2203(a)(4) is considered to be a timely request and fulfills the requirements of 40CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40CFR Part 190 and does not apply in any way to the other requirements for dose limitation of 10CFR Part 20, as addressed in Controls DLCO 3.6.15.a.(1) and DLCO 3.6.15.b.(1). An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

BASES FOR DLCO 3.6.16 AND DSR 4.6.16 RADIOACTIVE EFFLUENT TREATMENT SYSTEMS

Liquid Radwaste Treatment System

The requirement that the appropriate portions of this system be used provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This control implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. Projected doses are calculated on a batch rather than every 31 days due to the low frequency of releases.

Gaseous Effluent Treatment Systems

The operability of the Gaseous Radwaste Treatment System and the Ventilation Exhaust Treatment System ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that appropriate portions of these systems 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 control implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR 50 and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The control governing the use of appropriate portions of the Gaseous Radwaste Treatment System is based on time without treatment rather than dose, due to the wide variability in effluent with changing power conditions. Since the capability exists to operate within specification without use of the Gaseous Radwaste Treatment System, it is conceivable that due to unforeseen circumstances, limited operation without the system may be made sometime during the life of the plant. The control governing the use of appropriate portions of the Ventilation Exhaust Treatment System was specified as a suitable fraction of the dose design objectives set forth in II.C of Appendix I, 10CFR Part 50, for gaseous effluents.

BASES FOR DLCO 3.6.18 AND DSR 4.6.18 MARK I CONTAINMENT

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

BASES FOR DLCO 3.6.19 AND DSR 4.6.19 LIQUID HOLDUP TANKS

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

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than ten times the concentrations of 10CFR Part 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

BASES FOR DLCO 3.6.20 AND DSR 4.6.20 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

The radiological environmental monitoring program required by this control provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of members of the public resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

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

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

BASES FOR DLCO 3.6.21 AND DSR 4.6.21 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring for the purposes of Section IV.B.2 of Appendix I to 10CFR Part 50.

BASES FOR DLCO 3.6.22 AND DSR 4.6.22 LAND USE CENSUS

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

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

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