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Constellation Nuclear

Calvert Cliffs
Nuclear Power Plant

A Member of the
Constellation Energy Group

July 19, 2001

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Independent Spent Fuel Storage Installation Docket No. 72-8
2000 Radioactive Effluent Release Report and Dose Assessment

REFERENCES:
(a) Calvert Cliffs Unit Nos. 1 and 2 Technical Specification 5.6.3
(b) Independent Spent Fuel Storage Installation Technical Specification 6.3

As required by References (a) and (b), Enclosures (1) and (2) are provided. Meteorological data is being kept in our onsite file and will be available upon request.

Should you have questions regarding this matter, we will be pleased to discuss them with you.

Very truly yours,

Chris Earls

C. E. Earls
General Supervisor - Chemistry

CEE/MJY/bjd

Enclosures: (1) Calvert Cliffs Nuclear Power Plant Effluent and Waste Disposal 2000 Annual Report, Supplemental Information
Appendix (A) Calvert Cliffs Nuclear Power Plant Effluent and Waste Disposal Semi-Annual Report 2000
(2) Offsite Dose Calculation Manual for Calvert Cliffs Nuclear Power Plant, Revision 4

cc: R. I. McLean, DNR

(Without Enclosures)
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ENCLOSURE (1)

CALVERT CLIFFS NUCLEAR POWER PLANT

RADIOACTIVE EFFLUENT RELEASE REPORT

AND DOSE REPORT

Facility - Calvert Cliffs Nuclear Power Plant

Licensee – Calvert Cliffs Nuclear Power Plant, Inc.

I. REGULATORY LIMITS

A. Fission and Activation Gases

1. The instantaneous release rate of noble gases in gaseous effluents shall not result in a site boundary dose rate greater than 500 mRem/year to the whole body or greater than 3000 mRem/year to the skin (Offsite Dose Calculation Manual (ODCM) 3.11.2.1).

2. Gaseous Radwaste Treatment System and the Ventilation Exhaust Treatment System shall be used to reduce gaseous emissions when the calculated gamma-air dose due to gaseous effluents exceeds 1.20 mRad or the calculated beta-air dose due to gaseous effluents exceeds 2.40 mRad at the site boundary in a 92 day period (ODCM 3.11.2.4).

3. The air dose at the site boundary due to noble gases released in gaseous effluents shall not exceed (ODCM 3.11.2.2):

   10 mRad/qtr, gamma-air
   20 mRad/qtr, beta-air
   20 mRad/year, gamma-air
   40 mRad/year, beta-air

4. All of the above parameters are calculated according to the methodology specified in the ODCM.

B. Iodines and Particulates with Half Lives Greater than Eight Days

1. The instantaneous release rate of iodines and particulates in gaseous effluents shall not result in a site boundary dose in excess of 1500 mRem/year to any organ (ODCM 3.11.2.1).

2. The Gaseous Radwaste Treatment System and the Ventilation Exhaust Treatment System shall be used to reduce radioactive materials in gaseous effluents when calculated doses exceed 1.8 mRem to any organ in a 92 day period at or beyond the site boundary (ODCM 3.11.2.4).

3. The dose to a member of the public at or beyond the site boundary from iodine-131 and particulates with half lives greater than eight days in gaseous effluents shall not exceed (ODCM 3.11.2.3):

   15 mRem/qtr, any organ
   30 mRem/year, any organ
   less than 0.1% of the above limits as a result of burning contaminated oil.

4. All of the above parameters are calculated according to the methodology specified in the ODCM.

C. Liquid Effluents

1. The concentrations of radionuclides in liquid effluents from the plant shall not exceed the values specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for unrestricted areas (ODCM 3.11.1.1).

2. The liquid radwaste treatment system shall be used to reduce the concentration of radionuclides in liquid effluents from the plant when the calculated dose to unrestricted areas exceeds 0.36 mRem to the whole body, or 1.20 mRem to any organ in a 92 day period (ODCM 3.11.1.3).

3. The dose to a member of the public in unrestricted areas shall not exceed (ODCM 3.11.1.2):

   3 mRem/qtr, total body
   10 mRem/qtr, any organ
   6 mRem/year, total body
   20 mRem/year, any organ

4. All of the liquid dose parameters are calculated according to the methodology specified in the ODCM.

II. MAXIMUM PERMISSIBLE CONCENTRATIONS

A. Fission and Activation Gases

Prior to the batch release of gaseous effluents, a sample of the source is collected and analyzed by gamma spectroscopy for the principal gamma emitting radionuclides. The identified radionuclide concentrations are evaluated and an acceptable release rate is determined to ensure that the dose rate limits of ODCM 3.11.2.1 are not exceeded.

B. Iodines and Particulates with Half Lives Greater than Eight Days

Compliance with the dose rate limitations for iodines and particulates is demonstrated by analysis of the charcoal and particulate samples of the station main vents. The charcoal samples are analyzed by gamma spectroscopy for quantification of any release of radioiodines. The particulate samples are analyzed by gamma spectroscopy for quantification of particulate radioactive material. All of the above parameters are calculated according to the methodology specified in the ODCM.

C. Liquid Effluents

The Maximum Permissible Concentrations (MPCs) used for radioactive materials released in liquid effluents are in accordance with ODCM 3.11.1.1 and the values from 10 CFR Part 20, Appendix B, Table II, Column 2 including applicable table notes. In all cases, the more restrictive (lower) MPC found for each radionuclide is used regardless of solubility.

III. TECHNICAL SPECIFICATION REPORTING REQUIREMENTS

A. Calvert Cliffs Nuclear Power Plant (CCNPP), Technical Specification 5.6.3

1. 2000 Dose Assessment Summary

	Actual Value	Percent of ODCM limit	ODCM Limit
Liquid Waste:			
Maximum Annual Organ Dose (mRem) [1]	0.21	1.06	20
Maximum Whole Body Dose (mRem) [1]	0.02	0.30	6
Gaseous Waste:			
Noble Gases:			
Maximum Quarterly Gamma Air Dose (mRad)	0.001	0.014	10
Maximum Quarterly Beta Air Dose (mRad)	0.007	0.035	20
Iodines and Particulates:			
Maximum Annual Organ Dose Rate (mRem) [2]	0.012	<0.001	1500
Maximum Annual Organ Dose (mRem) [2]	0.076	0.254	30

1 The controlling pathway was the fish and shellfish pathway with adult as the controlling age group and the gastrointestinal tract representing the organ with the highest calculated dose during the calendar year of 2000.

2 The controlling pathway was the child-infant-thyroid pathway representing the organ with the highest calculated dose during the calendar year of 2000. There is currently no known milk pathway in existence.

2. 40 CFR 190 Total Dose Compliance

Based upon the calendar year 2000 and the ODCM calculations, the maximum exposed individual would receive less than 1% of the allowable dose. During the calendar year 2000, there were no on-site sources of direct radiation that would have contributed to a significant or measurable off-site dose. The direct radiation contribution is measured by both on-site and off-site thermoluminescent dosimeters (TLDs). The results of these measurements did not indicate any statistical increase in the off-site radiation doses attributable to on-site sources. Therefore, no increase in the calculated offsite dose is attributed to the direct exposure from on-site sources. A more detailed evaluation will be reported in the Annual Radiological Environmental Operating Report.

3. Solid Waste Report Requirements

During 2000, the types of radioactive solid waste shipped from Calvert Cliffs were dry compressible waste, filters, and dewatered resins which were shipped in High Integrity Containers (HICs) within NRC approved casks, Sealand containers, and steel boxes. Appendix A provides a detailed breakdown of the waste shipments for 2000 per the categories specified in Technical Specification 5.6.3. At CCNPP, methods of waste and materials segregation are used to reduce the volume of solid waste shipped offsite for processing, volume reduction and burial.

4. Offsite Dose Calculation Manual (ODCM) and Process Control Program (PCP) Changes

   The ODCM was revised on February 7, 2000, a copy is attached. There were no changes to the PCP.

B. Radioactive Gaseous Effluent Monitoring Instrumentation

No effluent pathway was unmonitored for more than 30 consecutive days in 2000.

C. Independent Spent Fuel Storage Installation (ISFSI), ISFSI Technical Specification 6.1

12 casks of spent fuel were transferred to the ISFSI during 2000. No quantity of radionuclides was released to the environment during the ISFSI operation in 2000. Additional information regarding the ISFSI radiation-monitoring program is included in the Annual Radiological Environmental Operation Report.

IV. AVERAGE ENERGY

Not Applicable.

V. MEASUREMENTS AND APPROXIMATIONS AND TOTAL RADIOACTIVITY

A. Fission and Activation Gases

1. Batch Releases

   Prior to each batch release of gas from a pressurized waste gas decay tank or containment, a sample is collected and analyzed by gamma spectroscopy using a Germanium (Ge) detector for the principal gamma emitting noble gas radionuclides. The total activity released is based on the pressure/volume relationship (gas laws).

2. Continuous Releases

   A gas sample is collected at least weekly from the main vents and analyzed by gamma spectroscopy using a Ge detector for the principal gamma emitting noble gas radionuclides. The total activity released for the week is based on the total sample activity decay corrected to the sample time multiplied by the main vent flow for the week.

   Prior to and after each containment purge, a gas sample is collected and analyzed by gamma spectroscopy using a Ge detector for the principal gamma emitting noble gas radionuclides. The total activity released is based on containment volume and purge rate.

   A monthly composite sample is collected from the main vents and analyzed by liquid scintillation for tritium. The total tritium release for the month is based on the sample analysis and the main vent flow.

B. Iodine and Particulates

1. Batch Releases

The total activities of radioiodines and particulates released from pressurized waste gas decay tanks, containment purges, and containment vents are accounted for by the continuous samplers on the main vent.

2. Continuous Releases

During the release of gas from the main vents, samples of iodines and particulates are collected using a charcoal and particulate filter, respectively. The filters are removed weekly and are analyzed by gamma spectroscopy using a Ge detector for significant gamma emitting radionuclides. The total activity released for the week is based on the total sample activity decay corrected to the midpoint of the sample period multiplied by the main vent flow for the week. The weekly particulate filters are then composited to form monthly and quarterly composites for the gross alpha and strontium 89 and 90 analyses.

C. Liquid Effluents

1. Batch Releases

Prior to the release of liquid from a waste tank, a sample is collected and analyzed by gamma spectroscopy for the principal gamma emitting radionuclides. To demonstrate compliance with the requirements addressed in Section I.C.1 above, the measured radionuclide concentrations are compared with the allowable MPCs; dilution in the discharge conduit is considered, and an allowable release rate is verified.

The total activity released in each batch is determined by multiplying the volume released by the concentration of each radionuclide. The actual volume released is based on the difference in tank levels prior to and after the release. A proportional composite sample is also withdrawn from each release, this is used in turn to prepare monthly tritium and quarterly gross alpha, iron 55, and strontium 89 and 90 samples for analysis.

2. Continuous Releases

To account for activity from continuous releases, a sample is collected and analyzed by gamma spectroscopy for the principal gamma emitting radionuclides. The measured radionuclide concentrations are compared with the allowable MPC concentrations in the discharge conduit, and an allowable release rate is verified.

When steam generator blowdown is discharged to the circulating water conduits, it is sampled at a minimum of three times per week and these samples are used in turn to prepare a weekly blowdown composite sample based on each day's blowdown. The weekly composite sample is analyzed by gamma spectroscopy for the principal gamma emitting radionuclides. These results are multiplied by the actual quantity of blowdown to determine the total activity released. The weekly composite is also used to prepare monthly composites for tritium analysis.

During periods of primary-to-secondary leakage, the secondary system becomes contaminated and subsequently, contaminates the turbine building sumps. The low-level activity water (predominantly tritium) contained in the turbine building sumps is released directly to the Chesapeake Bay. This water is sampled weekly and composited. The composite sample is analyzed at least monthly for tritium and principal gamma emitting radionuclides. The results are multiplied by the actual quantity of liquid released to determine the total activity released.

D. Estimation of Total Error

Total error for all releases was estimated using, as a minimum, the random counting error associated with typical releases. In addition to this random error, the following systematic errors were also examined:

1. Liquid
   a. Error in volume of liquid released prior to dilution during batch releases.
   b. Error in volume of liquid released via steam generator blowdown.
   c. Error in amount of dilution water used during the reporting period.

2. Gases
   a. Error in main vent release flow.
   b. Error in sample flow rate.
   c. Error in containment purge release flow.
   d. Error in gas decay tank pressure.

Where errors could be estimated they are usually considered additive.

VI. BATCH RELEASES

	2000			
	1ST QUARTER	2ND QUARTER	3RD QUARTER	4TH QUARTER
A. Liquid				
1. Number of batch releases	2.20E+01	1.70E+01	1.40E+01	1.30E+01
2. Total time period for batch releases (min)	8.32E+03	6.62E+03	9.37E+04	4.14E+03
3. Maximum time period for a batch release (min)	7.40E+02	6.48E+02	4.46E+04	6.54E+02
4. Average time period for batch releases (min)	3.78E+02	3.89E+02	6.69E+03	3.18E+02
5. Minimum time period for a batch release (min)	2.10E+01	2.60E+01	2.80E+01	2.70E+01
6. Average stream flow during periods of effluent into a flowing stream (liters/min of dilution water)	7.95E+06	8.27E+06	9.05E+06	9.01E+06
B. Gaseous				
1. Number of batch releases	1.00E+01	1.00E+01	0.00E+00	2.00E+00
2. Total time period for batch releases (min)	6.79E+03	3.61E+03	0.00E+00	2.64E+02
3. Maximum time period for a batch release (min)	3.27E+03	2.44E+03	0.00E+00	1.72E+02
4. Average time period for batch release (min)	6.79E+02	3.61E+02	0.00E+00	1.32E+02
5. Minimum time period for a batch release (min)	2.48E+00	1.20E+01	0. 00E+00	9.20E+01

VII. ABNORMAL RELEASES

	2000			
	1ST QUARTER	2ND QUARTER	3RD QUARTER	4TH QUARTER
A. Liquid				
1. Number of releases	- 0 -	- 0 -	- 0 -	- 0 -
2. Total activity released (Curies)	- 0 -	- 0 -	- 0 -	- 0 -
B. Gaseous				
1. Number of releases	- 0 -	- 0 -	- 0 -	- 0 -
2. Total activity releases (Curies)	- 0 -	- 0 -	- 0 -	- 0 -

TABLE 1A - REG GUIDE 1.21
GASEOUS EFFLUENTS - SUMMATION OF ALL RELEASES

A. FISSION AND ACTIVATION GASES	UNITS	1ST QUARTER	2ND QUARTER	EST. TOTAL ERROR, %
1. Total Release	Ci	7.24E+01	1.69E+01	±1.20E+01
2. Average release rate for period	μCi/sec	9.21E+00	2.15E+00	
3. Percent of Tech. Spec. limit (1)	%	9.40E-04	2.34E-04	
4. Percent of Tech. Spec. limit (2)	%	6.63E-04	2.08E-04	
5. Percent of Tech. Spec. limit (3)	%	1.38E-02	3.26E-03	
6. Percent of Tech. Spec. limit (4)	%	6.89E-03	1.63E-03	
7. Percent of Tech. Spec. limit (5)	%	3.48E-02	9.87E-03	
8. Percent of Tech. Spec. limit (6)	%	1.74E-02	4.94E-03	
B. IODINES				
1. Total Iodine - 131	Ci	1.39E-03	9.46E-04	±6.50E+00
2. Average release rate for period	μCi/sec	1.77E-04	1.20E-04	
3. Percent of Tech. Spec. limit (7)	%	4.20E-04	2.86E-04	
4. Percent of Tech. Spec. limit (8)	%	2.66E-01	1.81E-01	
5. Percent of Tech. Spec. limit (9)	%	1.33E-01	9.06E-02	
C. PARTICULATES				
1. Particulates with half lives greater than 8 days	Ci	(10)	(10)	N/A
2. Average release rate for period	μCi/sec	(10)	(10)	
3. Percent of Tech. Spec. limit (7)	%	(10)	(10)	
4. Percent of Tech. Spec. limit (8)	%	(10)	(10)	
5. Percent of Tech. Spec. limit (9)	%	(10)	(10)	
6. Gross alpha radioactivity	Ci	(10)	(10)	N/A

A. FISSION AND ACTIVATION GASES	UNITS	3RD QUARTER	4TH QUARTER	EST. TOTAL ERROR, %
1. Total Release	Ci	5.21E+00	1.27E+01	±1.20E+01
2. Average release rate for period	μCi/sec	6.55E-01	1.60E+00	
3. Percent of Tech. Spec. limit (1)	%	1.22E-04	3.73E-04	
4. Percent of Tech. Spec. limit (2)	%	4.66E-05	1.53E-04	
5. Percent of Tech. Spec. limit (3)	%	1.76E-03	5.26E-03	
6. Percent of Tech. Spec. limit (4)	%	8.81E-04	2.63E-03	
7. Percent of Tech. Spec. limit (5)	%	2.12E-03	6.03E-03	
8. Percent of Tech. Spec. limit (6)	%	1.06E-03	3.01E-03	
B. IODINES				
1. Total Iodine - 131	Ci	1.91E-04	1.27E-04	±6.50E+00
2. Average release rate for period	μCi/sec	2.40E-05	1.60E-05	
3. Percent of Tech. Spec. limit (7)	%	5.71E-05	3.80E-05	
4. Percent of Tech. Spec. limit (8)	%	3.66E-02	2.43E-02	
5. Percent of Tech. Spec. limit (9)	%	1.83E-02	1.22E-02	
C. PARTICULATES				
1. Particulates with half lives greater than 8 days	Ci	(10)	(10)	N/A
2. Average release rate for period	μCi/sec	(10)	(10)	
3. Percent of Tech. Spec. limit (7)	%	(10)	(10)	
4. Percent of Tech. Spec. limit (8)	%	(10)	(10)	
5. Percent of Tech. Spec. limit (9)	%	(10)	(10)	
6. Gross alpha radioactivity	Ci	(10)	(10)	N/A

TABLE 1A - REG GUIDE 1.21
GASEOUS EFFLUENTS - SUMMATION OF ALL RELEASES

D. TRITIUM	UNITS	1ST QUARTER	2ND QUARTER	EST. TOTAL ERROR, %
1. Total Release	Ci	4.44E+00	2.43E+00	±1.32E+01
2. Average release rate for period	μCi/sec	5.65E-01	3.09E-01	

D. TRITIUM	UNITS	3RD QUARTER	4TH QUARTER	EST. TOTAL ERROR, %
1. Total Release	Ci	3.54E+00	1.70E+00	±1.32E+01
2. Average release rate for period	μCi/sec	4.45E-01	2.14E-01	

NOTES TO TABLE 1A

(1) Percent of I.A.1 whole body dose rate limit (500 mRem/year)

(2) Percent of I.A.1 skin dose rate limit (3000 mRem/year)

(3) Percent of I.A.3 gamma quarterly dose limit (10 mRad)

(4) Percent of I.A.3 gamma yearly dose limit (20 mRad)

(5) Percent of I.A.3 beta quarterly dose limit (20 mRad)

(6) Percent of I.A.3 beta yearly dose limit (40 mRad)

(7) Percent of I.B.1 organ dose rate limit (1500 mRem/year)

(8) Percent of I.B.3 quarterly organ dose limit (15 mRem)

(9) Percent of I.B.3 yearly organ dose limit (30 mRem)

(10) Less than minimum detectable activity which meets the lower limit of detection (LLD) requirements of ODCM Surveillance Requirement 4.11.2.1.2.

TABLE 1C - REG GUIDE 1.21
GASEOUS EFFLUENTS - GROUND LEVEL RELEASES

			CONTINUOUS MODE		BATCH MODE	
		UNITS	1ST QUARTER	2ND QUARTER	1ST QUARTER	2ND QUARTER
1. FISSION AND ACTIVATION GASES						
Argon	-41	Ci	(2)	(2)	7.84E-04	2.58E-06
Krypton	-85	Ci	1.25E+01	(2)	1.15E+01	9.72E+00
Krypton	-85m	Ci	(2)	(2)	3.01E-05	(2)
Krypton	-87	Ci	(2)	(2)	7.76E-06	(2)
Krypton	-88	Ci	(2)	(2)	2.02E-05	(2)
Xenon	-131m	Ci	3.09E-01	(2)	9.11E-02	8.98E-02
Xenon	-133	Ci	4.52E+01	5.15E+00	1.25E+00	7.00E-01
Xenon	-133m	Ci	8.30E-02	(2)	1.78E-03	2.27E-03
Xenon	-135	Ci	1.48E+00	1.26E+00	1.39E-03	3.98E-03
Xenon	-135m	Ci	(2)	(2)	(2)	(2)
Xenon	-138	Ci	(2)	(2)	(2)	(2)
Total for Period		Ci	5.96E+01	6.41E+00	1.28E+01	1.05E+01
2. HALOGENS						
Iodine	-131	Ci	1.39E-03	9.46E-04	(1)	(1)
Iodine	-133	Ci	8.19E-04	1.65E-03	(1)	(1)
Bromine	-82	Ci	(2)	(2)	(1)	(1)
Total for Period		Ci	2.21E-03	2.60E-03	(1)	(1)

			CONTINUOUS MODE		BATCH MODE	
		UNITS	3RD QUARTER	4TH QUARTER	3RD QUARTER	4TH QUARTER
1. FISSION AND ACTIVATION GASES						
Argon	-41	Ci	(2)	(2)	(2)	(2)
Krypton	-85	Ci	(2)	(2)	(2)	1.00E+00
Krypton	-85m	Ci	(2)	(2)	(2)	(2)
Krypton	-87	Ci	(2)	(2)	(2)	(2)
Krypton	-88	Ci	(2)	(2)	(2)	(2)
Xenon	-131m	Ci	(2)	(2)	(2)	(2)
Xenon	-133	Ci	4.77E+00	9.55E+00	(2)	(2)
Xenon	-133m	Ci	(2)	(2)	(2)	(2)
Xenon	-135	Ci	4.39E-01	2.16E+00	(2)	(2)
Xenon	-135m	Ci	(2)	(2)	(2)	(2)
Xenon	-138	Ci	(2)	(2)	(2)	(2)
Total for Period		Ci	5.21E+00	1.17E+01	0.00E+00	1.00E+00
2. HALOGENS						
Iodine	-131	Ci	1.91E-04	1.27E-04	(1)	(1)
Iodine	-133	Ci	1.52E-03	5.44E-04	(1)	(1)
Bromine	-82	Ci	(2)	(2)	(1)	(1)
Total For Period		Ci	1.71E-03	6.71E-04	(1)	(1)

TABLE 1C - REG GUIDE 1.21
GASEOUS EFFLUENTS - GROUND LEVEL RELEASES

	UNITS	CONTINUOUS MODE 1ST QUARTER	CONTINUOUS MODE 2ND QUARTER	BATCH MODE 1ST QUARTER	BATCH MODE 2ND QUARTER
3. PARTICULATES					
Manganese -54	Ci	(2)	(2)	(1)	(1)
Iron -59	Ci	(2)	(2)	(1)	(1)
Cobalt -58	Ci	(2)	(2)	(1)	(1)
Cobalt -60	Ci	(2)	(2)	(1)	(1)
Zinc -65	Ci	(2)	(2)	(1)	(1)
Strontium -89	Ci	(2)	(2)	(1)	(1)
Strontium -90	Ci	(2)	(2)	(1)	(1)
Molybdenum -99	Ci	(2)	(2)	(1)	(1)
Cesium -134	Ci	(2)	(2)	(1)	(1)
Cesium -137	Ci	(2)	(2)	(1)	(1)
Cerium -141	Ci	(2)	(2)	(1)	(1)
Cerium -144	Ci	(2)	(2)	(1)	(1)
Gross Alpha Radioactivity	Ci	(2)	(2)	(1)	(1)
Total For Period	Ci	(2)	(2)	(1)	(1)

	UNITS	CONTINUOUS MODE 3RD QUARTER	CONTINUOUS MODE 4TH QUARTER	BATCH MODE 3RD QUARTER	BATCH MODE 4TH QUARTER
3. PARTICULATES					
Manganese -54	Ci	(2)	(2)	(1)	(1)
Iron -59	Ci	(2)	(2)	(1)	(1)
Cobalt -58	Ci	(2)	(2)	(1)	(1)
Cobalt -60	Ci	(2)	(2)	(1)	(1)
Zinc -65	Ci	(2)	(2)	(1)	(1)
Strontium -89	Ci	(2)	(2)	(1)	(1)
Strontium -90	Ci	(2)	(2)	(1)	(1)
Molybdenum -99	Ci	(2)	(2)	(1)	(1)
Cesium -134	Ci	(2)	(2)	(1)	(1)
Cesium -137	Ci	(2)	(2)	(1)	(1)
Cerium -141	Ci	(2)	(2)	(1)	(1)
Cerium -144	Ci	(2)	(2)	(1)	(1)
Gross Alpha Radioactivity	Ci	(2)	(2)	(1)	(1)
Total For Period	Ci	(2)	(2)	(1)	(1)

NOTES TO TABLE 1C

(1) Iodines and particulates in batch releases are accounted for with the main vent continuous samplers when the release is made through the plant main vent.

(2) Less than minimum detectable activity which meets the LLD requirements of ODCM Surveillance Requirement 4.11.2.1.2.

TABLE 2A - REG GUIDE 1.21
LIQUID EFFLUENTS - SUMMATION OF ALL RELEASES

	UNITS	1ST QUARTER	2ND QUARTER	EST. TOTAL ERROR, %
A. FISSION AND ACTIVATION GASES				
1. Total Release (not including tritium, gases, alpha)	Ci	3.03E-02	1.61E-01	±1.03E+01
2. Average diluted concentration during period	µCi/ml	2.91E-11	1.48E-10	
3. Percent of Tech. Spec. limit (1)	%	1.97E-01	5.59E-01	
4. Percent of Tech. Spec. limit (2)	%	9.87E-02	2.80E-01	
5. Percent of Tech. Spec. limit (3)	%	3.29E-02	1.37E-01	
6. Percent of Tech. Spec. limit (4)	%	1.65E-02	6.87E-02	
B. TRITIUM				
1. Total Release	Ci	4.31E+02	9.83E+01	
2. Average diluted concentration during period	µCi/ml	4.14E-07	9.07E-08	
3. Percent of applicable limit (5)	%	1.38E-02	3.02E-03	
C. DISSOLVED AND ENTRAINED GASES				
1. Total Release	Ci	1.89E-01	1.54E-03	
2. Average diluted concentration during period	µCi/ml	1.81E-10	1.42E-12	

	UNITS	3RD QUARTER	4TH QUARTER	EST. TOTAL ERROR, %
A. FISSION AND ACTIVATION GASES				
1. Total Release (not including tritium, gases, alpha)	Ci	9.56E-02	2.52E-02	±1.03E+01
2. Average diluted concentration during period	µCi/ml	7.98E-11	2.11E-11	
3. Percent of Tech. Spec. limit (1)	%	1.14E+00	2.15E-01	
4. Percent of Tech. Spec. limit (2)	%	5.71E-01	1.07E-01	
5. Percent of Tech. Spec. limit (3)	%	3.07E-01	1.13E-01	
6. Percent of Tech. Spec. limit (4)	%	1.54E-01	5.65E-02	
B. TRITIUM				
1. Total Release	Ci	1.92E+02	2.36E+02	
2. Average diluted concentration during period	µCi/ml	1.60E-07	1.98E-07	
3. Percent of applicable limit (5)	%	5.33E-03	6.59E-03	
C. DISSOLVED AND ENTRAINED GASES				
1. Total Release	Ci	2.24E-04	4.36E-04	
2. Average diluted concentration during period	µCi/ml	1.87E-13	3.66E-13	

TABLE 2A - REG GUIDE 1.21
LIQUID EFFLUENTS - SUMMATION OF ALL RELEASES

	UNITS	1ST QUARTER	2ND QUARTER	EST. TOTAL ERROR, %
D. GROSS ALPHA RADIOACTIVITY				
1. Total Release	Ci	2.81E-05	(6)	
E. VOLUME OF WASTE RELEASED (prior to dilution)	liters	5.77E+07	1.16E+08	±1.30E+00
F. VOLUME OF DILUTION WATER USED DURING PERIOD	liters	1.04E+12	1.08E+12	±1.64E+01

	UNITS	1ST QUARTER	2ND QUARTER	EST. TOTAL ERROR, %
D. GROSS ALPHA RADIOACTIVITY				
1. Total Release	Ci	(6)	(6)	
E. VOLUME OF WASTE RELEASED (prior to dilution)	Liters	6.04E+07	5.41E+07	±1.30E+00
F. VOLUME OF DILUTION WATER USED DURING PERIOD	Liters	1.04E+12	1.19E+12	±1.64E+01

NOTES TO TABLE 2A

(1) Percent of I.C.3 Quarterly Organ Dose Limit (10 mRem) to maximum exposed organ

(2) Percent of I.C.3 Yearly Organ Dose Limit (20 mRem) to maximum exposed organ

(3) Percent of I.C.3 Quarterly Whole Body Dose Limit (3 mRem)

(4) Percent of I.C.3 Yearly Whole Body Dose Limit (6 mRem)

(5) Limit used is 3×10^{-3} μCi/ml

(6) Less than minimum detectable activity which meets the LLD requirements of ODCM Surveillance Requirement 4.11.1.1.1.

TABLE 2B - REG GUIDE 1.21
LIQUID EFFLUENTS

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	Units	1ST QUARTER	2ND QUARTER	1ST QUARTER	2ND QUARTER
Beryllium – 7	Ci	(1)	(1)	(1)	(1)
Sodium – 22	Ci	(1)	(1)	(1)	(1)
Sodium – 24	Ci	(1)	(1)	(1)	(1)
Chromium - 51	Ci	(1)	(1)	(1)	9.80E-03
Manganese - 54	Ci	(1)	(1)	4.33E-04	3.87E-04
Iron – 55	Ci	(2)	(2)	6.72E-03	2.18E-02
Cobalt – 57	Ci	(1)	(1)	8.07E-07	(1)
Cobalt – 58	Ci	(1)	(1)	5.26E-03	6.30E-03
Iron – 59	Ci	(1)	(1)	4.57E-04	2.27E-03
Cobalt - 60	Ci	(1)	(1)	1.92E-03	1.36E-03
Zinc – 65	Ci	(1)	(1)	(1)	(1)
Strontium - 89	Ci	(1)	(1)	(1)	(1)
Strontium - 90	Ci	(1)	(1)	(1)	(1)
Strontium - 92	Ci	(1)	(1)	(1)	(1)
Niobium - 95	Ci	(1)	(1)	2.40E-03	4.87E-03
Zirconium - 95	Ci	(1)	(1)	1.07E-03	2.65E-03
Niobium - 97	Ci	(1)	(1)	(1)	(1)
Zirconium - 97	Ci	(1)	(1)	(1)	(1)
Molybdenum - 99	Ci	(1)	(1)	(1)	(1)
Technetium - 99m	Ci	(1)	(1)	(1)	(1)
Ruthenium - 103	Ci	(1)	(1)	(1)	5.01E-05
Rhodium - 105	Ci	(1)	(1)	(1)	(1)
Ruthenium - 105	Ci	(1)	(1)	(1)	(1)
Silver - 110m	Ci	(1)	(1)	2.10E-03	9.04E-03
Tin – 113	Ci	(1)	(1)	(1)	7.81E-04
Tin – 117m	Ci	(1)	(1)	(1)	4.86E-04
Antimony - 122	Ci	(1)	(1)	1.15E-06	(1)
Antimony - 124	Ci	(1)	(1)	(1)	2.45E-02
Antimony - 125	Ci	(1)	(1)	5.20E-04	3.67E-02
Tellurium – 125m	Ci	(1)	(1)	8.50E-03	(1)
Tellurium - 132	Ci	(1)	(1)	9.05E-05	5.21E-05
Iodine - 131	Ci	(1)	(1)	1.05E-04	3.34E-04
Iodine - 132	Ci	(1)	(1)	2.22E-04	(1)
Iodine – 133	Ci	(1)	(1)	1.82E-05	2.08E-06
Iodine – 135	Ci	(1)	(1)	(1)	(1)
Cesium – 134	Ci	(1)	(1)	9.83E-05	1.85E-02
Cesium – 136	Ci	(1)	(1)	(1)	2.67E-05
Cesium – 137	Ci	(1)	(1)	3.88E-04	2.08E-02
Barium – 140	Ci	(1)	(1)	(1)	(1)
Lanthanum - 140	Ci	(1)	(1)	(1)	(1)
Cerium – 144	Ci	(1)	(1)	(1)	(1)
Europium – 154	Ci	(1)	(1)	(1)	(1)
Europium – 155	Ci	(1)	(1)	(1)	(1)
Tungsten – 187	Ci	(1)	(1)	(1)	(1)
Total For Period	Ci	(1)	(1)	3.03E-02	1.61E-01

TABLE 2B - REG GUIDE 1.21
LIQUID EFFLUENTS

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	Units	3RD QUARTER	4TH QUARTER	3RD QUARTER	4TH QUARTER
Beryllium - 7	Ci	(1)	(1)	(1)	(1)
Sodium – 22	Ci	(1)	(1)	(1)	(1)
Sodium - 24	Ci	(1)	(1)	(1)	(1)
Chromium - 51	Ci	(1)	(1)	2.07E-03	(1)
Manganese - 54	Ci	(1)	(1)	1.35E-03	2.14E-04
Iron – 55	Ci	(2)	(2)	2.84E-02	1.03E-02
Cobalt - 57	Ci	(1)	(1)	4.87E-05	(1)
Cobalt - 58	Ci	(1)	(1)	5.29E-03	1.05E-03
Iron – 59	Ci	(1)	(1)	7.75E-04	4.47E-05
Cobalt - 60	Ci	(1)	(1)	3.53E-03	1.17E-03
Zinc – 65	Ci	(1)	(1)	(1)	(1)
Strontium - 89	Ci	(1)	(1)	(1)	(1)
Strontium - 90	Ci	(1)	(1)	(1)	1.88E-07
Strontium - 92	Ci	(1)	(1)	(1)	(1)
Niobium - 95	Ci	(1)	(1)	1.15E-03	1.34E-03
Zirconium - 95	Ci	(1)	(1)	2.28E-03	7.76E-04
Niobium - 97	Ci	(1)	(1)	(1)	(1)
Zirconium - 97	Ci	(1)	(1)	(1)	(1)
Molybdenum - 99	Ci	(1)	(1)	(1)	(1)
Technetium - 99m	Ci	(1)	(1)	(1)	(1)
Ruthenium - 103	Ci	(1)	(1)	(1)	(1)
Rhodium - 105	Ci	(1)	(1)	(1)	(1)
Ruthenium - 105	Ci	(1)	(1)	(1)	(1)
Silver - 110m	Ci	(1)	(1)	4.73E-02	4.77E-03
Tin – 113	Ci	(1)	(1)	4.55E-04	7.85E-05
Tin – 117m	Ci	(1)	(1)	7.83E-05	(1)
Antimony - 122	Ci	(1)	(1)	(1)	(1)
Antimony - 124	Ci	(1)	(1)	(1)	5.57E-05
Antimony - 125	Ci	(1)	(1)	2.72E-03	5.10E-03
Tellurium - 132	Ci	(1)	(1)	(1)	(1)
Iodine - 131	Ci	(1)	(1)	2.67E-06	1.09E-05
Iodine - 133	Ci	(1)	(1)	7.98E-06	1.50E-05
Iodine - 135	Ci	(1)	(1)	(1)	(1)
Cesium - 134	Ci	1.28E-04	(1)	1.74E-04	7.10E-05
Cesium - 136	Ci	(1)	(1)	(1)	(1)
Cesium - 137	Ci	1.83E-04	(1)	1.97E-04	2.45E-04
Barium - 140	Ci	(1)	(1)	(1)	(1)
Lanthanum - 140	Ci	(1)	(1)	(1)	(1)
Cerium - 144	Ci	(1)	(1)	(1)	(1)
Europium - 154	Ci	(1)	(1)	(1)	(1)
Europium - 155	Ci	(1)	(1)	(1)	(1)
Tungsten - 187	Ci	(1)	(1)	(1)	(1)
Total For Period	Ci	3.11E-04	(1)	9.53E-02	2.52E-02

TABLE 2B - REG GUIDE 1.21
LIQUID EFFLUENTS

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	Units	1ST QUARTER	2ND QUARTER	1ST QUARTER	2ND QUARTER
Krypton – 85	Ci	(1)	(1)	(1)	(1)
Xenon - 131m	Ci	(1)	(1)	4.12E-03	(1)
Xenon – 133	Ci	(1)	(1)	1.84E-01	1.54E-03
Xenon – 135	Ci	(1)	(1)	2.80E-05	(1)
Total For Period	Ci	(1)	(1)	1.89E-01	1.54E-03

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	Units	3RD QUARTER	4TH QUARTER	3RD QUARTER	4TH QUARTER
Krypton - 85	Ci	(1)	(1)	(1)	(1)
Xenon-131m	Ci	(1)	(1)	(1)	(1)
Xenon - 133	Ci	(1)	(1)	2.24E-04	4.35E-04
Xenon - 135	Ci	(1)	(1)	(1)	1.50E-06
Total For Period	Ci	(1)	(1)	2.24E-04	4.36E-04

NOTES TO TABLE 2B

(1) Less than minimum detectable activity which meets the LLD requirements of ODCM Surveillance Requirement 4.11.1.1.1.

(2) Continuous mode effluents are not analyzed for Fe-55.

TABLE 3A
SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

A. SOLID WASTE SHIPPED OFFSITE FOR BURIAL OR DISPOSAL (NOT IRRADIATED FUEL)

1.	Type of Waste		UNITS	12-MONTH PERIOD	EST. TOTAL ERROR %
	a.	Dewatered spent resin	m^3 Ci	1.49E+01 4.75E+02	 ±2.00E+01
	b.	Dry Compressible Waste (Shipped) Contaminated Equipment, etc.	m^3 Ci	7.12E+02 2.28E+00	 ±5.00E+01
	b. *	Dry Compressible Waste (Buried) Contaminated Equipment, etc.	m^3 Ci	1.67E+00 6.73E-01	 ±5.00E+01
	c.	Irradiated Components, Control Rods, etc.	m^3 Ci	0.00E+00 0.00E+00	 N/A
	d.	Other (Cartridge Filters)	m^3 Ci	0.00E+00 0.00E+00	 N/A

(b.) Volume shipped represents waste generated prior to offsite volume reduction.
(b. *) Represents waste buried after volume reduction at offsite processor.

2. **Estimate of Major Nuclides** (By Type of Waste - Only nuclides >1 % are reported)

a.	Be-7	1.18E+01%
	Mn-54	1.77E+00%
	Fe-55	8.37E+00%
	Co-58	1.13E+01%
	Co-60	3.62E+00%
	Ni-63	2.00E+01%
	Cs-134	1.69E+01%
	Cs-137	2.37E+01%
b.	Cr-51	1.14E+01%
	Fe-55	2.37E+01%
	Co-58	2.79E+01%
	Co-60	4.94E+00%
	Nb-95	7.92E+00%
	Zr-95	5.09E+00%
	Cs-134	1.97E+00%
	Cs-137	4.74E+00%

c. N/A

d. N/A

TABLE 3A
SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

3. Solid Waste Disposition

Number of Shipments	Mode of Transportation	Destination
6	Motor Surface Transit	Chem. Nuclear Systems, Inc. Barnwell, SC
11	Motor Surface Transit	GTS Duratek Oak Ridge, TN
38	Motor Surface Transit	US Ecology, Inc. Oak Ridge, TN

CALVERT CLIFFS NUCLEAR POWER PLANT
EFFLUENT AND WASTE DISPOSAL 2000 ANNUAL REPORT

TYPE WASTE: DAW

10 CFR PART 61 WASTE CLASS: A

SOURCE OF WASTE: Radiologically Controlled Areas

SHIPPING CONTAINER: 20' or 40' Sealand Containers, B-25 Metal Boxes, and 55-Gallon Drums

TOTAL CURIE QUANTITY: 2.28 Ci

HOW DETERMINED: Dose to curie content, conversion by volume based on generic distribution and scaling factors

TOTAL SHIPPED WASTE VOLUME: 25,142 ft^3

TOTAL BURIAL WASTE VOLUME: 59.2 ft^3

HOW DETERMINED: Container volume and number of containers shipped, Burial volume is determined from information provided by volume reduction processor.

SOLIDIFICATION AGENT OR ABSORBENT: None

TYPE WASTE: Dewatered Resin

10 CFR PART 61 WASTE CLASS: A (S), B, and C

SOURCE OF WASTE: Liquid Waste Processing Systems

SHIPPING CONTAINER: 8-120 High Integrity Container shipped in a 8-120A or 8-120B Shipping Cask

TOTAL CURIE QUANTITY: 475 Ci

HOW DETERMINED: Gamma scan analysis using resin sample, conversion by weight based on radionuclide distribution and scaling factors

TOTAL SHIPPED WASTE VOLUME: 526 ft^3

TOTAL BURIAL WASTE VOLUME: 721.8 ft^3

HOW DETERMINED: Waste volume determined by mass and assumed density of dewatered resin

SOLIDIFICATION AGENT OR ABSORBENT: None

ENCLOSURE (2)

OFFSITE DOSE CALCULATION MANUAL FOR CALVERT CLIFFS NUCLEAR POWER PLANT,

Revision 4

Calvert Cliffs Nuclear Power Plant – Units 1 & 2
Independent Spent Fuel Storage Installation
July 19, 2001

OFFSITE DOSE CALCULATION MANUAL
Revision 4, Change 0

For The
Baltimore Gas And Electric Company
Calvert Cliffs Nuclear Power Plant

PREPARED BY:	Robert A Krieger — Technical Writer	DATE: 1/18/00
REVIEWED BY:	RETS Program Manager	DATE: 1/24/00
REVIEWED BY:	Chris Earls — General Supervisor - Chemistry	DATE: 1/31/00
REVIEWED BY:	POSRC — Meeting Number 00-008	DATE: 1-31-00
APPROVED BY:	P.E. Katz — Plant General Manager	DATE: 2/7/00

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PART 1.0: PURPOSE AND APPLICABILITY/SCOPE

PURPOSE

A. The ODCM lists the radiological effluent controls established at Calvert Cliffs Nuclear Power Plant (CCNPP) for the purpose of ensuring the amount of radioactive materials released to the environment are as low as reasonably achievable.

B. The ODCM defines parameters and methodologies for calculating projected doses and dose rates resulting from the release of radioactive materials in liquid and gaseous effluents.

C. The ODCM defines parameters and methodologies for calculating alarm and trip setpoints for Technical Specification related radiation effluent monitoring systems.

D. The ODCM defines and establishes controls for the Radiological Environmental Monitoring Program.

APPLICABILITY/SCOPE

A. The information presented in this manual is applicable to any division, department, section, or unit within the Baltimore Gas And Electric Company (BGE) which is either wholly or partly responsible for performing any of the activities listed in the PURPOSES section of this procedure. Responsible organizations include, but are not limited to, the following:

1. BGE, Nuclear Energy Division (NED), CCNPP Department (CCNPPD), CHEMISTRY SECTION
2. BGE, Fossil Energy Division, Technical Services Section, Chemistry Unit
3. BGE, NED, CCNPPD, Electrical and Controls (E&C) Section
4. BGE, NED, CCNPPD, Operations Section
5. BGE, NED, CCNPPD, Radiation Safety Section

B. This manual is applicable to the determination of alarm and trip setpoints for the following radioactive gaseous effluent monitoring instrumentation:

1. 0-RE-2191
2. 1/2-RE-5415
3. 1/2-RE-5416

C. This manual is applicable to the determination of alarm and trip setpoints for the following radioactive liquid effluent monitoring instrumentation:

1. 0-RE-2201
2. 1/2-RE-4014

3. 1/2-RE-4095

D. This manual is applicable to the determination of the offsite doses and/or offsite dose rates due to the following:

1. Radioactive material in gaseous waste discharged from CCNPP

2. Radioactive material in liquid waste discharged from CCNPP

3. Radioactive material contained in outside storage tanks at CCNPP

E. This manual is applicable to the determination of the radiological effects on the environment due to the presence of Calvert Cliffs Nuclear Power Plant.

F. The ODCM is based on Technical Specifications, BGE's interpretation of industry standards and practices, and recommendations made by Combustion Engineering.

PART 2.0 : DEFINITIONS AND REFERENCES

DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Controls. Definitions linked to Technical Specification are designated by **TS** and the Technical Specification definition number it is linked to, e.g. **TS-1.0.**

ABNORMAL AND/OR UNANTICIPATED RADIOACTIVE RELEASE:

Any unplanned or uncontrolled release of radioactive material from the **SITE BOUNDARY**.

ACTION:

That part of a control which prescribes remedial measures required under designated conditions. **TS-1.0**

BATCH RELEASE:

A **BATCH RELEASE** is the discharge of liquid (or gaseous) waste of a discrete volume. (NUREG-0133, page 14)

CHANNEL CALIBRATION:

A **CHANNEL CALIBRATION** shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The **CHANNEL CALIBRATION** shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the **CHANNEL FUNCTIONAL TEST**. The **CHANNEL CALIBRATION** may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated. **TS-1.0**

CHANNEL CHECK:

A **CHANNEL CHECK** shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter. **TS-1.0**

CHANNEL FUNCTIONAL TEST:

a. An analog **CHANNEL FUNCTIONAL TEST** shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify **OPERABILITY** including alarm and/or trip functions. **TS-1.0**

b. A bistable **CHANNEL FUNCTIONAL TEST** shall be injection of a simulated signal into the channel sensor to verify **OPERABILITY** including alarm and/or trip functions. **TS-1.0**

COMPOSITE SAMPLE:

A **COMPOSITE SAMPLE** is a combination of individual samples obtained at intervals that are very short (e.g., hourly) in relation to the compositing time interval (e.g., monthly) to assure obtaining a representative sample. The sample volume should be proportionate to the volume of fluid, either liquid or gas, flowing through the system.

CONTINUOUS RELEASE

A **CONTINUOUS RELEASE** is the discharge of liquid (or gaseous) wastes of a nondiscrete volume; e.g., from a volume or system that has an input flow during the **CONTINUOUS RELEASE**. (NUREG-0133, P. 14.)

CONTINUOUS SAMPLING FREQUENCY

A **CONTINUOUS SAMPLING FREQUENCY** is a sampling arrangement that provides for uninterrupted sampling flow under normal operating conditions. During periods of sampling interruption, the requirement for continuous sampling is considered to be met, provided the interruption is less than one hour AND plant conditions are such that no adverse trend in effluents would be present during the period of interruption. Interruptions in continuous sampling during changing plant conditions OR interruptions in continuous sampling for greater than one hour must be evaluated to determine if an ODCM violation has occurred.

DOMINANT RADIONUCLIDE:

A **DOMINANT RADIONUCLIDE** is one whose activity is greater than 1% of the total activity found in a TYPICAL RELEASES of liquid or gaseous radwaste.

DOSE EQUIVALENT IODINE-131:

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." **TS-1.0**

FREQUENCY NOTATION:

The frequency notation specified for the performance of Surveillance Requirements shall correspond to the intervals defined in the following Table **TS-1.0**.

Frequency Notation Table **TS-table 1.0**

NOTATION	FREQUENCY
S	At least once per 12 hours
D	At least once per 24 hours
W	At least once per 7 days
M	At least once per 31 days
Q	At least once per 92 days
SA	At least once per 6 months
R	At least once per 18 months
S/U	Prior to each reactor startup
P	Completed prior to each release
N.A.	Not applicable
Refueling Interval	At least once per 24 months

GAMMA ISOTOPIC ANALYSIS:

A **GAMMA ISOTOPIC ANALYSIS** is a analytical method of measurement used for the identification and quantification of gamma-emitting radionuclides.

GASEOUS RADWASTE TREATMENT SYSTEM:

A **GASEOUS RADWASTE TREATMENT SYSTEM** is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment. **TS-1.0**

GSC:

GSC stands for General Supervisor Chemistry.

LESS DOMINANT RADIONUCLIDE:

A **LESS DOMINANT RADIONUCLIDE** is one whose activity is less than 1% of the total activity found in TYPICAL RELEASES of liquid or gaseous radwaste.

LIQUID RADWASTE TREATMENT SYSTEM:

A **LIQUID RADWASTE TREATMENT SYSTEM** is the system installed and designed to reduce radioactive liquid effluents. The minimum components necessary for reducing liquid radioactive effluents is either 13 or 14 Reactor Coolant Waste Ion Exchange; however, other equipment in the system may be used in the maintenance of ALARA for liquid radioactive effluents.

LOWER LIMIT OF DETECTION:

The LLD is the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

MEMBERS OF THE PUBLIC:

MEMBERS OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries **TS-1.0**. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL:

The **OFFSITE DOSE CALCULATION MANUAL** (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain the radioactive effluent controls, 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 Specification 6.6.2 (Improved Technical Specification 5.6.2) and 6.6.3 (Improved Technical Specification EC-01
5.6.3). **TS-1.0**

OPERABILITY:

A system, subsystem, train, component or device shall be operable or have **OPERABILITY** when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other required auxiliary equipment that are required for the system, subcomponent or device to perform its function(s) are also capable of performing their related support function(s). **TS-1.0**

OPERATIONAL MODE:

An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in below. **TS-1.0**

Operational Modes **TS-table 1.0**

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
POWER OPERATION	≥0.99	>5%	≥300 °F
STARTUP	≥0.99	≤5%	≥300 °F
HOT STANDBY	<0.99	0	≥300 °F
HOT SHUTDOWN	<0.99	0	300 °F > T_{avg} > 200 °F
COLD SHUTDOWN	<0.99	0	≤200 °F
REFUELING**	≤0.95	0	≤140 °F

* Excluding decay heat.
** Reactor vessel head unbolted or removed and fuel in the vessel.

PROCESS CONTROL PROGRAM:

PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71 and Federal and State and Local regulations governing the disposal of the radioactive waste.

PURGE OR PURGING:

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

RATED THERMAL POWER:

RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2700 MWt. **TS-1.0**

SIMULTANEOUS RELEASES:

Simultaneous liquid releases are discharges of liquid radwaste which occur at the same time. Simultaneous gaseous releases are discharges of gaseous radwaste which occur at the same time excluding main vent discharges.

SITE BOUNDARY:

The **SITE BOUNDARY** shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee. **TS-1.0**
(The **SITE BOUNDARY** is depicted on Attachment 18, "Environmental Monitoring Sites")

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. **TS-1.0**

THERMAL POWER:

The THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant. **TS-1.0**

TYPICAL RADWASTE RELEASES (OR TYPICAL RADWASTE EFFLUENTS):

TYPICAL RADWASTE RELEASES are defined as (1) all releases conducted during any calendar quarter while either Unit 1 or Unit 2 is in mode 1, and also includes (2) all releases conducted during any calendar quarter following mode 1 operation of either Unit 1 or Unit 2.

This definition of TYPICAL RADWASTE RELEASES is intended to ensure the concentrations of **DOMINANT RADIONUCLIDES** represent realistic, maximum-expected values. This definition is also intended to ensure that the concentrations of **DOMINANT RADIONUCLIDES** is not biased "low", as would be the case, if periods of extended outages--when the production of fission products is minimized--were included.

This definition may be modified by the **GSC**, however, the new definition shall be documented in accordance with the provisions outlined in the applicable section of the ODCM. (e.g., See the section which contains information related to documenting the fixed/adjustable setpoint.)

UNRESTRICTED AREA:

An UNRESTRICTED AREA shall be any area at or beyond the **SITE BOUNDARY** access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the **SITE BOUNDARY** used for residential quarters of/for industrial, commercial, institutional, and/or recreational purposes. **TS-1.0**

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 High Efficiency Particulate Air (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. **TS-1.0**

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. **TS-1.0**

WASTE GAS HOLDUP SYSTEM:

See **GASEOUS RADWASTE TREATMENT SYSTEM**.

REFERENCES

DEVELOPMENT REFERENCES

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2. Regulatory Guide 1.109, "Calculation Of Annual Doses To Man From Routine Release Of Reactor Effluents For The Purpose Of Compliance With 10 CFR Part 50, Appendix I," Revision 1, (October 1977).

3. 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).

4. CCNPP System Description Number 14B, "Reactor Coolant Waste Processing System Description."

5. Updated Final Safety Analysis Report, Chapter 11, "Waste Processing And Radiation Protection."

6. CCNPP System Description Number 14D, "Miscellaneous Liquid Waste Processing System Description."

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8. OI-17C, "Reactor Coolant Waste Processing System"

9. Title 10 of the Code of Federal Regulations, Part 20, Jan 1990 and May 1991

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11. Calvert Cliffs Nuclear Power Plant Semi-Annual Radiological Effluent Release Report (1986, 1987, 1988).

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18. CP-607, Revision 1, "Offsite Dose Calculation Manual"

19. NO-1-201, Calvert Cliffs Operating Manual

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21. "Plant Data Book", BGE CCNPP Units 1 and 2, Bechtel Power Corporation, Volume 1, Job 6750.

22. 50.59 Log No. 82-B-999-028-R00, Safety Analysis No. 2, FCR 82-1053, Supplement 1 (Component Cooling System) [B527]

23. 50.59 Log No. 90-0-074-011-R2, Activity MASE 90-7 (Plant Nitrogen System) [B527]

24. 50.59 Log No. 90-0-029-045-R1, Activity MASE 90-6 (Plant Heating System) [B527]

25. 50.59 Log No. 90-0-037-044-R1, Activity MASE 90-5 (Demineralized Water System) [B527]

26. 50.59 Log No. 90-B-012-043-R1, Activity MASE 90-4 (Service Water System) [B527]

27. 50.59 Log No. 90-0-027-037-R2, Activity MASE 90-8 (Auxiliary Boiler System) [B527]

28. Bechtel Power Corporation, Calculation Number M-90-20, "Allowable Radioactive Contamination Levels in the Plant Heating System" [B527]

29. Bechtel Power Corporation, Calculation Number M-90-18, "Allowable Radioactive Contamination Levels in the Nitrogen System Header" [B527]

30. Bechtel Power Corporation, Calculation Number M-90-04, "Allowable Radioactive Contamination Levels in Auxiliary Boiler Water" [B527]

31. Bechtel Power Corporation, Calculation Number M-90-21, "Allowable Radioactive Contamination Levels in the Demineralized Water System" [B527]

32. Bechtel Power Corporation, Calculation Number M-90-19, "Allowable Radioactive Contamination Levels in the Service Water System" [B527]

33. Bechtel Power Corporation, Job Number 11865, Calculation Number 7.4.3-15, "Allowable Radioactive Contamination Levels in the Condensate Storage Tank" [B527]

34. EN-1-100, Engineering Service Process Overview

35. OI-8A, Blowdown System

36. NUREG-1301, "**OFFSITE DOSE CALCULATION MANUAL**: Standard Radiological Effluent Controls for Pressurized Water Reactors", W. W. Meinke, and T. H. Essig, (Published April 1991).

37. Radiological Environmental Monitoring Program Annual Report for the Calvert Cliffs Nuclear Power Plant Units 1 and 2 January 1 - December 31, 1991", Baltimore Gas And Electric Company, March 1992

38. 50.59 Log No. 90-B-037-120-R2, Activity MASE 90-11 (Condensate Storage Tank) **[B527]**

39. Regulatory Guide 4.13, Performance Testing and Procedural Specifications for Thermoluminescence Dosimetry; Environmental Applications

PERFORMANCE REFERENCES

1. TE-001, Main Vent Stack Flow Measurement

2. Technical Specifications

3. Technical Requirements Manual

4. Regulatory Guide 4.13, Performance Testing and Procedural Specifications for Thermoluminescence Dosimetry; Environmental Applications

5. TE-006, Containment Purge Exhaust System HEPA Filter Test

PART 3.0 / 4.0 : CONTROLS AND SURVEILLANCE REQUIREMENTS

CONTROLS

3.0.1 Compliance with the Controls contained in the succeeding controls is required during the **OPERATIONAL MODES** or other conditions specified therein; except that upon failure to meet the Control, the associated **ACTION** requirements shall be met.

3.0.2 Noncompliance with a Control shall exist when the requirements of the Control and associated **ACTION** requirements are not met within the specified time intervals. If the Control is restored prior to expiration of the specified time intervals, completion of the **ACTION** requirements is not required.

3.0.3 When a Control is not met, except as provided in the associated **ACTION** requirements, within one hour **ACTION** shall be initiated to place the unit in a MODE in which the control does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the **ACTION** requirements, the **ACTION** may be taken in accordance with the specified time limits as measured from the time of failure to meet the Control. Exceptions to these requirements are stated in the individual controls.

3.0.4 Entry into an **OPERATIONAL MODE** or other specified condition shall not be made unless the conditions of the Control are met without reliance on provisions contained in the **ACTION** requirements. This provision shall not prevent passage through **OPERATIONAL MODES** as required to comply with **ACTION** requirements. Exceptions to these requirements are stated in the individual controls.

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Control, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are **OPERABLE**, or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied within 2 hours, action shall be initiated to place the unit in a **MODE** in which the applicable Control does not apply by placing it, as applicable in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This specification is not applicable in **MODES** 5 or 6.

SURVEILLANCE

4.0.1 Surveillance Requirements shall be applicable during the **OPERATIONAL MODES** or other conditions specified for individual Control unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Control 4.0.2, shall constitute noncompliance with the **OPERABILITY** requirements for a Control. The time limits of the **ACTION** requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. However, this time of applicability may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the **ACTION** requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an **OPERATIONAL MODE** or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Control have been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to **OPERATIONAL MODES** as required to comply with **ACTION** requirements.

MONITORING INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT

CONTROLS

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-12 shall be **OPERABLE** with their alarm/trip setpoints set to ensure that the limits of Control 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: **As shown in Table 3.3-12.**

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Control, without delay 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.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels **OPERABLE**, take the **ACTION** shown in Table 3.3-12. Exert best efforts to return the instruments to **OPERABLE** status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated **OPERABLE** by performance of the **CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION** and **CHANNEL FUNCTIONAL TEST** operations at the frequencies shown in Table 4.3-11.

TABLE 3.3-12
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	35
b. Effluent System Flow Rate Measuring Device	1	*	36
2. MAIN VENT SYSTEM			
a. Noble Gas Activity Monitor	1	*	37
b. Iodine Sampler	1	*	38
c. Particulate Sampler	1	*	38

* At all times.

TABLE 3.3-12 (Continued)

ACTION STATEMENTS

ACTION 35 - With the number of channels **OPERABLE** less than required by the Minimum Channels **OPERABLE** requirement, the contents of the tank(s) may be released to the environment:

a. Using the main vent monitor as a backup and recording RMS readings every 15 minutes during the release, or

b. Provided that prior to initiating the release, at least two independent samples of the tank's contents are analyzed, and at least two technically qualified members of the Facility Staff independently verify the release rate calculations and two qualified operators verify the discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - 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.

ACTION 37 - With the number of channels **OPERABLE** less than required by the Minimum Channels **OPERABLE** requirement, effluent releases via this pathway may continue provided either (1) grab samples are taken and analyzed for gross activity at least once per 24 hours, or (2) an equivalent monitor is provided.

ACTION 38 - With the number of channels **OPERABLE** less than required by the Minimum Channels **OPERABLE** requirement, effluent releases via the affected pathway may continue provided samples are continuously collected as required in Table 4.11-2 with auxiliary sampling equipment.

TABLE 4.3-11
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	SA(1)	*
b. Effluent System Flow Rate Measuring Device	D(4)	NA	R	NA	*
2. MAIN VENT SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	SA(2)	*
b. Iodine Sampler	W	NA	NA	NA	*
c. Particulate Sampler	W	NA	NA	NA	*

* At all times other than when the line is valved out and locked.

TABLE 4.3-11 (Continued)

TABLE NOTATION

(1) The **CHANNEL FUNCTIONAL TEST** shall also demonstrate the automatic isolation of this pathway and/or Control Room alarm annunciation occurs if the appropriate following condition(s) exists:

1. Instrument indicates measure levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(2) The **CHANNEL FUNCTIONAL TEST** shall also demonstrate that Control Room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(3) The initial **CHANNEL CALIBRATION** shall be performed using one or more of the reference standards traceable to NIST or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system within its intended range of energy and measurement range. For subsequent **CHANNEL CALIBRATION**, sources that have been related to the initial calibration can be used.

(4) The **CHANNEL CHECK** shall consist of verifying indication of flow during periods of release and shall be made at least once per 24 hours during periods in which effluent releases are made.

RADIOACTIVE LIQUID EFFLUENT

CONTROLS

3.3.3.10 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-13 shall be **OPERABLE** with their alarm/trip setpoints set to ensure that the limits of Control 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the **OFFSITE DOSE CALCULATION MANUAL (ODCM).**

APPLICABILITY: **As shown in Table 3.3-13.**

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Control, without delay 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.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels **OPERABLE**, take the **ACTION** shown in Table 3.3-13. Exert best efforts to return the instruments to **OPERABLE** status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated **OPERABLE** by performance of the **CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION** and **CHANNEL FUNCTIONAL TEST** operations at the frequencies shown in Table 4.3-12.

TABLE 3.3-13

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE			
a. Liquid Radwaste Effluent Line	1	(1)	28
b. Steam Generator Blowdown Effluent Line	1	(2)	29
2. FLOW RATE MEASUREMENT DEVICES			
a. Liquid Radwaste Effluent Line	1	(1)	30
b. Steam Generator Blowdown Effluent Line	1	(2)	30

Note: (1) At all times.
(2) At all times while process flow is present.

TABLE 3.3-13 (Continued)

ACTION STATEMENTS

ACTION 28 - 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:

a. At least two independent samples are analyzed in accordance with Control 4.11.1.1.1, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and two qualified operators verify the discharge valve line up.

ACTION 29 - 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 analyzed for gross radioactivity (beta or gamma) at the **LOWER LIMIT OF DETECTION** defined in Table 4.11-1:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microcurie/gram **DOSE EQUIVALENT I-131**.

b. At least once per 48 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcurie/gram **DOSE EQUIVALENT I-131**.

ACTION 30 - 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. Pump performance curves may be used to estimate flow.

TABLE 4.3-12
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line	D	P	R(2)	SA(1)
b. Steam Generator Blowdown Effluent Line	D	P	R(2)	SA(1)
2. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(3)	NA	R	NA
b. Steam Generator Blowdown Effluent Line	D(3)	NA	R	NA

TABLE 4.3-12 (Continued)

TABLE NOTATION

(1) The **CHANNEL FUNCTIONAL TEST** shall also demonstrate that automatic isolation of this pathway and/or Control Room alarm annunciation occur if the appropriate following condition(s) exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Circuit failure.

3. Instrument indicates a downscale failure.

(2) The initial **CHANNEL CALIBRATION** shall be performed using one or more of the reference standards traceable to NIST or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system within its intended range of energy and measurement range. For subsequent **CHANNEL CALIBRATION**, sources that have been related to the initial calibration can be used.

(3) **CHANNEL CHECK** shall consist of verifying indication of flow during periods of release. **CHANNEL CHECK** shall be made at least once per 24 hours during periods in which effluent releases are made.

RADIOACTIVE EFFLUENTS

LIQUID EFFLUENTS:

Concentration

CONTROLS

3.11.1.1 The concentration of radioactive material released in liquid effluents to **UNRESTRICTED AREAS** shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases.

APPLICABILITY: **At all times.**

ACTION:

a. With the concentration of radioactive material released in liquid effluents to **UNRESTRICTED AREAS** exceeding the above limits, without delay restore the concentration to within the above limits.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the **ODCM** to assure that the concentrations at the point of release are maintained within the limits of Control 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	MINIMUM SAMPLING FREQUENCY	ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[a] (mCi/ml)
A. Batch Waste Releases[b]	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
			Mo-99, Ce-144	2×10^{-6}
	P Each Batch	M Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[d]	Sr-89, Sr-90	5×10^{-8}
B. Turbine Building Sump	M	M	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
			Mo-99, Ce-144	2×10^{-6}

TABLE 4.11-1 (Continued)

TABLE NOTATION

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

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

$$LLD = \frac{4.66 s_b}{E \bullet V \bullet 2.22 \times 10^6 \bullet Y \bullet \exp(-\lambda \Delta t)}$$ **Error! Switch argument not specified.**

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 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.

It should be recognized that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement.

TABLE 4.11-1 (Continued)

TABLE NOTATION

b

Prior to sampling Reactor Coolant Waste and Miscellaneous Waste for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

c

The principal gamma emitters for which the LLD control applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Cs-134, Cs-137 and Ce-141. 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 Annual Radioactive Effluent Release Report pursuant to Technical Specification 6.6.3 (Improved Technical Specification 5.6.3) . EC-01

d

A **COMPOSITE SAMPLE** is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged in which the method of sampling employed results in a specimen that is representative of the liquids released.

LIQUID EFFLUENTS:

Dose

CONTROLS

3.11.1.2 The dose or dose commitment to a **MEMBER OF THE PUBLIC** from radioactive materials in liquid effluents released to **UNRESTRICTED AREAS** shall be limited:

a. During any calendar quarter to less than or equal to 3.0 mrems to the total body and to less than or equal to 10 mrems to any organ, and

b. During any calendar year to less than or equal to 6 mrems to the total body and to less than or equal to 20 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. 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 pursuant to 10 CFR 50.4 within 30 days 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.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Monthly cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 60 days.

LIQUID EFFLUENTS:

Liquid Radwaste Treatment System

CONTROLS

3.11.1.3 The **LIQUID RADWASTE TREATMENT SYSTEM** shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the calculated doses due to the liquid effluent to **UNRESTRICTED AREAS** exceeds 0.36 mrem to the total body or 1.20 mrem to any organ in a 92 day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission pursuant to 10 CFR 50.4 within 30 days a Special Report that includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. **ACTION**(s) taken to restore the inoperable equipment to **OPERABLE** status, and

   3. Summary description of **ACTION**(s) taken to prevent a recurrence.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3 Monthly doses due to liquid releases to **UNRESTRICTED AREAS** shall be calculated at least once per 60 days in accordance with the methodology and parameters in the **ODCM**.

GASEOUS EFFLUENTS:

Dose Rate

CONTROLS

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the **SITE BOUNDARY** (see figure in UFSAR Chapter 1) shall be limited to the following:

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

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

APPLICABILITY: **At all times.**

ACTION:

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

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the **ODCM**.

4.11.2.1.2 The dose rate due to iodine-131 and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the **ODCM** by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[a] (μCi/ml)
A. Waste Gas Storage[f] Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[b] (Gaseous Emissions Only)	1×10^{-4}
B. Containment Purge and Vent	P Each Batch Grab Sample	P Each Batch	Principal Gamma Emitters[b] (Gaseous Emissions Only)	1×10^{-4}
C. Main Vent	M[c] Grab Sample	M[C]	Principal Gamma Emitters[b] (Gaseous Emissions Only)	1×10^{-4}
	Continuous[d]	M	H-3	1×10^{-6}
	Continuous[d]	W Charcoal Sample[e]	I-131	1×10^{-12}
	Continuous[d]	W Particulate Sample[e]	Principal Gamma Emitters[b] (I-131, others)	1×10^{-11}
	Continuous[d]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[d]	Q Composite Particulate Sample	Sr-89, Sr90	1×10^{-11}
	Continuous[d]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}
D. Incinerated Oil[g]	P Each Batch[h]	P Each Batch[h]	Principal Gamma Emitters	5×10^{-7}

TABLE 4.11-2 (Continued)

TABLE NOTATION

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

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

$$LLD = \frac{4.66 s_b}{E \bullet V \bullet 2.22 \times 10^6 \bullet Y \bullet \exp(-\lambda \Delta t)} 2$$

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 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.

It should be recognized that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement.

TABLE 4.11-2 (Continued)

TABLE NOTATION

b

The principal gamma emitters for which the LLD control applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radioactive Effluent Release Report pursuant to Technical Specification 6.6.3 (Improved Technical Specification 5.6.3). EC-01

c

Sampling and analysis shall also be performed following shutdown, **STARTUP**, or a **THERMAL POWER** change exceeding 15 percent of **RATED THERMAL POWER** within one hour unless (1) analysis shows that the **DOSE EQUIVALENT I-131** concentration in the primary coolant has not increased more than a factor of 5, and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 5.

d

The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Controls 3.11.2.1, 3.11.2.2 and 3.11.2.3.

e

Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. When sample collection time is less than seven days, the corresponding LLDs may be increased by a proportional factor. This requirement does not apply if (1) analysis shows that the **DOSE EQUIVALENT I-131** concentration in the primary coolant has not increased more than a factor of 5, and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 5.

f

Collect sample and analyze daily for total Curie content per Technical Specification 3.11.1.2 (Technical Requirements Manual 9.11.2.1) when the Reactor Coolant System specific activity of Xe-133 is greater than 150 uCi/ml.

g

Incinerated oil may be discharged via points other than the main vent (e.g., Auxiliary Boiler). Releases shall be accounted for based on pre-release grab sample data.

h

Samples for incinerated oil releases shall be collected from and representative of filtered oil in liquid form.

GASEOUS EFFLUENTS:

Dose - Noble Gases

CONTROLS

3.11.2.2 The air dose due to noble gases released in gaseous effluents to areas at and beyond the **SITE BOUNDARY** (see figure in UFSAR Chapter 1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation and,

b. During any calendar year: Less than or equal to 20 mrads for gamma radiation and less than or equal to 40 mrads for beta radiation.

APPLICABILITY: **At all times.**

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission pursuant to 10 CFR 50.4 within 30 days 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.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Monthly cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the **ODCM** at least once per 60 days.

GASEOUS EFFLUENTS:

Dose - Iodine-131 & Radionuclides in Particulate Form

CONTROLS

3.11.2.3 The dose to a **MEMBER OF THE PUBLIC** from iodine-131 and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released to areas at and beyond the **SITE BOUNDARY** (see figure in UFSAR Chapter 1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 15 mrems to any organ and,

b. During any calendar year: Less than or equal to 30 mrems to any organ.

c. Less than 0.1% of the limits of 3.11.2.3(a) and (b) as a result of burning contaminated oil.

APPLICABILITY: **At all times.**

ACTION:

a. With the calculated dose from the release of iodine-131 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 pursuant to 10 CFR 50.4 within 30 days 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.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Monthly cumulative dose contributions for the current calendar quarter and the current calendar year for iodine-131 and radionuclides in particulate form with half lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 60 days.

GASEOUS EFFLUENTS:

Gaseous Radwaste Treatment System

CONTROLS

3.11.2.4 The **GASEOUS RADWASTE TREATMENT SYSTEM** and the **VENTILATION EXHAUST TREATMENT SYSTEM** shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the gaseous effluent air doses due to gaseous effluent releases, to areas at and beyond the **SITE BOUNDARY** (see figure in UFSAR Chapter 1) exceeds 1.20 mrad for gamma radiation and 2.4 mrad for beta radiation in a 92 day period. The **VENTILATION EXHAUST TREATMENT SYSTEM** shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the calculated doses due to gaseous effluent releases, to areas at and beyond the **SITE BOUNDARY** (see figure in UFSAR Chapter 1) exceeds 1.8 mrem to any organ in a 92 day period.

APPLICABILITY: **At all times.**

ACTION:

a. With gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission pursuant to 10 CFR 50.4 within 30 days a Special Report that includes the following information:

   1. Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. **ACTION**(s) taken to restore the inoperable equipment to **OPERABLE** status, and

   3. Summary description of **ACTION**(s) taken to prevent a recurrence.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4 Monthly doses due to gaseous releases shall be calculated at least once per 60 days in accordance with the methodology and parameters in the **ODCM**.

TOTAL DOSE

CONTROLS

3.11.4 The annual (calendar year) dose or dose commitment to any **MEMBER OF THE PUBLIC** due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: **At all times.**

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Control 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations shall be made including direct radiation contributions from the reactor units and outside storage tanks to determine whether the above limits of Control 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission pursuant to 10 CFR 50.4 within 30 days 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 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a **MEMBER OF THE PUBLIC** from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff **ACTION** on the request is complete.

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Controls 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the **ODCM**.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and outside storage tanks shall be determined in accordance with the methodology and parameters in the **ODCM**. This requirement is applicable only under conditions set forth in Control 3.11.4.a.

RADIOLOGICAL ENVIRONMENTAL MONITORING:

PROGRAM

CONTROLS

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Technical Specification 6.6.2 (Improved Technical Specification 5.6.2), a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. EC-01

b. With the level of radioactivity as the result of plant effluents in an environmental sample at a specified location exceeding the reporting levels of Table 3.12-2, prepare and submit to the Commission pursuant to 10 CFR 50.4 within 30 days after receiving the sample analysis a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective ACTIONs to be taken to reduce radioactive effluents so that the potential annual dose* to a MEMBER OF THE PUBLIC is less than the calendar year limits of Controls 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sample this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$ **Error! Switch argument not specified.**

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Controls 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

* The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

c. With fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, 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. Pursuant to Technical Specification 6.6.2 (Improved Technical Specification 5.6.2), identify the cause of the unavailability of samples and identify the new location(s) for obtaining the replacement samples in the next Annual Radiological Environmental Operating Report. EC-01

d. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. DIRECT RADIATION[b]	23 routine monitoring stations (DR1 - DR23) either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: an inner ring of stations, one in each meteorological sector in the general area of the **SITE BOUNDARY** (DR1 - DR09);[f] an outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site (DR10 - DR18); the remaining stations (DR19 - DR23) to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 area to serve as a control station.	At least Quarterly	Gamma dose at least quarterly.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. AIRBORNE Radioiodine and Particulates	Samples from 5 locations (A1-A5): 3 samples (A1-A3) from close to the 3 **SITE BOUNDARY** locations, in different sectors of the highest calculated annual average ground-level D/Q.[f] 1 sample (A4) from the vicinity of a community having the highest calculated annual average ground-level D/Q. 1 sample (A5) from a control location, as for example 15-30 km distant and in the least prevalent wind direction.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Canister I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;[c] Gamma isotopic analysis[d] of composite (by location) quarterly

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. WATERBORNE			
a. Surface	1 sample at intake area (Wa1) 1 sample at discharge area (Wa2)	COMPOSITE SAMPLE over 1-month period[e]	GAMMA ISOTOPIC ANALYSIS[d] monthly. Composite for tritium analysis quarterly.
b. Sediment from shoreline	1 sample from downstream area with existing or potential recreational value (Wb1)	Semiannually	GAMMA ISOTOPIC ANALYSIS[d] semiannually.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. INGESTION			
a. Fish and Invertebrates	3 samples of commercially and/or recreationally important species (2 fish species and 1 invertebrate species) in vicinity of plant discharge area (Ia1-Ia3).	Sample in season, or semiannually if they are not seasonal.	**GAMMA ISOTOPIC ANALYSIS**[d] on edible portions.
	3 samples of same species in areas not influenced by plant discharge (Ia4-Ia6).		
b. Food Products	Samples of 3 different kinds of broad leaf vegetation grown near the **SITE BOUNDARY** at 2 different locations of highest predicted annual average ground level D/Q (Ib1-Ib6).[f]	Monthly during growing season.	Gamma isotopic[d] and I-131 analysis.
	1 sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction (Ib7-Ib9).	Monthly during growing season.	Gamma isotopic[d] and I-131 analysis.

TABLE 3.12-1 (Continued)

TABLE NOTATION

a The code in parenthesis, e.g., DR1, A1, defines generic sample locations in this control that can be used to identify the specific locations in the map(s) and table in the **ODCM**. Specific parameters of distance and direction sector from the central point between the two containment buildings and additional description where pertinent, is provided for each sample location in Table 3.12-1, and in a table and figure(s) in the **ODCM**. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, October 1978", and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, and malfunction of automatic sampling equipment. If specimens are unobtainable due to sampling equipment malfunction, effort shall be made to complete corrective **ACTION** prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 6.6.2 (Improved Technical Specification 5.6.2). **EC-01** It is recognized that, at times, it may not be possible or practicable to continue 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 appropriate substitutions made within 30 days in the radiological environmental monitoring program. Pursuant to Technical Specification 6.6.2 (Improved Technical Specification 5.6.2), identify **EC-01** the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining samples in the next Annual Radiological Environmental Operating Report and also include in the report a revised figure(s) and table for the **ODCM** reflecting the new location(s).

TABLE 3.12-1 (Continued)

TABLE NOTATION

b One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The frequency of analysis or readout for TLD Systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading. Due to the geographical limitations, 9 sectors are monitored around the Calvert Cliffs Nuclear Power Plant.

c Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, **GAMMA ISOTOPIC ANALYSIS** shall be performed on the individual samples.

d **GAMMA ISOTOPIC ANALYSIS** means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

e A **COMPOSITE SAMPLE** is one in which the quantity (aliquot) of liquid sampled is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. In this program, **COMPOSITE SAMPLE** aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

f. Exception to these locations is in the South Sector where DR7, A1, 1b4, 1b5, and 1b6 are located approximately 0.7 km from the release point. This location is conservative with respect to the site boundary, which is located approximately 2.1 km from the release point.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/M^3)	FISH & INVERTEBRATES (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)
H-3	20,000#				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

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

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a,b]

LOWER LIMIT OF DETECTION (LLD)[c]

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/M^3)	FISH & INVERTEBRATES (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg,dry)
Gross Beta	4	0.01				
H-3	2,000[#]					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1[d]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

[#] If no drinking water pathway exists, a value of 3000 pCi/l may be used.

TABLE 4.12-1 (Continued)

TABLE NOTATION

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 6.6.2 (Improved Technical Specification 5.6.2). EC-01

b

Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

c

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

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

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

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, when applicable,

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

TABLE 4.12-1 (Continued)

TABLE NOTATION

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.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 6.6.2 (Improved Technical Specification 5.6.2). EC-01

d LLD for drinking water samples. If no drinking water pathway exists, the LLD of **GAMMA ISOTOPIC ANALYSIS** may be used.

RADIOLOGICAL ENVIRONMENTAL MONITORING:

LAND USE CENSUS

CONTROLS

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles), the location in each of the 9 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify within a distance of 5 km (3 miles) the locations in each of the 9 meteorological sectors of all milk animals and all gardens of greater than 50 m^2 producing broad leaf vegetation).

APPLICABILITY: **At all times.**

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Control 4.11.2.3, identify the new location(s) in the next Annual Radiological Environmental Operating Report, pursuant to Technical Specification 6.6.2 (Improved Technical Specification 5.6.2). **EC-01**

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Control 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Technical Specification 6.6.2 (Improved Technical Specification 5.6.2), identify the new location(s) in the next Annual Radiological Environmental Operating Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s). **EC-01**

c. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The land 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 by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Technical Specification 6.6.2 (Improved Technical Specification 5.6.2). **EC-01**

* Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the **SITE BOUNDARY** in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1.4b shall be followed, including analysis of control samples.

RADIOLOGICAL ENVIRONMENTAL MONITORING:

INTERLABORATORY COMPARISON PROGRAM

CONTROLS

3.12.3 Analyses shall be performed on all radioactive materials, supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission, that correspond to samples required by Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. 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 pursuant to Technical Specification 6.6.2 (Improved Technical Specification 5.6.2) . EC-01

b. The provisions of Controls 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the **ODCM**. 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 pursuant to Technical Specification 6.6.2 (Improved Technical Specification 5.6.2). EC-01

PART 5.0 : CALCULATIONAL METHODOLOGIES

RADIOACTIVE LIQUID EFFLUENTS

RELEASE PATHWAYS [B527]

1. Introduction

   a) Radioactive liquid waste generated as a result of operating the Calvert Cliffs Nuclear Power Plant (CCNPP) may be released to the Chesapeake Bay[1].

   b) There are four pathways by which all waste water, non-radioactive and radioactive, may be discharged from the site to the bay:

      (1) Outfall 001,

      (2) Outfall 002,

      (3) Outfall 003,

      (4) Outfall 004.

   c) A diagram which shows the location of each outfall is included as Attachment 1.

2. Outfall 001

   a) Water from the Chesapeake Bay is pumped through the condensers and is discharged to the Chesapeake Bay through the circulating water discharge conduits, known as outfall 001.

   b) The liquid radioactive waste is mixed with and diluted by the circulating water prior to exiting the discharge conduit.

   c) The circulating water discharge conduit extends 850 feet into the Chesapeake bay.

   d) The circulating water discharge conduit may accept liquid radioactive waste from various sources. Sources which may contribute radioactive material to the circulating water discharge conduit are tabulated in Attachment 2.

3. Outfalls 002, 003, and 004

   a) There are three other potential pathways for the release of radioactive liquids to the bay. These pathways are designated outfall 002, outfall 003, and outfall 004.

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1 The federal controls and administrative limits associated with the release of radioactive materials from CCNPP are discussed elsewhere in this document.

b) Sources which could potentially contribute radioactive material to each of these outfalls are tabulated in Attachment 3.

c) No radioactive materials are expected to be discharged from outfall 002 unless there is a primary to secondary leak, or unless there is an unforeseen and catastrophic failure of a condensate storage tank (CST).

d) No radioactive materials are expected to be released from outfalls 003 or 004 unless there is an unforeseen and catastrophic failure of a refueling water tank (RWT).

4. Unmonitored release paths not shown on Attachment 3 should be evaluated and added to the ODCM as necessary.

5. Safety evaluations have been conducted for operating the following systems after they have become contaminated:

a) component cooling water system

(1) In accordance with applicable safety evaluations (Ref. 22), continued operation of this system is allowed as long as the concentration of radionuclides in the component cooling water system is less than 3,000 MPCs.

b) plant heating system

(1) In accordance with applicable safety evaluations (Ref. 24 and Ref. 28), continued operation of this system is allowed as long as the concentration of radionuclides in the plant heating system is less than 0.3 MPCs.

c) auxiliary boiler system

(1) In accordance with applicable safety evaluations (Ref. 27 and Ref. 30) continued operation of this system is allowed as long as the concentration of radionuclides in the auxiliary boiler steam drum is less than 96 MPCs.

d) demineralized water system

(1) In accordance with applicable safety evaluations (Ref. 25 and Ref. 31), continued operation of this system is allowed as long as the concentration of radionuclides in the demineralized water system is less than 0.3 MPCs.

e) service water system

(1) In accordance with applicable safety evaluations (Ref. 26 and Ref. 32), continued operation of this system is allowed as long as the concentration of radionuclides in the service water system is less than 0.1 MPCs.

f) condensate storage tank

(1) In accordance with applicable safety evaluations (Ref. 33), continued operation of this system is allowed as long as the concentration of radionuclides in the condensate storage tank is less than 1.0 MPCs.

TYPES OF LIQUID RELEASES

1. Liquid radwaste discharges have been classified as **CONTINUOUS** or **BATCH** as shown on Attachments 2 and 3.

PROCESSING EQUIPMENT

1. Simplified Flow Diagram

a) An overview of the liquid waste processing system, including major equipment and (normal) flow paths, is outlined on Attachment 4.

b) There is no processing equipment for wastes discharged through outfalls 002, 003, and 004; however, the waste shall be analyzed for radioactivity in accordance with the analysis frequencies contained in approved CHEMISTRY SECTION procedures.

2. Modifications

a) Licensee initiated major changes to the Liquid Radioactive Waste System shall be reported to the Commission in the Radioactive Effluent Release Report for the period in which the modification to the waste system is completed. The discussion of each change shall contain:

(1) A description of the equipment, components and processes involved; and

(2) Documentation of the fact that the change, including the safety analysis, was reviewed and found acceptable by the onsite review function.

The report shall also include changes to the ODCM, in accordance with Technical Specification 6.6.3 (Improved Technical Specification 5.6.3) . EC-01

b) A "major" change or modification includes, but is not limited, to the removal or permanent bypass of any of the following:

(1) degassifier

(2) reactor coolant waste receiver tank

(3) reactor coolant waste monitor tank

(4) letdown filter

(5) reactor coolant waste ion exchanger

(6) miscellaneous waste ion exchanger

(7) miscellaneous waste filter

(8) miscellaneous waste receiver tank

(9) miscellaneous waste monitor tank

(10) evaporator

3. Detailed Description

a) A detailed description of the liquid waste processing system is beyond the scope of the ODCM.

b) For more information on the Miscellaneous Liquid Waste Processing System see FSAR and System Description Number 14D, "Miscellaneous Liquid Waste Processing System Description."

c) For more information on the Reactor Coolant Waste Processing System see System Description Number 14B, "Reactor Coolant Waste Processing System Description."

d) For more information see the Updated Final Safety Analysis Report, Chapter 11, "Waste Processing And Radiation Protection."

LIQUID EFFLUENT RADIATION MONITOR ALARM AND TRIP SETPOINTS

1. Liquid Waste Discharge Radiation Monitor (0-RE-2201)

a) General description

(1) number of radiation elements: one

(2) type of radiation element: in-line scintillation detector

(3) output: analog

(4) radiation indicator: 0-RI-2201

(5) units for radiation indicator: counts per minute

(6) supplier: Westinghouse

b) Functions of 0-RE-2201

(1) continuously measure the activity contained in liquid waste discharge line (Control 3.3.3.10)

(2) continuously indicate (via 0-RI-2201) the activity of liquids contained in liquid waste discharge line (Control 3.3.3.10)

(3) alarm (via 0-RI-2201) prior to exceeding 10 CFR 20, Appendix B, Table II, Column 2 limits (per Control 3.11.1.1)

(4) automatically terminate discharges from the liquid waste processing system by closing the discharge isolation valves (MWS-2201-CV, and MWS-2202-CV) whenever the radiation indicator (0-RI-2201) exceeds the fixed high radiation trip setpoint

c) **OPERABILITY** of 0-RE-2201

(1) This monitor shall be operable (or have **OPERABILITY**) when it is capable of performing its specified function(s)

(2) The functions of 0-RE-2201 are listed in paragraph (b) above.

d) Monitors equivalent to 0-RE-2201

(1) There are no equivalent monitors for 0-RE-2201.

e) Radiological effluent controls for 0-RE-2201

(1) Liquid waste discharges via this pathway may continue if any one of the following two conditions are satisfied (per Control 3.3.3.10).

(a) The liquid waste discharge monitor, 0-RE-2201, is operable and the alarm and trip setpoint for this monitor is set to ensure the concentrations of radioactive materials released in liquid effluents to **UNRESTRICTED AREAS** are less than the concentrations specified in 10 CFR 20, Appendix B, Table II, Column 2, or

(b) two independent samples are analyzed in accordance with Control 4.11.1.1.1; AND at least two technically qualified members of the Facility Staff independently verify the release rate calculations; AND two qualified operators verify the discharge valve line up.

f) Surveillances for 0-RE-2201

(1) Control 4.3.3.10 requires demonstrating the **OPERABILITY** of 0-RE-2201 by satisfying the checks, calibrations, and tests listed below:

(a) **CHANNEL CHECK** within the past 24 hours

(b) **SOURCE CHECK** prior to each release

(c) **CHANNEL CALIBRATION** within the past 18 months

(d) **CHANNEL FUNCTIONAL TEST** within the past 6 six months

g) Setpoints for 0-RI-2201

(1) There are three radiation alarm setpoints associated with, or otherwise related to, the liquid waste discharge monitor.

(a) 0-RI-2201 fixed high radiation alarm and automatic control trip setpoint

(b) 0-RI-2201 adjustable plant computer high radiation alarm and manual control trip setpoint

(c) 0-RI-2201 low radiation alarm setpoint

(2) In order to simplify the setpoint terminology, eliminate ambiguity, and minimize the possibility of misinterpretation, the ODCM will refer to these setpoints as follows:

(a) The 0-RI-2201 fixed high radiation alarm and automatic control trip setpoint will be referred to as the fixed setpoint.

(b) The 0-RI-2201 adjustable plant computer high radiation alarm and manual control trip setpoint will be referred to as the adjustable setpoint.

(c) The 0-RI-2201 low radiation alarm setpoint will be referred to as the low setpoint.

(3) Each of these alarm setpoints is described below.

h) Fixed setpoint for 0-RI-2201

(1) General information

(a) This setpoint is considered to be a fixed setpoint. This setpoint is not adjusted for each release.

(b) Whenever the fixed setpoint is exceeded, discharges from the liquid waste processing system will be automatically suspended.

(c) The fixed setpoint corresponds to the maximum concentration of radionuclides allowed in liquid waste discharged from the liquid waste processing system.

(d) The current value for the fixed setpoint is specified in the CCNPP Alarm Manual.

(e) The CCNPP Alarm Manual refers to this alarm and trip setpoint as the 0-RI-2201 Liquid Waste Discharge High Alarm.

(f) The fixed setpoint is integral to the liquid release discharge monitor, as purchased from the supplier.

(g) The fixed setpoint is administratively controlled by EN-1-100.

(h) The fixed setpoint shall be calculated as described below[1].

(2) Calculating the fixed setpoint for 0-RI-2201

(a) The fixed alarm and trip setpoint for 0-RI-2201 shall be calculated as follows:

FIXED ALARM AND TRIP SETPOINT FOR 0-RI-2201

$S_{fix} \leq K_{sf}\{ (F_d / F_u) \sum [(A_{iLn}) (e_i)] + Bkg \}$	Eq. 1L[2]

S_{fix} = the fixed alarm and trip setpoint for 0-RI-2201 (cpm)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP activity limit to the 10 CFR 20 MPC limit (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the fixed setpoint is always less than or equal to the limits of 10 CFR 20.

A safety factor of 1.00 will yield a fixed setpoint which corresponds to 1.0 MPC.

A safety factor of 0.500 will yield a fixed setpoint which corresponds to 0.5 MPCs.

It is recommended that a safety factor of 0.5 be used for calculating the fixed setpoint, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The use of a "safety margin" is in accordance with the provisions of NUREG-0133, section 4.1.1, which states, "The alarm and trip setpoints ... should correspond to a value(s) which represents a safe margin of assurance that the instantaneous liquid release limit of 10 CFR Part 20 is not exceeded."

This safety margin will prevent minor fluctuations in the nominal circulating water flow rate (and other statistical aberrations) from adversely impacting the calculated fixed setpoint.

F_d = the dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The dilution water flow rate is actually the sum of the minimum circulating water flow rate, the minimum salt water flow rate, and the maximum undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

1 Per Control 3.3.3.10.
2 This formula has been derived from NUREG-0133, Addendum, page AA-1.

DILUTION WATER FLOW RATE

$F_d = F_{cw} + F_{sw} + F_u$		Eq. 2L

F_{cw} = the minimum circulating water system flow rate necessary to conduct liquid releases

A minimum of two circulating water pumps (one circulating water pump per conduit accepting radwaste) shall be operable when discharging liquid radwaste -- via this monitor -- to outfall 001.

Additional circulating water pumps may be required as specified in approved Chemistry Procedures.

If a release of liquid radwaste is to be conducted using less than minimum circulating water flow rate necessary to conduct liquid releases, then the release shall not be allowed until a new setpoint has been calculated and entered into 0-RI-2201, or otherwise comply with the **ACTION** requirements of Control 3.3.3.10.

F_{sw} = the minimum salt water system flow rate necessary to release liquid radwaste

If the minimum salt water flow rate available for liquid releases is unknown, difficult to predict, or may decrease during a liquid release, a minimum salt water flow rate of 0 gpm should be used when calculating the fixed setpoint.

If a release of liquid radwaste is to be conducted using less than the minimum salt water flow rate necessary to conduct liquid releases, the release shall not be allowed until a new setpoint has been calculated and entered into 0-RI-2201, or otherwise comply with the **ACTION** requirements of Control 3.3.3.10.

F_u = maximum undiluted radwaste flow rate (gpm)

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachments 2 and 3.

Since the flow rate of undiluted liquid waste (120 gpm maximum) is insignificant relative to the circulating water flow rate (200,000 gpm per circulating water pump), the flow rate of undiluted liquid waste need not be considered when calculating the dilution water flow rate.

Since the maximum undiluted radwaste flow rate is used for calculating the fixed setpoint, a flow setpoint is not required--for the flow measuring device (0-FE-2199) in the effluent line--to verify compliance with Control 3.3.3.10.

A_{iLn} = specific activity limit for the release of radionuclide, i, to **UNRESTRICTED AREAS** (calculated in accordance with 10 CFR 20, Appendix B, Table II, Note 1 as described below; microcuries per milliliter).

In order to calculate a meaningful and accurate fixed setpoint, the specific radionuclides, i, chosen for calculating the fixed setpoint should correspond to those **DOMINANT** and **LESS DOMINANT RADIONUCLIDES** found in **TYPICAL RADWASTE RELEASES** from CCNPP.

Attachment 5 provides further guidance for determining the identity of those radionuclides, i, to be used to calculate a fixed setpoint.

Values for A_{iLn} shall be calculated, as described below, for each **DOMINANT RADIONUCLIDE** and for the collective total of all **LESS DOMINANT RADIONUCLIDEs**.

SPECIFIC ACTIVITY LIMIT FOR RADIONUCLIDE i

$A_{iLn} = (f_i)(A_{TLn})$	**Eq. 3L**

f_i = a fraction which represents the relative activity contribution of nuclide i to the average total effluent activity (unitless)

The average total effluent activity does not include tritium or dissolved and entrained noble gases.

This value may be obtained using the guidance provided on Attachment 5.

A_{TLn} = the sum of the total specific activities of all radionuclides found in a TYPICAL LIQUID RADWASTE RELEASE (microcuries/cm^3)

This value corresponds to 1 MPC.

A_{TLn} shall be calculated as shown below.

TOTAL SPECIFIC ACTIVITY EQUIVALENT TO 1 MPC

$\Sigma [(f_i)(A_{TLn}) / A_{iLt}] = 1$	**Eq. 4L**[1]

A_{iLt} = the specific activity limit for radionuclide, i, as obtained from 10 CFR 20, Appendix B, Table II, Column 2 (microcuries/cm^3)

For all the **DOMINANT RADIONUCLIDEs** found in **TYPICAL RADWASTE EFFLUENTS**, use the value from 10 CFR 20, Appendix B, Table II, Column 2.

For each of the **LESS DOMINANT RADIONUCLIDEs** found in **TYPICAL RADWASTE EFFLUENTS**, use 3E-8 microcuries per milliliter as the value for A_{iLt} (per 10 CFR 20, Appendix B, Note 2).

1 = the MPC limit (MPCs) for **UNRESTRICTED AREAs**

This value is based on the MPC limit from 10 CFR 20, Table II, Note 1.

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

[1] This formula has been derived from 10 CFR 20, Table II, Notes 1, 2, and 3.

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background prior to initiating the liquid release (cpm)

Instead of using an approximation of the detector background, a value of 0 cpm may be used as the detector background if so desired.

(3) Documenting the fixed setpoint for 0-RI-2201

(a) Whenever the fixed setpoint is calculated, the specific values chosen for each of the parameters shall be documented in accordance with EN-1-100.

(4) Changing the fixed setpoint for 0-RI-2201

(a) If the fixed setpoint calculated in accordance with equation 1L exceeds the maximum range of the monitor, the fixed setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(b) The fixed alarm and trip setpoint for 0-RI-2201 may be established at values lower than the maximum allowable setpoint, if desired.

(c) A setpoint change should be initiated whenever any of the parameters identified in equation 1L have changed.

(d) The fixed setpoint should not be changed unless one of the following occurs:

i) the relative activity[1] of any radionuclide in TYPICAL LIQUID EFFLUENTS, f_i, has changed by greater than 10%, and the new radionuclide mixture yields a fixed setpoint which is 10% (or more) lower than the current fixed setpoint,

ii) the minimum dilution water flow rate is not available for a liquid release,

iii) the values listed in 10 CFR 20, Table II, column 2 have changed,

iv) the radiation monitor has been recently calibrated, repaired, or otherwise altered, or

v) the monitor is not conservative in its function (see "Functions of 0-RE-2201" earlier in this section).

(e) EN-1-100 contains the administrative controls associated with changing and approving the fixed setpoint.

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[1] As determined in accordance with Attachment 5.

i) The adjustable alarm and trip setpoint for 0-RI-2201

(1) General information

(a) This setpoint is an adjustable setpoint. The adjustable setpoint is calculated and adjusted prior to each release from the liquid waste processing system.

(b) The adjustable setpoint is based on the specific activities of the radionuclides present in the undiluted liquid waste (as determined by radiochemical analysis per Control 4.11.1.1.1).

(c) Whenever the adjustable setpoint is exceeded, discharges from the liquid waste processing system will be manually suspended.

(d) See OI-17C, "Reactor Coolant Waste Processing System", or OI-17D, "Miscellaneous Waste Processing System," for a full list of operator **ACTION**s taken in response to this alarm.

(e) The adjustable setpoint corresponds to the maximum concentration of radionuclides anticipated or expected in discharges from the liquid waste processing system.

(f) The value for the adjustable setpoint is recorded on the liquid release permit in accordance with CHEMISTRY SECTION procedures.

(g) This alarm is not integral to the liquid release discharge monitor, as purchased from the supplier.

(h) This alarm is generated by the plant computer, which monitors output from 0-RI-2201, and provides an alarm to plant operators when the adjustable alarm and trip setpoint has been exceeded.

(i) A value for the adjustable alarm and trip setpoint shall be calculated prior to each release as shown below.

(2) Calculating the adjustable setpoint for 0-RI-2201

(a) The adjustable setpoint is based on the specific activities of the radionuclides in the undiluted liquid waste (as determined by radiochemical analysis per Control 4.11.1.1.1), and shall be calculated as shown below.

ADJUSTABLE SETPOINT FOR 0-RI-2201

$$S_{adj} \leq 1.50 \left[\sum (A_{iu}) (e_i) + Bkg \right] \qquad \text{Eq. 5L}$$

S_{adj} = the adjustable alarm and trip setpoint for 0-RI-2201 (cpm)

1.50 = a constant, actually a safety factor, which allows for fluctuations in radiation monitor response (unitless)

This safety factor helps ensure the release is not unnecessarily terminated due to (1) electronic anomalies which cause spurious monitor responses, (2) statistical fluctuations in disintegration rates, (3) statistical fluctuations in detector efficiencies, (4) errors associated with sample analysis, or (5) errors associated with monitor calibrations.[1]

A_{iu} = specific activity of radionuclide, i, in the undiluted waste stream (microcuries per milliliter)

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background (cpm)

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(3) Documenting the adjustable alarm and trip setpoint for 0-RI-2201

(a) Whenever the adjustable setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with approved CHEMISTRY SECTION procedures (e.g., CP-601).

(4) Changing the adjustable alarm and trip setpoint for 0-RI-2201

(a) In all cases, the adjustable alarm setpoint shall be set to a value which is less than or equal to the fixed setpoint.

(b) If the adjustable alarm and trip setpoint exceeds the maximum range of the monitor, the setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(c) CHEMISTRY SECTION procedures (e.g., CP-601) contain administrative controls associated with calculating and approving an adjustable setpoint.

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1 The "analysis errors" and "calibration errors" refer to errors which are within established quality assurance and quality control limits.

(d) Whenever 0-RI-2201 is operable, the calculated value for the adjustable setpoint shall be entered into the plant computer prior to each release from the liquid waste processing system.

(e) Plant Operating Instructions (e.g., OI-17C and OI-17D) contain administrative controls associated with entering the adjustable setpoint into the plant computer.

j) The low radiation alarm for 0-RI-2201

(1) This alarm is integral to the liquid release discharge monitor, as purchased from the supplier.

(2) The current value for the low alarm setpoint is specified in the CCNPP Alarm Manual.

(3) The low alarm setpoint may be used to determine the **OPERABILITY** of this monitor (per Control 4.3.3.10, **CHANNEL FUNCTIONAL TEST**).

(4) The alarm generated by the low alarm setpoint may be used to terminate a release in the event 0-RE-2201 fails (i.e., downscale failure or circuit failure) in accordance with Control 4.3.3.10.

(5) The low alarm setpoint calculations are not described in the ODCM.

(6) Changes to the low alarm setpoint are controlled by EN-1-100.

2. Steam Generator Blowdown Effluent Radiation Monitors (1/2-RE-4095)

a) General description of 1/2-RE-4095

(1) number of radiation elements: one per unit

(2) Type of radiation elements: in-line scintillation detectors

(3) output: analog

(4) Designations for radiation indicators

(a) 1-RI-4095

(b) 2-RI-4095

(5) units for radiation indicator: counts per minute

(6) supplier: Westinghouse

b) Functions of 1/2-RE-4095

(1) continuously measure the activity contained in steam generator blowdown effluent line (Control 3.3.3.10)

(2) continuously indicate (via 1/2-RI-4095) the activity of liquids contained in the steam generator blowdown effluent line (Control 3.3.3.10)

(3) alarm (via 1/2-RI-4095) prior to exceeding 10 CFR 20, Appendix B, Table II, Column 2 limits (per Control 3.11.1.1).

(4) automatically terminate steam generator blowdown releases to **UNRESTRICTED AREAS** when the radiation indicator (1/2-RI-4095) exceeds the fixed alarm setpoint

c) **OPERABILITY** of 1/2-RE-4095

(1) This monitor shall be operable (or have **OPERABILITY**) when it is capable of performing its specified function(s).

(2) The functions of this monitor are listed in section (b) above.

(3) It should be noted that if the steam generator blowdown processing system heat exchangers are bypassed, it is possible for blowdown flow to bypass 1/2-RE-4095. If blowdown flow is allowed to bypass 1/2-RE-4095, the minimum channels **OPERABILITY** requirement of Control 3.3.3.10 may not be satisfied.

d) Monitors equivalent to 1/2-RE-4095

(1) 1/2-RE-4014 is normally considered the PRIMARY MONITOR for measuring activity released via the steam generator blowdown processing system, and 1/2-RE-4095 is normally considered the BACKUP MONITOR for measuring activity released via the steam generator blowdown processing system.

(2) In the event PRIMARY MONITOR (1/2-RE-4014) is inoperable or otherwise unavailable, the BACKUP MONITOR (1/2-RE-4095) may fulfill the measuring, indicating, and alarming functions normally provided by the PRIMARY MONITOR.

(3) 1/2-RE-4014, Steam Generator Blowdown Tank Radiation Monitor is considered to be "equivalent" monitor to 1/2-RE-4095 as specified below.

(4) 1/2-RE-4014, Steam Generator Blowdown Tank Radiation Monitor, may perform measurement, indication, alarm, and isolation functions (see "Functions of 1/2-RE-4014" earlier in this section) which limit the concentration of radioactive materials released to **UNRESTRICTED AREAS** in accordance with Control 3.11.1.1 as long as the following conditions are satisfied:

(a) the **OPERABILITY** of 1/2-RE-4014 must be demonstrated in accordance with Control 4.3.3.10, Table 3.3-13(1b), and

(b) the blowdown ion exchangers are isolated (or the decontamination factors for all radionuclides are verified to be greater than or equal to one for the duration of the release), and

(c) the specific activities of radionuclides in the blowdown tank radiation monitor are representative of the activities of the radionuclides in the blowdown effluent line.

e) Radiological effluent controls for 1/2-RE-4095

(1) Steam generator blowdown releases via this pathway may continue if any one of the following two conditions are satisfied (per Control 3.3.3.10):

(a) A steam generator blowdown monitor (either 1/2-RE-4095 or 1/2-RE-4014) is **OPERABLE** (see "**OPERABILITY** of 1/2-RE-4095"; and "**OPERABILITY** of 1/2-RE-4014" earlier in this section) AND the alarm and trip setpoint for this monitor is set to ensure the concentrations of radioactive materials released in liquid effluents to **UNRESTRICTED AREAS** are less than the concentrations specified in 10 CFR 20, Appendix B, Table II, Column 2, or

(b) a grab sample is collected AND analyzed in accordance with the provisions described below:

i) analysis shall determine EITHER the gross gamma activity or gross beta activity of the sample

ii) sampling and analysis shall be completed at least once per 12 hours if the specific activity of the steam generator blowdown is greater than 0.01 microcuries per gram IODINE DOSE EQUIVALENT

iii) sampling and analysis shall be completed at least once per 48 hours if the specific activity of the steam generator blowdown is less than or equal to 0.01 microcuries per gram IODINE DOSE EQUIVALENT

iv) the LOWER LIMITS OF DETECTION (LLDs) for the gross gamma and/or gross beta analyses are in accordance with the LLDs of Control 4.11.1.1.1

f) Surveillances for 1/2-RE-4095

(1) Control 4.3.3.10 requires demonstrating the **OPERABILITY** of 1/2-RE-4095 by satisfying the checks, calibrations, and tests listed below

(a) **CHANNEL CHECK** within the past 24 hours

(b) **SOURCE CHECK** prior to each release

(c) **CHANNEL CALIBRATION** within the past 18 months

(d) **CHANNEL FUNCTIONAL TEST** within the past 6 six months

g) Setpoints for 1/2-RI-4095

(1) There are three radiation alarm setpoints associated with the Steam Generator Blowdown Effluent Radiation Monitor.

(a) 1/2-RI-4095 fixed high radiation alarm and automatic control trip setpoint

(b) 1/2-RI-4095 adjustable plant computer high radiation alarm and manual control trip setpoint

(c) 1/2-RI-4095 low radiation alarm setpoint

(2) In order to simplify the setpoint terminology, eliminate ambiguity, and minimize the possibility of misinterpretation, the ODCM will refer to these setpoints as follows:

(a) The 1/2-RI-4095 fixed high radiation alarm and automatic control trip setpoint will be referred to as the fixed setpoint.

(b) The 1/2-RI-4095 adjustable plant computer high radiation alarm and manual control trip setpoint will be referred to as the adjustable setpoint.

(c) The 1/2-RI-4095 low radiation alarm setpoint will be referred to as the low setpoint.

(3) Each of these alarm setpoints are described below.

h) The fixed setpoint for 1/2-RI-4095

(1) General information

(a) This setpoint is considered to be a fixed alarm and trip setpoint. The fixed setpoint is not adjusted for each release.

(b) Whenever the fixed setpoint is exceeded, discharges from the steam generator blowdown processing system to **UNRESTRICTED AREAS** will be automatically suspended.

(c) The fixed setpoint corresponds to the maximum concentration of radionuclides allowed in liquid waste discharged from the steam generator blowdown processing system.

(d) The current value for the fixed setpoint is specified in the CCNPP Alarm Manual.

(e) The CCNPP Alarm Manual refers to this alarm and trip setpoint as the 1/2-RI-4095 Steam Generator Blowdown Recovery High Alarm.

(f) This setpoint is integral to the steam generator blowdown discharge monitor, as purchased from the manufacturer.

(g) The fixed setpoint is administratively controlled by EN-1-100.

(h) The fixed alarm and trip setpoint shall be calculated as described below[1].

(2) Calculating the fixed setpoint for 1/2-RI-4095

(a) The fixed alarm setpoint for 1/2-RI-4095 shall be calculated in accordance with equation 1L.

FIXED SETPOINT FOR 1/2-RI-4095

$$S_{fix} \leq K_{sf} \{ (F_d/F_u) \Sigma [(A_{iLn}) (e_i)] + Bkg \} \quad \text{Eq. 1L}^{2}$$

S_{fix} = the fixed alarm and trip setpoint for 1/2-RI-4095 (cpm)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP activity limit to the 10 CFR 20 MPC limit (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the fixed setpoint is always less than or equal to the limits of 10 CFR 20.

A safety factor of 1.00 will yield a fixed setpoint which corresponds to 1.0 MPC.

A safety factor of 0.500 will yield a fixed setpoint which corresponds to 0.5 MPCs.

It is recommended that a safety factor of 0.5 be used for calculating the fixed setpoint, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The use of a "safety margin" is in accordance with the provisions of NUREG-0133, section 4.1.1, which states, "The alarm and trip setpoints ... should correspond to a value(s) which represents a safe margin of assurance that the instantaneous liquid release limit of 10 CFR Part 20 is not exceeded."

This safety margin will prevent minor fluctuations in the nominal circulating water flow rate (and other statistical aberrations) from adversely impacting the calculated fixed setpoint.

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1 Per Control 3.3.3.10.

2 Equation 1L has been derived from NUREG-0133, Addendum, page AA-1.

F_d = the dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

the dilution water flow rate is actually the sum of the minimum circulating water flow rate, the minimum salt water flow rate, and the maximum undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

DILUTION WATER FLOW RATE

$$F_d = F_{cw} + F_{sw} + F_u \qquad \textbf{Eq. 2L}$$

F_{cw} = the minimum circulating water system flow rate necessary to conduct liquid releases

A minimum of two circulating water pumps (one circulating water pump per conduit accepting radwaste) shall be operable when discharging liquid radwaste -- via this monitor -- to outfall 001.

The flow rate of each circulating water pump is 200,000 gallons per minute.

Additional circulating water pumps may be required as specified in approved CHEMISTRY SECTION procedures.

If a release of liquid radwaste from the steam generator blowdown system is to be conducted using less than minimum circulating water flow rate necessary to conduct liquid releases, the release shall not be allowed until a new setpoint has been calculated and entered into 1/2-RI-4095, or otherwise comply with the **ACTION** requirements of Control 3.3.3.10.

F_{sw} = the minimum salt water system flow rate necessary to release liquid radwaste

If the minimum salt water flow rate available for liquid releases is unknown, difficult to predict, or may decrease during a liquid release, a minimum salt water flow rate of 0 gpm should be used when calculating the fixed setpoint.

The maximum flow rate for one salt water pump is listed on Attachment 2.

If a release of liquid radwaste is to be conducted using less than the minimum salt water flow rate necessary to conduct liquid releases, the release shall not be allowed until a new setpoint has been calculated and entered into 1/2-RI-4095, or otherwise comply with the **ACTION** requirements of Control 3.3.3.10.

F_u = maximum undiluted radwaste flow rate (gpm)

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachments 2 and 3.

Since the steam generator blowdown flow rate (225 gpm maximum) is insignificant relative to the circulating water flow rate (200,000 gpm per circulating water pump), the steam generator blowdown flow rate need not be considered when calculating the dilution water flow rate.

Since the maximum undiluted radwaste flow rate is used for calculating the fixed setpoint, a flow setpoint is not required--for the flow measuring device in the effluent line--to verify compliance with Control 3.3.3.10.

A_{iLn} = specific activity limit for the release of radionuclide, i, to **UNRESTRICTED AREAS** (calculated in accordance with 10 CFR 20, Appendix B, Table II, Note 1 as described below; microcuries per milliliter)

In order to calculate a meaningful and accurate fixed setpoint, the specific radionuclides, i, chosen for calculating the fixed setpoint should correspond to those **DOMINANT** and **LESS DOMINANT RADIONUCLIDES** found in **TYPICAL RADWASTE RELEASES** from CCNPP.

Attachment 5 provides further guidance for determining which radionuclides, i, should be used to calculate a fixed setpoint.

Values for A_{iLn} shall be calculated, as described below, for each **DOMINANT RADIONUCLIDE** and for the collective total of all **LESS DOMINANT RADIONUCLIDEs**.

SPECIFIC ACTIVITY LIMIT FOR EACH RADIONUCLIDE i

$$A_{iLn} = (f_i)(A_{TLn}) \quad \textbf{Eq. 3L}$$

f_i = a fraction which represents the relative activity contribution of nuclide i to the average total effluent activity (unitless)

The average total effluent activity does not include tritium or dissolved and entrained noble gases.

This value may be obtained using the guidance provided on Attachment 5.

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

A_{TLn} = the sum of the total specific activities of all radionuclides found in a TYPICAL LIQUID RADWASTE RELEASE (microcuries/cm^3)

This value corresponds to 1 MPC.

A_{TLn} shall be calculated as shown below.

TOTAL SPECIFIC ACTIVITY EQUIVALENT TO 1 MPC

$$\Sigma [(f_i) (A_{TLn}) / A_{iLt}] = 1 \qquad \text{Eq. 4L}$$ [1]

A_{iLt} = the specific activity limit for radionuclide, i, as obtained from 10 CFR 20, Appendix B, Table II, Column 2 (microcuries/cm^3)

For all the **DOMINANT RADIONUCLIDEs** found in **TYPICAL RADWASTE EFFLUENTS**, use the value from 10 CFR 20, Appendix B, Table II, Column 2.

For each of the **LESS DOMINANT RADIONUCLIDEs** found in **TYPICAL RADWASTE EFFLUENTS**, use 3E-8 microcuries per milliliter as the value for A_{iLt} (per 10 CFR 20, Appendix B, Note 2).

1 = the MPC limit (MPCs) for **UNRESTRICTED AREAs**

This value is based on the MPC limit from 10 CFR 20, Table II, Note 1.

Bkg = an approximation of the detector background (cpm)

Instead of using an approximation of the detector background, a value of 0 cpm may be used as the detector background if so desired.

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(3) Documenting the fixed setpoint for 1/2-RI-4095

(a) Whenever the fixed setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with EN-1-100.

(4) Changing the fixed setpoint for 1/2-RI-4095

(a) If the fixed setpoint calculated in accordance with equation 1L exceeds the maximum range of the monitor, the fixed setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(b) The fixed setpoint may be established at values lower than the maximum allowable setpoint, if desired.

(c) A setpoint change should be initiated whenever any of the parameters identified in equation 1L have changed.

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1 This formula has been derived from 10 CFR 20, Table II, Notes 1, 2, and 3.

(d) The fixed setpoint should not be changed unless one of the following occurs:

i) the relative activity[1] of any radionuclide in TYPICAL LIQUID EFFLUENTS has changed by greater than 10%, and the new radionuclide mixture yields a fixed setpoint which is 10% (or more) lower than the current fixed setpoint,

ii) the minimum dilution water flow rate is not available for a liquid release,

iii) the values listed in 10 CFR 20, Table II, column 2 have changed,

iv) the radiation monitor has been recently calibrated, repaired, or otherwise altered, or

v) the monitor is not conservative in its function (see "Functions of 1/2-RE-4095" earlier in this section).

(e) EN-1-100 contains the administrative controls associated with changing and approving the fixed alarm and trip setpoint.

i) The adjustable alarm and trip setpoint for 1/2-RI-4095

(1) General information

(a) This setpoint is an adjustable setpoint. The adjustable setpoint is calculated and adjusted prior to each release from the steam generator blowdown processing system.

(b) The adjustable setpoint is based on the specific activities of the radionuclides present in the undiluted liquid waste (as determined by radiochemical analysis per Control 4.11.1.1.1).

(c) Whenever the adjustable setpoint is exceeded, discharges from the steam generator blowdown processing system will be manually suspended.

(d) See OI-8A for a full list of operator actions taken in response to this alarm.

(e) The adjustable setpoint corresponds to the maximum concentration of radionuclides anticipated or expected in discharges from the steam generator blowdown processing system.

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1 As determined in accordance with Attachment 5.

(f) The value for the adjustable setpoint is recorded on the liquid release permit in accordance with CHEMISTRY SECTION procedures.

(g) This alarm is not integral to the steam generator blowdown effluent monitor, as purchased from the supplier.

(h) This alarm is generated by the plant computer which monitors output from 1/2-RI-4095, and provides an alarm to plant operators when the adjustable alarm and trip setpoint has been exceeded.

(i) A value for the adjustable alarm and trip setpoint shall be calculated prior to each release as shown below.

(2) Calculating the adjustable setpoint for 1/2-RI-4095

(a) The adjustable setpoint is based on the specific activities of radionuclides in the undiluted liquid waste (as determined by radiochemical analysis per Control 4.11.1.1.1), and shall be calculated as shown below.

ADJUSTABLE SETPOINT FOR 1/2-RI-4095

$$S_{adj} \leq 1.50 \left[\sum (A_{iu})(e_i) + Bkg\right] \quad \text{Eq. 5L}^{1}$$

Where,

S_{adj} = the adjustable alarm and trip setpoint for 1/2-RI-4095 (cpm)

1.50 = a constant, actually a safety factor, which allows for fluctuation in radiation monitor response (unitless)

This safety factor helps ensure the release is not unnecessarily terminated due to (1) electronic anomalies which cause spurious monitor responses, (2) statistical fluctuations in disintegration rates, (3) statistical fluctuations in detector efficiencies, (4) errors associated with sample analysis, or (5) errors associated with monitor calibrations.[2]

A_{iu} = specific activity of radionuclide, i, in the undiluted waste stream (microcuries per milliliter)

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

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1 This formula may be derived from NUREG-0133, Addendum, page AA-1.

2 The "analysis errors" and "calibration errors" refer to errors which are within established quality assurance and quality control limits.

Bkg = an approximation of the detector background prior to initiating the liquid release (cpm)

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(3) Documenting the adjustable setpoint, 1/2-RI-4095

(a) Whenever the adjustable setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with approved CHEMISTRY SECTION procedures.

(4) Changing the adjustable alarm and trip setpoint for 1/2-RI-4095

(a) If the adjustable setpoint exceeds the maximum range of the monitor, the setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(b) In all cases, the adjustable setpoint shall be set to a value which is less than or equal to the fixed setpoint.

(c) Chemistry procedures contain administrative controls associated with calculating and approving an adjustable setpoint.

(d) Whenever 1/2-RI-4095 is operable, the calculated value for the adjustable setpoint shall be entered into the plant computer prior to each release from the steam generator blowdown processing system.

(e) Plant Operating Instructions (e.g., OI-8A) contain administrative controls associated with entering the adjustable setpoint in the plant computer.

j) The low setpoint for 1/2-RI-4095

(1) This alarm is integral to the steam generator blowdown effluent monitor, as purchased from the manufacturer.

(2) The current value for the low alarm setpoint is specified in the CCNPP Alarm Manual.

(3) The low alarm setpoint may be used to determine **OPERABILITY** of this monitor (per Control 4.3.3.10, **CHANNEL FUNCTIONAL TEST**).

(4) The low alarm setpoint may be used to terminate a release in the event 1/2-RE-4095 fails (i.e., downscale failure or circuit failure).

(5) Changes to the low alarm setpoint are controlled by EN-1-100.

(6) The ODCM does not address the calculations associated with the low alarm setpoint.

3. Steam Generator Blowdown Tank Radiation Monitors (1/2-RE-4014)

a) General description of 1/2-RE-4014

(1) number of radiation elements: one per unit

(2) Type of radiation element: off-line scintillation detector

(3) output: analog

(4) Designations for radiation indicators

(a) 1-RI-4014

(b) 2-RI-4014

(5) units for radiation indicator: counts per minute

(6) supplier: Westinghouse

(7) A 3 gallon per minute sample is cooled, passed through the detector, and pumped back into the steam generator blowdown tank.

b) Functions of 1/2-RE-4014

(1) continuously measure the activity contained in an off-line sample of the steam generator blowdown tank (Control 3.3.3.10)

(2) continuously indicate (via 1/2-RI-4014) the specific activity in an off-line sample of the steam generator blowdown tank (Control 3.3.3.10)

(3) alarm (via 1/2-RI-4014) prior to exceeding the 10 CFR 20, Appendix B, Table II, Column 2 limits (per Control 3.11.1.1)

(4) automatically terminate steam generator blowdown releases to **UNRESTRICTED AREAS** when the radiation indicator (1/2-RI-4014) exceeds the fixed alarm setpoint.

c) **OPERABILITY** of 1/2-RE-4014

(1) This monitor shall be operable (or have **OPERABILITY**) when it is capable of performing its specified function(s)

(2) The functions of this monitor are listed in section (b) above.

d) Monitors equivalent to 1/2-RE-4014

(1) 1/2-RE-4014 is normally considered the PRIMARY MONITOR for measuring activity released via the steam generator blowdown processing system, and 1/2-RE-4095 is normally considered the BACKUP MONITOR for measuring activity released via the steam generator blowdown processing system.

(2) In the event PRIMARY MONITOR (1/2-RE-4014) is inoperable or otherwise unavailable, the BACKUP MONITOR (1/2-RE-4095) may fulfill the measuring, indicating, and alarming functions normally provided by the PRIMARY MONITOR.

(3) 1/2-RE-4095, Steam Generator Blowdown Effluent Radiation Monitor, performs measurement, indication, alarm, and isolation functions equivalent to 1/2-RE-4014 (see "Function of 1/2-RE-4014" earlier in this section), unless 1/2-RE-4095 is bypassed as described below.

(4) It should be noted that if the steam generator blowdown processing system heat exchangers are bypassed, it is possible for blowdown flow to bypass 1/2-RE-4095 (the steam generator blowdown effluent radiation monitor). If blowdown flow is allowed to bypass 1/2-RE-4095, the minimum channels **OPERABILITY** requirement of Control 3.3.3.10 may not be satisfied (unless 1/2-RE-4014 is operable).

e) Radiological effluent controls for 1/2-RE-4014

(1) Steam generator blowdown releases via this pathway may continue if any one of the following two conditions are satisfied (per Control 3.3.3.10):

(a) A steam generator blowdown monitor (either 1/2-RE-4095 or 1/2-RE-4014) is **OPERABLE** (see "**OPERABILITY** of 1/2-RE-4095"; and "**OPERABILITY** of 1/2-RE-4014" earlier in this section) AND the alarm and trip setpoint for this monitor is set to ensure the concentrations of radioactive materials released in liquid effluents to **UNRESTRICTED AREAS** are less than the concentrations specified in 10 CFR 20, Appendix B, Table II, Column 2, or

(b) a grab sample is collected AND analyzed in accordance with the provisions described below:

i) analysis shall determine EITHER the gross gamma activity or gross beta activity of the sample

ii) sampling and analysis shall be completed at least once per 12 hours if the specific activity of the steam generator blowdown is greater than 0.01 microcuries per gram IODINE DOSE EQUIVALENT

iii) sampling and analysis shall be completed at least once per 48 hours if the specific activity of the steam generator blowdown is less than or equal to 0.01 microcuries per gram IODINE DOSE EQUIVALENT

iv) the LOWER LIMITS OF DETECTION for the gross gamma and/or gross beta analyses are in accordance with the LLDs of Control 4.11.1.1.1

f) Surveillances for 1/2-RE-4014

(1) Control 4.3.3.10 requires demonstrating the **OPERABILITY** of 1/2-RE-4014 by satisfying the checks, calibrations, and tests listed below

(a) **CHANNEL CHECK** within the past 24 hours

(b) **SOURCE CHECK** prior to each release

(c) **CHANNEL CALIBRATION** within the past 18 months

(d) **CHANNEL FUNCTIONAL TEST** within the past 6 six months

g) Setpoints for 1/2-RI-4014

(1) There are three radiation alarm setpoints associated with the Steam Generator Blowdown Tank Radiation Monitor

(a) 1/2-RI-4014 fixed high radiation alarm and automatic control trip setpoint

(b) 1/2-RI-4014 adjustable plant computer high radiation alarm and manual control trip setpoint

(c) 1/2-RI-4014 low radiation alarm setpoint

(2) In order to simplify the setpoint terminology, eliminate ambiguity, and minimize the possibility of misinterpretation, the ODCM will refer to these setpoints as follows:

(a) The 1/2-RI-4014 fixed high radiation alarm and automatic control trip setpoint will be referred to as the fixed setpoint.

(b) The 1/2-RI-4014 adjustable plant computer high radiation alarm and manual control trip setpoint will be referred to as the adjustable setpoint.

(c) The 1/2-RI-4014 low radiation alarm setpoint will be referred to as the low setpoint.

(3) Each of these alarms is described below.

h) The fixed alarm and trip setpoint for 1/2-RI-4014

(1) General Information

(a) This alarm and trip setpoint is considered to be a fixed setpoint. The fixed setpoint is not adjusted for each release.

(b) Whenever the fixed setpoint is exceeded, discharges from the steam generator blowdown processing system to **UNRESTRICTED AREAS** will be automatically suspended.

(c) The fixed setpoint corresponds to the maximum concentration of radionuclides allowed in liquid waste discharged from the steam generator blowdown processing system.

(d) The current value for the fixed alarm and trip setpoint is specified in the CCNPP Alarm Manual.

(e) The CCNPP Alarm Manual refers to the fixed setpoint as the 1/2-RI-4014 Steam Generator Blowdown High Alarm.

(f) The fixed setpoint is integral to the liquid release discharge monitor, as purchased from the manufacturer.

(g) The fixed alarm and trip setpoint is administratively controlled by EN-1-100.

(h) The fixed alarm and trip setpoint shall be calculated as described below[1].

(2) Calculating the fixed alarm and trip setpoint for 1/2-RI-4014

(a) The fixed alarm and trip setpoint for 1/2-RI-4014 shall be calculated as follows:

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1 Per Control 3.3.3.10.

FIXED ALARM AND TRIP SETPOINT FOR 1/2-RI-4014

$$S_{fix} \leq K_{sf} \{ (F_d / F_u) \Sigma [(A_{iLn}) (e_i)] + Bkg \} \qquad \textbf{Eq. 1L}^{1}$$

Where,

S_{fix} = the fixed alarm and trip setpoint for 1/2-RI-4014 (cpm)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP activity limit to the 10 CFR 20 MPC limit (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the fixed setpoint is always less than or equal to the limits of 10 CFR 20.

A safety factor of 1.00 will yield a fixed setpoint which corresponds to 1.0 MPC.

A safety factor of 0.500 will yield a fixed setpoint which corresponds to 0.5 MPCs.

It is recommended that a safety factor of 0.5 be used for calculating the fixed setpoint, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The use of a "safety margin" is in accordance with the provisions of NUREG-0133, SECTION 4.1.1, which states, "The alarm and trip setpoints ... should correspond to a value(s) which represents a safe margin of assurance that the instantaneous liquid release limit of 10 CFR Part 20 is not exceeded."

This safety margin will prevent minor fluctuations in the nominal circulating water flow rate (and other statistical aberrations) from adversely impacting the calculated fixed setpoint.

F_d = the dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The dilution water flow rate is actually the sum of the minimum circulating water flow rate, the minimum salt water flow rate, and the maximum undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

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[1] Equation 1L has been derived from NUREG-0133, Addendum, page AA-1.

DILUTION WATER FLOW RATE

F_d	=	$F_{cw} + F_{sw} + F_u$	Eq. 2L

F_{cw} = the minimum circulating water system flow rate necessary to conduct liquid releases

A minimum of two circulating water pumps (one circulating water pump per conduit accepting radwaste) shall be operable when discharging liquid radwaste -- via this monitor -- to outfall 001.

The flow rate of each circulating water pump is 200,000 gallons per minute.

Additional circulating water pumps may be required as specified in approved CHEMISTRY SECTION procedures.

If a release of liquid radwaste from the steam generator blowdown system is to be conducted using less than minimum circulating water flow rate necessary to conduct liquid releases, the release shall not be allowed until a new setpoint has been calculated and entered into 1/2-RI-4014, or otherwise comply with the **ACTION** requirements of Control 3.3.3.10.

F_{sw} = the minimum salt water system flow rate necessary to release liquid radwaste

If the minimum salt water flow rate available for liquid releases is unknown, difficult to predict, or may decrease during a liquid release, a minimum salt water flow rate of 0 gpm should be used when calculating the fixed setpoint.

The maximum flow rate for one salt water pump is listed on Attachment 2.

If a release of liquid radwaste is to be conducted using less than the minimum salt water flow rate necessary to conduct liquid releases, the release shall not be allowed until a new setpoint has been calculated and entered into 1/2-RI-4014, or otherwise comply with the **ACTION** requirements of Control 3.3.3.10.

F_u = maximum undiluted radwaste flow rate (gpm)

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachments 2 and 3.

Since the steam generator blowdown flow rate (225 gpm maximum) is insignificant relative to the circulating water flow rate (200,000 gpm per circulating water pump), the steam generator blowdown flow rate need not be considered when calculating the dilution water flow rate.

Since the maximum undiluted radwaste flow rate is used for calculating the fixed setpoint, a flow setpoint is not required--for the flow measuring device in the effluent line--to verify compliance with Control 3.3.3.10.

A_{iLn} = specific activity limit for the release of radionuclide, i, to **UNRESTRICTED AREAS** (calculated in accordance with 10 CFR 20, Appendix B, Table II, Note 1 as described below; microcuries per milliliter)

In order to calculate a meaningful and accurate fixed setpoint, the specific radionuclides, i, chosen for calculating the fixed setpoint should correspond to those **DOMINANT** and **LESS DOMINANT RADIONUCLIDES** found in **TYPICAL RADWASTE RELEASES** from CCNPP.

Attachment 5 provides further guidance for determining which radionuclides, i, should be used to calculate a fixed setpoint.

Values for A_{iLn} shall be calculated, as described below, for each **DOMINANT RADIONUCLIDE** and for the collective total of all **LESS DOMINANT RADIONUCLIDEs**.

SPECIFIC ACTIVITY LIMIT FOR RADIONUCLIDE, i

$$A_{iLn} = (f_i)(A_{TLn}) \qquad \textbf{Eq. 3L}$$

f_i = a fraction which represents the relative activity contribution of nuclide i to the average total effluent activity (unitless)

The average total effluent activity does not include tritium or dissolved and entrained noble gases.

This value may be obtained using the guidance provided on Attachment 5.

A_{TLn} = the sum of the total specific activities of all radionuclides found in a TYPICAL LIQUID RADWASTE RELEASE (microcuries/cm^3)

This value corresponds to 1 MPC.

Calculate the value of A_{TLn} as shown below.

TOTAL SPECIFIC ACTIVITY CORRESPONDING TO 1 MPC AT THE **SITE BOUNDARY**

$$\Sigma[(f_i)(A_{TLn})/A_{iLt}] = 1 \qquad \textbf{Eq. 4L}^{1}$$

A_{iLt} = the specific activity limit for radionuclide, i, as obtained from 10 CFR 20, Appendix B, Table II, Column 2 (microcuries/cm^3)

For all the **DOMINANT RADIONUCLIDEs** found in **TYPICAL RADWASTE EFFLUENTS**, use the value from 10 CFR 20, Appendix B, Table II, Column 2.

For each of the **LESS DOMINANT RADIONUCLIDEs** found in **TYPICAL RADWASTE EFFLUENTS**, use 3E-8 microcuries per milliliter as the value for A_{iLt} (per 10 CFR 20, Appendix B, Note 2).

1 This formula has been derived from 10 CFR 20, Table II, Notes 1, 2, and 3.

1 = the MPC limit (MPCs) for **UNRESTRICTED AREAs**

This value is based on the MPC limit from 10 CFR 20, Table II, Note 1.

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background (cpm)

Instead of using an approximation of the detector background, a value of 0 cpm may be used as the detector background if so desired.

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(3) Documenting the fixed setpoint for 1/2-RI-4014

(a) Whenever the fixed setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with EN-1-100.

(4) Changing the fixed setpoint for 1/2-RI-4014

(a) If the fixed setpoint calculated in accordance with equation 1L exceeds the maximum range of the monitor, the fixed setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(b) The fixed alarm setpoint may be established at values lower than the maximum allowable setpoint, if desired.

(c) A setpoint change should be initiated whenever any of the parameters identified in equation 1L (identified in this section of the ODCM) have changed.

(d) The fixed setpoint should not be changed unless one of the following occurs:

i) the relative activity[1] of any radionuclide in TYPICAL LIQUID EFFLUENTS has changed by greater than 10%, and the new radionuclide mixture yields a fixed setpoint which is 10% (or more) lower than the current fixed setpoint,

ii) the minimum dilution water flow rate is not available for a liquid release,

iii) the values listed in 10 CFR 20, Table II, column 2 have changed,

iv) the radiation monitor has been recently calibrated, repaired, or otherwise altered, or

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1 As determined in accordance with Attachment 5.

v) the monitor is not conservative in its function (see "Functions of 1/2-RE-4014" earlier in this section).

(e) EN-1-100 contains the administrative controls associated with changing and approving the fixed setpoint.

i) The adjustable alarm and trip setpoint for 1/2-RI-4014

(1) General information

(a) This setpoint is an adjustable setpoint. The adjustable setpoint is calculated and adjusted prior to each release from the steam generator blowdown processing system.

(b) The adjustable setpoint is based on the specific activities of the radionuclides present in the undiluted liquid waste (as determined by radiochemical analysis per Control 4.11.1.1.1).

(c) Whenever the adjustable setpoint is exceeded, discharges from the steam generator blowdown processing system will be manually suspended.

(d) See OI-8A for a full list of operator **ACTION**s taken in response to this alarm.

(e) The adjustable setpoint corresponds to the maximum concentration of radionuclides anticipated or expected in discharges from the steam generator blowdown processing system.

(f) The value for the adjustable setpoint is recorded on the liquid release permit in accordance with CHEMISTRY SECTION procedures.

(g) This alarm is not integral to the steam generator blowdown tank radiation monitor, as purchased from the supplier.

(h) This alarm is generated by the plant computer which monitors output from 1/2-RI-4014, and provides an alarm to plant operators when the 1/2-RI-4014 adjustable setpoint has been exceeded.

(i) A value for the adjustable alarm and trip setpoint shall be calculated prior to each release as shown below.

(2) Calculating the adjustable setpoint for 1/2-RI-4014

(a) The adjustable alarm and trip setpoint is based on the specific activity of the radionuclides in the undiluted liquid waste (as determined by radiochemical analysis per Control 4.11.1.1.1), and the alarm and trip setpoint is calculated as shown below.

ADJUSTABLE ALARM AND TRIP SETPOINT FOR 1/2-RI-4014

$$S_{adj} \leq 1.50\,[\Sigma\,(A_{iu})\,(e_i) + Bkg] \qquad \text{Eq. 5L}^1$$

S_{adj} = the adjustable alarm and trip setpoint for 1/2-RI-4014 (cpm)

1.50 = a constant, actually a safety factor, which allows for fluctuation in radiation monitor response (unitless)

This safety factor helps ensure the release is not unnecessarily terminated due to (1) electronic anomalies which cause spurious monitor responses, (2) statistical fluctuations in disintegration rates, (3) statistical fluctuations in detector efficiencies, (4) errors associated with sample analysis, and (5) errors associated with monitor calibrations.[2]

A_{iu} = specific activity of radionuclide, i, in the undiluted waste stream (microcuries per milliliter)

e_i = absolute detector efficiency for nuclide, i (cpm/microcuries per milliliter)

The detector efficiency for each radionuclide may be calculated from data collected during calibration of the radiation monitor.

Bkg = an approximation of the detector background (cpm)

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(3) Documenting the adjustable setpoint for 1/2-RI-4014

(a) Whenever the adjustable setpoint is calculated, the specific values chosen for each of the variables shall be documented in accordance with approved CHEMISTRY SECTION procedures.

(4) Changing the adjustable setpoint for 1/2-RI-4014

(a) If the adjustable setpoint exceeds the maximum range of the monitor, the setpoint shall be adjusted to a value which falls within the normal operating range of the monitor.

(b) In all cases, the adjustable setpoint shall be set to a value which is less than or equal to the fixed setpoint.

(c) Chemistry procedures contain administrative controls associated with calculating and approving an adjustable setpoint.

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1 Equation 5L has been derived from NUREG-0133, Addendum, page AA-1.

2 The "analysis errors" and "calibration errors" refer to errors which are within established quality assurance and quality control limits.

(d) Whenever 1/2-RI-4014 is operable, the calculated value for the adjustable setpoint shall be entered into the plant computer prior to each release from the steam generator blowdown and processing system.

(e) Plant Operating Instructions (e.g., OI-8A) contain administrative controls associated with entering the adjustable setpoint in the plant computer.

j) The low alarm setpoint for 1/2-RI-4014

(1) This alarm is integral to the steam generator blowdown tank monitor, as purchased from the manufacturer.

(2) The current value for the low alarm setpoint is specified in the CCNPP Alarm Manual.

(3) Changes to the low alarm setpoint are controlled by EN-1-100.

(4) The ODCM does not address the calculations associated with the low alarm setpoint.

LIMITS ON RADIONUCLIDE CONCENTRATIONS IN LIQUID EFFLUENTS

1. Introduction

a) 10 CFR 20, Appendix B, specifies concentration limits associated with the release of radioactive materials to **UNRESTRICTED AREAS**.

b) Radiological effluent controls have been established to implement the requirements of 10 CFR 20.

c) These radiological effluent controls are described below.

2. Radiological Effluent Controls

a) The concentration of radioactive material released in liquid effluents to **UNRESTRICTED AREAS** shall be limited to the concentrations specified in 10 CFR 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases (per Control 3.11.1.1).

b) It should be noted that NUREG-0133 specifies that the concentration of radioactive materials in liquid effluents to **UNRESTRICTED AREAS** shall be limited to 2 E-4 microcuries per milliliter for dissolved or entrained noble gases, this control has not been incorporated into the CCNPP Technical Specifications, and as a result, the ODCM does not include calculations for same.

c) The routine surveillances which are performed to verify compliance with these radiological effluent controls are described below.

3. Surveillance Requirements

a) Routine surveillances for **BATCH RELEASEs**

(1) Sample each "BATCH" of radioactive liquid waste prior to its release (per Control 4.11.1.1.1).

(2) Determine the concentrations of principle gamma emitters (including, but not limited to, Mn-54, Fe-59, Co-58, Co-60, Zn-65, Cs-134, Cs-137, Ce-141, I-131, Mo-99, and Ce-144) in the "BATCH" sample, prior to the "BATCH" release (per Control 4.11.1.1.1).

(3) Determine the concentration of tritium contained in a monthly **COMPOSITE SAMPLE** at least once per month (per Control 4.11.1.1.1).

(4) Determine the concentrations of Sr-89 and Sr-90 contained in a quarterly **COMPOSITE SAMPLE** at least once per quarter (per Control 4.11.1.1.1).

(5) Calculate the concentrations of radionuclides in the receiving waters at the point the liquid radioactive waste is released to **UNRESTRICTED AREAS** (per Control 4.11.1.1.2).

b) Routine surveillances for **CONTINUOUS RELEASEs**

(1) Sample **CONTINUOUS RELEASEs** of radioactive liquid waste at least once per month (per Control 4.11.1.1.1).

(2) Determine the concentrations of principle gamma emitters (including, but not limited to, Mn-54, Fe-59, Co-58, Co-60, Zn-65, Cs-134, Cs-137, Ce-141, I-131, Mo-99, and Ce-144) in the undiluted, CONTINUOUS waste stream sample (per Control 4.11.1.1.1).

(3) Calculate the concentrations of radionuclides in the receiving waters at the point the liquid radioactive waste is released to **UNRESTRICTED AREAS** (per Control 4.11.1.1.2).

4. Responsible Plant Organizations

a) The CHEMISTRY SECTION is responsible for performing the sampling, analysis, and calculations required by the above surveillances.

b) The conditions which initiate the required surveillances are contained in the following section.

5. Initiating Conditions

a) The surveillances for **BATCH RELEASEs** shall be performed prior to each **BATCH RELEASE**[1] and may be performed more often as specified in approved CHEMISTRY SECTION procedures.

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[1] Per Control 4.11.1.1.1.

b) The surveillances for **CONTINUOUS RELEASE**s shall be performed at least monthly[1] (until the **CONTINUOUS RELEASE** has been terminated) and, if activity is identified[2] in the CONTINUOUS waste stream, the surveillances may be performed more often as specified in approved CHEMISTRY SECTION procedures.

6. Calculation Methodology

a) At CCNPP, there are two methodologies for calculating the concentrations of radionuclides in the receiving waters, at the point of release to **UNRESTRICTED AREAS**, resulting from the discharge of liquid waste.

(1) The rigorous method shall be used IF a computer system and the appropriate software are available.

(2) The simplified method may be used IF a computer system and the appropriate software are NOT available.

(3) These methods, as well as additional supporting information, are presented in the following sections.

b) Rigorous method

(1) Solution of the following equation may prove too rigorous for routine use unless a computer system and appropriate software are available.

(2) If a computer system and the appropriate software are available, ensure the concentrations of radionuclides in **UNRESTRICTED AREAS** are less than one MPC by verifying the following inequality is true:

LIMIT ON CONCENTRATIONS OF RADIONUCLIDES IN **UNRESTRICTED AREAS** (RIGOROUS METHOD)

$$(F_u / F_d) \sum (A_{iu} / A_{iLt}) \leq 1 \qquad \textbf{Eq. 6L}^3$$

F_d = the actual dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The actual dilution water flow rate is the sum of the circulating water flow rate, the salt water flow rate, and the undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

1 Per Control 4.11.1.1.1.
2 The LOWER LIMITS OF DETECTION shall conform to Control 4.11.1.1.1, Table 4.11-1.
3 Equation 6L has been derived from 10 CFR 20, Appendix B, Table II, Note 1.

DILUTION WATER FLOW RATE

$F_d = F_{cw} + F_{sw} + F_u$	Eq. 2L

F_{cw} = the actual circulating water system flow rate

The flow rate of each circulating water pump is 200,000 gallons per minute.

F_{sw} = the salt water system flow rate

If the actual salt water flow rate is unknown or otherwise not readily available, a salt water flow rate of 0 gpm may be used in equation 2L.

The maximum flow rate for one salt water pump is listed on Attachment 2.

F_u = actual undiluted radwaste flow rate (gpm)

If the actual undiluted radwaste flow rate is less than 1% of the total dilution water flow rate, an undiluted radwaste flow rate of 0 gpm may be used in equation 2L.

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachments 2 and 3.

A_{iLt} = the specific activity limit for radionuclide, i, as obtained from 10 CFR 20, Appendix B, Table II, Column 2 (microcuries/cm^3)

A_{iu} = the specific activity of nuclide, i, in the undiluted liquid radwaste (microcuries per milliliter)

1 = the MPC limit (MPCs) for **UNRESTRICTED AREAs**

This value is based on the MPC limit from 10 CFR 20, Table II, Note 1.

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c) Simplified method

(1) Whenever a computer system and appropriate software are unavailable to perform the rigorous calculations described in the previous section, ensure the concentrations of radionuclides in **UNRESTRICTED AREAS** are less than one MPC by verifying the following inequality is true.

LIMIT ON CONCENTRATIONS OF RADIONUCLIDES IN **UNRESTRICTED AREAS** (SIMPLIFIED METHOD)

$\{ F_u / [(F_d) (A_{I131Lt}) (K_{sf})] \} \sum A_{iu} \leq 1$	**Eq. 7L[1]**

F_u = actual undiluted radwaste flow rate (gpm)

If the actual undiluted radwaste flow rate is less than 1% of the total dilution water flow rate, an undiluted radwaste flow rate of 0 gpm may be used in equation 2L.

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachments 2 and 3.

F_d = the actual dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The actual dilution water flow rate is the sum of the circulating water flow rate, the salt water flow rate, and the undiluted radwaste flow rate.

DILUTION WATER FLOW RATE

$F_d = F_{cw} + F_{sw} + F_u$	**Eq. 2L**

F_{cw} = the actual circulating water system flow rate

The flow rate of each circulating water pump is 200,000 gallons per minute.

F_{sw} = the actual salt water system flow rate

If the actual salt water flow rate is unknown or otherwise not readily available a salt water flow rate of 0 gpm may be used in equation 2L.

The maximum flow rate for one salt water pump is listed on Attachment 2.

A_{I131Lt} = the specific activity limit for I-131 corresponding to the limiting concentration specified in 10 CFR 20, Appendix B, Table II, Column 2

This value is 3E-7 microcuries per milliliter.

A_{iu} = the specific activity of nuclide, i, in the undiluted liquid radwaste (microcuries per milliliter)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP activity limit to the activity limit of 10 CFR 20, Appendix B, Table II, Column 2, (unitless) (per Control 3.11.1.1)

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1 This equation has been derived from 10 CFR 20, Appendix B, Table II, Note 1.

The safety factor chosen shall be less than or equal to 1.00. This ensures the activity is always less than or equal to the activity limit of 10 CFR 20, Appendix B, Table II, Column 2.

A safety factor of 1.00 will yield a activity which corresponds to the 10 CFR 20, Appendix B, Table II, Column 2 activity limit.

A safety factor of 0.500 will yield an activity which corresponds to one-half the activity limit of 10 CFR 20, Appendix B, Table II, Column 2.

It is recommended that a safety factor of 1.0 be used for calculating the activity, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a safety factor does provide plant personnel with a degree of administrative control over the use of simplified equations for generating radioactive liquid release permits. This administrative control is designed to minimize the possibility of violating 10 CFR 20, Appendix B, Table II, Column 2 limits whenever simplifying assumptions are used.

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

This safety factor has been included in equation 7L to account for any potential nonconservatism associated with applying the I-131 MPC limit to all radionuclides identified in the liquid release. Such nonconservatism could conceivable be present whenever radionuclides which have an MPC value higher than the I-131 MPC limit are present in a liquid release.

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d) Once the rigorous or simplified calculations have been completed, the calculation results are compared to the applicable limits and corrective **ACTION**s are initiated as described below.

7. Corrective **ACTION**s

a) CHEMISTRY SECTION surveillance procedures (e.g., CP-212) shall contain and/or reference administrative and/or Technical Specification limits for concentration of radionuclides in liquid effluents and shall specify corrective **ACTION**s to be initiated when these limits are exceeded.

b) Refer to Control 3.11.1.1 for **ACTION**s to be taken in the event the concentrations of radionuclides in **UNRESTRICTED AREAS** exceed one MPC.

LIMITS ON CUMULATIVE TOTAL BODY DOSES AND CUMULATIVE ORGAN DOSES FOR LIQUID EFFLUENTS

1. Introduction

   a) Appendix I to 10 CFR 50 specifies total body dose limits and organ dose limits associated with the release of radioactive liquids to **UNRESTRICTED AREAS.**

   b) Radiological effluent controls have been established to implement the requirements of 10 CFR 50, Appendix I.

   c) These radiological effluent controls are described below.

2. Radiological Effluent Controls

   a) The total body dose to **MEMBERS OF THE PUBLIC**, from liquid waste discharged to **UNRESTRICTED AREAS**, shall be less than 3 mrems per calendar quarter and 6 mrems per calendar year (Control 3.11.1.2).

   b) The organ doses to **MEMBERS OF THE PUBLIC**, from liquid waste discharged to **UNRESTRICTED AREAS**, shall be less than 10 mrems per calendar quarter and 20 mrems per calendar year (Control 3.11.1.2).

   c) The routine surveillances which are performed to verify compliance with these radiological effluent controls are described below.

3. Surveillance Requirement(s)

   a) Cumulative total body doses to **MEMBERS OF THE PUBLIC** in **UNRESTRICTED AREAS**--for the current calendar month, the calendar quarter, the current calendar year, and the previous 92 days--shall be calculated, in accordance with equation 8L, at least once per 60 days (per Control 4.11.1.2 and 4.11.1.3).

   b) Cumulative organ doses to **MEMBERS OF THE PUBLIC** in **UNRESTRICTED AREAS**--for the current calendar month, the current calendar quarter, the current calendar year, and the previous 92 days--shall be calculated, in accordance with equation 8L, at least once per 60 days (per Control 4.11.1.2 and 4.11.1.3).

4. Responsible Plant Organizations

   a) The CHEMISTRY SECTION is responsible for performing the surveillances, listed above, whenever the appropriate initiating conditions, listed below, are present.

5. Initiating Conditions

a) Perform the surveillances, listed above, at least once per 60 days (per Control 4.11.1.2).

b) For **BATCH RELEASE**s, perform the surveillances, listed above, prior to each **BATCH RELEASE** of radioactive liquid waste in accordance with approved CHEMISTRY SECTION procedures.

c) For **CONTINUOUS RELEASE**s, perform the surveillances, listed above, at least monthly (until the **CONTINUOUS RELEASE** has been terminated) and, if activity is identified[1] in the CONTINUOUS waste stream, the surveillances may be performed more often as specified in approved CHEMISTRY SECTION procedures.

d) Whenever the correct initiating conditions are present, the calculations required by the above mentioned surveillance(s) shall be completed in accordance with the methodology listed in the next section.

6. Calculation Methodology

a) The cumulative total body dose and the cumulative organ doses (for the current calendar month, current calendar quarter, previous 92 days, and current calendar year) shall be calculated as follows:

CUMULATIVE TOTAL BODY OR ORGAN, o, DOSE FROM LIQUID RELEASES, r

$$D_{ToL} = \sum D_{or} \qquad \text{Eq. 8L}$$

D_{ToL} = the sum total of the total body or organ, o, dose for all liquid releases discharged during the applicable time interval

D_{or} = the organ or total body dose resulting from release, r

Calculate the values of D_{or} for each liquid release as described below.

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b) At CCNPP, two methods exist for calculating D_{or} (i.e., the total body and organ doses resulting from any single release of radioactive liquid to an **UNRESTRICTED AREA**).

(1) The rigorous method shall be used IF a computer system and the appropriate software are available.

(2) The simplified method may be used IF a computer system and the appropriate software are NOT available.

(3) These methods, as well as additional supporting information, are presented in the following sections.

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1 The LOWER LIMITS OF DETECTION shall conform to Control 4.11.1.1.1, Table 4.11-1.

c) Rigorous method

(1) Solution of the following equation may prove too rigorous for routine use unless a computer system and appropriate software are available.

(2) If a computer system and the appropriate software are available, the dose commitments due to each release of radioactive liquid to an **UNRESTRICTED AREA** shall be calculated in accordance with the following equation:

DOSE TO THE TOTAL BODY OR ORGAN, o, FROM A LIQUID RELEASE (RIGOROUS EQUATION)

$$D_o = \{V_u / [(60)(F_d)]\} \Sigma\{(A_{iu})(C_{io})\} \qquad \text{Eq. 9L}^1$$

Where,

F_d = actual dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The actual dilution water flow rate is the sum of the circulating water flow rate, the salt water flow rate, and the undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

DILUTION WATER FLOW RATE

$$F_d = F_{cw} + F_{sw} + F_u \qquad \text{Eq. 2L}$$

F_{cw} = actual circulating water system flow rate

The flow rate of each circulating water pump is 200,000 gallons per minute.

F_{sw} = actual salt water system flow rate

If the actual salt water flow rate is unknown or otherwise not readily available a salt water flow rate of 0 gpm may be used in equation 2L.

The maximum flow rate for one salt water pump is listed on Attachment 2.

F_u = actual undiluted radwaste flow rate (gpm)

If the actual undiluted radwaste flow rate is less than 1% of the total dilution water flow rate, an undiluted radwaste flow rate of 0 gpm may be used in equation 2L.

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachments 2 and 3.

[1] Equation 9L has been derived from NUREG-0133, 4.3.

V_u = volume of undiluted radwaste (gallons)

A_{iu} = specific activity of nuclide, i, in the undiluted liquid radwaste (microcuries per milliliter)

C_{io} = liquid release dose factor for nuclide, i, and organ, o (mrem/hr per microcurie/ml)

The liquid release dose factors for principal gamma and beta emitters are listed in Attachment 6.

The liquid release dose factors for principal gamma and beta emitters were obtained in accordance with the methodology of NUREG-0133, section 4.3.1 (for salt water sites).

60 = a constant, the number of minutes per hour

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(3) In the event a computer system is unavailable, a simplified equation may be used to calculate the total body and organ dose commitments due to individual liquid releases.

(4) The simplified method is presented below.

d) Simplified methods

(1) Whenever a computer system is unavailable to perform the rigorous total body dose calculations described in the previous section, the total body dose commitments--due to each release of radioactive liquid to an **UNRESTRICTED AREA**--may be calculated in accordance with equation 10L.

(2) Whenever a computer system is unavailable to perform the rigorous organ dose calculations described in the previous section, the dose commitments--to the maximum exposed organ, due to each release of radioactive liquid to an **UNRESTRICTED AREA**--may be calculated in accordance with equation 11L.

TOTAL BODY DOSE FROM A LIQUID RELEASE (SIMPLIFIED EQUATION)

$$D_{tb} = \{ [(58.6)(V_u)] / [(K_{sf})(F_d)] \} \Sigma A_{iu} \qquad \text{Eq. 10L}^1$$

D_{tb} = dose commitment, to the total body, due to each release of radioactive liquid to an **UNRESTRICTED AREA**

58.6 = a conversion constant which includes:

1) 1.33E4 mrem/hr per microcurie/ml (the total body dose factor for Cs-134)

2) 1.000 hr/60.00 min

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1 Equations 10L has been derived from NUREG-0133, 4.3.

3) 1.000 gallons/3.785 liters

V_u = volume of undiluted radwaste (liters)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP total body dose limit to the total body dose limit of Control 3.11.1.2, (unitless)

The safety factor chosen shall be less than or equal to 1.00. This ensures the total body dose is always less than or equal to the total body dose limit of Control 3.11.1.2.

A safety factor of 1.00 will yield a total body dose which corresponds to the total body dose limit of Control 3.11.1.2.

A safety factor of 0.500 will yield an total body dose which corresponds to one-half the total body dose limit of Control 3.11.1.2.

It is recommended that a safety factor of 1.0 be used for calculating the total body dose, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a safety factor does provide plant personnel with a degree of administrative control over the use of simplified equations for generating radioactive liquid release permits. This administrative control is designed to minimize the possibility of violating Control 3.11.1.2 when simplifying assumptions are used.

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

This safety factor has been included in equation 10L to account for any potential nonconservatism associated with applying the Cs-134 total body dose conversion factor to all radionuclides identified in the liquid release. Such nonconservatism could conceivable be present whenever radionuclides having a dose conversion factor greater than that of Cs-134 are present in a liquid release.

F_d = actual dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The actual dilution water flow rate is the sum of the circulating water flow rate, the salt water flow rate, and the undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

DILUTION WATER FLOW RATE

$$F_d = F_{cw} + F_{sw} + F_u \qquad \text{Eq. 2L}$$

F_{cw} = actual circulating water system flow rate

The flow rate of each circulating water pump is 200,000 gallons per minute.

F_{sw} = actual salt water system flow rate

If the actual salt water flow rate is unknown or otherwise not readily available, a salt water flow rate of 0 gpm may be used in equation 2L.

The maximum flow rate for one salt water pump is listed on Attachment 2.

F_u = actual undiluted radwaste flow rate (gpm)

If the actual undiluted radwaste flow rate is less than 1% of the total dilution water flow rate, an undiluted radwaste flow rate of 0 gpm may be used in equation 2L.

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachments 2 and 3.

A_{iu} = specific activity of nuclide, i, in the undiluted liquid radwaste (microcuries per milliliter)

DOSE TO THE ORGAN, o, FROM A LIQUID RELEASE (SIMPLIFIED EQUATION)

$$D_o = \{ [(449)(V_u)] / [(K_{sf})(F_d)] \} \Sigma A_{iu} \qquad \text{Eq. 11L}^1$$

D_o = the dose commitment to the maximum exposed ORGAN due to each release of radioactive liquid waste to an **UNRESTRICTED AREA**

449 = a conversion constant which includes:

1) 1.02E5 mrem/hr per microcurie/ml (the dose factor for I-131 to the thyroid)

2) 1.000 hr/60.00 min

3) 1.000 gallons/3.785 liters

V_u = volume of undiluted radwaste (liters)

K_{sf} = a constant, actually a safety factor, which is the ratio of the CCNPP organ dose limit to the organ dose limit of Control 3.11.1.2, (unitless)

[1] Equations 11L has been derived from NUREG-0133, 4.3.

The safety factor chosen shall be less than or equal to 1.00. This ensures the organ dose is always less than or equal to the organ dose limit of Control 3.11.1.2.

A safety factor of 1.00 will yield a organ dose which corresponds to the organ dose limit of Control 3.11.1.2.

A safety factor of 0.500 will yield an organ dose which corresponds to one-half the organ dose limit of Control 3.11.1.2.

It is recommended that a safety factor of 1.0 be used for calculating the organ dose, however, other values--not to exceed 1.00--may be used as directed by the General Supervisor Chemistry.

The particular value selected for the safety factor is somewhat arbitrary, however a safety factor does provide plant personnel with a degree of administrative control over the use of simplified equations for generating radioactive liquid release permits. This administrative control is designed to minimize the possibility of violating Control 3.11.1.2 when simplifying assumptions are used.

The use of a safety factor is consistent with the ALARA philosophy that licensees should make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to **UNRESTRICTED AREAS**, as low as is reasonably achievable.

This safety factor has been included in equation 11L to account for any potential nonconservatism associated with applying the I-131 thyroid dose conversion factor to all radionuclides identified in the liquid release. Such nonconservatism could conceivable be present whenever radionuclides having a dose conversion factor greater than that of I-131 are present in a liquid release.

F_d = dilution water flow rate (gpm) prior to the point of release to **UNRESTRICTED AREAS**

The actual dilution water flow rate is the sum of the circulating water flow rate, the salt water flow rate, and the undiluted radwaste flow rate.

The dilution water flow rate shall be calculated in accordance with equation 2L.

DILUTION WATER FLOW RATE

$$F_d = F_{cw} + F_{sw} + F_u \qquad \text{Eq. 2L}$$

F_{cw} = actual circulating water system flow rate

The flow rate of each circulating water pump is 200,000 gallons per minute.

F_{sw} = salt water system flow rate

If the actual salt water flow rate is unknown or otherwise not readily available, a salt water flow rate of 0 gpm may be used in equation 2L.

The maximum flow rate for one salt water pump is listed on Attachment 2.

F_u = actual undiluted radwaste flow rate (gpm)

If the actual undiluted radwaste flow rate is less than 1% of the total dilution water flow rate, an undiluted radwaste flow rate of 0 gpm may be used in equation 2L.

Values of maximum undiluted radwaste flow rates for various waste streams are tabulated in Attachments 2 and 3.

A_{iu} = specific activity of nuclide, i, in the undiluted liquid radwaste (microcuries per milliliter)

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7. Corrective ACTIONs

   a) CHEMISTRY SECTION surveillance procedures (e.g., CP-212) shall contain and/or reference administrative and/or Control limits for cumulative total body doses or cumulative organ doses resulting from liquid effluents and shall specify corrective actions to be initiated when these limits are exceeded.

   b) Refer to Control 3.11.1.2 for actions to be taken in the event the calculated cumulative total body dose exceeds 3 mrem per calendar quarter or 6 mrem per calendar year.

   c) Refer to Control 3.11.1.2 for actions to be taken in the event the calculated cumulative organ dose--for any organ--exceeds 10 mrem per calendar quarter or 20 mrem per calendar year.

LIMITS FOR THE LIQUID WASTE PROCESSING SYSTEM

1. Introduction

   a) 10 CFR 50.36a requires licensees to maintain and use the equipment installed in the liquid waste processing system for the purpose of controlling effluents to the environment.

   b) Radiological effluent controls have been established to implement the requirements of 10 CFR 50.36a.

   c) These radiological effluent controls are described below.

2. Radiological Effluent Controls

   a) The liquid radwaste processing system shall be used to reduce the quantity of radioactive materials in liquid waste released to the environment whenever the total body dose to **MEMBERS OF THE PUBLIC**, from liquid waste discharged to **UNRESTRICTED AREAS**, is greater than 0.36 mrem for the previous 92 days (per Control 3.11.1.3).

   b) The liquid radwaste processing system shall be used to reduce the quantity of radioactive materials in liquid waste released to the environment whenever the organ doses to **MEMBERS OF THE PUBLIC**, from liquid waste discharged to **UNRESTRICTED AREAS**, is greater than 1.20 mrem for the previous 92 days (per Control 3.11.1.3).

   c) The routine surveillances which are performed to verify compliance with this radiological effluent controls are described below.

3. Surveillance Requirement(s)

   a) The previous 92-day, cumulative, total body dose--to **MEMBERS OF THE PUBLIC** in **UNRESTRICTED AREAS**--shall be calculated, as described in the ODCM, at least once per 60 days (per Control 4.11.1.3).

   b) The previous 92-day, cumulative, organ doses--to **MEMBERS OF THE PUBLIC** in **UNRESTRICTED AREAS**--shall be calculated, as described in the ODCM, at least once per 60 days (per Control 4.11.1.3).

   c) The CCNPP organization(s) responsible for performing these surveillances are identified in the next section.

4. Responsible Plant Organization(s)

   a) The CHEMISTRY SECTION is responsible for performing the surveillances, listed above, whenever the appropriate initiating conditions, listed below, are present.

5. Initiating Conditions

   a) Perform the surveillances, listed above, at least once per 60 days (per Control 4.11.1.2).

   b) For **BATCH RELEASEs**, perform the surveillances, listed above, prior to each **BATCH RELEASE** of radioactive liquid waste in accordance with approved CHEMISTRY SECTION procedures.

   c) For **CONTINUOUS RELEASEs**, perform the surveillances, listed above, at least monthly (until the **CONTINUOUS RELEASE** has been terminated) and, if activity is identified[1] in the CONTINUOUS waste stream, the surveillances may be performed more often as specified in approved CHEMISTRY SECTION procedures.

[1] The LOWER LIMITS OF DETECTION shall conform to Control 4.11.1.1.1, Table 4.11-1.

d) Whenever the correct initiating conditions are present, the calculations required by the above mentioned surveillance(s) shall be completed in accordance with the methodology contained in the following section.

6. Calculation Methodology

a) The cumulative total body dose and the cumulative organ doses for the previous 92 days shall be calculated in accordance with equation 8L found in the section, "Limits On Cumulative Total Body Doses And Cumulative Organ Doses For Liquid Effluents", found elsewhere in the ODCM.

7. Corrective **ACTIONs**

a) CHEMISTRY SECTION surveillance procedures (e.g., CP-212) shall contain/and or reference administrative and/or Control limits for cumulative total body doses or cumulative organ doses resulting from liquid effluents and shall specify corrective **ACTIONs** to be initiated when these limits are exceeded.

b) Refer to Control 3.11.1.3 for **ACTIONs** to be taken in the event the calculated cumulative total body dose exceeds 0.36 mrem for the previous 92 days.

c) Refer to Control 3.11.1.3 for **ACTIONs** to be taken in the event the calculated cumulative organ dose--for any organ--exceeds 1.2 mrem for the previous 92 days.