VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

April 22, 1996

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Serial No. 96-186
NL/RPC
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNITS 1 AND 2
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

Pursuant to Technical Specification 6.9.1.9, enclosed is the Annual Radioactive Effluent Release Report for North Anna Power Station Units 1 and 2 for the reporting period of January 1, 1995 to December 31, 1995.

If you have any questions or require additional information, please contact us.

Very truly yours,

M L Bowling

M. L. Bowling, Manager
Nuclear Licensing and Operations Support

Attachment

cc: U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, N. W.
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Mr. R. D. McWhorter
NRC Senior Resident Inspector
North Anna Power Station

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ATTACHMENT

ANNUAL RADIOACTIVE
EFFLUENT RELEASE REPORT

NORTH ANNA POWER STATION UNITS 1 AND 2
LICENSE NOS. NPF-4 AND NPF-7

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

NORTH ANNA POWER STATION

(JANUARY 01, 1995 TO DECEMBER 31, 1995)

PREPARED BY: ____________________

Supervisor
Radiological Analysis

REVIEWED BY: E. ____________________

Supervisor
Technical Services

APPROVED BY: A H Stafford

Superintendent
Radiological Protection

F O R W A R D

This report is submitted as required by Appendix A to Operating License Nos. NPF-4 and NPF-7, Technical Specifications for North Anna Power Station, Units 1 and 2, Virginia Electric and Power Company, Docket Nos. 50-338, 50-339, Section 6.9.1.9.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

FOR THE

NORTH ANNA POWER STATION

JANUARY 01, 1995 TO DECEMBER 31, 1995

I N D E X

1.0 EXECUTIVE SUMMARY

The Annual Radiological Effluent Release Report describes the radiological effluent control program conducted at the North Anna Power station during the 1995 calendar year. This document summarizes the quantities of radioactive liquid and gaseous effluents and solid waste released from the North Anna Power Station in accordance with R.G. 1.21 during the period January 1 through December 31, 1995, and includes an assessment of radiation doses to the maximum exposed member of the public due to radioactive liquid and gaseous effluents.

There were no unplanned liquid or gaseous effluent releases classified according to the criteria in the Offsite Dose Calculation Manual during this reporting period.

Based on the 1995 effluent release data, 10 CFR 50, Appendix I dose calculations were performed in accordance with the Offsite Dose Calculation Manual. The results of these pathway dose calculations indicate the following:

1. The total body dose due to liquid effluents was 2.81E-01 mrem, which is 4.68% of the dose limit and the critical organ (liver) dose due to liquid effluents was 2.89E-01 mrem, which is 1.4% of the dose limit.

2. The air dose due to noble gases was 2.29E-02 mrad gamma, which is 0.1% of the annual gamma dose limit, and 1.07E-02 mrad beta, which is .03% of the annual beta dose limit.

1.0 EXECUTIVE SUMMARY (cont.)

3. The critical organ dose for I-131, I-133, H-3, and Particulates with half-lives greater than 8 days was 2.18E-02 mrem, which is .07% of the annual dose limit.

There war (1) one major change to radioactive liquid, gaseous, and solid waste treatment systems during this reporting period. This was installation of a high capacity blowdown capability for each unit's steam generators. A brief description of this system and a summary of the safety evaluations are provided in Attachment 4.

There were changes to the Offsite Dose Calculation Manual, VPAP-2103, during this reporting period which were implemented in PN-1 and Revision 7 on January 3, 1995 and October 31, 1995 respectively. Attachment 3 provides the changes to VPAP-2103.

Based on the levels of radioactivity observed during this reporting period and the dose calculations performed, the operations of the North Anna Nuclear Power Station Units 1 and 2 have resulted in negligible dose consequences to the maximum exposed member of the public in unrestricted areas.

2.0 PURPOSE AND SCOPE

The Radioactive Effluent Release Report includes, in Attachment 1, a summary of the quantities of radioactive liquid and gaseous effluents and solid waste as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", Revision 1, June 1974, with data

2.0 **PURPOSE AND SCOPE** (cont).

summarized on a quarterly basis following the format of Tables 1, 2 and 3 of Appendix B thereof. The report submitted before May 1st of each year includes an assessment of radiation doses to the maximum exposed member of the public due to radioactive liquid and gaseous effluents released from the site during the previous calendar year. The report also includes a list of unplanned releases during the reporting period, in Attachment 6.

As required by Technical Specification 6.15, changes to the Offsite Dose Calculation Manual (ODCM) for the time period covered by this report are included in Attachment 3.

Major changes to radioactive liquid, gaseous and solid waste treatment systems are reported in Attachment 4, as required by the ODCM, section 6.7.2.a.4. Information to support the reason(s) for the change(s) and a summary of the 10 CFR 50.59 evaluation are included. In lieu of reporting major changes in this report, major changes to the radioactive waste treatment systems may be submitted as part of the annual FSAR update.

As required by the ODCM, sections 6.2.2.b.2 and 6.3.2.b.3, a list and explanation for the inoperability of radioactive liquid and/or gaseous effluent monitoring instrumentation is provided in Attachment 5 of this report.

3.0 DISCUSSION

The basis for the calculation of the percent of technical specification for the critical organ in Table 1A of Attachment 1 is the ODCM, section 6.3.1, which requires that the dose rate for iodine-131 & iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days shall be less than or equal to 1500 mrem/yr to the critical organ at or beyond the site boundary. The critical organ is the child's thyroid via the inhalation pathway.

The basis for the calculation of percent of technical specification for the total body and skin in Table 1A of Attachment 1 is the ODCM, section 6.3.1, which requires that the dose rate for noble gases to areas at or beyond site boundary shall be less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin.

The basis for the calculation of the percent of technical specification in Table 2A in Attachment 1 is the ODCM, section 6.2.1, which states that the concentrations of radioactive material released in liquid effluents to unrestricted areas shall be limited to 10 times the concentrations specified in 10 CFR 20, Appendix B, Table 2, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2.0E-4 microcuries/ml.

Percent of technical specification calculations are based on the total gaseous or liquid effluents released for that respective quarter.

3.0 **DISCUSSION** (cont.)

The annual and quarterly doses, as reported in Attachment 2, were calculated according to the methodology presented in the ODCM. The beta and gamma air doses due to noble gases released from the site were calculated at site boundary. The maximum exposed member of the public from the releases of airborne iodine-131 & iodine-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days, is defined as an infant, exposed through the grass-cow-milk pathway, with the critical organ being the thyroid gland. The maximum exposed member of the public from radioactive materials in liquid effluents in unrestricted areas is defined as an adult, exposed by either the invertebrate or fish pathway, with the critical organ being the liver. The total body dose was also determined for this individual.

Presented in Attachment 6 is a list of unplanned gaseous and liquid releases meeting the requirements of 6.7.2.a.3 of the ODCM.

The typical Lower Limit of Detection (LLD) capabilities of the radioactive effluent analysis instrumentation are presented in Attachment 7. These LLD values are based upon conservative conditions (i.e., minimum sample volume and maximum delay time prior to analysis). Actual LLD values may be lower. If a radioisotope was not detected when effluent samples were analyzed, then the activity of that radioisotope was reported as Not Detectable (N/D) on Attachment 1 of this report. If an analysis for an isotope was not performed, then the activity was reported as Not Applicable (N/A).

4.0 SUPPLEMENTAL INFORMATION

As required by the ODCM, section 6.6.2, evaluation of the Land Use Census is made to determine if new location(s) have been identified for the radiological environmental monitoring program pursuant to the ODCM, Section 6.6.2 requirements. Evaluation of the Land Use Census conducted in 1995 identified no change in sample locations for the radiological environmental monitoring program.

Changes to some TLD locations were made to resolve differences in emergency plan sectors and radiological environmental monitoring sectors. These changes were implemented with PN-1 to VPAP-2101, Offsite Dose Calculation Manual, included in Attachment 3 of this report.

Section 6.6.1.b.4 of the ODCM requires identification of the cause(s) for the unavailability of milk or leafy vegetation samples, and the identification of new locations for obtaining replacement samples. Milk samples, as required by the ODCM, section 6.6.1, were available during the time period covered by this report. The leafy vegetation samples for vegetation station 14, 15, 16, 21 and 23 were not collected for the months of January, February, March, November and December 1995 due to seasonal unavailability. All other samples were obtained and analyzed as required during the time period covered by this report.

ATTACHMENT 1

EFFLUENT RELEASE DATA

(01/95 - 12/95)

This attachment includes a summary of the quantities of radioactive liquid and gaseous effluents and solid waste, as outlined in Regulatory Guide 1.21, Appendix B.

TABLE 1A
NORTH ANNA POWER STATION
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
SUMMATION OF ALL GASEOUS EFFLUENT RELEASES FOR (01/95 - 12/95)

	UNITS	1ST QUARTER	2ND QUARTER	ESTIMATED TOTAL PERCENT ERROR (%)
A. Fission and Activation Gases:				
1. Total Release.	Curies	2.53E+1	3.49E+0	1.80E+1
2. Average Release Rate for Period	μCi/sec	3.25E+0	4.44E-1	
B. Iodines:				
1. Total Iodine-131 Release.	Curies	7.85E-5	1.56E-4	2.80E+1
2. Average Release Rate for Period	μCi/sec	1.01E-5	1.99E-5	
C. Particulates (T½ > 8 days):				
1. Total Particulate (T½ > 8 days) Release	Curies	2.17E-6	6.35E-5	2.80E+1
2. Average Release Rate for Period	μCi/sec	2.79E-7	8.07E-6	
3. Gross Alpha Radioactivity Release	Curies	8.52E-6	5.74E-6	
D. Tritium:				
1. Total Release	Curies	1.14E+1	1.08E+2	3.10E+1
2. Average Release Rate for Period	μCi/sec	1.47E+0	1.38E+1	
E. Percentage of Technical Specification Limits				
1. Total Body Dose Rate	%	9.28E-3	1.98E-4	
2. Skin Dose Rate	%	2.32E-3	7.59E-5	
3. Critical Organ Dose Rate	%	1.02E-3	9.79E-3	

TABLE 1A
NORTH ANNA POWER STATION
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
SUMMATION OF ALL GASEOUS EFFLUENT RELEASES FOR (01/95 - 12/95)

	UNITS	3rd QUARTER	4th QUARTER	ESTIMATED TOTAL PERCENT ERROR (%)
A. Fission and Activation Gases:				
1. Total Release.	Curies	2.24E+0	5.39E+0	1.80E+1
2. Average Release Rate for Period	μCi/sec	2.82E-1	6.78E-1	
B. Iodines:				
1. Total Iodine-131 Release.	Curies	8.18E-9	N/D	2.80E+1
2. Average Release Rate for Period	μCi/sec	1.03E-9	N/D	
C. Particulates (T½ > 8 days):				
1. Total Particulate (T½ > 8 days) Release	Curies	1.02E-5	1.66E-7	2.80E+1
2. Average Release Rate for Period	μCi/sec	1.29E-6	2.09E-8	
3. Gross Alpha Radioactivity Release	Curies	4.62E-6	3.48E-6	
D. Tritium:				
1. Total Release	Curies	7.03E+1	1.27E+1	3.10E+1
2. Average Release Rate for Period	μCi/sec	8.85E+0	1.60E+0	
E. Percentage of Technical Specification Limits				
1. Total Body Dose Rate	%	1.13E-4	7.85E-3	
2. Skin Dose Rate	%	8.79E-5	1.91E-3	
3. Critical Organ Dose Rate	%	6.01E-3	8.81E-4	

TABLE 1B
NORTH ANNA POWER STATION
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
MIXED MODE GASEOUS EFFLUENT RELEASES FOR 01/95 - 12/95

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	UNITS	1st QUARTER	2nd QUARTER	1st QUARTER	2nd QUARTER
Fission & Activation Gases:					
Krypton - 85	Ci	N/D	N/D	1.22E+0	7.88E-1
Krypton - 85m	Ci	N/D	N/D	4.63E-3	N/D
Krypton - 87	Ci	N/D	N/D	3.69E-4	N/D
Krypton - 88	Ci	N/D	N/D	3.71E-3	N/D
Xenon - 131m	Ci	N/D	N/D	2.11E-1	2.08E-2
Xenon - 133	Ci	7.24E+0	6.79E-1	9.55E+0	5.48E-2
Xencn - 133m	Ci	N/D	N/D	1.43E-1	N/D
Xenon - 135	Ci	9.78E-2	N/D	2.42E-2	N/D
Xenon - 135m	Ci	N/D	N/D	N/D	N/D
Xenon - 138	Ci	N/D	N/D	N/D	N/D
Other (Specify)					
Argon - 41	Ci	7.27E-3	1.56E-2	1.21E-2	N/D
Total for Period	Ci	7.35E+0	6.95E-1	1.12E+1	8.64E-1
Iodines:					
Iodine - 130	Ci	N/D	N/D	N/D	N/D
Iodine - 131	Ci	1.33E-6	1.46E-7	N/D	N/D
Iodine - 132	Ci	N/D	N/D	N/D	N/D
Iodine - 133	Ci	3.72E-7	1.38E-8	N/D	N/D
Iodine - 134	Ci	N/D	N/D	N/D	N/D
Iodine - 135	Ci	N/D	N/D	N/D	N/D
Total for Period	Ci	1.70E-6	1.60E-7	N/D	N/D
Particulates:					
Manganese - 54	Ci	N/D	N/D	N/D	N/D
Iron - 55	Ci	N/A	N/A	N/A	N/A
Cobalt - 58	Ci	N/D	N/D	N/D	N/D
Iron - 59	Ci	N/D	N/D	N/D	N/D
Cobalt - 60	Ci	8.83E-8	2.96E-8	N/D	N/D
Zinc - 65	Ci	N/D	N/D	N/D	N/D

TABLE 1B

NORTH ANNA POWER STATION

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

MIXED MODE GASEOUS EFFLUENT RELEASES FOR 01/95 - 12/95

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	UNITS	1st QUARTER	2nd QUARTER	1st QUARTER	2nd QUARTER
Particulates: (cont.)					
Strontium - 89	Ci	N/D	N/D	N/D	N/D
Strontium - 90	Ci	N/D	N/D	N/D	N/D
Cesium - 134	Ci	5.39E-9	N/D	N/D	N/D
Cesium - 136	Ci	N/D	N/D	N/D	N/D
Cesium - 137	Ci	2.64E-7	1.37E-7	N/D	N/D
Barium - 140	Ci	N/D	N/D	N/D	N/D
Cerium - 141	Ci	N/D	N/D	N/D	N/D
Cerium - 144	Ci	N/D	N/D	N/D	N/D
Other (Specify)					
Molybdenum - 99 ($T_{1/2}$ < 8 days)	Ci	N/D	N/D	N/D	N/D
Antimony - 122 ($T_{1/2}$ < 8 days)	Ci	N/D	N/D	N/D	N/D
Total for Period ($T_{1/2}$ > 8 days)	Ci	3.58E-7	1.67E-7	N/D	N/D
Total for Period ($T_{1/2}$ < 8 days)	Ci	N/D	N/D	N/D	N/D
Total for Period	Ci	3.58E-7	1.67E-7	N/D	N/D
GROSS ALPHA:	Ci	1.90E-9	2.33E-9	N/A	N/A
TRITIUM:	Ci	1.45E+0	1.34E+0	1.02E-4	5.70E-5

TABLE 1B
NORTH ANNA POWER STATION
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
MIXED MODE GASEOUS EFFLUENT RELEASES FOR 01/95 - 12/95

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	UNITS	3rd QUARTER	4th QUARTER	3rd QUARTER	4th QUARTER
Fission & Activation Gases:					
Krypton - 85	Ci	N/D	N/D	1.59E+0	3.88E-1
Krypton - 85m	Ci	N/D	N/D	N/D	N/D
Krypton - 87	Ci	N/D	N/D	N/D	N/D
Krypton - 88	Ci	N/D	N/D	N/D	N/D
Xenon - 131m	Ci	N/D	N/D	N/D	N/D
Xenon - 133	Ci	5.13E-1	1.16E+0	N/D	3.36E-5
Xenon - 133m	Ci	N/D	N/D	N/D	N/D
Xenon - 135	Ci	N/D	N/D	N/D	N/D
Xenon - 135m	Ci	N/D	N/D	N/D	N/D
Xenon - 138	Ci	N/D	N/D	N/D	N/D
Other (Specify)					
Argon - 41	Ci	N/D	6.02E-2	N/D	N/D
Total for Period	Ci	5.13E-1	1.22E+0	1.59E+0	3.88E-1
Iodines:					
Iodine - 131	Ci	8.18E-9	N/D	N/D	N/D
Iodine - 133	Ci	N/D	N/D	N/D	N/D
Iodine - 135	Ci	N/D	N/D	N/D	N/D
Total for Period	Ci	8.18E-9	N/D	N/D	N/D
Particulates:					
Manganese - 54	Ci	N/D	N/D	N/D	N/D
Cobalt - 58	Ci	N/D	N/D	N/D	N/D
Iron - 59	Ci	N/D	N/D	N/D	N/D
Cobalt - 60	Ci	6.64E-8	3.28E-8	N/D	N/D
Zinc - 65	Ci	N/D	N/D	N/D	N/D
Strontium - 85	Ci	N/D	N/D	N/D	N/D

TABLE 1B
NORTH ANNA POWER STATION
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
MIXED MODE GASEOUS EFFLUENT RELEASES FOR 01/95 - 12/95

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	UNITS	3rd QUARTER	4th QUARTER	3rd QUARTER	4th QUARTER
Particulates: (cont.)					
Strontium - 89	Ci	N/D	N/D	N/A	N/A
Strontium - 90	Ci	N/D	N/D	N/A	N/A
Cesium - 134	Ci	N/D	N/D	N/D	N/D
Cesium - 136	Ci	N/D	N/D	N/D	N/D
Cesium - 137	Ci	3.28E-7	1.33E-7	N/D	N/D
Barium - 140	Ci	N/D	N/D	N/D	N/D
Lanthanum - 140	Ci	N/D	N/D	N/D	N/D
Cerium - 141	Ci	N/D	N/D	N/D	N/D
Cerium - 144	Ci	N/D	N/D	N/D	N/D
Other (Specify)					
Molybdenum - 99 (T½ < 8 days)	Ci	N/D	N/D	N/D	N/D
Cerium - 143 (T½ < 8 days)	Ci	3.63E-8	N/D	N/D	N/D
Niobium - 95 (T½ > 8 days)	Ci	N/D	N/D	N/D	N/D
Total for Period (T½ > 8 days)	Ci	3.94E-7	1.66E-7	N/D	N/D
Total for Period (T½ < 8 days)	Ci	3.63E-8	N/D	N/D	N/D
Total for Period	Ci	4.30E-7	1.66E-7	N/D	N/D
GROSS ALPHA:	Ci	8.09E-10	2.20E-9	N/A	N/A
TRITIUM:	Ci	2.43E+0	3.10E+0	7.37E-4	1.35E-4

TABLE 1C
NORTH ANNA POWER STATION
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
GROUND LEVEL GASEOUS EFFLUENT RELEASES FOR 01/95 - 12/95

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	UNITS	1st QUARTER	2nd QUARTER	1st QUARTER	2nd QUARTER
Fission & Activation Gases:					
Krypton - 85	Ci	N/D	N/D	3.17E-2	3.24E-2
Krypton - 85m	Ci	9.31E-4	N/D	N/D	N/D
Krypton - 87	Ci	1.38E-3	N/D	N/D	N/D
Krypton - 88	Ci	N/D	N/D	N/D	N/D
Xenon - 131m	Ci	N/D	N/D	1.35E-2	5.53E-3
Xenon - 133	Ci	3.65E-2	N/D	2.44E+0	1.86E+0
Xenon - 133m	Ci	N/D	N/D	8.36E-3	4.15E-4
Xenon - 135	Ci	1.03E-2	N/D	6.99E-3	N/D
Xenon - 135m	Ci	1.10E-2	N/D	6.50E-8	N/D
Xenon - 138	Ci	2.33E-3	N/D	N/D	N/D
Other (Specify)					
Argon - 41	Ci	4.21E+0	2.71E-2	3.71E-5	N/D
Total for Period	Ci	4.27E+0	2.71E-2	2.50E+0	1.90E+0
Iodines:					
Iodine - 130	Ci	N/D	N/D	N/D	N/D
Iodine - 131	Ci	7.72E-5	1.56E-4	9.63E-10	N/D
Iodine - 132	Ci	N/D	N/D	5.13E-9	8.09E-6
Iodine - 133	Ci	N/D	N/D	1.96E-8	N/D
Iodine - 134	Ci	N/D	N/D	3.01E-9	N/D
Iodine - 135	Ci	N/D	N/D	1.36E-8	N/D
Total for Period	Ci	7.72E-5	1.56E-4	4.23E-8	8.09E-6
Particulates:					
Manganese - 54	Ci	N/D	N/D	N/D	N/D
Iron - 55	Ci	N/A	N/A	N/A	N/A
Cobalt - 58	Ci	1.81E-6	4.71E-5	N/D	1.98E-6
Iron - 59	Ci	N/D	N/D	N/D	N/D
Cobalt - 60	Ci	N/D	5.16E-6	N/D	N/D
Zinc - 65	Ci	N/D	N/D	N/D	N/D

TABLE 1C
NORTH ANNA POWER STATION
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
GROUND LEVEL GASEOUS EFFLUENT RELEASES FOR 01/95 - 12/95

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	UNITS	1st QUARTER	2nd QUARTER	1st QUARTER	2nd QUARTER
Particulates: (cont.)					
Strontium - 89	Ci	N/D	N/D	N/D	N/D
Strontium - 90	Ci	N/D	N/D	N/D	N/D
Cesium - 134	Ci	N/D	N/D	N/D	N/D
Cesium - 136	Ci	N/D	N/D	N/D	N/D
Cesium - 137	Ci	N/D	8.44E-6	N/D	N/D
Barium - 140	Ci	N/D	N/D	N/D	N/D
Cerium - 141	Ci	N/D	N/D	N/D	N/D
Cerium - 144	Ci	N/D	N/D	N/D	N/D
Other (Specify)					
Molybdenum - 99 ($T\frac{1}{2}$ < 8 days)	Ci	N/D	N/D	N/D	N/D
Antimony - 122 ($T\frac{1}{2}$ < 8 days)	Ci	N/D	N/D	N/D	N/D
Niobium - 95 ($T\frac{1}{2}$ > 8 days)	Ci	N/D	N/D	N/D	5.79E-7
Total for Period ($T\frac{1}{2}$ > 8 days)	Ci	1.81E-6	6.07E-5	N/D	2.56E-6
Total for Period ($T\frac{1}{2}$ < 8 days)	Ci	N/D	N/D	N/D	N/D
Total for Period	Ci	1.81E-6	6.07E-5	N/D	2.56E-6
GROSS ALPHA:	Ci	8.52E-6	5.13E-6	N/D	6.09E-7
TRITIUM:	Ci	9.95E+0	1.07E+2	3.21E-2	9.92E-1

TABLE 1C

NORTH ANNA POWER STATION

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

GROUND LEVEL GASEOUS EFFLUENT RELEASES FOR 01/95 - 12/95

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	UNITS	3rd QUARTER	4th QUARTER	3rd QUARTER	4th QUARTER
Fission & Activation Gases:					
Krypton - 85	Ci	N/D	N/D	N/D	N/D
Krypton - 85m	Ci	N/D	N/D	N/D	N/D
Krypton - 87	Ci	N/D	N/D	N/D	N/D
Krypton - 88	Ci	N/D	N/D	N/D	N/D
Xenon - 131m	Ci	N/D	N/D	N/D	N/D
Xenon - 133	Ci	N/D	N/D	N/D	2.99E-5
Xenon - 133m	Ci	N/D	N/D	N/D	N/D
Xenon - 135	Ci	N/D	N/D	N/D	N/D
Xenon - 135m	Ci	N/D	N/D	N/D	N/D
Xenon - 138	Ci	N/D	N/D	N/D	N/D
Other (Specify) Xe-137	Ci	1.07E-1	N/D	N/D	N/D
Argon - 41	Ci	3.50E-2	3.78E+0	N/D	N/D
Total for Period	Ci	1.42E-1	3.78E+0	N/D	2.99E-5
Iodines:					
Iodine - 131	Ci	N/D	N/D	N/D	N/D
Iodine - 132	Ci	N/D	N/D	N/D	N/D
Iodine - 133	Ci	N/D	N/D	N/D	N/D
Iodine - 134	Ci	N/D	N/D	N/D	N/D
Iodine - 135	Ci	N/D	N/D	N/D	N/D
Total for Period	Ci	N/D	N/D	N/D	N/D
Particulates:					
Manganese - 54	Ci	N/D	N/D	N/D	N/D
Iron - 55	Ci	N/A	N/A	N/A	N/A
Cobalt - 58	Ci	2.86E-6	N/D	N/D	N/D
Iron - 59	Ci	N/D	N/D	N/D	N/D
Cobalt - 60	Ci	4.36E-6	N/D	N/D	N/D
Zinc - 65	Ci	N/D	N/D	N/D	N/D
Niobium - 95	Ci	N/D	N/D	N/D	N/D
Zirconium - 95	Ci	N/D	N/D	N/D	N/D

TABLE 1C

NORTH ANNA POWER STATION

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

GROUND LEVEL GASEOUS EFFLUENT RELEASES FOR 01/95 - 12/95

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	UNITS	3rd QUARTER	4th QUARTER	3rd QUARTER	4th QUARTER
Particulates: (cont.)					
Strontium - 89	Ci	N/D	N/D	N/D	N/D
Strontium - 90	Ci	N/D	N/D	N/D	N/D
Cesium - 134	Ci	N/D	N/D	N/D	N/D
Cesium - 137	Ci	2.61E-6	N/D	N/D	N/D
Barium - 140	Ci	N/D	N/D	N/D	N/D
Lanthanum - 140	Ci	N/D	N/D	N/D	N/D
Cerium - 141	Ci	N/D	N/D	N/D	N/D
Cerium - 144	Ci	N/D	N/D	N/D	N/D
Other (Specify)					
Molybdenum - 99 (T½ < 8 days)	Ci	N/D	N/D	N/D	N/D
Antimony - 122 (T½ < 8 days)	Ci	N/D	N/D	N/D	N/D
Cesium - 138 (T½ < 8 days)	Ci	N/D	N/D	N/D	N/D
Total for Period (T½ > 8 days)	Ci	9.83E-6	N/D	N/D	N/D
Total for Period (T½ < 8 days)	Ci	N/D	N/D	N/D	N/D
Total for Period	Ci	9.83E-6	N/D	N/D	N/D
GROSS ALPHA:	Ci	4.62E-6	3.49E-6	N/D	N/D
TRITIUM:	Ci	6.74E+1	9.60E+0	5.32E-1	9.66E-6

TABLE 2A
NORTH ANNA POWER STATION
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
LIQUID EFFLUENTS-SUMMATION OF ALL RELEASES FOR (01/95 - 12/95)

	UNITS	1st QUARTER	2nd QUARTER	ESTIMATED TOTAL PERCENT ERROR (%)
A. Fission and Activation Products:				
1. Total release (not including tritium, noble gas, and gross alpha).	Curies	1.12E-1	7.47E-2	2.00E+1
2. Average diluted concentration during the period.	μCi/ml	2.51E-10	1.51E-10	
3. Percent of applicable limit (T.S.)	%	1.85E-3	1.64E-4	
B. Tritium:				
1. Total release activity.	Curies	3.49E+2	1.77E+2	2.00E+1
2. Average diluted concentration during the period.	μCi/ml	7.81E-7	3.57E-7	
3. Percent of applicable limit (T.S.)	%	7.82E-3	3.59E-3	
C. Dissolved and Entrained Gases:				
1. Total release activity.	Curies	3.51E-4	4.60E-5	2.00E+1
2. Average diluted concentration during the period.	μCi/ml	7.85E-13	9.27E-14	
3. Percent of applicable limit (T.S.)	%	3.92E-7	4.65E-8	
D. Gross Alpha Radioactivity:				
1. Total release activity.	Curies	1.42E-4	N/D	2.00E+1
E. Volume of waste released: (prior to dilution).	Liters	7.15E+7	5.90E+7	3.00E+0
F. Total volume of dilution water used during the period.	Liters	4.47E+11	4.96E+11	3.00E+0

TABLE 2A
NORTH ANNA POWER STATION
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
LIQUID EFFLUENTS-SUMMATION OF ALL RELEASES FOR (01/95 - 12/95)

	UNITS	3rd QUARTER	4th QUARTER	ESTIMATED TOTAL PERCENT ERROR (%)
A. **Fission and Activation Products**:				
1. Total release (not including tritium, noble gas, and gross alpha).	Curies	3.66E-2	1.29E-1	2.00E+1
2. Average diluted concentration during the period.	μCi/ml	4.20E-11	1.83E-10	
3. Percent of applicable limit (T.S.)	%	8.06E-5	2.37E-4	
B. **Tritium**:				
1. Total release activity.	Curies	5.47E+1	3.96E+2	2.00E+1
2. Average diluted concentration during the period.	μCi/ml	6.28E-8	5.63E-7	
3. Percent of applicable limit (T.S.)	%	6.28E-4	5.62E-3	
C. **Dissolved and Entrained Gases**:				
1. Total release activity.	Curies	3.00E-4	N/D	2.00E+1
2. Average diluted concentration during the period.	μCi/ml	3.44E-13	N/D	
3. Percent of applicable limit (T.S.)	%	1.72E-7	N/A	
D. **Gross Alpha Radioactivity**:				
1. Total release activity.	Curies	N/D	N/D	2.00E+1
E. **Volume of waste released: (prior to dilution).**	Liters	7.08E+7	5.41E+7	3.00E+0
F. **Total volume of dilution water used during the period.**	Liters	8.71E+11	7.04E+11	3.00E+0

TABLE 2B
NORTH ANNA POWER STATION
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
LIQUID EFFLUENT RELEASES FOR 01/95 - 12/95

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	UNITS	1st QUARTER	2nd QUARTER	1st QUARTER	2nd QUARTER
Fission & Activation Products:					
Manganese - 54	Ci	7.47E-4	4.26E-4	N/D	N/D
Iron - 55	Ci	4.95E-2	N/D	N/D	N/D
Cobalt - 58	Ci	1.96E-2	2.99E-2	N/D	N/D
Cobalt - 60	Ci	2.52E-2	1.04E-2	N/D	N/D
Strontium - 89	Ci	N/D	N/D	N/D	N/D
Strontium - 90	Ci	N/D	N/D	N/D	N/D
Niobium - 95	Ci	2.11E-3	2.08E-3	N/D	N/D
Ruthenium - 106	Ci	1.94E-4	1.14E-4	N/D	N/D
Silver - 110m	Ci	6.55E-3	1.16E-2	N/D	N/D
Iodine - 131	Ci	7.57E-5	N/D	N/D	N/D
Iodine - 133	Ci	4.82E-5	N/D	N/D	N/D
Cesium - 134	Ci	N/D	N/D	N/D	N/D
Cesium - 136	Ci	N/D	N/D	N/D	N/D
Cesium - 137	Ci	3.24E-4	5.63E-4	N/D	N/D
Barium - 140	Ci	N/D	N/D	N/D	N/D
Cerium - 141	Ci	N/D	N/D	N/D	N/D
Antimony - 125	Ci	1.57E-3	N/D	N/D	N/D
Other (Specify)					
Sodium - 24 (T½ < 8 days)	Ci	N/D	N/D	N/D	N/D
Antimony - 122 (T½ < 8 days)	Ci	N/D	N/D	N/D	N/D
Chromium - 51	Ci	4.36E-3	1.29E-2	N/D	N/D
Iron - 59	Ci	6.87E-4	4.69E-3	N/D	N/D
Zirconium - 95	Ci	3.25E-4	1.25E-3	N/D	N/D
Indium-113m (T½ < 8 days)	Ci	N/D	3.64E-4	N/D	N/D
Zinc - 65	Ci	N/D	N/D	N/D	N/D
Rhodium - 106 (T½ < 8 days)	Ci	1.94E-4	1.14E-4	N/D	N/D
Tellurium - 129	Ci	N/D	N/D	N/D	N/D
Tellurium - 129m	Ci	N/D	N/D	N/D	N/D
Tin-113	Ci	N/D	3.64E-4	N/D	N/D
Total for Period	Ci	1.12E-1	7.47E-2	N/D	N/D

TABLE 2B
NORTH ANNA POWER STATION
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
LIQUID EFFLUENT RELEASES FOR 01/95 - 12/95

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	UNITS	1st QUARTER	2nd QUARTER	1st QUARTER	2nd QUARTER
Dissolved & Entrained Noble Gases:					
Xenon - 133	Ci	8.40E-5	N/D	N/D	N/D
Xenon - 133m	Ci	N/D	N/D	N/D	N/D
Xenon - 135	Ci	N/D	N/D	N/D	N/D
Xenon - 135m	Ci	N/D	N/D	N/D	N/D
Other (Specify)					
Argon - 41	Ci	2.67E-4	4.60E-5	N/D	N/D
Total for Period	Ci	3.51E-4	4.60E-5	N/D	N/D
Tritium	Ci	3.49E+2	1.77E+2	N/D	1.81E-3
Gross Alpha	Ci	1.42E-4	N/D	N/D	N/D

TABLE 2B

NORTH ANNA POWER STATION

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

LIQUID EFFLUENT RELEASES FOR 01/95 - 12/95

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	UNITS	3rd QUARTER	4th QUARTER	3rd QUARTER	4th QUARTER
Fission & Activation Products:					
Manganese - 54	Ci	2.89E-4	1.18E-3	N/D	N/D
Iron - 55	Ci	N/D	5.26E-2	N/D	N/D
Cobalt - 58	Ci	7.34E-3	9.12E-3	N/D	N/D
Cobalt - 60	Ci	1.21E-2	3.29E-2	N/D	N/D
Strontium - 89	Ci	N/D	N/D	N/D	N/D
Strontium - 90	Ci	N/D	N/D	N/D	N/D
Niobium - 95	Ci	1.52E-3	1.74E-3	N/D	N/D
Ruthenium - 106	Ci	N/D	N/D	N/D	N/D
Silver - 110m	Ci	1.47E-2	2.58E-2	N/D	N/D
Iodine - 131	Ci	N/D	N/D	N/D	N/D
Iodine - 133	Ci	N/D	N/D	N/D	N/D
Cesium - 134	Ci	N/D	N/D	N/D	N/D
Cesium - 137	Ci	8.49E-5	1.33E-4	N/D	N/D
Barium - 140	Ci	N/D	N/D	N/D	N/D
Lanthanum - 140	Ci	N/D	N/D	N/D	N/D
Cerium - 141	Ci	N/D	N/D	N/D	N/D
Antimony - 125	Ci	4.00E-4	5.23E-3	N/D	N/D
Other (Specify)					
Antimony - 122 (T½ < 8 days)	Ci	N/D	N/D	N/D	N/D
Chromium - 51	Ci	N/D	N/D	N/D	N/D
Iron - 59	Ci	N/D	N/D	N/D	N/D
Zirconium - 95	Ci	1.64E-4	1.67E-4	N/D	N/D
Indium - 113m (T½ < 8 days)	Ci	N/D	N/D	N/D	N/D
Zinc - 65	Ci	N/D	N/D	N/D	N/D
Rhodium - 106 (T½ < 8 days)	Ci	N/D	N/D	N/D	N/D
Cobalt - 57	Ci	N/D	N/D	N/D	N/D
Nickel - 65	Ci	N/D	N/D	N/D	N/D
Technetium - 99m	Ci	N/D	N/D	N/D	N/D
Ruthenium - 103	Ci	N/D	N/D	N/D	N/D
Tin - 113	Ci	N/D	N/D	N/D	N/D
Total for Period	Ci	3.66E-2	1.29E-1	N/D	N/D

TABLE 2B
NORTH ANNA POWER STATION
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
LIQUID EFFLUENT RELEASES FOR 01/95 - 12/95

		CONTINUOUS MODE		BATCH MODE	
NUCLIDES RELEASED	UNITS	3rd QUARTER	4th QUARTER	3rd QUARTER	4th QUARTER
Dissolved & Entrained Noble Gases:					
Xenon - 133	Ci	N/D	N/D	N/D	N/D
Xenon - 133m	Ci	N/D	N/D	N/D	N/D
Xenon - 135	Ci	4.81E-5	N/D	N/D	N/D
Xenon - 135m	Ci	N/D	N/D	N/D	N/D
Other (Specify)					
Argon - 41	Ci	2.52E-4	N/D	N/D	N/D
Total for Period	Ci	3.00E-4	N/D	N/D	N/D
Tritium	Ci	5.47E+1	3.96E+2	N/D	N/D
Gross Alpha	Ci	N/D	N/D	N/D	N/D

TABLE 3

NORTH ANNA POWER STATION

RADIOACTIVE EFFLUENT RELEASE REPORT

SUMMATION OF SOLID RADIOACTIVE WASTE AND IRRADIATED FUEL SHIPMENTS

FOR 01-01-95 THROUGH 12-31-95

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

1.	Type of Waste	Unit	12-Month Period	Estimated Total Percent Error (%)
a.	Spent resins, sludges, filters, filter sludge, evaporator bottoms, etc.	m^3	6.98E+1*	2.50E+1
		Ci	8.05E+2	2.50E+1
b.	Dry compressible waste, contaminated equipment, etc.	m^3	6.61E+2**	2.50E+1
		Ci	1.58E+1	2.50E+1
c.	Irradiated components, control rods, etc.	m^3	0.00E0	0.00E0
		Ci	0.00E0	0.00E0
d.	Other (describe) Waste Oil	m^3	4.10E+0***	2.50E+1
		Ci	1.91E-2	2.50E+1

2. Estimate of major nuclide composition (by type of waste)

a.	Ni-63	35.9%	2.87E+2	2.50E+1
	Co-60	28.7%	2.29E+2	2.50E+1
	Cs-137	5.7%	4.59E+1	2.50E+1
	Fe-55	12.0%	9.56E+1	2.50E+1
	Cs-134	2.8%	2.27E+1	2.50E+1
	Co-58	13.1%	1.05E+2	2.50E+1
	Mn-54	1.3%	1.03E+1	2.50E+1
b.	Mn-54	2.0%	3.15E-1	2.50E+1
	Co-58	10.1%	1.59E0	2.50E+1
	Fe-55	77.6%	1.22E+1	2.50E+1
	Co-60	4.3%	6.76E-1	2.50E+1
	Ni-63	4.9%	7.77E-1	2.50E+1
c.				

TABLE 3
NORTH ANNA POWER STATION
RADIOACTIVE EFFLUENT RELEASE REPORT
SUMMATION OF SOLID RADIOACTIVE WASTE AND IRRADIATED FUEL SHIPMENTS
FOR 01-01-95 THROUGH 12-31-95

2. Estimate of major nuclide composition (by type of waste) (cont.)	Unit	12-Month Period	Estimated Total Percent Error (%)
d. Ce-144	1.2%	2.24E-4	2.50E+1
Cs-137	9.6%	1.84E-3	2.50E+1
Co-60	3.1%	5.85E-4	2.50E+1
Ni-63	38.0%	7.27E-3	2.50E+1
Fe-55	32.8%	6.27E-3	2.50E+1
Sr-90	13.6%	2.60E-3	2.50E+1
Ag-110m	1.8%	3.48E-4	2.50E+1

3. Solid Waste Disposition

Number of Shipments	Mode of Transportation	Destination
11	Truck	Barnwell, SC
14	Truck	Oak Ridge, TN (SEG)
4	Truck	Wampum, PA (ALARON)

B. Irradiated Fuel Shipments (Disposition)

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

* 1 shipment of resin and 2 shipments of sludge were shipped from North Anna to a Licensed Waste Processor for volume reduction. Therefore, the volume listed for this type is not representative of actual volume buried. The total volume buried for this reporting period was 74.3 m^3.

** 16 shipments of dry compressible waste/contaminated equipment were shipped from North Anna to Licensed Waste Processors for volume reduction. Therefore, the volume listed for this type is not representative of the actual volume buried. The total volume buried for this reporting period was 54.1 m^3.

*** 2 shipments of waste oil and 1 shipment of grease were shipped from North Anna to a Licensed Waste Processor for incineration. Therefore, the volume listed for this type is not representative of the actual volume buried. The total volume buried for this reporting period was 0.00 m^3.

ATTACHMENT 2

ANNUAL AND QUARTERLY DOSES

(01/95 - 12/95)

An assessment of radiation doses to the maximum exposed member of the public due to radioactive liquid and gaseous effluents released from the site for each calendar quarter for the calendar year of this report, along with an annual total of each effluent pathway will be made pursuant to the ODCM Section 6.7.2.

Liquid Effluents:

	1st Quarter	2nd Quarter	3rd Quarter	4th Quarter	Annual Total
Total Body Dose (mrem)	9.90E-2	5.64E-2	1.[illegible]-2	1.09E-1	2.81E-1
Critical Organ Dose (mrem)	1.02E-1	6.09E-2	1.68E-2	1.10E-1	2.89E-1

Gaseous Effluents:

	1st Quarter	2nd Quarter	3rd Quarter	4th Quarter	Annual Total
Noble Gas Gamma Dose (mrad)	1.21E-2	2.85E-4	1.51E-4	1.04E-2	2.29E-2
Noble Gas Beta Dose (mrad)	5.67E-3	7.16E-4	5.72E-4	3.74E-3	1.07E-2
Critical Organ Dose for I-131, I-133, H-3, Particulates with T½ > 8 days (mrem)	4.17E-3	1.31E-2	3.79E-3	6.19E-4	2.18E-2

ATTACHMENT 3

REVISIONS TO OFFSITE DOSE CALCULATION MANUAL

(ODCM)

(01/95 - 12/95)

As required by Technical Specification 6.15, revisions to the ODCM, effective for the time period covered by this report, are summarized in this attachment.

There are two procedure changes to Revision 6 and one procedure revision implemented during the period January 1 through December 31, 1995. Included in this attachment are the revision summaries and associated page changes to the ODCM corresponding to North Anna's procedure changes. Revision 7, effective October 31, 1995, incorporated the two procedure changes and additional changes described in the revision summary.

The changes which relate to North Anna Power Station are the following:

1. 6-PN1 - a correction of Emergency TLD locations and reference to Deviation Report, DR N94-1137, which was implemented January 3, 1995, and

2. Revision 7 - correction of units of I-131 concentration contained in the definition of Dose Equivalent I-131, which was implemented October 31, 1995.

FOR INFORMATION ONLY

VIRGINIA POWER

Station Administrative Procedure

Title: **Offsite Dose Calculation Manual**

Lead Department: **Radiological Protection**

Procedure Number	Revision Number	Effective Date
VPAP-2103	6-PN1	1-3-95

Revision Summary

- Reference commitment 3.2.2 in Step 6.6.1.a.2 and Attachment 23 to ensure the commitment is not deleted at a later date.
- Correct page number referenced in Attachment 6 page 1 of 5 definitions fi and Ai. Add definition of "21" in equation Ai on Attachment 6 page 2 of 5.
- On Attachment 7 corrected "Total Body Ai and Critical Organ Ai" to "Total Body Bi and Liver Bi".
- Added alphabetic identifiers to first three items in first column on Attachment 10 and 11.
- Changed note 1 on Attachment 17 page 3 of 3 from "Automatic isolation of this pathway" to "Automatic actuation of the valves in this pathway".
- Change step 6.7.2.a.3 to include classification of unplanned liquid and gaseous effluent releases for the Annual Radioactive Effluent Release Report.
- Delete Bi-monthly River Water samples from Attachment 22 page 3 of 4.
- Change location of sample points for Oysters.
- Delete sediment requirements at Burwell's Bay and Newport News
- Delete sample point of clams at Jamestown. The number of required samples was reduced from 5 to 4.
- Change collection frequency for fish and invertebrates in sections a and b on Attachment 20 page 2 of 3 from Bi-monthly to Semi-Annually.
- Correct distance location of TLD on Attachment 22 page 1 of 4 from .33 to .29 miles.
- Incorporate PN&S1. Change equipment mark numbers to new format.
- Changed Attachments 14 and 16 to include mark numbers for VG-RM-104 and ventilation flow rate monitors.
- **E-Par PN1 Correct TLD locations on Attachment 23 and referenced DR N94-1137. North Anna Only**

Surry Power Station	North Anna Power Station
Approved by: N/A — SNSOC Chairman N/A — Date	Approved by: W. R. Matthews — SNSOC Chairman 1-3-95 — Date
Approved by: N/A — Station Manager N/A — Date	Approved by: J. A. Stall — Station Manager 1-3-95 — Date

Approved by: N/A — Vice President-Nuclear Operations — N/A — Date

VIRGINIA POWER

Station Administrative Procedure

Title: Offsite Dose Calculation Manual

Lead Department: Radiological Protection

Procedure Number	Revision Number	Effective Date
VPAP-2103	7	10-31-95

Revision Summary

- Incorporates PS1: Changes ATTACHMENT 22, Surry Environmental Sampling Locations to replace Walnut Point oyster sampling location with Kingsmill oyster sampling location on page 128
- Incorporates PN1: Changes ATTACHMENT 23, North Anna Environmental Sampling Locations to correct Emergency TLD locations on pages 129 and 130, adds reference 3.1.22, Deviation Report N94-1137, Improper Placement of Emergency TLDs to page 9
- Corrects units of I^{131} concentration to µCi/cc at 4.5, Dose Equivalent I-131 on page 10
- Reflects Surry Core Uprate to 2546 MWt at 4.13, Rated Thermal Power on page 12

Surry Power Station		North Anna Power Station	
Approved by: Bryan C. Stamm (signature) SNSOC Chairman	9/21/95 Date	Approved by: DAHeal (signature) SNSOC Chairman	10-5-95 Date
Approved by: (signature) Station Manager	9-21-95 Date	Approved by: (signature) Station Manager	10/5/95 Date

Approved by: N/A
Vice President-Nuclear Operations
Date

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1.0 PURPOSE

The Offsite Dose Calculation Manual (ODCM) establishes requirements for the Radioactive Effluent and Radiological Environmental Monitoring Programs. Methodology and parameters are provided to calculate offsite doses resulting from radioactive gaseous and liquid effluents, to calculate gaseous and liquid effluent monitoring alarm/trip setpoints, and to conduct the Environmental Monitoring Program. Requirements are established for the Annual Radiological Environmental Operating Report and the Annual Radioactive Effluent Release Report required by Station Technical Specifications. Calculation of offsite doses due to radioactive liquid and gaseous effluents are performed to assure that:

- Concentration of radioactive liquid effluents to the unrestricted area will be limited to ten times the effluent concentration values of 10 CFR 20, Appendix B, Table 2, Column 2, for radionuclides other than dissolved or entrained noble gases and 2E-4 μCi/ml for dissolved or entrained noble gases.
- Exposure to the maximum exposed member of the public in the unrestricted area from radioactive liquid effluents will not result in doses greater than the liquid dose limits of 10 CFR 50, Appendix I
- Dose rate at and beyond the site boundary from radioactive gaseous effluents will be limited to:
  - Noble gases – less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin
  - I^{131}, I^{133}, and H^3, and all radionuclides in particulate form with half-lives greater than 8 days – less than or equal to a dose rate of 1500 mrem/yr to any organ
- Exposure from radioactive gaseous effluents to the maximum exposed member of the public in the unrestricted area will not result in doses greater than the gaseous dose limits of 10 CFR 50, Appendix I, and
- Exposure to a real individual will not exceed 40 CFR 190 dose limits

2.0 SCOPE

This procedure applies to the Radioactive Effluent and Environmental Monitoring Programs at Surry and North Anna Stations.

3.0 REFERENCES/COMMITMENT DOCUMENTS

3.1 References

3.1.1 10 CFR 20, Standards for Protection Against Radiation

3.1.2 10 CFR 50, Domestic Licensing of Production and Utilization Facilities

3.1.3 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operations

3.1.4 TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites

3.1.5 Regulatory Guide 1.21, Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants, Rev. 1, U.S. NRC, June 1974

3.1.6 Regulatory Guide 1.109, Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR 50, Appendix I, Rev. 1, U.S. NRC, October 1977

3.1.7 Regulatory Guide 1.111, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors, Rev. 1, U.S. NRC, July 1977

3.1.8 Surry and North Anna Technical Specifications (Units 1 and 2)

3.1.9 NUREG-0324, XOQDOQ, Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations, U.S. NRC, September 1977

3.1.10 NUREG/CR-1276, Users Manual for the LADTAP II Program, U.S. NRC, May, 1980

3.1.11 NUREG-0597, User's Guide to GASPAR Code, U.S. NRC, June, 1980

3.1.12 Radiological Assessment Branch Technical Position on Environmental Monitoring, November, 1979, Rev. 1

3.1.13 NUREG-0133, Preparation of Radiological Effluent Technical Specifications for Nuclear Power Stations, October, 1978

3.1.14 NUREG-0543, February 1980, Methods for Demonstrating LWR Compliance With the EPA Uranium Fuel Cycle Standard (40 CFR Part 190)

3.1.15 NUREG-0472, Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors, Rev. 3, March 1982

3.1.16 Environmental Measurements Laboratory, DOE HASL 300 Manual

3.1.17 NRC Generic Letter 89-01, Implementation of Programmatic Controls for Radiological Effluent Technical Specifications (RETS) in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program

3.1.18 UFSAR (Surry and North Anna)

3.1.19 Nuclear Reactor Environmental Radiation Monitoring Quality Control Manual, IWL-0032-361

3.1.20 VPAP-2802, Notifications and Reports

3.1.21 NAPS Circulating Water System Modifications

a. DC-85-37-1 Unit 1

b. DC-85-38-2 Unit 2

3.1.22 Deviation Report N94-1137, Improper Placement of Emergency TLDs

3.2 Commitment Documents

3.2.1 Quality Assurance Audit Report Number C 90-22, Management Safety Review Committee, Observation 03C, January 17, 1991

3.2.2 Quality Assurance Audit Report Number 91-03, Observation 08N

3.2.3 Quality Assurance Audit Report Number 92-03, Observation 02N

3.2.4 Quality Assurance Audit Report Number 92-03, Observation 04NS (Item 2)

4.0 DEFINITIONS

4.1 Channel Calibration

Adjustment, as necessary, of the channel output so it responds with the necessary range and accuracy to known values of the parameter the channel monitors. It encompasses the entire channel, including the sensor and alarm and/or trip functions and the Channel Functional Test. The Channel Calibration can be performed by any series of sequential, overlapping, or total channel steps so the entire channel is calibrated.

4.2 Channel Check

A qualitative assessment, by observation, of channel behavior during operation. This assessment includes, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation channels measuring the same parameter.

4.3 Channel Functional Test

There are two types of Channel Functional Tests.

4.3.1 Analog Channel

Injection of a simulated signal into a channel, as close to the sensor as practicable, to verify Operability, including alarm and/or trip functions.

4.3 Channel Functional Test (continued)

4.3.2 Bistable Channel

Injection of a simulated signal into a sensor to verify Operability, including alarm and/or trip functions.

4.4 Critical Organ

That organ, which has been determined to be the maximum exposed organ based on an effluent pathway analysis, thereby ensuring the dose and dose rate limitations to any organ will not be exceeded.

4.5 Dose Equivalent I-131

That concentration of I^{131} (µCi/cc) that 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. Thyroid dose conversion factors for this calculation are listed in Table III of TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites. Thyroid dose conversion factors from NRC Regulatory Guide 1.109, Revision 1, may be used (**Surry**).

4.6 Frequency Notations

NOTE: Frequencies are allowed a maximum extension of 25 percent.

NOTATION	FREQUENCY
D - Daily	At least once per 24 hours
W - Weekly	At least once per 7 days
M - Monthly	At least once per 31 days
Q - Quarterly	At least once per 92 days
SA - Semi-annually	At least once per 184 days
R - Refueling	At least once per 18 months
S/U - Start-up	Prior to each reactor start-up
P - Prior to release	Completed prior to each release
N.A. - Not applicable	Not applicable
DR - During the release	At least once during each release

4.7 Gaseous Radwaste Treatment System

A system that reduces radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing delay or holdup to reduce total radioactivity prior to release to the environment. The system comprises the waste gas decay tanks, regenerative heat exchanger, waste gas charcoal filters, process vent blowers, waste gas surge tanks, and waste gas diaphragm compressor (**North Anna**).

4.8 General Nomenclature

χ = Chi: concentration at a point at a given instant (curies per cubic meter)

D = Deposition: quantity of deposited radioactive material per unit area (curies per square meter)

Q = Source strength (instantaneous; grams, curies)

= Emission rate (continuous; grams per second, curies per second)

= Emission rate (continuous line source; grams per second per meter)

4.9 Lower Limit of Detection (LLD)

The smallest concentration of radioactive material in a sample that will yield a net count (above system background) that can be detected with 95 percent probability with only 5 percent probability of falsely concluding that a blank observation represents a "real" signal.

4.10 Members of the Public

Individuals who, by virtue of their occupational status, have no formal association with the Station. This category includes non-employees of Virginia Power who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with Station functions. This category does not include non-employees such as vending machine servicemen or postal workers who, as part of their formal job function, occasionally enter an area that is controlled by Virginia Power to protect individuals from exposure to radiation and radioactive materials.

4.11 Operable - Operability

A system, subsystem, train, component, or device is operable or has operability when it is capable of performing its specified functions and all necessary, attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its functions are also capable of performing their related support functions.

4.12 Purge - Purging

Controlled discharge of air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, so that replacement air or gas is required to purify the confinement.

4.13 Rated Thermal Power

Total reactor core heat transfer rate to reactor coolant.

- Surry – 2546 Megawatts Thermal (MWt)
- North Anna – 2893 MWt

4.14 Site Boundary

The line beyond which Virginia Power does not own, lease, or otherwise control the land.

4.15 Source Check

A qualitative assessment of channel response when a channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

4.16 Special Report

A report to NRC to comply with Subsections 6.2, 6.3, or 6.5 of this procedure. Also refer to VPAP-2802, Notifications and Reports.

4.17 Thermal Power

Total reactor core heat transfer rate to the reactor coolant.

4.18 Unrestricted Area

Any area at or beyond the site boundary, access to which is neither limited nor controlled by Virginia Power for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the site boundary used for residential quarters or for industrial, commercial, institutional or recreational purposes.

4.19 Ventilation Exhaust Treatment System

A system that reduces gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and High Efficiency Particulate Air (HEPA) filters to remove iodines and particulates from a gaseous exhaust stream prior to 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 Ventilation Exhaust Treatment System components.

5.0 RESPONSIBILITIES

5.1 Superintendent Radiological Protection

The Superintendent Radiological Protection is responsible for:

5.1.1 Establishing and maintaining procedures for surveying, sampling, and monitoring radioactive effluents and the environment.

5.1.2 Surveying, sampling, and analyzing plant effluents and environmental monitoring, and documenting these activities.

5.1.3 Analyzing plant effluent trends and recommending actions to correct adverse trends.

5.1.4 Preparing Effluent and Environmental Monitoring Program records.

5.2 Superintendent Operations

The Superintendent Operations is responsible for requesting samples, analyses, and authorization to release effluents.

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6.0 INSTRUCTIONS

NOTE: Meteorological, liquid, and gaseous pathway analyses are presented in Attachments 28 and 29, Meteorological, Liquid, and Gaseous Pathway Analysis.

6.1 Sampling and Monitoring Criteria

6.1.1 Surveys, sampling, and analyses shall use instruments calibrated for the type and range of radiation monitored and the type of discharge monitored.

6.1.2 Installed monitoring systems shall be calibrated for the type and range of radiation or parameter monitored.

6.1.3 A sufficient number of survey points shall be used or samples taken to adequately assess the status of the discharge monitored.

6.1.4 Samples shall be representative of the volume and type of discharge monitored.

6.1.5 Surveys, sampling, analyses, and monitoring records shall be accurately and legibly documented, and sufficiently detailed that the meaning and intent of the records are clear.

6.1.6 Surveys, analyses, and monitoring records shall be reviewed for trends, completeness, and accuracy.

6.2 Liquid Radioactive Waste Effluents

6.2.1 **Liquid Effluent Concentration Limitations**

a. Liquid waste concentrations discharged from the Station shall not exceed the following limits:

1. For radionuclides (other than dissolved or entrained noble gases), liquid effluent concentrations released to unrestricted areas shall not exceed ten times the effluent concentration values specified in 10 CFR 20, Appendix B, Table 2, Column 2.

2. For dissolved or entrained noble gases, concentrations shall not exceed 2E-4 μCi/ml.

b. If the concentration of liquid effluent exceeds the limits in 6.2.1.a., promptly reduce concentrations to within limits.

6.2.1 **Liquid Effluent Concentration Limitations (continued)**

c. Daily concentrations of radioactive materials in liquid waste released to unrestricted areas shall meet the following:

$$\frac{\text{Volume of Waste Discharged} + \text{Volume of Dilution Water}}{\text{Volume of Waste Discharged} \times \sum_{i} \frac{\mu Ci/ml_i}{ACW_i}} \geq 1 \quad (1)$$

where:

$\mu Ci/ml_i$ = the concentration of nuclide i in the liquid effluent discharge

ACW_i = ten times the effluent concentration value in unrestricted areas of nuclide i, expressed as μCi/ml from 10 CFR 20, Appendix B, Table 2, Column 2 for radionuclides other than noble gases, and 2E-4 μCi/ml for dissolved or entrained noble gases

6.2.2 **Liquid Monitoring Instrumentation**

a. **Radioactive Liquid Effluent Monitoring Instrumentation**

Radioactive liquid effluent monitoring instrumentation channels shown on Attachments 1 and 2, Radioactive Liquid Effluent Monitoring Instrumentation, shall be operable with their alarm/trip setpoints set to ensure that 6.2.1.a. limits are not exceeded.

1. Alarm/trip setpoints of these channels shall be determined and adjusted in accordance with 6.2.2.d., Setpoint Calculation.
2. If a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required by 6.2.2.a., perform one of the following:
   - Promptly suspend release of radioactive liquid effluents monitored by the affected channel
   - Declare the channel inoperable
   - Change the setpoint to an acceptable, conservative value

b. **Radioactive Liquid Effluent Monitoring Instrumentation Operability**

Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated operable by performing a Channel Check, Source Check, Channel Calibration, and Channel Functional Test at the frequencies shown in Attachments 3 and 4, Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements.

1. If the number of operable channels is less than the minimum required by the tables in Attachment 1 or 2, perform the action shown in those tables.
2. Attempt to return the instruments to operable status within 30 days. If unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. **Applicable Monitors**

Liquid effluent monitors for which alarm/trip setpoints shall be determined are:

Release Point	Instrument Number	
	North Anna	Surry
Liquid Radwaste Effluent Line	1-LW-RM-111	N/A
Service Water System Effluent Line	1-SW-RM-108	1-SW-RM-107 A, B, C, D
Condenser Circulating Water Line	1-SW-RM-130 2-SW-RM-230	1-SW-RM-120 2-SW-RM-220
Radwaste Facility Effluent Line	N/A	1-RRM-RITS-131

6.2.2 **Liquid Monitoring Instrumentation (continued)**

d. **Setpoint Calculation**

NOTE: This methodology does not preclude use of more conservative setpoints.

1. Maximum setpoint values shall be calculated by:

$$S = \frac{CF_D}{F_E} \tag{2}$$

where:

S = the setpoint, in µCi/ml, of the radioactivity monitor measuring the radioactivity concentration in the effluent line prior to dilution

C = the effluent concentration limit for the monitor used to implement 10 CFR 20 for the Station, in µCi/ml

F_E = maximum design pathway effluent flow rate

F_D = dilution water flow rate calculated as:

(Surry) D = F_E + (200,000 gpm x number of circ. pumps in service)

(N. Anna) D = F_E + (218,000 gpm x number of circ. pumps in service)

2. Each of the condenser circulating water channels (Surry: SW-120, SW-220) (North Anna: SW-130, SW-230) monitors the effluent (service water, including component cooling service water, circulating water, and liquid radwaste) in the circulating water discharge tunnel beyond the last point of possible radioactive material addition. No dilution is assumed for this pathway. Therefore, Equation (2) becomes:

$$S = C \tag{3}$$

The setpoint for Station monitors used to implement 10 CFR 20 for the site becomes the effluent concentration limit.

3. In addition, for added conservatism, setpoints shall be calculated for the liquid radwaste effluent line (North Anna: LW-111), the service water system effluent line (Surry: SW-107 A, B, C, and D, North Anna: SW-108), and the Radwaste Facility effluent line (Surry: RRM-131).

4. For the liquid radwaste effluent line, Equation (2) becomes:

$$S = \frac{CF_D K_{LW}}{F_E} \tag{4}$$

where:

K_{LW} = The fraction of the effluent concentration limit, used to implement 10 CFR 20 for the site, attributable to the liquid radwaste effluent line pathway

5. For the service water system effluent line, Equation (2) becomes:

$$S = \frac{CF_D K_{SW}}{F_E} \tag{5}$$

where:

K_{SW} = The fraction of the effluent concentrat'on limit, used to implement 10 CFR 20 for the Station, attributable to the service water effluent line pathway

6. For the Radwaste Facility effluent line, Equation (2) becomes:

$$S = \frac{CF_D K_{RW}}{F_E} \tag{6}$$

where:

K_{RW} = The fraction of the effluent concentration limit, used to implement 10 CFR 20 attributable to the Radwaste Facility effluent line pathway

7. The sum $K_{LW} + K_{SW} + K_{RW}$ shall not be greater than 1.0.

6.2.3 **Liquid Effluent Dose Limit**

a. **Requirement**

At least once per 31 days, perform the dose calculations in 6.2.3.c. and 6.2.3.d. to ensure the dose or dose commitment to the maximum exposed member of the public from radioactive materials in liquid releases (from each reactor unit) to unrestricted areas is limited to:

1. During any calendar quarter:
   - Less than or equal to 1.5 mrem to the total body
   - Less than or equal to 5 mrem to the critical organ
2. During any calendar year:
   - Less than or equal to 3 mrem to the total body
   - Less than or equal to 10 mrem to the critical organ

b. **Action**

If the calculated dose from release of radioactive materials in liquid effluents exceeds any of the above limits, prepare and submit to the NRC, within 30 days, a special report in accordance with VPAP-2802, Notifications and Reports, that identifies causes for exceeding limits and defines corrective actions taken to reduce releases of radioactive materials in liquid effluents to ensure that subsequent releases will be in compliance with the above limits.

c. **Surry Dose Contribution Calculations**

NOTE: Thyroid and GI-LLI organ doses must be calculated to determine which is the critical organ for the period being considered.

Dose contributions shall be calculated for all radionuclides identified in liquid effluents released to unrestricted areas based on the equation:

$$D = t\,F\,M \sum_i C_i A_i \qquad (7)$$

where:

Subscripts = i, refers to individual radionuclide

D = the cumulative dose commitment to the total body or critical organ from the liquid effluents for the period t, in mrem

t = the period for which C_i and F are averaged for all liquid releases, in hours

M = the mixing ratio (reciprocal of the dilution factor) at the point of exposure, dimensionless, 0.2 from Appendix 11A, Surry UFSAR

F = the near field average dilution factor for C_i during any liquid effluent release; the ratio of the average undiluted liquid waste flow during release to the average flow from the site discharge structure to unrestricted areas

C_i = the average concentration of radionuclide, i, in undiluted liquid effluent during the period t, from all liquid releases, in µCi/ml

A_i = the site-related ingestion dose commitment factor to the total body or critical organ of an adult for each identified principal gamma and beta emitter in mrem-ml per hr-µCi. Values for A_i are given in Attachment 5, Liquid Ingestion Pathway Dose Factors For Surry Power Station.

$$A_i = 1.14\text{ E+05 } (21BF_i + 5BI_i)\, DF_i \qquad (8)$$

where:

1.14 E+05 = 1 E+06 pCi/µCi x 1 E+03 ml/kg/(8760 hr/yr), units conversion factor

21 = adult fish consumption, kg/yr, from NUREG-0133

5 = adult invertebrate consumption, kg/yr, from NUREG-0133

BI_i = the bioaccumulation factor for nuclide i, in invertebrates, pCi/kg per pCi/l, from Table A-1 of Regulatory Guide 1.109, Rev. 1

BF_i = the bioaccumulation factor for nuclide i, in fish, pCi/kg per pCi/l, from Table A-1 of Regulatory Guide 1.109, Rev. 1

DF_i = the critical organ dose conversion factor for nuclide i, for adults, in mrem/pCi, from Table E-11 of Regulatory Guide 1.109, Rev. 1

6.2.3 **Liquid Effluent Dose Limit (continued)**

d. **North Anna Dose Contribution Calculations**

NOTE: Attachment 6, North Anna Liquid Ingestion Pathway Dose Factor Calculation provides the derivation for Equation (9).

Dose contribution shall be calculated for all radionuclides identified in liquid effluents released to unrestricted areas based on:

$$D = \sum_i Q_i \times B_i \tag{9}$$

Where:

Subscripts = i, refers to individual radionuclide

D = the cumulative dose commitment to the total body or critical organ from the liquid effluents for the period t, in mrem

B_i = Dose Commitment Factors (mrem/Ci) for adults. Values for B_i are provided in Attachment 7, North Anna Liquid Ingestion Pathway Dose Commitment Factors for Adults

Q_i = Total released activity for the considered period and the ith nuclide

$$Q_i = t \times C_i \times \text{Waste Flow} \tag{10}$$

Where:

t = the period for which C_i and F are averaged for all liquid releases, in hours

C_i = the average concentration of radionuclide, i, in undiluted liquid effluent during the period, t, from any liquid releases, in μCi/ml

e. **Quarterly Composite Analyses**

For radionuclides not determined in each batch or weekly composite, dose contribution to current monthly or calendar quarter cumulative summation may be approximated by assuming an average monthly concentration based on previous monthly or quarterly composite analyses. However, for reporting purposes, calculated dose contribution shall be based on the actual composite analyses.

6.2.4 **Liquid Radwaste Treatment**

a. **Requirement**

1. The Liquid Radwaste Treatment System and/or the Surry Radwaste Facility Liquid Waste System shall be used to reduce the radioactive materials in liquid waste prior to discharge when projected dose due to liquid effluent, from each reactor unit, to unrestricted areas would exceed 0.06 mrem to total body or 0.2 mrem to the critical organ in a 31-day period.

2. Doses due to liquid releases shall be projected at least once per 31 days.

b. **Action**

If radioactive liquid waste is discharged without treatment and in excess of the above limits prepare and submit to the NRC, within 30 days, a special report in accordance with VPAP-2802, Notifications and Reports, that includes the following:

1. An explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or sub-system, and the reason for the inoperability.

2. Actions taken to restore inoperable equipment to operable status.

3. Summary description of actions taken to prevent recurrence.

c. **Projected Total Body Dose Calculation**

1. Determine D_{TB}, the total body dose from liquid effluents in the previous 31-day period, per Equation (7) or Equation (9) (Surry and North Anna, respectively).

2. Estimate R_1, the ratio of the estimated volume of liquid effluent releases in the present 31-day period to the volume released in the previous 31-day period.

3. Estimate F_1, the ratio of the estimated liquid effluent radioactivity concentration in the present 31-day period to liquid effluent concentration in the previous 31-day period (µCi/ml).

4. Determine PD_{TB}, the projected total body dose in a 31-day period.

$$PD_{TB} = D_{TB}(R_1 F_1) \tag{11}$$

6.2.4 **Liquid Radwaste Treatment (continued)**

d. **Projected Critical Organ Dose Calculation**

Historical data pertaining to the volumes and radioactivity of liquid effluents released in connection with specific Station functions, such as maintenance or refueling outages, shall be used in projections as appropriate.

1. Determine D_o, the critical organ dose from liquid effluents in the previous 31-day period, per Equation (7) or Equation (9) (Surry and North Anna, respectively).
2. Estimate R_1 as in 6.2.4.c.2.
3. Estimate F_1 as in 6.2.4.c.3.
4. Determine PD_o = projected critical organ dose in a 31-day period.

$$PD_O = D_O(R_1F_1) \tag{12}$$

6.2.5 **Liquid Sampling**

Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis requirements in Attachments 8 and 9, Radioactive Liquid Waste Sampling and Analysis Program (Surry and North Anna, respectively).

6.3 Gaseous Radioactive Waste Effluents

6.3.1 **Gaseous Effluent Dose Rate Limitation**

a. **Requirement**

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

1. The dose rate limit for noble gases shall be $\leq$ 500 mrem/year to the total body and $\leq$ 3000 mrem/year to the skin.
2. The dose rate limit for I^{131}, I^{133}, for tritium, and for all radioactive materials in particulate form with half-lives greater than 8 days shall be $\leq$ 1500 mrem/year to the critical organ.

b. **Action**

1. If dose rates exceed 6.3.1.a. limits, promptly decrease the release rate to within the above limits.

2. Dose rates due to noble gases in gaseous effluents shall be determined, continuously, to be within 6.3.1.a. limits.

3. Dose rates due to I^{131}, I^{133}, tritium, and all radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents shall be determined to be within the above limits by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified on Attachments 10 and 11, Radioactive Gaseous Waste Sampling and Analysis Program.

6.3.1 **Gaseous Effluent Dose Rate Limitation (continued)**

c. **Calculations of Gaseous Effluent Dose Rates**

1. The dose rate limit for noble gases shall be determined to be within the limit by limiting the release rate to the lesser of:

$$\sum_i [K_{ivv}\dot{Q}_{ivv} + K_{ipv}\dot{Q}_{ipv}] \leq 500\text{mrem/yr to the total body} \quad (13)$$

OR

$$\sum_i [(L_{ivv} + 1.1M_{ivv})\dot{Q}_{ivv} + (L_{ipv} + 1.1M_{ipv})\dot{Q}_{ipv}] \leq 3000\text{mrem/yr to the skin} \quad (14)$$

where:

Subscripts	=	vv, refers to vent releases from the building ventilation vent, including Radwaste Facility Ventilation Vent; pv, refers to the vent releases from the process vent; i, refers to individual radionuclide
K_{ivv}, K_{ipv}	=	The total body dose factor for ventilation vents or process vent release due to gamma emissions for each identified noble gas radionuclide i, in mrem/yr per Curie/sec. Factors are listed in Attachments 12 and 13, Gaseous Effluent Dose Factors (Surry and North Anna, respectively)
L_{ivv}, L_{ipv}	=	The skin dose factor for ventilation vents or process vent release due to beta emissions for each identified noble gas radionuclide i, in mrem/yr per Curie/sec. Factors are listed in Attachments 12 and 13
M_{ivv}, M_{ipv}	=	The air dose factor for ventilation vents or process vent release due to gamma emissions for each identified noble gas radionuclide, i, in mrad/yr per Curie/sec. Factors are listed in Attachments 12 and 13
$\dot{Q}_{ivv}\dot{Q}_{ipv}$	=	The release rate for ventilation vents or process vent of noble gas radionuclide i, in gaseous effluents in Curie/sec (per site)
1.1	=	The unit conversion factor that converts air dose to skin dose, in mrem/mrad

2. The dose rate limit for I^{131}, I^{133}, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days, shall be determined to be within the limit by restricting the release rate to:

$$\sum_i [P_{ivv}\dot{Q}_{ivv} + P_{ipv}\dot{Q}_{ipv}] \leq 1500 \text{mrem/yr to the critical organ} \quad (15)$$

where:

P_{ivv}, P_{ipv} = The critical organ dose factor for ventilation vents or process vent for I^{131}, I^{133}, H^3, and all radionuclides in particulate form with half-lives greater than 8 days, for the inhalation pathway, in mrem/yr per Curie/sec. Factors are listed in Attachments 12 and 13

$\dot{Q}_{ivv}\dot{Q}_{ipv}$ = The release rate for ventilation vents or process vent of I^{131}, I^{133}, H^3, and all radionuclides i, in particulate form with half-lives greater than 8 days, in gaseous effluents in Curie/sec (per site)

3. All gaseous releases, not through the process vent, are considered ground level and shall be included in the determination of $\dot{Q}_{ivv}$.

6.3.2 Gaseous Monitoring Instrumentation

a. **Requirement**

1. The radioactive gaseous effluent monitoring instrumentation channels shown in Attachment 14 or 15, Radioactive Gaseous Effluent Monitoring Instrumentation, shall be operable with alarm/trip setpoints set to ensure that 6.3.1.a. noble gas limits are not exceeded. Alarm/trip setpoints of these channels shall be determined and adjusted in accordance with 6.3.2.d.

2. Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated operable by Channel Checks, Source Checks, Channel Calibrations, and Channel Functional Tests at the frequencies shown in Attachment 16 or 17, Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements.

6.3.2 **Gaseous Monitoring Instrumentation (continued)**

b. **Action**

1. If a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required by 6.3.2.a.1, promptly:
   - Suspend the release of radioactive gaseous effluents monitored by the affected channel **and** declare the channel inoperable
   - or
   - Change the setpoint so it is acceptably conservative

2. If the number of operable channels is less than the minimum required by tables in Attachment 14 and 15, take the action shown in those tables.

3. Return instruments to operable status within 30 days. If unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. **Applicable Monitors**

Radioactive gaseous effluent monitors for which alarm/trip setpoints shall be determined are:

Release Point	Instrument Number	
	North Anna	Surry
Process Vent	1-GW-RM-102 1-GW-RM-178-1	1-GW-RM-102 1-GW-RM-130-1
Condenser Air Ejector	1-SV-RM-121 2-SV-RM-221	1-SV-RM-111 2-SV-RM-211
Ventilation Vent A	1-VG-RM-104 1-VG-RM-179-1	N/A
Ventilation Vent B	1-VG-RM-113 1-VG-RM-180-1	N/A
Ventilation Vent No. 1	N/A	1-VG-RM-104
Ventilation Vent No. 2	N/A	1-VG-RM-110 1-VG-RM-131-1
Radwaste Facility Vent	N/A	RRM-101

d. **Setpoint Calculations**

1. Setpoint calculations for each monitor listed in 6.3.2.c. shall maintain this relationship:

$$D \geq D_{pv} + D_{cae} + D_{vv} \tag{16}$$

where:

D = Step 6.3.1.a. dose limits that implement 10 CFR 20 for the Station, mrem/yr

D_{pv} = The noble gas site boundary dose rate from process vent gaseous effluent releases, mrem/yr

D_{cae} = The noble gas site boundary dose rate from condenser air ejector gaseous effluent releases, mrem/yr

D_{vv} = The noble gas site boundary dose rate from:

Surry: Summation of the Ventilation Vents 1, 2, and the Radwaste Facility vent gaseous effluent releases, mrem/yr

North Anna: Summation of Ventilation Vent A plus B gaseous effluent releases, mrem/yr

2. Setpoint values shall be determined by:

$$C_m = \frac{R_m \times 2.12\text{ E-03}}{F_m} \tag{17}$$

where:

m	=	The release pathway, process vent (pv), ventilation vent (vv) condenser air ejector (cae), or Radwaste Facility (rv)
C_m	=	The effluent concentration limit implementing 6.3.1.a. for the Station, µCi/ml
R_m	=	The release rate limit for pathway m determined from methodology in 6.3.1.c., using Xe^{133} as nuclide to be released, µCi/sec
2.12E-03	=	CFM per ml/sec
F_m	=	The maximum flow rate for pathway m, CFM

NOTE: According to NUREG-0133, the radioactive effluent radiation monitor alarm/trip setpoints should be based on the radioactive noble gases. It is not practicable to apply instantaneous alarm/trip setpoints to integrating monitors sensitive to radioiodines, radioactive materials in particulate form, and radionuclides other than noble gases.

6.3.3 Noble Gas Effluent Air Dose Limit

a. **Requirement**

1. The air dose in unrestricted areas due to noble gases released in gaseous effluents from each unit at or beyond the site boundary shall be limited to:
   - During any calendar quarter: ≤ 5 mrads for gamma radiation and ≤ 10 mrads for beta radiation
   - During any calendar year: ≤ 10 mrads for gamma radiation and ≤ 20 mrads for beta radiation

2. Cumulative dose contributions for noble gases for the current calendar quarter and current calendar year shall be determined in accordance with 6.3.3.c. at least once per 31 days.

b. **Action**

If the calculated air dose from radioactive noble gases in gaseous effluents exceeds any of the above limits, prepare and submit to the NRC, within 30 days, a special report in accordance with VPAP-2802, Notifications and Reports, that identifies the causes for exceeding the limits and defines corrective actions that have been taken to reduce releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the limits in 6.3.3.a.

c. **Noble Gas Effluent Air Dose Calculation**

Gaseous releases, not through the process vent, are considered ground level and shall be included in the determination of $\overline{Q}_{ivv}$.

The air dose to areas at or beyond the site boundary due to noble gases shall be determined by the followi. ~·

For gamma radiation:

$$D_g = 3.17E\text{-}08 \sum_i [M_{ivv}\overline{Q}_{ivv} + M_{ipv}\overline{Q}_{ipv}] \quad (18)$$

For beta radiation:

$$D_b = 3.17E\text{-}08 \sum_i [N_{ivv} \bar{Q}_{ivv} + N_{ipv} \bar{Q}_{ipv}] \qquad (19)$$

Where:

Subscripts	=	vv, refers to vent releases from the building ventilation vents, including the Radwaste Facility Ventilation Vent and air ejectors pv, refers to the vent releases from the process vent i, refers to individual radionuclide
D_g	=	the air dose for gamma radiation, in mrad
D_b	=	the air dose for beta radiation, in mrad
M_{ivv}, M_{ipv}	=	the air dose factors for ventilation vents or process vent release due to gamma emissions for each identified noble gas radionuclide i, in mrad/yr per Curie/sec. Factors are listed in Attachments 12 and 13
N_{ivv}, N_{ipv}	=	the air dose factor for ventilation vents or process vent release due to beta emissions for each identified noble gas radionuclide i, in mrad/yr per Curie/sec. Factors are listed in Attachments 12 and 13
$\bar{Q}_{ivv}$, $\bar{Q}_{ipv}$	=	the release for ventilation vents or process vent of noble gas radionuclide i, in gaseous effluents for 31 days, quarter, or year as appropriate in Curies (per site)

6.3.4 I-131, 133, H-3 & Radionuclides In Particulate Form Effluent Dose Limit

a. **Requirement**

1. Methods shall be implemented to ensure that the dose to any organ of a member of the public from I^{131}, I^{133}, tritium, and all radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents released from the site to unrestricted areas from each reactor unit shall be:
   - During any calendar quarter: $\leq$ 7.5 mrem to the critical organ
   - During any calendar year: $\leq$ 15 mrem to the critical organ

2. Cumulative dose contributions to a member of the public from I^{131}, I^{133}, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents released to unrestricted areas for the current calendar quarter and current calendar year shall be determined at least once per 31 days in accordance with 6.3.4.c. or 6.3.4.d.

b. **Action**

If the calculated dose from the release of I^{131}, I^{133}, tritium, and radionuclides in particulate form, with half-lives greater than 8 days, in gaseous effluents exceeds any of the above limits, prepare and submit to the NRC within 30 days, a special report in accordance with VPAP-2802, Notifications and Reports, that contains the:

1. Causes for exceeding limits.
2. Corrective actions taken to reduce releases.
3. Proposed corrective actions to be taken to assure that subsequent releases will be in compliance with limits stated in 6.3.4.a.

c. **Surry Dose Calculations**

Gaseous releases, not through the process vent, are considered ground level and shall be included in the determination of $\tilde{Q}_{ivv}$. Historical data pertaining to the volumes and radioactive concentrations of gaseous effluents released in connection to specific Station functions, such as containment purges, shall be used in the estimates, as appropriate.

1. The dose to the maximum exposed member of the public, attributable to gaseous effluents at and beyond the site boundary that contain I^{131}, I^{133}, tritium, and particulate-form radionuclides with half-lives greater than 8 days, shall be determined by:

$$D_r = 3.17E\text{-}08 \sum_i [(RM_{ivv}\tilde{Q}_{ivv} + RM_{ipv}\tilde{Q}_{ipv}) + (RI_{ivv}\tilde{Q}_{ivv} + RI_{ipv}\tilde{Q}_{ipv})] \quad (20)$$

Where:

Subscripts	=	vv, refers to vent releases from the building ventilation vents, including the Radwaste Facility Ventilation Vent and air ejectors; pv, refers to the vent releases from the process vent
D_r	=	the dose to the critical organ of the maximum exposed member of the public in mrem
RM_{ivv}, RM_{ipv}	=	the cow-milk pathway dose factor for ventilation vents or process vent release due to I^{131}, I^{133}, tritium, and from all particulate-form radionuclides with half-lives greater than 8 days, in mrem/yr per Curie/sec. Factors are listed in Attachment 18, Critical Organ and Inhalation Dose Factors For Surry
RI_{ivv}, RI_{ipv}	=	the inhalation pathway dose factor for ventilation vents or process vent release due to I^{131}, I^{133}, tritium, and from all particulate-form radionuclides with half-lives greater than 8 days, in mrem/yr per Curie/sec. Factors are listed in Attachment 18
$\tilde{Q}_{ivv}\tilde{Q}_{ipv}$	=	the release for ventilation vents or process vent of I^{131}, I^{133}, tritium, and from all particulate-form radionuclides with half-lives greater than 8 days in Curies
3.17 E-08	=	the inverse of the number of seconds in a year

6.3.4 **I-131, I-133, H-3, and Radionuclides In Particulate Form Effluent Dose Limit (continued)**

d. **North Anna Dose Calculations**

Gaseous releases, not through the process vent, are considered ground level and shall be included in the determination of $\tilde{Q}_{ivv}$. Historical data pertaining to the volumes and radioactive concentrations of gaseous effluents released in connection to specific Station functions, such as containment purges, shall be used in the estimates as appropriate.

1. The dose to the maximum exposed member of the public, attributable to gaseous effluents at and beyond the site boundary, that contain I^{131}, I^{133}, tritium, and particulate-form radionuclides with half-lives greater than 8 days, shall be determined by:

$$)_r = 3.17E\text{-}08 \sum_i [RM_{ivv}\tilde{Q}_{ivv} + RM_{ipv}\tilde{Q}_{ipv} \quad (21)$$

Where:

Subscripts = vv, refers to vent releases from the building ventilation vents; pv, refers to the vent releases from the process vent

D_r = the dose to the critical organ of the maximum exposed member of the public, in mrem

RM_{ivv}, RM_{ipv} =the cow-milk dose factor for ventilation vents or process vent release due to I^{131}, I^{133}, tritium, and from all particulate-form radionuclides with half-lives greater than 8 days, in mrem/yr per Curie/sec. Factors are listed in Attachment 19, Critical Organ Dose Factors for North Anna

$\tilde{Q}_{ivv}\tilde{Q}_{ipv}$ = the release for ventilation vents or process vent of I^{131}, I^{133}, tritium, and from all particulate-form radionuclides with half-lives greater than 8 days, in Curies

3.17 E-08 = the inverse of the number of seconds in a year

6.3.5 **Gaseous Radwaste Treatment**

Historical data pertaining to the volumes and radioactive concentrations of gaseous effluents released in connection with specific Station functions, such as containment purges, shall be used to calculate projected doses, as appropriate.

a. **Requirement**

1. The Gaseous Radwaste Treatment System and the Ventilation Exhaust Treatment System shall be used to reduce radioactive material in gaseous waste before its discharge, when projected gaseous effluent air doses due to gaseous effluent releases, from each unit to areas at and beyond the site boundary, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation, averaged over 31 days. (**North Anna**)

2. Appropriate portions of the Gaseous Radwaste Treatment System shall be used to reduce radioactive materials in gaseous waste before its discharge, when the projected gaseous effluent air doses due to gaseous effluent releases, from each unit to areas at and beyond the site boundary, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation, averaged over 31 days. (**Surry**)

3. The Ventilation Exhaust Treatment System shall be used to reduce radioactive materials in gaseous waste before its discharge, when the projected doses due to gaseous effluent releases, from each unit to areas at and beyond the site boundary, would exceed 0.3 mrem to the critical organ, averaged over 31 days.

4. Doses due to gaseous releases from the site shall be projected at least once per 31 days, based on the calculations in 6.3.5.c., and 6.3.5.d.

b. **Action**

If gaseous waste that exceeds the limits in 6.3.5.a. is discharged without treatment, prepare and submit to the NRC within 30 days, a special report in accordance with VPAP-2802, Notifications and Reports, that includes:

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

2. Actions taken to restore the inoperable equipment to operable status.

3. Summary description of actions taken to prevent recurrence.

6.3.5 **Gaseous Radwaste Treatment (continued)**

c. **Projected Gamma Dose**

1. Determine D_g, the 31-day gamma air dose for the previous 31-day period, per Equation (18).
2. Estimate R_g, the ratio of the estimated volume of gaseous effluent in the current 31-day period to the volume released during the previous 31-day period.
3. Estimate F_g, the ratio of the estimated noble gas effluent activity in the current 31-day period to the noble gas effluent activity during the previous 31-day period (μCi/ml).
4. Determine PD_g, the projected 31-day gamma air dose.

$$PD_g = D_g(R_g \times F_g) \tag{22}$$

d. **Projected Beta Dose**

1. Determine D_b, the 31-day beta air dose in the previous 31 days, per Equation (19).
2. Estimate R_g and F_g as in 6.3.5.c.2. and 6.3.5.c.3.
3. Determine PD_b, the projected 31-day beta air dose.

$$PD_b = D_b(R_g \times F_g) \tag{23}$$

e. **Projected Maximum Exposed Member of the Public Dose**

1. Determine D_{max}, the 31-day maximum exposed member of the public dose in the previous 31-day period, per Equation (20) or Equation (21), where $D_r = D_{max}$.
2. Estimate F_i, the ratio of the estimated activity from I^{131}, I^{133}, radioactive materials in particulate form with half-lives greater than 8 days, and tritium in the current 31-day period to the activity of I^{131}, I^{133}, radioactive materials in particulate form with half-lives greater than 8 days, and tritium in the previous 31-day period (μCi/ml).

3. Determine PD_{max}, the projected 31-day maximum exposed member of the public dose.

$$PD_{max} = D_{max}(R_g \times F_i) \tag{24}$$

6.4 Radioactive Liquid and Gaseous Release Permits

RP shall maintain procedures for Liquid and Gaseous Release Permits to ensure effluent dose limits are not exceeded when making releases.

6.4.1 Liquid Waste Batch Release Permits

Operations shall obtain RP authorization before initiating batch releases of radioactive liquids. Examples of batch releases include:

a. **Surry Batch Releases**

Release of contents from the following tanks/sumps other than transfers to the Surry Radwaste Facility shall have a Liquid Waste Batch Release Permit before the discharge:

- Boron Recovery Test Tank (BRTT)
- Low Level Waste Drain Tank (LLWDT)
- High Level Waste Drain Tank (HLWDT)
- Liquid Waste Test Tank (LWTT)
- Contaminated Drain Tank (CDT)
- Laundry Drain Surge Tank (LDST)
- Turbine Building Sumps when RP determines that source activity requires placing pumps in manual mode
- Condensate Polishing Building Sumps when RP determines the presence of contamination from primary-to-secondary leakage

6.4.1 **Liquid Waste Batch Release Permits (continued)**

b. **North Anna Batch Releases**

NOTE: If the clarifier is in service, releases from tanks processed through the clarifier are considered continuous releases.

A Batch Release Permit is required for a release from any tanks/sumps which contain (or potentially contain) radioactive liquid. Tanks/sumps include:

- BRTT
- LLWDT
- HLWDT
- Turbine Building Sumps when secondary coolant activity exceeds 1.0 E-5 μCi/ml
- CDT

6.4.2 **Continuous Release Permit**

Operations shall obtain RP authorization before initiating continuous releases of radioactive liquids.

a. **Surry Continuous Releases**

A Continuous release permit is required at Surry for:

- Steam generator blowdown
- Component Cooling Water (CCW) heat exchanger to service water leakage, if applicable
- Turbine Building sumps and/or subsurface drains if source activity concentrations are sufficiently low to allow continuous release

b. **North Anna Continuous Releases**

A Continuous Release Permit is required at North Anna for:

- Clarifier, unless being bypassed
- Steam generator blowdown when clarifier is bypassed
- Containment mat sumps and service water reservoir when clarifier is bypassed

6.4.3 **Waste Gas Decay Tank (WGDT) Release Permit**

Operations shall obtain RP authorization before initiating WGDT releases.

6.4.4 **Reactor Containment Release Permits**

Operations shall obtain authorization from RP before initiating containment purges or containment hogging. Reactor Containment Release Permits shall be valid from start of purge/hog until:

- Routine termination
- Terminated for cause by RP
- Receipt of Radiation Monitoring System (RMS) Containment Gas Monitor high alarm

6.4.5 **Miscellaneous Gaseous Release Permit**

Operations shall obtain RP authorization before initiating releases of noble gases that may not be accounted for by routine sampling, or any planned release not being routed through the Process Vent or Ventilation Vents (e.g., steam driven auxiliary feedwater pump testing if primary to secondary leakage exists).

6.4.6 **Radioactive Liquid and Gaseous Release Controls**

a. Operations shall notify RP of pending releases and request RP to initiate the appropriate release permit. Operations shall provide the necessary information to complete the required release permit.

b. A representative sample shall be obtained of the source to be released.

   1. Operations shall provide RP with liquid samples and sample information (e.g., time of sample) for samples obtained outside the Primary Sample Room, except Clarifier Proportional Tank and Clarifier Grab Samples at North Anna.

   2. Chemistry shall provide RP with liquid samples and sample information for samples obtained from inside the Primary Sample Room.

   3. RP shall obtain gaseous samples.

c. RP shall perform required sample analyses.

d. RP shall calculate and record the following information on a release permit:

   - Maximum authorized release rate
   - Maximum authorized release rate in percentage of limits specified by the ODCM
   - Applicable conditions or controls pertaining to the release

6.4.6 **Radioactive Liquid and Gaseous Release Controls (continued)**

e. RP shall notify the Shift Supervisor if it is determined that a release may not be within the effluent dose limits.

f. Upon receipt of a release permit from RP, Operations shall:

1. Verify the correct source is authorized for release.
2. Note maximum authorized release rate.
3. Note percent of Technical Specification limits the release represents.
4. Note and ensure compliance with any indicated controls or conditions applicable to the release.

g. When commencing release, Operations shall provide RP with required information. As appropriate, required information shall include:

- Date and time release was started
- Starting tank/sump level
- Beginning pressure
- Release flow rate
- Dilution water flow rate

h. Upon terminating the release, Operations shall return the permit to RP and provide information necessary for completion of permit. As appropriate, required information shall include:

- Date and time release was stopped
- Tank/sump ending level
- Release flow rate just prior to termination
- Ending pressure
- Volume released

6.5 Total Dose Limit to Public From Uranium Fuel Cycle Sources

6.5.1 **Requirement**

The annual (calendar year) dose or dose commitment to a real individual due to releases of radioactivity and radiation from uranium fuel cycle sources shall not exceed 25 mrem to the total body or the critical organ (except the thyroid, which shall not exceed 75 mrem).

6.5.2 **Action**

a. If the calculated doses from release of radioactive materials in liquid or gaseous effluents exceed twice the limits in 6.2.3.a., 6.3.3.a., or 6.3.4.a., calculate (including direct radiation contribution from the units and from outside storage tanks) whether limits in 6.5.1 have been exceeded.

b. If the limits in 6.5.1 have been exceeded, prepare and submit to the NRC within 30 days, a special report in accordance with VPAP-2802, Notifications and Reports, that defines the corrective action to be taken to reduce subsequent releases and to prevent recurrence, and includes a schedule for achieving conformance with the limits. Special reports, as defined in 10 CFR 20.2203(a)(4), shall include:

   1. An analysis that estimates the radiation exposure (dose) to a real individual from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the releases covered by the report.

   2. A description of the levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations.

   3. If the estimated dose exceeds the limits in 6.5.1, and if the release condition that violates 40 CFR 190 has not already been corrected, the special report shall include a request for a variance in accordance with the provisions of 40 CFR 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

6.6 Radiological Environmental Monitoring

6.6.1 Monitoring Program

a. **Requirement**

1. The Radiological Environmental Monitoring Program shall be conducted as specified in Attachments 20 or 21, Radiological Environmental Monitoring Program.

2. Samples shall be collected from specific locations specified in Attachment 22 or 23, Environmental Sample Locations. **[Commitment 3.2.2]**

3. Samples shall be analyzed in accordance with:
   - Attachment 20 or 21 requirements
   - Detection capabilities required by Attachment 24 or 25, Detection Capabilities for Environmental Sample Analysis
   - Guidance of the Radiological Assessment Branch Technical Position on Environmental Monitoring dated November, 1979, Revision No. 1

b. **Action**

1. If the Radiological Environmental Monitoring Program is not being conducted as required in 6.6.1.a., report the situation in accordance with VPAP-2802, Notifications and Reports, by preparing and submitting to the NRC, in the Annual Radiological Environmental Operating Report required by Technical Specification (Surry Technical Specification 6.6.B.2 and North Anna Technical Specification 6.9.1.8), a description of the reasons for not conducting the program as required, and the plan for precluding recurrence.

2. If, when averaged over any calendar quarter, radioactivity exceeds the reporting levels of Attachment 26 or 27, Reporting Levels for Radioactivity Concentrations in Environmental Samples, prepare and submit to the NRC within 30 days, a special report in accordance with VPAP-2802, Notifications and Reports, that:
   - Identifies the causes for exceeding the limits, 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 6.2.3, 6.3.3, and 6.3.4

When more than one of the radionuclides listed in Attachment 26 or 27 are detected in the sampling medium, the report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0 \qquad (25)$$

3. When radionuclides other than those listed in Attachments 26 and 27 are detected and are the result of plant effluents, the 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 6.2.3, 6.3.3, and 6.3.4. The report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, report and describe the condition in the Annual Radiological Environmental Operating Report in accordance with VPAP-2802, Notifications and Reports.

4. If milk or fresh leafy vegetable samples are unavailable from one or more of the sample locations required by Attachment 20 or 21, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Identify the cause of the unavailability of samples and identify the new locations for obtaining replacement samples in the next Annual Radioactive Effluent Release Report in accordance with VPAP-2802, Notifications and Reports. Include in the report a revised figure and table for the ODCM to reflect the new locations.

6.6.2 **Land Use Census**

a. **Requirement**

A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the following:

- Nearest milk animal
- Nearest residence
- Nearest garden greater than 50 m^2 (500 ft^2) that produces broad leaf vegetation

1. The land use census shall be conducted during the growing season, at least once per 12 months, using methods that will provide the best results (e.g., door-to-door survey, aerial survey, local agriculture authorities). Land use census results shall be included in the Annual Radiological Environmental Operating Report in accordance with VPAP-2802, Notifications and Reports.
2. In lieu of the garden census, road 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 ground deposition (D/Qs). Specifications for broad leaf vegetation sampling in Attachment 20 or 21 shall be followed, including analysis of control samples.

b. **Action**

1. If a land use census identifies locations that yield a calculated dose or dose commitment greater than the values currently being calculated in 6.3.4.a.2, identify the new locations in the next Annual Radioactive Effluent Release Report in accordance with VPAP-2802, Notifications and Reports.
2. If a land use census identifies locations that yield a calculated dose or dose commitment (via the same exposure pathway) 20 percent (**Surry**) or 25 percent (**North Anna**) greater than at a location from which samples are currently being obtained, add the new locations to the Radiological Environmental Monitoring Program within 30 days. Sampling locations, excluding the control station location, that have the lowest calculated dose or dose commitments (via the same exposure pathway) may be deleted from the monitoring program. Identify new locations in the next Annual Radioactive Effluent Release Report and include in the report revised figures and tables reflecting the new locations in accordance with VPAP-2802, Notifications and Reports. **[Commitment 3.2.4]**

6.6.3 Interlaboratory Comparison Program

a. **Requirement**

Radioactive materials (which contain nuclides produced at the Stations), supplied as part of an Interlaboratory Comparison Program that has been approved by the NRC, shall be analyzed.

b. **Action**

1. Analyses shall be performed as part of the Environmental Protection Agency's Environmental Radioactivity Laboratory Intercomparison Studies (Cross Check) Program and include:

Program	Cross-Check of
Milk	I^{131}, Gamma, K, Sr^{89} and Sr^{90}
Water	Gross Beta, Gamma, I^{131}, H^3 (Tritium), Sr^{89} and Sr^{90} (blind—any combinations of above radionuclides)
Air Filter	Gross Beta, Gamma, Sr^{90}

2. If analyses are not performed as required by 6.6.3.b., report in the Annual Radiological Environmental Operating Report in accordance with VPAP-2802, Notifications and Reports, the corrective actions taken to prevent recurrence.

6.6.3 **Interlaboratory Comparison Program (continued)**

c. **Methodology and Results**

1. Methodology and results of the cross-check program shall be maintained in the contractor-supplied Nuclear Reactor Environmental Radiation Monitoring Quality Control Manual, IWL-0032-361.

2. Results shall be reported in the Annual Radiological Environmental Monitoring Report in accordance with VPAP-2802, Notifications and Reports.

6.7 Reporting Requirements

6.7.1 Annual Radiological Environmental Operating Report

Routine Radiological Environmental Operating Reports covering the operation of the units during the previous calendar year shall be submitted prior to May 1 of each year. A single submittal may be made for the Station. Radiological Environmental Operating Reports shall include:

a. Summaries, interpretations, and analysis of trends of results of radiological environmental surveillance activities for the report period, including:
   - A comparison (as appropriate) with preoperational studies, operational controls, and previous environmental surveillance reports
   - An assessment of the observed impacts of the plant operation on the environment
   - Results of land use census per 6.6.2

b. Results of analysis of radiological environmental samples and of environmental radiation measurements taken per 6.6.1, Monitoring Program. Results shall be summarized and tabulated in the format of the table in the Radiological Assessment Branch Technical Position on Environmental Monitoring.

   1. If some individual results are not available for inclusion with the report, the report shall be submitted, noting and explaining reasons for missing results.

   2. Missing data shall be submitted in a supplementary report as soon as possible.

c. A summary description of the radiological environmental monitoring program.

d. At least two legible maps covering sampling locations, keyed to a table giving distances and directions from the centerline of one reactor. One map shall cover stations near the site boundary; a second shall include more distant stations.

e. Results of Station participation in the Interlaboratory Comparison Program, per 6.6.3.

f. Discussion of deviations from the Station's environmental sampling schedule per Attachment 20 or 21.

g. Discussion of analyses in which the lower limit of detection (LLD) required by Attachment 24 or 25 was not achievable.

6.7.2 **Annual Radioactive Effluent Release Report**

a. **Requirement**

Radioactive Effluent Release Reports covering operation of the units during the previous 12 months of operation shall be submitted before May 1 of each year. A single submittal may be made for the Station and should combine those sections that are common to both units. Radioactive Effluent Release Reports shall include:

1. A summary of quantities of radioactive liquid and gaseous effluents and solid waste released. Data shall be summarized on a quarterly basis following the format of Regulatory Guide 1.21, Appendix B.

2. An assessment of radiation doses to the maximum exposed members of the public due to the radioactive liquid and gaseous effluents released from the Station during the previous calendar year. This assessment shall be in accordance with 6.7.2.b.

3. A list and description of unplanned releases from the site to unrestricted areas, during the reporting period, which meet the following criteria:
   - Unplanned releases that exceeded the limits in 6.2.1 and 6.3.1
   - Unplanned releases which require a Deviation Report and involve the discharge of contents of the wrong Waste Gas Decay Tank or the wrong liquid radwaste release tank
   - Unplanned releases from large leaks due to unexpected valve or pipe failures that result in a quantity of release such that a 10 CFR 50.72, Immediate Notification Requirements for Operating Nuclear Power Reactors or 10 CFR 50.73, Licensee Event Report System, report is required
   - Unplanned releases as determined by Radiation Protection Supervision, which may or may not require a Deviation Report

6.7.2 **Annual Radioactive Effluent Release Report (continued)**

4. Major changes to radioactive liquid, gaseous, and solid waste treatment systems during the reporting period.

5. Changes to VPAP-2103, Offsite Dose Calculation Manual (see 6.7.4).

6. A listing of new locations for dose calculations or environmental monitoring identified by the land use census (see 6.6.2).

b. **Dose Assessment**

1. Radiation dose to individuals due to radioactive liquid and gaseous effluents from the Station during the previous calendar year shall either be calculated in accordance with this procedure or in accordance with Regulatory Guide 1.109. Population doses shall not be included in dose assessments.

2. The dose to the maximum exposed member of the public due to radioactive liquid and gaseous effluents from the Station shall be incorporated with the dose assessment performed above. If the dose to the maximum exposed member of the public exceeds twice the limits of 6.2.3.a.1, 6.2.3.a.2, 6.3.3.a.1, or 6.3.4.a.1, the dose assessment shall include the contribution from direct radiation.

NOTE: NUREG-0543 states: "There is reasonable assurance that sites with up to four operating reactors that have releases within Appendix I design objective values are also in conformance with the EPA Uranium Fuel Cycle Standard, 40 CFR Part 190."

3. Meteorological conditions during the previous calendar year or historical annual average atmospheric dispersion conditions shall be used to determine gaseous pathway doses.

6.7.3 **Annual Meteorological Data**

a. Meteorological data collected during the previous year shall be in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.

b. Meteorological data shall be retained in a file on site and shall be made available to NRC upon request.

6.7.4 **Changes to the ODCM**

Changes to the ODCM shall be:

a. Reviewed and approved by SNSOC and the Station Manager before implementation.

b. Documented. Records of reviews shall be retained as Station records. Documentation shall include:

   1. Sufficient information to support changes, together with appropriate analyses or evaluations justifying changes.

   2. A determination that a change will not adversely impact the accuracy or reliability of effluent doses or setpoint calculations, and will maintain the level of radioactive effluent control required by:

      - 10 CFR 20 Subpart D
      - 40 CFR 190
      - 10 CFR 50.36a
      - 10 CFR 50, Appendix I

c. Submitted to NRC in the form of a complete, legible copy of the entire ODCM as a part of, or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

d. Submitted to the Management Safety Review Committee (MSRC) Coordinator. **[Commitment 3.2.1]**

e. Submitted to NRC in accordance with VPAP-2802, Notifications and Reports.

7.0 RECORDS

7.1 The following individual and packaged documents and copies of any related correspondence completed as a result of the performance or implementation of this procedure are records. They shall be submitted to Records Management in accordance with VPAP-1701, Records Management. Prior to transmittal to Records Management, the sender shall assure that:

- Each record is packaged when applicable,
- QA program requirements have been fulfilled for Quality Assurance records,
- Each record is legible, completely filled out, and adequately identifiable to the item or activity involved,
- Each record is stamped, initialed, signed, or otherwise authenticated and dated, as required by this procedure.

7.1.1 **Individual Records**

- None

7.1.2 Record Packages

- Records of changes to the ODCM in accordance with 6.7.4
- Records of meteorological data in accordance with 6.7.3
- Records of sampling and analyses
- Records of radioactive materials and other effluents released to the environment
- Records of preventive maintenance, surveillances, and calibrations

7.2 The following documents completed as a result of the implementation of this procedure are not records and are not required to be transmitted to Records Management.

None

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ATTACHMENT 1

(Page 1 of 1)

Surry Radioactive Liquid Effluent Monitoring Instrumentation

Instrument	Minimum Operable Channels	Action
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE (a) Radwaste Facility Liquid Effluent Line RM-RRM-131	1	1
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE (a) Circulating Water Discharge Line Unit 1: 1-SW-RM-120 Unit 2: 2-SW-RM-220	2	2
(b) Component Cooling Service Water Effluent Line 1-SW-RM-107A 1-SW-RM-107B 1-SW-RM-107C 1-SW-RM-107D	4	2
3. FLOW RATE MEASUREMENT DEVICES Radwaste Facility Liquid Effluent Line Instrument Loop RLW-153	1	3

ACTION 1: If the number of operable channels is less than required, effluent releases shall be suspended.

ACTION 2: If the number of operable channels is less than required, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for principal gamma emitters, as defined in Attachment 8, Surry Radioactive Liquid Waste Sampling and Analysis Program.

ACTION 3: If the number of operable channels is less than required, effluent releases via this pathway shall be suspended.

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ATTACHMENT 2

(Page 1 of 2)

North Anna Radioactive Liquid Effluent Monitoring Instrumentation

Instrument	Minimum Operable Channels	Action
1. **Liquid Radwaste Effluent**		
(a) 1-RM-LW-111, Liquid Radwaste Effluent Monitor	1	1
(b) 1-LW-FT-104, Liquid Radwaste Effluent Total Flow Measuring Device	1	2
(c) 1-LW-SOV-121, Clarifier Effluent Line Continuous Composite Sampler and Sampler Flow Monitor	1	1
(d) 1-LW-TK-20, Liquid Waste Effluent Sample Vessel	1	1
(e) 1-LW-1130, Liquid Waste Effluent Proportional Sample Valve	1	1
(f) 1-RM-SW-108, Service Water Effluent Monitor	1	1
(g) 1-RM-SW-130, Unit 1 Circulating Water System Effluent Line Monitor	1	4
(h) 2-RM-SW-230, Unit 2 Circulating Water System Effluent Line Monitor	1	4
2. **Tank Level Indicating Devices** (Note 1)		
(a) Refueling Water Storage Tanks		
Unit 1 1-QS-LT-100A 1-QS-LT-100B 1-QS-LT-100C 1-QS-LT-100D	1	3
Unit 2 2-QS-LT-200A 2-QS-LT-200B 2-QS-LT-200C 2-QS-LT-200D	1	3
(b) Casing Cooling Storage Tanks		
Unit 1 1-RS-LT-103A 1-RS-LT-103B	1	3
Unit 2 2-RS-LT-203A 2-RS-LT-203B	1	3
(c) PG Water Storage Tanks (Note 2)		
1-BR-LT-116A (1-PG-TK-1A)	1	3
1-BR-LT-116B (1-PG-TK-1B)	1	3
(d) Boron Recovery Test Tanks (Note 2)		
1-BR-LT-112A (1-BR-TK-2A)	1	3
1-BR-LT-112B (1-BR-TK-2B)	1	3

ATTACHMENT 2

(Page 2 of 2)

North Anna Radioactive Liquid Effluent Monitoring Instrumentation

ACTION 1: If the number of operable channels is less than required, effluent releases via this pathway may continue if, at least once within 12 hours, grab samples are collected and analyzed for gross radioactivity (beta and gamma) at an LLD of at least 1×10^{-7} μCi/g or an isotopic radioactivity at an LLD of at least 5×10^{-7} μCi/g.

ACTION 2: If the number of operable channels is less than required, effluent releases via this pathway may continue if the flow rate is estimated at least once per 4 hours during actual releases. Design capacity performance curves generated in situ may be used to estimate flow.

ACTION 3: If the number of operable channels is less than required, liquid additions to this tank may continue if the tank liquid level is estimated during all liquid additions to the tank.

ACTION 4: If the number of operable channels is less than required, make repairs as soon as possible. Grab samples cannot be obtained via this pathway.

NOTE 1: Tanks included in this requirement are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents, and do not have overflows and surrounding area drains connected to the liquid radwaste treatment system.

NOTE 2: This is a shared system between Unit 1 and Unit 2.

ATTACHMENT 3

(Page 1 of 1)

Surry Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements

Channel Description	Channel Check	Source Check	Channel Calibration	Channel Functional Test
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
(a) Radwaste Facility Liquid Effluent Line RM-RRM-131	D	P	R	Q
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
(a) Circulating Water Discharge Line Unit 1: 1-SW-RM-120 Unit 2: 2-SW-RM-220	D	M	R	Q
(b) Component Cooling Service Water Effluent Line 1-SW-RM-107A 1-SW-RM-107B 1-SW-RM-107C 1-SW-RM-107D	D	M	R	Q
3. FLOW RATE MEASUREMENT DEVICES				
Radwaste Facility Liquid Effluent Line Instrument Loop RLW-153	DR	N/A	R	N/A

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ATTACHMENT 4

(Page 1 of 2)

North Anna Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements

Channel Description	Channel Check	Source Check	Channel Calibration	Channel Functional Test
1. **Liquid Radwaste Effluent**				
(a) 1-RM-LW-111, Liquid Radwaste Effluent Monitor	D	D	R	Q (NOTE 1)
(b) 1-LW-FT-104, Liquid Radwaste Effluent Total Flow Measuring Device	D (NOTE 3)	N/A	R	Q
(c) 1-LW-SOV-121, Clarifier Effluent Line Continuous Composite Sampler and Sampler Flow Monitor	N/A	N/A	R	N/A
(d) 1-LW-TK-20, Liquid Waste Effluent Sample Vessel	D (NOTE 9)	N/A	N/A	N/A
(e) 1-LW-1130, Liquid Waste Effluent Proportional Sample Valve	D (NOTE 9)	N/A	N/A	N/A
(f) 1-RM-SW-108, Service Water System Effluent Monitor	D	M	R	Q (NOTE 2)
(g) 1-RM-SW-130, Unit 1 Circulating Water System Effluent Line Monitor	D	M	R	Q (NOTE 2)
(h) 2-RM-SW-230, Unit 2 Circulating Water System Effluent Line Monitor	D	M	R	Q (NOTE 2)
2. **Tank Level Indicating Device** (NOTE 6)				
(a) Refueling Water Storage Tanks				
Unit 1 1-QS-LT-100A 1-QS-LT-100B 1-QS-LT-100C 1-QS-LT-100D	D (NOTE 4)	N/A	R	Q (NOTE 7)
Unit 2 2-QS-LT-200A 2-QS-LT-200B 2-QS-LT-200C 2-QS-LT-200D	D (NOTE 4)	N/A	R	Q (NOTE 7)
(b) Casing Cooling Storage Tanks				
Unit 1 1-RS-LT-103A 1-RS-LT-103B	D (NOTE 4)	N/A	R	Q (NOTE 7)
Unit 2 2-RS-LT-203A 2-RS-LT-203B	D (NOTE 4)	N/A	R	Q (NOTE 7)
(c) PG Water Storage Tanks (NOTE 5)				
1-BR-LT-116A (1-PG-TK-1A)	D (NOTE 4)	N/A	R	Q (NOTE 8)
1-BR-LT-116B (1-PG-TK-1B)	D (NOTE 4)	N/A	R	Q (NOTE 8)
(d) Boron Recovery Test Tanks (NOTE 5)				
1-BR-LT-112A (1-BR-TK-2A)	D (NOTE 4)	N/A	R	Q (NOTE 8)
1-BR-LT-112B (1-BR-TK-2B)	D (NOTE 4)	N/A	R	Q (NOTE 8)

ATTACHMENT 4

(Page 2 of 2)

North Anna Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements

NOTE 1: The Channel Functional Test shall demonstrate:

a. Automatic isolation of this pathway and Control Room alarm annunciation occur if the instrument indicates measured levels above alarm/trip setpoint.

b. Alarm annunciation occurs if the instrument controls are not set in "operate" mode.

NOTE 2: The Channel Functional Test shall demonstrate that Control Room alarm annunciation occurs if any of the following conditions exists:

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

b. Instrument controls not set in "operate" mode.

NOTE 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 on days on which continuous, periodic, or batch releases are made.

NOTE 4: During liquid additions to the tank, verify indication of level change.

NOTE 5: This is a shared system between Unit 1 and Unit 2.

NOTE 6: Tanks included in this requirement are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and do not have overflows and surrounding area drains connected to the liquid radwaste treatment system.

NOTE 7: The Channel Functional Test shall demonstrate that automatic isolation of this pathway and Control Room alarm annunciation occur if instrument indicates measured levels outside the alarm/trip setpoint. Demonstration of automatic isolation may consist of verifying the appropriate signal is generated. Valves need not be operated for this test.

NOTE 8: The Channel Functional Test shall demonstrate that Control Room alarm annunciation occurs if the instrument indicates measured levels are outside alarm setpoint.

NOTE 9: Channel Check shall consist of verifying that proportional flow exceeds 0.5 mls/gallon.

ATTACHMENT 5

(Page 1 of 1)

Liquid Ingestion Pathway Dose Factors for Surry Station Units 1 and 2

Radionuclide	Total Body A_i $\frac{mrem/hr}{\mu Ci/ml}$	Thyroid A_i $\frac{mrem/hr}{\mu Ci/ml}$	GI-LLI $\frac{mrem/hr}{\mu Ci/ml}$
H-3	2.82E-01	2.82E-01	2.82E-01
Na-24	4.57E-01	4.57E-01	4.57E-01
Cr-51	5.58E+00	3.34E-01	1.40E+03
Mn-54	1.35E+03	-	2.16E+04
Fe-55	8.23E+03	-	2.03E+04
Fe-59	7.27E+04	-	6.32E+05
Co-58	1.35E+03	-	1.22E+04
Co-60	3.82E+03	-	3.25E+04
Zn-65	2.32E+05	-	3.23E+05
Rb-86	2.91E+02	-	1.23E+02
Sr-89	1.43E+02	-	8.00E+02
Sr-90	3.01E+04	-	3.55E+03
Y-91	2.37E+00	-	4.89E+04
Zr-95	3.46E+00	-	1.62E+04
Zr-97	8.13E-02	-	5.51E+04
Nb-95	1.34E+02	-	1.51E+06
Mo-99	2.43E+01	-	2.96E+02
Ru-103	4.60E+01	-	1.25E+04
Ru-106	2.01E+02	-	1.03E+05
Ag-110m	8.60E+02	-	5.97E+05
Sb-124	1.09E+02	6.70E-01	7.84E+03
Sb-125	4.20E+01	1.79E-01	1.94E+03
Te-125m	2.91E+01	6.52E+01	8.66E+02
Te-127m	6.68E+01	1.40E+02	1.84E+03
Te-129m	1.47E+02	3.20E+02	4.69E+03
Te-131m	5.71E+01	1.08E+02	6.80E+03
Te-132	1.24E+02	1.46E+02	6.24E+03
I-131	1.79E+02	1.02E+05	8.23E+01
I-132	9.96E+00	9.96E+02	5.35E+00
I-133	3.95E+01	1.90E+04	1.16E+02
I-134	5.40E+00	2.62E+02	1.32E-02
I-135	2.24E+01	4.01E+03	6.87E+01
Cs-134	1.33E+04	-	2.85E+02
Cs-136	2.04E+03	-	3.21E+02
Cs-137	7.85E+03	-	2.32E+02
Cs-138	5.94E+00	-	5.12E-05
Ba-140	1.08E+02	-	3.38E+03
La-140	2.10E-01	-	5.83E+04
Ce-141	2.63E-01	-	8.86E+03
Ce-143	4.94E-02	-	1.67E+04
Ce-144	9.59E+00	-	6.04E+04
Np-239	1.91E-03	-	7.11E+02

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ATTACHMENT 6

(Page 1 of 5)

North Anna Liquid Ingestion Pathway Dose Factor CalculationUnits 1 and 2

1.0 Equation (6)

$$D = t\,F\sum_{i} f_i C_i A_i \qquad (6\text{-}1)$$

where:

D = cumulative dose commitment to the total body or critical organ, from the liquid effluents for the period t, in mrem

t = period for which C_i and F are averaged for all liquid releases, in hours

F = the near field average dilution factor for C_i during any liquid effluent release. Defined as the ratio of the average undiluted liquid waste flow during release to the average flow from the Station discharge structure to unrestricted areas

f_i = the individual dilution multiplication factor to account for increases in concentration of long-lived nuclides due to recirculation, listed on page 5 of this attachment. "f_i" is the ratio of the total dilution flow over the effective dilution flow

C_i = average concentration of radionuclide i, in undiluted liquid effluent during the period t, from any liquid releases, in µCi/ml

A_i = the site-related ingestion dose commitment factor to the total body or critical organ of an adult for each identified principal gamma and beta emitter listed on page 5 of this attachment, in mrem/hr per µCi/ml

$$A_i = 1.14\text{ E+05 } (730/D_W + 21\ BF_i/D_a)\ DF_i \qquad (6\text{-}2)$$

where:

1.14 E+05 = (1 E+06 pCi/µCi x 1 E+03 ml/kg)/8760 hr/yr, units conversion factor

730 = adult water consumption, kg/yr, from NUREG-0133

ATTACHMENT 6

(Page 2 of 5)

North Anna Liquid Ingestion Pathway Dose Factor CalculationUnits 1 and 2

D_w = dilution factor from the near field area within one-quarter mile of the release point to the potable water intake for the adult water consumption. D_w includes the dilution contributions from the Lake Anna Dam to Doswell (0.73), the WHTF (C_L/C_C), and Lake Anna (C_R/C_L). The potable water mixing ratio is calculated as:

$$1 / (C_L/C_C)(C_R/C_L) \times 0.73 = C_C/(C_R \times 0.73) \qquad (6\text{-}3)$$

where C_C, C_L and C_R are the respective concentrations for the considered nuclide in the discharge channel, WHTF (Lagoon) and the Lake. Calculation is per expressions 11.2-5, 11.2-6, and 11.2-8 of the North Anna UFSAR

21 = adult fish consumption rate, kg/yr, from NUREG 0133

BF_i = the bioaccumulation factor for nuclide i, in fish, pCi/kg per pCi/l, from Table A-1 of Regulatory Guide 1.109, Rev. 1

D_a = dilution factor for the fish pathway, calculated as $1/(C_L/C_C)$ where C_L and C_c are the concentrations for the considered nuclide in the discharge channel and the WHTF (Lagoon). Calculation is per Expressions 11.2-5, and 11.2-6 of North Anna's UFSAR

DF_i = the critical organ dose conversion factor for nuclide i, for adults, in mrem/pCi, from Table E-11 of Regulatory Guide 1.109, Rev. 1

ATTACHMENT 6

(Page 3 of 5)

North Anna Liquid Ingestion Pathway Dose Factor CalculationUnits 1 and 2

2.0 Equation (9)

Equation (6) is simplified for actual dose calculations by introducing:

$$F = \frac{\text{WASTE FLOW}}{\text{CIRC. (WATER) FLOW + WASTE FLOW}} \approx \frac{\text{WASTE FLOW}}{\text{CIRC. FLOW}} \quad (6\text{-}4)$$

and

$$f_i = \frac{\text{CIRC. FLOW}}{\text{EFFECTIVE DIL. FLOW}_i} \quad (6\text{-}5)$$

Effective dilution flow rates for individual nuclides "i" are listed on Attachment 7, North Anna Liquid Pathway Dose Commitment Factors for Adults. Then the total released activity (Q_i) for the considered period and the ith nuclide is written as:

$$Q_i = t \times C_i \times \text{WASTE FLOW} \quad (6\text{-}6)$$

and Equation (6) reduces to:

$$D = \sum_i Q_i \frac{A_i}{\text{EFF. DIL. FLOW}_i} \quad (6\text{-}7)$$

For the long-lived, dose controlling nuclides, the effective dilution flow is essentially the over (dam) flow rate out of the Lake Anna system (i.e., the liquid pathway dose is practically independent from the circulating water flow rate. However, to accurately assess long range average effects of reduced circulating water flow rates during outages or periods of low lake water temperatures, calculations are based on an average of 7 out of 8 circulating water pumps running at 218,000 gpm = 485.6 cft/sec per pump.

ATTACHMENT 6

(Page 4 of 5)

North Anna Liquid Ingestion Pathway Dose Factor CalculationUnits 1 and 2

NOTE: The 218,000 gpm flow rate per Circulating Water pump is based on Reference 3.1.21. The choice of seven Circulating Water pumps is considered realistic. Compared to this, the NAPS UFSAR, Chapter 11.2 (Reference 3.1.18), contains an extremely conservative consideration based on the minimum flow in accordance with Reference 3.1.21 with only two Circulating Water pumps operating. Even at such a low flow rate, which cannot be sustained during power generation, liquid pathway effluent dose factors increase only slightly for the dose controlling nuclides (i.e., Cs^{134} 19 percent, Cs^{137} 15 percent).

By defining $B_i = A_i$ / EFF. DIL. FLOW$_i$, the dose calculation is reduced to a two factor formula:

$$D = \sum_i Q_i \times B_i \qquad (6\text{-}8)$$

Values for B_i (mrem/Ci) and EFF. DIL. FLOW$_i$ are listed in Attachment 7.

ATTACHMENT 6

(Page 5 of 5)

North Anna Liquid Ingestion Pathway Dose Factor CalculationUnits 1 and 2

Radionuclide	Individual Dilution Multiplication Factor (f_i)	Total Body A_i $\frac{mrem/hr}{\mu Ci/ml}$	Liver A_i $\frac{mrem/hr}{\mu Ci/ml}$
H-3	14.9	6.18E+00	6.18E+00
Na-24	1.0	3.71E+01	3.71E+01
Cr-51	1.7	1.10E+00	-
Mn-54	7.0	8.62E+02	4.52E+03
Fe-55	11.3	1.30E+02	5.56E+02
Fe-59	2.2	9.47E+02	2.47E+03
Co-58	2.8	2.49E+02	1.11E+02
Co-60	13.3	8.27E+02	3.75E+02
Zn-65	6.1	3.28E+04	7.25E+04
Rb-86	1.5	3.53E+04	7.59E+04
Sr-89	2.3	8.[illegible]E+02	-
Sr-90	15.8	2.39E+05	-
Y-91	2.5	3.42E-01	-
Zr-95	2.7	2.98E-01	4.41E-01
Zr-97	1.0	1.50E-04	3.27E-04
Nb-95	1.9	1.13E+02	2.10E+02
Mo-99	1.0	7.48E+00	3.93E+01
Ru-103	2.0	4.10E+00	-
Ru-106	7.6	2.65E+01	-
Ag-110m	6.2	4.94E+00	8.32E+00
Sb-124	2.6	4.37E+01	2.08E+00
Sb-125	11.4	2.46E+01	1.16E+00
Te-125m	2.5	3.23E+02	8.73E+02
Te-127m	3.7	7.82E+02	2.29E+03
Te-129m	1.9	1.52E+03	3.58E+03
Te-131m	1.0	1.12E+02	1.35E+02
Te-132	1.0	5.04E+02	5.37E+02
I-131	1.2	9.66E+01	1.69E+02
I-132	1.0	1.03E-01	2.95E-01
I-133	1.0	3.47E+00	1.14E+01
I-134	1.0	2.15E-02	6.00E-02
I-135	1.0	6.58E-01	1.78E+00
Cs-134	10.3	5.80E+05	7.09E+05
Cs-136	1.3	6.01E+04	8.35E+04
Cs-137	15.8	3.45E+05	5.26E+05
Cs-138	1.0	9.18E-01	1.85E+00
Ba-140	1.3	2.65E+01	5.08E-01
La-140	1.0	4.47E-03	1.69E-02
Ce-141	1.8	2.14E-02	1.89E-01
Ce-143	1.0	1.35E-04	1.22E+00
Ce-144	6.6	1.41E+00	1.10E+01
Np-239	1.0	5.13E-04	9.31E-04

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ATTACHMENT 7

(Page 1 of 1)

North Anna Liquid Pathway Dose Commitment Factors for Adults

$B_i = A_i F_i/\text{CIRC FLOW} = (A_i/\text{Effective Dilution Flow}_i) \times 9.81\text{E-3 hr ft3 } \mu\text{Ci / sec ml Ci}$

Radionuclide	Effective Dilution Flow (cft/sec)	Total Body B_i (mrem/Ci)	Liver B_i (mrem/Ci)
H-3	2.28E+02	2.66E-04	2.66E-04
Na-24	3.39E+03	1.07E-04	1.07E-04
Cr-51	1.99E+03	5.44E-06	N/A
Mn-54	4.88E+02	1.73E-02	9.08E-02
Fe-55	3.01E+02	4.23E-03	1.81E-02
Fe-59	1.57E+03	5.93E-03	1.55E-02
Co-58	1.20E+03	2.04E-03	9.10E-04
Co-60	2.55E+02	3.18E-02	1.44E-02
Zn-65	5.60E+02	5.74E-01	1.27E+00
Rb-86	2.34E+03	1.48E-01	3.18E-01
Sr-89	1.46E+03	5.84E-03	N/A
Sr-90	2.16E+02	1.09E+01	N/A
Y-91	1.34E+03	2.50E-06	N/A
Zr-95	1.27E+03	2.30E-06	3.40E-06
Zr-97	3.39E+03	4.33E-10	9.46E-10
Nb-95	1.78E+03	6.24E-04	1.16E-03
Mo-99	3.30E+03	2.22E-05	1.17E-04
Ru-103	1.68E+03	2.40E-05	N/A
Ru-106	4.48E+02	5.80E-04	N/A
Ag-110m	5.52E+02	8.78E-05	1.48E-04
Sb-124	1.32E+03	3.25E-04	1.55E-05
Sb-125	2.98E+02	8.10E-04	3.80E-05
Te-125m	1.35E+03	2.35E-03	6.35E-03
Te-127m	9.16E+02	8.37E-03	2.46E-02
Te-129m	1.82E+03	8.19E-03	1.93E-02
Te-131m	3.38E+03	3.27E-04	3.92E-04
Te-132	3.27E+03	1.51E-03	1.61E-03
I-131	2.94E+03	3.22E-04	5.62E-04
I-132	3.40E+03	2.98E-07	8.51E-07
I-133	3.39E+03	1.00E-05	3.29E-05
I-134	3.40E+03	6.19E-08	1.73E-07
I-135	3.40E+03	1.90E-06	5.15E-06
Cs-134	3.29E+02	1.73E+01	2.11E+01
Cs-136	2.62E+03	2.25E-01	3.12E-01
Cs-137	2.15E+02	1.57E+01	2.40E+01
Cs-138	3.40E+03	2.65E-06	5.34E-06
Ba-140	2.65E+03	9.83E-05	1.[illegible]8E-06
La-140	3.36E+03	1.31E-08	4.94E-08
Ce-141	1.85E+03	1.14E-07	1.00E-06
Ce-143	3.37E+03	3.93E-10	3.55E-06
Ce-144	5.14E+02	2.70E-05	2.10E-04
Np-239	3.32E+03	1.51E-09	2.75E-09

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Surry Radioactive Liquid Waste Sampling and Analysis Program

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml), (Note 1)
Batch Releases (Note 2)	P (Each Batch)	P (Each Batch)	Principle Gamma Emitters (Note 3)	5×10^{-7}
			I^{131}	1×10^{-6}
	P (One Batch/M)	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P (Each Batch)	M Composite (Note 4)	H^3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P (Each Batch)	Q Composite (Note 4)	Sr^{89} and Sr^{90}	5×10^{-8}
			Fe^{55}	1×10^{-6}
Continuous Releases (Note 5)	Continuous (Note 6)	W Composite (Note 6)	Principal Gamma Emitters (Note 6)	5×10^{-7}
			I^{131}	1×10^{-6}
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Continuous (Note 6)	M Composite (Note 6)	H^3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Continuous (Note 6)	Q Composite (Note 6)	Sr^{89} and Sr^{90}	5×10^{-8}
			Fe^{55}	1×10^{-6}

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Surry Radioactive Liquid Waste Sampling and Analysis Program

NOTE 1: For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \bullet V \bullet 2.22E{+}06 \bullet Y \bullet e^{-(\lambda \Delta t)}} \tag{8-1}$$

Where:

LLD = the "a priori" (before the fact) Lower Limit of Detection (as microcuries per unit mass or volume) (see 4.8)

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

E = the counting efficiency (as counts per disintegration)

V = the sample size (in units of mass or volume)

2.22E+06 = the number of disintegrations per minute (dpm) per microcurie

Y = the fractional radiochemical yield (when applicable)

λ = the radioactive decay constant for the particular radionuclide

Δt = the elapsed time between the midpoint of sample collection and time of counting

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

The LLD is an "a priori" (before the fact) limit representing the capability of a measurement system and not a "posteriori" (after the fact) limit for a particular measurement.

NOTE 2: A batch release is the discharge of liquid wastes of a discrete volume. Before sampling for analyses, each batch shall be isolated, and appropriate methods will be used to obtain a representative sample for analysis.

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Surry Radioactive Liquid Waste Sampling and Analysis Program

NOTE 3: The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn^{54}, Fe^{59}, Co^{58}, Co^{60}, Zn^{65}, Mo^{99}, Cs^{134}, Cs^{137}, Ce^{141}, and Ce^{144}. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, at levels exceeding the LLD, together with the above nuclides, shall also be identified and reported.

NOTE 4: A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and for which the method of sampling employed results in a specimen that is representative of the liquids released.

NOTE 5: A continuous release is the discharge of liquid wastes of a non-discrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

NOTE 6: To be representative of the quantities and concentrations of radioactive materials in liquid effluents, composite sampling shall employ appropriate methods which will result in a specimen representative of the effluent release.

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North Anna Radioactive Liquid Waste Sampling and Analysis Program

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (µCi/ml), (Note 1)
Batch Releases (Notes 2 and 7)	P (Each Batch)	P (Each Batch)	Principle Gamma Emitters (Note 3)	5×10^{-7}
			I^{131}	1×10^{-6}
	P (One Batch/M)	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P (Each Batch)	M Composite (Note 4)	H^3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P (Each Batch)	Q Composite (Note 4)	Sr^{89} and Sr^{90}	5×10^{-8}
			Fe^{55}	1×10^{-6}
Continuous Releases (Note 5)	Continuous (Note 6)	W Composite (Note 6)	Principal Gamma Emitters (Note 6)	5×10^{-7}
			I^{131}	1×10^{-6}
			Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Continuous (Note 6)	M Composite (Note 6)	H^3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Continuous (Note 6)	Q Composite (Note 6)	Sr^{89} and Sr^{90}	5×10^{-8}
			Fe^{55}	1×10^{-6}

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North Anna Radioactive Liquid Waste Sampling and Analysis Program

NOTE 1: For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \bullet V \bullet 2.22E{+}06 \bullet Y \bullet e^{-(\lambda \Delta t)}} \qquad (9\text{-}1)$$

Where:

LLD	=	the "a priori" (before the fact) Lower Limit of Detection as defined above (as microcuries per unit mass or volume) (see 4.8)
s_b	=	the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute, cpm)
E	=	the counting efficiency (as counts per disintegration)
V	=	the sample size (in units of mass or volume)
2.22E+06	=	the number of disintegrations per minute (dpm) per microcurie
Y	=	the fractional radiochemical yield (when applicable)
λ	=	the radioactive decay constant for the particular radionuclide
Δt	=	the elapsed time between the midpoint of sample collection and time of counting

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

The LLD is an "a priori" (before the fact) limit representing the capability of a measurement system and not a "posteriori" (after the fact) limit for a particular measurement.

NOTE 2: A batch release is the discharge of liquid wastes of a discrete volume. Before sampling for analyses, each batch shall be isolated, and then thoroughly mixed as the situation permits, to assure representative sampling.

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North Anna Radioactive Liquid Waste Sampling and Analysis Program

NOTE 3: The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn^{54}, Fe^{59}, Co^{58}, Co^{60}, Zn^{65}, Mo^{99}, Cs^{134}, Cs^{137}, Ce^{141}, and Ce^{144}. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, at levels exceeding the LLD, together with the above nuclides, shall also be identified and reported.

NOTE 4: A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and for which the method of sampling employed results in a specimen that is representative of the liquids released.

NOTE 5: A continuous release is the discharge of liquid wastes of a non-discrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

NOTE 6: To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent releases.

NOTE 7: Whenever the secondary coolant activity exceeds 10^{-5} μCi/ml, the turbine building sump pumps shall be placed in manual operation and samples shall be taken and analyzed prior to release. Secondary coolant activity samples shall be collected and analyzed on a weekly basis. These samples are analyzed for gross activity or gamma isotopic activity within 24 hours.

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Surry Radioactive Gaseous Waste Sampling and Analysis Program

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml), (Note 1)
A. Waste Gas Storage Tank	Prior to Release (Each Tank) (Grab Sample)	Prior to Release (Each Tank)	Principal Gamma Emitters (Note 2)	1×10^{-4}
B. Containment Purge	Prior to Release (Each PURGE) (Grab Sample)	Prior to Release (Each PURGE)	Principle Gamma Emitters (Note 2)	1×10^{-4}
			H^3	1×10^{-6}
C. Ventilation (1)Process Vent (2)Vent Vent #1 (3)Vent Vent #2 (4)SRF Vent	Weekly (Crab Sample) (Note 3)	Weekly (Note 3)	Principle Gamma Emitters (Note 2)	1×10^{-4}
			H^3	1×10^{-6}
All Release Types as listed in A, B, and C	Continuous (Note 4)	Weekly (Note 5) (Charcoal Sample)	I^{131}	1×10^{-12}
			I^{133}	1×10^{-10}
	Continuous (Note 4)	Weekly (Note 5) Particulate Sample	Principal Gamma Emitter (Note 2)	1×10^{-11}
	Continuous (Note 4)	Weekly Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous (Note 4)	Quarterly Composite Particulate	Sr^{89} and Sr^{90}	1×10^{-11}
	Continuous (Note 4)	Noble Gas Monitor	Noble Gases Gross Beta and Gamma	1×10^{-6}
Condenser Air Ejector	Weekly Grab Sample (Note 3)	Weekly (Note 3)	Principle Gamma Emitters (Note 2)	1×10^{-4}
			H^3	1×10^{-6}

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Surry Radioactive Gaseous Waste Sampling and Analysis Program

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (µCi/ml), (Note 1)
Containment Hog Depres-surization	Prior to Release (Grab Sample)	Prior to Release (Each Release)	Principle Gamma Emitters	1×10^{-4}
			H^3	1×10^{-6}
	Continuous (Note 4)	Charcoal Sample (Note 6)	I^{131}	1×10^{-11}
			I^{133}	1×10^{-10}
	Continuous (Note 4)	Particulate Sample (Note 6)	Principal Gamma Emitter (Note 2)	1×10^{-10}
	Continuous (Note 4)	Composite Particulate Sample (Note 6)	Gross Alpha	1×10^{-10}
	Continuous (Note 4)	Composite Particulate Sample (Note 6)	Sr^{89} and Sr^{90}	1×10^{-10}

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Surry Radioactive Gaseous Waste Sampling and Analysis Program

NOTE 1: For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \bullet V \bullet 2.22E{+}06 \bullet Y \bullet e^{-(\lambda \Delta t)}} \tag{10-1}$$

Where:

LLD = the "a priori" (before the fact) Lower Limit of Detection as defined above (as microcuries per unit mass or volume) (see 4.8).

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

E = the counting efficiency (as counts per disintegration).

V = the sample size (in units of mass or volume).

2.22E+06 = the number of disintegrations per minute (dpm) per microcurie.

Y = the fractional radiochemical yield (when applicable).

λ = the radioactive decay constant for the particular radionuclide.

Δt = the elapsed time between the midpoint of sample collection and time of counting.

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

The LLD is an "a priori" (before the fact) limit representing the capability of a measurement system and not a "posteriori" (after the fact) limit for a particular measurement.

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Surry Radioactive Gaseous Waste Sampling and Analysis Program

NOTE 2: The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr^{87}, Kr^{88}, Xe^{133}, Xe^{133m}, Xe^{135}, Xe^{135m}, 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 detected and reported. Other nuclides with half lives greater than 8 days, that are measurable and identifiable at levels exceeding the LLD, together with the above nuclides, shall also be identified and reported.

NOTE 3: Sampling and analysis shall also be performed following shutdown, start-up, and whenever a thermal power change exceeding 15 percent of the rated thermal power occurs within any one-hour period, when:

a. Analysis shows that the dose equivalent I^{131} concentration in the primary coolant has increased more than a factor of 3; and

b. The noble gas activity monitor shows that effluent activity has increased by more than a factor of 3.

NOTE 4: The ratio of the sample flow rate to the sampled stream flow rate shall be known for the period covered by each dose or dose rate calculation made in accordance with 6.3.1, 6.3.3, and 6.3.4.

NOTE 5: Samples shall be changed at least once per seven days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least seven days following each shutdown, start-up, or thermal power change exceeding 15 percent of rated thermal power in one hour, and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement applies if:

a. Analysis shows that the dose equivalent I^{131} concentration in the primary coolant has increased by a factor of 3; and

b. Noble gas monitor shows that effluent activity has increased more than a factor of 3.

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Surry Radioactive Gaseous Waste Sampling and Analysis Program

NOTE 6: To be representative of the quantities and concentrations of radioactive materials in gaseous effluents, composite sampling shall employ appropriate methods that will result in a specimen representative of the effluent release.

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North Anna Radioactive Gaseous Waste Sampling and Analysis Program

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml), (Note 1)
A. Waste Gas Storage Tank	Prior to Release (Each Tank Grab Sample)	Prior to Release (Each Tank)	Principal Gamma Emitters (Note 2)	1×10^{-4}
B. Containment Purge	Prior to Release (Each PURGE Grab Sample)	Prior to Release (Each PURGE)	Principle Gamma Emitters (Note 2)	1×10^{-4}
			H^3	1×10^{-6}
C. Ventilation **(1) Process Vent** **(2) Vent Vent A** **(3) Vent Vent B**	Monthly (Grab Sample) (Notes 3, 4, and 5)	Monthiy (Note 3)	Principle Gamma Emitters (Note 2)	1×10^{-4}
			H^3	1×10^{-6}
All Release Types as listed in A, B, and C	Continuous (Note 4)	Weekly (Charcoal Sample)	I^{131}	1×10^{-12}
			I^{133}	1×10^{-10}
	Continuous (Note 4)	Weekly Particulate Sample	Principal Gamma Emitter (Note 2)	1×10^{-11}
	Continuous (Note 4)	Monthly Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous (Note 4)	Quarterly Composite Particulate	Sr^{89} and Sr^{90}	1×10^{-11}
	Continuous (Note 4)	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}
Condenser Air Ejector/Steam Generator Blowdown Vent	Weekly Grab Sample (Note 6)	Weekly	Principle Gamma Emitters (Note 7)	1×10^{-4}
			H^3	1×10^{-6}
Containment Vacuum Steam Ejector (Hogger)	Prior to Release (Grab Sample) (Note 8)	Prior to each Release	Principle Gamma Emitters (Note 2)	1×10^{-4}
			H^3	1×10^{-6}

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North Anna Radioactive Gaseous Waste Sampling and Analysis Program

NOTE 1: For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \bullet V \bullet 2.22E{+}06 \bullet Y \bullet e^{-(\lambda \Delta t)}} \qquad (11\text{-}1)$$

Where:

LLD	=	the "a priori" (before the fact) Lower Limit of Detection as defined above (as microcuries per unit mass or volume) (see Subsection 4.9)
s_b	=	the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute, cpm)
E	=	the counting efficiency (as counts per disintegration)
V	=	the sample size (in units of mass or volume)
2.22E+06	=	the number of disintegrations per minute (dpm) per microcurie
Y	=	the fractional radiochemical yield (when applicable)
λ	=	the radioactive decay constant for the particular radionuclide
Δt	=	the elapsed time between the midpoint of sample collection and time of counting

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

The LLD is an "a priori" (before the fact) limit representing the capability of a measurement system and not as "posteriori" (after the fact) limit for a particular measurement.

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North Anna Radioactive Gaseous Waste Sampling and Analysis Program

NOTE 2: The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr^{87}, Kr^{88}, Xe^{133}, Xe^{133m}, Xe^{135}, Xe^{135m}, 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 detected and reported. Other peaks that are measurable and identifiable, at levels exceeding the LLD, together with the above nuclides, shall also be identified and reported.

NOTE 3: Sampling and analysis shall also be performed following shutdown, start-up, and whenever a thermal power change exceeding 15 percent of the rated thermal power occurs within any one-hour period, if:

a. Analysis shows that the dose equivalent I^{131} concentration in the primary coolant is greater than 1.0 μCi/gm; and

b. The noble gas activity monitor shows that effluent activity has increased by more than a factor of 3.

NOTE 4: The ratio of the sample flow rate to the sampled stream flow rate shall be known for the period covered by each dose or dose rate calculation made in accordance with 6.3.1, 6.3.3, and 6.3.4.

NOTE 5: Samples shall be changed at least once per seven days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least seven days following each shutdown, start-up or thermal power change exceeding 15 percent of rated thermal power in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement applies if:

a. Analysis shows that the dose equivalent I^{131} concentration in the primary coolant is greater than 1.0 μCi/gm and;

b. Noble gas monitor shows that effluent activity has increased more than a factor of 3.

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North Anna Radioactive Gaseous Waste Sampling and Analysis Program

NOTE 6: Whenever the secondary coolant activity exceeds 10^{-5} µCi/ml, samples shall be obtained and analyzed weekly. Secondary coolant activity samples shall be collected and analyzed on a weekly basis. These samples are analyzed for gross activity or gamma isotopic activity within 24 hours.

NOTE 7: The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr^{87}, Kr^{88}, Xe^{133}, Xe^{133m}, Xe^{135}, Xe^{135m}, and Xe^{138} for gaseous emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, at levels exceeding the LLD together with the above nuclides, shall also be identified and reported.

NOTE 8: If the secondary coolant activity level in any Steam Generator supplying steam to the Hogger exceeds 1.0E-5 µCi/ml, Steam Generator samples shall be obtained and analyzed prior to release.

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Gaseous Effluent Dose Factors for Surry

(Gamma and Beta Dose Factors)

χ/Q = 6.0E-05 sec/m^3 at 499 meters N Direction

Dose Factors for Ventilation Vent

Noble Gas Radionuclide	K_{ivv} Total Body mrem/yr / Curie/sec	L_{ivv} Skin mrem/yr / Curie/sec	M_{ivv} Gamma Air mrad/yr / Curie/sec	N_{ivv} Beta Air mrad/yr / Curie/sec
Kr-83m	4.54E+00	-	1.16E+03	1.73E+04
Kr-85m	7.02E+04	8.76E+04	7.38E+04	1.18E+05
Kr-85	9.66E+02	8.04E+04	1.03E+03	1.17E+05
Kr-87	3.55E+05	5.84E+05	3.70E+05	6.18E+05
Kr-88	8.82E+05	1.42E+05	9.12E+05	1.76E+05
Kr-89	9.96E+05	6.06E+05	1.04E+06	6.36E+05
Kr-90	9.36E+05	4.37E+05	9.78E+05	4.70E+05
Xe-131m	5.49E+03	2.86E+04	9.36E+03	6.66E+04
Xe-133m	1.51E+04	5.96E+04	1.96E+04	8.88E+04
Xe-133	1.76E+04	1.84E+04	2.12E+04	6.30E+04
Xe-135m	1.87E+05	4.27E+04	2.02E+05	4.43E+04
Xe-135	1.09E+05	1.12E+05	1.15E+05	1.48E+05
Xe-137	8.52E+04	7.32E+05	9.06E+04	7.62E+05
Xe-138	5.30E+05	2.48E+05	5.53E+05	2.85E+05
Ar-41	5.30E+05	1.61E+05	5.58E+05	1.97E+05

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Gaseous Effluent Dose Factors for Surry

(Gamma and Beta Dose Factors)
χ/Q = 1.0E-06 sec/m^3 at 644 meters N Direction
Dose Factors for Process Vent

Noble Gas Radionuclide	K_{ipv} Total Body mrem/yr / Curie/sec	L_{ipv} Skin mrem/yr / Curie/sec	M_{ipv} Gamma Air mrad/yr / Curie/sec	N_{ipv} Beta Air mrad/yr / Curie/sec
Kr-83m	7.56E-02	-	1.93E+01	2.88E+02
Kr-85m	1.17E+03	1.46E+03	1.23E+03	1.97E+03
Kr-85	1.61E+01	1.34E+03	1.72E+01	1.95E+03
Kr-87	5.92E+03	9.73E+03	6.17E+03	1.03E+04
Kr-88	1.47E+04	2.37E+03	1.52E+04	2.93E+03
Kr-89	1.66E+04	1.01E+04	1.73E+04	1.06E+04
Kr-90	1.56E+04	7.29E+03	1.63E+04	7.83E+03
Xe-131m	9.15E+01	4.76E+02	1.56E+02	1.11E+03
Xe-133m	2.51E+02	9.94E+02	3.27E+02	1.48E+03
Xe-133	2.94E+02	3.06E+02	3.53E+02	1.05E+03
Xe-135m	3.12E+03	7.11E+02	3.36E+03	7.39E+02
Xe-135	1.81E+03	1.86E+03	1.92E+03	2.46E+03
Xe-137	1.42E+03	1.22E+04	1.51E+03	1.27E+04
Xe-138	8.83E+03	4.13E+03	9.21E+03	4.75E+03
Ar-41	8.84E+03	2.69E+03	9.30E+03	3.28E+03

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Gaseous Effluent Dose Factors for Surry

(Inhalation Pathway Dose Factors)

Ventilation Vent $\chi/Q = 6.0E\text{-}05\ sec/m^3$ at 499 meters N Direction

Process Vent $\chi/Q = 1.0E\text{-}06\ sec/m^3$ at 644 meters S Direction

Radionuclide	P_{ivv} mrem/yr / Curie/sec	P_{ipv} mrem/yr / Curie/sec
H-3	6.75E+04	1.12E+03
Cr-51	5.13E+03	8.55E+01
Mn-54	ND	ND
Fe-59	ND	ND
Co-58	ND	ND
Co-60	ND	ND
Zn-65	ND	ND
Rb-86	ND	ND
Sr-90	ND	ND
Y-91	ND	ND
Zr-95	ND	ND
Nb-95	ND	ND
Ru-103	ND	ND
Ru-106	ND	ND
Ag-110m	ND	ND
Te-127m	3.64E+05	6.07E+03
Te-129m	3.80E+05	6.33E+03
Cs-134	ND	ND
Cs-136	ND	ND
Cs-137	ND	ND
Ba-140	ND	ND
Ce-141	ND	ND
Ce-144	ND	ND
I-131	9.75E+08	1.62E+07
I-133	2.31E+08	3.85E+06

ND - No data for dose factor according to Regulatory Guide 1.109, Revision 1

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Gaseous Effluent Dose Factors for North Anna

(Gamma and Beta Dose Factors)
χ/Q = 9.3E-06 sec/m^3 at 1416 meters SE Direction
Dose Factors for Ventilation Vent

Noble Gas Radionuclide	K_{ivv} Total Body mrem/yr / Curie/sec	L_{ivv} Skin mrem/yr / Curie/sec	M_{ivv} Gamma Air mrad/yr / Curie/sec	N_{ivv} Beta Air mrad/yr / Curie/sec
Kr-83m	7.03E-01	-	1.79E+02	2.68E+03
Kr-85m	1.09E+04	1.36E+04	1.14E+04	1.83E+04
Kr-85	1.50E+02	1.25E+04	1.60E+02	1.81E+04
Kr-87	5.51E+04	9.05E+04	5.74E+04	9.58E+04
Kr-88	1.37E+05	2.20E+04	1.41E+05	2.72E+04
Kr-89	1.54E+05	9.39E+04	1.61E+05	9.86E+04
Kr-90	1.45E+05	6.78E+04	1.52E+05	7.28E+04
Xe-131m	8.51E+02	4.43E+03	1.45E+03	1.03E+04
Xe-133m	2.33E+03	9.24E+03	3.04E+03	1.38E+04
Xe-133	2.73E+03	2.85E+03	3.28E+03	9.77E+03
Xe-135m	2.90E+04	6.61E+03	3.12E+04	6.87E+03
Xe-135	1.68E+04	1.73E+04	1.79E+04	2.29E+04
Xe-137	1.32E+04	1.13E+05	1.40E+04	1.18E+05
Xe-138	8.21E+04	3.84E+04	8.57E+04	4.42E+04
Ar-41	8.22E+04	2.50E+04	8.65E+04	3.05E+04

ATTACHMENT 13

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Gaseous Effluent Dose Factors for North Anna

(Gamma and Beta Dose Factors)

χ/Q = 1.2E-06 sec/m^3 at 1513 meters S Direction

Dose Factors for Process Vent

Noble Gas Radionuclide	K_{ipv} Total Body mrem/yr / Curie/sec	L_{ipv} Skin mrem/yr / Curie/sec	M_{ipv} Gamma Air mrad/yr / Curie/sec	N_{ipv} Beta Air mrad/yr / Curie/sec
Kr-83m	9.07E-02	-	2.32E+01	3.46E+02
Kr-85m	1.40E+03	1.75E+03	1.48E+03	2.36E+03
Kr-85	1.93E+01	1.61E+03	2.06E+01	2.34E+03
Kr-87	7.10E+03	1.17E+04	7.40E+03	1.24E+04
Kr-88	1.76E+04	2.84E+03	1.82E+04	3.52E+03
Kr-89	1.99E+04	1.21E+04	2.08E+04	1.27E+04
Kr-90	1.87E+04	8.75E+03	1.96E+04	9.40E+03
Xe-131m	1.10E+02	5.71E+02	1.87E+02	1.33E+03
Xe-133m	3.01E+02	1.19E+03	3.92E+02	1.78E+03
Xe-133	3.53E+02	3.67E+02	4.24E+02	1.26E+03
Xe-135m	3.74E+03	8.53E+02	4.03E+03	8.87E+02
Xe-135	2.17E+03	2.23E+03	2.30E+03	2.95E+03
Xe-137	1.70E+03	1.46E+04	1.81E+03	1.52E+04
Xe-138	1.06E+04	4.96E+03	1.11E+04	5.70E+03
Ar-41	1.06E+04	3.23E+03	1.12E+04	3.94E+03

ATTACHMENT 13

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Gaseous Effluent Dose Factors for North Anna

(Inhalation Pathway Dose Factors)

Ventilation Vent χ/Q = 9.3E-06 sec/m^3 at 1416 meters SE Direction

Process Vent χ/Q = 1.2E-06 sec/m^3 at 1513 meters S Direction

Radionuclide	P_{ivv} <u>mrem/yr</u> Curie/sec	P_{ipv} <u>mrem/yr</u> Curie/sec
H-3	1.05E+04	1.35E+03
Cr-51	7.95E+02	1.02E+02
Mn-54	ND	ND
Fe-59	ND	ND
Co-58	ND	ND
Co-60	ND	ND
Zn-65	ND	ND
Rb-86	ND	ND
Sr-90	ND	ND
Y-91	ND	ND
Zr-95	ND	ND
Nb-95	ND	ND
Ru-103	ND	ND
Ru-106	ND	ND
Ag-110m	ND	ND
Te-127m	5.64E+04	7.28E+03
Te-129m	5.88E+04	7.59E+03
Cs-134	ND	ND
Cs-136	ND	ND
Cs-137	ND	ND
Ba-140	ND	ND
Ce-141	ND	ND
Ce-144	ND	ND
I-131	1.51E+08	1.95E+07
I-133	3.58E+07	4.62E+06

ND - No data for dose factor according to Regulatory Guide 1.109, Revision 1

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ATTACHMENT 14

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Surry Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT	MINIMUM OPERABLE CHANNELS	ACTION
1. PROCESS VENT SYSTEM		
(a) Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release 1-GW-RM-102 1-GW-RM-130-1	1	1
(b) Iodine Sampler Process Vent Continuous HP Sampler, or 1-GW-RM-130-1	1	2
(c) Particulate Sampler Process Vent Continuous HP Sampler, or 1-GW-RM-130-1	1	2
(d) Process Vent Flow Rate Monitor 1-GW-FT-100	1	3
(e) Sampler Flow Rate Measuring Device KAMAN Flow Rate Measuring Device (Parameter #19), or HP Sampler Rotometer	1	3
2. CONDENSER AIR EJECTOR SYSTEM		
(a) Gross Activity Monitor 1-SV-RM-111 2-SV-RM-211	2 (one per unit)	1
(b) Air Ejector Flow Rate Measuring Device Unit 1: 1-VP-FI-1A 1-VP-FI-1B Unit 2: 2-VP-FI-1A 2-VP-FI-1B	2 (one per unit)	3
3. VENTILATION VENT SYSTEM		
(a) Noble Gas Activity Monitor		
SRF: RRM-101	1	1
SPS: Vent #1 1-VG-RM-104	1	1
Vent #2, 1-VG-RM -110 , or 1-VG-RM-131-1	1	1

ATTACHMENT 14

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Surry Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT	MINIMUM OPERABLE CHANNELS	ACTION
(b) Iodine Sampler		
SRF: RRM-101	1	2
SPS: Vent #1, 1-VG-RM-104	1	2
Vent #2, Continuous HP Sampler, or 1-VG-RM-131-1	1	2
(c) Particulate Sampler		
SRF: RRM-101	1	2
SPS: Vent #1, VG-RM-104	1	2
Vent #2, HP Continuous Sampler, or 1-VG-RM-131-1	1	2
(d) Ventilation Vent Flow Rate Monitor		
SRF: 01-RHV-FT-156	1	3
SPS: Vent #1, 1-VS-FT-119	1	3
Vent #2, 1-VS-FT-116	1	3
(e) Sampler Flow Rate Measuring Device		
SRF: RRM-101	1	3
SPS: Vent #1, 1-VG-RM-104	1	3
Vent #2, KAMAN Flow Rate Measuring Device (Parameter #19), or HP Sampler Rotometer	1	3

ATTACHMENT 14

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Surry Radioactive Gaseous Effluent Monitoring Instrumentation

ACTION 1: If the number of operable channels is less than required, effluent releases via this path may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 2: If the number of operable channels is less than required, effluent releases via the effected path may continue provided samples are continuously collected within one hour with auxiliary sampling equipment as required in Attachment 10.

ACTION 3: If the number of operable channels is less than required, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

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ATTACHMENT 15

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North Anna Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT	MINIMUM OPERABLE CHANNELS	ACTION
1. PROCESS VENT SYSTEM		
(a) Noble Gas Activity Monitor		
1-RM-GW-102 1-RM-GW-178-1	1	2, 4
(b) Iodine Sampler		
1-RM-GW-178-1 Process Vent Continuous HP Sampler	1	2, 5
(c) Particulate Sampler		
1-RM-GW-178-1 Process Vent Continuous HP Sampler	1	2, 5
(d) Total Flow Monitor		
1-GW-FT-108	1	1
(e) Sampler Flow Rate Measuring Device		
KAMANS Flow Rate Measuring Device (Parameter 19) HP Sampler Rotameter	1	1
2. CONDENSER AIR EJECTOR SYSTEM		
(a) Gross Activity Monitor		
Unit 1 1-SV-RM-121 Unit 2 2-SV-RM-221	1	3
(b) Flow Rate Measuring Device		
Unit 1 1-SV-FI-100A 1-SV-FI-101A 1-SV-FI-100B 1-SV-FI-101B	1 (NOTE 1)	1
Unit 2 2-SV-FI-200A 2-SV-FI-201A 2-SV-FI-200B 2-SV-FI-201B	1 (NOTE 2)	1

ATTACHMENT 15

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North Anna Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT	MINIMUM OPERABLE CHANNELS	ACTION
3. VENTILATION VENT A		
(a) Noble Gas Activity Monitor		
1-RM-VG-104 1-RM-VG-179-1	1	2
(b) Iodine Sampler		
1-RM-VG-179-1 Vent Vent A Continuous HP Sampler	1	2
(c) Particulate Sampler		
1-RM-VG-179-1 Vent Vent A Continuous HP Sampler	1	2
(d) Total Flow Monitor		
1-HV-FT-1212A	1	1
(e) Sampler Flow Rate Measuring Device		
KAMANS Flow Rate Measuring Device (Parameter 19) HP Sampler Rotameter	1	1
4. VENTILATION VENT B		
(a) Noble Gas Activity Monitor		
1-RM-VG-113 1-RM-VG-180-1	1	2
(b) Iodine Sampler		
1-RM-VG-180-1 Vent Vent B Continuous HP Sampler	1	2
(c) Particulate Sampler		
1-RM-VG-180-1 Vent Vent B Continuous HP Sampler	1	2
(d) Total Flow Monitor		
1-HV-FT-1212B	1	1
(e) Sampler Flow Rate Measuring Device		
KAMANS Flow Rate Measuring Device (Parameter 19) HP Sampler Rotameter	1	1

ATTACHMENT 15

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North Anna Radioactive Gaseous Effluent Monitoring Instrumentation

ACTION 1: If the number of operable channels is less than required, effluent releases, via this path, may continue if the flow rate is estimated at least once per four hours.

ACTION 2: If the number of operable channels is less than required, effluent releases, via this path, may continue if grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity or gamma isotopic activity within 24 hours.

ACTION 3: If the number of operable channels is less than required, effluent releases, via this path, may continue if the frequency of the grab samples provided by Technical Specification requirement 4.4.6.3.b is increased to at least once per four hours and these samples are analyzed for gross activity or gamma isotopic activity within eight hours.

ACTION 4: If the number of operable channels is less than required, the contents of the Waste Gas Decay Tanks may be released to the environment provided that prior to initiation of the release:

a. At least two independent samples of the tank's contents are analyzed, and:

b. At least two technically qualified members of the Station staff independently verify the release rate calculations and discharge valve lineup.

ACTION 5: If the number of operable channels is less than required, effluent releases from the Waste Gas Decay Tank may continue provided samples are continuously collected with auxiliary sampling equipment as required in Attachment 11.

NOTE 1: A channel shall consist of:

a. The flow instrument installed in the ejector through which the discharge is routed; either Train A (1-SV-FI-100A, 101A), or Train B (1-SV-FI-100B, 101B) or both.

b. Flow instruments 101A and 101B provide low range measurement. Flow instruments 100A and 100B provide high range measurement.

NOTE 2: A channel shall consist of:

a. The flow instrument installed in the ejector through which the discharge is routed; either Train A (2-SV-FI-200A, 201A), or Train B (2-SV-FI-200B, 201B) or both.

b. Flow instruments 201A and 201B provide low range measurement. Flow instruments 200A and 200B provide high range measurement.

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ATTACHMENT 16

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Surry Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

CHANNEL DESCRIPTION	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. PROCESS VENT SYSTEM				
(a) Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release 1-GW-RM-102 1-GW-RM-130-1	D	M, *	R	Q
(b) Iodine Sampler Process Vent Continuous HP Sampler, or 1-GW-RM- 130-1	W	N/A	N/A	N/A
(c) Particulate Sampler Process Vent Continuous HP Sampler, or 1-GW-RM- 130-1	W	N/A	N/A	N/A
(d) Process Vent Flow Rate Monitor 1-GW-FT-100	D	N/A	R	N/A
(e) Sampler Flow Rate Measuring Device HP Sampler Rotometer, or	D	N/A	SA	N/A
KAMAN Flow Rate Measuring Device (Parameter #19)	D	N/A	R	N/A
2. CONDENSER AIR EJECTOR SYSTEM				
(a) Gross Activity Monitor Unit 1: 1-SV-RM-111 Unit 2: 2-SV-RM-211	D	M	R	Q
(b) Air Ejector Flow Rate Measuring Device Unit 1: 1-VP-FI-1A 1-VP-FI-1B Unit 2: 2-VP-FI-1A 2-VP-FI-1B	D	N/A	R	N/A
3. VENTILATION VENT SYSTEM				
(a) Noble Gas Activity Monitor SRF: RRM-101 SPS: 1-VG-RM -110 1-VG-RM -131-1	D	M	R	Q

ATTACHMENT 16

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Surry Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

CHANNEL DESCRIPTION	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
(b) Iodine Sampler SRF: RRM-101 SPS: Vent #1, 1-VG-RM-104 Vent #2, Continuous HP Sampler, or 1-VG-RM-131-1	W	N/A	N/A	N/A
(c) Particulate Sampler SRF: RRM-101 SPS: Vent #1, 1-VG-RM-104 Vent #2, Continuous HP Sampler, or 1-VG-RM-131-1	W	N/A	N/A	N/A
(d) Ventilation Vent Flow Rate Monitor SRF: 01-RHV-FT-156 SPS: Vent #1, 1-VS-FT-119 Vent #2, 1-VS-FT-116	D	N/A	R	N/A
(e) Sampler Flow Rate Measuring Device				
SRF: RRM-101	D	N/A	R	N/A
SPS: Vent #1, 1-VG-RM-104	D	N/A	R	N/A
Vent #2, KAMAN Flow Rate Measuring Device (Parameter #19), or	D	N/A	R	N/A
HP Sampler Rotometer	D	N/A	S/A	N/A

* Prior to each Waste Gas Decay Tank release

ATTACHMENT 17

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North Anna Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

CHANNEL DESCRIPTION	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. PROCESS VENT SYSTEM				
(a) Noble Gas Activity Monitor				
1-RM-GW-102	D	M (NOTE 5)	R	Q (NOTE 1)
1-RM-GW-178-1	D	M (NOTE 5)	R	Q (NOTE 1)
(b) Iodine Sampler				
1-RM-GW-178-1	W	N/A	N/A	N/A
Process Vent Continuous HP Sampler	D (NOTE 3)	N/A	N/A	N/A
(c) Particulate Sampler				
1-RM-GW-178-1	W	N/A	N/A	N/A
Process Vent Continuous HP Sampler	D (NOTE 3)	N/A	N/A	N/A
(d) Total Flow Monitor				
1-GW-FT-108	D	N/A	R	Q
(e) Sampler Flow Rate Measuring Device				
KAMANS Flow Rate Measuring Device (Parameter 19)	D (NOTE 3)	N/A	R	N/A
HP Sampler Rotameter	D (NOTE 3)	N/A	SA	N/A
2. CONDENSER AIR EJECTOR SYSTEM				
(a) Noble Gas Activity Monitor				
Unit 1 1-SV-RM-121 Unit 2 2-SV-RM-221	D	M	R	Q (NOTE 1)
(b) Flow Rate Measuring Device				
Unit 1 1-SV-FI-100A 1-SV-FI-101A 1-SV-FI-100B 1-SV-FI-101B	D	N/A	R	N/A
Unit 2 2-SV-FI-200A 2-SV-FI-201A 2-SV-FI-200B 2-SV-FI-201B	D	N/A	R	N/A

ATTACHMENT 17

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North Anna Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

CHANNEL DESCRIPTION	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
3. VENTILATION VENT A				
(a) Noble Gas Activity Monitor				
1-RM-VG-104	D	M	R	Q (NOTE 2)
1-RM-VG-179-1	D	M (NOTE 5)	R	Q (NOTE 2)
(b) Iodine Sampler				
1-RM-VG-179-1	W	N/A	N/A	N/A
Vent Vent A Continuous HP Sampler	D (NOTE 3)	N/A	N/A	N/A
(c) Particulate Sampler				
1-RM-VG-179-1	W	N/A	N/A	N/A
Vent Vent A Continuous HP Sampler	D (NOTE 3)	N/A	N/A	N/A
(d) Total Flow Monitor				
1-HV-FT-1212A	D	N/A	R	Q
(e) Sampler Flow Rate Measuring Device				
KAMANS Flow Rate Measuring Device (Parameter 19)	D (NOTE 3)	N/A	R	N/A
HP Sampler Rotameter	D (NOTE 3)	N/A	SA	N/A
4. VENTILATION VENT B				
(a) Noble Gas Activity Monitor				
1-RM-VG-113	D	M	R	Q (NOTE 4)
1-RM-VG-180-1	D	M (NOTE 5)	R	Q (NOTE 2)
(b) Iodine Sampler				
1-RM-VG-180-1	W	N/A	N/A	N/A
Vent Vent B Continuous HP Sampler	D (NOTE 3)	N/A	N/A	N/A
(c) Particulate Sampler				
1-RM-VG-180-1	W	N/A	N/A	N/A
Vent Vent B Continuous HP Sampler	D (NOTE 3)	N/A	N/A	N/A
(d) Total Flow Monitor				
1-HV-FT-1212B	D	N/A	R	Q
(e) Sampler Flow Rate Measuring Device				
KAMANS Flow Rate Measuring Device (Parameter 19)	D (NOTE 3)	N/A	R	N/A
HP Sampler Rotameter	D (NOTE 3)	N/A	SA	N/A

ATTACHMENT 17

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North Anna Radioactive Gaseous Effluent Monitoring InstrumentationSurveillance Requirements

NOTE 1: The Channel Functional Test shall demonstrate:

a. Automatic actuation of the valves in this pathway and Control Room alarm annunciation occur if the instrument indicates measured levels above the alarm/trip setpoint.

b. Alarm annunciation occurs if the instrument controls not set in "operate" mode.

NOTE 2: The Channel Functional Test shall demonstrate:

a. Control Room alarm annunciation occurs if the instrument indicates measured levels are above the alarm/trip setpoint.

b. Alarm annunciation occurs if the instrument controls not set in "operate" mode.

NOTE 3: Channel Checks shall consist of verifying indication of flow during periods of release. Channel Checks shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

NOTE 4: The Channel Functional Test shall demonstrate that:

a. Control Room alarm annunciation occurs if the instrument indicates measured levels are above alarm/trip setpoint.

b. The Instrument mode selection control automatically resets to "operate" mode when released.

NOTE 5: Monitors 1-RM-GW-178-1, 1-RM-VG-179-1, and 1-RM-VG-180-1 are electronically source checked using an LED.

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ATTACHMENT 18

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Critical Organ and Inhalation Dose Factors for Surry

(Critical Pathway Dose Factors)
Ventilation Vent D/Q = 9.0E-10 m^{-2} at 5150 meters S Direction
Process Vent D/Q = 4.3E-10 m^{-2} at 5150 meters S Direction

Radionuclide	RM_{ivv} mrem/yr / Curie/sec	RM_{ipv} mrem/yr / Curie/sec
H-3	7.20E+02	3.12E+02
Mn-54	ND	ND
Fe-59	ND	ND
Cr-51	6.45E+01	3.08E+01
Co-58	ND	ND
Co-60	ND	ND
Zn-65	ND	ND
Rb-86	ND	ND
Sr-89	ND	ND
Sr-90	ND	ND
Y-91	ND	ND
Zr-95	ND	ND
Nb-95	ND	ND
Ru-103	ND	ND
Ru-106	ND	ND
Ag-110m	ND	ND
Te-127m	8.06E+04	3.85E+04
Te-129m	1.25E+05	5.98E+04
I-131	6.21E+08	2.97E+08
I-133	5.79E+06	2.77E+06
Cs-134	ND	ND
Cs-136	ND	ND
Cs-137	ND	ND
Ba-140	ND	ND
Ce-141	ND	ND
Ce-144	ND	ND

ND - No data for dose factor according to Regulatory Guide 1.109, Revision 1

ATTACHMENT 18

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Critical Organ and Inhalation Dose Factors for Surry

(Inhalation Pathway Dose Factors)
Ventilation Vent χ/Q = 3.0E-07 sec/m^3 at 5150 meters S Direction
Process Vent χ/Q = 1.3E-07 sec/m^3 at 5150 meters S Direction

Radionuclide	RM_{ivv} mrem/yr / Curie/sec	RM_{ipv} mrem/yr / Curie/sec
H-3	1.94E+02	8.41E+01
Cr-51	1.73E+01	7.48E+00
Mn-54	ND	ND
Fe-59	ND	ND
Co-58	ND	ND
Co-60	ND	ND
Zn-65	ND	ND
Rb-86	ND	ND
Sr-89	ND	ND
Sr-90	ND	ND
Y-91	ND	ND
Zr-95	ND	ND
Nb-95	ND	ND
Ru-103	ND	ND
Ru-106	ND	ND
Ag-110m	ND	ND
Te-127m	1.46E+03	6.33E+02
Te-129m	1.64E+03	7.12E+02
I-131	4.45E+06	1.93E+06
I-133	1.07E+06	4.63E+05
Cs-134	ND	ND
Cs-136	ND	ND
Cs-137	ND	ND
Ba-140	ND	ND
Ce-141	ND	ND
Ce-144	ND	ND

ND - No data for dose factor according to Regulatory Guide 1.109, Revision 1

ATTACHMENT 19

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Critical Organ Dose Factors for North Anna

(Critical Pathway Dose Factors)
Ventilation Vent D/Q = 2.4E-09 m^{-2} at 3250 meters N Direction
Process Vent D/Q = 1.1E-09 m^{-2} at 3250 meters N Direction

Radionuclide	RM_{ivv} <u>mrem/yr</u> Curie/sec	RM_{ipv} <u>mrem/yr</u> Curie/sec
H-3	1.73E+03	9.36E+02
Mn-54	ND	ND
Fe-59	ND	ND
Cr-51	1.50E+02	6.89E+01
Co-58	ND	ND
Co-60	ND	ND
Zn-65	ND	ND
Rb-86	ND	ND
Sr-89	ND	ND
Sr-90	ND	ND
Y-91	ND	ND
Zr-95	ND	ND
Nb-95	ND	ND
Ru-103	ND	ND
Ru-106	ND	ND
Ag-110m	ND	ND
Te-127m	1.97E+05	9.04E+04
Te-129m	2.95E+05	1.35E+05
I-131	1.45E+09	6.72E+08
I-133	1.33E+07	6.12E+06
Cs-134	ND	ND
Cs-136	ND	ND
Cs-137	ND	ND
Ba-140	ND	ND
Ce-141	ND	ND
Ce-144	ND	ND

ND - No data for dose factor according to Regulatory Guide 1.109, Revision 1

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ATTACHMENT 20

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Surry Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Sample and Sample Location	Collection Frequency	Type and Frequency of Analysis
1. DIRECT RADIATION	About 40 Routine Monitoring Stations to be placed as follows: 1) Inner Ring in general area of site boundary with station in each sector 2) Outer Ring 6 to 8 km from the site with a station in each sector 3) The balance of the 8 dosimeters should be placed in special interest areas such as population centers, nearby residents, schools, and in 2 or 3 areas to serve as controls	Quarterly	GAMMA DOSE Quarterly
2. AIRBORNE Radioiodines and Particulates	Samples from 7 locations: a) 1 sample from close to the site boundary location of the highest calculated annual average ground level D/Q b) 5 sample locations 6-8 km distance located in a concentric ring around the Station c) 1 sample from a control location 15-30 km distant, providing valid background data	Continuous Sampler operation with sample collection weekly	Radioiodine Canister I^{131} Analysis Weekly Particulate Sampler Gross beta radioactivity analysis following filter change; Gamma isotopic analysis of composite (by location) quarterly

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Surry Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Sample and Sample Location	Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a) Surface	a) 1 sample upstream b) 1 sample downstream	Monthly Sample	Gamma isotopic analysis monthly; Composite for tritium analysis quarterly
b) Ground	Sample from 1 or 2 sources	Quarterly	Gamma isotopic and tritium analysis quarterly
c) Sediment from shoreline	1 sample from downstream area with existing or potential recreational value	Semi-Annually	Gamma isotopic analysis semi-annually
d) Silt	5 samples from vicinity of the Station	Semi-Annually	Gamma isotopic analysis semi-annually
4. INGESTION			
a) Milk (NOTE 1)	a) 3 samples from milking animals in the vicinity of the Station b) 1 sample from milking animals at a control location (15-30 km distant)	Monthly	Gamma isotopic and I^{131} analysis monthly
b) Fish and Invertebrates	a) 2 samples of oysters in the vicinity of the Station	Semi-Annually	Gamma isotopic on edibles
	b) 4 samples of clams in the vicinity of the Station	Semi-Annually	Gamma isotopic on edibles
	c) 1 sampling of crabs from the vicinity of the Station	Annually	Gamma isotopic on edibles
	d) 2 samples of fish from the vicinity of the Station (catfish, white perch, eel)	Semi-Annually	Gamma isotopic on edibles

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Surry Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Sample and Sample Location	Collection Frequency	Type and Frequency of Analysis
4. INGESTION (Continued)			
c) Food Products	a) 1 sample corn b) 1 sample soybeans c) 1 sample peanuts	Annually	Gamma isotopic on edible portion
	d) 1 sample of a broadleaf vegetation of two different available offsite locations with highest annual average ground level D/Q, if one or more milk samples are unavailable e) 1 sample of a broadleaf vegetation grown 15-30 km in the available, least prevalent wind direction, if one or more milk samples are unavailable	Monthly, if available, or at harvest	Gamma isotopic and I^{131} analysis

NOTE 1: If milk sampling cannot be performed, use item 4.c (d).

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ATTACHMENT 21

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North Anna Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Sample and Sample Location (NOTE 2)	Collection Frequency	Type and Frequency of Analysis
1. DIRECT RADIATION (NOTE 3)	36 routine monitoring stations, either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, to be placed as follows: 1) An inner ring of stations, one in each emergency meteorological sector within the site boundary 2) An outer ring of stations, one in each emergency meteorological sector within 8 km range from the site 3) The balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations	Quarterly	GAMMA DOSE Quarterly

ATTACHMENT 21

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North Anna Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	**Number of Sample and Sample Location (NOTE 2)**	**Collection Frequency**	**Type and Frequency of Analysis**
2. AIRBORNE			
Radioiodines and Particulates	Samples from 5 locations: a) 3 samples from close to the 3 site boundary locations (in different sectors) of the highest calculated historical annual average ground level D/Q b) 1 sample from the vicinity of a community having the highest calculated annual average ground level D/Q c) 1 sample from a control location 15-40 km distant and in the least prevalent wind direction	Continuous sampler, operation with sample collection weekly	Radioiodine Canister I^{131} Analysis, weekly Particulate Sampler Gross beta radioactivity analysis following filter change; (NOTE 4) Gamma isotopic analysis of composite (by location) quarterly (NOTE 5)
3. WATERBORNE			
a) Surface	Samples from 3 locations: a) 1 sample upstream b) 1 sample downstream c) 1 sample from cooling lagoon	Grab Monthly	Gamma isotopic analysis monthly; (NOTE 5) Composite for tritium analysis quarterly
b) Ground	Sample from 1 or 2 sources only if likely to be affected	Grab Quarterly	Gamma isotopic and tritium analysis quarterly (NOTE 5)
c) Sediment	1 sample from downstream area with existing or potential recreational value	Semi-Annually	Gamma isotopic analysis semi-annually (NOTE 5)

ATTACHMENT 21

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North Anna Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Sample and Sample Location (NOTE 2)	Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a) Milk (NOTE 7)	a) Samples from milking animals in 3 locations within 5 km that have the highest potential. If there are none, then 1 sample from milking animals in each of 3 areas between 5 to 8 km where doses are calculated to be greater than 1 mrem per yr (NOTE 6) b) 1 sample from milking animals at a control location (15-30 km in the least prevalent wind direction)	Monthly at all times	Gamma isotopic (NOTE 5) and I^{131} analysis monthly
b) Fish and Invertebrates	a) 1 sample of commercially and recreationally important species (bass, sunfish, catfish) in vicinity of plant discharge area b) 1 sample of same species in areas not influenced by plant discharge	Semiannually	Gamma isotopic on edible portions
c) Food Products	a) Samples of an edible broad leaf vegetation grown nearest each of two different offsite locations of highest predicted historical annual average ground level D/Q if milk sampling is not performed b) 1 sample of broad leaf vegetation grown 15-30 km in the least prevalent wind direction if milk sampling is not performed	Monthly if available, or at harvest	Gamma isotopic (NOTE 5) and I^{131} analysis

ATTACHMENT 21

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North Anna Radiological Environmental Monitoring Program

NOTE 1: The number, media, frequency, and location of samples may vary from site to site. This table presents an acceptable minimum program for a site at which each entry is applicable. Local site characteristics must be examined to determine if pathways not covered by this table may significantly contribute to an individual's dose and be included in the sampling program.

NOTE 2: For each and every sample location in Attachment 21, specific parameters of distance and direction sector from the centerline of the reactor, and additional description where pertinent, shall be provided in Attachment 23. Refer to Radiological Assessment Branch Technical Positions and to NUREG-0133, Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plant. Deviations are permitted from the required sampling schedule if specimens are unattainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unattainable due to sampling equipment malfunction, every effort shall be made to complete corrective action before 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 6.7.1. 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. In lieu of a Licensee Event Report and pursuant to 6.7.2, identify the cause of the unavailability of samples for that pathway and identify the new locations for obtaining replacement samples in the next Annual Radioactive Effluent Release Report, and include revised figures and tables from the ODCM reflecting the new locations in the report.

ATTACHMENT 21

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North Anna Radiological Environmental Monitoring Program

NOTE 3: 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 36 stations are not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations, e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. 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.

NOTE 4: 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.

NOTE 5: Gamma isotopic analysis is the identification and quantification of gamma-emitting radionuclides that may be attributable to effluents from the facility.

NOTE 6: The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

NOTE 7: If milk sampling cannot be performed, use item 4.c (Page 3 of 5, Attachment 21)

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ATTACHMENT 22

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Surry Environmental Sampling Locations

SAMPLE MEDIA	LOCATION	DISTANCE (MILES)	DIRECTION	REMARKS
Air Charcoal and Particulate	Surry Station (SS)	0.37	NNE	Site Boundary Location at Sector with Highest D/Q
	Hog Island Reserve (HIR)	2.0	NNE	
	Bacons Castle (BC)	4.5	SSW	
	Alliance (ALL)	5.1	WSW	
	Colonial Parkway (CP)	3.7	NNW	
	Dow Chemical (DOW)	5.1	ENE	
	Fort Eustis (FE)	4.8	ESE	
	Newport News (NN)	16.5	ESE	Control Location
Environmental TLDs	Control (00)			Onsite **
	West North West (02)	0.17	WNW	Site Boundary
	Surry Station Discharge (03)	0.6	NW	Site Boundary
	North North West (04)	0.4	NNW	Site Boundary
	North (05)	0.29	N	Site Boundary
	North North East (06)	0.28	NNE	Site Boundary
	North East (07)	0.31	NE	Site Boundary
	East North East (08)	0.43	ENE	Site Boundary
	East (Exclusion) (09)	0.31	E	Onsite
	West (10)	0.40	W	Site Boundary
	West South West (11)	0.45	WSW	Site Boundary
	South West (12)	0.30	SW	Site Boundary
	South South West (13)	0.43	SSW	Site Boundary
	South (14)	0.48	S	Site Boundary
	South South East (15)	0.74	SSE	Site Boundary
	South East (16)	1.00	SE	Site Boundary
	East (17)	0.57	E	Site Boundary
	Station Intake (18)	1.23	ESE	Site Boundary
	Hog Island Reserve (19)	1.94	NNE	Near Resident

ATTACHMENT 22

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Surry Environmental Sampling Locations

SAMPLE MEDIA	LOCATION		DISTANCE (MILES)	DIRECTION	REMARKS
Environmental TLDs	Bacons Castle	(20)	4.45	SSW	Approx. 5 miles
	Route 633	(21)	3.5	SW	Approx. 5 miles
	Alliance	(22)	5.1	WSW	Approx. 5 miles
	Surry	(23)	8.0	WSW	Population Center
	Route 636 and 637	(24)	4.0	W	Approx. 5 miles
	Scotland Wharf	(25)	5.0	WNW	Approx. 5 miles
	Jamestown	(26)	6.3	NW	Approx. 5 miles
	Colonial Parkway	(27)	3.7	NNW	Approx. 5 miles
	Route 617 and 618	(28)	5.2	NNW	Approx. 5 miles
	Kingsmill	(29)	4.8	N	Approx. 5 miles
	Williamsburg	(30)	7.8	N	Population Center
	Kingsmill North	(31)	5.6	NNE	Approx. 5 miles
	Budweiser	(32)	5.7	NNE	Population Center
	Water Plant	(33)	4.8	NE	Approx. 5 miles
	Dow	(34)	5.1	ENE	Approx. 5 miles
	Lee Hall	(35)	7.1	ENE	Population Center
	Goose Island	(36)	5.0	E	Approx. 5 miles
	Fort Eustis	(37)	4.8	ESE	Approx. 5 miles
	Newport News	(38)	16.5	ESE	Population Center
	James River Bridge	(39)	14.8	SSE	Control
	Benn's Church	(40)	14.5	S	Control
	Smithfield	(41)	11.5	S	Control
	Rushmere	(42)	5.2	SSE	Approx. 5 miles
	Route 628	(43)	5.0	S	Approx. 5 miles
Milk	Epp's		4.8	SSW	
	Colonial Parkway		3.7	NNW	
	Judkin's		6.2	SSW	
	William's		22.5	S	Control Location

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Surry Environmental Sampling Locations

SAMPLE MEDIA	LOCATION	DISTANCE (MILES)	DIRECTION	REMARKS
Well Water	Surry Station			Onsite***
	Hog Island Reserve	2.0	NNE	
Crops (Corn, Peanuts, Soybeans)	Slade's Farm	2.4	S	
	Brock's Farm	3.8	S	
Crops (Cabbage, Kale)	Spratley's Garden	3.2	S	
	Carter's Grove Garden	4.8	NE	
	Lucas's Garden			Control Location (Chester, Va.)
River Water (Monthly)	Surry Discharge	0.17	NW	
	Scotland Wharf	5.0	WNW	Control Location
Sediment (Silt)	Chickahominy River	11.2	WNW	Control Location
	Surry Station Intake	1.9	ESE	
	Surry Station Discharge	1.0	NNW	
	Hog Island Point	2.4	NE	
	Point of Shoals	6.4	SSE	

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Surry Environmental Sampling Locations

SAMPLE MEDIA	LOCATION	DISTANCE (MILES)	DIRECTION	REMARKS
Clams	Chickahominy River	11.2	WNW	Control Location
	Surry Station Discharge	1.3	NNW	
	Hog Island Point	2.4	NE	
	Lawne's Creek	2.4	SE	
Oysters	Kingsmill	2.9	NE	
	Mulberry Point	4.9	EESE	
Crabs	Surry Station Discharge	0.6	NW	
Fish	Surry Station Discharge	0.6	NW	
Shoreline Sediment	Hog Island Reserve	0.8	N	

** Onsite Location - in Lead Shield

*** Onsite sample of Well Water—taken from tap-water at Surry Environmental Building

ATTACHMENT 23

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North Anna Environmental Sampling Locations

Distance and Direction From Unit No. 1

Sample Media	Location	Station No.	Distance (Miles)	Direction	Collection Frequency	REMARKS
Environmental TLDs	NAPS Sewage Treatment Plant	01	0.20	NE	Quarterly & Annually	On-Site
	Frederick's Hall	02	5.30	SSW	Quarterly & Annually	
	Mineral, VA	03	7.10	WSW	Quarterly & Annually	
	Wares Crossroads	04	5.10	WSW	Quarterly & Annually	
	Route 752	05	4.20	NNE	Quarterly & Annually	
	Sturgeon's Creek Marina	05A	3.20	N	Quarterly & Annually	
	Levy, VA	06	4.70	ESE	Quarterly & Annually	
	Bumpass, VA	07	7.30	SSE	Quarterly & Annually	
	End of Route 685	21	1.00	WNW	Quarterly & Annually	Site Boundary
	Route 700	22	1.00	WSW	Quarterly & Annually	Site Boundary
	"Aspen Hills"	23	0.93	SSE	Quarterly & Annually	Site Boundary
	Orange, VA	24	22.00	NW	Quarterly & Annually	Control
	Bearing Cooling Tower	N-1/33	0.06	N	Quarterly	On-Site
	Sturgeon's Creek Marina	N-2/34	3.20	N	Quarterly	
	Parking Lot "C"	NNE-3/35	0.24	NNE	Quarterly	On-Site
	Good Hope Church	NNE-4/36	4.96	NNE	Quarterly	
	Parking Lot "B"	NE-5/37	0.20	NE	Quarterly	On-Site
	Bogg's Drive	NE-6/38	1.46	NE	Quarterly	
	Weather Tower Fence	ENE-7/39	0.36	ENE	Quarterly	On-Site
	Route 689	ENE-8/40	2.43	ENE	Quarterly	
	Near Training Facility	E-9/41	0.30	E	Quarterly	On-Site
	"Morning Glory Hill"	E-10/42	2.85	E	Quarterly	
	Island Dike	ESE-11/43	0.12	ESE	Quarterly	On-Site
	Route 622	ESE-12/44	4.70	ESE	Quarterly	

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North Anna Environmental Sampling Locations

Distance and Direction From Unit No. 1

Sample Media	Location	Station No.	Distance (Miles)	Direction	Collection Frequency	REMARKS
Environmental TLDs	Biology Lab	SE-13/45	0.75	SE	Quarterly	On-Site
	Route 701 (Dam Entrance)	SE-14/46	5.88	SE	Quarterly	
	"Aspen Hills"	SSE-15/47	0.93	SSE	Quarterly	Site Boundary
	Elk Creek	SSE-16/48	2.33	SSE	Quarterly	
	NAPS Access Road	S-17/49	0.47	S	Quarterly	On-Site
	Elk Creek Church	S-18/50	1.55	S	Quarterly	
	NAPS Access Road	SSW-19/51	0.42	SSW	Quarterly	On-Site
	Route 618	SSW-20/52	5.30	SSW	Quarterly	
	500KV Tower	SW-21/53	0.60	SW	Quarterly	On-Site
	Route 700	SW-22/54	4.36	SW	Quarterly	
	NAPS Radio Tower	WSW-23/55	0.38	WSW	Quarterly	On-Site
	Route 700	WSW-24/56	1.00	WSW	Quarterly	Site Boundary
	South Gate of Switchyard	W-25/57	0.32	W	Quarterly	On-Site
	Route 685	W-26/58	1.55	W	Quarterly	
	End of Route 685	WNW-27/59	1.00	WNW	Quarterly	Site Boundary
	Route 685	WNW-28/60	1.40	WNW	Quarterly	
	Laydown Area North Gate	NW-29/61	0.45	NW	Quarterly	On-Site
	Lake Anna Campground	NW-30/62	2.54	NW	Quarterly	
	#1/#2 Intake	NNW-31/63	0.07	NNW	Quarterly	On-Site
	Route 208	NNW-32/64	3.43	NNW	Quarterly	
	Bumpass Post Office	C-1/2	7.30	SSE	Quarterly	Control
	Orange, VA	C-3/4	22.00	NW	Quarterly	Control
	Mineral, VA	C-5/6	7.10	WSW	Quarterly	Control
	Louisa, VA	C-7/8	11.54	WSW	Quarterly	Control

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North Anna Environmental Sampling Locations

Distance and Direction From Unit No. 1

Sample Media	Location	Station No.	Distance (Miles)	Direction	Collection Frequency	REMARKS
Airborne Particulate and Radioiodine	NAPS Sewage Treatment Plant	01	0.20	NE	Weekly	On-Site
	Frederick's Hall	02	5.30	SSW	Weekly	
	Mineral, VA	03	7.10	WSW	Weekly	
	Wares Crossroads	04	5.10	WNW	Weekly	
	Route 752	05	4.20	NNE	Weekly	
	Sturgeon's Creek Marina	05A	3.20	N	Weekly	
	Levy, VA	06	4.70	ESE	Weekly	
	Bumpass, VA	07	7.30	SSE	Weekly	
	End of Route 685	21	1.00	WNW	Weekly	Site Boundary
	Route 700	22	1.00	WSW	Weekly	Site Boundary
	"Aspen Hills"	23	0.93	SSE	Weekly	Site Boundary
	Orange, VA	24	22.00	NW	Weekly	Control
Surface Water **[Commitment 3.2.2]**	Waste Heat Treatment Facility (Second Cooling Lagoon)	08	1.10	SSE	Monthly	
	North Anna River (upstream) Rt 669 Bridge (Brook's Bridge)	09A	12.9	WNW	Monthly	Control
	North Anna River (downstream)	11	5.80	SE	Monthly	
Ground Water (well water)	Biology Lab	01A	0.75	SE	Quarterly	
Aquatic Sediment	Waste Heat Treatment Facility (Second Cooling Lagoon)	08	1.10	SSE	Semi-Annually	
	North Anna River (upstream) Rt 669 Bridge (Brook's Bridge)	09A	12.9	WNW	Semi-Annually	Control
	North Anna River (downstream)	11	5.80	SE	Semi-Annually	
Shoreline Soil	Lake Anna (upstream)	09	2.20	NW	Semi-Annually	
Soil	NAPS Sewage Treatment Plant	01	0.20	NE	Once per 3 yrs	On-Site

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North Anna Environmental Sampling Locations

Distance and Direction From Unit No. 1

Sample Media	Location	Station No.	Distance (Miles)	Direction	Collection Frequency	REMARKS
Soil (continued)	Fredericks Hall	02	5.30	SSW	Once per 3 yrs	
	Mineral, VA	03	7.10	WSW	Once per 3 yrs	
	Wares Crossroads	04	5.10	WNW	Once per 3 yrs	
	Route 752	05	4.20	NNE	Once per 3 yrs	
	Sturgeon's Creek Marina	05A	3.20	N	Once per 3 yrs	
	Levy, VA	06	4.70	ESE	Once per 3 yrs	
	Bumpass, VA	07	7.30	SSE	Once per 3 yrs	
	End of Route 685	21	1.00	WNW	Once per 3 yrs	Site Boundary
	Route 700	22	1.00	WSW	Once per 3 yrs	Site Boundary
	"Aspen Hills"	23	0.93	SSE	Once per 3 yrs	Site Boundary
	Orange, VA	24	22.00	NW	Once per 3 yrs	Control
Milk	Holladay Dairy (R.C. Goodwin)	12	8.30	NW	Monthly	
	Terrell's Dairy (Frederick's Hall)	13	5.60	SSE	Monthly	
Fish	Waste Heat Treatment Facility (Second Cooling Lagoon)	08	1.10	SSE	Semi-Annually	
	Lake Orange	25	16.50	NW	Semi-Annually	Control
Food Products (Broad Leaf vegetation)	Route 713	14	varies	NE	Monthly if available, or at harvest	
	Route 614	15	varies	SE		
	Route 629/522	16	varies	NW		Control
	Route 685	21	varies	WNW		
	"Aspen Hills" Area	23	varies	SSE		

ATTACHMENT 24

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Detection Capabilities for Surry Environmental Sample Analysis

LOWER LIMIT OF DETECTION (LLD)

Analysis (NOTE 2)	**Water (pCi/l)**	**Airborne Particulate or Gases (pCi/m^3)**	**Fish (pCi/kg) (wet)**	**Milk (pCi/l)**	**Food Products (pCi/kg) (wet)**	**Sediment (pCi/kg) (wet)**
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-95	30					
Nb-95	15					
I-131	(NOTE 3) 1	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-140	60			60		
La-140	15			15		

NOTE 1: Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

NOTE 2: This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

NOTE 3: LLD for the Ground (drinking) Water Samples. The LLD for the surface (non-drinking) water samples is 10 pCi/l.

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Detection Capabilities for Surry Environmental Sample Analysis

LOWER LIMIT OF DETECTION (LLD)

NOTE 1: For a particular measurement system (which may include radiochemical separation):

$$\mathrm{LLD} = \frac{4.66\ s_b}{E \bullet V \bullet 2.22E{+}06 \bullet Y \bullet e^{-(\lambda \Delta t)}} \qquad (24\text{-}1)$$

Where:

LLD	=	the "a priori" (before the fact) Lower Limit of Detection as defined above (as microcuries per unit mass or volume) (see 4.8)
s_b	=	the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute, cpm)
E	=	the counting efficiency (as counts per disintegration)
V	=	the sample size (in units of mass or volume)
2.22E+06	=	the number of disintegrations per minute (dpm) per microcurie
Y	=	the fractional radiochemical yield (when applicable)
λ	=	the radioactive decay constant for the particular radionuclide
Δt	=	the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples)

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

The LLD is an "a priori" (before the fact) limit representing the capability of a measurement system and not a "posteriori" (after the fact) limit for a particular measurement.

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Detection Capabilities for North Anna Environmental Sample Analysis

LOWER LIMIT OF DETECTION (LLD)

Analysis (NOTE 2)	**Water (pCi/l)**	**Airborne Particulate or Gases (pCi/m^3)**	**Fish (pCi/kg) (wet)**	**Milk (pCi/l)**	**Food Products (pCi/kg) (wet)**	**Sediment (pCi/kg) (wet)**
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-95	30					
Nb-95	15					
I-131	(NOTE 3) 1	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-140	60			60		
La-140	15			15		

NOTE 1: Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

NOTE 2: This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

NOTE 3: LLD for the ground (drinking) water samples. The LLD for the surface (non-drinking) water samples is 10 pCi/l.

ATTACHMENT 25

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Detection Capabilities for North Anna Environmental Sample Analysis

LOWER LIMIT OF DETECTION (LLD) (NOTE 3)

NOTE 3: For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \bullet V \bullet 2.22E{+}06 \bullet Y \bullet e^{-(\lambda \Delta t)}} \qquad (25\text{-}1)$$

Where:

LLD	=	the "a priori" (before the fact) Lower Limit of Detection as defined above (as microcuries per unit mass or volume) (see 4.9)
s_b	=	the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute, cpm)
E	=	the counting efficiency (as counts per disintegration)
V	=	the sample size (in units of mass or volume)
2.22E+06	=	the number of disintegrations per minute (dpm) per microcurie
Y	=	the fractional radiochemical yield (when applicable)
λ	=	the radioactive decay constant for the particular radionuclide
Δt	=	the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples)

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

The LLD is an "a priori" (before the fact) limit representing the capability of a measurement system and not a "posteriori" (after the fact) limit for a particular measurement.

ATTACHMENT 26

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Reporting Levels for Radioactivity Concentrations in Environmental Samples at Surry

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	30,000				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	(NOTE 1) 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	

NOTE 1: Reporting level for the ground (drinking) water samples required by Attachment 20. The reporting level for the surface (non-drinking) water samples required by Attachment 20 is 20 pCi/l.

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ATTACHMENT 27

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Reporting Levels for Radioactivity Concentrations in Environmental Samples at North Anna

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	(NOTE 1) 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	

NOTE 1: For drinking water samples

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ATTACHMENT 28

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Surry Meteorological, Liquid, and Gaseous Pathway Analysis

1.0 METEORLOGICAL ANALYSIS

1.1 Purpose

The purpose of the meteorological analysis was to determine the annual average χ/Q and D/Q values at critical locations around the Station for ventilation vent (ground level) and process vent (mixed mode) releases. The annual average χ/Q and D/Q values were used in a dose pathway analysis to determine both the maximum exposed individual at site boundary and member of the public. The χ/Q and D/Q values resulting in the maximum exposures were incorporated into the dose factors in Attachments 12 and 18.

1.2 Meteorological Data, Parameters, and Methodology

Onsite meteorological data for the period January 1, 1979, through December 31, 1981, were used in calculations. These data included wind speed, wind direction, and differential temperature for the purpose of determining joint frequency distributions for those releases characterized as ground level (i.e., ventilation vent), and those characterized as mixed mode (i.e., process vent). The portions of release characterized as ground level were based on $\Delta T_{158.9ft-28.2ft}$ and 28.2 foot wind data, and the portions characterized as mixed mode were based on $\Delta T_{158.9ft-28.2ft}$ and 158.9 ft wind data.

χ/Qs and D/Qs were calculated using the NRC computer code "XOQDOQ - Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations", September, 1977. The code is based upon a straight line airflow model implementing the assumptions outlined in Section C (excluding C1a and C1b) of Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors."

The open terrain adjustment factors were applied to the χ/Q values as recommended in Regulatory Guide 1.111. The site region is characterized flat terrain such that open terrain correction factors are considered appropriate. The ground level ventilation vent release calculations included a building wake correction based on a 1516 m^2 containment minimum cross-sectional area. The effective release height used in mixed mode release calculations was based on a process vent release height of 131 ft, and plume rise due to momentum for a vent diameter of 3 in. with plume exit velocity of 100 ft/sec.

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Surry Meteorological, Liquid, and Gaseous Pathway Analysis

Ventilation vent, and vent releases other than from the process vent, are considered ground level as specified in Regulatory Guide 1.111 for release points less than the height of adjacent solid structures. Terrain elevations were obtained from Surry Power Station Units 1 and 2 Virginia Electric and Power Company Updated Final Safety Analysis Report Table 11A-8.

χ/Q and D/Q values were calculated for the nearest site boundary, resident, milk cow, and vegetable garden by sector for process vent and ventilation vent releases. χ/Q values were also calculated for the nearest discharge canal bank for process and ventilation vent releases.

According to the definition for short term in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Stations," October, 1978, some gaseous releases may fit this category, primarily waste gas decay tank releases and containment purges. However, these releases are considered long term for dose calculations as past releases were both random in time of day and duration as evidenced by reviewing past release reports. Therefore, the use of annual average concentrations is appropriate according to NUREG-0133.

1.3 Results

The χ/Q value that resulted in the maximum total body, skin, and inhalation exposure for ventilation vent releases was 6.0E-05 sec/m^3 at a site boundary location 499 meters N sector. For process vent releases, the site boundary χ/Q value was 1.0E-06 sec/m^3 at a location 644 meters S sector. The discharge canal bank χ/Q value that resulted in the maximum inhalation exposure for ventilation vent releases was 7.8E-05 sec/m^3 at a location 290 meters NW sector. The discharge canal bank χ/Q value for process vent was 1.6E-06 sec/m^3 at a location 290 meters NW sector.

Pathway analysis indicated that the maximum exposure from I^{131}, I^{133}, and from all radionuclides in particulate form with half-lives greater than 8 days, was through the grass-cow-milk pathway. The D/Q value from ventilation vent releases resulting in the maximum exposure was 9.0E-10 per m^2 at a location 5150 meters S sector. For process vent releases, the D/Q value was 4.3E-10 per m^2 at a location 5150 meters S sector. For tritium, the χ/Q value from ventilation vent releases resulting in the maximum exposure for the milk pathway was 3.0E-07 sec/m^3, and 1.3E-07 sec/m^3 for process vent releases at a location 5150 meters S sector. The inhalation pathway is the only other pathway existing at this location. Therefore, the χ/Q values given for tritium also apply for the inhalation pathway.

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Surry Meteorological, Liquid, and Gaseous Pathway Analysis

2.0 LIQUID PATHWAY ANALYSIS

2.1 Purpose

The purpose of the liquid pathway analysis was to determine the maximum exposed member of the public in unrestricted areas as a result of radioactive liquid effluent releases. The analysis included a determination of most restrictive liquid pathway, most restrictive age group, and critical organ. This analysis is required for Subsection 6.2, Liquid Radioactive Waste Effluents.

2.2 Data, Parameters, and Methodology

Radioactive liquid effluent release data for the years 1976, 1977, 1978, 1979, 1980, and 1981 were compiled from the Surry Power Station effluent release reports. The data for each year, along with appropriate site specific parameters and default selected parameters, were entered into the NRC computer code LADTAP as described in NUREG-0133.

Liquid radioactive effluents from both units are released to the James River via the discharge canal. Possible pathways of exposure for release from the Station include ingestion of fish and invertebrates and shoreline activities. The irrigated food pathway and potable water pathway do not exist at this location. Access to the discharge canal by the general public is gained two ways: bank fishing, controlled by the Station and limited to Virginia Power employees or guests of employees, and by boat as far upstream as the inshore end of the discharge canal groin. It has been estimated that boat sport fishing would be performed a maximum of 800 hours per year, and that bank fishing would be performed a maximum of 160 hours per year.

For an individual fishing in the discharge canal, no river dilution was assumed for the fish pathway. For an individual located beyond the discharge canal groins, a river dilution factor of 5 was assumed as appropriate according to Regulatory Guide 1.109, Rev. 1, and the fish, invertebrate, and shoreline pathways were considered to exist. Dose factors, bioaccumulation factors, and shore width factors given in Regulatory Guide 1.109, Rev. 1, and in LADTAP were used, as were usage terms for shoreline activities and ingestion of fish and invertebrates. Dose to an individual fishing on the discharge bank was determined by multiplying the annual dose calculated with LADTAP by the fractional year the individual spent fishing in the canal.

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Surry Meteorological, Liquid, and Gaseous Pathway Analysis

2.3 Results

For the years 1976, 1977, 1979, 1980, and 1981, the invertebrate pathway resulted in the largest dose. In 1978 the fish pathway resulted in the largest dose. The maximum exposed member of the public was determined to utilize the James River. The critical age group was the adult and the critical organ was either the thyroid or GI-LLI. The ingestion dose factor, A_i, in 6.2.3 includes the fish and invertebrate pathways. A_i dose factors were calculated for the total body, thyroid, and GI-LLI organs.

3.0 GASEOUS PATHWAY ANALYSIS

3.1 Purpose

A gaseous effluent pathway analysis was performed to determine the location that would result in the maximum doses due to noble gases, for use in demonstrating compliance with 6.3.1.a. and 6.3.3.a. The analysis also included a determination of the location, pathway, and critical organ, of the maximum exposed member of the public, as a result of the release of I^{131}, I^{133}, tritium, and for all radionuclides in particulate form with half-lives greater than eight days for use in demonstrating compliance with 6.3.4.a. In addition, the analysis included a determination of the critical organ, maximum age group, and sector location of an exposed individual through the inhalation pathway from I^{131}, I^{133}, tritium, and particulates to demonstrate compliance with 6.3.1.a..

3.2 Data, Parameters, and Methodology

Annual average χ/Q values were calculated, as described in Section 1 of this attachment, for the nearest site boundary in each directional sector and at other critical locations accessible to the public inside site boundary. The largest χ/Q value was determined to be 6.0E-05 sec/m^3 at site boundary for ventilation vent releases at a location 499 meters N direction, and 1.0E-06 sec/m^3 at site boundary for process vent releases at a location 644 meters S direction. The maximum doses to total body and skin, and air doses for gamma and beta radiation due to noble gases would be at these site boundary locations. The doses from both release points are summed in calculations to calculate total maximum dose.

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Surry Meteorological, Liquid, and Gaseous Pathway Analysis

Step 6.3.1.a.2 dose limits apply specifically to the inhalation pathway. Therefore, the locations and χ/Q values determined for maximum noble gas doses can be used to determine the maximum dose from I^{131}, I^{133}, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days for the inhalation pathway.

The NRC computer code GASPAR, "Evaluation of Atmospheric Releases," Revised 8/19/77, was run using 1976, 1977, 1978, 1979, 1980, and 1981 Surry Power Station gaseous effluent release report data. Doses from I^{131}, I^{133}, tritium, and particulates for the inhalation pathway were calculated using the 6.0E-05 sec/m^3 site boundary χ/Q. Except for the source term data and the χ/Q value, computer code default parameters were used. Results for each year indicated that the critical age group was the child and the critical organ was the thyroid for the inhalation pathway. In 1979, the teen was the critical age group. However, the dose calculated for the teen was only slightly greater than for the child and the doses could be considered equivalent.

The gamma and beta dose factors K_{ivv}, L_{ivv}, M_{ivv}, and N_{ivv} in Attachment 12 were obtained by performing a units conversion of the appropriate dose factors from Table B-1, Regulatory Guide 1.109, Rev. 1, to mrem/yr per Ci/m^3 or mrad/yr per Ci/m^3, and multiplying by the ventilation vent site boundary χ/Q value of 6.0E-05 sec/m^3. The same approach was used to calculate the gamma and beta dose factors K_{ipv}, L_{ipv}, M_{ipv}, and N_{ipv} in Attachment 12, using the process vent site boundary χ/Q value of 1.0E-06 sec/m^3.

Inhalation pathway dose factors P_{ivv} and P_{ipv} in Attachment 12 were calculated using the equation:

$$P_i = K'(BR)\,DFA_i(\chi/Q)(\text{mrem/yr per Curie/sec}) \qquad (28\text{-}1)$$

where:

K' = a constant of unit conversion, 1E+12 pCi/Ci

BR = the breathing rate of the child age group, 3700 m^3/yr, from Table E-5, Regulatory Guide 1.109, Rev.1

DFA_i = the thyroid organ inhalation dose factor for child age group for the ith radionuclide, in mrem/pCi, from Table E-9, Regulatory Guide 1.109, Rev. 1

χ/Q = the ventilation vent site boundary χ/Q, 6.0E-5 sec/m^3, or the process vent site boundary χ/Q, 1.0E-06 sec/m^3, as appropriate

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Surry Meteorological, Liquid, and Gaseous Pathway Analysis

Step 6.3.4.a., requires that the dose to the maximum exposed member of the public from I^{131}, I^{133}, tritium, and from all radionuclides in particulate form with half-lives greater than 8 days be less than or equal to the specified limits. Dose calculations were performed for an exposed member of the public within site boundary unrestricted areas, discharge canal bank, and to an exposed member of the public beyond site boundary at real residences with the largest χ/Q values using the NRC computer code GASPAR. Doses to members of the public were also calculated for the vegetable garden, meat animal, and milk-cow pathways with the largest D/Q values using the NRC computer code GASPAR.

It was determined that the member of the public within site boundary would be using the discharge canal bank for fishing a maximum of 160 hours per year. The maximum annual χ/Q at this location was determined to be 7.8E-05 sec/m^3 at 290 meters NW direction. After applying a correction for the fractional part of year an individual would be fishing at this location, the dose was calculated to be less than an individual would receive at site boundary.

The member of the public receiving the largest dose beyond site boundary was determined to be located 5150 meters S sector. The critical pathway was the grass-cow-milk, the maximum age group was the infant, and the critical organ the thyroid. For each year 1976, 1977, 1978, 1979, 1980 and 1981 the dose to the infant from the grass-cow-milk pathway was greater than the dose to the member of the public within site boundary, nearest residence, vegetable or meat pathways. Therefore, the maximum exposed member of the public was determined to be the infant, exposed through the grass-cow-milk pathway, critical organ thyroid, at a location 5150 meters S sector. The only other pathway existing at this location for the infant is inhalation.

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Surry Meteorological, Liquid, and Gaseous Pathway Analysis

The RM_{ivv} and RM_{ipv} dose factors, except for tritium, in Attachment 18 were calculated by multiplying the appropriate D/Q value with the following equation:

$$RM_i = K' \frac{Q_F(U_{ap})}{\lambda_i + \lambda_w} F_m \,(r)\,(DFL_i) \left[\frac{f_p f_s}{Y_p} + \frac{(1 - f_p f_s)\, e^{-\lambda_i t_h}}{Y_s} \right] e^{-\lambda_i t_f} \qquad (28\text{-}2)$$

where:

K'	=	a constant of unit conversion, 1E+12 pCi/Ci
Q_F	=	cow's consumption rate, 50, in Kg/day (wet weight)
U_{ap}	=	infant milk consumption rate, 330, liters/yr
Y_p	=	agricultural productivity by unit area of pasture feed grass, 0.7 Kg/m^2
Y_s	=	agricultural productivity by unit area of stored feed, 2.0, in Kg/m^2
F_m	=	stable element transfer coefficients, from Table E-1, Regulatory Guide 1.109, Rev. 1
r	=	fraction of deposited activity retained on cow's feed grass, 1.0 for radioiodine, and 0.2 for particulates
DFL_i	=	thyroid ingestion dose factor for the ith radionuclide for the infant, in mrem/pCi, from Table E-14, Regulatory Guide 1.109, Rev.1
λ_i	=	decay constant for the ith radionuclide, in sec^{-1}, from Table of Isotopes, Lederer, Hollander, and Perlman, sixth Edition.
λ_w	=	decay constant for removal of activity of leaf and plant surfaces by weathering, 5.73E-07 sec^{-1} (corresponding to a 14 day half-life)
t_f	=	transport time from pasture to cow, to milk, to receptor, 1.73+05, in seconds
t_h	=	transport time from pasture, to harvest, to cow, to milk, to receptor, 7.78E+06, in seconds
f_p	=	fraction of year that cow is on pasture, 0.67 (dimensionless), 7.78E+06 in seconds
f_s	=	fraction of cow feed that is pasture grass while cow is on pasture, 1.0, dimensionless

Parameters used above were obtained from NUREG-0133 and Regulatory Guide 1.109, Rev.1.

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Since the concentration of tritium in milk is based on the airborne concentration rather than the deposition, the following equation is used:

$$R_{H^3} = K'K'''F_m Q_F U_{ap}(DFL_{H^3})\left[0.75(0.5/H)\right]\chi/Q \qquad (28\text{-}3)$$

where:

K''' = a constant of unit conversion 1E+03 gm/kg

H = absolute humidity of the atmosphere, 8.0, gm/m^3

0.75 = the fraction of total feed that is water

0.5 = the ratio of the specific activity of the feed grass to the atmospheric water

χ/Q = the annual average concentration at a location 5150 meters S sector, 3.0E-07 sec/m^3 for ventilation vent releases, and 1.3E-07 sec/m^3 for the process vent releases

Other parameters have been previously defined.

The inhalation pathway dose factors RI_{ivv} and RI_{ipv} in Attachment 18 were calculated using the following equation:

$$RI_i = K'(BR)\,DFA_i(\chi/Q)\ \text{(mrem/yr per Curie/sec)} \qquad (28\text{-}4)$$

where:

K' = a constant of unit conversion, 1E+12 pCi/Ci

BR = breathing rate of the infant age group, 1400 m^3/yr, from Table E-5, Regulatory Guide 1.109, Rev.1

DFA_i = thyroid organ inhalation dose factor for infant age group for the ith radionuclide, in mrem/pCi, from Table E-10, Regulatory Guide 1.109, Rev.1

χ/Q = ventilation vent χ/Q, 3.0E-07 sec/m^3, or the process vent site boundary χ/Q, 1.3E-07 sec/m^3, at a location 5150 meters S sector.

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North Anna Meteorological, Liquid, and Gaseous Pathway Analysis

1.0 METEOROLOGICAL ANALYSIS

1.1 Purpose

The purpose of the meteorological analysis was to determine the annual average χ/Q and D/Q values at critical locations around the Station for ventilation vent (ground level) and process vent (mixed mode) releases. The annual average χ/Q and D/Q values were used to perform a dose pathway analysis to determine both the maximum exposed individual at site boundary and member of the public. The χ/Q and D/Q values resulting in the maximum exposures were incorporated into the dose factors in Attachments 13 and 19.

1.2 Meteorological Data, Parameters, and Methodology

Onsite meteorological data for the period January 1, 1981, through December 31, 1981, were used in calculations. These data included wind speed, wind direction, and differential temperature for the purpose of determining joint frequency distributions for those releases characterized as ground level (e.g., ventilation vent), and those characterized as mixed mode (i.e., process vent). The portions of release characterized as ground level were based on $\Delta T_{158.9ft\text{-}28.2ft}$ and 28.2 foot wind data, and the portions characterized as mixed mode were based on $\Delta T_{158.9ft\text{-}28.2ft}$ and 158.9 ft wind data.

χ/Q's and D/Q's were calculated using the NRC computer code "XOQDOQ - Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations," September, 1977. The code is based upon a straight line airflow model implementing the assumptions outlined in Section C (excluding C1a and C1b) of Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors."

The open terrain adjustment factors were applied to the χ/Q values as recommended in Regulatory Guide 1.111. The site region is characterized by gently rolling terrain so open terrain correction factors were considered appropriate. The ground level ventilation vent release calculations included a building wake correction based on a 1516 m^2 containment minimum cross-sectional area.

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North Anna Meteorological, Liquid, and Gaseous Pathway Analysis

The effective release height used in mixed mode release calculations was based on a process vent release height of 157.5 ft, and plume rise due to momentum for a vent diameter of 3 in. with plume exit velocity of 100 ft/sec. Ventilation vent, and vent releases other than from the process vent, are considered ground level as specified in Regulatory Guide 1.111 for release points less than the height of adjacent solid structures. Terrain elevations were obtained from North Anna Power Station Units 1 and 2, Virginia Electric and Power Company Final Safety Analysis Report Table 11C.2-8.

χ/Q and D/Q values were calculated for the nearest site boundary, resident, milk cow, and vegetable garden by sector for process vent and ventilation vent releases at distances specified from North Anna Power Station Annual Environmental Survey Data for 1981. χ/Q values were also calculated for the nearest lake shoreline by sector for the process vent and ventilation vent releases.

According to the definition for short term in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Stations," October, 1978, some gaseous releases may fit this category, primarily waste gas decay tank releases and containment purges. However, these releases are considered long term for dose calculations as past releases were both random in time of day and duration as evidenced by reviewing past release reports. Therefore, the use of annual average concentrations is appropriate according to NUREG-0133.

The χ/Q and D/Q values calculated from 1981 meteorological data are comparable to the values presented in the North Anna Power Station UFSAR.

1.3 Results

The χ/Q value that resulted in the maximum total body, skin and inhalation exposure for ventilation vent releases was 9.3E-06 sec/m^3 at a site boundary location 1416 meters SE sector. For process vent releases, the site boundary χ/Q value was 1.2E-06 sec/m^3 at a location 1513 meters S sector. The shoreline χ/Q value that resulted in the maximum inhalation exposure for ventilation vent releases was 1.0E-04 sec/m^3 at a location 274 meters NNE sector. The shoreline χ/Q value for process vent was 2.7E-06 sec/m^3 at a location 274 meters NNE sector.

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North Anna Meteorological, Liquid, and Gaseous Pathway Analysis

Pathway analysis indicated that the maximum exposure from I-131, I^{133}, and from all radionuclides in particulate form with half-lives greater than 8 days was through the grass-cow-milk pathway. The D/Q value from ventilation vent releases resulting in the maximum exposure was 2.4E-09 per m^2 at a location 3250 meters N sector. For process vent releases, the D/Q value was 1.1E-09 per m^2 at a location 3250 meters N sector. For tritium, the χ/Q value from ventilation vent releases resulting in the maximum exposure for the milk pathway was 7.2E-07 sec/m^3, and 3.9E-07 sec/m^3 for process vent releases at a location 3250 meters N sector.

2.0 LIQUID PATHWAY ANALYSIS

2.1 Purpose

The purpose of the liquid pathway analysis was to determine the maximum exposed member of the public in unrestricted areas as a result of radioactive liquid effluent releases. The analysis includes a determination of most restrictive liquid pathway, most restrictive age group, and critical organ. This analysis is required for Subsection 6.2.

2.2 Data, Parameters, and Methodology

Radioactive liquid effluent release data for the years 1979, 1980, and 1981 were compiled from the North Anna Power Station semi-annual effluent release reports. The data for each year, along with appropriate site specific parameters and default selected parameters, were entered into the NRC computer code LADTAP as described in NUREG-0133.

Re-concentration of effluents using the small lake connected to larger water body model was selected with the appropriate parameters determined from Table 3.5.3.5, Design Data for Reservoir and Waste Heat Treatment Facility from Virginia Electric and Power Company, Applicant's Environmental Report Supplement, North Anna Power Station, Units 1 and 2, March 15, 1972. Dilution factors for aquatic foods, shoreline, and drinking water were set to one. Transit time calculations were based on average flow rates. All other parameters were defaults selected by the LADTAP computer code.

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2.3 Results

For each year, the fish pathway resulted in the largest dose. The critical organ each year was the liver, and he adult and teenage age groups received the same organ dose. However, since the adult total body dose was greater than the teen total body dose for each year, the adult was selected as the most restrictive age group. Dose factors in Attachment 7 are for the maximum exposed member of the public, an adult, with the critical organ being the liver.

3.0 GASEOUS PATHWAY ANALYSIS

3.1 Purpose

A gaseous effluent pathway analysis was performed to determine the location that would result in the maximum doses due to noble gases for use in demonstrating compliance with 6.3.1.a. and 6.3.3.a. The analysis also included a determination of the critical pathway, location of maximum exposed member of the public, and the critical organ for the maximum dose due to I^{131}, I^{133}, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days for use in demonstrating compliance with requirements in 6.3.4.a.1 . In addition, the analysis included a determination of the critical pathway, maximum age group, and sector location of an exposed individual through the inhalation pathway from I^{131}, I^{133}, tritium, and particulates with half-lives greater than 8 days to demonstrate compliance with 6.3.1.a..

3.2 Data, Parameters, and Methodology

Annual average χ/Q values were calculated, as described in Section 1 of this attachment, for the nearest site boundary in each directional sector and at other critical locations beyond the site boundary. The largest χ/Q value was determined to be 9.3E-06 sec/m^3 at site boundary for ventilation vent releases at a location 1416 meters SE direction, and 1.2E-06 sec/m^3 at site boundary for process vent releases at a location 1513 meters S direction. The maximum doses to total body and skin, and air doses for gamma and beta radiation due to noble gases, would be at these site boundary locations. The doses from both release points are summed in calculations to calculate total maximum dose.

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North Anna Meteorological, Liquid, and Gaseous Pathway Analysis

Step 6.3.1.a.2 dose limits apply specifically to the inhalation pathway. Therefore, the locations and χ/Q values determined for maximum noble gas doses can be used to determine the maximum dose from I^{131}, I^{133}, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days for the inhalation pathway.

The NRC computer code GASPAR, "Evaluation of Atmospheric Releases," Revised 8/19/77, was run using 1979, 1980 and 1981 North Anna Power Station Gaseous Effluent Release Report data. Doses from I^{131}, I^{133}, tritium, and particulates for the inhalation pathway were calculated using the 9.3E-06 sec/m^3 site boundary χ/Q. Except for the source term data and the χ/Q value, computer code default parameters were used. Results for each year indicated that the critical age group was the child and the critical organ was the thyroid for the inhalation pathway.

The gamma and beta dose factors K_{ivv}, L_{ivv}, M_{ivv}, and N_{ivv} in Attachment 13 were obtained by performing a units conversion of the appropriate dose factors from Table B-1, Regulatory Guide 1.109, Rev. 1, to mrem/yr per Ci/m^3 or mrad/yr per Ci/m^3, and multiplying by the ventilation vent site boundary χ/Q value of 9.3E-06 sec/m^3. The same approach was used in calculating the gamma and beta dose factors K_{ipv}, L_{ipv}, M_{ipv}, and N_{ipv} in Attachment 13 using the process vent site boundary χ/Q value of 1.2E-06 sec/m^3.

The inhalation pathway dose factors P_{ivv} and P_{ipv} in Attachment 13 were calculated using the following equation:

$$P_i = K'(BR)\, DFA_i\, (\chi/Q) \quad \text{(mrem/yr per Curie/sec)} \tag{29-1}$$

where:

K' = a constant of unit conversion, 1E+12 pCi/Ci

BR = the breathing rate of the child age group, 3700 m^3/yr, from Table E-5, Regulatory Guide 1.109, Rev.1

DFA_i = the thyroid organ inhalation dose factor for child age group for the ith radionuclide, in mrem/pCi, from Table E-9, Regulatory Guide 1.109, Rev. 1

χ/Q = the ventilation vent site boundary χ/Q, 9.3E-06 sec/m^3, or the process vent site boundary χ/Q, 1.2E-06 sec/m^3, as appropriate.

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North Anna Meteorological, Liquid, and Gaseous Pathway Analysis

Step 6.3.4.a., requires that the dose to the maximum exposed member of the public from I^{131}, I^{133}, tritium, and from all radionuclides in particulate form with half-lives greater than 8 days be less than or equal to the specified limits. Dose calculations were performed for an exposed member of the public within site boundary unrestricted areas, and to an exposed member of the public beyond site boundary at locations identified in the North Anna Power Station Annual Environmental Survey Data for 1981.

It was determined that the member of the public within site boundary would be using Lake Anna for recreational purposes a maximum of 2232 hours per year. It is assumed that this member of the public would be located the entire 2232 hours at the lake shoreline with the largest annual χ/Q of 1.0E-04 at a location 274 meters NNE sector. The NRC computer code GASPAR was run to calculate the inhalation dose to this individual. The GASPAR results were corrected for the fractional year the member of the public would be using the lake.

Using the NRC computer code GASPAR and annual average χ/Q and D/Q values obtained as described in Section 1 of this attachment, the member of the public receiving the largest dose beyond site boundary was determined to be located 3250 meters N sector. The critical pathway was the grass-cow-milk, the maximum age group was the infant, and the critical organ the thyroid. For each year 1979, 1980, and 1981 the dose to the infant from the grass-cow-milk pathway was greater than the dose to the member of the public within site boundary. Therefore, the maximum exposed member of the public was determined to be the infant, exposed through the grass-cow-milk pathway, critical organ thyroid, at a location 3250 meters N sector.

Pathway analysis results indicate that existing pathways, including ground and inhalation, within five miles of North Anna Power Station, yield R_i dose factors less than those determined for the cow-milk pathway. Although the cow-milk pathway does not exist within five miles of the Station, NUREG-0133 requires the use of cow-milk R_i dose factors since these values result in the most limiting doses. There is no requirement to include the other pathways.
[Commitment 3.2.3]

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North Anna Meteorological, Liquid, and Gaseous Pathway Analysis

The RM_{ivv} and RM_{ipv} dose factors, except for tritium, in Attachment 19 were calculated by multiplying the appropriate D/Q value with the following equation:

$$RM_i = K' \frac{Q_F(U_{ap})}{\lambda_i + \lambda_w} F_m \,(r)\,(DFL_i)\left[\frac{f_p f_s}{Y_p} + \frac{(1 - f_p f_s)\, e^{-\lambda_i t_h}}{Y_s}\right] e^{-\lambda_i t_f} \qquad (29\text{-}2)$$

where:

K' = a constant of unit conversion, 1E+12 pCi/Ci

Q_F = cow's consumption rate, 50, in Kg/day (wet weight)

U_{ap} = infant milk consumption rate, 330 liters/yr

Y_p = agricultural productivity by unit area of pasture feed grass, 0.7 Kg/m^2

Y_s = agricultural productivity by unit area of stored feed, 2.0, in Kg/m^2

F_m = stable element transfer coefficients, from Table E-1, Regulatory Guide 1.109, Rev. 1

r = fraction of deposited activity retained on cow's feed grass, 1.0 for radioiodine, and 0.2 for particulates

DFL_i = thyroid ingestion dose factor for the ith radionuclide for the infant, in mrem/pCi, from Table E-14, Regulatory Guide 1.109, Rev. 1

λ_i = decay constant for the ith radionuclide, in sec^{-1}, from Table of Isotopes, Lederer, Hollander, and Perlman, sixth Edition.

λ_w = decay constant for removal of activity of leaf and plant surfaces by weathering, 5.73E-07 sec^{-1} (corresponding to a 14 day half-life)

t_f = transport time from pasture to cow, to milk, to receptor, 1.73E+05, in seconds

t_h = transport time from pasture, to harvest, to cow, to milk, to receptor, 7.78E+06, in seconds

f_p = fraction of year that cow is on pasture, 0.58 (dimensionless), 7 months per year from NUREG-0597

f_s = fraction of cow feed that is pasture grass while cow is on pasture, 1.0, dimensionless

Parameters used in the above equation were obtained from NUREG-0133 and Regulatory Guide 1.109, Rev.1.

ATTACHMENT 29

(Page 8 of 8)

North Anna Meteorological, Liquid, and Gaseous Pathway Analysis

Since the concentration of tritium in milk is based on the airborne concentration rather than the deposition, the following equation is used:

$$RM_{H^3} = K'K'''F_m Q_F U_{ap}(DFL_{H^3})\left[0.75(0.5/H)\right]\chi/Q \qquad (29\text{-}3)$$

where:

K''' = a constant of unit conversion 1E+03 gm/kg

H = absolute humidity of the atmosphere, 8.0, gm/m^3

0.75 = the fraction of total feed that is water

0.5 = the ratio of the specific activity of the feed grass to the atmospheric water

χ/Q = the annual average concentration at a location 3250 meters N sector, 7.2E-07 sec/m^3 for ventilation vent releases, and 3.9E-07 sec/m^3 for the process vent releases

Other parameters have been previously defined.

ATTACHMENT 4

MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS

(01/95 - 12/95)

As required by the ODCM, Section 6.7.2.a.4, major changes to radioactive liquid, gaseous and solid waste treatment systems for the time period covered by this report are synopsized in this attachment. Supporting information as to the reason(s) for the change(s) and a summary of the 10 CFR 50.59 evaluations are included, as applicable.

The high capacity blowdown system was installed in Unit 1 in September 1995 and in Unit 2 in December 1995. It is used in conjunction with the chemical feed of the feedwater system to control steam generator chemistry.

Each unit's high capacity blowdown system consists of a flash tank, a blowdown cooler, a radiation monitor and associated piping and valves. The system discharges to the Circulating Water Discharge Tunnel through an existing discharge point.

The high capacity blowdown system is automatically terminated in the event of High-High flash tank level, High flash tank pressure, or high condenser pressure. It is also automatically terminated on a containment isolation signal or a radiation monitor High-High alarm.

In the event of a high capacity steam generator radiation monitor alarm procedures require that a sample be taken and analyzed for principal gamma emitters and tritium. If the presence of any primary-to-secondary leakage is confirmed, the release via the high capacity blowdown will be terminated, and blowdown will be diverted through the normal blowdown pathway.

The high capacity blowdown system is described in NAPS UFSAR Section 10.4.6, Secondary Vent and Drain Systems.

VIRGINIA POWER **INFORMATION ONLY**

1. Safety Evaluation Number	2. Applicable Station	3. Applicable Unit
95-SE-MOD-58	[X] North Anna Power Station [ ] Surry Power Station	[X] Unit 1 [ ] Unit 2 [ ] Unit 1 [ ] Unit 2

PART A - Resolution Summary Report

4. List the governing documents for which this safety evaluation was performed.

DC-94-003, Steam Generator Blowdown System Upgrades, North Anna Unit 1

5. Summarize the change, test, or experiment evaluated.

See Page 1A and 1B of 12.

6. State the purpose for this change, test, or experiment.

This change makes the existing abandoned high capacity S/G BD system operational without the requirement of continuous condensate polishing for the secondary side of the Unit. The existing control instrumentation will be replaced and the drains from the BD flash tank will be routed through a heat exchanger to recover heat and cool the effluent prior to discharge into the circulating water discharge tunnel. The original design routed the flash tank drains to the condenser hot well. The original design concept was to have the impurities in the blowdown removed by the condensate polishing system. Since the condensate polishing system is not continuously operated at power, the original high capacity blowdown system, operated in the Unit's current operational configuration without continuous condensate polishing, would result in zero net blowdown.

7. List all limiting conditions and special requirements identified or assumed by this safety analysis. For each item, indicate the formal tracking mechanism that will be used to ensure that these conditions and/or requirements will be met.

The installation of all the upgrades to the S/G BD equipment and piping, except for the final tie-in of the BD flash tank drains cooler condensate return line divert to the condenser, including the removal of the previously installed tie-in stub weld caps, may be performed during any mode of operation with the modification being planned during non-outage time. Installation of the BD cooler condensate return line divert to the condenser will require a Unit outage, mode 5 or 6.

Existing fire barriers from the Turbine Building to the ESGR and from the ESGR to the TSC will be temporarily breeched for cable pulling during installation. Re-installation of the fire barriers will be completed upon completion of the cable pulls. All work on the fire barriers shall be done in accordance with approved Station procedures.

8. Will the proposed activity/condition result in or constitute an unreviewed safety question, an unreviewed environmental question, a change to the Fire Protection Program that affects the ability of the station to achieve and maintain safe shutdown in the event of a fire, or require a license amendment or Technical Specifications change? [ ] Yes [X] No

9. Preparer Name (Print)	10. Preparer Signature	11. Date
W. A. Thomas, Jr.	W. A. Thomas, Jr.	9-12-95
12. Cognizant Supervisor Name (Print)	**13. Cognizant Supervisor Signature**	**14. Date**
E. W. May	[signature]	9/12/95

15. Disposition

[X] Approved [ ] Disapproved [ ] Approved As Modified [ ] Requires Further Evaluation

16. SNSOC Chairman Signature	17. Date
DAH	9-14-95

Comments

Note: Attach a Copy of Part A, Resolution Summary Report, to the Change/Activity Documentation Package.
Send a Copy of Part A to Licensing for Submittal to the NRC in Accordance With VPAP-2802, Reporting Requirements.
Send a copy of the completed Safety Evaluation to the Independent Review Coordinator (for the MSRC).
Send the completed Safety Evaluation Original to Records Management.
Use "Safety Evaluation, Supplemental Page" Form No. 730928, if additional Space is Needed.

Key: MSRC-Management Safety Review Committee

PART A - Resolution Summary Report, Item 5. Continuation

The abandoned high capacity S/G BD system will be made operational without requiring continuous condensate polishing for the secondary side of the Unit. The existing control instrumentation will be replaced and the drains from the BD flash tank will be cooled, then rerouted to the circulating water discharge tunnel, instead of the original design of routing the drains to the condenser hot well. The original design concept was to have the impurities in the blowdown removed by the condensate polishing system. Since the condensate polishing system is not continuously operated at power, the original high capacity blowdown system, operated in the Unit's current operational configuration without continuous condensate polishing, would result in zero net blowdown.

The drains from the BD flash tank will be cooled via a new shell and tube heat exchanger (the BD flash tank drains cooler) with condensate as the cooling medium. Thus, energy from the flash tank drains will be recovered to the steam cycle, while cooling the BD discharge to approximately the same temperature as the circulating water to which it is discharging. These upgrades to the high capacity BD system will permit the system to operate at its design capacity of approximately 100,000 lb/hr (200 gpm at 60F), or 67 gpm per S/G, if sufficient makeup water capacity is available. Present makeup water capacity allows a continuous S/G BD rate of approximately 45 gpm per S/G with both Units operating.

In addition to the BD flash tank drains cooler and its associated piping, this Design Change will also install a continuous radiation monitor and a sampling system in the cooled blowdown line upstream of the point of discharge to the circulating water discharge tunnel. The point of discharge of the cooled blowdown effluent from the high capacity S/G BD system is into the 20" - Class 121 piping discharging to the circulating water discharge tunnel. The low level liquid waste clarifier effluent also discharges to this 20" pipe in the same location. Therefore, the existing low capacity S/G BD system effluent (from the clarifier discharge) and the upgraded high capacity S/G BD system effluent enter the circulating water discharge tunnel at the same location.

The S/G BD is monitored for radioactivity by existing radiation monitors in sample lines off of the three S/G BD lines in the Auxiliary Building. The primary function of these radiation monitors is to detect radioactivity in the BD resulting from S/G tube leaks/ruptures. Currently the discharge of the Auxiliary Building S/G BD line radiation monitors (normal operating flow rate is less than 1 gpm per S/G) is routed through individual sample coolers, then combined into one line (1") for discharge into the low capacity S/G BD tank, and finally routed to the low level liquid waste system (entering the clarifier hold up tank inlet piping). Since the low capacity BD system will serve as a back-up system and is not planned for continuous operation after implementation of this Design Change, an alternate discharge path for the BD radiation monitors/sample coolers is included in the design. The alternate discharge path added by this Design Change is a 1" line teed off of the existing 1" line near the low capacity BD tank, routed to bypass the BD tank, the low capacity BD coolers, and the low capacity BD pumps. The Unit 2 BD radiation monitors/sample coolers discharge line will also tee into the alternate discharge path when DC 95-015 is implemented. Thus, when the low capacity system is not in operation, the S/G BD line radiation monitors/sample coolers can be manually aligned to discharge through the alternate radiation monitors/sample coolers discharge path to the clarifier hold up tank inlet piping downstream of the low capacity BD pumps. Since these new lines are located in the Auxiliary Building in the vicinity of various safety-related equipment, they will be seismically supported non-safety related lines. The current BD radiation monitors/sample coolers discharge path (BD tanks to low capacity BD coolers to BD pumps to clarifier hold up tank) may still be used with the high capacity BD system in operation, if necessary.

The new radiation monitor in the high capacity S/G BD system cooled blowdown effluent line (similar to the clarifier effluent radiation monitor, 1-LW-RM-111) does not perform a safety function, but is included in the design as added protection against release of radioactivity to the environment. The high capacity S/G BD system will be automatically isolated if the Hi-Hi trip setpoint of the cooled BD effluent radiation monitor is exceeded. This radiation monitor will provide local indication as well as local alarms. The trip and Hi-Hi alarm setpoints will be 5E-7 μCi/cc and the Hi alarm setpoint will be lower to preclude a release of radioacitivity to the environment.

The high capacity BD system as currently designed will be operated only when the Unit's steam generators have no identified primary to secondary tube leakage. Should leakage be identified at a level greater than 5E-7 μCi/cc at the effluent radiation monitor, the system will isolate on the high radiation trip. Upon the trip signal all valves except ~~with~~ the BD cooler outlet LCV will fail in the closed position, thus isolating BD flow and draining the BD flash tank. The tank contents at the time of the trip signal will empty to the circulating water discharge. This release will be considered an unplanned, but monitored release. Upon receipt of a Hi alarm signal, Operations will contact Health Physics to obtain analysis of the high capacity blowdown effluent and Steam Generator Blowdown streams to determine the source of the leakage. If S/G tube leakage is determined for a given steam generator during routine sampling, the steam generator shall be isolated from the high capacity BD system. The low capacity BD system will still be available for use on steam generators with discernable primary to secondary leakage in accordance with the requirements of the ODCM. An alarm setpoint for the high capacity BD system effluent radiation monitor will be established to provide early warning of negative trends. From a maintenance and surveillance perspective, the new radiation monitor will be consistent with the requirements for the clarifier discharge monitor, RM-LW-111, since it serves a similar purpose.

In the future, additional sampling will be provided for the high capacity BD system which will allow the use of the system when low levels of primary to secondary leakage exists within the current technical specification limits.

PART A - Resolution Summary Report, Item 5. Continuation

Two new divert lines to the condenser will be installed by this Design Change to help control the high capacity S/G BD system during operational periods other than 100% power (steady state), i. e., start-up, transients, or low power operation. One of the divert lines will be a BD flash tank outlet steam divert to the condenser, and the other will be a BD flash tank drains cooler condensate return line divert to the condenser. The steam discharge divert line (including a new PCV) will be installed to tee off of the existing BD flash tank steam discharge line for directing flashed steam from the flash tank to the condenser, if additional capacity is needed, or if the normal steam discharge line to the 3rd point extraction lines is not available for use. The section of the original BD flash tank drain line from downstream of the original LCV to the condenser will be re-installed for the majority of the piping run for this new steam divert line. The BD flash tank drains cooler condensate return divert line will tee off of the normal condensate return line, include a TCV and a check valve, then run to condenser 1-CN-SC-1B, connecting to spare penetration number 59. The TCV in this line will open on high BD (tubeside) outlet temperature from the BD flash tank drains cooler. High cooler BD outlet temperature is indicative of low condensate (cooling) flow, in most cases resulting from low main condensate flow, which produces low pressure drop (motive force for the condensate flow to the cooler) across the condensate side of the cooler. Low main condensate flow occurs during times of condensate pump recirculation operation or low power operation. Thus, these two new divert lines will enable the high capacity BD system to be operational at full normal capacity during either condensate system upset or low power operation conditions.

The existing UFSAR (page 10.4-33) references the operational concern that the High Capacity Blowdown system as originally designed did not isolate on Hi-Hi Feedwater Heater level or on Turbine trip. This concern has been corrected via this DCP. The 1-ES-NRV-103A, 1-ES-NRV-103B Non-Return valves are providing an input to the Process Control System, which will trip 1-BD-PCV-100. With 1-BD-PCV-100 closed, pressure control of the system will provided via 1-BD-PCV-101, which diverts the steam from the blowdown tank to the condenser. The Non-return valves receive trip close signals for a Turbine trip and a Hi-Hi Water Level in the associated Feedwater heater. Therefore, the isolation of the supplemental heater steam from the high capacity BD system is accomplished via the new control system functions and the original intent of the UFSAR to prevent overspeed of the turbine an inadvertant water intrusion are accomplished by this design change.

F.C. #2

The control system is equipped with a Bypass/Run function. The "Bypass" part of the Bypass/Run function will allow system start up by disabling the Lo-Lo inlet flow and level Trip signals and allow small inlet flows to warm up and fill the system. Once the system has warmed up and operationally defined blowdown flow rates and operationally defined levels have been established, the Bypass/Run function will be placed in "Run" which will enable the trip functions. Use of this function will be administratively controlled by Operating procedures.

Part A - Resolution Summary Report

18. Summarize from Part D, Unreviewed Safety Question Determination, the major issues considered: state the reason the change, test, or experiment should be allowed; and state why an unreviewed safety question does or does not exist (a simple conclusion statement is insufficient).

This Design Change does not constitute an unreviewed safety question as defined in 10CFR50.59 since:

1. This modification does not affect or impact any safety related equipment or systems. Therefore, this Design Change does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

2. This Design Change is consistent with the affected systems' design bases and existing design basis criteria. The systems affected by this Design Change do not include any systems important to safety or required for accident mitigation. Therefore, this modification does not create a possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

3. This Design Change does not impact or change the basis of any of the Technical Specifications, and, therefore, the margin of safety as defined in the bases of the Technical Specifications remains unchanged.

The new steam divert and the new BD cooler condensate return divert to the condenser, the new flash tank drains line to the BD cooler, and the new BD cooler condensate supply and return lines are additional high energy lines that were not present in the original S/G BD system design. The HELB analysis in the turbine building area will not be affected, however, for the following reasons: a break in the steam divert line (as with a break in the normal steam outlet line to the 3rd point extraction) is bounded by the 3rd point extraction line break analysis; a break in the flash tank drains line to the BD cooler will be quickly mitigated by a shutdown of the high capacity S/G BD system upon loss of level in the flash tank due to the break; and breaks in the BD cooler condensate supply, return, and divert lines are all bounded by the main condensate line break analysis. The routing for the new high energy lines added by this Design Change are not in the vicinity of any safety-related, or any Appendix R safe-shutdown equipment, and, thus, could not impact the aforementioned equipment during a postulated HELB.

The existing UFSAR (page 10.4-33) references the operational concern that the High Capacity Blowdown system as originally designed did not isolate on Hi-Hi Feedwater Heater level or on Turbine trip. This concern has been corrected via this DCP. The 1-ES-NRV-103A, 1-ES-NRV-103B Non-Return valves are providing an input to the Process Control System, which will trip 1-BD-PCV-100. With 1-BD-PCV-100 closed, pressure control of the system will provided via 1-BD-PCV-101, which diverts the steam from the blowdown tank to the condenser. The Non-return valves receive trip close signals for a Turbine trip and a Hi-Hi Water Level in the associated Feedwater heater. Therefore, the isolation of the supplemental heater steam from the high capacity BD system is accomplished via the new control system functions and the original intent of the UFSAR to prevent overspeed of the turbine an inadvertant water intrusion are accomplished by this design change.

The control system is equipped with a Bypass/Run function. The "Bypass" part of the Bypass/Run function will allow system start up by disabling the Lo-Lo inlet flow and level Trip signals and allow small inlet flows to warm up and fill the system. Once the system has warmed up and operationally defined blowdown flow rates and operationally defined levels have been established, the Bypass/Run function will be placed in "Run" which will enable the trip functions. Use of this function will be administratively controlled by Operating procedures. F.C. #2

The modifications to the high capacity S/G BD system (non-safety related system) installed by this Design Change will have no physical effect on levels of radiation or airborne radioactivity during installation, since none of the work will be done in the RCA. During operation if the radiation monitors detect radioactivity in the S/G BD, the S/G BD system will be isolated prior to potential for GDC-19 concerns.

The high capacity BD system as currently designed will be operated only when the Unit's steam generators have no identified primary to secondary tube leakage. Should leakage be identified at a level greater than 5E-7 μCi/cc at the effluent radiation monitor, the system will isolate on the high radiation trip. Upon the trip signal all valves except ~~with~~ the BD cooler outlet LCV will fail in the closed position, thus isolating BD flow and draining the BD flash tank. The tank contents at the time of the trip signal will empty to the circulating water discharge. This release will be considered an unplanned, but monitored release. Upon receipt of a Hi alarm signal, Operations will contact Health Physics to obtain analysis of the high capacity blowdown effluent and Steam Generator Blowdown streams to determine the source of the leakage. If S/G tube leakage is determined for a given steam generator during routine sampling, the steam generator shall be isolated from the high capacity BD system. The low capacity BD system will still be available for use on steam generators with discernable primary to secondary leakage in accordance with the requirements of the ODCM. An alarm setpoint for the high capacity BD system effluent radiation monitor will be established to provide early warning of negative trends. From a maintenance and surveillance perspective, the new radiation monitor will be consistent with the requirements for the clarifier discharge monitor, RM-LW-111, since it serves a similar purpose.

In the future, additional sampling will be provided for the high capacity BD system which will allow the use of the system when low levels of primary to secondary leakage exists within the current technical specification limits. Additionally, the new S/G BD effluent discharge locations resulting from this Design Change are being included in the new Virginia Pollution Discharge Elimination System (VPDES) permit for North Anna Power Station.

GN02-5475

1. Safety Evaluation Number	2. Applicable Station	3. Applicable Unit
95-SE-MOD-81	[X] North Anna Power Station [ ] Surry Power Station	[ ] Unit 1 [X] Unit 2 [ ] Unit 1 [ ] Unit 2

PART A - Resolution Summary Report

4. List the governing documents for which this safety evaluation was performed.

DC-94-015, Steam Generator Blowdown System Upgrades, North Anna Unit 2

5. Summarize the change, test, or experiment evaluated.

See Page 1A and 1B of 12.

6. State the purpose for this change, test, or experiment.

This change makes the existing abandoned high capacity S/G BD system operational without the requirement of continuous condensate polishing for the secondary side of the Unit. The existing control instrumentation will be replaced and the drains from the BD flash tank will be routed through a heat exchanger to recover heat and cool the effluent prior to discharge into the circulating water discharge tunnel. The original design routed the flash tank drains to the condenser hot well. The original design concept was to have the impurities in the blowdown removed by the condensate polishing system. Since the condensate polishing system is not continuously operated at power, the original high capacity blowdown system, operated in the Unit's current operational configuration without continuous condensate polishing, would result in zero net blowdown.

7. List all limiting conditions and special requirements identified or assumed by this safety analysis. For each item, indicate the formal tracking mechanism that will be used to ensure that these conditions and/or requirements will be met.

- The installation of all the upgrades to the S/G BD equipment and piping, including the removal of the previously installed tie-in stub weld caps, may be performed during any mode of operation with the modification being planned during non-outage time.
- PROCEDURAL CONTROLS ARE REQUIRED TO BE IN PLACE PRIOR TO PLACING SYSTEM IN SERVICE. THIS IS TRACKED BY THE CAR. (RJA 12-12-95)

8. Will the proposed activity/condition result in or constitute an unreviewed safety question, an unreviewed environmental question, a change to the Fire Protection Program that affects the ability of the station to achieve and maintain safe shutdown in the event of a fire, or require a license amendment or Technical Specifications change? [ ] Yes [X] No

9. Preparer Name (Print)	10. Preparer Signature	11. Date
R. J. Atkinson	[signature]	12/7/95
12. Cognizant Supervisor Name (Print)	13. Cognizant Supervisor Signature	14. Date
N.T. McBride	W. T. McBride	12/8/95

15. Disposition

[X] Approved [ ] Disapproved [ ] Approved As Modified [ ] Requires Further Evaluation

16. SNSOC Chairman Signature		17. Date
	DAH	12-12-95

Comments

Note: Attach a Copy of Part A, Resolution Summary Report, to the Change/Activity Documentation Package.
Send a Copy of Part A to Licensing for Submittal to the NRC in Accordance With VPAP-2802, Reporting Requirements.
Send a copy of the completed Safety Evaluation to the Independent Review Coordinator (for the MSRC).
Send the completed Safety Evaluation Original to Records Management.
Use "Safety Evaluation, Supplemental Page" Form No. 730928, if additional Space is Needed.

Key: MSRC-Management Safety Review Committee

PART A - Resolution Summary Report, Item 5. Continuation

The abandoned high capacity S/G BD system will be made operational without requiring continuous condensate polishing for the secondary side of the Unit. The existing control instrumentation will be replaced and the drains from the BD flash tank will be cooled, then rerouted to the circulating water discharge tunnel, instead of the original design of routing the drains to the condenser hot well. The original design concept was to have the impurities in the blowdown removed by the condensate polishing system. Since the condensate polishing system is not continuously operated at power, the original high capacity blowdown system, operated in the Unit's current operational configuration without continuous condensate polishing, would result in zero net blowdown.

The drains from the BD flash tank will be cooled via a new shell and tube heat exchanger (the BD flash tank drains cooler) with condensate as the cooling medium. Thus, energy from the flash tank drains will be recovered to the steam cycle, while cooling the BD discharge to approximately the same temperature as the circulating water to which it is discharging. These upgrades to the high capacity BD system will permit the system to operate at its design capacity of approximately 100,000 lb/hr (200 gpm at 60F), or 67 gpm per S/G, if sufficient makeup water capacity is available. Present makeup water capacity allows a continuous S/G BD rate of approximately 45 gpm per S/G with both Units operating.

In addition to the BD flash tank drains cooler and its associated piping, this Design Change will also install a continuous radiation monitor and a sampling system in the cooled blowdown line upstream of the point of discharge to the circulating water discharge tunnel. The point of discharge of the cooled blowdown effluent from the high capacity S/G BD system is into the 20" - Class 121 piping discharging to the circulating water discharge tunnel. The low level liquid waste clarifier effluent also discharges to this 20" pipe in the same location. Therefore, the existing low capacity S/G BD system effluent (from the clarifier discharge) and the upgraded high capacity S/G BD system effluent enter the circulating water discharge tunnel at the same location.

The S/G BD is monitored for radioactivity by existing radiation monitors in sample lines off of the three S/G BD lines in the Auxiliary Building. The primary function of these radiation monitors is to detect radioactivity in the BD resulting from S/G tube leaks/ruptures. Currently the discharge of the Auxiliary Building S/G BD line radiation monitors (normal operating flow rate is less than 1 gpm per S/G) is routed through individual sample coolers, then combined into one line (1") for discharge into the low capacity S/G BD tank, and finally routed to the low level liquid waste system (entering the clarifier hold up tank inlet piping). Since the low capacity BD system will serve as a back-up system and is not planned for continuous operation after implementation of this Design Change, an alternate discharge path for the BD radiation monitors/sample coolers is included in the design. The alternate discharge path added by this Design Change is a 1" line teed off of the existing 1" line near the low capacity BD tank, routed to bypass the BD tank, the low capacity BD coolers, and the low capacity BD pumps. The Unit 1 BD radiation monitors/sample coolers discharge line is also teed into the alternate discharge path (by DC 94-003). Thus, when the low capacity system is not in operation, the S/G BD line radiation monitors/sample coolers can be manually aligned to discharge through the alternate radiation monitors/sample coolers discharge path to the clarifier hold up tank inlet piping downstream of the low capacity BD pumps. Since these new lines are located in the Auxiliary Building in the vicinity of various safety-related equipment, they will be seismically supported non-safety related lines. The current BD radiation monitors/sample coolers discharge path (BD tanks to low capacity BD coolers to BD pumps to clarifier hold up tank) may still be used with the high capacity BD system in operation, if necessary.

The new radiation monitor in the high capacity S/G BD system cooled blowdown effluent line (similar to the clarifier effluent radiation monitor, 1-LW-RM-111) does not perform a safety function, but is included in the design as added protection against release of radioactivity to the environment. The high capacity S/G BD system will be automatically isolated if the Hi-Hi trip setpoint of the cooled BD effluent radiation monitor is exceeded. This radiation monitor will provide local indication as well as local alarms. The trip and Hi-Hi alarm setpoints will be 5E-7 µCi/cc and the Hi alarm setpoint will be lower to preclude a release of radioactivity to the environment.

The high capacity BD system as currently designed will be operated only when the Unit's steam generators have no identified primary to secondary tube leakage. Should leakage be identified at a level greater than 5E-7 µCi/cc at the effluent radiation monitor, the system will isolate on the high radiation trip. Upon the trip signal all valves except the BD cooler outlet LCV will fail in the closed position, thus isolating BD flow and draining the BD flash tank. The tank contents at the time of the trip signal will empty to the circulating water discharge. This release will be considered an unplanned, but monitored release. Upon receipt of a Hi alarm signal, Operations will contact Health Physics to obtain analysis of the high capacity blowdown effluent and Steam Generator Blowdown streams to determine the source of the leakage. If S/G tube leakage is determined for a given steam generator during routine sampling, the steam generator shall be isolated from the high capacity BD system. The low capacity BD system will still be available for use on steam generators with discernable primary to secondary leakage in accordance with the requirements of the ODCM. An alarm setpoint for the high capacity BD system effluent radiation monitor will be established to provide early warning of negative trends. From a maintenance and surveillance perspective, the new radiation monitor will be consistent with the requirements for the clarifier discharge monitor, RM-LW-111, since it serves a similar purpose.

In the future, additional sampling will be provided for the high capacity BD system which will allow the use of the system when low levels of primary to secondary leakage exists within the current technical specification limits.

Part A - Resolution Summary Report

18. Summarize from Part D, Unreviewed Safety Question Determination, the major issues considered; state the reason the change, test, or experiment should be allowed; and state why an unreviewed safety question does or does not exist (a simple conclusion statement is insufficient).

This Design Change does not constitute an unreviewed safety question as defined in 10CFR50.59 since:

1. This modification does not affect or impact any safety related equipment or systems. Therefore, this Design Change does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR.

2. This Design Change is consistent with the affected systems' design bases and existing design basis criteria. The systems affected by this Design Change do not include any systems important to safety or required for accident mitigation. Therefore, this modification does not create a possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

3. This Design Change does not impact or change the basis of any of the Technical Specifications, and, therefore, the margin of safety as defined in the bases of the Technical Specifications remains unchanged.

The new steam divert and the new BD cooler condensate return divert to the condenser, the new flash tank drains line to the BD cooler, and the new BD cooler condensate supply and return lines are additional high energy lines that were not present in the original S/G BD system design. The HELB analysis in the turbine building area will not be affected, however, for the following reasons: a break in the steam divert line (as with a break in the normal steam outlet line to the 3rd point extraction) is bounded by the 3rd point extraction line break analysis; a break in the flash tank drains line to the BD cooler will be quickly mitigated by a shutdown of the high capacity S/G BD system upon loss of level in the flash tank due to the break; and breaks in the BD cooler condensate supply, return, and divert lines are all bounded by the main condensate line break analysis. The routing for the new high energy lines added by this Design Change are not in the vicinity of any safety-related, or any Appendix R safe-shutdown equipment, and, thus, could not impact the aforementioned equipment during a postulated HELB.

The existing UFSAR (page 10.4-33) references the operational concern that the High Capacity Blowdown system as originally designed did not isolate on Hi-Hi Feedwater Heater level or on Turbine trip. This concern has been corrected via this DCP. The 2-ES-NRV-203A, 2-ES-NRV-203B Non-Return valves are providing an input to the Process Control System, which will trip 2-BD-PCV-200. With 2-BD-PCV-200 closed, pressure control of the system will provided via 2-BD-PCV-201, which diverts the steam from the blowdown tank to the condenser. The Non-return valves receive trip close signals for a Turbine trip and a Hi-Hi Water Level in the associated Feedwater heater. Therefore, the isolation of the supplemental heater steam from the high capacity BD system is accomplished via the new control system functions and the original intent of the UFSAR to prevent overspeed of the turbine an inadvertent water intrusion are accomplished by this design change.

The control system is equipped with a Bypass/Run function. The "Bypass" part of the Bypass/Run function will allow system start up by disabling the Lo-Lo inlet flow and the Lo-Lo level Trip signals and allow small flow rates to warm up the system. Once the system has warmed up and operationally defined blowdown flow rates have been established, the Bypass/Run function will be placed in "Run" which will enable the trip functions. Use of this function will be administratively controlled by Operating procedures.

The modifications to the high capacity S/G BD system (non-safety related system) installed by this Design Change will have no physical effect on levels of radiation or airborne radioactivity during installation, since none of the work will be done in the RCA, except for the installation of a short (less than 10' length) section of 1" pipe in the Aux. Bldg. Work done in the Aux. Bldg. will be performed in the low general dose rate areas. The number of man-hours expected to complete the installation is minimal. Therefore, the accumulated exposure is expected to be minimal and no further calculation for personnel exposure is required. During operation if the radiation monitors detect radioactivity in the S/G BD, the S/G BD system will be isolated prior to potential for GDC-19 concerns.

The high capacity BD system as currently designed will be operated only when the Unit's steam generators have no identified primary to secondary tube leakage. Should leakage be identified at a level greater than 5E-7 μCi/cc at the effluent radiation monitor, the system will isolate on the high radiation trip. Upon the trip signal all valves except the BD cooler outlet LCV will fail in the closed position, thus isolating BD flow and draining the BD flash tank. The tank contents at the time of the trip signal will empty to the circulating water discharge. This release will be considered an unplanned, but monitored release. Upon receipt of a Hi alarm signal, Operations will contact Health Physics to obtain analysis of the high capacity blowdown effluent and Steam Generator Blowdown streams to determine the source of the leakage. If S/G tube leakage is determined for a given steam generator during routine sampling, the steam generator shall be isolated from the high capacity BD system. The low capacity BD system will still be available for use on steam generators with discernable primary to secondary leakage in accordance with the requirements of the ODCM. An alarm setpoint for the high capacity BD system effluent radiation monitor will be established to provide early warning of negative trends. From a maintenance and surveillance perspective, the new radiation monitor will be consistent with the requirements for the clarifier discharge monitor, RM-LW-111, since it serves a similar purpose.

PART A - Resolution Summary Report, Item 5. Continuation

Two new divert lines to the condenser will be installed by this Design Change to help control the high capacity S/G BD system during operational periods other than 100% power (steady state), i. e., start-up, transients, or low power operation. One of the divert lines will be a BD flash tank outlet steam divert to the condenser, and the other will be a BD flash tank drains cooler condensate return line divert to the condenser. The steam discharge divert line (including a new PCV) will be installed to tee off of the existing BD flash tank steam discharge line for directing flashed steam from the flash tank to the condenser, if additional capacity is needed, or if the normal steam discharge line to the 3rd point extraction lines is not available for use. The section of the original BD flash tank drain line from downstream of the original LCV to the condenser will be re-installed for the majority of the piping run for this new steam divert line. The BD flash tank drains cooler condensate return divert line will tee off of the normal condensate return line, include a TCV and a check valve, then run to condenser 2-CN-SC-1B, connecting to spare penetration number 57. The TCV in this line will open on high BD (tubeside) outlet temperature from the BD flash tank drains cooler. High cooler BD outlet temperature is indicative of low condensate (cooling) flow, in most cases resulting from low main condensate flow, which produces low pressure drop (motive force for the condensate flow to the cooler) across the condensate side of the cooler. Low main condensate flow occurs during times of condensate pump recirculation operation or low power operation. Thus, these two new divert lines will enable the high capacity BD system to be operational at full normal capacity during either condensate system upset or low power operation conditions.

The existing UFSAR (page 10.4-33) references the operational concern that the High Capacity Blowdown system as originally designed did not isolate on Hi-Hi Feedwater Heater level or on Turbine trip. This concern has been corrected via this DCP. The 2-ES-NRV-203A, 2-ES-NRV-203B Non-Return valves are providing an input to the Process Control System, which will trip 2-BD-PCV-200. With 2-BD-PCV-200 closed, pressure control of the system will provided via 2-BD-PCV-201, which diverts the steam from the blowdown tank to the condenser. The Non-return valves receive trip close signals for a Turbine trip and a Hi-Hi Water Level in the associated Feedwater heater. Therefore, the isolation of the supplemental heater steam from the high capacity BD system is accomplished via the new control system functions and the original intent of the UFSAR to prevent overspeed of the turbine an inadvertent water intrusion are accomplished by this design change.

The control system is equipped with a Bypass/Run function. The "Bypass" part of the Bypass/Run function will allow system start up by disabling the Lo-Lo inlet flow and Lo-Lo level Trip signals and allow small flow rates to warm up the system. Once the system has warmed up and operationally defined blowdown flow rates have been established, the Bypass/Run function will be placed in "Run" which will enable the trip functions. Use of this function will be administratively controlled by Operating procedures.

ATTACHMENT 5

INOPERABILITY OF RADIOACTIVE LIQUID AND GASEOUS EFFLUENT MONITORING INSTRUMENTATION

(01/95 - 12/95)

As required by the ODCM, Sections 6.2.2.b.2 and 6.3.2.b.3, a list and explanation for extended inoperability of radioactive liquid and/or gaseous effluent monitoring instrumentation is provided in this attachment.

No extended periods of inoperability occurred with any of the Liquid or Gaseous Effluent Monitoring Instrumentation specified in the ODCM, Attachments 2 and 15, for the time period covered by this report.

ATTACHMENT 6

UNPLANNED RELEASES

(01/95 - 12/95)

As required by the ODCM, Section 6.7.2.a.3, a list of unplanned releases, from the site to unrestricted areas, of radioactive material in gaseous and liquid effluents occurring during the reporting period, is made in this attachment.

No unplanned releases, as defined by the criteria presented in the ODCM, Section 6.7.2.a.3, occurred during the time period covered by this report.

ATTACHMENT 7

LOWER LIMITS OF DETECTION FOR EFFLUENT SAMPLE ANALYSIS

(01/95 - 12/95)

Gaseous Effluents:

Radioisotope	Required L.L.D. (μCi/ml)	Typical L.L.D. (μCi/ml)
Krypton - 87	1.00E-4	4.40E-8 - 3.98E-7
Krypton - 88	1.00E-4	7.09E-8 - 5.70E-7
Xenon - 133	1.00E-4	4.48E-8 - 3.23E-7
Xenon - 133m	1.00E-4	1.51E-7 - 1.38E-6
Xenon - 135	1.00E-4	1.79E-8 - 1.50E-7
Xenon - 135m	1.00E-4	6.84E-8 - 5.83E-7
Xenon - 138	1.00E-4	1.64E-7 - 1.96E-6
Iodine - 131	1.00E-12	5.52E-14 - 7.59E-14
Manganese - 54	1.00E-11	3.64E-14 - 5.32E-14
Cobalt - 58	1.00E-11	4.05E-14 - 5.21E-14
Iron - 59	1.00E-11	7.84E-14 - 1.01E-13
Cobalt - 60	1.00E-11	5.40E-14 - 1.52E-13
Zinc - 65	1.00E-11	9.09E-14 - 1.19E-13
Strontium - 89	1.00E-11	4.00E-15 - 5.00E-15
Strontium - 90	1.00E-11	7.00E-16 - 1.00E-15
Molybdenum - 99	1.00E-11	2.72E-13 - 3.47E-13
Cesium - 134	1.00E-11	5.32E-14 - 1.61E-13
Cesium - 137	1.00E-11	4.64E-14 - 5.77E-14
Cerium - 141	1.00E-11	4.54E-14 - 6.79E-14
Cerium - 144	1.00E-11	2.04E-13 - 3.22E-13
Gross Alpha	1.00E-11	6.90E-15 - 1.20E-14
Tritium	1.00E-6	1.12E-7 - 1.37E-7

ATTACHMENT 7
LOWER LIMITS OF DETECTION FOR EFFLUENT SAMPLE ANALYSIS
(01/95 - 12/95)

Liquid Effluents:

Radioisotope	Required L.L.D. (μCi/ml)	Typical L.L.D. (μCi/ml)
Krypton - 87	1.00E-5	5.93E-8 - 7.29E-8
Krypton - 88	1.00E-5	9.38E-8 - 1.16E-7
Xenon - 133	1.00E-5	6.08E-8 - 8.83E-8
Xenon - 133m	1.00E-5	2.04E-7 - 2.69E-7
Xenon - 135	1.00E-5	2.81E-8 - 3.14E-8
Xenon - 135m	1.00E-5	8.92E-8 - 1.19E-7
Xenon - 138	1.00E-5	2.17E-7 - 3.41E-7
Iodine - 131	1.00E-6	2.63E-8 - 3.43E-8
Manganese - 54	5.00E-7	2.45E-8 - 3.39E-8
Iron - 55	1.00E-6	9.70E-9 - 8.00E-7
Cobalt - 58	5.00E-7	2.61E-8 - 3.25E-8
Iron - 59	5.00E-7	5.04E-8 - 5.86E-8
Cobalt - 60	5.00E-7	2.86E-8 - 8.43E-8
Zinc - 65	5.00E-7	5.81E-8 - 6.83E-8
Strontium - 89	5.00E-8	3.00E-8 - 5.00E-8
Strontium - 90	5.00E-8	5.00E-9 - 1.00E-8
Molybdenum - 99	5.00E-7	1.82E-7 - 2.03E-7
Cesium - 134	5.00E-7	3.59E-8 - 1.00E-7
Cesium - 137	5.00E-7	3.32E-8 - 3.79E-8
Cerium - 141	5.00E-7	4.07E-8 - 5.47E-8
Cerium - 144	5.00E-7	1.81E-7 - 2.63E-7
Gross Alpha	1.00E-7	2.15E-8 - 3.72E-8
Tritium	1.00E-5	2.78E-6 - 3.32E-6