1717 Wakonade Drive
Welch, MN 55089

May 14, 2019

L-PI-19-012
TS 5.5.1.c
TS 5.6.3

ATTN: Document Control Desk
Nuclear Regulatory Commission
Washington, DC 20555-0001

Prairie Island Nuclear Generating Plant, Units 1 and 2
Docket Nos. 50-282 and 50-306
Renewed Facility Operating License Nos. DPR-42 and DPR-60

2018 Annual Radioactive Effluent Report and Offsite Dose Calculation Manual

Pursuant to the applicable Prairie Island Nuclear Generating Plant (PINGP) Technical Specifications (TS), Appendix A to Renewed Operating Licenses DPR-42 and DPR-60, and the requirements of the H4, Offsite Dose Calculation Manual (ODCM), Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), submits the 2018 Annual Radioactive Effluent Report which is comprised of the following:

- Enclosure 1 contains the Off-Site Radiation Dose Assessment for the period January 1, 2018, through December 31, 2018, in accordance with ODCM section 8.1.1.
- Enclosure 2 contains the Annual Radioactive Effluent Report, Supplemental Information, for the period January 1, 2018, through December 31, 2018, in accordance with TS 5.6.3 and ODCM section 8.1.1.
- Enclosure 3 contains a complete and legible copy of the entire H4, ODCM revision 32, issued on May 25, 2018, in accordance with ODCM section 8.1.1.
- Enclosure 4 contains a complete and legible copy of the D59, Process Control Program for Processing/Dewatering of Radioactive Waste from Liquid Systems revision 12, issued on January 26, 2018, in accordance with ODCM section 8.1.1.

Summary of Commitments

This letter makes no new commitments and no revisions to existing commitments.

Scott Sharp
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosures (4)

cc: Administrator, Region III, USNRC Project
Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
Department of Health, State of Minnesota
PI Dakota Community Environmental Coordinator

ENCLOSURE 1

OFF-SITE RADIATION DOSE ASSESSMENT

January 1, 2018 - December 31, 2018

7 pages to follow

PRAIRIE ISLAND NUCLEAR GENERATING PLANT OFF-SITE RADIATION DOSE ASSESSMENT FOR

January 1, 2018 - December 31, 2018

An Assessment of the 2018 radiation dose, due to operation of The Prairie Island Nuclear Generating Plant, was performed in accordance with the Offsite Dose Calculation Manual, and as required by Technical Specifications. Computed doses were well below the 40 CFR Part 190 Standards and 10 CFR Part 50 Appendix I Guidelines.

Off-site dose calculation formulas and historical meteorological data were used in making this assessment. Source terms were obtained from the Annual Radioactive Effluent and Waste Disposal Report and prepared for NRC review, for the year of 2018.

OFFSITE DOSES FROM GASEOUS RELEASE:

Computed doses due to gaseous releases are reported in Table 1. Critical receptor location and pathways for organ doses are reported in Table 2. Gaseous release doses are a small percentage of Appendix I Guidelines.

OFFSITE DOSES FROM LIQUID RELEASE:

Computed doses due to liquid releases are reported in Table 1. Critical receptor information is reported in Table 2. Liquid release doses, both whole body and organ, are a small percentage of Appendix I Guidelines.

DOSES TO INDIVIDUALS DUE TO ACTIVITIES INSIDE THE SITE BOUNDARY:

Occasionally sportsmen enter the Prairie Island Site Boundary for recreational activities. These individuals are not expected to spend more than a few hours per year within the site boundary. Commercial and recreational river traffic exists through this area.

For purposes of estimating the dose due to recreational and river water transportation activities within the site boundary it is assumed that the limiting dose within the site boundary would be received by an individual who spends a total of seven days per year on the river just off-shore from the plant buildings (ESE at 0.2 miles). The gamma and beta doses from noble gas releases and the maximum organ doses from the inhalation pathway due to Iodine 131, Iodine-133, tritium, long-lived particulates and Carbon-14 were calculated for this location and occupancy time. These doses are reported in Table 1.

Critical Receptor location and pathways for organ doses are reported in Table 2.

<u>40 CFR 190 COMPLIANCE:</u>

REMP environmental TLD results for 2018 were reviewed per ANSI/HPS N13.37-2014 methodology for determining any plant effect above ambient gamma radiation measurements. All measurements are considered to be within the range of variations in natural background radiation.

Neutron sky shine dose from the ISFSI was evaluated. The maximum neutron sky shine dose was determined to be 0.79 mrem, to the nearest resident, at 724 meters from the ISFSI. Neutron sky shine dose is greater than the effluent dose to the Critical Receptor, therefore, 40 CFR190 compliance was evaluated to the location of the maximum neutron sky shine dose.

The 40 CFR 190 evaluation location was determined to be 0.7 miles west of the plant. Dose due to gaseous effluents was calculated to the 40 CFR 190 evaluation location.

	MREM
Gamma Direct Radiation Dose:	0.00E+00
Neutron Sky Shine Dose:	7.90E-01
Noble *Gas Gamma Dose:*	8.87E-07
Noble Gas Beta Dose:	3.33E-06
Iodine, particulate, H-3 and C-14 Dose:	4.16E-03*

*Calculated values were identical for Whole Body, Thyroid and Maximum "Other" Organs

SUMMATION OF 40 CFR 190 DOSE:

	40 CFR 190 LIMIT (MREM)	**40 CFR 190 DOSE (MREM)**
WHOLE BODY	**25**	**7.94E-01**
THYROID	**75**	**7.94E-01**
OTHER ORGANS (TEEN - WHOLE BODY)	**25**	**7.94E-01**

ABNORMAL RELEASES:

There were zero (0) abnormal releases in 2018.

SAMPLING, ANALYSIS AND LLD REQUIREMENTS:

The lower limit of detection (LLD) requirements, as specified in ODCM Table 2.1 and 3.1, were met for 2018. The minimum sampling frequency requirements, as specified in ODCM Table 2.1, were met for 2018.

The minimum sampling frequency requirements, as specified in ODCM Table 3.1, were not met for 2018, on one (1) occasion.

I-131, I-133, Principal Gamma Emitters and Gross Alpha were not evaluated for Rad Waste Building Ventilation, for the week of 6/25/18 to 7/2/18.

Particulate and charcoal filters are collected and evaluated on a weekly basis, for all gaseous continuous release pathways, in accordance with ODCM Table 3.1.

The sample line is equipped with a flow integrator, to determine the uncorrected sample volume. The sample line is equipped with a vacuum gauge, to correct the flow integrator uncorrected sample volume.

During weekly samples collection, the Rad Waste Building Ventilation vacuum gauge reading was found elevated at 13" and the sample line flow integrator indicated no sample had been collected.

It was determined that the sample line was plugged and that no sample had been collected. A temporary sampling rig was established in the Rad Waste Building until repairs could be accomplished. Repairs were completed on 7/13/18.

A review of Rad Waste Ventilation samples was performed, including all samples since the beginning of the year and sample results for the week following the missed sample. No activity was quantified on particulate or charcoal filter samples.

A review of work in the Rad Waste Building during the week noted no work that would contribute to airborne activity levels.

The weekly release permit was completed attributing no particulate or iodine activity.

MONITORING INSTRUMENTATION:

For 2018, there were zero (0) occurrences, when less than the minimum required radioactive liquid and/or gaseous effluent monitoring instrumentation channels were operable, as required by ODCM Tables 2.2 and 3.2.

DOSES TO INDIVIDUALS DUE TO EFFLUENT RELEASES FROM THE INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI):

Four (4) fuel casks were loaded and placed in the ISFSI during the 2018 calendar year. The total number of casks in the ISFSI, as of 12/31/18, was forty-four (44). There were zero (0) releases of radioactive effluents from the ISFSI.

CURRENT OFFSITE DOSE CALCULATIONS MANUAL (ODCM) REVISION:

The Offsite Dose Calculation Manual was revised once in 2018. The current revision, Revision 32, is dated May 25, 2018. Revision 32 is submitted in its entirety with the 2018 report.

The methodology for bounding tritium in set point calculations, as developed in revision 27, has been expanded. It now includes hard-to-detect nuclides, which do not generate a radiation monitor response.

PROCESS CONTROL PROGRAM:

D59, The Process Control Program for Solidification/Dewatering of Radioactive Waste from Liquid Systems, was revised in 2018. Revision 12 is the current revision. The revision date is January 26, 2018. Revision 12 is submitted with the 2018 report.

INDUSTRY INITIATIVE ON GROUND WATER PROTECTION:

For 2018, there was zero (0) events for inclusion in the Annual Effluent Report, as part of the NEI Ground Water Initiative.

CRITICAL RECEPTOR:

Based on the Annual Land Use Census, the critical receptor did not change. The critical receptor is defined as The Suter Residence, at 0.6 miles, in the SSE sector.

LOW LEVEL WASTE DISPOSAL ANNUAL REPORT SOLID WASTE AND IRRADIATED COMPONENTS SHIPMENTS PERIOD: 1/1/18 TO 12/31/18 LICENSE NUMBER: DPR-42/60

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

Resins, Filters and Evaporator Bottoms	Volume		Curies Shipped
Waste Class	FT3	M3	Curies
A	4.24E+02	1.20E+01	1.70E+01
B	0.00+00	0.00E+00	0.00+00
C	0.00+00	0.00E+00	0.00E+00
ALL	4.24E+02	1.20E+01	1.70E+01
Major Nuclides	H-3, C-14, Mn-54, Fe-55, Co-58, Co-60, Ni-59, Ni-63, Sr-90, Zr-95, Nb-94, Nb-95, Tc-99, Ag-110m, Sb-125, I-129, Cs-137, Ce-144, Pu-238, Pu-239, Pu-240, Pu-241, Am-241, Cm-242, Cm-243, Cm-244		

Dry Active Waste	Volume		Curies Shipped
Waste Class	FT3	M3	Curies
A	1.92E+04	5.45E+02	1.94E-01
B	0.00E+00	0.00E+00	0.00E+00
C	0.00E+00	0.00E+00	0.00E+00
ALL	1.92E+04	5.45E+02	1.94E-01
Major Nuclides	H-3, C-14, Fe-55, Co-58, Co-60, Ni-63, Zr-95, Nb-95, Tc-99, I-129, Cs-137, Ce-144		

Irradiated Components	Volume		Curies Shipped
Waste Class	FT3	M3	Curies
A	0.00E+00	0.00E+00	0.00E+00
B	0.00E+00	0.00E+00	0.00E+00
C	0.00E+00	0.00E+00	0.00E+00
ALL	0.00E+00	0.00E+00	0.00E+00
Major Nuclides			

Other Waste	Volume		Curies Shipped
Waste Class	FT3	M3	Curies
A	1.50E+01	4.25E-01	5.20E-01
B	0.00E+00	0.00E+00	0.00E+00
C	0.00E+00	0.00E+00	0.00E+00
ALL	1.50E+01	4.25E-01	5.20E-01
Major Nuclides	H-3, C-14, Co-60, Ni-63, Sr-90, Tc-99, I-129, Cs-137, Pu-238, Pu-139, Pu-140, Pu-241, Am-241, Cm-242, Cm-243, Cm-244		

Sum of All Low Level Waste Shipped from Site	Volume		Curies Shipped
Waste Class	FT3	M3	Curies
A	1.97E+04	5.57E+02	1.77E+01
B	0.00E+00	0.00E+00	0.00E+00
C	0.00E+00	0.00E+00	0.00E+00
ALL	1.97E+04	5.57E+02	1.77E+01
Major Nuclides	H-3, C-14, Mn-54, Fe-55, Co-58, Co-60, Ni-59, Ni-63, Sr-90, Zr-95, Nb-94, Nb-95, Tc-99, Ag-110m, Sb-125, I-129, Cs-137, Ce-144, Pu-238, Pu-239, Pu-240, Pu-241, Am-241, Cm-242, Cm-243, Cm-244		

Total curie quantity and principal radionuclides identification are calculated estimates determined for packaged waste using gross gamma radiation measurements, direct sample data or swipe data within WMG's Radman Suite Software. Characterization of radioactive waste is performed in accordance with 10 CFR 20, 10 CFR 61, and NRC's Branch Technical Positions.

Table 1

OFF-SITE RADIATION DOSE ASSESSMENT

JANUARY 2018 THROUGH DECEMBER 2018

	DOSE	LIMIT*
Gaseous Releases		
Maximum Site Boundary Gamma Air Dose (mrad)	2.23E-06	20
Maximum Site Boundary Beta Air Dose (mrad)	8.48E-06	40
Maximum Off-site Dose to any Organ (mrem)**	6.37E-02	30
Organ	Child – bone	
Offshore Location		
Maximum Site Boundary Gamma Air Dose (mrad)	3.62E-07	20
Maximum Site Boundary Beta Air Dose (mrad)	1.89E-06	40
Maximum Off-site Dose to any Organ (mrem)**	4.62E-04	30
Organ	Teen – Total Body	
Liquid Releases		
Maximum Off-site Dose Total Body (mrem)	2.39E-03	6
Maximum Off-site Dose to any Organ (mrem)	2.44E-03	20
Organ	Adult – Gi-LLi	

*10 CFR part 50, Appendix I Guidelines (2-unit site per year)

**Long Lived Particulate, I-131, I-133, Tritium and C-14

Table 2

OFF-SITE RADIATION DOSE ASSESSMENT- PRAIRIE ISLAND SUPPLEMENTAL INFORMATION

January 1, 2018 - December 31, 2018

<u>Gaseous Releases</u>

Maximum Site Boundary Dose Location (From Building Vents)	
Sector	W
Distance (miles)	0.36
Offshore Location Within Site Boundary	
Sector	ESE
Distance (miles)	0.2
Pathway	Inhalation
Critical Receptor Location	
Sector	SSE
Distance (miles)	0.60
Pathways	Ground Inhalation Vegetable
Age Group	Child

<u>Liquid Releases</u>

Maximum Off-site Dose Location	
Sector	SSE
Distance (miles)	0.43
Pathway	Fish

ENCLOSURE 2

ANNUAL RADIOACTIVE EFFLUENT REPORT
SUPPLEMENTAL INFORMATION

January 1, 2018 - December 31, 2018

8 pages to follow

ANNUAL RADIOACTIVE EFFLUENT REPORT

SUPPLEMENTAL INFORMATION

01-JAN-18 THROUGH 31-DEC-18

Facility: Prairie Island Nuclear Generating Plant

Licensee: Northern States Power Company

License Numbers: DPR-42 & DPR-60

A. Regulatory Limits

1. Liquid Effluents:

   a. The dose or dose commitment to an individual from radioactive materials in liquid effluents released from the site shall be limited to:

for the quarter	3.0 mrem to the total body 10.0 mrem to any organ
for the year	6.0 mrem to the total body 20.0 mrem to any organ

2. Gaseous Effluents:

   a. The dose rate due to radioactive materials released in gaseous effluents from the site shall be limited to:

noble gases	500 mrem/year total body 3000 mrem/year skin
I-131, I-133, H-3, LLP, C-14	1500 mrem/year to any organ

   b. The dose due to radioactive gaseous effluents released from the site shall be limited to:

noble gases	10 mrad/quarter gamma 20 mrad/quarter beta 20 mrad/year gamma 40 mrad/year beta
I-131, I-133, H-3, LLP, C-14	15 mrem/quarter to any organ 30 mrem/year to any organ

B. Effluent Concentration

1. Fission and activation gases in gaseous releases:

   10 CFR 20, Appendix B, Table 2, Column 1

2. Iodine and particulates with half-lives greater than 8 days in gaseous releases:

   10 CFR 20, Appendix B, Table 2, Column 1

3. Liquid effluents for radionuclides other than dissolved or entrained gases:

   10 CFR 20, Appendix B, Table 2, Column 2

4. Liquid effluent dissolved and entrained gases:

   Offsite Dose Calculation Manual

C. Average Energy

Not applicable to Prairie Island regulatory limits.

D. Measurements and approximations of total activity

1. Fission and activation gases in gaseous releases:	Total Nuclide	HPGe	±25%
2. Iodines in gaseous releases:	Total Nuclide	HPGe	±25%
3. Particulates in gaseous releases:	Total Nuclide	HPGe	±25%
4. Liquid effluents	Total Nuclide	HPGe	±25%

E. Manual Revisions

1. Offsite Dose Calculations Manual:

   Latest Revision number: 32

   Revision date May 25, 2018

Prairie Island Nuclear Generating Station
PI 2018 Annual Release Summary

Batch Release Summary

Liquid Releases	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Year
Number of Releases:	20	26	60	65	171
Total Time for All Releases (Minutes):	1485.0	1990.0	4686.0	5314.0	13475.0
Maximum Time for All Releases (Minutes):	87.0	109.0	127.0	118.0	127.0
Average Time for All Releases (Minutes):	74.3	76.5	78.1	81.8	78.8
Minimum Time for All Releases (Minutes):	60.0	62.0	60.0	60.0	60.0

Gaseous Releases	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Year
Number of Releases:	1	1	10	19	31
Total Time for All Releases (Minutes):	129600.0	131040.0	139533.0	156440.0	556613.0
Maximum Time for All Releases (Minutes):	129600.0	131040.0	132480.0	132480.0	132480.0
Average Time for All Releases (Minutes):	129600.0	131040.0	13953.3	8233.7	17955.3
Minimum Time for All Releases (Minutes):	129600.0	131040.0	1.0	460.0	1.0

Abnormal Release Summary

Liquid Releases	
Number of Abnormal Releases:	0
Total Activity for Abnormal Releases:	0.00E+00 Curies

Gaseous Releases	
Number of Abnormal Releases:	0
Total Activity for Abnormal Releases:	0.00E+00 Curies

Prairie Island Nuclear Generating Station
PI 2018 Annual Release Summary

Gaseous Effluents-Summation of All Releases

Type of Effluent	Units	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Total Error, %
A. Fission & Activation Gases						
1. Total Release	Curies	1.53E-03	3.24E-04	2.39E-02	6.50E-04	2.50E+01
2. Average Release Rate for Period	µCi/sec	1.96E-04	4.12E-05	3.01E-03	8.17E-05	
3. Percent of Applicable Limit	%	8.44E-06	1.16E-06	3.18E-05	2.38E-06	
B. Iodines						
1. Total Iodine-131	Curies	0.00E+00	0.00E+00	1.32E-06	5.20E-08	2.50E+01
2. Average Release Rate for Period	µCi/sec	0.00E+00	0.00E+00	1.65E-07	6.54E-09	
3. Percent of Applicable Limit	%	0.00E+00	0.00E+00	1.32E-04	5.20E-06	
C. Particulates						
1. Total Particulates (Half-lives > 8 days)	Curies	0.00E+00	0.00E+00	1.29E-09	1.27E-06	2.50E+01
2. Average Release Rate for Period	µCi/sec	0.00E+00	0.00E+00	1.62E-10	1.59E-07	
3. Percent of Applicable Limit	%	0.00E+00	0.00E+00	1.17E-08	6.50E-06	
4. Gross Alpha Activity	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.50E+01
D. Tritium						
1. Total Release	Curies	5.80E+00	6.78E+00	8.83E+00	9.80E+00	2.50E+01
2. Average Release Rate for Period	µCi/sec	7.46E-01	8.63E-01	1.11E+00	1.23E+00	
3. Percent of Applicable Limit	%	1.06E-02	1.24E-02	1.63E-02	1.80E-02	
E. Carbon-14						
1. Total Release	Curies	2.85E+00	2.90E+00	2.73E+00	2.49E+00	2.50E+01

Prairie Island Nuclear Generating Station

PI 2018 Annual Release Summary

Gaseous Effluents - Ground Level Releases

		Continuous Mode				Batch Mode			
Nuclides Released	Units	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Qtr 1	Qtr 2	Qtr 3	Qtr 4
1. Fission and Activation Gases									
Ar-41	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.07E-05	0.00E+00	8.21E-06	4.22E-06
Kr-85	Curies	0.00E+00	0.00E+00	9.31E-03	0.00E+00	2.11E-04	0.00E+00	1.18E-04	0.00E+00
Kr-85M	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.50E-06	0.00E+00	0.00E+00	0.00E+00
Kr-88	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.23E-06	0.00E+00	0.00E+00	0.00E+00
Xe-131m	Curies	0.00E+00	0.00E+00	1.07E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	Curies	0.00E+00	0.00E+00	1.37E-02	0.00E+00	1.10E-03	3.14E-04	6.27E-04	6.29E-04
Xe-133m	Curies	0.00E+00	0.00E+00	2.13E-06	0.00E+00	2.14E-05	4.78E-06	1.35E-05	0.00E+00
Xe-135	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.22E-04	4.90E-06	2.41E-05	1.61E-05
Total for Period	Curies	0.00E+00	0.00E+00	2.32E-02	0.00E+00	1.53E-03	3.24E-04	7.90E-04	6.50E-04
2. Iodines									
I-131	Curies	0.00E+00	0.00E+00	1.32E-06	5.20E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total For Period	Curies	0.00E+00	0.00E+00	1.32E-06	5.20E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
3. Particulates									
Cd-109	Curies	0.00E+00	0.00E+00	0.00E+00	3.50E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	Curies	0.00E+00	0.00E+00	1.29E-09	8.29E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	Curies	0.00E+00	0.00E+00	0.00E+00	3.02E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	Curies	0.00E+00	0.00E+00	0.00E+00	5.64E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total for Period	Curies	0.00E+00	0.00E+00	1.29E-09	1.27E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00
4. Tritium									
H-3	Curies	5.80E+00	6.78E+00	8.81E+00	9.80E+00	3.91E-04	1.84E-04	1.56E-02	3.18E-03
5. Carbon-14									
C-14	Curies	2.85E+00	2.90E+00	2.73E+00	2.49E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Prairie Island Nuclear Generating Station

PI 2018 Annual Release Summary

Liquid Effluents - Summation of All Releases

Type of Effluent	Units	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Total Error, %
A. Fission & Activation Products						
1. Total Release (not including Tritium, Gases, and Alpha)	Curies	1.99E-03	2.05E-03	9.80E-04	1.58E-03	2.50E+01
2. Average Diluted Concentration During Period	µCi/ml	2.42E-11	2.65E-11	1.22E-11	1.64E-11	
3. Percent of Applicable Limit	%	3.98E-02	4.10E-02	1.96E-02	3.17E-02	
B. Tritium						
1. Total Release	Curies	1.38E+02	1.40E+02	2.99E+02	1.80E+02	2.50E+01
2. Average Diluted Concentration During Period	µCi/ml	1.68E-06	1.81E-06	3.72E-06	1.86E-06	
3. Percent of Applicable Limit	%	1.68E-01	1.81E-01	3.72E-01	1.86E-01	
C. Dissolved and Entrained Gases						
1. Total Release	Curies	1.01E-05	4.01E-06	9.48E-05	6.46E-05	2.50E+01
2. Average Diluted Concentration During Period	µCi/sec	1.23E-13	5.19E-14	1.18E-12	6.68E-13	
3. Percent of Applicable Limit	%	6.14E-08	2.60E-08	5.91E-07	3.34E-07	
D. Gross Alpha Radioactivity						
1. Total Release	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.50E+01
E. Waste Volume Released (Pre-Dilution)	Liters	3.90E+07	3.99E+07	3.12E+07	4.48E+07	2.50E+01
F. Volume of Dilution Water Used	Liters	8.22E+10	7.73E+10	8.02E+10	9.67E+10	2.50E+01

Prairie Island Nuclear Generating Station
PI 2018 Annual Release Summary

Liquid Effluents

		Continuous Mode				Batch Mode			
Nuclides REleased	Units	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Qtr 1	Qtr 2	Qtr 3	Qtr 4
Ag-110m	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.46E-05	1.91E-05	1.11E-04	2.38E-05
Ar-41	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.75E-06	0.00E+00
As-76	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.74E-06
Co-58	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.75E-04	7.50E-05	5.92E-05	2.53E-04
Co-60	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.10E-04	3.06E-05	2.39E-04	1.39E-04
Cr-51	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.33E-05	0.00E+00	0.00E+00	2.41E-05
Fe-55	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.50E-03	1.76E-03	4.64E-05	0.00E+00
Fe-59	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.37E-06	0.00E+00	0.00E+00	0.00E+00
H-3	Curies	2.01E-01	6.80E-02	1.28E-01	7.53E-02	1.38E+02	1.40E+02	2.99E+02	1.80E+02
Mn-54	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.51E-06	0.00E+00	1.60E-05	3.25E-06
Nb-95	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.99E-06	0.00E+00	0.00E+00	7.03E-06
Nb-97	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.31E-06	1.58E-05	2.09E-05	9.33E-06
Sb-125	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.45E-05	1.47E-04	4.85E-04	7.95E-04
Sn-113	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.61E-07	2.10E-06
Sr-92	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.09E-07	1.41E-06	5.74E-07
Te-123M	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.38E-06	3.49E-06	1.42E-06	3.21E-04
Xe-133	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.01E-05	4.01E-06	8.78E-05	6.26E-05
Xe-135	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.33E-06	1.99E-06
Zr-95	Curies	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.75E-06	0.00E+00	0.00E+00	3.82E-06
Total for Period	Curies	2.01E-01	6.80E-02	1.28E-01	7.53E-02	1.38E+02	1.40E+02	2.99E+02	1.80E+02

Prairie Island Nuclear Generating Station
2018 Annual Dose Summary

Gaseous Effluents

	Parameter	Location	Dose	Dose Limit	% of Limit
Qtr 1	Gamma Air Dose (mrad)	0.58 km W	8.44E-07	1.00E+01	0.00
	Beta Air Dose (mrad)	0.58 km W	1.41E-06	2.00E+01	0.00
	Total Body Dose (mrem)	0.58 km W	7.70E-07	5.00E+00	0.00
	Skin Dose (mrem)	0.58 km W	1.62E-06	1.50E+01	0.00
	Max Organ Dose (mrem) Child - Liver	0.97 km SSE	1.59E-03	1.50E+01	0.01
Qtr 2	Gamma Air Dose (mrad)	0.58 km W	8.11E-08	1.00E+01	0.00
	Beta Air Dose (mrad)	0.58 km W	2.32E-07	2.00E+01	0.00
	Total Body Dose (mrem)	0.58 km W	6.82E-08	5.00E+00	0.00
	Skin Dose (mrem)	0.58 km W	1.62E-07	1.50E+01	0.00
	Max Organ Dose (mrem) Child - Bone	0.97 km SSE	2.65E-02	1.50E+01	0.18
Qtr 3	Gamma Air Dose (mrad)	0.58 km W	1.11E-06	1.00E+01	0.00
	Beta Air Dose (mrad)	0.58 km W	6.36E-06	2.00E+01	0.00
	Total Body Dose (mrem)	0.58 km W	9.35E-07	5.00E+00	0.00
	Skin Dose (mrem)	0.58 km W	4.42E-06	1.50E+01	0.00
	Max Organ Dose (mrem) Child - Bone	0.97 km SSE	3.72E-02	1.50E+01	0.25
Qtr 4	Gamma Air Dose (mrad)	0.58 km W	1.95E-07	1.00E+01	0.00
	Beta Air Dose (mrad)	0.58 km W	4.76E-07	2.00E+01	0.00
	Total Body Dose (mrem)	0.58 km W	1.67E-07	5.00E+00	0.00
	Skin Dose (mrem)	0.58 km W	3.70E-07	1.50E+01	0.00
	Max Organ Dose (mrem) Child - Kidney	0.97 km SSE	2.70E-03	1.50E+01	0.02
Year	Gamma Air Dose (mrad)	0.58 km W	2.23E-06	2.00E+01	0.00
	Beta Air Dose (mrad)	0.58 km W	8.48E-06	4.00E+01	0.00
	Total Body Dose (mrem)	0.58 km W	1.94E-06	1.00E+01	0.00
	Skin Dose (mrem)	0.58 km W	6.58E-06	3.00E+01	0.00
	Max Organ Dose (mrem) Child - Bone	0.97 km SSE	6.37E-02	3.00E+01	0.21

Liquid Effluents

	Parameter	Max Receptor	Dose	Dose Limit	% of Limit
Qtr 1	Max Organ Dose (mrem)	Adult - Gi-LLi	4.64E-04	1.00E+01	0.00
	Total Body Dose (mrem)	Adult - Total Body	4.46E-04	3.00E+00	0.01
Qtr 2	Max Organ Dose (mrem)	Adult - Liver	7.37E-04	1.00E+01	0.01
	Total Body Dose (mrem)	Adult - Total Body	7.23E-04	3.00E+00	0.02
Qtr 3	Max Organ Dose (mrem)	Adult - Gi-LLi	6.49E-04	1.00E+01	0.01
	Total Body Dose (mrem)	Adult - Total Body	6.36E-04	3.00E+00	0.02
Qtr 4	Max Organ Dose (mrem)	Adult - Gi-LLi	5.86E-04	1.00E+01	0.01
	Total Body Dose (mrem)	Adult - Total Body	5.82E-04	3.00E+00	0.02
Year	Max Organ Dose (mrem)	Adult - Gi-LLi	2.44E-03	2.00E+01	0.01
	Total Body Dose (mrem)	Adult - Total Body	2.39E-03	6.00E+00	0.04

ENCLOSURE 3

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

H4 - OFFSITE DOSE CALCULATION MANUAL (ODCM)

DATED – MAY 25, 2018

162 pages to follow

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

OFFSITE DOSE CALCULATION MANUAL (ODCM)

DOCKET NO. 50-282 AND 50-306

INFORMATION USE
• *Procedure may be performed from memory.*
• *User remains responsible for procedure adherence.*
• *Procedure should be available, but not necessarily at the work location.*

PORC REVIEW DATE:	APPROVAL:
5/25/18	PCR #: 616000000031

TABLE OF CONTENTS

TABLE OF CONTENTS [CONTINUED]

LIST OF TABLES

TABLE OF CONTENTS [CONTINUED]

LIST OF FIGURES

LIST OF APPENDICES

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RECORD OF REVISIONS

Revision No.	Date	Reason for Revision
Original	June 7, 1979	
1	April 15, 1980	Incorporation of NRC Staff comments and corrections of miscellaneous errors.
2	August 6, 1982	Incorporation of NRC Staff comments.
3	February 21, 1983	Change in milk sampling location.
4	November 14, 1983	Change in milk sampling location and change in cooling tower blowdown.
5	March 27, 1984	Change Table 3.2-1
6	February 14, 1986	Change in location to collect cultivated crops (leafy green veg.) and removal of meat animals from land use census.
7	July 31, 1986	Retype and format ODCM. No change in content.
8	January 8, 1987	Addition of discharge Canal monitor setpoint calculation.
9	June 29, 1987	Change inhalation dose factor to child and address change in land use survey.
10	April 27, 1989	Change in method for calculating liquid effluent monitor setpoints. Fix of various typing errors. Change in location of two REMP sampling locations. Deletion of one REMP sampling location.
11	October 5, 1989	Change in Tables 3.3-6 thru 3.3-16. Appendix C equations corrected. Section 5 figures replaced. Sample point definitions corrected.
12	June 17, 1991	Change in REMP sampling locations Tables 5.1-1. Added text to address the increased volume of the new discharge pipe.
13	September 27, 1995	Incorporation of RETS as defined in PINGP Technical Specifications in accordance with GL 89-01 as directed by NUREG-1301. Change grab sampling frequency from 8 hours to 12 hours when required on line monitoring equipment is out of service. Define liquid and gaseous monitor calibration. Define radiological effluent and environmental reporting and records retention.
14	May 15, 1996	Correct typing errors and Tech. Spec. references. Update dose factor tables.
15	August 30, 1999	Revised Tech Spec references. Added reference to TBS Landlock. Changed environmental LLDs and reporting level values to reflect "Drinking Water Pathway." Consistent usage of Site Boundary and Unrestricted Area.

RECORD OF REVISIONS [CONTINUED]

Revision No.	Date	Reason for Revision
16	August 1, 2001	Reformatted to M.S. Word. References to Northern States Power Company removed.
17	October 12, 2002	Revised to comply with Improved Technical Specifications. Changed T.S. references, redefined monthly as at least every 31 days, removed all references to the OLD 10 CFR-20 and the MPC liquid release rate limits, increased the size of the airborne release dose factor tables to include all nuclides listed in Reg Guide 1.109, changed REMP milk sampling description to comply NUREG 1301, and a few typographical errors were corrected.
18	June 26, 2003	Adopted airborne radio iodine and particulate sampler locations from NUREG 1301.
19	July 8, 2005	For out-of-service effluent monitoring instrumentation, removed operational time constraints, and added reporting requirements, IAW NUREG 1301. Applicability requirement, for condensate storage tank level instrumentation, was clarified. Updated Site Boundary Map for Liquid Effluents to reflect extension of discharge piping. Various editorial changes.
20	November 6, 2006	Clarification was added to the Basis section, providing guidance for review and approval of monitor set point changes. Direction is that the Operations Committee (OC) will review and approve changes to the ODCM, which includes the methodologies for set point determination. Specific set point changes made in accordance with theses OC reviewed and approved methodologies need not be reviewed by the OC.
21	April 20, 2007	Added the NEI Industry Initiative on Groundwater Protection recommended reporting protocol to Section 8.0, Reporting Requirements. This addition lowers the threshold for reporting of groundwater contamination and clarifies the reporting protocol.
22	June 11, 2008	Revised record retention length for various documents from 5 to Life of the Insurance Policy plus 10 years. NRC Branch Technical Position, Rev 1, November 1979 added to the Critical Receptor Identification, as a compliant alterative approach, when this approach proves to be conservative with regards to dose. Various typographical errors with no change to intent.

RECORD OF REVISIONS [CONTINUED]

Revision No.	Date	Reason for Revision
23	May 29, 2009	Revised Section 8.4 based on guidance in NEI 07-07, "Industry Ground Water Protection Initiative – Final Guidance Document," August, 2007. This revision included the addition of four definitions to the "Definitions" section, an additional condition of Plant Manager discretion for voluntary communication to State and Local official, and the addition of NEI to the list of entities notified in the event of a spill or leak.
24	9/17/09	µ Symbol shows up as an empty box (□)
25	10/21/2010	Revised sections 2.11 and 4.2.1 to remove references to release of Turbine Building Sump water via the land locked discharge pathway. Release to the land locked area was no longer allowed as of 1/8/10. Added Section 8.5 and 8.6 to direct the processing of correspondence with the NRC and other government agencies to be IAW corporate directives.

RECORD OF REVISIONS [CONTINUED]

Revision No.	Date	Reason for Revision
26	4/07/2011	Adopted the language of Technical Specification SR 3.0.2, for section ODCM 1.2, "Surveillance Requirements". The phase operability requirements, "for a Control for operation" was deleted, as undefined and unsupported. In section 2.11, "LANDLOCK AREA", reference to NSP was changed to Prairie Island Nuclear Generating Plant. Methodology for quantification of Carbon-14 curies generated and dose attributed, was added as section 3.5.1. Removed "at least once per" from "The Land Use Census" frequency to read, "between the dates of May 1 and October 31" Entered new calculations for C-14 dose based on Regulatory Guide 1.109 and NUREG -0133 methodologies. - Calculation B.2-9 Moved Ri tables, Historical Meteorological Joint Frequency Tables and dispersion tables to reference document H4.2, "OFFSITE DOSE CALCULATION MANUAL (ODCM) SUPPORTING DATA"

RECORD OF REVISIONS [CONTINUED]

Revision No.	Date	Reason for Revision
27	6/27/2012	The definition of CHANNEL RESPONSE TEST has been deleted. There is no requirement to perform a CHANNEL RESPONSE TEST; therefore the inclusion of the definition is extraneous. The statement that C-14 will not be included in the totals when assessing compliance with specification 3.7.1.A (section 3.5.1.F) has been deleted as inappropriate. The term "GALE Code" is defined and referenced. Beyond the definition, all subsequent use of the term "GALE Code", GALE Code Mix or PWR GALE Code, has been changed to "source term". Corrected DEI definition. Conversion factor basis has always been T1D-14844. Non-gamma emitters, previously treated as a subtraction in the liquid radiation process monitor setpoint calculations will now be treated as a factor, in a similar fashion to gamma emitters. This will reduce calculated setpoints from those previously generated by past methodology generated values. Tritium will be accounted for by bounding calculation. Section 4 and section 5 equations have been restructured to reflect source documentation, with NO change in methodology, other than those identified. This was done to enhance auditability. Discharge Canal Monitor definition was revised to reflect USAR, and to direct the maintaining of the alarm setpoint low, reflecting its function as an atypical release monitor. Table 5.1 specific dispersion factor values have been removed. Dispersion factors are identified as long or short term. The ODCM defines methodology. Specific values generated by the prescribed methodology, will be maintained in supporting documentation (H4.2)

RECORD OF REVISIONS [CONTINUED]

Revision No.	Date	Reason for Revision
28	11/27/2013	Corrected DEI definition. DEI is a Reactor Coolant System Radiochemistry Technical Specification parameter. With the implementation of Alternate Source Term (License Amendment 206/193) the source document for the determination of conversion factors will change from TID-14844 to EPA Federal Guidance Report No. 11. Added the extra tritium samples taken for the NEI Groundwater Protection Initiative to the list of REMP waterborne samples in Table 7.1.
29	8/24/14	Included various reporting criteria, not previously captured in the ODCM: • Reporting criteria when the concentration of liquid effluents exceeds 20 times the limits, as averaged over 1 hour. • Reporting criteria when the dose rates of airborne effluents exceeds 20 times the limits, as averaged over 1 hour. • ISFSI Annual Environmental Report. • ISFSI Annual Radiological Environmental Monitoring Report.
30	2/27/2016	Added paragraph to link to ISFSI License Renewal aging management program.

RECORD OF REVISIONS [CONTINUED]

Revision No.	Date	Reason for Revision
31	10/14/2016	Various typographical and editorial errors Changed the terms “Operational” and “Operability”, to “Functional” and “Functionality” with regards to grading ODCM required equipment availability. Revised various definitions to the standard language of Reg Guide 1.21 and NUREG 0133, with no change to requirements or intent. Revised Surveillance Requirements of section 1.2.2 to standard language of NUREG 1301, with no change to requirements or intent. Changed dose rate limits reference to Technical Specificaitons, rather than 10 CFR 20 Clarified C-14 approach, and reporting, with no change to methodology or reporting Added requirement and methodology to assess Total Dose annually, for demonstrating compliance with 40 CFR 190. Changed the completion due date for the Annual Land Use Census from October 31 to September 30 Added requirement to Annual Radioactive Effluent Report, to include reporting of description of all leaks or spills that are communicated per the Industry Initiative on Groundwater Protection Aligned the ODCM with NUREG 1301, with regards to Ce-144 required LLD Removal of R-15 from the Radioactive Gaseous Effluent Monitoring Instrumentation tables Removed the reporting requirement for Waste Gas Instrumentation in the Annual Radioactive Effluent Report.

Align the ODCM

RECORD OF REVISIONS [CONTINUED]

Revision No.	Date	Reason for Revision
31 (CONT.)		Align the ODCM with NUREG 1301 and Technical Specifications, with regards to CHANNEL CHECK definition. Remove the requirement in Appendix A, to determine dispersion factors in 0.1 mile increments.
32	5/29/18	The methodology for bounding tritium in set point calculations, as developed in revision 27, has been expanded. It now includes hard to detect nuclides, which do not generate a radiation monitor response.

OFFSITE DOSE CALCULATIONS MANUAL INTRODUCTION

The Offsite Dose Calculation Manual (ODCM) describes the methodologies and parameters used in: 1) the calculation of offsite doses resulting from radioactive gaseous and liquid effluents; 2) the calculation of gaseous and liquid effluent monitoring instrumentation Alarm/Trip Setpoints. The methodology stated in this manual is acceptable for use in demonstrating compliance with 10 CFR 20.1301(a)(1), 10 CFR 50.36A, 10 CFR 50, Appendix A (GDC 60 & 64) and Appendix I, and 40 CFR 190.

The ODCM is based on "Radiological Effluent Technical Specification of PWR's (NUREG-0472, October 1978)", "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants (NUREG-0133, October 1978)", and "Offsite Dose Calculation Manual Guidance (NUREG-1301, April 1991). Specific plant procedures have been developed to implement the ODCM.

This manual also includes information related to the RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP). Tables 7.1, 7.2 and 7.3 designate specific sample types, reporting levels and lower limits of detection currently used to satisfy the sampling requirements for the REMP.

This procedure is credited by the ISFSI License Renewal Aging Management Programs for implementing monitoring and for the methodologies and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints, and information related to the REMP all of which aid in determining that the ISFSI radiation levels remain within acceptable limits. Therefore, consider ISFSI License Renewal requirements when revising this procedure (Ref. H76, ISFSI License Renewal Implementation and Aging Management Programs)

Licensee initiated changes to the ODCM:

1. **SHALL** be documented and records of reviews performed **SHALL** contain:

    1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s).

    2. A determination that the change(s) maintain the level of radioactive effluent control required by 10 CFR20.1301(a)(1), 10 CFR 50.36A, 40 CFR 190, 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose or setpoint calculations.

2. **SHALL** become effective upon review and acceptance by the Operations Committee.

3. **SHALL** be submitted to the NRC in the form of a complete legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Report for the period of the report in which the change in the ODCM 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. The date (i.e., month and year) of the change **SHALL** be clearly indicated on the "Record of Revision" page.

DEFINITIONS

- **ABNORMAL RELEASE**

  An unplanned or uncontrolled release of radioactive material from the plant. A release which results from procedural or equipment inadequacies, or personnel errors, that could indicate a deficiency.

- **ACTION**

  ACTION **SHALL** be that part of a specification which prescribes remedial measures required under designated conditions.

- **BATCH RELEASE**

  A BATCH RELEASE is a discharge of liquid or gaseous radioactive effluents of a discrete volume. Prior to release, each batch **SHALL** be isolated and thoroughly mixed for sampling and analysis.

- **CHANNEL CALIBRATION**

  A CHANNEL CALIBRATION **SHALL** be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION **SHALL** encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

- **CHANNEL CHECK**

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

- **CHANNEL FUNCTIONAL TEST**

  A CHANNEL FUNCTIONAL TEST consists of injecting a simulated signal into the channel as close to the primary sensor as practicable to verify that it is FUNCTIONAL, including alarm and/or trip initiating action.

- **CONTINUOUS RELEASE**

  A CONTINUOUS RELEASE is the discharge of liquid or gaseous radioactive effluents of a nondiscrete volume of a system that usually has makeup flow during the release. CONTINUOUS RELEASES are normally sampled and analyzed either during or following the release.

- **DOSE EQUIVALENT I-131**

  DOSE EQUIVALENT I-131 is that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present.

  DEI is a Reactor Coolant System Radiochemistry Technical Specification parameter. With the implementation of Alternate Source Term (License Amendment 206/193) the source document for the determination of conversion factors changes from TID-14844 to EPA Federal Guidance Report No. 11.

- **EXCLUSION AREA BOUNDARY**

  The EXCLUSION AREA is the area encompassed by the EXCLUSION AREA BOUNDARY at a minimum distance of 715 meters from the center of either reactor.

- **GALE CODE**

  GALE (Gaseous and Liquid Effluents) Code refers to the computer modeling of plant effluents, using a combination of input data and hard wired parameters to calculate source terms. The gaseous and liquid source terms presented in the ODCM are calculated using the GALE Code and referenced to USAR tables 9.3-1 and 9.2-3. Throughout the ODCM, the use of the terms "Liquid Source Term" or "Gaseous Source Term" will mean source terms generated using the Gale code.

- **GASEOUS RADWASTE TREATMENT SYSTEM**

  The GASEOUS RADWASTE TREATMENT SYSTEM **SHALL** be any system designated and installed to reduce atmospherically released radioactive effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

GROUNDWATER

All water in the surface soil, the subsurface soil, or any other subsurface water. Ground water is simply water in the ground regardless of its quality, including saline, brackish, or fresh water. Ground water can be moisture in the ground that is above the regional water table in the unsaturated (or vadose) zone, or ground water can be at and below the water table in the saturated zone.

- **LIQUID RADWASTE TREATMENT SYSTEM**

  The LIQUID RADWASTE TREATMENT SYSTEM **SHALL** be any system designated and installed to reduce radioactive effluents by holdup or collecting radioactive materials by means of filtering, evaporation, ion exchange or chemical reaction for the purpose of reducing the total radioactivity prior to release to the environment.

- **LONG TERM RELEASE**

  LONG TERM RELEASES refer to releases that are generally continuous and stable in release rate with some anticipated variation in release rate, such as is experienced in normal ventilation system effluents at a nuclear power plant. Determination of doses due to long-term releases should use the historical annual average relative concentration (X/Q), based on meteorological data summarized.

- **MEMBER OF THE PUBLIC**

  MEMBER OF THE PUBLIC means any individual except when that individual is receiving an occupational dose.

- **FUNCTIONAL - FUNCTIONALITY**

  A system, subsystem, train, component, or device shall be FUNCTIONAL or have FUNCTIONALITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

- **POTENTIAL TO REACH GROUNDWATER**

  SPILLS OR LEAKS with the POTENTIAL TO REACH GROUNDWATER include:

- SPILL OR LEAK directly onto native soil or fill,
- SPILL OR LEAK onto an artificial surface (i.e. concrete or asphalt) if the surface is cracked or the material is porous or unsealed, or
- A SPILL OR LEAK that is directed into unlined on non-impervious ponds or retention basins (i.e., water hydrologically connected to GROUNDWATER).

A SPILL OR LEAK inside a building or containment unit is generally unlikely to reach GROUNDWATER, particularly if the building or containment unit has a drain and sump system.

- **PURGE - PURGING**

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

- **RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP)**

  The RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM is established for monitoring the radiation and radionuclides in the environs of the plant. The program **SHALL** provide representative measurements of radioactivity in the highest potential exposure pathways and verification of the accuracy of potential exposure pathways and verification of the accuracy of the effluent monitoring program and modeling of the environmental exposure pathways. The current methodology used in the conduct of the specifications of the REMP described in the ODCM are defined in the RPIP 4700 series of Radiation Protection Implementing Procedures.

- **SHORT TERM RELEASE**

  SHORT TERM RELEASES refers to releases that are intermittent in radionuclide concentration or flow and defined as less than 500 hours per calendar year, but not more than 150 hours in any quarter, and is subject to more restrictive dispersion factors than long term releases.

- **SITE BOUNDARY**

  That line beyond which the land or property is not owned, leased or otherwise controlled by the licensee. The SITE BOUNDARIES for liquid and gaseous releases are defined in Figures 3.1 and 3.2.

- **SPILL OR LEAK**

  An inadvertent event or perturbation in a system or component performance that releases liquid outside the system or component.

- **SOURCE CHECK**

  A SOURCE CHECK **SHALL** be the quantitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

- **SOURCE CONTAINING LICENSED MATERIAL**

  A liquid, including steam, for which a statistically valid positive result is obtained when the sample is analyzed to the lower limits of detection that are required for radioactive liquid effluents for the isotopes listed in Table 2.1.

- **UNRESTRICTED AREA**

  An UNRESTRICTED AREA **SHALL** be any area, access to which is neither limited nor controlled by the licensee.

- **URANIUM FUEL CYCLE**

  The URANIUM FUEL CYCLE is defined in 40 CFR 190.02(b) as: "The operation of milling of uranium ore, chemical conversion of uranium, isotopic enrichment of uranium, fabrication of uranium fuel, generation of electricity by a light-water-cooled nuclear power plant using uranium fuel, and reprocessing of spent uranium fuel, to the extent that these directly support the production of electrical power for public use utilizing nuclear energy, but excludes mining operations, operations at waste disposal sites, transportation of any radioactive material in support of these operations, and the use of recovered non-uranium special nuclear and by-product materials from the cycle."

- **VENTILATION EXHAUST TREATMENT SYSTEM**

  A VENTILATION EXHAUST TREATMENT SYSTEM **SHALL** be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered safety feature atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

- **VENTING**

  VENTING **SHALL** be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is NOT provided or required during VENTING. Vent, used in system names, does not imply a venting process. The release of air or gases via sampling equipment or instrumentation is not considered a controlled process.

1.0 RADIOLOGICAL EFFLUENT SPECIFICATIONS AND SURVEILLANCE REQUIREMENTS

APPLICABILITY AND SURVEILLANCE REQUIREMENTS

1.1 Specifications

1.1.1 Compliance with the Controls contained within the succeeding text is required during the conditions specified. Upon failure to meet the specifications, the associated ACTION requirements **SHALL** be met.

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

1.2 Surveillance Requirements

1.2.1 Surveillance Requirement **SHALL** be met during the conditions specified for individual specifications unless otherwise stated in an individual Surveillance Requirement.

1.2.2 Each Surveillance Requirement **SHALL** be performed within the specified time interval with the following exceptions:

A. A maximum allowable extension not to exceed 25% of the surveillance interval, but

B. The combined time interval, for any three consecutive surveillance intervals, **SHALL NOT** exceed 3.25 times the specified surveillance interval.

1.2.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 1.2.2, **SHALL** constitute noncompliance with the functionality requirements for a specification. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on nonfunctional equipment.

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2.0 LIQUID EFFLUENTS

2.1 SPECIFICATION – CONCENTRATION

2.1.1 In accordance with T.S. 5.5.4.b the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS **SHALL** be limited to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402 other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration **SHALL** be limited to 2 x 10^{-4} μCi/ml total activity.

2.1.2 **APPLICABILITY**: At all times.

2.1.3 **ACTION**:

A. When the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeds the above limits, immediately restore the concentration to within the above limits.

B. Report all deviations in the Annual Radioactive Effluent Release Report.

C. With liquid effluent release, averaged over 1 hour, exceeding 20 times the limits of concentration in unrestricted areas (10 CFR 20, App. B, Table 2, Column 2) to UNRESTRICTED AREAS, a Licensee Event Report (LER) SHALL be submitted within 60 days

2.2 SURVEILLANCE REQUIREMENTS - CONCENTRATION

2.2.1 Radioactive liquid wastes **SHALL** be sampled and analyzed according to the sampling and analysis program of Table 2.1.

2.2.2 The results of radioactive analysis **SHALL** be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 2.1.1.

2.3 SPECIFICATION – DOSE

2.3.1 In accordance with T.S. 5.5.4.d the dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS **SHALL** be limited to:

A. During any calendar quarter to $\leq$3 mrem to the total body and to $\leq$10 mrem to any organ, and

B. During any calendar year to $\leq$6 mrem to the total body and to $\leq$20 mrem to any organ.

2.3.2 **APPLICABILITY**: At all times.

2.3.3 **ACTION**:

A. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days a Special Report that includes the following information:

1. Identifies the cause(s) for exceeding the limit(s).

2. Defines the corrective actions taken to reduce the release.

3. Defines the corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

2.4 SPECIFICATION REQUIREMENT - DOSE

2.4.1 Cumulative dose contributions for the current calendar quarter and current calendar year **SHALL** be determined at least every 31 days in accordance with the methodology and parameters in Section 4.3of the ODCM.

2.5 SPECIFICATIONS - LIQUID RADWASTE TREATMENT SYSTEMS

2.5.1 In accordance with T.S. 5.5.4.f the LIQUID RADWASTE TREATMENT SYSTEM **SHALL** be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses, due to the liquid effluent, to UNRESTRICTED AREAS would exceed 0.12 mrem to the whole body or 0.4 mrem to any organ in a 31 day period.

2.5.2 **APPLICABILITY**: At all times.

2.5.3 **ACTION**:

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

1. Explanation of why liquid radioactive waste was being discharged without treatment, identification of any non-functional equipment or subsystems, and the reason for the non-functionality.

2. Action(s) taken to restore the non-functional equipment to FUNCTIONAL status, and

3. Summary description of action(s) taken to prevent recurrence.

2.6 SURVEILLANCE REQUIREMENTS - LIQUID RADWASTE TREATMENT SYSTEMS

2.6.1 Doses due to liquid releases **SHALL** be projected at least every 31 days in accordance with the methodology and parameters in Section 4.3 of the ODCM.

2.6.2 The installed LIQUID RADWASTE TREATMENT SYSTEM **SHALL** be considered FUNCTIONAL by meeting the Controls specified in 2.1.1 and 2.3.1.

2.7 SPECIFICATIONS -RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

2.7.1 In accordance with T.S. 5.5.4.a the radioactive liquid effluent monitoring instrumentation channels shown in Table 2.2 **SHALL** be FUNCTIONAL with their alarm/trip setpoints set to ensure that the limits of Specification 2.1.1 are not exceeded. The alarm/trip setpoints of these channels **SHALL** be determined in accordance with the methodology in Section 4.1 of the ODCM.

2.7.2 **APPLICABILITY**: During release via the monitored pathway.

2.7.3 **ACTION**:

A. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive effluents monitored by the effected channel, or declare the channel non-functional, or change the setpoint so it is acceptably conservative.

B. With less than the minimum required radioactive liquid effluent monitoring instrumentation channels FUNCTIONAL, take the Action shown in Table 2.2

C. Report all deviations in the Annual Radioactive Effluent Release Report.

2.8 SURVEILLANCE REQUIREMENTS - RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

2.8.1 Each radioactive liquid effluent monitoring instrumentation channel **SHALL** be demonstrated FUNCTIONAL by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 2.3.

2.9 SPECIFICATION - LIQUID STORAGE TANKS

2.9.1 In accordance with T.S. 5.5.10.c the quantity of radioactive material contained in each of the following tanks **SHALL** be limited to 10 Curies, excluding tritium and dissolved or entrained gases:

A. Condensate Storage Tanks

B. Outside Temporary Storage Tanks

2.9.2 **APPLICABILITY**: At all times.

2.9.3 **ACTION**:

A. With the quantity of radioactive material contained in any of the above listed tanks exceeding the limit in 2.9.1 above, immediately suspend all additions of radioactive materials to the tank and within 48 hours reduce the contents to within the limit.

2.10 SURVEILLANCE REQUIREMENTS - LIQUID STORAGE TANKS

2.10.1 The quantity of radioactive material contained in each of the tanks listed in Specification 2.9.1 **SHALL** be determined to be within the limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

2.11 SPECIFICATION - LANDLOCKED AREA

2.11.1 In accordance with 10CFR20.2001 and NRC interpretations, soil removed from the landlocked area for free release to the UNRESTRICTED AREA **SHALL NOT** contain licensed radioactivity, i.e., radionuclides are detected when the soil sample is analyzed to the LLDs listed in Table 7.3 for sediment.

2.11.2 APPLICABILITY:

A. When the soil in the landlocked area is disturbed (construction occurs in the area or the soil is moved to a new location) and during plant decommissioning.

B. The landlocked area is located near the southwest corner of the Prairie Island reactor building proper. The landlocked area is fully contained within an area controlled by Prairie Island Nuclear Generating Plant.

2.11.3 ACTION:

A. With the quantity of radioactive material contained in the soil exceeding the limit in 2.11.1 above, describe the landlocked location in the 10CFR50.75.g file, conduct a dose assessment, and remediate, as required by applicable regulation.

2.12 SURVEILLANCE REQUIREMENTS – LANDLOCKED AREA

2.12.1 The presence of licensed radioactive material described in specification 2.11.1 **SHALL** be determined by analyzing soil samples of the affected landlocked area when the area is disturbed and during plant decommissioning, as required by applicable regulations.

3.0 GASEOUS EFFLUENTS

3.1 SPECIFICATION - DOSE RATE

3.1.1 In accordance with T.S.5.5.4.g the dose rate due to radioactive materials released in gaseous effluents from the site to areas at or beyond the gaseous SITE BOUNDARY (Figure 3.2) **SHALL** be limited to the following:

A. For Noble Gases: $\leq$500 mrem/yr to the whole body and $\leq$3000 mrem/yr to the skin, and

B. For Iodine-131, Iodine-133, Tritium, and Particulates with half-lives greater than 8 days: $\leq$1500 mrem/yr to any organ.

3.1.2 **APPLICABILITY**: At all times.

3.1.3 **ACTION**:

A. With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limits(s).

B. Report all deviations in the Annual Radioactive Effluent Report.

C. With airborne radioactive release, averaged over 1 hour, exceeding 20 times the limits of concentration in unrestricted areas (10 CFR 20, App. B, Table 2, Column1) a Licensee Event Report (LER) SHALL be submitted within 60 days.

3.2 SURVEILLANCE REQUIREMENTS – DOSE RATE

3.2.1 The dose rate due to noble gases in effluents **SHALL** be determined to be within the above limits of 3.1.1 in accordance with the methodology and parameters in Section 5.2 of the ODCM.

3.2.2 The dose rate due to Iodine-131, Iodine-133, Tritium, and Particulates with half-lives greater than 8 days in gaseous effluents **SHALL** be determined to be within the above limits of 3.1.1 in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 3.1.

3.3 SPECIFICATION - NOBLE GAS AIR DOSE

3.3.1 In accordance with T.S.5.5.4.h the air dose due to noble gases released in gaseous effluents to areas at or beyond the gaseous SITE BOUNDARY (Figure 3.2) **SHALL** be limited to the following:

A. During any calendar quarter: $\leq$10 mrad for gamma radiation and $\leq$20 mrad for beta radiation, and

B. During any calendar year: $\leq$20 mrad for gamma radiation and $\leq$40 mrad for beta radiation.

3.3.2 **APPLICABILITY**: At all times.

3.3.3 **ACTION**:

A. With the calculated dose from the release of radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days a Special Report that includes the following:

1. Identifies the cause(s) for exceeding the limit(s).

2. Defines the corrective actions taken to reduce the release.

3. Defines the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

3.4 SURVEILLANCE REQUIREMENTS – NOBLE GAS AIR DOSE

3.4.1 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases **SHALL** be determined at least every 31 days in accordance with the methodology and parameters in Section 5.3 of the ODCM.

3.5 SPECIFICATION - IODINE-131, IODINE-133, TRITIUM AND PARTICULATE DOSE

3.5.1 In accordance with T.S.5.5.4.i the dose to any organ of a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, Tritium, and all radioactive particulates with a half-life greater than 8 days in gaseous effluents released to areas at or beyond the gaseous SITE BOUNDARY (Figure 3.2) **SHALL** be limited to the following:

A. During any calendar quarter: $\leq$15 mrem to any organ, and

B. During any calendar year: $\leq$30 mrem to any organ.

3.5.2 Carbon 14

A. Carbon-14 contribution to dose **SHALL** be included in the total dose from Iodine-131, Iodine-133, Tritium and Particulates, as specified and defined in section 3.5.1. It is recognized that Carbon-14 is NOT a 10 CFR 50 appendix I dose.

B. Carbon 14 dose contribution **SHALL** be included in the total dose, in determination of total uranium fuel cycle dose, as required by specification 6.2

C. Carbon-14 total curies generated, for a given time period, **SHALL** be determined by calculation, in accordance with the methodologies of "EPRI Estimation of Carbon-14 in Nuclear Power Plant Gaseous Effluents".

D. Carbon-14 total curies released, for a given time period, **SHALL** be equal to the Carbon-14 determined to have been generated. No credit for holdup in the Waste Gas Decay Tanks **SHALL** be taken.

E. Only the portion of Carbon-14 in the Carbon Dioxide (CO2) form is available to enter a viable dose pathway. This is via photosynthesis and incorporation into vegetation, during the daylight growing season. Credit **SHALL** be taken for the portion of Carbon-14 that is in the CO2 form, when performing dose calculations.

3.5.3 APPLICABILITY: - At all times.

3.5.4 ACTION:

A. With the calculated dose from the release of Iodine-131, Iodine-133, Tritium, and Particulates with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days a Special Report that includes the following:

1. Identifies the cause(s) for exceeding the limit(s).

2. Defines the corrective actions taken to reduce the release.

3. Defines the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

3.6 SURVEILLANCE REQUIREMENTS - IODINE-131, IODINE-133, TRITIUM AND PARTICULATE DOSE

3.6.1 Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131, Iodine-133, Tritium, and Particulates with half-lives greater than 8 days **SHALL** be determined at least every 31 days in accordance with the methodology and parameters in Section 5.0 of the ODCM.

3.7 SPECIFICATION - GASEOUS RADWASTE TREATMENT SYSTEMS

3.7.1 In accordance with T.S.5.5.4.f the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM **SHALL** be used to reduce releases of radioactivity when the projected doses due to the gaseous effluents to areas at or beyond the gaseous SITE BOUNDARY (Figure 3.2) would exceed any of the following controls over a 31 day period:

A. 0.4 mrad to air from gamma radiation, or

B. 0.8 mrad to air from beta radiation, or

C. 0.6 mrem to any organ of a MEMBER OF THE PUBLIC.

3.7.2 In accordance with T.S.5.5.10.b the quantity of radioactivity contained in each gas storage tank **SHALL** be limited to $\leq$ 78,800 Curies of noble gases (considered as dose equivalent Xe-133).

3.7.3 The radioactive gas contained in the Waste Gas Treatment System **SHALL NOT** be deliberately discharged to the environment during unfavorable wind conditions when the cooling towers are in operation. For purposes of this specification, unfavorable wind conditions are defined as wind from 5° West of North to 45° East of North at 10 miles per hour or less.

3.7.4 **APPLICABILITY:** At all times.

3.7.5 **ACTION**:

A. With radioactive gaseous waste being discharged without treatment and in excess of the above limits of 3.7.1, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days a Special Report that includes the following information:

1. Identification of any non-functional equipment or subsystems, and the reason for the non-functionality.

2. Action(s) taken to restore the non-functional equipment to FUNCTIONAL status, and

3. Summary description of action(s) taken to prevent recurrence.

B. With the quantity of radioactive material in any gas storage tank exceeding the limits of 3.7.2, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

3.8 SURVEILLANCE REQUIREMENTS - GASEOUS RADWASTE TREATMENT SYSTEMS

3.8.1 Doses due to gaseous releases at and beyond the SITE BOUNDARY **SHALL** be projected at least every 31 days in accordance with the methodology and parameters in the ODCM. A projected dose in excess of the limits of 3.7.1 indicates that appropriate components or subsystems of the GASEOUS RADWASTE TREATMENT SYSTEM must be placed in service to reduce radioactive materials in the gaseous effluents.

3.8.2 The installed Waste Gas Treatment System and the VENTILATION EXHAUST TREATMENT SYSTEM **SHALL** be considered FUNCTIONAL by meeting the Controls specified in 3.1.1, 3.3.1 AND 3.5.1.

3.8.3 The quantity of radioactive material contained in each gas storage tank in use **SHALL** be determined to be within the limit specified in 3.7.2 at least every 31 days. If the inventory of any tank exceeds 10,000 curies, daily sampling when making additions **SHALL** be performed.

3.9 SPECIFICATIONS - EXPLOSIVE GAS MONITORING INSTRUMENTATION

3.9.1 In accordance with T.S.5.5.10.a the explosive gas monitoring instrumentation channels shown in Table 3.2 **SHALL** be FUNCTIONAL with their Alarm/Trip Setpoints set to ensure the limits of 3.9.2 are not exceeded.

3.9.2 The concentration of oxygen at the outlet of each operating recombiner **SHALL** be maintained to $\leq$2% by volume.

3.9.3 **APPLICABILITY**: As shown in Table 3.2.

3.9.4 **ACTION**:

A. With an explosive gas monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the specification of 3.9.2, declare the channel non-functional and take the ACTION shown in Table 3.2.

B. With less than the minimum required explosive gas monitoring instrumentation channels FUNCTIONAL, restore the non-functional instrumentation to FUNCTIONAL status within 30 days. If unsuccessful, in lieu of a License Event Report, prepare and submit a Special Report to the Commission explaining why this non-functionality was not corrected in a timely manner. The Special Report SHALL be submitted within 60 days.

C. With the concentration of oxygen measured at the outlet of operating recombiner(s) >2% by volume but <4% by volume, restore the concentration of oxygen to ≤2% by volume within 48 hours.

D. With the concentration of oxygen measured at the outlet of operating recombiner(s) >4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to ≤2% within one hour.

3.10 SURVEILLANCE REQUIREMENTS EXPLOSIVE GAS MONITORING INSTRUMENTATION

3.10.1 Each explosive gas monitoring instrumentation channel **SHALL** be demonstrated FUNCTIONAL by performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION at the frequencies shown in Table 3.3.

3.11 SPECIFICATION - RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

3.11.1 In accordance with T.S.5.5.4.a the radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.2 **SHALL** be FUNCTIONAL with their alarm/trip setpoints set to ensure that the limits of Specification 3.1.1are not exceeded. The alarm/trip setpoints of these channels **SHALL** be determined in accordance with the methodology in Section 5.0 of the ODCM.

3.11.2 **APPLICABILITY**: As shown in Table 3.2.

3.11.3 ACTION:

A. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive effluents monitored by the effected channel, or declare the channel non-functional, or change the setpoint so it is acceptably conservative.

B. With less than the minimum required radioactive gaseous effluent monitoring instrumentation channels FUNCTIONAL, take the Action shown in Table 3.2.

C. Report all deviations in the Annual Radioactive Effluent Release Report.

3.12 SURVEILLANCE REQUIREMENTS - RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

3.12.1 Each radioactive gaseous effluent monitoring instrumentation channel **SHALL** be demonstrated FUNCTIONAl by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 3.3.

3.13 SPECIFICATION - ATMOSPHERIC STEAM DUMP MONITORING

3.13.1 The dose to a MEMBER OF THE PUBLIC from Iodine-131 released, via one steam dump operation, in gaseous effluents from the site at or beyond the gaseous SITE BOUNDARY (Figure 3.2) **SHALL NOT** be greater than twice the limit specified in 3.5.1.

3.13.2 **APPLICABILITY**: During atmospheric steam dump operations with detectable Iodine-131 activity in the Steam Generator bulk water.

3.13.3 ACTION:

A. When the calculated dose from the release of Iodine-131 in gaseous effluents via steam dump operations exceeds the above limit:

1. The milk from dairy cows grazing in the downwind area **SHALL** be sampled and analyzed for a period of 5 days following the release. The downwind area shall include the 22 1/2 degree sector of a circle having it's center at the plant and a 2 mile radius.

2. The Iodine-131 concentration in the milk **SHALL** be determined daily utilizing instrumentation with a minimum Iodine-131 detection limit of 1.0 pCi/ml.

3.14 SURVEILLANCE REQUIREMENTS - ATMOSPHERIC STEAM DUMP MONITORING

3.14.1 The Iodine-131 activity released via atmospheric steam dumps **SHALL** be sampled and analyzed according to the sample and analysis program of Table 3.1.

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4.0 LIQUID EFFLUENT CALCULATIONS

4.1 Monitor Alarm Setpoint Determination

This procedure determines the monitor alarm setpoint that indicates if the concentration of radionuclides in the liquid effluent released to UNRESTRICTED AREAS exceeds the specification defined in Section 2.1.1.

Fe-55, Ni-63 Sr-89, Sr-90, and alpha concentrations are determined from composite samples. Composite results will be used as input to the liquid monitor setpoint determinations, using the most recent values available, or performing bounding calculations per the methodology of section 4.1.1.

Monitor high alarm or isolation setpoints will be established or verified each time a release permit is generated, by the methodology described in section 4.1.1 and 4.1.2. Nuclide mix input to the high alarm or isolation setpoint will be:

1. The Liquid Source Terms (Table 4.1).
    1. Used in the case that no gamma emitters are identified in the batch tank pre-release samples or for continuous releases which are not anticipated to have gamma emitters and are not evaluated pre-release.
2. Based on analysis prior to discharge.
    1. Used in the case of that gamma emitters are identified in the batch tank pre-release samples.

In the event that no release is made for a given liquid release pathway and therefore no evaluation of the associated liquid process radiation monitor is made, a setpoint calculation will be performed based on Liquid Source Terms (Table 4.1) at least once every 31 days.

If the calculated setpoint is less than the existing monitor setpoint, the setpoint **SHALL** be reduced to the new value. If the calculated setpoint is greater than the existing setpoint, the setpoint may remain at the lower value or increased to the calculated value.

Setpoint calculations are performed each time a release permit is generated.

H3 and hard to detect nuclides, such as Fe-55 and Ni-63, may constitute a significant portion of the nuclide mix, however will not generate a monitor response (cpm). A bounding calculation may be performed, and a correction factor generated, as a basis for negating these nuclides from the set point calculations, as defined in section 4.1.1

4.1.1 Setpoint Safety Factor Determination:

A. Determine the maximum postulated activity for H3 and select hard to detect nuclides.

B. Determine the maximum postulated diluted concentration in the discharged effluent, based on 110% of rated pump discharge flow rate and minimum allowable dilution flow rate, for each nuclide:

$$ACT_{DIL} = ACT_{MAX} * F_{PMP} / F_{DIL} \quad (4.1\text{-}1)$$

Where:

ACT_{DIL}	-	Diluted Activity concentration (µCi/ml)
ACT_{MAX}	-	Maximum postulated Activity (µCi/ml)
F_{PMP}	-	110% of rated pump flow (gpm)
F_{DIL}	-	Minimum dilution flow (gpm)

C. Determine maximum postulated contribution to ECL fraction:

Maximum postulated contribution to ECL fraction =

$$\frac{\text{H3 } ACT_{DIL}}{\text{H3 ECL}} + \frac{\text{Nuclide 1 } ACT_{DIL}}{\text{Nuclide 1 ECL}} + \frac{\text{Nuclide 2 } ACT_{DIL}}{\text{Nuclide 2 ECL}} \quad (4.1\text{-}2)$$

D. Determine the Setpoint Safety Factor (SPSF): (4.1-3)

SPSF = 1 - Maximum postulated contribution to ECL Fraction

The ECL limit (typically 1) is reduced to account for a nuclide's or nuclides' bounding contribution to ECL and is represented in the set point calculation as a reduced value of the Setpoint Safety Factor (SPSF).

The bounding calculation will be maintained in supporting documentation and validated as appropriate.

4.1.2 Liquid Effluent Monitor Setpoints

The following methodology applies when determining the isolation setpoints for:

- Waste Effluent Liquid Monitor (R-18)
- Steam Generator Blowdown Liquid Monitor - Unit 1 (1R-19)
- Steam Generator Blowdown Liquid Monitor - Unit 2 (2R-19)

The following calculations assume the radioactive waste liquid discharge flow rate will be maintained constant, at the maximum design flow rate and that dilution flow will be maintained constant at a minimum flow rate.

A. Nuclide "mix" Determination (representative liquid source terms of the liquid effluent)

1. For short term batch releases, the gamma and tritium source terms will be determined, by analysis, prior to release.

2. In the absence of quantification of gamma source terms, the Liquid Source Terms (Table 4.1) may be used, generating a default set point.

3. Hard to detect nuclides, such as Fe-55, Ni-63, Sr-89 and Sr-90, will be determined by quarterly composite samples. Input to the source term will be the most recent values, unless they have been bounded and accounted for, as described in section 4.1.1.

B. Required Dilution Factor (RDF) Determination

$$RDF = \frac{\Sigma(AC_i/ECL_i)}{DSF} \qquad (4.1\text{-}4)$$

Where:

RDF - Require Dilution Factor (unitless)
AC_i - Activity Concentration of nuclide "i" (µCi/ml)
ECL_i - Effluent Concentration Limit of nuclide "i" (µCi/ml)
DSF - Dilution Safety Factor; 0.8

C. Specific Activity (SP) Determination

1. Specific Activity equates all nuclide activities determined to be in the mix to the equivalent Cs-137 activity, based on monitor response.

2. If the gamma activity concentration is 0, then the Specific Activity need not be determined. The setpoint will be the default setpoint, based on the Liquid Source Terms (Table 4.1).

3. The allocation factor is the fraction of the radioactivity from the site that may be released via each release point to ensure that the unrestricted area limit is not exceeded due to simultaneous releases from multiple release points. The summation of all the allocation factors for active release points **SHALL NOT** be greater than one.

$$SP = \frac{\Sigma(AC_i * CsEq_i) * SPSF * AF * (F_{DIL} + F_{WST})}{RDF * F_{WST}} \qquad (4.1\text{-}5)$$

Where:

SP - Specific Activity, adjusted for monitor response (µCi/ml)
AC_i - Activity Concentration of nuclide "i" (µCi/ml)
$CsEq_i$ - Cs-137 Equivalence, monitor response of nuclide "i"
SPSF - Setpoint Safety Factor
AF - Allocation Factor
F_{DIL} - Dilution Flow (gpm)
F_{WST} - Waste Flow (gpm)
RDF - Required Dilution Factor

D. Liquid Set Point (LSP) Determination

1. If no gamma emitter activity is quantified, then the Liquid Set Point Calculation is the default Liquid Set Point value, based on Liquid Source Terms (Table 4.1).

2. If the Waste Flow exceeds the Maximum Permissible Waste Flow then the Liquid Set Point is zero and no release is permitted.

$$LSP = e^{COA + (COB * ln(SP))} \qquad (4.1\text{-}6)$$

Where:

LSP - Liquid Set Point (cpm)
COA - Monitor Calibration Coefficient A
COB - Monitor Calibration Coefficient B
SP Specific Activity adjusted for monitor response . (µCi/ml)

3. The monitor high alarm setpoint above background (ncpm), SHALL be set at or below the LSP value.

4.1.3 Maximum Waste Flow Determination

As a further verification, and to ensure the release concentration limits are not exceeded, and to identify flow throttling requirements, a maximum waste flow rate is determined, as follows:

If the Required Dilution Factor is < 1, the release rate limit is unlimited.

If the Required Dilution Factor is $\geq$ 1, then:

$$F_{MAX} = \frac{F_{DIL} * (1 - \Sigma(DAC_i / ECL_i)) * AF * SPSF}{RDF} \tag{4.1-7}$$

Where:

F_{MAX} - Maximum Permissible Waste Flow (gpm)
F_{DIL} - Dilution Flow (gpm)
DAC_i - Dilution Activity concentration of nuclide "i" (µCi/ml)
ECL_i - Effluent Concentration of nuclide "i" (µCi/ml)
AF - Allocation Factor
SPSF - Setpoint Safety Factor
RDF - Required Dilution Factor

In the absence of activity in the dilution water, the equation becomes:

$$F_{MAX} = \frac{F_{DIL} * AF * SPSF}{RDF} \tag{4.1-8}$$

4.1.4 Discharge Canal Monitor (R-21)

The Discharge Canal Monitor (R-21) provides direct measurement of the diluted plant effluent concentration, monitoring the various streams feeding the discharge canal, with the exception of the Waste Liquid Discharge Header.

The Waste Liquid Discharge Header is extended to the end of the discharge canal to a point just upstream of the river release sluice gates. This line effectively bypasses R-21.

The Waste Liquid Discharge Header effluents are Steam Generator Blowdown releases and Radioactive Liquid Waste Tank Batch releases from the Auxiliary Building.

The activity of effluents released from the site, other than the Waste Liquid Discharge Header are typically negligible. R-21 is an atypical release monitor, for the purpose of identifying unanticipated releases.

The Discharge Canal Monitor alarm set point is set at a minimal value to detect minimal activity, without generating spurious alarms.

4.1.5 Rad Effluent Monitor Calibration

Liquid effluent monitors are calibrated periodically using a Cs-137 standard. Since the actual isotopic mixes of the liquids released may contain nuclides with different gamma energies and yields than the calibration standard, the response of the monitor varies with respect to the actual energies and abundances of the nuclides in the mix being monitored when compared to Cs-137.

Setpoint determinations or expected monitor readings during or prior to a release are compensated for the difference in gamma energies and yields. The monitor setpoint calculations and predicted monitor readings are adjusted according to reflect the actual nuclide mix.

The assumption is made that the monitor's response is directly proportional to the gamma energies and yields.

The cumulative errors associated with the monitor calibration methodology are not accounted for in the determination of individual monitor setpoints. Sufficient conservatism is built into the monitor setpoint determination, such as the required dilution safety factor. Additionally, the use of allocation factors would require that all release paths exceed their respective monitor set points before the limits of ten times the water effluent concentrations of 10 CFR 20, Appendix B, Table 2, Column 2 (ECLs) were challenged.

4.2 Compliance With 10 CFR 20

In order to comply with 10 CFR 20, in accordance with T.S.5.5.4.b, the concentrations of radionuclides in liquid effluents will not exceed 10 times the water effluent concentrations as defined in Appendix B, Table 2, Column 2 of 10 CFR 20. For CONTINUOUS RELEASES, the alarm trip setpoints discussed in Section 4.1 will assure that these concentrations are not exceeded. For BATCH RELEASES, concentrations of diluted effluents will be compared to effluent concentrations limits pre-release, providing protection in addition to the alarm trip setpoints discussed in Section 4.1.

4.2.1 Continuous Releases

Continuous liquid releases can occur from PINGP through Steam Generator Blowdown. The alarm trip setpoints discussed in Section 4.1 will assure that releases from this pathway will not exceed the limits of ten times the water effluent concentrations of 10 CFR 20, Appendix B, Table 2, Column 2.

Other continuous releases occur at PINGP, through the turbine building sump system. These releases are minor. A continuous composite sample will be maintained at the discharge from the turbine building sump with samples being taken and analyzed weekly. If these samples indicate significant levels of radionuclides, the methodologies given in section 4.2.2 will be applied to the turbine sump weekly releases and the limit in Equation 4.1-6, as input to Steam Generator Blowdown and BATCH RELEASES, will be lowered to account for this source term. This will be done by the adjustment of allocation factors to these releases.

4.2.2 Batch Releases

To demonstrate compliance with 10 CFR 20, Appendix B, Table 2, Column 2, the radioactivity content of each BATCH RELEASE will be determined prior to release. The concentration of the various radionuclides in the BATCH RELEASE prior to dilution, is divided by the minimum dilution flow to obtain the concentration at the UNRESTRICTED AREA.

$$Conc_i = \frac{C_i \, F_{WST}}{F_{DIL}} \qquad (4.2\text{-}1)$$

Where:

$Conc_i$	-	Concentration of radionuclide i at the site boundary (µCi/ml)
C_i	-	Concentration of radionuclide i in the potential batch release
F_{WST}	-	Release rate of the batch
F_{DIL}	-	minimum dilution flow (65,900 gpm)

In accordance with T.S.5.5.4.b, the projected concentration at the UNRESTRICTED AREA is compared to the ten times the water effluent concentrations of Appendix B, Table 2, Column 2 of 10 CFR 20. Before a release may occur, Equation 4.2-2 must be met for all isotopes.

$$\Sigma_i = \frac{Conc_i}{ECL_i} \qquad (4.2\text{-}2)$$

ECL_i - Ten times the water effluent concentration of radionuclide i, from 10 CFR 20, Appendix B, Table 2, Column 2 (µCi/ml)

The summation of all source terms, as input to the total contribution to ECL **SHALL NOT** be greater than 1.0.

The volume of the discharge pipe could contain the volume of 2 to 3 waste batch tanks. To ensure compliance with 10CFR20 when the maximum acceptable discharge flow rate, as calculated in section 4.1.3, is less than the maximum possible release rate from all release sources, the discharge pipe **SHALL** be flushed with a volume of at least the volume of the discharge pipe. The flush rate **SHALL NOT** exceed the maximum discharge flow rate and may be accomplished with water from other release paths. If more than one waste batch tank requiring flushing are to be released, the discharge pipe may be flushed following the final tank release.

Volume of discharge pipe = 15,500 gal.

4.3 Liquid Effluent Dose - Compliance with 10 CFR 50

Doses resulting from liquid effluents will be calculated at least every 31 days to show compliance with 10 CFR 50. A cumulative summation of total body and organ doses for each calendar quarter and calendar year will be maintained as well as projected doses for the next month.

Since Fe-55, Sr-89, Sr-90, and alpha concentrations are determined from composite samples, the monthly liquid effluent dose calculations and comparisons to quarterly and annual limits should be completed using the most recent available composite sample results. The quarterly and annual dose calculations **SHALL** be completed using the actual composite sample results.

The limits of 10 CFR 50 are on a per reactor unit basis. The liquid radwaste system at PINGP is shared by both reactor units making it impossible to separate the releases of the two units. The releases that can be separated by unit, for steam generator blowdown and turbine building sump releases, contribute a very small portion of the total liquid releases from PINGP. Therefore, for compliance with 10 CFR 50, the releases from both units will be summed and the limits of Appendix I will be doubled.

4.3.1 Dose Calculations

The dose contribution from the release of liquid effluents will be calculated for each release permit and will be assessed at least every 31 days. The dose contribution will be calculated using the following:

A. Waste Flow Fraction Determination

To determine doses from liquid effluents, the waste flow fraction for the period of release must be calculated. This dilution factor must be calculated for each BATCH RELEASE and each CONTINUOUS RELEASE mode. The waste flow fraction is determined by:

$$WFF = \frac{F_{WST}}{(F_{DIL} + F_{WST}) * MF} \qquad (4.3\text{-}1)$$

where:

WFF - Waste Flow Fraction (unitless)
F_{WST} - Waste Flow (gpm)
F_{DIL} - Dilution Flow (gpm)
MF - Mixing Factor*

* The value of MF is the site specific factor for the mixing effect of the PINGP discharge structure. This value is 10 for PINGP while operating in the closed cycle cooling mode. In once through, or helper mode, the value of MF is 1.0.

A waste flow value of the rated pump flow and a dilution value of 65,900 gpm (147 CFS) is used on dose projections when generating a release permit. Actual values are used for final reported dose.

B. Effluent Activity Determination

$$EC_i = WC_i * WFF \qquad (4.3\text{-}2)$$

Where:

EC_i - Effluent Concentration for nuclide i (µCi/ml)
WC_i - Waste Concentration for nuclide I (µCi/ml)
WFF - Waste Flow Fraction (unitless).

C. Effluent Concentration Duration Determination

$$ECD_i = EC_i * T \qquad (4.3\text{-}3)$$

Where:

ECD$_i$ - Effluent Concentration Duration (µci-hr/ml)
T - Release Duration (hours)

D. Dose Determination

$$D_T = \frac{\Sigma(ECD_i * U_{ap} * Bi_p * DF_{ari}) * 10^9}{DIL * 8760} \qquad (4.3\text{-}4)$$

Where:

D_T - the dose commitment to the total body or any organ τ, from the liquid effluents for the period of release (mrem)
ECD$_i$ - Effluent Concentration Duration for nuclide i (µCi-hr/ml)
U_{ap} - Consumption rate for age group a, pathway p (Kg/year)
B_{ip} - Bioaccumulation Factor for nuclide i for pathway p
DF_{ari} - Dose Factor for age group a, receptor r, nuclide i (mrem/hour / pCi/L)
10^9 - Conversion factor (µCi/ml to pCi/L)
DIL - Dilution Factor between discharge to collection point
8760 - hours per year

The factor $A_{i\tau}$ assesses and accounts for the site specific inputs to dose. The factor $A_{i\tau}$ also includes the correction factors of equation 4.3-4. A_{iT} must be reassessed and updated if assessment of specific site dose inputs should change, as identified in the Annual Land Use Census. For instance, a change to A_{iT} could be required in the event that the Mississippi River were to be used as a potable water source.

By employing the A_{iT} factor the dose equation can be reduced to:

$$D_T = \Sigma(ECD_i * A_{iT} * DF_{ARI}) \qquad (4.3\text{-}5)$$

$A_{i\tau}$ is the site related ingestion dose commitment factor to the total body or any organ τ for each identified principal gamma and beta emitter (mrem/hr per µCi/ml)

The dose factor $A_{i\tau}$ was calculated for an adult for each isotope using the following equation:

$$A_{i\tau} = 1.14x10^5[21B_{if}DF_{Ai\tau}] \qquad (4.3\text{-}6)$$

Where:

1.14×10^5	-	10E+06 pCi/uCi * 1E+03 ml/L * 1 year/8,760 hours
21	-	adult fish consumption (Kg/yr)
B_{if}	-	Bioaccumulation factor for radionuclide i, pathway fish, from Table A-1 of Regulatory Guide 1.109 Rev. 1 (5) (pCi/Kg per pCi/l)
$DF_{Ai\tau}$	-	Dose conversion factor for radionuclide i for adults for a particular organ τ from Table E-11 of Regulatory Guide 1.109 Rev. 1, (5) (mrem/pCi)

Mississippi River water is not used as a potable water supply within 300 miles downstream of the PINGP. Wells are used for irrigation downstream of the plant.

Applicable pathway(s) and age group(s) are determined by the Annual Land Use Census. If changes to the $A_{i\tau}$ is required, calculations are performed using the methodologies of Regulatory Guide 1.109 Rev. 1. The current values are captured in Table 4.2.

A table of $A_{i\tau}$ values, for an adult age group and a fish pathway, are presented in Table 4.2.

4.3.2 Accumulation of Doses

Doses calculated at least every 31 days will be summed for comparison with quarterly and annual limits. The monthly results should be added to the doses cumulated from the other months in the quarter of interest and in the year of interest for the combined releases of both reactor units and compared to the limits given in Section 2.3.

The quarterly limits represent one half of the annual design objective. If these quarterly or annual limits are exceeded, a special report should be submitted to the USNRC identifying the cause and corrective action to be taken. If twice the quarterly or annual limits are exceeded, a special report **SHALL** be submitted showing compliance with 40 CFR 190.

4.3.3 Projection of Doses

Anticipated doses resulting from the release of liquid effluents will be projected monthly. If the projected doses would exceed 2 percent of the limit specified in Section 2.3.1, the liquid radwaste treatment system will be used to process waste (T.S. 5.5.4.f).

Projected dose will be the dose for the preceding 31 days, as calculated by Equation 4.3-4.

The total source term utilized for the most recent dose calculation should be used for the projections unless information exists indicating that actual future releases could differ significantly. In this case, the source term would be adjusted to reflect this information and the justification for the adjustment noted. This adjustment should account for any radwaste equipment which was operated during the previous month that could be out of service in the coming month.

4.4 References

4.4.1 "Prairie Island Final Environmental Statement," USAEC, May, 1973, p. V-26.

4.4.2 "Prairie Island Nuclear Generating Plant, Appendix I Analysis – Supplement No.1 – Docket No. 50-282 and 50-306," Table 2.1-1.

4.4.3 "10 CFR 20," Appendix B, Table II, Column 2.

4.4.4 "Prairie Island Nuclear Generating Plant, Appendix I Analysis – Supplement No. 1 - docket 50-282 and 50-306," July 21, 1976, Table 2.1-2.

4.4.5 U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.109 – Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Compliance with 10 CFR 50, Appendix I," Rev. 1, 1977.

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5.0 GASEOUS EFFLUENT CALCULATIONS

5.1 Monitor Alarm Setpoint Determination

This procedure determines the monitor alarm setpoint that indicates if the dose rate beyond the SITE BOUNDARY due to noble gas radionuclides in the gaseous effluent released from the site exceeds 500 mrem/year to the whole body or exceeds 3000 mrem/year to the skin.

Monitor high alarms or isolation setpoints will be established each time a release permit is generated, calculated by the methodology described in section 5.1.1. Nuclide mix input to the high alarm or isolation setpoint will be:

1. The Gaseous Source Terms (Table 5.2).
    1. Used in the case that no gamma emitters are identified in the pre-release samples or for continuous releases which are not anticipated to have gamma emitters and are not evaluated pre-release.
2. Pre-release analysis results
    1. Used in the case that gamma emitters are identified in the batch release pre-release samples.

If the calculated setpoint is less than the existing monitor setpoint, the setpoint **SHALL** be reduced to the calculated setpoint. If the calculated setpoint is greater than the existing setpoint, the setpoint may remain at the lower value or increased, not to exceed the calculated set point.

5.1.1 Effluent Monitors

The following method applies when determining the isolation or high alarm setpoint for the monitors listed in Table 5.1.

A. Determine the “mix” (noble gas radionuclides and composition) of the gaseous effluent. This is the gaseous source terms that are representative of the gaseous effluent. Gaseous source terms are the total curies of each noble gas. If measured gas source terms are below the lower limits of detection (LLD), Table 5.2 source terms are the mix.

B. Determine the maximum effluent release rate in µCi/sec, for Whole Body Dose Limits.

$$MRR_{WB} = \frac{500 * SPSF * AF * \sum(NG_i)}{\sum(NG_i * K_i) * X/Q} \qquad (5.1\text{-}1)$$

Where:

MRR_{WB}	-	Max Effluent Release Rate based on Whole Body Dose Rate Limit (µCi/sec)
500	-	Whole Body Dose Rate Limit (mrem/year)
SPSF	-	Setpoint Safety Factor; ($\leq$ 1)
AF	-	Allocation Factor
NG_i	-	Noble Gas "i" Concentration (µci/cc)
K_i	-	The total whole body dose factor due to gamma emissions from noble gas radionuclide "i" from Table 5.4 (mrem/year/µci/m^3)
X/Q	-	Highest calculated annual average relative concentration of effluents released via the plant vents for any area at or beyond the site boundary, for all sectors. (Table 5.1)

C. Determine the maximum effluent release rate in µCi/sec, for Skin Dose Limits.

$$MRR_{SKIN}(uCi/\sec) = \frac{3000 * SPSF * AF * \Sigma(NG_i)}{\Sigma(NG_i * (L_1 + 1.1M_i)) * X/Q} \qquad (5.1\text{-}2)$$

Where:

MRR_{SKIN}	-	Max Effluent Release Rate based on Skin Dose Rate Limit (µCi/sec)
3000	-	Skin Dose Rate Limit (mrem/year)
SPSF	-	Setpoint Safety Factor
AF	-	Allocation Factor
NG_i	-	Noble Gas "i" Concentration (µCi/cc)
$L_i + 1.1M_i$	-	The total skin dose factor due to gamma and beta emissions from noble gas radionuclide "i" from Table 5.4 (mrem/year/µCi/m^3)
X/Q	-	Highest calculated annual average relative concentration of effluents released via the plant vents for any area at or beyond the site boundary, for all sectors, from Table 5.1

D. Define the limiting maximum effluent release rate (MRR), as the lesser value generated (µCi/sec), per equations 5.1-1 and 5.1-2, as input to subsequent calculations.

E. Determine the maximum effluent concentration (µCi/cc).

$$\mathrm{MEC} = \frac{\mathrm{MRR}}{(\mathrm{FWST} + \mathrm{FDIL}) * 472} \qquad (5.1\text{-}3)$$

MEC	-	Maximum effluent concentration (µCi/cc)
MRR	-	Maximum Effluent Release Rate (µCi/sec)
FWST	-	Waste Flow (cfm)
FDIL	-	Dilution Flow (cfm)
472	-	Conversion factor

In the absence of dilution flow (typical), the equation becomes:

$$\mathrm{MEC} = \frac{\mathrm{MRR}}{\mathrm{FWST} * 472} \qquad (5.1\text{-}4)$$

F. Determine Xe-133 fraction of the total radioactivity in the gaseous effluent.

$$S_{xe} = \frac{A_{xe\text{-}133}}{A_{total}} \qquad (5.1\text{-}5)$$

Where:

$A_{xe\text{-}133}$ - The radioactivity of Xe-133 in the gaseous effluent (µCi/cc)
A_{total} - The total radioactivity of noble gas radionuclides in the gaseous effluent (µCi/cc)

G. Determine Non-Xe-133 fraction of the total radioactivity in the gaseous effluent comprised by all noble gases, excluding Xe-133.

$$S_i = \frac{A_i}{A_{total}} \qquad (5.1\text{-}6)$$

Where:

A_i - The total radioactivity in the gaseous effluent comprised by all noble gases, excluding Xe-133.
A_{total} - The total radioactivity of noble gas radionuclides in the gaseous effluent.

H. Determine the setpoint contribution from Xe-133 (cpm).

$$SP_{Xe\text{-}133} = e^{(XCOA + (XCOB * \ln(MEC * Sxe-133)))} \qquad (5.1\text{-}7)$$

WHERE:

$SP_{Xe\text{-}133}$ - Gas Set Point Contribution from Xe-133 (cpm)
XCOA - Monitor Xe-133 Calibration Coefficient A
XCOB - Monitor Xe-133 Calibration Coefficient B
MEC - Maximum Effluent Concentration

I. Determine the Monitor High Set Point.

$$HSP = [SP_{Xe-133} + e^{COA+(COB*\ln(MEC*S))}] * (29 - VAC/29) \qquad (5.1\text{-}8)$$

Where:

HSP	-	High Set Point (cpm)
COA	-	Monitor Non-Xe-133 Calibration Coefficient A
COB	-	Monitor Non-Xe-133 Calibration Coefficient B
MEC	-	Maximum Effluent Concentration
29-VAC/29	-	Correction for vacuum of the monitor

The isolation or high alarm setpoint above background (ncpm) for the monitors should be set at or below the HSP value.

5.1.2 Air Ejector Monitors

Radiation monitors 1R-15 and 2R-15 provide an indication of gross noble gas activity at the main condenser air ejector of Unit 1 and Unit 2, respectively. These monitors are provided to give rapid indication of steam generator tube leakage. They are not effluent monitors since the air ejectors are vented to the auxiliary building vents during normal plant operation and releases are monitored by the auxiliary building vent monitoring system.

5.1.3 Monitor Calibration

Gaseous effluent monitors are calibrated periodically. Available gas mixes existing in plant systems may be used. Since effluent gas mixes vary in isotopic ratios and the energies of those isotopes span a range of energies, more than one gas mix is used during the calibration.

One calibration is performed with a mix that is predominately Xe-133 with lower level beta and gamma energies. A second calibration is performed with a mix containing longer lived plant gases that more accurately represent the higher beta energy range.

The result of this method of calibration is two separate calibration curves for each monitor. The Xe-133 curve is applied to setpoint calculations for the Xe-133 activity. The second curve is applied to setpoint calculations for balance of noble gases activities.

Setpoint determination and projected monitor reading during release utilize a combination of the two calibration curves, according to the actual nuclide mix.

The cumulative errors associated with the monitor calibration methodology are not accounted for in the determination of the individual monitor setpoints. There is sufficient conservatism built into the selection of the actual monitor setpoint. Additionally, the monitor fractions used in the setpoint determination equation make it necessary for all the effluent monitors to be in alarm before the limits of 10CFR Part 20 would be exceeded.

5.2 Gaseous Effluent Dose Rate - Compliance with 10 CFR 20

Dose rates resulting from the release of noble gases, and radioiodines and particulates must be calculated to show compliance with 10 CFR 20. The limits of 10 CFR 20 must be met on an instantaneous basis at the hypothetical worst case location, and apply on a per site basis.

Releases made via the shield building vents as a result of routine surveillance tests or scheduled short term maintenance/work activities of 2 hours or less do not require the sampling and analysis of shield building vent stack samples described in Table 3.1 for the following reasons:

1. Shield building effluent particulates and iodines are filtered through a PAC (Particulate Absolute Charcoal) system and the auxiliary building vent normal ventilation has no filtration.

2. The lower limit of detection limits specified in Table 3.1 can not be obtained on all the specified nuclides with normal sample flow and sample duration of less than 2 hours.

3. Shield building vent releases are monitored via a noble gas monitor.

4. Auxiliary building normal ventilation flow is higher than the special ventilation fans that vent via the shield building vent stack.

Therefore, it is conservative to assume that the auxiliary building normal ventilation system would continue to run during the testing/maintenance period. The surveillance test or maintenance/work being performed should be evaluated to ensure the airborne activity in the affected areas will not increase during the evolution. If this evaluation indicates a possible increase in airborne effluents, or radiation monitors or continuous air monitors in the affected buildings indicate higher than normal background airborne activity before the evolution begins, the shield building vent stack sample **SHALL** be sampled and analyzed as described in Table 3.1.

Since Sr-89 and Sr-90 concentrations are determined from composite samples, the pre-release, weekly and monthly airborne dose calculations and comparisons to quarterly and annual limits should be completed using the most recent available composite sample results. The quarterly dose values and critical receptors reported to the USNRC **SHALL** be calculated using the actual composite results.

5.2.1 Noble Gases

5.2.2 To comply with the 10 CFR 20 dose limit of 100 mrem TEDE to MEMBERS OF THE PUBLIC, the dose rate at the SITE BOUNDARY resulting from noble gas effluents is limited, per Techical Specification 5.5.4.g.1, to 500 mrem/yr to the total body and 3000 mrem/yr to the skin. The setpoint determinations discussed in the previous section are based on the dose calculation method presented in NUREG-0133. They represent a backward solution to the limiting dose equations in NUREG-0133. Setting monitor alarm trip points in this manner will assure that the limits of Technical Specification 5.5.4 are met for noble gas releases. Therefore, no routine dose calculations for noble gases will be needed to show compliance with this part. Routine calculations will be made for doses from noble gas releases to show compliance with 10 CFR 50, Appendix I as discussed in Section 5.3.1.

5.2.3 Radioiodine, Radioactive Particulates, and Other Radionuclides

For compliance with Techical Specification 5.5.4.g.2, the dose rate at the SITE BOUNDARY resulting from the release of radioiodine and particulates with half lives greater than 8 days is limited to 1500 mrem/yr to any organ. Calculations showing compliance with this dose rate limit will be performed for BATCH RELEASES prior to the release. To show compliance, Equations 5.2-1 will be evaluated using I-131, I-133, tritium, and radioactive particulates with half-lives greater than eight days.

$$\sum P_i [Q_{iv} * X/Q_v] \ * < 1500 \text{ mrem/year} \qquad (5.2\text{-}1)$$

Where:

P_i - Child critical organ dose parameter for radionuclide i for the inhalation pathway, from Table 5.3 (mrem/yr per μCi/m³)

(X/Q_v) - Annual average relative concentration for LONG TERM release at the critical location, from H4.2, "Offsite Dose Calculation Manual (ODCM) Supporting Data" (sec/m³)

Q_{iv} - Total release rate of radionuclide i from all vents from both units for the batch or week of interest (μCi/sec)

Radioiodines, tritium, and radioactive particulates will be released from up to six individual vents all within 300 feet of each other. For showing compliance with Techical Specification 5.5.4.g.2, calculations based on Equation 5.2-1 will be made each release. The source terms (Q_{iv}) will be determined from the results of analysis of vent particulate filters and charcoal canisters and vent flow rate. These source terms include all gaseous releases from PINGP.

5.2.4 Critical Receptor Identification

Compliance with 10 CFR 20 radiation dose limits for individual MEMBERS OF THE PUBLIC will be demonstrated by identifying critical receptor locations based on 10 CFR 50 App I ALARA design objectives. Since the doses associated with 10 CFR 50 are more restrictive than the 10 CFR 20 limits, this method satisfies the 10 CFR 20 requirements.

5.3 Gaseous Effluents - Compliance with 10 CFR 50

Doses resulting from the release of noble gases, radioiodines and particulates must be calculated to show compliance with Appendix I of 10 CFR 50. The calculations will be performed at least every 31 days for all gaseous effluents.

The limits of 10 CFR 50 are on a per reactor unit basis. The GASEOUS RADWASTE TREATMENT SYSTEM and the auxiliary building at PINGP is shared by both reactor units making it impossible to separate the releases of the two units. The releases that can be separated by unit contribute a very small portion of the total gaseous releases from PINGP. Therefore, for compliance with 10 CFR 50 the releases from both units will be summed and the limits of Appendix I will be doubled.

Releases made via the shield building vents as a result of routine surveillance tests or scheduled short term maintenance/work activities of 2 hours or less do not require the sampling and analysis of shield building vent stack samples described in Table 3.1 for the following reasons:

1. Shield building effluent particulates and iodines are filtered through a PAC (Particulate Absolute Charcoal) system and the auxiliary building vent normal ventilation has no filtration.
2. The lower limit of detection limits specified in Table 3.1 can not be obtained on all the specified nuclides with normal sample flow and a sample duration of less than 2 hours.
3. Shield building vent releases are monitored via noble gas monitor.
4. Auxiliary building normal ventilation flow is higher than the special ventilation fans that vent via the shield building vent stack.

Therefore, it is conservative to assume that the auxiliary building normal ventilation system would continue to run during the testing/maintenance period. The surveillance test or maintenance/work being performed should be evaluated to ensure the airborne activity in the affected areas will not increase during the evolution. If this evaluation indicates a possible increase in airborne effluents, or radiation monitors or continuous air monitors in the affected buildings indicate higher than normal background airborne activity before the evolution begins, the shield building vent stack sampled **SHALL** be sampled and analyzed as described in Table 3.1.

Since Sr-89 and Sr-90 concentrations are determined from composite samples, the pre-release, weekly and monthly airborne dose calculations and comparisons to quarterly and annual limits should be completed using the most recent available composite sample results. The quarterly dose values and critical receptors reported to the USNRC **SHALL** be calculated using the actual composite results.

5.3.1 Noble Gas

A. Dose Equations

The air dose at the critical receptor due to noble gases released in gaseous effluents is determined by Equations 5.3-1 and 5.3-2. The critical receptor will be identified as described in Section 5.3.4. For gamma radiation:

$$GammaAirDose = 3.17X10^{-8}\sum[M_i * (X/Q_v * Q_{iv}) + (x/q_v * q_{iv}))] \tag{5.3-1}$$

Gamma Air Dose Limits:

< 10 mrad for any calendar quarter
< 20 mrad for any calendar year

$$BetaAirDose = 3.17X10^{-8}\sum[N_i * ((X/Q_v * Q_{iv}) + (x/q_v * q_{iv}))] \tag{5.3-2}$$

Beta Air Dose Limits:

< 20 mrad for any calendar quarter
< 40 mrad for any calendar year

Where:

3.17×10^{-8} The inverse of the number of seconds in a year.

M_i - The air dose factor due to gamma emission for each identified noble gas radionuclide i, from Table 5.4 (mrad/yr per μCi/m^3)

N_i - The air dose factor due to beta emission for each identified noble gas radionuclide i, from Table 5.4 (mrad/yr per μCi/m^3)

X/Q_v - The annual average relative concentration for areas at or beyond the restricted area boundary for LONG TERM vent releases from H4.2, "Offsite Dose Calculation Manual (ODCM) Supporting Data" Table 6.0,(sec/m^3).

Q_{iv} - The total release of noble gas radionuclide i in gaseous effluents for LONG-TERM vent releases from both units (µCi)

$(x/q)_v$ - The relative concentration for areas at or beyond the restricted area boundary for SHORT-TERM vent releases, from H4.2, "Offsite Dose Calculation Manual (ODCM) Supporting Data" Table 6.0,(sec/m^3).

q_{iv} - The total release of noble gas radionuclide in gaseous effluents for SHORT-TERM vent releases from both units (µCi)

Noble gases will be released from PINGP from up to six vents.

LONG-TERM (X/Q) and SHORT-TERM (x/q) dispersion factors were calculated using the USNRC computer code "XOQDOQ" assuming 100 hours per year SHORT TERM RELEASES (H4.2, "Offsite Dose Calculation Manual (ODCM) Supporting Data"). Values of M and N are taken directly from Reg Guide 1.109 and are given in Table 5.4.

B. **Accumulation of Doses**

Doses calculated monthly will be summed for comparison with quarterly and annual limits. The monthly results will be added to the doses calculated from the other months in the quarter of interest and the year of interest and compared to the limits given in Section 3.3. If these limits are exceeded, a special report will be submitted to the USNRC. If twice the limits are exceeded, a special report showing compliance with 40 CFR 190 will be submitted.

5.3.2 Radioiodine, Particulates, and Other Radionuclides

A. Dose Equations

The worst case dose to an individual from I-131, I-133, tritium, and radioactive particulates with half-lives greater than eight days in gaseous effluents released beyond the SITE BOUNDARY is determined by the following expressions:

Dose due to I-131, I-133, Tritium and Radioactive Particulates with half-lives greater than eight days =

$$3.17x10^{-8} \sum_j \sum_i R_{ijak}\left[(W_v * Q_{IV}) + (w_v * q_{iv}\right] \qquad (5.3\text{-}3)$$

< 15 mrem (per quarter)

< 30 mrem (per calendar year)

Where:

The W and w values are in terms of X/Q (sec/m^3) for the inhalation pathways, tritium and Carbon-14. For all other pathways and/or nuclides the W and w values are in terms of D/Q (1/m^2). Current dispersion factors are maintained in H4.2, "Offsite Dose Calculation Manual (ODCM) Supporting Data"

3.17×10^{-8}	-	the inverse of the number of seconds in a year (sec^{-1})
R_{ijak}	-	the dose factor for each identified radionuclide i, pathway j, age group a, and organ k, m^2 mrem/yr per µCi/sec or mrem/yr per µCi/ m^3.
W_v	-	Dispersion (deposition) parameter for estimating the dose to an individual at the controlling location for LONG-TERM vent releases
Q_{iv}	-	release of radionuclide i for LONG-TERM vent releases from both units (µCi)
w_v	-	Dispersion (deposition) parameter for estimating the dose to an individual at the controlling location for SHORT-TERM vent releases
q_{iv}	-	release of radionuclide i for SHORT-TERM purge releases from both units (µCi)

Equation 5.3-3 will be applied to each combination of age group and organ. Values of R_{ijak} have been calculated using the methodology given in NUREG-0133 and are maintained in H4.2, "Offsite Dose Calculation Manual (ODCM) Supporting Data". Dose factors for isotopes not listed will be determined in accordance with the methodology in Appendix B.

B. **Accumulation of Doses**

Doses calculated monthly will be summed for comparison with quarterly and annual limits. The monthly results should be added to the doses cumulated from the other months in the quarter of interest and in the year of interest and compared with the limits in Section 3.5.1. If these limits are exceeded, a special report will be submitted to the USNRC. If twice the limits are exceeded, a special report showing compliance with 40 CFR 190 will be submitted.

5.3.3 Projection of Doses

Doses resulting from the release of gaseous effluents will be projected at least every 31 days. The doses calculated for the present month will be used as the projected doses unless information exists indicating that actual releases could differ significantly in the next month. In this case the source terms will be adjusted to reflect this information and the justification for the adjustment noted. If the projected release of noble gases for the month exceeds 2 percent of the calendar year limits of section 3.3.1, additional waste gas treatment will be provided. If the projected release of I-131, I-133, tritium, and radioactive particulates with half-lives greater than 8 days exceeds 2 percent of the calendar year limit of equation 5.3-3, operation of the ventilation exhaust treatment equipment is required if not currently in use.

5.3.4 Critical Receptor Identification

To demonstrate compliance with 10 CFR 50 App I ALARA design objectives, two locations of interest are identified.

For noble gases effluents, the location identified for demonstrating compliance will be based on the beta and gamma air doses only. This location will be the offsite location with the highest long term vent X/Q values maintained in H4.2, "Offsite Dose Calculation Manual (ODCM) Supporting Data". This location will remain the same unless meteorological data is reevaluated or the SITE BOUNDARY changes.

For the I-131, I-133, tritium, long-lived particulate and C-14 effluents, the location indentfied for demonstrating compliance will be based on maxiumum organ dose. It will be known as the Critical Receptor Location. The selection will follow the annual land use census, performed within 5 miles of the PINGP. Each of the following locations will be evaluated as potential critical receptors.

1. Residence in each sector
2. Vegetable garden producing leafy green vegetables
3. All identified milk animal locations

Following the annual survey, doses will be calculated using Equation 5.3-3 for all new identified receptors and those receptors whose characteristics have changed significantly. The calculation will include appropriate information about each new location. The dispersion parameters given in this manual should be employed. The total releases reported for the previous calendar year should be used as the source terms.

In certain cases, the Critical Receptor identified may not produce conservative doses in comparison to a past Critical Receptor. A past Critical Receptor may no longer qualify, based on such criteria as discontinuing the maintenance of a qualifying garden. In this case the option to consider a qualifying garden to still exist may be chosen, when doses may be proven to be conservative, with regards to the newly identified Critical Receptor, based on radioactive effluent releases. This position complies with the U.S. Nuclear Regulatory Commission Branch Technical Position, Revision 1, dated November, 1979.

5.4 References

5.4.1 "Prairie Island Nuclear Generating Plant, Appendix I Analysis - Supplement No. 1 -Docket No. 50-282 and 50-306", Table 2.1-4.

5.4.2 "10 CFR 20"

5.4.3 "10 CFR 50" Appendix I

5.4.4 U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.109 Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Compliance with 10 CFR 50, Appendix I", Rev. 1, 1977.

5.4.5 U.S. Nuclear Regulatory Commission, NUREG 0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants", dated October, 1978.

6.0 TOTAL DOSE FROM RADIOACTIVE RELEASES AND URANIUM FUEL SOURCES

6.1 SPECIFICATION – TOTAL DOSE

6.1.1 In accordance with T.S.5.5.4.j and 40 CFR 190 the annual dose or dose commitment to any MEMBER OF THE PUBLIC, beyond the SITE BOUNDARY, due to releases of radioactivity and to radiation from URANIUM FUEL CYCLE sources **SHALL** be limited to less than or equal to 25 mrems to the whole body or any organ, except the thyroid, which **SHALL** be limited to less than or equal to 75 mrems.

6.1.2 **APPLICABILITY**: At all times

6.1.3 **ACTION**:

A. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 2.3.A, 2.3.B, 3.3.A, 3.3.B, 3.5.A, or 3.5.B, calculations **SHALL** be made including direct radiation contributions from the reactor units (including outside storage tanks and the ISFSI) to determine whether the above limits have been exceeded and to determine compliance with 40 CFR 190. If such is the case, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, a Special Report that includes the following:

1. Defines the corrective action(s) to be taken to reduce subsequent releases to prevent reoccurrence of exceeding the above limits.

2. Includes the schedule for achieving conformance with the above limits.

3. This special report as defined in 10 CFR 20.2203(a), **SHALL** include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report.

4. Describe levels of radiation and concentrations of radioactive material involved, and cause of the exposure levels and concentrations.

5. If the estimated dose(s) exceed the above limits, and if the release condition resulting in violation of 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.

6.2 SURVEILLANCE REQUIREMENTS – TOTAL DOSE

6.2.1 Cumulative dose contributions from liquid and gaseous effluents **SHALL** be determined in accordance with Surveillance Requirements 2.4, 3.4, and 3.6, and in accordance with the methodology and parameters in the ODCM.

6.2.2 Cumulative direct radiation dose contributions, from the reactor units and the ISFSI **SHALL** be determined:

A. When the conditions set forth in ACTION (a) of Specification 6.1 are met, AND

B. In demonstration of compliance with the limits of 40 CFR 190, for preparation of the Annual Radioactive Effluent Report.

6.3 METHODOLOGY – TOTAL DOSE

6.3.1 Direct dose component SHALL be assessed

A. Direct dose component, due to gamma radiation, SHALL be assessed per the methodology of ANSI/HPS N13.37-2014, Environmental Dosimetry – Criteria for System Design and Implementation.

B. Neutron shine dose, due to the ISFSI, SHALL be assessed.

6.3.2 Organ dose contribution, due to release of Iodine-131, Iodine-133, Tritium, Particulates with half-lives greater than 8 days and Carbon-14 in gaseous effluents SHALL be assessed.

6.3.3 An evaluation of the dose contributions from the sources noted in 6.3.1, 6.3.2 and 6.3.3 will be performed to determine the maxium individual dose location.

6.3.4 Noble gas dose (mrem) to the maximum individual dose location will be determined as follows:

A. Gamma Dose (mrem)

= 3.17E-08 * SF * ((X/Q * ∑(K * Q)) + (x/q *∑(K * q)))

Where:

3.17E-08 - years/sec
SF - occupancy factor (typically 1)
X/Q - Long Term Release Dispersion Factor (sec/m3)
Ki - Nuclide i Total Body Dose factor due to gamma emissions (mrem/year per μCi/m3)
Qi - Total Release of noble gas i, released as a continuous release (μCi)
x/q - Short Term Release Dispersion Factor (sec/m3)
qi - Total Release of noble gas i, released as a batch release (μCi)

B. Beta Dose (mrem) =

3.17E-08 * SF * ((X/Q * ∑((L+1.1M) * Q)) + (x/q *∑((L+1.1M) * q)))

Where:
3.17E-08 - years/sec
SF - occupancy factor (typically 1)
X/Q - Long Term Release Dispersion Factor (sec/m3)
Li - Skin Dose factor due to beta emissions for nuclide i (mrem/year per μCi/m3)
Mi - Air Dose Factor due to gamma emissions for nuclide i (mrad/year per μCi/m3)
1.1 - mrem/mrad

Qi - Total Release of noble gas i, released as a continuous release (μCi)
x/q - Short Term Release Dispersion Factor (sec/m3)
qi - Total Release of noble gas i, released as a batch release (μCi)

6.3.5 Doses SHALL be totaled from the direct radiation dose and effluent dose contributors, for the maximum individual dose location.

6.3.6 Liquid effluent dose is not required to be assessed for the determination of compliance with 40 CFR 190 limits.

7.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

MONITORING PROGRAM

SPECIFICATIONS

7.1 In accordance with T.S.5.5.1 the Radiological Environmental Monitoring Program (REMP) **SHALL** be conducted as specified in Table 7.1.

APPLICABILITY At all times.

ACTION

1. Whenever the Radiological Environmental Monitoring Program is not being conducted as described in Table 7.1 the Annual Radiological Environmental Monitoring Report **SHALL** include a description of the reasons for not conducting the program as required and the plans for the prevention of a recurrence.

2. Deviations are permitted from the required sampling schedule if samples are unobtainable due to hazardous conditions, seasonable unavailability, or to malfunctions of automatic sampling equipment. If the latter occurs, every effort **SHALL** be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule **SHALL** be reported in the Annual Radiological Environmental Monitoring Report.

3. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 7.2 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Special Report that includes the following:

    1. Identifies the cause(s) for exceeding the limit(s).

    2. Defines the corrective actions that have been taken to reduce radioactive effluents so that the potential annual dose[1] to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 2.3.1, 3.3.1, or 3.5.1.

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[1] The Methodology and parameters used to estimate the potential annual dose to a member of the public **SHALL** be indicated in the report.

When more than one of the nuclides in Table 7.2 are detected in the sampling medium, this report **SHALL** be submitted if:

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

When nuclides other than those in Table 7.2 are detected and are the result of plant effluents, this report **SHALL** be submitted if the potential annual dose to a MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Specifications 2.3.1, 3.3.1, or 3.5.1 This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition **SHALL** be reported and described in the Annual Radiological Environmental Monitoring Report.

4. Although deviations from the sampling schedule are permitted under Paragraph b. above, whenever milk or leafy vegetation samples can no longer be obtained from the designated sample locations required by Table 7.1, the Annual Radiological Environmental Monitoring Report **SHALL** explain why the samples can no longer be obtained and identify the new locations added to and deleted from the monitoring program.

SURVEILLANCE REQUIREMENTS

7.2 The radiological environmental monitoring samples **SHALL** be collected pursuant to Table 7.1 from the specific locations of the radiological environmental monitoring sampling program described in the Radiation Protection Implementing Procedure (RPIP) 4700, and **SHALL** be analyzed pursuant to the requirements of Table 7.1 and the detection capabilities required by Table 7.3.

LAND USE CENSUS

SPECIFICATIONS

7.3 A Land Use Census **SHALL** be conducted and **SHALL** identify:

1. The location of the nearest milk animal, the nearest residence, and the nearest garden of greater than 500 ft^2 producing fresh leafy vegetation in each of the 16 meteorological sectors within a distance of 5 miles.

2. Fields or gardens of greater than 500 ft^2 producing corn that are irrigated with water taken from the Mississippi River between the plant and a point 5 miles downstream.

APPLICABILITY

At all times.

ACTION

1. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 3.6, in lieu of a Licensee Event Report, identify the new location(s) in the next Annual Radiological Environmental Monitoring Report.

2. With the Land Use Census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 7.1, add the new location(s) to the Radiological Environmental Monitoring Program within 30 days. The sampling location(s) excluding the control station location, having a lower calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program. Identify the new location(s) in the next Annual Radiological Environmental Monitoring Report.

3. If fields or gardens larger than 500 ft^2 producing corn are being irrigated with Mississippi River water, appropriate samples **SHALL** be collected and analyzed per Table 7.1.

SURVEILLANCE REQUIREMENTS

7.4 The Land Use Census **SHALL** be conducted during the growing season, between the dates of May 1 and September 30 by door to door survey, aerial survey, or by consulting local agricultural authorities or associations. A summary of the results of the land use census **SHALL** be included in the Annual Radiological Environmental Monitoring Report.

INTERLABORATORY COMPARISON PROGRAM

SPECIFICATIONS

7.5 An analysis **SHALL** be performed on radioactive materials, supplied by an NRC approved crosscheck program. This program involves the analyses of samples provided by a control laboratory as well as with other laboratories which receive portions of the same samples. Media used in this program (air, milk, water, etc.) **SHALL** be limited to those found in the radiation environmental monitoring program.

APPLICABILITY

At all times.

ACTION

1. When required analyses are not performed, corrective action **SHALL** be reported in the Annual Radiological Environmental Monitoring Report.

SURVEILLANCE REQUIREMENTS

7.6 The summary results of analyses performed as part of the above required Interlaboratory Comparison Program **SHALL** be included in the Annual Radiological Environmental Monitoring Report.

8.0 REPORTING REQUIREMENTS

8.1 Annual Radioactive Effluent Report

Annual Radioactive Effluent Reports include, T.S.5.6.3 "The Annual Radioactive Effluent Report" and ISFSI T.S 5.3, "The ISFSI Annual Environmental Report". These reports have different submittal dates and are generated separately.

8.1.1 The Annual Radioactive Effluent Report

In accordance with T.S.5.6.3 the Annual Radioactive Effluent Report covering the operation of the units **SHALL** be submitted in accordance with 10 CFR 50.36A and **SHALL** include:

1. The Annual Radioactive Effluent Report covering the operation of the plant during the previous calendar year **SHALL** be submitted by May 15 of each calendar year to the Administrator of the appropriate Regional NRC office or designee.

2. The Annual Radioactive Effluent Report **SHALL** include a summary of the quantities of radioactive liquid and gaseous effluents released from the plant as outlined in Appendix B of Regulatory Guide 1.21, Revision 1, June, 1974, with data summarized on a quarterly basis. In the event that some results are not available for inclusion with the report, the report **SHALL** be submitted noting and explaining the reasons for the missing results. The missing data **SHALL** be submitted as soon as possible in a supplementary report.

3. The Annual Radioactive Effluent Report **SHALL** include an assessment of the radiation doses from radioactive effluents released from the plant during the previous calendar year. The report **SHALL** also include an assessment of the radiation doses from radioactive liquids and gaseous effluents to individuals due to their activities inside the SITE BOUNDARY (Figures 3.1 and 3.2) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) **SHALL** be included in the report.

4. The Annual Radioactive Effluent Report **SHALL** include the following information for solid waste shipped offsite during the report period.
    1. Container volume,
    2. Total curie quantity (specify whether determined by measurement or estimate),
    3. Principal radionuclides (specify whether determined by measurement or estimate),

4. Type of waste (e.g., spent resin, compacted dry waste, evaporated bottoms),
5. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and
6. Solidification agent (e.g., cement, urea formaldehyde).

5. The Annual Radioactive Effluent Report **SHALL** include ABNORMAL RELEASES from the site of radioactive materials in gaseous and liquid effluents on a quarterly basis.

6. The Annual Radioactive Effluent Report **SHALL** also include an assessment of radiation doses to the most likely exposed MEMBER OF THE GENERAL PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show compliance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation

7. The Annual Radioactive Effluent Report **SHALL** include a description (including cause, response and prevention of reoccurrence) of occurrences when the sampling frequency, minimum analysis frequency, or lower limit of detection requirements specified in Tables 2.1 and 3.1 were not met.

8. The Annual Radioactive Effluent Report **SHALL** include a description of occurrences when less than the minimum required radioactive liquid and/or gaseous effluent monitoring instrumentation channels were functional as required in Tables 2.2 and 3.2.

9. The Annual Radioactive Effluent Report **SHALL** include a description of the circumstances which caused the failure to complete the minimum sample and/or analysis frequency required by Tables 2.1 and 3.1. The report **SHALL** include the actions taken to restore the sampler, actions taken to prevent recurrence, and a summary of the occurrences effect on the analysis validity.

10. The Annual Radioactive Effluent Report **SHALL** include a description of the circumstances which result in LLD's higher than those listed in Tables 2.1 and 3.1.

11. The Annual Radioactive Effluent Report **SHALL** include an assessment of the radiation doses from radioactive effluents released from the ISFSI during the previous calendar year.

12. Licensee initiated changes to the ODCM **SHALL** be submitted to the NRC in the form of a complete legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Report for the period of the report in which the change in the ODCM 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. The date (i.e., month and year) of the change **SHALL** be clearly indicated on the Record of Revisions page.

13. The Annual Radioactive Effluent Report **SHALL** include description of changes to the Process Control Program.

n. The Annual Radioactive Effluent Report SHALL include a description of all leaks or spills that are communicated per section 8.4 below, including a dose assessment.

8.1.2 ISFSI Annual Environmental Report

In accordance with ISFSI T.S 5.3, the ISFSI Annual Environmental Report SHALL be submitted to the NRC Region III, Office, with a copy to the Director, Office of Nuclear Material Safety and Safeguards in accordance with 10 CFR 72.44(d)(3) and SHALL include:

1. The ISFSI Annual Environmental Report SHALL be submitted within 60 days after January 1 of each year.

b. This report should specify the quantity of each of the principal radionuclides released to the environment in liquid and in gaseous effluents during the previous year of operation and such other information as may be required by the Commission to estimate maximum potential radiation dose commitment to the public resulting from effluent release.

8.2 Annual Radiological Environmental Monitoring Report

The Annual Radiological Environmental Monitoring Report, covering the operation of the offsite monitoring program is submitted in accordance with T.S.5.6.2. The ISFSI Annual Radiological Environmental Monitoring Report, covering the operation of the ISFSI monitoring program is submitted in accordance with ISFSI T.S.5.2. These reports are typically submitted as a single report. The two reports are inclusively referred to as the Annual Radiological Environmental Monitoring Report.

1. The Annual Radiological Environmental Monitoring Report covering the operation of the plant during the previous calendar year **SHALL** be submitted by May 15 of each year to the Administrator of the appropriate Regional NRC office or his designee.

2. The Annual Radiological Environmental Monitoring Report **SHALL** include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report **SHALL** be submitted noting and explaining the reasons for the missing results. The missing data **SHALL** be submitted as soon as possible in a supplementary report.

3. The Annual Radiological Environmental Monitoring Report **SHALL** include summaries, interpretations, and an analysis of trends of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The report **SHALL** also include a summary of the results of the land use census. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report **SHALL** provide an analysis of the problem and a planned course of action to alleviate the problem.

4. The Annual Radiological Environmental Monitoring Report **SHALL** also include the following: a summary description of the radiological environmental monitoring program; a map of sampling locations within a distance of five miles keyed to a table giving distances and directions from the reactor; and the results of licensee participation in the Interlaboratory Comparison Program.

5. The Annual Radiological Environmental Monitoring Report **SHALL** include reasons for all deviations from the REMP sampling program as specified in Table 7.1 and plans for the prevention of a recurrence, if applicable.

6. The Annual Radiological Environmental Monitoring Report **SHALL** contain a description of when and why milk or leafy vegetable samples specified in Table 7.1 cannot be obtained from the designated sample locations, and identify the new locations added to and deleted from the monitoring program.

7. If the level of radioactivity in an environmental sampling medium at a specified location exceeds the reporting levels of Table 7.2 for the sample type specified in Table 7.1 and is NOT the results of plant effluents, the condition **SHALL** be reported in the Annual Radiological Environmental Monitoring Report.

8. A summary of the Interlaboratory Comparison Program **SHALL** be included in the Annual Radiological Environmental Monitoring Report. If the required Interlaboratory Comparison Program analyses are NOT performed, corrective action **SHALL** be reported in the Annual Radiological Environmental Monitoring Report

9. The Annual Radiological Environmental Monitoring Report **SHALL NOT** include the Complete Analysis Data Tables. These contain the results of each sample analysis and **SHALL** be maintained by the licensee.

10. The Annual Radiological Environmental Monitoring Report **SHALL** include all on-site and off-site groundwater sample results taken in support of the Industry Initiative.

11. The Annual Radiological Environmental Monitoring Report **SHALL** include a description of all leaks or spills that are communicated per section 8.4 below.

8.3 Annual Summary of Meteorological Data

An annual summary of meteorological data **SHALL** be submitted, at the request of the Commission, for the previous calendar year in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.

8.4 Industry Initiative on Groundwater Protection

NOTE: For purposes of this section, groundwater is defined as any subsurface moisture or water, regardless of where it is locked beneath the earth's surface; any water located in wells, regardless of depth, type, or whether it is potable; water in storm drains, unless it has been demonstrated that the storm drains do not leak to ground; and water in sumps that communicate with subsurface water.

1. 30-day Report to the NRC

    1. Submit to the NRC within 30 days, a special report for any on-site or off-site GROUNDWATER sample that:

        - Exceeds the ODCM criteria for 30-day reporting for off-site samples(see Section 7.0); and

- Has a POTENTIAL TO REACH GROUNDWATER that is or could be used in the future as a source of drinking water. Any GROUNDWATER that is potable should be considered as a potential source of drinking water.

The initial discovery of GROUNDWATER contamination greater than the REMP reporting criterion is the event documented in a written 30-day report. It is not expected that a written 30-day report will be generated each time a subsequent sample(s) suspected to be from the same "plume" identifies concentrations greater than any of the REMP criteria as described in the ODCM. Evaluate the need for additional reports or communications based on unexpected changes in conditions.

2. The 30-day special report should include:

   - A statement that the report is being submitted in support of the Groundwater Protection Initiative,
   - A list of the contaminant(s) and verified concentration(s),
   - Description of the action(s) taken.
   - An estimate of the potential or bounding annual dose to a member of the public, and
   - Corrective action(s), if necessary, that will be taken to reduce the projected annual dose to a member of the public to less than the limits in 10 CFR 50 Appendix I.

3. Concurrently, provide copies of the 30-day written report to the designated State and Local Officials.

2. Voluntary Communications to State and Local Officials

   1. Make informal communications by end of next business day to the designated State and Local officials if a SPILL OR LEAK has the POTENTIAL TO REACH GROUNDWATER and exceeds any of the following criteria:

      - If a SPILL OR LEAK exceeding 100 gallons from a source containing licensed material,
      - If the volume of a SPILL OR LEAK cannot be quantified but is likely to exceed 100 gallons from a source containing licensed material, or
      - Any SPILL OR LEAK, regardless of volume or activity, is deemed by the Plant Manager or designee to warrant voluntary communication.

2. Communication with the designated State and Local officials **SHALL** be made before the end of the next business day for a water sample result of:

   - Off-site GROUNDWATER or surface water that exceeds any of the REMP reporting criteria for water as described in the ODCM (see Section 7.0), or

   - On-site surface water that is hydrologically connected to GROUNDWATER, or GROUNDWATER that is or could be used as a source of drinking water, that exceeds any of the REMP reporting criteria for water as described in the ODCM.

   Document the basis for concluding that on-site GROUNDWATER is not or would not be considered a source of drinking water. Examples of a defensible basis are documents from the regulatory agency with jurisdiction over GROUNDWATER use.

3. When communicating with State and Local officials, be clear and precise when quantifying the actual release information as it applies to the appropriate regulatory criteria (i.e. put it in perspective). The following information should be provided as part of the information communication:

   - A statement that the communication is being made as part of the NEI Groundwater Protective Initiative,
   - The date and time of the SPILL OR LEAK, or sample result(s),
   - Whether or not the spill has been contained or the leak has been stopped,
   - If known, the location of the SPILL OR LEAK or water sample(s),
   - The source of the SPILL OR LEAK, if known,
   - A list of the contaminant(s) and the verified concentration(s),
   - Description of the action(s) already taken and a general description of future actions,
   - An estimate of the potential or bounding annual dose to a member of the public if available at this time, and
   - An estimated time/date to provide additional information or follow-up.

4. Following communication with State/Local officials, complete a 4-hour 10CRF50.72 NRC notification.

5. Contact NEI by email address GW_Notice@nei.org with the information provided to the State Local Officials.

8.5 Record Retention

8.5.1 Records will be retained for the "Life of Insurance Policy, plus ten (10) years".

8.5.2 Records to be retained include, but not limited to, the following:

A. Periodic checks, inspections, tests and calibrations of components and systems as related to the specifications and treatment systems defined in the ODCM.

B. Records of wind speed and direction.

C. Liquid and airborne radioactive releases to the environment.

D. Off-site environmental monitoring surveys.

E. Records of reviews performed for changes made to the Offsite Dose Calculation Manual.

8.6 Official correspondences with the NRC and other government agencies **SHALL** be processed IAW:

1. CP 0061
2. CP 0067
3. FP-R-LIC-13

8.7 Reporting Errata in Effluent Release Reports

8.7.1 Small errors should be corrected within one year of discovery and the correction may be submitted with the next normally scheduled submittal of the ARERR (Annual Radiological Effluent Release Report). Small errors criteria are:

- Inaccurate reporting of dose that equates to < 10% of the applicable 10 CFR 50 Appendix I design objectives of < 10% of the EPA public dose criterion.
- Inaccurate reporting of curies, release rates, volumes, etc., that equate to < 10% of the affected curie total, release rate, volume, etc., after correction.
- Omissions that do not impede the NRC's ability to adequately assess the information supplied.
- Typographical errors or other errors that do not alter the intent of the report.

8.7.2 Large errors should be corrected within 90 days of discovery and the correction should be submitted within 90 days of the discovery. The correction may be submitted with the next ARERR, if the next ARERR is to be submitted within 90 days of the discovery. Large error criteria, are those which do not meet the criteria of a small error.

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BASIS

2.0 LIQUID EFFLUENTS

2.1/2.2 CONCENTRATION

This control is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to UNRESTRICTED AREAS will be less than ten times the concentration levels specified in 10 CFR 20, Appendix B, Table 2, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will not result in exposures exceeding (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, and (2) ten times the limits of 10 CFR 20. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

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

Secondary condenser drains were not included in the routine sampling requirements of Table 2.1. Operating experience has shown that the condenser activity during plant transients normally consists of very low levels of tritium. Condensers are normally only released directly to the environment during plant startups and shutdowns and these volumes combined with the low levels of activity are insignificant when compared to the waste tank activities. Condenser releases should be sampled and analyzed during a significant plant event (i.e. steam generator tube rupture, or steam dump to the condenser with a primary to secondary leak >725 gpd).

2.3/2.4 DOSE

Provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable". Considering that the nearest drinking water supply using the river for drinking water is more than 300 miles downstream, there is reasonable assurance that the operation of the facility will not result in radioactive concentrations in the drinking water that are in excess of the 40 CFR 141 limits.

2.5/2.6 LIQUID RADWASTE TREATMENT SYSTEMS

Provides assurance that the liquid radwaste treatment system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirements that appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents be kept "as low as reasonably achievable". This control implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR 50 and the design objective given in Section II.D of Appendix I to 10 CFR 50. The limits governing the use of appropriate portions of the liquid radwaste system were specified as a suitable fraction of the guide set forth in Section II.A of Appendix I, 10 CFR 50, for liquid effluents.

The liquid radwaste treatment system is shared by both units. It is not practical to determine the contribution from each unit to liquid radwaste releases. For this reason, liquid radwaste releases will be allocated equally to each unit.

2.7/2.8 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoint for these instruments **SHALL** be calculated and adjusted in accordance with the methodologies and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding ten times the water effluent concentration limits of 10 CFR 20. The FUNCTIONALITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR 50.

Radiation monitor set points are calculated to provide alarm and trip functions to ensure concentration of radioactive materials in liquid waste effluents released from the site to UNRESTRICTED AREAS, does not exceed the noted specific limits. The methodology prescribed in the ODCM for these calculations is acceptable for use in demonstrating compliance with 10 CFR 20.1301(a)(1), 10 CFR 50.36a, 10 CFR 50, Appendix A (GDC 60 & 64) and Appendix I, and 40 CFR 190.

Revision to the ODCM requires Operations Committee review and approval to ensure the revision continues to demonstrate compliance.

Specific monitor set point changes, when performed in accordance the methodology as reviewed and approved by the Operations Committee need not be reviewed by the Operations Committee. Specific monitor set point changes will be reviewed and approved by the Department Manager administering the ODCM program and the Radiation Monitor Engineer. The calculation sheet supporting the set point change is submitted to engineering for documentation.

2.9/2.10 LIQUID STORAGE TANKS

Restricting the quantities of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the contents of the tank, the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, in an UNRESTRICTED AREA.

3.0 GASEOUS EFFLUENTS

3.1/3.2 DOSE RATE

This specification provides reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure to a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA either at or beyond the SITE BOUNDARY in excess of the design objectives of appendix I to 10 CFR 50. This specification is provided to ensure that gaseous effluents from all units on the site will be appropriately controlled. It provides operational flexibility for releases gaseous effluents to satisfy the Section II.A and II.C design objectives of appendix I to 10 CFR 50. For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the SITE BOUNDARY to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to less than or equal to 1500 mrem/year at or beyond the SITE BOUNDARY.

This control applies to the release of radioactive materials in gaseous effluent from all units at the site.

3.3/3.4 DOSE FROM NOBLE GAS

This control is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR 50. The Limiting Conditions for Operation implement the guides set forth in Section II.B of Appendix I. The ACTION statement provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the release of radioactive material in gaseous effluents will be kept “as low as reasonably achievable”.

3.5/3.6 DOSE FROM IODINE 131, IODINE 133, TRITIUM & PARTICULATES

Implements the requirements of Section II.C, III.A and IV.A of Appendix I, 10 CFR 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTIONS statement provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as reasonable achievable". The release rate specifications for I-131, I-133, tritium and radioactive particulates with half-lives greater than eight days are dependent on the existing radionuclide pathways to MEMBERS OF THE PUBLIC in the UNRESTRICTED AREA, using child dose conversion factors. The pathways which are examined in the development of these calculations are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3.7/3.8 GASEOUS RADWASTE TREATMENT SYSTEMS

This control provides assurance that the Waste Gas Treatment System and the VENTILATION EXHAUST TREATMENT SYSTEMS will be available for use whenever gaseous wastes are released to the environment. The requirement that the appropriate portions of the Waste Gas Treatment System be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as reasonably achievable". This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR 50, and the design objective given in Section II.D of Appendix I to 10 CFR 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the guide set forth in Sections II.B and II.C of Appendix I, 10 CFR 50, for gaseous effluents.

The Waste Gas Treatment System, containment purge release vent, and spent fuel pool are shared by both units. Experience has also shown that contributions from both units are released from each auxiliary building vent. For these reasons, it is not practical to allocate releases to a specific unit. All releases will be allocated equally in determining conformance to the design objectives of 10 CFR 50, Appendix I.

Restricting the quantities of radioactivity which can be stored in one decay tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest EXCLUSION AREA BOUNDARY will not exceed 0.5 rem.

The cooling towers at Prairie Island are located to the south of the plant and are within 500 to 2000 feet from the point of release. At low wind velocities (below 10 mph) the gaseous activity released from the gaseous radwaste system could be at or near ground level near the cooling towers and remain long enough to be drawn into the circulating water in the tower. This control minimizes the possibility of releases of gaseous effluents from entering the river from cooling tower scrubbing.

3.9/3.10 EXPLOSIVE GAS MONITORING INSTRUMENTATION

To ensure the concentration of potentially explosive gas mixtures contained in the waste gas treatment system is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. Maintaining the concentrations below the flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR 50.

The system is operated with flow through the recombiners and with excess hydrogen in the system. By verifying that oxygen is less than or equal to 2% at the recombiner outlet, there will be no explosive mixtures in the system. Waste gas system oxygen is monitored by the two recombiner oxygen analyzers and the 121 gas analyzer. The 121 gas analyzer only monitors the low level loop of the waste gas system. If the required gas analyzers are not functional, the oxygen to the recombiner will be isolated to prevent oxygen from entering the system from this source. Tanks that may undergo maintenance are normally purged with nitrogen before placing them in service to eliminate this as a source of oxygen.

3.11/3.12 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoint for these instruments **SHALL** be calculated and adjusted in accordance with the methodologies and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR 20. The FUNCTIONALITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR 50.

Radiation monitor set points are calculated to provide alarm and trip functions to ensure concentration of radioactive materials in airborne effluents released from the site do not exceed the noted specific limits. The methodology prescribed in the ODCM for these calculations is acceptable for use in demonstrating compliance with 10 CFR 20.1301(a)(1), 10 CFR 50.36A, 10 CFR 50, Appendix A (GDC 60 & 64) and Appendix I, and 40 CFR 190.

Revision to the ODCM requires Operations Committee review and approval to ensure the revision continues to demonstrate compliance.

Specific monitor set point changes, when performed in accordance the methodology as reviewed and approved by the Operations Committee need not be reviewed by the Operations Committee. Specific monitor set point changes will be reviewed and approved by the Department Manager administering the ODCM program and the Radiation Monitor Engineer. The calculation sheet supporting the set point change is submitted to engineering for documentation.

6.0 TOTAL DOSE

This control is provided to meet the dose limitations of 40 CFR 190 that have been incorporated into 10 CFR 20 by FR 18525. The control requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and to radiation from uranium fuel cycle sources exceed 25 mrems to the whole body or to any organ, except the thyroid, which **SHALL** be limited to less than or equal to 75 mrems.

For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the units (including outside storage tanks, etc.) are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within 40 CFR 190 limits.

With the advent of the ISFSI, direct dose SHALL be assessed in demonstrating compliance with 40 CFR 190, for Special Reporting and for inclusion in the Annual Radioactive Effluent Report.

If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 & 10 CFR 20.2203(a)(4), is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR 20, as addressed in Specification 2.1 and 3.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle. Demonstration of compliance with the limits of 40 CFR 190 will be considered to demonstrate compliance with the 0.1 rem limit of 10 CFR 20.1301.

7.0 RADIOLOGICAL ENVIRONMENTAL MONITORING

7.1/7.2 MONITORING PROGRAM

Provides measurements of radiation and radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the plant operation. This program thereby supplements the radiological effluent monitoring by verifying that the measurable concentrations of radioactive materials and levels are not higher than expected in the bases of the effluent measurements and modeling of the environmental exposure pathways.

The detection capabilities required by Table 7.1 are state-of-the art for routine environmental measurements in industrial laboratories and the LLDs for drinking water meet the requirement of 40 CFR 141.

7.3/7.4 LAND USE CENSUS

This control is provided to ensure that changes in the use of off site areas are identified and that modifications to the monitoring program are made if required by the results of the census. The best survey information from door-to-door, aerial or consulting with local agricultural authorities **SHALL** be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used: 1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/square meter.

7.5/7.6 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an interlaboratory comparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

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Table 2.1 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)[a, d]
Batch Releases[g]: Waste Tanks	Each Batch (Prior to Release)	Each Batch (Prior to Release)	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	One Batch Each Month	One Batch Each Month	Dissolved and Entrained Gases	1×10^{-5}
	Each Batch	Monthly Composite[b]	H-3	1×10^{-5}
			Gross alpha	1×10^{-7}
	Each Batch	Quarterly Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
Continuous Release[e]: Turbine Building Sumps	Continuous$_{j,h,k}$	Weekly Composite[f]	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	Weekly Grab Sample	Each Sample	Dissolved and Entrained Gases	1×10^{-5}
	Continuous$_{j,k}$	Monthly Composite[f]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Continuous$_{j,k}$	Quarterly Composite[f]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
Continuous Release[e]: Steam Generator Blowdown	Weekly Grab Sample During Releases[i]	Each Sample Composite[b]	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	Grab Sample Each Month During Releases	Each Sample	Dissolved and Entrained Gases	1×10^{-5}
	Weekly Grab Sample During Releases[i]	Monthly Composite[b]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Weekly Grab Sample During Releases[i]	Quarterly Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

Table 2.1 Radioactive Liquid Waste Sampling and Analysis Program

Table Notations

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

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

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

where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

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

E = the counting efficiency (counts per disintegration),

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

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

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

$\Delta\tau$ = the elapsed time between the midpoint of sample collection and the time of counting (sec).

Typical values of E, V, Y, and $\Delta\tau$ should be used in the calculation.

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

Table 2.1 Radioactive Liquid Waste Sampling and Analysis Program

Table Notations [Cont'd]

2. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharge and in which the method of sampling employed results in a specimen which is representative of the liquids released.

3. The principal gamma emitters for which the LLD control applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137 and Ce-141. Ce-144 shall also be measured, but with an LLD of 5×10^{-6}. This list does not mean that only these nuclides are to be detected and reported. Other gamma peaks which are measurable and identifiable, together with the above nuclides, **SHALL** also be identified and reported.

4. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level. When unusual circumstances result in LLDs higher than required, the reasons **SHALL** be documented in the Annual Radioactive Effluent Report.

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

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 release.

7. A BATCH RELEASE is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch **SHALL** be isolated, and then thoroughly mixed to assure representative sampling.

8. Daily grab samples from the turbine building sumps **SHALL** be collected and analyzed for principal gamma emitters, including I-131, whenever primary to secondary leakage exceeds 150 gpd in any steam generator. This sampling is provided in lieu of continuous monitoring with automatic isolation.

9. Grab samples **SHALL** be collected at least once per 12 hours when steam generator blowdown releases are being made and the specific activity of the secondary coolant is $\geq$0.01 µCi/gram DOSE EQUIVALENT I-131 or primary to secondary leakage exceeds 150 gpd.

Table 2.1 Radioactive Liquid Waste Sampling and Analysis Program

Table Notations [Cont'd]

10. A continuous sample is one in which the sampling media is in place at all times during the release period, with the exception of periods necessary to change sampling media and scheduled short term equipment maintenance. If the sample media is not in place during the entire release period, an explanation of the occurrence, actions taken to restore the sampler and to prevent recurrence, and a summary description to explain the occurrence's effect on the analysis validity **SHALL** be included in the Annual Radioactive Effluent Report.

11. Continuous samples of the Turbine Building Sumps are collected via on-line composite samplers. These samplers function on timers and collect a predetermined volume of effluent whenever the TBS pumps are in operation. Samples from these compositors are collected daily and saved for the preparation of a weekly composite prepared utilizing volumes proportional to the sample volumes collected daily by the compositor. If the use of a submersible pump is necessary to maintain sump level, that pump should be positioned above the normal TBS pump controlling level and include a timer to allow the calculation of the additional release volume.

Table 2.2 Radioactive Liquid Effluent Monitoring Instrumentation

With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels FUNCTIONAL, take the actions directed in Table 2.2. Restore the non-functional instrumentation to FUNCTIONAL status within 30 days. If instrumentation is not restored within 30 days, explain in the next Annual Radioactive Effluent Release Report, why this non-functionality was not corrected in a timely manner.

INSTRUMENT	MINIMUM CHANNELS FUNCTIONAL	APPLICABILITY	ACTION
1. Gross Radioactivity Monitors Providing Automatic Termination of Release			
a. Liquid Radwaste Effluent Line	1	During releases	1
b. Steam Generator Blowdown Effluent Line	1/Unit	During releases	2
2. Flow Rate Measurement Devices			
a. Liquid Radwaste Effluent Line	1	During releases requiring throttling of flow	4
b. Steam Generator Blowdown Flow	1/Gen	During releases	4
3. Continuous Composite Samplers			
a. Each Turbine Building Sump Effluent Line	1/Unit	During releases	3
4. Discharge Canal Monitor	1	At all times	6
5. Tank Level Monitor			
a. Condensate Storage Tanks	1/Unit	When containing radioactive material	5
b. Temporary Outdoor Tanks Holding Radioactive Liquid	1/Tank	When tanks are in use	5
6. Discharge Canal Flow System (Daily determination and following changes in flow)	NA	At all times	

Table 2.2 Radioactive Liquid Effluent Monitoring Instrumentation Table Notations

ACTION 1 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue, provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 2.2.1, and

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 2 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue provided grab samples are analyzed for radioactivity at a lower limit of detection of not more than that specified in Table 2.1 for Principal Gamma Emitters.

1. At least once per 12 hours when the specific activity of the secondary coolant is $\geq$0.01 μCi/gram DOSE EQUIVALENT I-131, or
2. At least once per 24 hours when the specific activity of the secondary coolant is <0.01 μCi/gram DOSE EQUIVALENT I-131.

ACTION 3 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue provided that at least once per 12 hours, grab samples are collected and saved for weekly composition and analysis in accordance with Table 2.1.

ACTION 4 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue provided that the flow rate is estimated at least once per four (4) hours during actual releases. Pump curves may be used to estimate flow.

ACTION 5 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue provided that tank liquid level is estimated during all liquid additions.

ACTION 6 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for gamma emitters.

Table 2.3 Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements

Instrument	CHANNEL CHECK Frequency (4)	SOURCE CHECK Frequency	FUNCTIONAL TEST Frequency	CALIBRATION Frequency
Liquid Radwaste Effluent Line Gross Radioactivity Monitor	Daily during releases	Prior to each release	Quarterly(1)	At least once every 18 months(3)
Liquid Radwaste Effluent Line Flow Instrument	Daily during releases	----	----	At least once every 18 months
Steam Generator Blowdown Gross Radioactivity Monitors	Daily during releases	Monthly	Quarterly(1)	At least once every 18 months(3)
Steam Generator Blowdown Flow	Daily during releases	----	----	At least once every 18 months
Turbine Building Sump Continuous Composite Samplers	Daily during releases (Includes sample volume check)	----	----	----
Discharge Canal Monitor	Daily during releases	Monthly	Quarterly(2)	At least once every 18 months(3)
Discharge Canal Flow Instruments	Daily during releases	----	----	At least once every 18 months
Condensate Storage Tank Level Monitors	Daily	----	Quarterly	At least once every 18 months
Level Monitors for Temporary Outdoor Tanks Holding Radioactive Liquid	Daily when in use	----	Quarterly when in use	At least once every 18 months when in use

Table 2.3 Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements

Table Notations

1. The CHANNEL FUNCTIONAL TEST **SHALL** also demonstrate that automatic isolation of this pathway and control room annunciation occurs if any of the following conditions exists:

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

    2. Circuit failure (if provided).

    3. Instrument indicates a downscale failure (if provided).

    4. Instrument controls not set in operate mode (if provided).

2. The CHANNEL FUNCTIONAL TEST **SHALL** also demonstrate that alarm annunciation occurs if any of the following conditions exists:

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

    2. Circuit failure (if provided).

    3. Instrument indicates a downscale failure (if provided).

    4. Instrument controls not set in operate mode (if provided).

3. The initial CHANNEL CALIBRATION **SHALL** be performed using one or more of the reference standards certified by the National Institute of Standards and Technology or using sources traceable to NIST standards. These standards **SHALL** permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATIONS, sources that have been related to the initial calibration **SHALL** be used.

4. The CHANNEL CHECK **SHALL** consist of verifying indication of flow during periods of release. A CHANNEL CHECK **SHALL** be made at least once daily on any day on which continuous, periodic, or batch releases are made.

Table 3.1 Radioactive Gaseous Waste Sampling and Analysis Program

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit[a, f] of Detection (LLD)(μCi/ml)
CONTINUOUS RELEASE Points:	Weekly [b, i] Gas Grab Sample	Weekly	Principal Gamma Emitters [e]	1×10^{-4}
Plant Vents:	Continuous [g, i, h]	Weekly [c] Charcoal Sample	I-131, I-133	1×10^{-12}
Unit 1 Aux Bldg. Unit 2 Aux Bldg. Radwaste Bldg.	Continuous [g, i, h]	Weekly [c] Particulate Sample	Principal Gamma Emitters [e]	1×10^{-11}
Spent Fuel Pool Unit 1 Shield Bldg. Unit 2 Shield Bldg.	Continuous [g, i, h]	Monthly Silica Gel Sample	H-3	1×10^{-6}
	Continuous [g, i, h]	Each Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous [g, i, h]	Quarterly [d] Particulate Composite	Sr-89, Sr-90	1×10^{-11}
	Continuous [g]	Noble Gas Monitor	Noble Gases, Gross beta and gamma	1×10^{-4}
Atmospheric Steam Releases [k]	Daily [j] Grab Sample During Release	Each Sample	Principal Gamma Emitters [e]	5×10^{-7}
			I-131, I-133	1×10^{-6}
	Daily [j] Grab Sample During Release	Monthly [l] Composite	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Daily [j] Grab Sample During Release	Quarterly [l] Composite	Sr-89, Sr-90	5×10^{-8}

Table 3.1 Radioactive Gaseous Waste Sampling and Analysis Program

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit[a, f] of Detection (LLD)(□μCi/ml)
Containment Purge[m]	Gas Grab Sample Prior to each Purge	Each Sample (Prior to Release)	Principal Gamma Emitters [e]	1×10^{-4}
	Grab[g, h, m] Prior to Release and Continuous	Each Sample	H-3	1×10^{-6}
	Grab[g,h, m] Prior to Release and Continuous	Charcoal Sample	I-131, I-133	1×10^{-12}
	Grab[g, h, m] Prior to Release and Continuous	Particulate Sample	Principal Gamma Emitters [e]	1×10^{-11}
	Grab[g, h, m] Prior to Release and Continuous	Each Particulate Sample	Gross Alpha	1×10^{-11}
	Grab[g, h, m] Prior to Release and Continuous	Quarterly [d] Particulate Composite	Sr-89, Sr-90	1×10^{-11}
Waste Gas Storage Tanks	Gas Grab Sample Prior to each Release	Each Sample (Prior to Release)	Principal Gamma Emitters [e]	1×10^{-4}
	Grab Sample Prior to each Release	Each Sample (Prior to Release)	H-3	1×10^{-6}

Table 3.1 - Radioactive Gaseous Waste Sampling and Analysis Program

Table Notations

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

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

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

where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume.

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

E = the counting efficiency (counts per disintegration),

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

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

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

$\Delta\tau$ = the elapsed time between the midpoint of sample collection and the time of counting (sec).

Typical values of E, V, Y, and $\Delta\tau$ should be used in the calculation.

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

Table 3.1 - Radioactive Gaseous Waste Sampling and Analysis Program

Table Notations [Cont'd]

2. Grab samples taken at the ventilation exhausts are generally below minimum detectable levels for most nuclides with existing analytical equipment. If this is the case, Gaseous Source Terms (Table 5.2) noble gas isotopic ratios may be assumed.

3. With >1 µCi/gm DOSE EQUIVALENT I-131 in either Unit 1 or Unit 2 reactor coolant system, the iodine and particulate collection devices for all release points **SHALL** be removed and analyzed daily until it is shown that a pattern exists which can be used to predict the release rate. Sampling may then revert to weekly. When samples collected for one day are analyzed, the corresponding LLD's may be increased by a factor of 10. Samples **SHALL** be analyzed within 48 hours after removal.

4. To be representative of the average quantities and concentrations of radioactive materials in particulate form in gaseous effluents, samples should be collected in proportion to the rate of flow of the effluent streams.

5. The principal gamma emitters for which the LLD control applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for noble gas analysis and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 in iodine and particulate analysis. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, **SHALL** also be detected and reported.

6. Nuclides which are below the LLD for analyses should not be reported as being present at the LLD level for that nuclide. When unusual circumstances result in LLD's higher than reported, the reasons **SHALL** be documented in the Annual Radioactive Effluent Report.

7. For continuous samples, the ratio of the sample flow rate to the samples stream flow rate **SHALL** be known for the time period sampled (Conservative assumptions may be used). Design flow rates may be used for building exhaust vent flow rates.

8. A continuous sample is one in which the sampling media is in place at all times during the release period, with the exception of periods necessary to change sampling media and scheduled short term equipment maintenance of two hours or less. If the sample media is not in place during the entire release period (except as described above), an explanation of the occurrence, actions taken to restore the sampler and to prevent reoccurrence, and a summary description to explain the occurrence's effect on the analysis validity **SHALL** be included in the Annual Radioactive Effluent Report.

Table 3.1 Radioactive Gaseous Waste Sampling and Analysis Program

Table Notations [Cont'd]

9. Releases are made via the shield building vents only during PURGING, or operation of special ventilation systems. When ventilation fans in any vent path are not in service for the entire sample period, in lieu of weekly removal and analysis of iodine and particulate collection devices, these devices may be removed and analyzed following each release provided that the release lasts less than one week. Releases made via the plant ventilation paths as a result of routine surveillance tests, operational testing or scheduled short term maintenance activities of 2 hours or less do not require special sampling and analysis provided that plant conditions do not indicate the completion of these activities would cause an increase in the release of activity. Removal and analysis of collection devices is not required if releases are not being made.

10. Grab samples for atmospheric steam releases are representative liquid grab samples from the respective steam generator.

11. Atmospheric steam releases are the timed releases of steam from the steam generators to the atmosphere via either the power operated reliefs, steam dump valves or flash tank vents. It does not include steam dumped via the condenser.

12. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of steam released and in which the method of sampling employed results in a specimen which is representative of the total steam released form the respective steam generator.

13. Containment Purges includes PURGE releases with either the Inservice Purge or Containment Purge Fans and also VENTING of containment utilizing the Post Loca Vent System. When the release is completed via the Post Loca Vent, the pre-release tritium, particulate and charcoal samples should be used for all analyses, and continuous samples collected during the release are not required. During Cold Shutdown periods, the availability of ventilation systems and the position of containment air-lock doors may require that portions of the required samples be collected with installed continuous monitors or portable sampling equipment.

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Table 3.2 Radioactive Gaseous Effluent Monitoring Instrumentation

INSTRUMENT	MINIMUM CHANNELS FUNCTIONAL	APPLICABILITY	ACTION
1. Waste Gas Holdup System Explosive Gas (Oxygen) Monitors	2	During system operation	2
2. Effluent Release Points Unit 1 Aux Bldg. Unit 2 Aux Bldg. Rad Waste Bldg. Spent Fuel Pool Unit 1 Shield Bldg. Unit 2 Shield Bldg.			
a. Noble Gas Activity Monitor*	1	During releases	3, 4, 6
b. Iodine Sampler Cartridge	1	During releases	2
c. Particulate Sampler Filter	1	During releases	2
d. Sampler Flow Integrator	1	During releases	1

* Noble gas activity monitors providing automatic termination of releases (except the Radwaste Building which has no automatic isolation function).

With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels FUNCTIONAL, take the actions directed in Table 3.2. Restore the non-functional instrumentation to FUNCTIONAL status within 30 days.

If Radioactive Gaseous Effluent Monitoring Instrumentation is not restored within 30 days, explain in the next Annual Radioactive Effluent Report, why this non-functionality was not corrected in a timely manner. The Waste Gas Holdup System Explosive Gas (Oxygen) Monitors are NOT subject to this requirement. They are subject to the Special Reporting requirements, as defined in section 3.9.4.

Table 3.2 Radioactive Gaseous Effluent Monitoring Instrumentation

Table Notations

ACTION 1 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue provided that the flow rate is estimated at least once per 12 hours.

ACTION 2 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, operating of this system may continue for up to 14 days. With two channels not FUNTIONAL, manually isolate the oxygen addition line.

ACTION 3 With the numbers of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue provided that samples are collected with auxiliary sample equipment as required in Table 3.1.

ACTION 4 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue provided that samples are taken and analyzed to LLD per Table 3.1, at least once per 12 hours.

ACTION 5 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, immediately suspend Purging of radioactive effluents via this pathway during periods when containment integrity is required or the primary system is initially opened to the atmosphere. (applicable to Reactor Building Vents)

ACTION 6 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, the contents of the waste gas decay tanks may be released to the environment provided that prior to initiating 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 Facility Staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway (applicable to Unit 2 Auxiliary Building Vent).

Table 3.3 - Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

Instrument	CHANNEL CHECK Frequency	SOURCE CHECK Frequency	FUNCTIONAL TEST Frequency	CALIBRATION Frequency
Waste Gas Holdup System Explosive Gas (Oxygen) Monitors	Daily during system operation	----	Monthly(2)	Quarterly(5)
Effluent Release Points Unit 1 Aux Bldg. Unit 2 Aux Bldg. Rad Waste Bldg. Spent Fuel Pool Unit 1 Shield Bldg. Unit 2 Shield Bldg.				
Noble Gas Activity Monitor (4) (Except Radwaste Building)	Daily during releases	Monthly*	Quarterly(1)	At least once every 18 months(3)
Noble Gas Activity Monitor Radwaste Building (4)	Daily during releases	Monthly	Quarterly(2)	At least once every 18 months(3)
Iodine and Particulate Samplers	Weekly	----		----
Sampler Flow Rate Monitor	Weekly	----	----	At least once every 18 months

* A SOURCE CHECK of the applicable nobles gas monitor **SHALL** be conducted prior to each waste gas decay tank or containment purge release.

Table 3.3 - Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements

Table Notations

1. The CHANNEL FUNCTIONAL TEST **SHALL** also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following exists.

    1. Instrument indicates measured levels above the alarm/trip setpoint.
    2. Circuit failure (if provided).
    3. Instrument indicates a downscale failure (if provided).
    4. Instrument controls not set in operate mode (if provided).

2. The CHANNEL FUNCTIONAL TEST **SHALL** also demonstrate that alarm annunciation occurs if any of the following conditions exists:

    1. Instrument indicates measured levels above the alarm/trip setpoint.
    2. Circuit failure (if provided).
    3. Instrument indicates a downscale failure (if provided).
    4. Instrument controls not set in operate mode (if provided).

3. The initial CHANNEL CALIBRATION **SHALL** be performed using one or more of the reference standards certified by the National Institute of Standards and Technology or using sources traceable to NIST standards. These standards **SHALL** permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATIONS, sources that have been related to the initial calibration **SHALL** be used.

4. Noble gas monitor in the Radwaste Building vent not provided with automatic isolation trip.

5. The CHANNEL CALIBRATION **SHALL** include the use of a nitrogen zero gas and an oxygen span gas with a nominal concentration suitable for the range of the instrument.

Table 4.1 Liquid Source Terms

RADIONUCLIDE	EFFLUENT CONCENTRATION LIMIT (µCi/ml) *	WASTE EFFLUENT A_i (Ci/Yr)	SGBD A_i (Ci/Yr)
Mo-99	2E-4	6.42E-3	1.415E-2
I-131	1E-5	3.061E-2	4.11E-2
Te-132	9E-5	2.12E-3	3.61E-3
I-132	1E-3	2.83E-3	1.88E-2
I-133	1E-6	2.365E-2	4.856E-2
Cs-134	9E-6	1.464E-1	4.047E-2
I-135	3E-4	4.84E-3	1.792E-2
Cs-136	6E-5	5.743E-2	1.862E-2
Cs-137	1E-5	8.214E-2	2.69E-2
All Others	1E-7	0	2E-5
H-3	1E-2	1.89E2	1.41E2
Noble gases	2E-4	---	---
TOTAL		1.894E2	1.412E2

* Effluent Concentration Limit =

Ten times the values listed in 10 CFR-20.1001-20.2402, App. B, Table 2, Column 2.

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Table 4.2 - Adult Ingestion Dose Values (A_{it}) for the Prairie Island Nuclear Generating Plant (mrem/Hr Per μCi/ml)

NUCLIDE	BONE	LIVER	T. BODY	THYROID	KIDNEY	LUNG	GI-LLI
H-3	0.00E-01	2.26E-01	2.26E-01	2.26E-01	2.26E-01	2.26E-01	2.26E-01
C-14	3.13E 04	6.26E 03	6.26E 03	6.26E 03	6.26E 03	6.26E 03	6.26E 03
NA-24	4.07E 02	4.07E 02	4.07E 02	4.07E 02	4.07E 02	4.07E 02	4.07E 02
CR-51	0.00E-01	0.00E-01	1.27E 00	7.61E-01	2.81E-01	1.69E 00	3.20E 02
MN-54	0.00E-01	4.38E 03	8.35E 02	0.00E-01	1.30E 03	0.00E-01	1.34E 04
MN-56	0.00E-01	1.10E 02	1.95E 01	0.00E-01	1.40E 02	0.00E-01	3.51E 03
FE-55	6.58E 02	4.55E 02	1.06E 02	0.00E-01	0.00E-01	2.54E 02	2.61E 02
FE-59	1.04E 03	2.44E 03	9.36E 02	0.00E-01	0.00E-01	6.82E 02	8.14E 03
CO-57	0.00E-01	2.10E 01	3.48E 01	0.00E-01	0.00E-01	0.00E-01	5.32E 02
CO-58	0.00E-01	8.92E 01	2.00E 02	0.00E-01	0.00E-01	0.00E-01	1.81E 03
CO-60	0.00E-01	2.56E 02	5.65E 02	0.00E-01	0.00E-01	0.00E-01	4.81E 03
NI-63	3.11E 04	2.16E 03	1.04E 03	0.00E-01	0.00E-01	0.00E-01	4.50E 02
NI-65	1.26E 02	1.64E 01	7.49E 00	0.00E-01	0.00E-01	0.00E-01	4.17E 02
CU-64	0.00E-01	9.97E 00	4.68E 00	0.00E-01	2.51E 01	0.00E-01	8.50E 02
ZN-65	2.32E 04	7.37E 04	3.33E 04	0.00E-01	4.93E 04	0.00E-01	4.64E 04
ZN-69	4.93E 01	9.43E 01	6.56E 00	0.00E-01	6.13E 01	0.00E-01	1.42E 01
BR-83	0.00E-01	0.00E-01	4.04E 01	0.00E-01	0.00E-01	0.00E-01	5.82E 01
BR-84	0.00E-01	0.00E-01	5.24E 01	0.00E-01	0.00E-01	0.00E-01	4.11E-04
BR-85	0.00E-01	0.00E-01	2.15E 00	0.00E-01	0.00E-01	0.00E-01	1.01E-15
RB-86	0.00E-01	1.01E 05	4.71E 04	0.00E-01	0.00E-01	0.00E-01	1.99E 04
RB-88	0.00E-01	2.90E 02	1.54E 02	0.00E-01	0.00E-01	0.00E-01	4.00E-09
RB-89	0.00E-01	1.92E 02	1.35E 02	0.00E-01	0.00E-01	0.00E-01	1.12E-11
SR-89	2.21E 04	0.00E-01	6.35E 02	0.00E-01	0.00E-01	0.00E-01	3.55E 03
SR-90	5.44E 05	0.00E-01	1.34E 05	0.00E-01	0.00E-01	0.00E-01	1.57E 04
SR-91	4.07E 02	0.00E-01	1.64E 01	0.00E-01	0.00E-01	0.00E-01	1.94E 03
SR-92	1.54E 02	0.00E-01	6.68E 00	0.00E-01	0.00E-01	0.00E-01	3.06E 03
Y-90	5.76E-01	0.00E-01	1.54E-02	0.00E-01	0.00E-01	0.00E-01	6.10E 03
Y-91M	5.44E-03	0.00E-01	2.11E-04	0.00E-01	0.00E-01	0.00E-01	1.60E-02
Y-91	8.44E 00	0.00E-01	2.26E-01	0.00E-01	0.00E-01	0.00E-01	4.64E 03
Y-92	5.06E-02	0.00E-01	1.48E-03	0.00E-01	0.00E-01	0.00E-01	8.86E 02
Y-93	1.60E-01	0.00E-01	4.43E-03	0.00E-01	0.00E-01	0.00E-01	5.09E 03
ZR-95	2.40E-01	7.70E-02	5.21E-02	0.00E-01	1.21E-01	0.00E-01	2.44E 02
ZR-97	1.33E-02	2.68E-03	1.22E-03	0.00E-01	4.04E-03	0.00E-01	8.30E 02
NB-95	4.47E 02	2.48E 02	1.34E 02	0.00E-01	2.46E 02	0.00E-01	1.51E 04
NB-97	3.76E 00	9.48E-01	3.46E-01	0.00E-01	1.11E 00	0.00E-01	3.50E 03
MO-99	0.00E-01	1.03E 02	1.96E 01	0.00E-01	2.34E 02	0.00E-01	2.39E 02
TC-99M	8.87E-03	2.51E-02	3.19E-01	0.00E-01	3.81E-01	1.23E-02	1.48E 01
TC-101	9.12E-03	1.31E-02	1.29E-01	0.00E-01	2.37E-01	6.72E-03	3.95E-14
RU-103	4.43E 00	0.00E-01	1.91E 00	0.00E-01	1.69E 01	0.00E-01	5.17E 02
RU-105	3.69E-01	0.00E-01	1.46E-01	0.00E-01	4.76E 00	0.00E-01	2.26E 02
RU-106	6.58E 01	0.00E-01	8.33E 00	0.00E-01	1.27E 02	0.00E-01	4.26E 03

Table 4.2 - Adult Ingestion Dose Values (A_{it}) for the Prairie Island Nuclear Generating Plant (mrem/Hr Per μCi/ml)

NUCLIDE	BONE	LIVER	T. BODY	THYROID	KIDNEY	LUNG	GI-LLI
RH-105	2.92E 00	2.12E 00	1.40E 00	0.00E-01	9.00E 00	0.00E-01	3.38E 02
AG-110M	8.81E-01	8.15E-01	4.84E-01	0.00E-01	1.60E 00	0.00E-01	2.9E 02
SB-124	6.74E 00	1.27E-01	2.66E-01	1.63E-02	0.00E-01	5.23E 00	1.91E 02
SB-125	5.34E 00	5.75E-02	1.07E 00	4.74E-03	0.00E-01	5.58E 02	4.72E 01
SB-126	2.75E 00	5.60E-02	9.94E-01	1.69E-02	0.00E-01	1.69E 00	2.25E 02
TE-125M	2.57E 03	9.30E 02	3.44E 02	7.72E 02	1.04E 04	0.00E-01	1.02E 04
TE-127M	6.48E 03	2.32E 03	7.90E 02	1.66E 03	2.63E 04	0.00E-01	2.17E 04
TE-127	1.05E-02	3.78E 01	2.28E 01	7.80E 01	4.29E 02	0.00E-01	8.31E 03
TE-129M	1.10E 04	4.11E 03	1.74E 03	3.78E 03	4.60E 04	0.00E-01	5.54E 04
TE-129	3.01E 01	1.13E 01	7.33E 00	2.31E 01	1.26E 02	0.00E-01	2.27E 01
TE-131M	1.66E 03	8.10E 02	6.75E 02	1.28E 03	8.21E 03	0.00E-01	8.04E 04
TE-131	1.89E 01	7.88E 00	5.96E 00	1.55E 01	8.26E 01	0.00E-01	2.67E 00
5TE-132	2.41E 03	1.56E 03	1.47E 03	1.72E 03	1.50E 04	0.00E-01	7.38E 04
I-130	2.71E 01	8.01E 01	3.16E 01	6.79E 03	1.25E 02	0.00E-01	6.89E 01
I-131	1.49E 02	2.14E 02	1.22E 02	7.00E 04	3.66E 02	0.00E-01	5.64E 01
I-132	7.29E 00	1.95E 01	6.82E 00	6.82E 02	3.11E 01	0.00E-01	3.66E 00
I-133	5.10E 01	8.87E 01	2.70E 01	1.30E 04	1.55E 02	0.00E-01	7.97E 01
I-134	3.81E 00	1.03E 01	3.70E 00	1.79E 02	1.64E 01	0.00E-01	9.01E-03
I-135	1.59E 01	4.17E 01	1.54E 01	2.75E 03	6.68E 01	0.00E-01	4.70E 01
CS-134	2.98E 05	7.09E 05	5.79E 05	0.00E-01	2.29E 05	7.61E 04	1.24E 04
CS-136	3.12E 04	1.23E 05	8.86E 04	0.00E-01	6.85E 04	9.38E 03	1.40E 04
CS-137	3.82E 05	5.22E 05	3.42E 05	0.00E-01	1.77E 05	5.89E 04	1.01E 04
CS-138	2.64E 02	5.22E 02	2.59E 02	0.00E-01	3.84E 02	3.79E 01	2.23E-03
BA-139	9.29E-01	6.62E-04	2.72E-02	0.00E-01	6.19E-04	3.75E-04	1.65E 00
BA-140	1.94E 02	2.44E-01	1.27E 01	0.00E-01	8.30E-02	1.40E-01	4.00E 02
BA-141	4.51E-01	3.41E-0.4	1.52E-02	0.00E-01	3.17E-04	1.93E-04	2.13E-10
BA-142	2.04E-01	2.10E-04	1.28E-02	0.00E-01	1.77E-04	1.19E-04	2.37E-19
LA-140	1.50E-01	7.54E-02	1.99E-02	0.00E-01	0.00E-01	0.00E-01	5.54E 03
LA-142	7.66E-03	3.48E-03	8.68E-04	0.00E-01	0.00E-01	0.00E-01	2.54E 01
CE-141	2.24E-02	1.52E-02	1.72E-03	0.00E-01	7.04E-03	0.00E-01	5.79E 01
CE-143	3.95E-03	2.92E 00	3.23E-04	0.00E-01	1.29E-03	0.00E-01	1.09E 02
CE-144	1.17E 00	4.88E-01	6.27E-02	0.00E-01	2.90E-01	0.00E-01	3.95E 02
PR-143	5.51E-01	2.21E-01	2.73E-02	0.00E-01	1.27E-01	0.00E-01	2.41E 03
PR-144	1.80E-03	7.48E-04	9.16E-05	0.00E-01	4.22E-04	0.00E-01	2.59E-10
ND-147	3.76E-01	4.35E-01	2.60E-02	0.00E-01	2.54E-01	0.00E-01	2.09E 03
W-187	2.96E 02	2.47E 02	8.65E 01	0.00E-01	0.00E-01	0.00E-01	8.10E 04
NP-239	2.85E-02	2.80E-03	1.54E-03	0.00E-01	8.74E-03	0.00E-01	5.75E 02

The values in the above table are calculated utilizing an adult fish consumption of 21 Kg/yr.

Table 5.1 - Monitor Alarm Setpoint Determination for PINGP

MONITOR	RELEASE POINT	SOURCE OF RELEASE	SOURCE TERMS (A_i) (TABLE 5.2)	Dispersion factor selection* X/Q (sec/m^3)	EFFLUENT FLOW RATE (F) (cfm)	RELEASE FRACTION (Tm)
1R-30 and 1R-37	Aux. Bldg. Vent - Unit 1	Aux. Bldg. Unit 1 Exhaust	Aux. Bldg.	Long Term Release	2.9E+4	0.2
		Air Ejector Unit 1	Air Ejector	NA	2.9E+4	
2R-30 and 2R-37	Aux. Bldg. Vent - Unit 2	Aux. Bldg. - Unit 2 Exhaust	Aux. Bldg.	Long Term Release	4.1E+4	0.3
		Gas Decay Tanks	Xe-133 (100%)	Short Term Release	4.1E+4	
		Air Ejector Unit 2	Air Ejector	NA	4.1E+4	
1R-12 and 1R-22	Shield Bldg. Vent - Unit 1	Cont. - Units 1&2 Purge, Unit 1 Inservice Purge	Shield Bldg.	Short Term Release	3.2E+4 (Note 2)	0.3
2R-12 and 2R-22	Shield Bldg. Vent - Unit 2	Cont. - Unit 2 Inservice Purge	Shield Bldg.	Short Term Release	4.6E+3	0.3
R-35	Radwaste Bldg. Vent	Radwaste Bldg. Exhaust	Aux. Bldg.	Long Term Release	6.1E+3	0.1
R-25 and R-31	Spent Fuel Pool Air Vent	Spent Fuel Pool Air Exhaust	Aux. Bldg.	Long Term Release	1.8E+4	0.1

* Current dispersion factors are maintained in H4.2, "Offsite Dose Calculation Manual (ODCM) Supporting Data".

NOTE:	Values listed for T_m are nominal values only. They may be adjusted as necessary to allow a reasonable margin to the monitor setpoint. Duplicate values of T_m are assigned to both Shield Building vents since only one containment will be purged at any one time. The assigned T_m values of all active release points SHALL NOT be greater than unity.

NOTE:	When purging the Unit 1 containment via the inservice purge system, the monitor setpoints may be based on 4.6E+3 cfm for the duration of the release.

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Table 5.2 Gaseous Source Terms

RADIONUCLIDE	AUX. BLDG A_i (Ci/Yr)	SHIELD BLDG. A_i (Ci/Yr)	AIR EJECTOR A_i (Ci/Yr)
Kr-85m	3E0	-	2E0
Kr-85	2E0	2.2E1	-
Kr-87	1E0	-	-
Kr-88	5E0	1E0	3E0
Xe-131m	2E0	2.1E1	1E0
Xe-133m	5E0	2E1	3E0
Xe-133	3.7E2	2.7E3	2.3E2
Xe-135	8E0	6E0	5E0
Xe-138	1E0	-	-
TOTAL	3.97E2	2.77E3	2.44E2

"-" indicates that the release is less than 1 Ci/yr.

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Table 5.3 Critical Organ Dose Values (P_i) for Child

ISOTOPE	(P_i) $\frac{\text{mrem/yr}}{\mu\text{Ci/m}^3}$
H-3	1.12 E 3
Cr-51	1.70 E 4
Mn-54	1.58 E 6
Fe-59	1.27 E 6
Co-58	1.11 E 6
Co-60	7.07 E 6
Zn-65	9.95 E 5
Rb-86	1.98 E 5
Sr-89	2.16 E 6
Sr-90	1.01 E 8
Y-91	2.63 E 6
Zr-95	2.23 E 6
Nb-95	6.14 E 5
Ru-103	6.62 E 5
Ru-106	1.43 E 7
Ag-110m	5.48 E 6
Te-127m	1.48 E 6
Te-129m	1.76 E 6
Cs-134	1.01 E 6
Cs-136	1.71 E 5
Cs-137	9.07 E 5
Ba-140	1.74 E 6
Ce-141	5.44 E 5
Ce-144	1.20 E 7
I-131	1.62 E 7

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Table 5.4 Dose Factors for Noble Gases *

Radionuclide	Total Body Dose Factor Ki (mrem/yr per µCi/m^3)	Skin Dose Factor Li (mrem/yr per µCi/m^3)	Gamma Air Dose Factor Mi (mrad/yr per µCi/m^3)	Beta Air Dose Factor Ni (mrad/yr per µCi/m^3)
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	9.73+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	163E+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

* The listed dose factors are for radionuclides that may be detected in gaseous effluents. All others are 0.

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Table 7.1 Radiological Environmental Monitoring Program Sample Collection and Analysis

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE Radioiodine and Particulates	Samples from 5 locations: a. Three samples from close to the three SITE BOUNDARY locations (in different sectors) of the highest calculated annual average ground level D/Q; b. One sample from the vicinity of a community having the highest calculated annual average ground level D/Q. c. One sample from a control location specified in the REMP.	Continuous Sampler operation with sample collection weekly	Radioiodine analysis weekly for I-131 Particulate: Gross beta activity on each filter weekly*. Analysis **SHALL** be performed more than 24 hours following filter change. Perform gamma isotopic analysis on composite (by location) sample quarterly.
2. DIRECT RADIATION	32 TLD stations established with duplicate dosimeters placed at the following locations: 1. Using the 16 meteorological wind sectors as guidelines, an inner ring of stations in the general area of the site boundary is established and an outer ring of stations in the 4 to 5 mile distance from the plant site is established. Because of inaccessibility, seven sectors in the inner and outer rings are not covered	Quarterly	Gamma dose quarterly

* If Gross beta activity in any indictor sample exceeds 10 times the yearly average of the control sample, a gamma isotopic analysis is required.

** Sample locations are further described by the REMP.

Table 7.1 Radiological Environmental Monitoring Program
Sample Collection and Analysis

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
2. DIRECT RADIATION [Cont'd]			
	2. Seven dosimeters are established at special interest areas and a control station.		
3. WATERBORNE			
a. Surface	Upstream & downstream locations	Monthly Composite of weekly samples (water & ice conditions permitting)	Gamma isotopic analysis of each monthly composite Tritium analysis of quarterly composites of monthly composites
b. Ground	3 samples from wells within 5 miles of the plant site and 1 sample from a well greater than 10 miles from the plant site	Quarterly	Gamma isotopic and tritium analyses of each sample
c. Drinking	1 sample from the City of Red Wing water supply	Monthly Composite of weekly samples	I-131 Analysis and Gross beta and gamma isotopic analyses of each monthly composite Tritium analysis of quarterly composites of monthly composites
d. All	Surface, ground, and drinking water samples in support of the NEI Groundwater Protection Initiative	Monthly, Quarterly, or Annually	Tritium analysis, Gamma analysis, and hard-to-detect analysis of selected samples

** Sample locations are further described by the REMP.

Table 7.1 Radiological Environmental Monitoring Program
Sample Collection and Analysis

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE [Cont'd]			
d. Sediment from shoreline	One sample upstream of plant, one sample downstream of plant, and one from shoreline of recreational area.	Semiannually	Gamma isotopic analysis of each sample
4. INGESTION			
a. Milk	One sample from dairy farm having highest D/Q, one sample from each of three dairy farms calculated to have doses from I-131 > 1 mRem/yr, and one sample from 10-20 miles	Semimonthly when animals are on pasture; monthly at other times.	Gamma isotopic and I-131 analysis of each sample
b. Fish and Invertebrates	One sample of one game specie of fish located upstream and downstream of the plant site One sample of Invertebrates upstream and downstream of the plant site	Semiannually	Gamma isotopic analyses on each sample (edible portion only on fish)

** Sample locations are further described by the REMP.

Table 7.1 Radiological Environmental Monitoring Program
Sample Collection and Analysis

Exposure Pathway and/or Sample	Number of Samples and Sample Locations**	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION [Cont'd]			
c. Food Products	One sample of corn from any field that is irrigated by water into which liquid plant wastes have been discharged***	At time of harvest	Gamma isotopic analysis of edible portion of each sample
	One sample of broad leaf vegetation from highest D/Q garden and one sample from 10-20 miles	At time of harvest	I-131 analyses of edible portion of each sample

** Sample locations are further described by the REMP.
*** As determined by methods outlined in the ODCM.

Table 7.2 - Reporting Levels for Radioactivity Concentration in Environmental Samples

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	20,000[(a)]				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400[(b)]				
I-131	2[(a)]	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[(b)]			300[(b)]	

(a) Drinking water pathway level.

(b) Total for parent and daughter.

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Table 7.3 - Detection Capabilities for Environmental Sample Analysis Lower Limit of Detection (LLD)[(a)]

ANALYSIS	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, dry)
Gross Beta	4	0.01				
H-3	2,000[(b)]					
Mn-54	15		130			
Fe-59	30		260			
Co-58, 60	15		130			
Zn-65	30		260			
Zr-Nb-95	15 [(c)]					
I-131[(d)]	1[(b)]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15[(c)]			15[(c)]		

Table 7.3 - Table Notation

a - The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

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

Where:

LLD is the appropriate lower limit of detection as defined above (as Pico curie per unit mass or volume), s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute). In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background **SHALL** include the typical contributing of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and $\Delta\tau$ **SHALL** be used in the calculations.

E is the counting efficiency (as counts per transformation),

2.22 is the number of transformation per minute per Pico curie,

Y is the fractional radiochemical yield (when applicable),

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

$\Delta\tau$ is the elapsed time between sample collection (or end of the sample collection period) and time of counting.

b - Drinking water pathway limit.
c - Total for parent and daughter
d - These LLDs apply only where "^{131}I analysis" is specified.
e - Where "Gamma Isotopic Analysis" is specified, the LLD specification applies to the following radionuclides: ^{54}Mn, ^{59}Fe, ^{58}Co, ^{60}Co, ^{65}Zn, ^{95}Zr-Nb, ^{137}Cs, ^{134}Cs, and ^{140}Ba-La. Other peaks which are measurable and identifiable, together with the above nuclides, **SHALL** also be identified and reported.

Figure 3.1 - Prairie Island Nuclear Generating Plant Site Boundary For Liquid Effluents

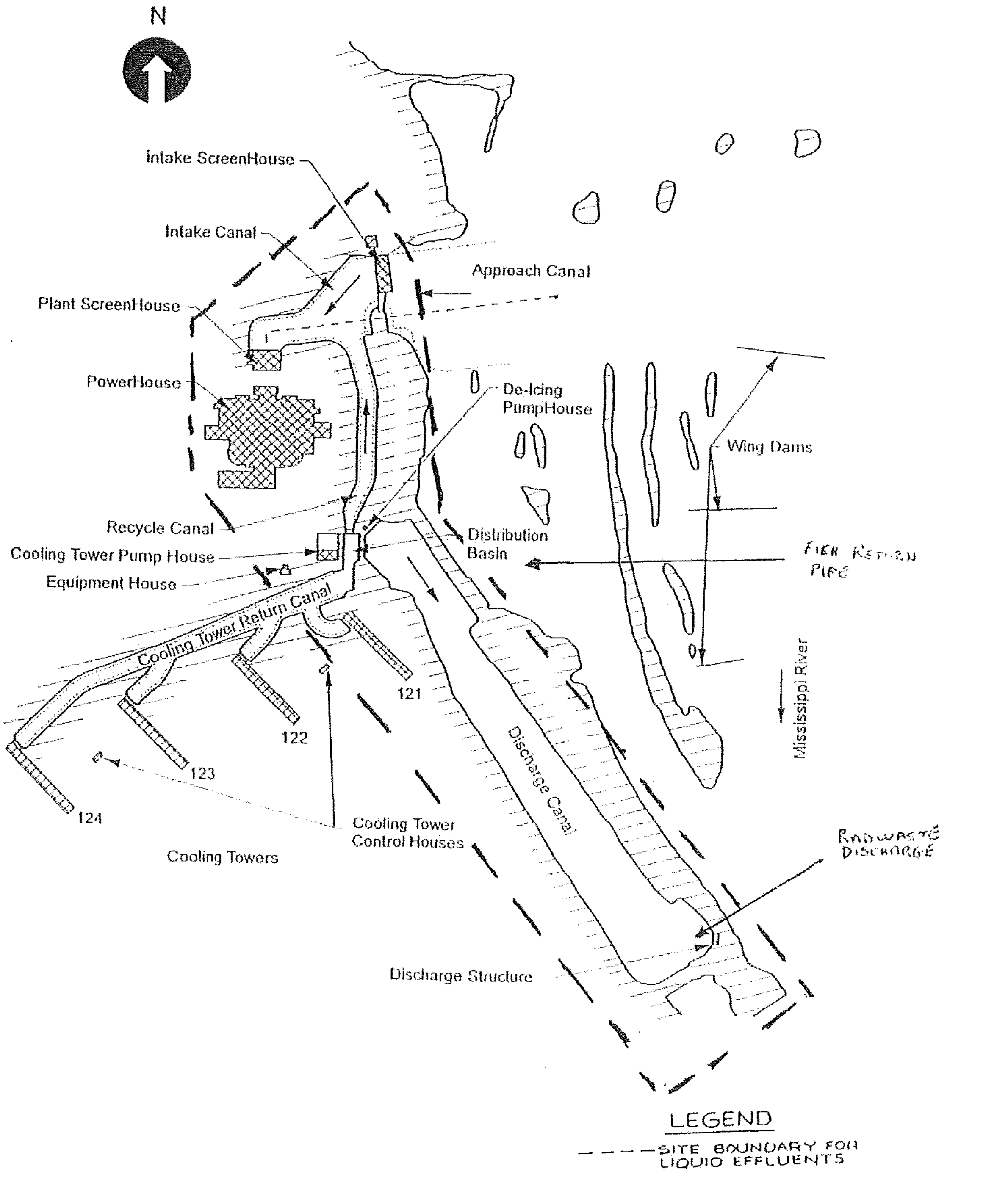

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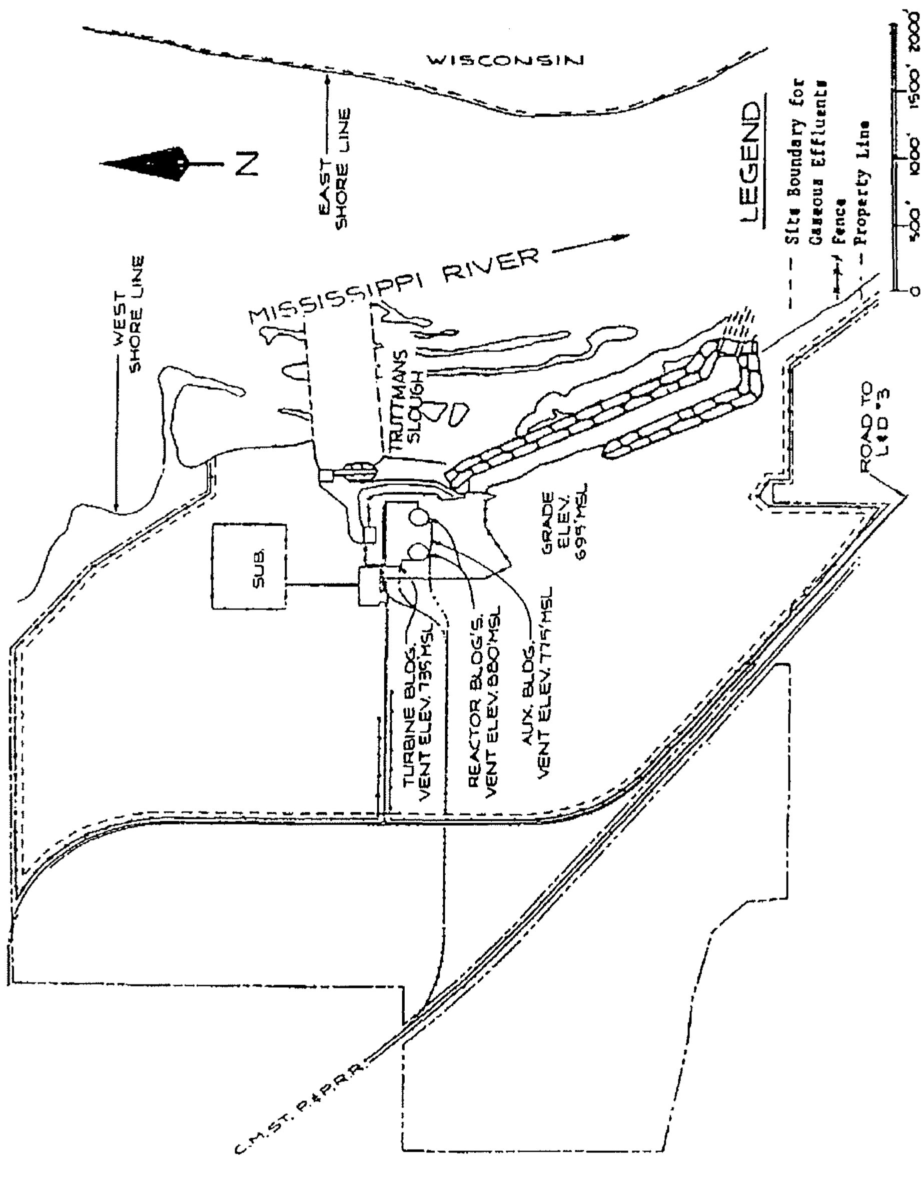
Figure 3.2 - Prairie Island Nuclear Generating Plant Site Boundary For Gaseous Effluents



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Appendix A Meteorological Analyses

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Appendix A

Summary of Dispersion Calculation Procedures

Undepleted, undecayed dispersion parameters were computed using the computer program XOQDOQ (Sagendorf and Goll, 1977). Specifically, sector average χ/Q and D/Q values were obtained for a sector width of 22.5 degrees. Building wake corrections were used to adjust calculations for ground-level releases. Standard open terrain recirculation correction factors were also applied as available as default values in XOQDOQ.

Dispersion calculations were based on ground level releases for the shield buildings, turbine buildings, and auxiliary building (hereafter referred to as the plant complex). A summary of release conditions used as input to XOQDOQ is presented in Table A-1 and controlling site boundary distances are defined in Table A-2. Computed X/Q and D/Q values for site boundary locations (relative to release points) and for standard distances to five miles are maintained in H4.2, “Offsite Dose Calculation Manual (ODCM) Supporting Data”.

Onsite meteorological data is collected over a representative time period. A 5 year period is suggested to ensure year to year variances do not bias the data set. This data reduced to joint frequency tables and used as input to the XOQDOQ determinations. Data is collected and delta-T stability classes are defined in conformance with NRC Regulatory Guide 1.23. Dispersion calculations for the plant complex is based on delta-T for 60 meter and 10 meter (joint data recovery of 90 percent. Joint frequency tables and resultant XOQDOQ determinations are maintained H4.2, "OFFSITE DOSE CALCULATION MANUAL (ODCM) SUPPORTING DATA". Meteorological data may be reassessed periodically to assure proper representation of local meteorological profiling.

REFERENCES

1. Sagendorf, J.F. and Goll, J.T., XOQDOQ Program for the Evaluation of Routine Effluent Releases at Nuclear Power Stations, NUREG-0324, U.S. Nuclear Regulatory Commission, September 1977.

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Table A-1 Prairie Island Release Conditions

	Shield Buildings	Auxiliary Building	Turbine Building
Type Release	Ground Level (Long Term and Short Term)	Ground Level (Long Term)	Ground Level (Long Term)
Release Point Height (m)	56.4	24.4	33.6, 12.2
Adjacent Building Height	62.2	62.2*	62.2*
Relative Location to Adjacent Structures	Adjacent to Auxiliary Building	Adjacent to Auxiliary Building	Adjacent to Auxiliary Building
Exit Velocity (m/sec)	N.A.	N.A.	N.A.
Internal Stack Diameter (m)	N.A.	N.A.	N.A.
Building Cross-Sectional Area (m^2)	2,170	2,170**	2,170**
Purge Frequency *** (times/yr)	20	N.A.	N.A.
Purge Duration*** (hours/release)	5	N.A.	N.A.

* Height of Shield Buildings

** Shield Building cross-sectional area

*** Applied to short-term calculations only

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Table A-2 Distances (Miles) to Controlling Site Boundary Locations As Measured from Edge of Plant Complex

Sector	Distance
N	0.28
NNE	0.26
NE	0.84*
ENE	0.62*
E	0.59*
ESE	0.61*
SE	0.67
SSE	0.43
S	0.43
SSW	0.40
SW	0.40
WSW	0.37
W	0.36
WNW	0.36
NW	0.43
NNW	0.48

*Over-water distances

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Appendix B Dose Parameters for Radioiodines, Particulates and Tritium

This appendix contains the methodology which was used to calculate the dose parameters for radioiodines, particulates, and tritium to show compliance with 10 CFR 20 and Appendix I of 10 CFR 50 for gaseous effluents. These dose parameters, P_i and R_i, were calculated using the methodology outlines in NUREG-0133 along with Regulatory Guide 1.109 Revision 1. The following sections provide the specific methodology which was utilized in calculating the P_i and R_i values for the various exposure pathways.

B.1 Calculation of P_i

The parameter, P_i, contained in the radioiodine and particulates portion of Section 5.2, includes pathway transport parameters of the ith radionuclide, the receptor's usage of the pathway media and the dosimetry of the exposure. Pathway usage rates and the internal dosimetry are functions of the receptor's age: however, the child age group, will always receive the maximum dose under the exposure conditions assumed.

B.1.1 Inhalation Pathway

$$P_i = K' (BR) DFA_i \qquad (B.1\text{-}1)$$

Where:

P_i = dose parameter for radionuclide i for the inhalation pathway, mrem/yr per µci/m3;

K′ = a constant of unit conversion:

= 10^6 pCi/µCi;

BR = the breathing rate of the child age group, m^3/yr;

DFA_i = the maximum organ inhalation dose factor for the child age group for radionuclide i, mrem/pCi.

Appendix B Dose Parameters for Radioiodines, Particulates and Tritium

The age group considered is the child group. The child's breathing rate is taken as 3700 m^3/yr from Table E-5 of Regulatory Guide 1.109 Revision 1. The inhalation dose factors for the child DFA_i, are presented in Table E-9 of Regulatory Guide 1.109 in units of mrem/pCi. The total body is considered as an organ in the selection of DFA_i. The incorporation of breathing rate of the child and the unit conversion factor results in the following:

$$P_i \quad = \quad 3.7 \times 109 \text{ DFAi} \qquad \text{(B.1-2)}$$

B.2 Calculation of R_i

The radioiodine and particulate specification is applicable to the location in the unrestricted area where the combination of existing pathways and receptor age groups indicates the maximum potential exposure occurs. The inhalation and ground plane exposure pathways **SHALL** be considered to exist at all locations. The grass-goat-milk, the grass-cow-milk, grass-cow-meat, and vegetation pathways are considered based on their existence at the various locations. R_i values have been calculated for the adult, teen, child, and infant age groups for the ground plane, cow milk, goat milk, vegetable and beef ingestion pathways. The methodology which was utilized to calculate these values is presented below.

B.2.1 <u>Inhalation Pathway</u>

$$R_i^l = K' \, (BR)_a \; (DFA_i)_a \qquad \text{(B.2-1)}$$

where:

R_i^l = dose factor for each identified radionuclide I of the organ of interest, mrem/yr per µCi/m^3

K′ = a constant of unit conversion:

= 10^6 pCi/µCi;

$(BR)_a$ = breathing rate of the receptor of age group a, m^3/yr;

Appendix B Dose Parameters for Radioiodines, Particulates and Tritium

$(DFA_i)_a$ = organ inhalation dose factor for radionuclide i for the receptor of age group a, mrem/pCi.

The breathing rates $(BR)_a$ for the various age groups are tabulated below, as given in Table E-5 of the Regulatory Guide 1.109 Revision 1.

Age Group (a)	Breathing Rate (m^3/yr)
Infant	1400
Child	3700
Teen	8000
Adult	8000

Inhalation dose factors $(DFA_i)_a$ for the various age groups are given in Tables E-7 through E-10 of Regulatory Guide 1.109 Revision 1.

B.2.2 Ground Plane Pathway

$$R_i^G = I_i K' K''(SF) DFG_i (1 - e^{-\lambda_i t})\lambda_i \quad \text{(B.2-2)}$$

where:

R_i^G = dose factor for the ground plane pathway for each identified radionuclide i for the organ of interest, m^2 -mrem/yr per μCi/sec per;

K′ = a constant of unit conversion;

= 10^6 pCi/μCi;

K″ = a constant of unit conversion;

= 8760 hr/year;

λ_i = the radiological decay constant for radionuclide i, sec^{-1};

t = the exposure time, sec;

= 4.73×10^8 sec (15 years)'

DFG_i = the ground plant dose conversion factor for radionuclide i; mrem/hr per pCi/m^2;

Appendix B Dose Parameters for Radioiodines, Particulates and Tritium

SF = the shielding factor (dimensionless)

I_i = factor to account for fractional deposition of radionuclide i.

For radionuclides other than iodine, the factor I_i is equal to one. For radioiodines, the value of I_i may vary. However, a value of 1.0 was used in calculating the R values in H4.2, "Offsite Dose Calculation Manual (ODCM) Supporting Data".

A shielding factor of 0.7 from Table E-15 of Regulatory Guide 1.109 Revision 1 is used. A tabulation of DFG_i values is presented in Table E-6 of Regulatory Guide 1.109 Revision 1.

B.2.3 Grass-Cow or Goat-Milk Pathway

$$R_i^M = I_i K' Q_F U_{ap} F_m (DFL_i)_a\ e^{-\lambda_i t_f} \left[f_p f_s \left[\frac{r\left(1-e^{-\lambda_{E_i} t_{ep}}\right)}{Y_p \lambda_{E_i}} + \frac{B_{iv}\left(1-e^{-\lambda_i t_b}\right)}{P_{\lambda_i}}\right] + \right.$$

$$\left. \left(1 - f_p f_s\right)\left[\frac{r\left(1-e^{-\lambda_{E_i} t_{es}}\right)}{Y_s \lambda_{E_i}} + \frac{B_{iv}\left(1-e^{-\lambda_i t_b}\right)}{P_{\lambda_i}}\right] e^{-\lambda_i t_h} \right] \quad \text{(B.2-3)}$$

where:

$R_{i\ M}^{M}$ = dose factor for the cow milk or goat milk pathway, for each identified radionuclide i for the organ of interest, m^2 - mrem/yr per µCi/sec;

K′ = a constant of unit conversion;

= 10^6 pCi/µCi;

Q_F = the cow's or goat's feed consumption rate, kg/day (wet weight);

U_{ap} = the receptor's milk consumption rate for age group a, liters/yr;

Appendix B Dose Parameters for Radioiodines, Particulates and Tritium

Y_P = the agricultural productivity by unit area of pasture feed grass, kg/m^2;

Y_S = the agricultural productivity by unit areas of stored feed, kg/m^2;

F_m = the stable element transfer coefficients, pCi/liter per pCi/day;

r = fraction of deposited activity retained on cow's feed grass;

$(DFL_i)_a$ = the organ ingestion dose factor for radionuclide I for the receptor in age group a, mrem/pCI;

λ_E = $\lambda_i + \lambda_w$;

λ_i = the radiological decay constant for radionuclide I, sec^{-1};

λ_w = the decay constant for removal of activity on leaf and plant surfaces by weathering, sec^{-1};

= $5.73 \text{ X } 10^{-7}$ sec^{-1} (corresponding to a 14 day half-lift);

t_f = the transport time from feed to cow or goat to milk to receptor, sec;

t_h = the transport time from harvest, to cow or goat, to consumption, sec;

t_b = period of time that activity builds up in soil, sec;

B_{iv} = concentration factor for uptake of radionuclide i from the soil by the edible parts of crops, pCi/kg (wet weight) per PCi/kg (dry soil);

P = effective surface density for soil, (dry weight) kg/m^2;

f_p = fraction of the year that the cow or goat is on pasture;

f_s = fraction of the cow feed that is pasture grass while the cow is on pasture;

Appendix B Dose Parameters for Radioiodines, Particulates and Tritium

t_{ep} = period of pasture grass exposure during the growing season, sec;

t_{es} = period of crop exposure during the growing season, sec;

I_i = factor to account for fractional deposition of radionuclide i.

For radionuclides other than iodine, the factor I_i is equal to one. For radioiodines, the value of I_i may vary. However, a value of 1.0 was used in calculating the R values in H4.2, "Offsite Dose Calculation Manual (ODCM) Supporting Data".

Milk cattle and goats are considered to be fed from two potential sources, pasture grass and stored feeds. Following the development in Regulatory Guide 1.109 Revision 1, the value of f_s was considered unity in lieu of site-specific information. The value of f_p was 0.5 based upon a 6-month grazing period.

Table B-1 contains the appropriate parameter values and their source in Regulatory Guide 1.109 Revision 1.

The concentration of tritium in milk is based on the airborne concentration rather than the deposition. Therefore, the R_i is based on X/Q:

$$R^M_{H-3} = K'K'' \; F_m Q_F U_{ap} (DFL_i)_a \; 0.75 \; (0.5/H) \qquad \text{(B.2-4)}$$

where:

R^M_{H-3} = dose factor for the cow or goat milk pathway for tritium for the organ of interest, mrem/yr per µCi/m^3;

K" = a constant of unit conversion;

= 10^3 gm/kg;

H = absolute humidity of the atmosphere, 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.

and other parameters and values are given below. A value of H of 8 grams/meter^3, was used in lieu of site-specific information.

Appendix B Dose Parameters for Radioiodines, Particulates and Tritium

B.2.4 Grass-Cow-Meat Pathway

The integrated concentration in meat follows in a similar manner to the development for the milk pathway, therefore:

$$R_i^B = I_i K'Q_F U_{ap} F_f (DFL_i)_a e^{-\lambda_i t_s} \left[f_p f_s \left[\frac{r\left(1-e^{-\lambda_{E_i} t_{ep}}\right)}{Y_p \lambda_{E_i}} + \frac{B_{iv}\left(1-e^{-\lambda_i t_b}\right)}{P \lambda_i} \right] \right.$$

$$\left. + \left(1 - f_p f_s\right) \left[\frac{r\left(1-e^{-\lambda_{E_i} t_{es}}\right)}{Y_s \lambda_{E_i}} + \frac{B_{iv}\left(1-e^{-\lambda_i t_b}\right)}{P \lambda_i} \right] e^{-\lambda_i t_h} \right] \quad \text{(B.2-5)}$$

where:

R_i^B = dose factor for the meat ingestion pathway for radionuclide i for any organ of interest, m^2 - mrem/yr per µCi/sec;

F_f = the stable element transfer coefficients, pCi/Kg per pCi/day;

U_{ap} = the receptor's meat consumption rate for age group a, kg/yr;

t_s = the transport time from slaughter to consumption, sec;

t_h = the transport time from harvest to animal consumption, sec;

t_{ep} = period of pasture grass exposure during the growing season, sec;

t_{es} = period of crop exposure during the growing season, sec;

I_i = factor to account for fractional disposition of radionuclide i.

For radionuclides other than iodine, the factor I_i is equal to one. For radioiodines, the value of I_i may vary. However, a value of 1.0 was used in calculating the R values in H4.2, "Offsite Dose Calculation Manual (ODCM) Supporting Data".

Appendix B Dose Parameters for Radioiodines, Particulates and Tritium

All other terms remain the same as defined in Equation B.2-3. Table B-2 contains the values which were used in calculating R_i for the meat pathway.

The concentration of tritium in meat is based on its airborne concentration rather than the deposition. Therefore, the R_i is based on X/Q.

$$R^{B}_{H-3} = K'K''\, F_f Q_F U_{ap} (DFL_i)_a\, 0.75\, (0.5/H) \quad \text{(B.2-6)}$$

where:

R_{T_B} = dose factor for the meat ingestion pathway for tritium for any organ of interest, mrem/yr per µCi/m^3.

All other terms are defined in Equation B.2-4 and B.2-5, above.

B.2.5 Vegetation Pathway

The integrated concentration in vegetation consumed by man follows the expression developed in the derivation of the milk factor. Man is considered to consume two types of vegetation (fresh and stored) that differ only in the time period between harvest and consumption, therefore:

$$R_i^V = I_i\, K'(DFL_i)a \left[U_a^L f_L e^{-\lambda_i t_L} \left[\frac{r\left(1-e^{-\lambda_{E_i} t_e}\right)}{Y_V \lambda_{E_i}} + \frac{B_{iv}(1-e^{-\lambda_i t_b})}{P\lambda_i} \right] + \right.$$

$$\left. U_a^S f_g e^{-\lambda_i t_h} \left[\frac{r\left(1-e^{-\lambda_{E_i} t_e}\right)}{Y_V \lambda_{E_i}} + \frac{B_{iv}(1-e^{-\lambda_i t_b})}{P\lambda_i} \right] \right] \quad \text{(B.2-7)}$$

where:

$R_i^V{}_V$ = dose factor for vegetable pathway for radionuclide i for organ of interest, m^2 - mrem/yr per µCi/sec;

K′ = a constant of unit conversion;

= 10^6 pCi/µCi;

Appendix B Dose Parameters for Radioiodines, Particulates and Tritium

U_a^L = the consumption rate of fresh leafy vegetation by the receptor in age group a, kg/yr;

U_a^S = the consumption the or stored vegetation by the receptor in age group a, kg/yr;

f_L = the fraction of the annual intake of fresh leafy vegetation grown locally;

f_g = the fraction of the annual intake of stored vegetation grown locally;

t_L = the average time between harvest of leafy vegetation and its consumption, sec;

t_h = the average time between harvest of stored vegetation and its consumption, sec;

Y_V = the vegetation aerial density, kg/m^2;

t_e = period of leafy vegetable exposure during growing season, sec;

I_i = factor to account for fractional deposition of radionuclide i.

For radionuclides other than iodine, the factor I_i is equal to one. For radioiodines, the value of I_i may vary. However, a value of 1.0 was used in calculating the R values in H4.2, "Offsite Dose Calculation Manual (ODCM) Supporting Data".

All other factors were defined above.

Table B-3 presents the appropriate parameter values and their source in Regulatory Guide 1.109 Revision 1.

In lieu of site-specific data default values for f_L and f_g, 1.0 and 0.76, respectively were used in the calculation of R_i. These values were obtained from Table E-15 of Regulatory Guide 1.109 Revision 1.

Appendix B Dose Parameters for Radioiodines, Particulates and Tritium

The concentration of tritium in vegetation is based on the airborne concentration rather than the deposition. Therefore, the R_i is based on X/Q:

$$R_{H-3}^{V} = K'K'' [U_a^L f_L + U_a^S f_g] (DFL_i)_a \; 0.75 \; (0.5/H) \qquad \text{(B.2-8)}$$

where:

R_{H-3}^{V} = dose factor for the vegetable pathway for tritium for any organ of interest, m^2 - mrem/yr per Ci/m^3.

All other terms remain the same as those in Equations B.2-4 and B.2-7.

The concentration of Carbon-14 in milk, meat, or vegetation, is based on the airborne concentration rather than the deposition. Therefore, the Ri is based on X/Q:

$$(R_{C-14})_{aj} = 10^9 * U_{C-14} * 0.11 * (DFL_{C-14})_{aj} * \; 1/0.19 \qquad \text{(B.2-9)}$$

where:

$(R_{C-14})_{aj}$ = Site specific Carbon-14 Dose Factor, for age group a, organ j , mrem/yr per µCi/m3

10^9 = a constant of unit conversion (pCi/uCi, gm/Kg)

U_{C-14} = Annual Carbon Ingestion via specific Pathway in Kg-Carbon per year for age group a

0.11 = Carbon Fraction (regulatory guide 1.109, Revision 1)

$(DFL_{C-14})_{aj}$ = C-14 Ingestion Dose Factor in mrem/pCi for age group a and organ j

0.19 = Atmospheric Concentration of Natural Carbon in gm/m^3
*based on 383 ppm

*stated value in Regulatory Guide 1.109, Revision 1, is 0.16. Due to atmospheric changes, latest EPA data is 0.19.

Table B-1 Parameters for Cow and Goat Milk Pathways

Parameter	Value	Reference (Reg. Guide 1.109 Rev. 1)
Q_F (kg/day)	50 (cow) 6 (goat)	Table E-3 Table E-3
Y_P (kg/m^2)	0.7	Table E-15
t_f (seconds)	1.73×10^5 (2 days)	Table E-15
r	1.0 (radioiodines) 0.2 (particulates)	Table E-15 Table E-15
$(DFL_i)_a$ (mrem/pCi)	Each radionuclide	Tables E-11 to E-14
F_m (pCi/day per pCi/liter)	Each stable element	Table E-1 (cow) Table E-2 (goat)
t_b (seconds)	4.73×10^8 (15 yr)	Table E-15
Y_s (kg/m^2)	2.0	Table E-15
Y_p (kg/m^2)	0.7	Table E-15
t_h (seconds)	7.78×10^6 (90 days)	Table E-15
U_{ap} (liters/yr)	330 infant 330 child 400 teen 310 adult	Table E-5 Table E-5 Table E-5 Table E-5
t_{ep} (seconds)	2.59×10^6 (30 days)	Table E-15
t_{es} (seconds)	5.18×10^6 (60 days)	Table E-15
B_{iv} (pCi/Kg (wet weight) per pCi/Kg (dry soil))	Each stable element	Table E-1
P (Kg/m^2 (dry weight))	240	Table E-15

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Table B-2 Parameters for the Meat Pathway

Parameter	Value	Reference (Reg. Guide 1.109 Rev. 1)
r	1.0 (radioiodines) 0.2 (particulates)	Table E-15 Table E-15
F_f (pCi/Kg per pCi/day)	Each stable element	Table E-1
U_{ap} (Kg/yr)	0 infant 41 child 65 teen 110 adult	Table E-5 Table E-5 Table E-5 Table E-5
$(DFL_i)_a$ (mrem/pCi)	Each radionuclide	Tables E-11 to E-14
Y_p (kg/m^2)	0.7	Table E-15
Y_s (kg/m^2)	2.0	Table E-15
t_b (seconds)	4.73×10^8 (15 yr)	Table E-15
t_s (seconds)	1.73×10^6 (20 days)	Table E-15
t_h (seconds)	7.78×10^6 (90 days)	Table E-15
t_{ep} (seconds)	2.59×10^6 (30 days)	Table E-15
t_{es} (seconds)	5.18×10^6 (60 days)	Table E-15
Q_f (kg/day)	50	Table E-3
B_{iv} (pCi/Kg (wet weight) per pCi/Kg (dry soil))	Each stable element	Table E-1
P (Kg/m^2 (dry weight))	240	Table E-15

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Table B-3 Parameters for the Vegetable Pathway

Parameter		Value	Reference (Reg. Guide 1.109 Rev. 1)
r (dimensionless)		1.0 (radioiodines)	Table E-1
		0.2 (particulates)	Table E-1
$(DFL_i)_a$ (mrem/pCi)		Each radionuclide	Tables E-11 to E-14
U_a^L (kg/yr)	- Infant	0	Table E-5
	- Child	26	Table E-5
	- Teen	42	Table E-5
	- Adult	64	Table E-5
U_a^S (kg/yr)	- Infant	0	Table E-5
	- Child	520	Table E-5
	- Teen	630	Table E-5
	- Adult	520	Table E-5
t_L (seconds)		8.6×10^4 (1 day)	Table E-15
t_h (seconds)		5.18×10^6 (60 days)	Table E-15
Y_v (kg/m^2)		2.0	Table E-15
t_e (seconds)		5.18×10^6 (60 days)	Table E-15
t_b (seconds)		4.73×10^8 (15 yr)	Table E-15
P(Kg/m^2 (dry weight))		240	Table E-15
B_{iv} (pCi/Kg (wet weight) per pCi/kg (dry soil))		Each stable element	Table E-1

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ENCLOSURE 4

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

D59 – PROCESS CONTROL PROGRAM FOR PROCESSING/DEWATERING OF RADIOACTIVE WASTE FROM LIQUID SYSTEMS

DATED – JANUARY 26, 2018

11 pages to follow

D	PROCESS CONTROL PROGRAM FOR PROCESSING/DEWATERING OF RADIOACTIVE WASTE FROM LIQUID SYSTEMS	NUMBER: D59
		REV: 12

Working Copy Verified			
Initial	Date	Initial	Date

WO: ______________

RESULTS/COMMENTS:

Work Request Initiated: YES ________ NO ______ Number: ____________________

PERFORMERS NAMES AND INITIALS

Print Name:	Initials:	Date:	Print Name:	Initials:	Date:
Print Name:	Initials:	Date:	Print Name:	Initials:	Date:
Print Name:	Initials:	Date:	Print Name:	Initials:	Date:

REFERENCE USE

- ***Procedure segments may be performed from memory.***
- ***Use the procedure to verify segments are complete.***
- ***Mark off steps within segment before continuing.***
- ***Procedure should be available at the work location.***

PORC REVIEW DATE:	APPROVAL:
1/26/18	PCR #: 6PCR01552638

TABLE OF CONTENTS

TABLE OF CONTENTS

1.0 GENERAL

1.1 Purpose

The purpose of this Process Control Program (PCP) is to detail the means by which the dewatering of radioactive waste from liquid systems can be assured, in accordance with applicable federal regulations and other requirements governing the disposal or processing solid radioactive waste.

1.2 Scope

This PCP includes the following processes:

1.2.1 Dewatering of bead resin.

1.2.2 Dewatering of powdered resin.

1.2.3 Dewatering of spent filter elements.

1.2.4 Reporting Requirements.

1.3 Definitions

1.3.1 Dewatering

The process of removing water from a substance to meet specific limits.

2.0 PROCESSING OF CERTAIN WASTE LIQUIDS THRU SPENT BEAD RESIN

2.1 Purpose

To establish an alternate method of processing certain waste liquids in lieu of solidification. This method utilizes spent bead resin to filter out suspended particulates allowing normal processing of the resultant liquid. Disposal volumes and personnel exposures are thus reduced.

2.2 Applicability

The following waste liquids may be processed using this procedure:

2.2.1 Laundry sludge

2.2.2 Decon solutions

2.2.3 Filter sludge

2.2.4 Mop bucket slurry

2.2.5 Tank bottoms

2.2.6 Sump bottoms

NOTE:	**Evaporator Concentrates may not be processed using this procedure.**

The above list is not to be considered complete. Items may be added or deleted upon evaluation of the Rad Materials Shipping Coordinator.

2.3 Sequence of Operation

2.3.1 Ensure there is a layer of bead resin in the liner to act as a filter (the type of liner is determined by the activity of the material to be disposed of).

2.3.2 Ensure adequate volume for the quantity of material to be processed.

2.3.3 Pump/pour liquid slurry into liner.

2.3.4 Flush drum and/or container, pump and hoses to liner.

2.4 Dewatering Procedure

Dewater as per Section 3.0 "Dewatering of Bead Resin" to ensure there is no free standing water in either the resin or the sludge.

3.0 DEWATERING OF BEAD RESIN

3.1 Purpose

To describe the process used to provide reasonable assurance that bead resin is dewatered to meet applicable disposal criteria.

3.2 Applicability

This section of the PCP is applicable disposal site or processor's criteria to all radioactively contaminated bead resin which is intended to be shipped dewatered (not solidified) for disposal.

3.3 Dewatering Procedure

The dewatering procedure varies with the supplier of the resin liner, with the type of liner, whether a steel liner or a high integrity container (HIC), and with the dewatering requirement of the disposal site. Individual shipping procedures unique to the particular container and disposal site or processor refer to the appropriate dewatering procedure.

In general, however, the dewatering process normally consists of the following steps after the liner has been filled:

3.3.1 Initial pumpdown with the diaphragm pump until suction is lost.

3.3.2 A waiting period (twenty hours, for example).

3.3.3 Final dewatering consisting of one or more pumpdowns using a diaphragm pump or a vacuum pumping system.

3.4 Verification of Dewatering

Preceding shipment, connect and operate the dewatering pump as before. IF no water is present, THEN the dewatering process is complete.

IF water is found, THEN pump until vacuum is lost. Repeat the pump/wait cycle as required. WHEN no more water can be removed, THEN the dewatering process is complete.

4.0 DEWATERING OF POWDERED RESIN

4.1 Purpose

To describe the process used to provide reasonable assurance that powdered resin is dewatered to meet applicable disposal site criteria.

4.2 Applicability

This section of the PCP is applicable to all radioactively contaminated powdered resin which is intended to be shipped for burial or processing.

4.3 System Description

Contaminated powdered resin originates in the Condensate Polishing System Filter Demineralizers of both units.

Spent resin is purged from the Filter Demineralizers to the Backwash Waste Receiving Tank where it awaits the dewatering/drying process.

The dewatering/drying process takes place in the Clamshell Backwash Waste Filter ("Clamshell").

There are two Clamshells to serve the needs of both units, each capable of being aligned to either unit. It is the function of the clamshells to filter the powdered resin out of the water-resin slurry that is pumped from the Backwash Waste Receiving Tank, thru the Clamshells. When a cake of resin develops in the Clamshell to a predetermined thickness, the filtering process automatically switches to a purge phase followed by a forced air drying phase. The duration of the air drying phase can be adjusted. Experience, however, has demonstrated that a drying cycle of approximately 12 minutes produces a product sufficiently dry to meet disposal site requirements yet not so dry as to create an airborne contamination hazard.

When the air-dry cycle is completed, the resin is dumped from the Clamshell into a hopper from which it is conducted down an enclosed chute to a container below. If the resin is insufficiently dried it will not flow freely down the chute.

4.4 Disposal

Powdered resin which has been processed thru the Clamshell system does not normally receive further dewatering treatment. Powdered resin may, therefore, be shipped in a container not fitted with dewatering equipment such as a steel drum or box. Because processed powdered resin is sufficiently dry to flow freely, and because powdered resin is normally very low in specific activity, if approved, it may be used to fill interstitial space in shipments of non-compatible trash or to fill voids in other shipping containers where they occur.

5.0 PROCESSING/DEWATERING OF SPENT FILTER ELEMENTS

5.1 Purpose

This section describes the method for processing spent filter elements and the process used to provide reasonable assurance that spent filter shipments are dewatered to meet applicable disposal site or processor's criteria.

5.2 Applicability

This section of the PCP is applicable to all radioactively contaminated filter elements intended for shipment for disposal or processing in the dewatered state (not solidified). Procedures specific to the appropriate type of container **SHALL** be employed.

5.3 Description of Filling Process

5.3.1 Verify that the container to be used is approved by the manufacturer for disposal of filter elements.

5.3.2 Ensure a dewatering element with an attached hose is installed in the container. The dewatering elements must be compatible with the dewatering pump.

5.3.3 Filter elements should be drained of excess water prior to placing in the container.

5.3.4 Place filter elements into the container while attempting to avoid bridging of filters and observing the principles of ALARA.

5.4 Dewatering

The dewatering process may vary with type and manufacture of container and with requirements of the disposal facility. Typically, however, the dewatering process consists of the following steps:

5.4.1 Allow wait period (typically 20 to 24 hours) for water if present to migrate to the bottom of the container.

5.4.2 Connect the dewatering pump to the dewatering element hose. Conduct the pump discharge hose to a container to enable monitoring of discharge volume.

5.4.3 Start the dewatering pump. IF no water is found, THEN the container may be considered to be dewatered.

IF water is found, THEN pump until vacuum is lost, THEN stop the pump and begin another wait period.

Repeat the pump/wait cycle until no more water can be removed.

5.5 Verification of Dewatering

Preceding shipment, connect and operate the dewatering pump as before. IF no water is present, THEN the dewatering process is complete.

IF water is found, THEN pump until vacuum is lost. Repeat the pump/wait cycle as required. WHEN no more water can be removed, THEN the dewatering process is complete.

6.0 REPORTING REQUIREMENTS

6.1 Purpose

This section of the PCP sets forth the reporting requirements are as they apply to this PCP to ensure that the reports are completed accurately and in a timely manner.

6.2 Applicability

This section of the PCP, in whole or part, applies to all sections of the PCP.

6.3 References

Waste Form Technical Position, Revision 1. United States Nuclear Regulatory Commission.

6.4 PCP Revisions

Whenever the PCP is revised or changed, a description of the changes AND justifications **SHALL** be included in the Annual Radioactive Effluent Release Report.

6.5 Reports of Mishaps

Waste form mishaps **SHALL** be reported to the NRC (Director of the Division of Low-Level Waste Management and Decommissioning) AND the designated State disposal site regulatory authority within 30 days of knowledge of the incident. Mishaps are defined as failure of misuse of stabilized waste forms or containers that provide stability (HIC's). Such mishaps include, but are not necessarily limited to, the following:

6.5.1 The failure of high integrity containers used to ensure structural stability.

6.5.2 The misuse of high integrity containers, as evidenced by excessive free liquid, or excessive void space within the container.

6.5.3 Production of a solidified Class B or Class C waste form that exhibits any of the characteristics listed in the Waste Form Technical Position, Revision 1.

6.6 PCP Specimen Summary Reports

WHENEVER cement stabilization (as defined by 10 CFR 61) of low-level waste is necessary, THEN PCP test specimens are required for verification and surveillance. Verification specimens are intended to provide assurance that the formulations used in the qualification testing program correspond to those actually used in the field. Surveillance specimens are intended to provide verification that the waste forms remain stable with time. A summary report **SHALL** be prepared annually and submitted to the NRC (Director, Division of Low-Level Waste Management and Decommissioning) documenting the results of tests performed on the cement-stabilized waste form surveillance specimens during the calendar year.

The annual report should be submitted within 90 days of the end of each calendar year.

7.0 ATTACHMENTS

NONE