ATTACHMENT 1

Nuclear Management Company
Letter L-PI-03-052

Off-Site Radiation Dose Assessment for
January through December 2002

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

January through December 2002

An Assessment of the radiation dose due to releases from Prairie Island Nuclear Generating Plant during 2002 was performed in accordance with the Offsite Dose Calculations Manual 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 meteorological data from the Off-site Dose Calculation Manual were used in making this assessment. Source terms were obtained from the Annual Radioactive Effluent and Waste Disposal Report prepared for NRC review for the year of 2002.

Off-site 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. Doses are a small percentage of Appendix I Guidelines.

Off-site Doses from Liquid Release

Computed doses due to liquid releases are reported in Table 1. Critical receptor information is reported in Table 2. 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 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 dose from noble gas releases and the whole body and organ doses from the inhalation pathway due to Iodine 131, Iodine-133, tritium and long-lived particulates were calculated for this location and occupancy time. These doses were reported in Table 1.

Doses to Individuals Due to Effluent Releases from the ISFSI

Three fuel casks were loaded and placed in the storage facility during the 2002 calendar year. The total number of casks in the ISFSI is seventeen. There has been no release of radioactive effluents from the ISFSI.

CURRENT ODCM REVISION

The Offsite Dose Calculations Manual was revised this year. The current revision is 17. The revision date is October 13, 2002. A copy is included with this report.

AIRBORNE ABNORMAL RELEASE

There were no abnormal airborne releases for the year 2002.

Deviation in Definition of Reporting Periods

In May of 2002, during a review of the Offsite Dose Calculation Manual (ODCM), a discrepancy was noted, between the requirements and current practice. Limits in the ODCM, for airborne effluent noble gas, iodine, tritium and long-lived particulates are stated by calendar quarter and calendar year. Prairie Island's practice was to calculate offsite dose based on a 13 week rolling quarter, with 4 quarters defining an effluent year. This means that a Prairie Island effluent quarter and year may straddle the calendar period by as much as 6 days.

Cause: Based on the chemistry technician's routine of evaluating airborne releases on a weekly basis, this approach worked well in practice, made sense and was in compliance with the Radioactive Effluents Technical Specifications (RETS).

The RETS were moved from Technical Specifications to the ODCM in 1995. The terminology of NUREG 1301 was adopted. The change in terminology to "calendar quarter" and "calendar year" was not noted. The 2002 review determined Prairie Island did not meet the verbatim requirement.

Corrective Actions: The event was documented and assessed by Chemistry Management through the Condition Reporting System, Condition Report number CAP023342.

This issue was discussed with the Region 3 NRC Radiation Protection Inspector, Radiation Protection Manager and Chemistry Manager.

The associated software was modified to divide the dose by calendar quarter and year as required.

Significance: For previous year's reporting (pre 2001), no release or dose was unaccounted for. The end of one reporting period was the beginning of the next reporting period.

For the years bracketing the change in definition of quarters and years (2001-2002), no release or dose was unaccounted for. By the previous year definition, the 2001 annual report covered a period ending on January 6, 2002. By the current year definition, the 2002 annual report covers a period beginning on January 1, 2002. Six days are doubly accounted for.

This discrepancy did not impose upon the health and safety of the public.

Table 1

OFF-SITE RADIATION DOSE ASSESSMENT - PRAIRIE ISLAND

PERIOD: JANUARY through DECEMBER 2002

		10 CFR Part 50 Appendix I Guidelines for a 2-unit site per year
Gaseous Releases		
Maximum Site Boundry Gamma Air Dose (mrad)	1.30E-04	20
Maximum Site Boundry Beta Air Dose (mrad)	1.48E-02	40
Maximum Off-site Dose to any organ (mrem)*	6.34E-02	30
Offshore Location		
Gamma Dose (mrad)	6.68E-06	
Total Body (mrem)*	2.95E-03	
Organ (mrad)*	2.95E-03	30
Liquid Releases		
Maximum Off-site Dose Total Body (mrem)	2.48E-03	6
Maximum Off-site Dose Organ - GI TRACT (mrem)	3.46E-03	20
Limiting Organ Dose Organ - TOTAL BODY (mrem)	2.48E-03	6

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

Table 2

OFF-SITE RADIATION DOSE ASSESSMENT - PRAIRIE ISLAND
SUPPLEMENTAL INFORMATION

PERIOD: JANUARY through DECEMBER 2002

Gaseous Releases

Maximum Site Boundary
Dose Location
(From Building Vents)

Sector		WNW
Distance	(miles)	0.4

Offshore Location
Within Site Boundary

Sector		ESE
Distance	(miles)	0.2
Pathway		Inhalation

Maximum Off-site

Sector		SSE
Distance	(miles)	0.6
Pathways		Plume, Ground, Inhalation, Vegetables
Age Group		Child

Liquid Releases

Maximum Off-site Dose
Location Downstream

Pathway	Fish

ATTACHMENT 2

Nuclear Management Company
Letter L-PI-03-052

Annual Radioactive Effluent Report
Supplemental Information
Revision 0

ANNUAL RADIOACTIVE EFFLUENT REPORT

01-JAN-02 THROUGH 31-DEC-02

SUPPLEMENTAL INFORMATION

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	≤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	≤15 mrem/quarter to any organ
	≤30 mrem/year to any organ

B. Water 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:

   2.0E-04 uCi/ml Total Activity

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	Gem Gem	±25%
2.	Iodines in gaseous releases:	Total Nuclide	Gem Gem	±25%
3.	Particulates in gaseous releases:	Total Nuclide	Gem Gem	±25%
4.	Liquid effluents	Total Nuclide	Gem Gem	±25%

E. Manual Revisions

1. Offsite Dose Calculations Manual latest Revision number: 10/13/02

   Revision date : 17

1.0 BATCH RELEASES (LIQUID)

	QTR: 01	QTR: 02	QTR: 03	QTR: 04
1.1 NUMBER OF BATCH RELEASES	5.60E+01	1.90E+01	3.10E+01	7.40E+01
1.2 TOTAL TIME PERIOD (HRS)	9.82E+01	3.26E+01	5.29E+01	1.35E+02
1.3 MAXIMUM TIME PERIOD (HRS)	2.75E+00	2.00E+00	2.40E+00	3.50E+00
1.4 AVERAGE TIME PERIOD (HRS)	1.75E+00	1.71E+00	1.71E+00	1.83E+00
1.5 MINIMUM TIME PERIOD (HRS)	1.43E+00	1.50E+00	2.50E-01	7.00E-01
1.6 AVERAGE MISSISSIPPI RIVER FLOW (CFS)	1.09E+04	3.37E+04	2.94E+04	1.97E+04

2.0 BATCH RELEASES (AIRBORNE)

	QTR: 01	QTR: 02	QTR: 03	QTR: 04
2.1 NUMBER OF BATCH RELEASES	1.50E+01	1.00E+00	0.00E+00	1.10E+01
2.2 TOTAL TIME PERIOD (HRS)	1.90E+02	6.22E+00	0.00E+00	1.09E+02
2.3 MAXIMUM TIME PERIOD (HRS)	2.40E+01	6.22E+00	0.00E+00	2.40E+01
2.4 AVERAGE TIME PERIOD (HRS)	1.27E+01	6.22E+00	0.00E+00	9.94E+00
2.5 MINIMUM TIME PERIOD (HRS)	1.28E-01	6.22E+00	0.00E+00	1.87E-02

3.0 ABNORMAL RELEASES (LIQUID)

	QTR: 01	QTR: 02	QTR: 03	QTR: 04
3.1 NUMBER OF BATCH RELEASES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
3.2 TOTAL ACTIVITY RELEASED (CI)	0.00E+00	0.00E+00	0.00E+00	0.00E+00
3.3 TOTAL TRITIUM RELEASED (CI)	0.00E+00	0.00E+00	0.00E+00	0.00E+00

4.0 ABNORMAL RELEASES (AIRBORNE)

	QTR: 01	QTR: 02	QTR: 03	QTR: 04
4.1 NUMBER OF BATCH RELEASES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
4.2 TOTAL ACTIVITY RELEASED (CI)	0.00E+00	0.00E+00	0.00E+00	0.00E+00

TABLE 1A
GASEOUS EFFLUENTS - SUMMATION OF ALL RELEASES

	QTR: 01	QTR: 02	QTR: 03	QTR: 04
5.0 FISSION AND ACTIVATION GASES				
5.1 TOTAL RELEASE (CI)	1.27E+00	5.42E-01	0.00E+00	0.00E+00
5.2 AVERAGE RELEASE RATE (UCI/SEC)	1.61E-01	6.90E-02	0.00E+00	0.00E+00
5.3 GAMMA DOSE (MRAD)	9.13E-05	3.90E-05	0.00E+00	0.00E+00
5.4 BETA DOSE (MRAD)	1.03E-02	4.42E-03	0.00E+00	0.00E+00
5.5 PERCENT OF GAMMA TECH SPEC (%)	9.13E-04	3.90E-04	0.00E+00	0.00E+00
5.6 PERCENT OF BETA TECH SPEC (%)	5.17E-02	2.21E-02	0.00E+00	0.00E+00
6.0 IODINES				
6.1 TOTAL I-131 (CI)	9.21E-07	0.00E+00	0.00E+00	0.00E+00
6.2 AVERAGE RELEASE RATE (UCI/SEC)	1.17E-07	0.00E+00	0.00E+00	0.00E+00
7.0 PARTICULATES				
7.1 TOTAL RELEASE (CI)	1.99E-08	0.00E+00	0.00E+00	7.31E-06
7.2 AVERAGE RELEASE RATE (UCI/SEC)	2.53E-09	0.00E+00	0.00E+00	9.30E-07
8.0 TRITIUM				
8.1 TOTAL RELEASE (CI)	5.07E+00	8.57E+00	9.07E+00	8.76E+00
8.2 AVERAGE RELEASE RATE (UCI/SEC)	6.45E-01	1.09E+00	1.15E+00	1.11E+00
9.0 TOTAL IODINE, PARTICULATE AND TRITIUM (UCI/SEC)	6.45E-01	1.09E+00	1.15E+00	1.11E+00
10.0 DOSE FROM IODINE, LLP, AND TRITIUM (MREM)	1.30E-02	1.53E-02	1.62E-02	1.88E-02
11.0 PERCENT OF TECH SPEC (%)	8.70E-02	1.02E-01	1.08E-01	1.25E-01
12.0 GROSS ALPHA (CI)	0.00E+00	0.00E+00	0.00E+00	0.00E+00

TABLE 1C
GASEOUS EFFLUENTS - GROUND LEVEL RELEASES (CI)

13.0 FISSION AND ACTIVATION GASES

		CONTINUOUS MODE				BATCH MODE			
NUCLIDE	UNITS	QTR: 01	QTR: 02	QTR: 03	QTR: 04	QTR: 01	QTR: 02	QTR: 03	QTR: 04
KR-85	CI					1.27E+00	5.42E-01		
TOTALS	CI	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.27E+00	5.42E-01	0.00E+00	0.00E+00

14.0 IODINES

		CONTINUOUS MODE				BATCH MODE			
NUCLIDE	UNITS	QTR: 01	QTR: 02	QTR: 03	QTR: 04	QTR: 01	QTR: 02	QTR: 03	QTR: 04
I-131	CI	9.21E-07							
TOTALS	CI	9.21E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

TABLE 1C
GASEOUS EFFLUENTS - GROUND LEVEL RELEASES (CONTINUED)

15.0 **PARTICULATES**

		CONTINUOUS MODE				BATCH MODE			
NUCLIDE	UNITS	QTR: 01	QTR: 02	QTR: 03	QTR: 04	QTR: 01	QTR: 02	QTR: 03	QTR: 04
CO-58	CI				7.31E-06				
CS-134	CI					1.46E-09			
CS-137	CI					1.84E-08			
TOTALS	CI	0.00E+00	0.00E+00	0.00E+00	7.31E-06	1.99E-08	0.00E+00	0.00E+00	0.00E+00

TABLE 1A
LIQUID EFFLUENTS - SUMMATION OF ALL RELEASES

	QTR: 01	QTR: 02	QTR: 03	QTR: 04
16.0 VOLUME OF WASTE PRIOR TO DILUTION (LITERS)	5.82E+07	5.26E+07	6.22E+07	9.50E+07
17.0 VOLUME OF DILUTION WATER (LITERS)	1.69E+11	1.10E+11	2.67E+11	1.76E+11
18.0 **FISSION AND ACTIVATION PRODUCTS**				
18.1 TOTAL RELEASES W/O H-3, RADGAS, ALPHA (CI)	2.04E-02	9.50E-03	1.64E-02	6.16E-02
18.2 AVERAGE DILUTION CONCENTRATION (UCI/ML)	1.21E-10	8.68E-11	6.14E-11	3.51E-10
19.0 **TRITIUM**				
19.1 TOTAL RELEASE (CI)	9.89E+01	1.12E+02	1.44E+02	1.61E+02
19.2 AVERAGE DILUTION CONCENTRATION (UCI/ML)	5.86E-07	1.02E-06	5.40E-07	9.16E-07
20.0 **DISSOLVED AND ENTRAINED GASES**				
20.1 TOTAL RELEASE (CI)	7.65E-04	0.00E+00	7.67E-05	2.65E-04
20.2 AVERAGE DILUTION CONCENTRATION (UCI/ML)	4.54E-12	0.00E+00	2.87E-13	1.51E-12
21.0 GROSS ALPHA (CI)	0.00E+00	0.00E+00	0.00E+00	0.00E+00
22.0 TOTAL TRITIUM, FISSION & ACTIVATION PRODUCTS (UCI/ML)	5.86E-07	1.02E-06	5.40E-07	9.17E-07
23.0 TOTAL BODY DOSE (MREM)	5.21E-04	2.70E-04	3.40E-04	1.35E-03
24.0 **CRITICAL ORGAN**				
24.1 DOSE (MREM)	5.21E-04	2.70E-04	3.40E-04	1.35E-03
24.2 ORGAN	TOT BODY	TOT BODY	TOT BODY	TOT BODY
25.0 PERCENT OF TECHNICAL SPECIFICATIONS LIMIT (%)	1.74E-02	9.00E-03	1.13E-02	4.49E-02
26.0 PERCENT OF CRITICAL ORGAN TECH SPEC LIMIT (%)	1.74E-02	9.00E-03	1.13E-02	4.49E-02

TABLE 2A
LIQUID EFFLUENTS - SUMMATION OF ALL RELEASES (CI)

27.0 **INDIVIDUAL LIQUID EFFLUENT**

		CONTINUOUS MODE				BATCH MODE			
NUCLIDE	UNITS	QTR: 01	QTR: 02	QTR: 03	QTR: 04	QTR: 01	QTR: 02	QTR: 03	QTR: 04
AG-110M	CI					2.22E-04	1.37E-03	1.53E-03	1.51E-03
BE-7	CI								7.29E-06
CO-57	CI					1.49E-05	1.23E-05	7.99E-07	1.49E-05
CO-58	CI					6.01E-03	2.38E-03	9.02E-04	1.71E-02
CO-60	CI					9.05E-04	7.20E-04	7.16E-04	9.90E-04
CR-51	CI					8.66E-04		6.75E-05	3.53E-03
CS-134	CI					3.75E-06			8.70E-05
CS-137	CI	6.72E-05				5.88E-06	2.49E-06	2.04E-06	1.14E-04
CU-64	CI								5.24E-04
FE-55	CI					7.36E-03	3.58E-03	6.30E-03	8.30E-03
FE-59	CI					7.03E-04	7.67E-05	5.62E-05	6.99E-04
I-133	CI								2.42E-07
LA-140	CI					6.53E-05			2.88E-04
MN-54	CI					1.37E-05	1.81E-05	2.59E-05	5.22E-05
NB-95	CI						3.50E-05	1.81E-05	1.05E-04
NB-97	CI					9.99E-06	9.55E-07	1.30E-06	
RH-105	CI					7.32E-06			
RU-103	CI								2.11E-06
SB-122	CI					5.33E-04			1.84E-04
SB-124	CI					1.03E-03	3.91E-04	1.61E-03	2.57E-03
SB-125	CI					1.58E-03	8.51E-04	5.06E-03	1.35E-02
SB-126	CI					1.83E-06			1.04E-05
SN-113	CI					2.57E-06	5.96E-06	9.10E-06	9.08E-05
SR-85	CI					1.41E-06			
SR-92	CI					3.40E-07	2.15E-05	4.45E-05	2.06E-05

(CONTINUED)

TABLE 2A
LIQUID EFFLUENTS - SUMMATION OF ALL RELEASES (CONTINUED)

27.0 INDIVIDUAL LIQUID EFFLUENT

		CONTINUOUS MODE				BATCH MODE			
NUCLIDE	UNITS	QTR: 01	QTR: 02	QTR: 03	QTR: 04	QTR: 01	QTR: 02	QTR: 03	QTR: 04
TE-123M	CI					6.95E-07			1.04E-04
TE-125M	CI					9.94E-04			1.17E-02
W-187	CI						1.47E-05	5.30E-05	8.30E-06
Y-91M	CI								1.21E-06
ZN-65	CI					1.53E-06	1.71E-06		6.07E-05
ZR-95	CI						2.79E-05	1.81E-05	6.59E-05
TOTALS	CI	6.72E-05	0.00E+00	0.00E+00	0.00E+00	2.03E-02	9.50E-03	1.64E-02	6.16E-02

28.0 DISSOLVED AND ENTRAINED GASES

		CONTINUOUS MODE				BATCH MODE			
NUCLIDE	UNITS	QTR: 01	QTR: 02	QTR: 03	QTR: 04	QTR: 01	QTR: 02	QTR: 03	QTR: 04
KR-85	CI					3.25E-04			
KR-85M	CI					3.93E-07			
XE-133	CI					4.39E-04		7.48E-05	2.65E-04
XE-135	CI					1.17E-06		1.91E-06	
TOTALS	CI	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.65E-04	0.00E+00	7.67E-05	2.65E-04

ATTACHMENT 3

Nuclear Management Company
Letter L-PI-03-052

Effluent and Waste Disposal Annual Report
Solid Waste and Irradiated Fuel Shipments

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
NORTHERN STATES POWER

Period: 01/01/02 to 12/31/02
License No. DPR-42/60

EFFLUENT AND WASTE DISPOSAL ANNUAL REPORT
SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

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

1. Solid Waste Total Volumes and Total Curie Quantities:

TYPE OF WASTE	UNITS	PERIOD TOTALS (0.00 E00)	EST. TOTAL ERROR, % (0.00 E00)	CONTAINER DISPOSAL VOL (ft^3) (LIST)
A. Resins	m^3	2.11E+01		
Bead Resin	ft^3	7.43E+02		179.4
Powdex Resin	Ci	1.73E+00	+/- 2.50E+01	94
B. Dry-Compacted	m^3			
	ft^3			
	Ci			
C. Non-Compacted	m^3	9.58E+01		
DAW (DAW/Metal/Wood)	ft^3	3.38E+03		94
	Ci	6.03E-01	+/- 2.50E+01	
D. Filter Media	m^3			
	ft^3			
	Ci			
S. Other (furnish description)	m^3	1.55E+01		
Combined Package (Resin/Filters)	ft^3	5.47E+02		179.4
Sludge/Soil	Ci	2.60E+01	+/- 2.50E+01	94

NOTE:	**The solid waste information provided in this report is the volume and activity of the low-level waste leaving the Prairie Island site. No allowance is made for off-site volume reduction prior to disposal.**

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
NORTHERN STATES POWER

Period: 01/01/02 to 12/31/02
License No. DPR-42/60

EFFLUENT AND WASTE DISPOSAL ANNUAL REPORT
SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

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

2. Principal Radionuclide Composition by Type of Waste:
(Bold letter designation from Page 1)

TYPE	Nuclide	Percent % Abundance (0.00E0)
A	*H-3	4.94E+00
	Mn-54	1.05E+00
	*Fe-55	4.20E+00
	Co-58	1.10E+00
	Co-60	2.17E+00
Note: 1% cutoff	*Ni-63	3.97E+00
	Cs-134	4.28E+01
	Cs-137	3.85E+01
C	Mn-54	1.12E+00
	*Fe-55	6.82E+01
	Co-58	3.27E+00
	Co-60	1.04E+01
	*Ni-63	9.62E+00
Note: 1% cutoff	Zr-95	1.63E+00
	Sb-125	1.15E+00
	Cs-134	2.13E+00

* = Inferred - Not Measured on Site

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
NORTHERN STATES POWER

Period: 01/01/02 to 12/31/02
License No. DPR-42/60

EFFLUENT AND WASTE DISPOSAL ANNUAL REPORT
SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

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

2. Principal Radionuclide Composition by Type of Waste (Continuation):
(Bold letter designation from Page 1)

TYPE	Nuclide	Percent % Abundance (0.00E0)
S	*C-14	4.21E+00
	*Fe-55	5.01E+01
	Co-58	4.30E+00
	Co-60	1.87E+01
	*Ni-63	1.91E+01
Note: 1% cutoff	Cs-137	1.13E+00

* = Inferred - Not Measured on Site

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
NORTHERN STATES POWER

Period: 01/01/02 to 12/31/02
License No. DPR-42/60

EFFLUENT AND WASTE DISPOSAL ANNUAL REPORT
SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

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

3. Solid Waste Disposition:

Number of Shipments	Mode	Destination
3	Hittman Transport	Barnwell Disposal Facility
3	Kindrick Trucking	US Ecology, Inc. NMMC

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
NORTHERN STATES POWER

Period: 01/01/02 to 12/31/02
License No. DPR-42/60

EFFLUENT AND WASTE DISPOSAL ANNUAL REPORT
SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

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

4. Shipping Container and Solidification Method:

No.	Disposal Volume (Ft^3/m^3)	Activity (Ci)	Type of Waste	Container Code	Solidif. Code
02-001	1410/39.9	00.337	C	L	N/A
02-002	1410/39.9	00.221	C	L	N/A
02-003	1316/37.3	00.045	C	L	N/A
02-017	179.4/5.08	13.600	S	L	N/A
02-018	179.4/5.08	12.400	S	L	N/A
02-019	179.4/5.08	1.730	A	L	N/A
TOTALS 6	4674/132	28.300			

CONTAINER CODES: (Shipment type)
- L = LSA
- A = Type A
- B = Type B
- Q = Highway Route Controlled Quantity

SOLIDIFICATION CODES:
- C = Cement

TYPES OF WASTES:
- A = Resins
- B = Dry Compacted
- C = Non-Compacted
- D = Filter Media
- S = Other

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
NORTHERN STATES POWER

Period: 01/01/02 to 12/31/02
License No. DPR-42/60

EFFLUENT AND WASTE DISPOSAL ANNUAL REPORT
SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

B. IRRADIATED FUEL SHIPMENTS (DISPOSITION)

Number of Shipments	Mode	Destination
0		

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
NORTHERN STATES POWER

Period: 01/01/02 to 12/31/02
License No. DPR-42/60

EFFLUENT AND WASTE DISPOSAL ANNUAL REPORT
SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

C. PROCESS CONTROL PROGRAM CHANGES

TITLE: Process Control for Solidification/Dewatering of Radioactive Waste from Liquid Systems

Current Revision Number: 8 Effective Date: 08/25/1999

NOTE:	**If the effective date of the PCP is within the period covered by this report, then a description and justification of the changes to the PCP is required (T.S.6.5.D) (IT.S.5.5.4). Attach the sidelined pages to this report.**

Changes/Justification: N/A

ATTACHMENT 4

Nuclear Management Company
Letter L-PI-03-052

Offsite Dose Calculation Manual
Revision 17

SUMMARY OF CHANGES TO OFFSITE DOSE CALCULATION MANUAL

REV 17

The Offsite Dose Calculations Manual, ODCM, was revised to support the Improved Technical Specification update. Those changes include:

1) Numerous Technical Specification reference changes. Every T.S. reference in the manual has changed.

2) Change of the surveillance period of "monthly" to "at least every 31 days".

3) All references to the OLD 10 CFR-20 and Maximum Permissible Concentration have been removed, including Table 4.3, which was the old 10 CFR-20 Appendix B, Table 2, Column 2 data. Liquid release limits are now based on ten times the current 10 CFR-20 water effluent concentration values as defined in ITS.

Other ODCM revisions not related to ITS include:

1) References to Northern States Power were removed.

2) The airborne pathway dose factors for all pathways (Tables 5.5-1 through 5.5-19) were expanded to include all the nuclides listed in Regulatory Guide 1.109.

3) Table 1.1, Operational Modes, was deleted since it was not referenced in the ODCM.

4) The Radiological Environmental Monitoring Plan, REMP, milk sampling frequency was revised to comply with NUREG 1301.

5) A typographical error was corrected in Appendix C, Table C-3 to change the particulate value for "r" from 2.0 to 0.2.

Evaluation of the above listed changes resulted in the determination that the changes maintain the levels of radioactive effluent control required by 10 CFR 20.1301(a), 10 CFR 50.36a, 10 CFR 50, Appendix A (GDC 60 & 64) and Appendix I, 40 CFR 190, and do not adversely impact the accuracy or reliability of effluent dose or setpoint calculations at Prairie Island.

H	OFFSITE DOSE CALCULATION MANUAL (ODCM)	NUMBER: H4 REV: 17 Page 1 of 221

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.*

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Section Title Page

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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.

RECORD OF REVISIONS [CONTINUED]

Revision No.	Date	Reason for Revision
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.
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.

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 10CFR 20.1301(a)(1), 10CFR 50.36A, 10CFR 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 for implementation of this manual are provided in the Count Room Manual, (Radiation Protection Implementing Procedures 4000 Series).

Also included in this manual is 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.

Licensee initiated changes to the ODCM:

1. **SHALL** be documented and records of reviews performed shall contain:
   a. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s).
   b. A determination that the change(s) maintain the level of radioactive effluent control required by 10CFR20.1301(a)(1), 10CFR50.36A, 40CFR190, 10CFR50, 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**

  CHANNEL CHECK is a quantitative determination of acceptable operability by observation of channel behavior during operation. This determination **SHALL** include comparison of the channel with other independent channels measuring the same variable.

- **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 OPERABLE, including alarm and/or trip initiating action.

- **CHANNEL RESPONSE TEST**

  A CHANNEL RESPONSE TEST consists of injecting a simulated signal into the channel as near the sensor as practicable to measure the time for electronics and relay actions, and trip functions.

- **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. The dose conversion factors used for this calculation **SHALL** be the child thyroid factors listed in Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

- **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.

- **GASEOUS RADWASTE TREATMENT SYSTEM**

  The GASEOUS RADWASTE TREATMENT SYSTEM **SHALL** be any system designated and installed to reduce 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.

- **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 are usually airborne CONTINUOUS RELEASES. The term "Long Term" comes from the reference to utilizing the long term dispersion factor (X/Q) from Table 5.1.

- **MEMBER OF THE PUBLIC**

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

- **OPERABLE - OPERABILITY**

  As defined in the Technical Specifications.

- **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 usually refers to airborne BATCH RELEASES. The term "Short Term" comes from the reference to utilizing the short term dispersion factor (X/Q) from Table 5.1.

- **SITE BOUNDARY**

  The SITE BOUNDARIES for liquid and gaseous releases are defined in Figures 3.1 and 3.2.

- **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.

- **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 Part 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.

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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. Specified time intervals between tests may be adjusted plus or minus 25% to accommodate normal test schedules.

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 OPERABILITY requirements for a Control for operation. 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 inoperable equipment.

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2.0 LIQUID EFFLUENTS

CONCENTRATION

SPECIFICATIONS

2.1 In accordance with T.S. 5.5.4.b the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS **SHALL** conform 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 2×10^{-4} μCi/ml total activity.

APPLICABILITY

At all times.

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.

2.2 SURVEILLANCE REQUIREMENTS

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.

DOSE

SPECIFICATIONS

2.3 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 ≤3 mrem to the total body and to ≤10 mrem to any organ, and

b. During any calendar year to ≤6 mrem to the total body and to ≤20 mrem to any organ.

APPLICABILITY

At all times.

ACTION

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, 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.

SURVEILLANCE REQUIREMENTS

2.4 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 4.0 of the ODCM.

LIQUID RADWASTE TREATMENT SYSTEMS

SPECIFICATIONS

2.5 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.

APPLICABILITY

At all times.

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 inoperable equipment or subsystems, and the reason for the inoperability.
2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and
3. Summary description of action(s) taken to prevent recurrence.

2.6 SURVEILLANCE REQUIREMENTS

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 4.0 of the ODCM.

2.6.2 The installed LIQUID RADWASTE TREATMENT SYSTEM **SHALL** be considered OPERABLE by meeting the Controls specified in 2.1 and 2.3.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

SPECIFICATIONS

2.7 In accordance with T.S. 5.5.4.a the radioactive liquid effluent monitoring instrumentation channels shown in Table 2.2 **SHALL** be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 2.1 are not exceeded. The alarm/trip setpoints of these channels **SHALL** be determined in accordance with the methodology in Section 4.0 of the ODCM.

APPLICABILITY

During release via the monitored pathway.

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 inoperable, or change the setpoint so it is acceptably conservative.

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

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

SURVEILLANCE REQUIREMENTS

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

LIQUID STORAGE TANKS

SPECIFICATIONS

2.9 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:

Condensate Storage Tanks
Outside Temporary Storage Tanks

APPLICABILITY

At all times.

ACTION

a. With the quantity of radioactive material contained in any of the above listed tanks exceeding the limit in 2.9 above, immediately suspend all additions of radioactive materials to the tank and within 48 hours reduce the contents to within the limit.

SURVEILLANCE REQUIREMENTS

2.10 The quantity of radioactive material contained in each of the tanks listed in specification 2.9 **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.

LANDLOCKED AREA

SPECIFICATIONS

2.11 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 analysis is analyzed to the LLDs listed in Table 7.3 for sediment.

APPLICABILITY

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.

The landlocked area is located near the southwest corner of the Prairie Island reactor building proper. It is designed to distribute water, discharged from the turbine building sumps, over a large area as the water enters the soil. The landlocked area is fully contained within an area controlled by NSP. When high levels of suspended solids are expected or detected in the turbine building sumps, the sump discharge is directed to the landlocked area. Sump sampling is conducted per Tables 2.1 and 2.2 to verify that the radioactivity concentration limits in specification 2.1 are not exceeded.

ACTION

a. With the quantity of radioactive material contained in the soil exceeding the limit in 2.11 above, describe the landlocked location in the 10CFR50.75.g file, conduct a dose assessment, and remediate, as required by applicable regulation.

SURVEILLANCE REQUIREMENTS

2.12 The presence of licensed radioactive material described in specification 2.11 **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

DOSE RATE

SPECIFICATIONS

3.1 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: ≤500 mrem/yr to the whole body and ≤3000 mrem/yr to the skin, and

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

APPLICABILITY

At all times.

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.

3.2 SURVEILLANCE REQUIREMENTS

3.2.1 The dose rate due to noble gases in effluents **SHALL** be determined to be within the above limits in accordance with the methodology and parameters in Section 5.0 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 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.

DOSE - NOBLE GASES

SPECIFICATIONS

3.3 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: ≤10 mrad for gamma radiation and ≤20 mrad for beta radiation, and

b. During any calendar year: ≤20 mrad for gamma radiation and ≤40 mrad for beta radiation.

APPLICABILITY

At all times.

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.

SURVEILLANCE REQUIREMENTS

3.4 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 5.0 of the ODCM.

DOSE - IODINE-131, IODINE-133, TRITIUM AND PARTICULATES

SPECIFICATIONS

3.5 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: ≤15 mrem to any organ, and

b. During any calendar year: ≤30 mrem to any organ.

APPLICABILITY

At all times.

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.

SURVEILLANCE REQUIREMENTS

3.6 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.

GASEOUS RADWASTE TREATMENT SYSTEMS

3.7 SPECIFICATIONS

3.7.1 In accordance with T.S.5.5.4.f the Waste Gas 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 ≤ 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.

APPLICABILITY

At all times.

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 inoperable equipment or subsystems, and the reason for the inoperability.
2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and
3. Summary description of action(s) taken to prevent 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

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 additional 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 OPERABLE by meeting the Controls specified in 3.1, 3.3 AND 3.5.

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.

EXPLOSIVE GAS MONITORING INSTRUMENTATION

3.9 SPECIFICATIONS

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 OPERABLE 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 ≤2% by volume.

APPLICABILITY

As shown in Table 3.2.

ACTION

a. With an explosive gas monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, declare the channel inoperable and take the ACTION shown in Table 3.2.

b. With less than the minimum required explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.2. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, in lieu of a License Event Report, prepare and submit a Special Report to the Commission to explain why this inoperability was not corrected in a timely manner.

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.

SURVEILLANCE REQUIREMENTS

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

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

SPECIFICATIONS

3.11 In accordance with T.S.5.5.4.a the radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.2 **SHALL** be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.1 are not exceeded. The alarm/trip setpoints of these channels **SHALL** be determined in accordance with the methodology in Section 5.0 of the ODCM.

APPLICABILITY

As shown in Table 3.2.

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 inoperable, or change the setpoint so it is acceptably conservative.

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

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

SURVEILLANCE REQUIREMENTS

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

ATMOSPHERIC STEAM DUMP MONITORING

SPECIFICATIONS

3.13 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.

APPLICABILITY

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

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

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.

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.

Since Fe-55, Sr-89, Sr-90, and alpha concentrations are determined from composite samples, the liquid monitor setpoint determinations should be completed using the most recent available composite sample results.

Monitor high alarm or isolation setpoints will be established by one of the following:

a. Calculation of setpoints using the methodology of Sections 4.1.1 and 4.1.3 at least every 31 days.

b. Calculation of alarm setpoint based on analysis prior to discharge using methodology of Section 4.1.2.

c. Alarm setpoint determined using methodology of Section 4.1.1 and 4.1.3 assuming all radionuclides have a concentration of 1E-7 μCi/ml. No recalculation of setpoints is necessary unless an increase in alarm setpoint is desired.

PWR GALE Code source terms (Table 4.1) may be used if there were no detectable isotopes in the previous month or in the analysis prior to release. If the newly 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 increase to the new value.

4.1.1 Liquid Effluent Monitor Setpoints

The following method applies when determining the isolation setpoints for the Waste Effluent Liquid Monitor (R-18), Steam Generator Blowdown Liquid Monitor - Unit 1 (1R-19), and Steam Generator Blowdown Liquid Monitor - Unit 2 (2R-19) during all operational conditions when the radwaste discharge flow rate is maintained constant at the maximum design flow rate.

A. Determine the "mix" (radionuclides and composition) of the liquid effluent.

1. Determine the liquid source terms that are representative of the "mix" of the liquid effluent. Liquid source terms are the total curies of each isotope released during the previous month. Table 4.1 source terms may be used if there have been no liquid releases.

2. Determine the activity concentrations (AC_i) of all non-gamma emitters including H-3, Sr-89, Sr-90, Fe-55, and alpha activity.

3. Determine NGF (the total fraction of the MPC in the liquid effluent) for all non-gamma emitting nuclides.

$$NGF = \sum_i \frac{AC_i}{MPC_i} \qquad (4.1\text{-}1)$$

where: AC_i = Activity concentration of nuclide 'i' in the liquid effluent (μCi/ml).

MPC_i = Ten times the water effluent concentration limit for radionuclide "i" (μCi/ml) from 10CFR20 Appendix B, Table 2, Column 2.

4. Determine Si (the fraction of the gamma emitting radioactivity in the liquid effluent comprised by radionuclides 'i') for each individual radionuclide in the liquid effluent.

$$S_i = \frac{A_i}{\sum_i A_i} \qquad (4.1\text{-}2)$$

where: A_i = the radioactivity of gamma emitting radionuclide 'i' in the liquid effluent.

5. Determine WGF (the sum of fractional activities weighted by the MPC) for the gamma emitting nuclides in the liquid effluent.

$$WFG = \sum_i \frac{S_i}{MPC_i} \qquad (4.1\text{-}3)$$

where: MPC_i = Ten times the water effluent concentration limit for radionuclide "i" (μCi/ml) from 10CFR20 Appendix B, Table 2, Column 2.

B. Determine C_t (the maximum acceptable total radioactivity concentration of gamma emitting nuclides in the liquid effluent prior to dilution (μCi/ml).

$$C_t = \frac{1}{WGF} X \left(\frac{F}{f} - NGF \right) \qquad (4.1\text{-}4)$$

where: F = Dilution water flow rate (gpm)

= 67,300 gpm from cooling tower blowdown

f = The maximum attainable discharge flow rate prior to dilution (gpm)

= 60 gpm from the ADT tank pump

= 100 gpm from the CVCS tank pump

= 60 gpm from the SGBD tank pump

C. Determine C.R. (the calculated monitor count rate above background attributed to the radionuclides (ncpm)).

C.R. is obtained by using the applicable Effluent Monitor Efficiency Curve located in the Radiation Monitor Calibration file. C.R. is the count rate that corresponds to the "adjusted" total radioactivity concentration (C_t).

D. Determine HSP (the monitor high alarm setpoint above background (ncpm)).

$$HSP = T_m C.R. \qquad (4.1\text{-}5)$$

T_m = 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 several release points.

= 0.75 for the Waste Effluent Liquid Monitor (R-18)

= 0.25 for the Steam Generator Blowdown Liquid Monitor - Unit 1 (1R-19)

= 0.25 for the Steam Generator Blowdown Liquid Monitor - Unit 2 (2R-19)

T_m values may be revised from the values given above. The summation of all the T_m values for active release points **SHALL NOT** be greater than unity.

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

4.1.2 Setpoint Based on Analysis of Liquid Prior to Discharge (Optional)

This method may be used in lieu of the method in Section 4.1.1 to determine the setpoints for the maximum acceptable discharge flow rate prior to dilution and to determine the associated high alarm setpoint based on this flow rate for the Waste Effluent Liquid Monitor (R-18), Steam Generator Blowdown Liquid Monitor - Unit 1 (1R-19), and Steam Generator Blowdown Liquid Monitor - Unit 2 (2R-19), during all operational conditions.

A. Determine f (the maximum acceptable discharge flow rate prior to dilution (gpm)).

$$f = \frac{0.8\,FT_m}{\sum_i \frac{C_i}{MPC_i}} \qquad (4.1\text{-}6)$$

F = Dilution water flow rate (gpm)

= 67,300 gpm from cooling tower blowdown

C_i = Concentration of radionuclide "i" in the liquid effluent prior to dilution (μCi/ml) from analysis of the liquid effluent to be released.

MPC_i = Ten times the water effluent concentration limit for radionuclide "i" (μCi/ml) from 10CFR20, Appendix B, Table 2, Column 2.

T_m = 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 several release points. Refer to Section 4.1.1.D.

B. Determine the monitor setpoint based on the radionuclide mix of the liquid effluent.

1. Determine C.R. (the calculated monitor count rate above background attributed to the radionuclides (ncpm)).

   C.R. is obtained by using the applicable Effluent Monitor Efficiency Curve located in the Radiation Monitor Calibration file. C.R. is the count rate point that corresponds to the "adjusted" total radioactivity concentration (C_t).

   C_t=Total radioactivity concentration of the radionuclides (minus tritium and other radionuclides that are only beta emitters) in the liquid discharge prior to dilution (μCi/ml) as determined using Equation 4.1-4.

2. Determine HSP (the monitor high alarm setpoint above background (ncpm)).

   $$HSP = \frac{C.R.}{0.8} \qquad (4.1\text{-}7)$$

   0.8 = A correction factor to increase the monitor setpoint to prevent spurious alarms caused by deviations in the mixture of radionuclides that affects monitor response.

3. The monitor high alarm setpoint above background **SHALL** be set at or below this HSP value when this optional method is selected. The maximum discharge flow **SHALL NOT** exceed the value of f as determined in Section 4.1.2.A. when this optional method is selected.

4.1.3 Discharge Canal Monitor

The following method determines the high alarm setpoint for the Discharge Canal Monitor (R-21) during all operational conditions.

A. Determine the "mix" (radionuclides and composition) of the liquid effluent.

1. Determine the liquid source terms that are representative of the "mix" of all liquids released into the discharge canal. Liquid source terms are the total curies of each isotope released during the previous month. Table 4.1 source terms may be used if there have been no liquid releases.

2. Determine the activity concentrations (AC_i) of all non-gamma emitters including H-3, Sr-89, Sr-90, Fe-55, and alpha activity.

3. Determine NGF (the total fraction of the MPC in the liquid released to the discharge canal) for all non-gamma emitting nuclides. The volume used to calculate the non-gamma emitting activity concentrations is the volume released via cooling tower blowdown during a one month period at the minimum flow rate of 67,300 gpm.

$$NGF = \sum_i \frac{AC_i}{MPC_i}$$

where: AC_i = Activity concentration of nuclide 'i' released to the discharge canal (μCi/ml)

MPC_i = Ten times the water effluent concentration limit for radionuclide "i" (μCi/ml) from 10CFR20, Appendix B, Table 2, Column 2.

4. Determine Si (the fraction of the gamma emitting radioactivity in the liquid released to the discharge canal comprised by radionuclide 'i') for each individual radionuclide released to the discharge canal.

$$S_i = \frac{A_i}{\sum_i A_i}$$

where: A_i=The radioactivity of gamma emitting radionuclide "i" released to the discharge canal.

5. Determine WGF (the sum of fractional activities weighted by the MPC) for the gamma emitting nuclides released to the discharge canal.

$$WGF = \sum_i \frac{S_i}{MPC_i}$$

where: MPC_i = Ten times the water effluent concentration limit for radionuclide "i" (μCi/ml) from 10CFR-20. Appendix B, Table 2, Column 2.

B. Determine C_t (the maximum acceptable total radioactivity concentration of gamma emitting nuclides released to the discharge canal (μCi/ml).

$$C_t = \frac{1-NGF}{WGF}$$

C. Determine C.R. (the calculated monitor count rate above background attributed to the radionuclides (ncpm)).

C.R. is obtained by using the applicable Effluent Monitor Efficiency Curve located in the Radiation Monitor Calibration file. C.R. is the count rate that corresponds to the "adjusted" total radioactivity concentration (C_t).

D. The monitor high alarm setpoint above background (ncpm) **SHALL** be set at or below the C.R. value.

4.1.4 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.

Effluent release computer calculations that compute setpoint determinations or expected monitor readings during or prior to a release compensate for the difference in gamma energies and yields and adjust the monitor setpoint or predicted monitor reading according to the actual nuclide mix. The assumption is made that the monitor's response is directly proportional to the gamma energies.

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; plus the fact that the monitor fractions used in the setpoint determination equation determine that it would be necessary for all of the effluent monitors to be in alarm before the limits of ten times the water effluent concentrations of 10CFR Part 20, Appendix B, Table 2, Column 2 would be exceeded.

4.2 Compliance With 10CFR20

In order to comply with 10CFR20, 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 10CFR20. 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 radioactivity in effluents prior to dilution will be determined, providing protection in addition to the alarm trip setpoint discussed in Section 4.1. Concentration in diluted effluents will be calculated using these results.

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 10CFR Part 20, Appendix B, Table 2, Column 2 would be exceeded .

Other minor releases of a continuous nature have occurred at PINGP through the turbine building sump system. These releases were minor and are not expected to occur in the future. However, 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 detectable 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 4.2-2 will be lowered to account for this source term. Doses from radionuclides contained in turbine building sump effluent diverted to the landlocked area are adequately accounted for since the calculations conservatively assume all the activity is discharged to the river per Section 4.2.2.

4.2.2 Batch Releases

To further show compliance with 10CFR20, 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. This calculation is shown in the following equation:

$$Conc_i = \frac{C_i R}{MDF} \qquad (4.2\text{-}1)$$

where

$Conc_i$ = concentration of radionuclide i at the site boundary, μCi/ml;

C_i = concentration of radionuclide i in the potential batch release, μCi/ml;

R = release rate of the batch

MDF = minimum dilution flow (=67,300 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 10CFR20. Before a release may occur, Equation 4.2-2 must be met for all isotopes.

$$\sum_i \frac{Conc_i}{MPC_i} \leq 0.9 \qquad (4.2\text{-}2)$$

MPC_i = Ten times the water effluent concentration of radionuclide i from 10CFR20, Appendix B, Table 2, Column 2, uCi/ml.

The summation has been reduced from 1.0 to 0.9 to account for simultaneous CONTINUOUS RELEASES from steam generator blowdown as given in Section 4.1.1.E. As noted earlier, this fraction may be adjusted based on experience. The summation of all source terms **SHALL NOT** be greater than 1.0.

Since the volume of the discharge pipe will 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 4.1.2, 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 10CFR50

Doses resulting from liquid effluents will be calculated at least every 31 days to show compliance with 10CFR50. 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 10CFR50 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, steam generator blowdown and turbine building sump releases, contribute a very small portion of the total liquid releases from PINGP. Therefore, for compliance with 10CFR50 the releases from both units will be summed and the limits of Appendix I will be doubled.

4.3.1 Determination of Liquid Effluent Dilution

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

$$F_k = \frac{R_k}{X\,ADF_k} \qquad (4.3\text{-}1)$$

where:

R_k = release rate of the batch or continuous release during the period, k, gpm.

ADF_k =average dilution flow during the time period of release k, gpm.

The value of X 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. The product of X and ADF_k is limited to 1000 cfs (4.5×10^5 gpm). Therefore, since blowdown flow in closed cycle is 150 cfs, the denominator of Equation 4.3-1 is always 4.5×10^5 in closed cycle. In once through or helper mode, the value if X is reduced to 1.0.

4.3.2 Dose Calculations

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

where:

$$D_\tau = \sum_k \sum_i A_{i\tau} t_k C_{ik} F_k \qquad (4.3\text{-}2)$$

where:

D_τ = the dose commitment to the total body or any organ τ, from the liquid effluents for the period of release, mrem;

C_{ik} = the average concentration of radionuclide, i, in undiluted liquid effluent for liquid release k, μCi/ml;

$A_{i\tau}$ = 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;

F_k = the near field average dilution factor for C_{ik} during liquid effluent release k,

t_k = the duration of release k, hours.

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

$$A_{i\tau} = 1.14 \times 10^5 [21 BF_i \ DF_{i\tau}] \qquad (4.3\text{-}3)$$

where:

$$1.14 \times 10^5 = 10^6 \ \frac{pCi}{\mu Ci} \times 10^3 \ \frac{ml}{l} \times \frac{1\,yr}{8760\,hr};$$

21 = adult fish consumption, Kg/yr;

BF_i = bio accumulation factor for radionuclide i in fish from Table A-1 of Regulatory Guide 1.109 Rev. 1 ([5]) pCi/Kg per pCi/l;

$DF_{i\tau}$ = 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.

A table of $A_{i\tau}$ values for an adult at the PINGP are presented in Table 4.2. 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.

4.3.3 Cumulation 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 40CFR190.

4.3.4 Projection of Doses

Anticipated doses resulting from the release of liquid effluents will be projected monthly. If the projected doses for the month exceed 2 percent of the limit specified in Section 2.3.b, additional components of the liquid radwaste treatment system will be used to process waste. The projected doses will be calculated using Equation 4.3-2. The dilution factor, F_k, will be calculated by replacing the term ADF_k in Equation 4.3-1 with the term MDF from Equation 4.2-1. The total source term utilized for the most recent dose calculation should be used for the projections unless information exists indicating that actual releases could differ significantly in the next month. 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

1. "Prairie Island Final Environmental Statement," USAEC, May, 1973, p. V-26.
2. "Prairie Island Nuclear Generating Plant, Appendix I Analysis - Supplement No. 1 - Docket No. 50-282 and 50-306," Table 2.1-1.
3. "10CFR20," Appendix B, Table II, Column 2.
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.
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 10CFR50, Appendix I," Rev. 1, 1977.

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 alarm or isolation setpoints will be established in one of the following ways:

a. Calculation of setpoint every 31 days using the methodology of Section 5.1.1 for CONTINUOUS RELEASES using previous month releases as source term.

b. Prior to each containment PURGE, recalculation of the setpoint using the methodology of Section 5.1.1 based on the sample taken prior to PURGING.

c. In lieu of (5.1.a) and (5.1.b) above, alarm setpoints may be established using the methodology of Section 5.1.1 using conservative assumptions (e.g., 100% Kr-89). No recalculation of setpoints is necessary unless an increase is desired.

PWR GALE Code source terms (Table 5.2) may be used if there were no detectable isotopes in the previous month or in the analysis prior to PURGING. If the newly 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 new value.

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.

1. Determine the gaseous source terms that are representative of the gaseous effluent. Gaseous source terms are the total curies of each noble gas released during the previous month or a representative analysis of the gaseous effluent. Table 5.2 source terms may be used if the releases for the previous month were below the lower limits of detection (LLD).

2. Determine S_i (the fraction of the total radioactivity in the gaseous effluent comprised by noble gas radionuclide "i") for each individual noble gas radionuclide in the gaseous effluent.

$$S_i = \frac{A_i}{\Sigma_i A_i} \qquad (5.1\text{-}1)$$

A_i = The radioactivity of noble gas radionuclide "i" in the gaseous effluent from either the previous months releases or from Table 5.2 if there were no releases during the previous month.

B. Determine Q_t (the maximum acceptable total release rate of all noble gas radionuclides in the gaseous effluent (μCi/sec)) based upon the whole body exposure limit.

$$Q_t = \frac{500}{(\chi/Q)\ \Sigma_i\ K_i S_i} \qquad (5.1\text{-}2)$$

(χ/Q) = The 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 (sec/m^3) from the "χ/Q" column in Table 5.1.

K_i = The total whole body dose factor due to gamma emissions from noble gas radionuclide "i" (mrem/year/μCi/m^3) from Table 5.4.

C. Determine Q_t based upon the skin exposure limit.

$$Q_t = \frac{3000}{(\chi/Q)\Sigma_i\ (L_i + 1.1\,M_i)\ S_i} \qquad (5.1\text{-}3)$$

$L_i + 1.1\ M_i$ = The total skin dose factor due to gamma and beta emissions from noble gas radionuclide "i" (mrem/year/μCi/m^3) from Table 5.4.

D. Determine C_t (the maximum acceptable total radioactivity concentration of all noble gas radionuclides in the gaseous effluent (/μCi/cc)).

$$C_t = \frac{2.12\ E\text{-}3\ Q_t}{F} \qquad (5.1\text{-}4)$$

NOTE: **Use the lower of the Q_t values obtained in Section 5.1.1.B and 5.1.1.C.**

F = The maximum effluent flow rate at the point of release (cfm) from the "Effluent Flow Rate" column in Table 5.1.

2.12 E-3 = Unit conversion constant to convert μCi/sec/cfm to μCi/cc.

E. Determine C.R. (the calculated monitor count rate above background attributed to the noble gas radionuclides (ncpm)).

C.R. is obtained by using the applicable Effluent Monitor Efficiency Curve located in the Radiation Monitor Calibration file.

C.R. is the count rate point that corresponds to the total radioactivity concentration (C_t).

F. Determine HSP (the monitor high alarm setpoint above background (ncpm)).

$$HSP = T_m\ C.R. \qquad (5.1\text{-}5)$$

T_m = Fraction of the total radioactivity from the site that may be released via each release point to ensure that the SITE BOUNDARY limit is not exceeded due to simultaneous releases from several release points from the "Release Fraction" column in Table 5.1.

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

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 using available gas mixes existing in plant systems. Since the available 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 mix is predominantly Xe-133 with lower level beta and gamma energies and a second mix which contains a larger variety of 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. One curve to be used when the isotopic mix being monitored is primarily Xe-133 and the other curve is for use when the mix is unknown or is known to contain a mixture of other fission and activation gases.

Effluent release computer calculations that compute setpoint determinations or expected monitor readings during or prior to a release utilize the correct calibration curves and adjust the monitor setpoint or predicted monitor reading 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; plus the fact that the monitor fractions used in the setpoint determination equation determine that it would be 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 10CFR20

Dose rates resulting from the release of noble gases, and radioiodines and particulates must be calculated to show compliance with 10CFR20. The limits of 10CFR20 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:

a. 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.

b. 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.

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

d. 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

To comply with the 10CFR20 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 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 calculational method presented in NUREG-0133. They represent a backward solution to the limiting dose equations in NUREG-0133. Setting alarm set trip points in this manner will assure that the limits of 10CFR20 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 10CFR50, Appendix I as discussed in Section 5.3.1.

5.2.2 Radioiodine, Radioactive Particulates, and Other Radionuclides

For compliance with 10CFR20, the dose rate at the SITE BOUNDARY resulting from the release of radioiodines 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 and weekly for all releases. To show compliance, Equations 5.2-1 will be evaluated for I-131, I-133, tritium, and radioactive particulates with half-lives greater than eight days.

$$\Sigma P_{I_i} \left[\left(\chi/Q_V\right) Q_{iv}\right] < 1500 \text{ mrem/yr} \qquad (5.2\text{-}1)$$

where:

P_{I_i} = child critical organ dose parameter for radionuclide i for the inhalation pathway, mrem/yr per $\mu Ci/m^3$ (Table 5.3);

(χ/Q_v) = annual average relative concentration for LONG-TERM release at the critical location, sec/m^3 (Appendix A, Table A-3);

Q_{iv} = the total release rate of radionuclide i from all vents form both units for the batch or week of interest, $\mu Ci/sec$;

Radioiodines, tritium, and radioactive particulates will be released form up to six individual vents all within 300 feet of each other. For showing compliance with 10CFR20, calculations based on Equation 5.2-1 will be made once per week. 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.

Significant short-term BATCH RELEASES of long-lived radioactive particulates and tritium will result from containment PURGES. Calculations will be made for these releases separately to further assure compliance with 10CFR Part 20 prior to release. These calculations will be used only to determine whether or not the PURGE release will be allowed to occur. Source terms will be determined from the results of isotopic analyses of samples from containment prior to release. Equation 5.2.1 will be used in conjunction with the following relationship to demonstrate that the BATCH RELEASE does not exceed the dose rate limit:

$$BL = 1500 - (D_v - D_p) \quad (5.2\text{-}2)$$

where:

BL = limiting dose rate for the batch, mrem/yr;

D_v = previous week's dose rate from all continuous and batch releases mrem/yr;

D_p = previous week's dose rate from all PURGE releases mrem/yr.

5.2.3 Critical Receptor Identification

Compliance with 10CFR20 radiation dose limits for individual MEMBERS OF THE PUBLIC will be demonstrated by identifying critical receptor locations based on 10CFR50 App I ALARA design objectives. Since the doses associated with 10CFR50 are more restrictive than the 10CFR20 limits, this method satisfies the 10CFR20 requirements.

5.3 Gaseous Effluents - Compliance with 10CFR50

Doses resulting from the release of noble gases, radioiodines and particulates must be calculated to show compliance with Appendix I of 10CFR50. The calculations will be performed at least every 31 days for all gaseous effluents.

The limits of 10CFR50 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 form PINGP. Therefore, for compliance with 10CFR50 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:

a. 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.

b. 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.

c. Shield building vent releases are monitored via noble gas monitor.

d. 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:

$$3.17 \times 10^{-8} \sum_i M_i \left[(\chi/Q)_v \, Q_{iv} + (\chi/q)_v q_{iv} \right]$$

< 10 mrad for any calendar quarter
< 20 mrad for any calendar year (5.3-1)

For beta radiation:

$$3.17 \times 10^{-8} \sum_i N_i \left[(\chi/Q)_v \, Q_{iv} + (\chi/q)_v q_{iv} \right]$$

< 20 mrad for any calendar quarter
< 40 mrad for any calendar year (5.3-2)

where:

M_i = The air dose factor due to gamma emission for each identified noble gas radionuclide i, mrad/yr per $\mu Ci/m^3$; (Table 5.4)

N_i = The air dose factor due to beta emissions for each identified noble gas radionuclide i, mrad/yr per $\mu Ci/m^3$; (Table 5.4)

$(\chi/Q)_v$ = the annual average relative concentration for areas at or beyond the restricted area boundary for LONG-TERM vent releases (greater than 500 hr/year), sec/m^3 (Appendix A, Table A-4);

$(\chi/q)_v$ = The relative concentration for areas at or beyond the restricted area boundary for SHORT-TERM vent releases (equal to or less than 500 hrs/year), sec/m^3 (Appendix A, Table A-7);

q_{iv} = The total release of noble gas radionuclide i in gaseous effluents for SHORT-TERM vent releases from both units (equal to or less than 500 hrs/year), μCi;

Q_{iv} = the total release of noble gas radionuclide i in gaseous effluents for LONG-TERM vent releases from both units (greater than 500 hrs/yr), μCi;

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

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

LONG-TERM χ/Q's were given in Appendix A. SHORT-TERM χ/q's were calculated using the USNRC computer code "XOQDOQ" assuming 100 hours per year SHORT TERM RELEASES and are given in Appendix A (Table A-7). Values of M and N are taken directly from Reg Guide 1.109 and are given in Table 5.4.

B. Cumulation 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 40CFR190 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:

During any calendar quarter or year -

$$3.17 \times 10^{-8} \Sigma_j \Sigma_i R_{ijak} [W_v Q_{iv} + w_v q_{iv}] \quad (5.3\text{-}3)$$

< 15 mrem (per quarter)
< 30 mrem (per calendar year)

where:

Q_{iv} = release of radionuclide i for LONG-TERM vent releases from both units (greater than 500 hrs/yr), μCi;

q_{iv} = release of radionuclide i for SHORT-TERM purge releases from both units (equal to or less than 500 hrs/yr); μCi;

W_v = the dispersion parameter for estimating the dose to an individual at the controlling location for LONG-TERM vent releases (greater than 500 hrs/yr);

w_v = the dispersion parameter for estimating the dose to an individual at the controlling location for SHORT-TERM vent releases (equal to or less than 500 hrs/yr);

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

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.

The above equation 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 given in Tables 5.5-1 through 5.5-19. Dose factors for isotopes not listed will be determined in accordance with the methodology in Appendix C. Equations 5.3-3 will be applied to a controlling location which will have one or more of the following: residence, vegetable garden and milk animal. The selection of the actual receptor is discussed in Section 5.3.4. The source terms and dispersion parameters in Equation 5.3-3 are obtained in the same manner as in Section 5.2. The W values are in terms of $\chi/Q(sec/m^3)$ for the inhalation pathways and for tritium (Tables A-4 and A-7) and in terms of $D/Q(1/m^2)$ for all other pathways (Tables A-5 and A-8).

B. Cumulation 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. 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 40CFR190 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 equation 5.3-1 or 5.3-2, 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

For compliance with 10CFR50 App I ALARA design objectives, two critical receptor locations will be identified to demonstrate compliance with limits on dose to air or individual MEMBERS OF THE PUBLIC in unrestricted areas from plant effluents.

For noble gases the critical location will be based on the beta and gamma air doses only. This location will be the offsite location with the highest long term vent χ/Q values given in Appendix A, Table A-3. This location will remain the same unless meteorological data is reevaluated or the SITE BOUNDARY changes.

The critical location for the I-131, I-133, tritium, and long-lived particulate pathway will be selected once each year. 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.

5.4 References

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

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6.0 TOTAL DOSE FROM RADIOACTIVE RELEASES AND URANIUM FUEL SOURCES

SPECIFICATIONS

6.1 In accordance with T.S.5.5.4.j 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.

APPLICABILITY At all times.

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) to determine whether the above limits have been exceeded. 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 10CFR20.2203(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 Part 190 has not already been corrected, the special report **SHALL** include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

SURVEILLANCE REQUIREMENTS

6.2 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.

6.3 Cumulative dose contributions from direct radiation from the reactor units **SHALL** be determined. This application is applicable only under conditions set forth in ACTION (a) of Specification 6.1 above.

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

a. 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.

b. 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.

c. 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, 3.3, or 3.5.

[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[2] to a MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Specifications 2.3, 3.3, or 3.5. 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.

d. 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.

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

LAND USE CENSUS

SPECIFICATIONS

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

a. 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.

b. 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

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

b. 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.

c. 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 at least once per 12 months between the dates of May 1 and October 31 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

a. 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

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 10CFR50.36A and **SHALL** include:

a. 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.

b. 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.

c. 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.

d. 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).

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

f. If the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeds twice the limits of 10 CFR 50, Appendix I, 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 40CFR190, Environmental Radiation Protection Standards for Nuclear Power Operation.

g. 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 exceeded.

h. 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 operable as required in Tables 2.2 and 3.2.

i. 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.

j. 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.

k. 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.

l. 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.

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

8.2 Annual Radiological Environmental Monitoring Report

In accordance with T.S.5.6.2 the Annual Radiological Environmental Monitoring Report covering the operation of the offsite monitoring program **SHALL** include:

a. 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.

b. 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.

c. 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.

d. 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.

e. 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.

f. 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.

g. 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.

h. 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

i. 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.

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 Record Retention

a. Records retained for Five Years:

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

2. Records of wind speed and direction.

b. The following records **SHALL** be retained for the Life of the Plant:

1. Liquid and airborne radioactive releases to the environment.

2. Off-site environmental monitoring surveys.

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

BASES

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 10CFR20, 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, 10CFR Part 50, and (2) ten times the limits of 10CFR20. 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, 10CFR Part 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 40CFR141 requirements.

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 10CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10CFR Part 50 and the design objective given in Section II.D of Appendix I to 10CFR Part 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, 10CFR Part 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 10CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10CFR Part 50.

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 10CFR Part 20, Appendix B, Table 2, Column 2, in an UNRESTRICTED AREA.

3.0 GASEOUS EFFLUENTS

3.1/3.2 DOSE RATE

This control is provided to ensure that the dose rate at any time at the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10CFR Part 20 for UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10CFR 20, Appendix B, Table 2, Column 1. These limits provide reasonable assurance that the radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an UNRESTRICTED AREA to annual average concentrations exceeding limits specified in Appendix B, Table 2 of 10CFR Part 20. For individuals 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, 10CFR Part 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, 10CFR Part 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 10CFR 50.36a, General Design Criterion 60 of Appendix A to 10CFR Part 50, and the design objective given in Section II.D of Appendix I to 10CFR Part 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, 10CFR Part 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 10CFR Part 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 10CFR Part 50.

The waste gas treatment system is a pressurized system with two potential sources of oxygen: 1) oxygen added for recombiner operation, and 2) placing tanks vented for maintenance back on the system. 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 operable, 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 10CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10CFR Part 50.

6.0 TOTAL DOSE

This control is provided to meet the dose limitations of 10CFR Part 190 that have been incorporated into 10CFR 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 40CFR Part 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 40CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40CFR Part 190 have not already been corrected), in accordance with the provisions of 40CFR 190.11 & 10CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10CFR Part 20, as addressed in 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.

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 40CFR Part 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 10CFR Part 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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