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Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

MAY 1 4 2007

L-PI-07-033
TS 5.5.1.c
TS 5.6.3

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282, 50-306 and 72-10
License Nos. DPR-42, DPR-60 and SNM-2506

2006 Annual Radioactive Effluent Report and Offsite Dose Calculation Manual

Pursuant to the applicable Prairie Island Nuclear Generating Plant (PINGP) Technical Specifications (TS), Appendix A to Operating Licenses DPR-42 and DPR-60, and the requirements of the Offsite Dose Calculation Manual (ODCM), Nuclear Management Company, LLC (NMC) submits the 2006 Annual Radioactive Effluent Report which is comprised of the following reports:

Enclosure 1 contains the Off-Site Radiation Dose Assessment for the period January 1, 2006 through December 31, 2006 in accordance with the requirements of the ODCM.

Enclosure 2 contains the Annual Radioactive Effluent Report, Supplemental Information, for the period January 1, 2006 through December 31, 2006 in accordance with the requirements of TS 5.6.3 and the ODCM.

Enclosure 3 contains the Effluent and Waste Disposal Annual Report, Solid Waste and Irradiated Fuel Shipments, for the period January 1, 2006 through December 31, 2006 in accordance with the requirements of TS 5.6.3 and the ODCM.

Enclosure 4 is an attachment to the 2006 Annual Effluent Report which contains an Amended Liquid Pathway Dose Calculation for the third quarter of 2006. In accordance with industry guidelines on groundwater monitoring, the report includes the dose calculation and dose report for a secondary steam condensate leakage which occurred in August 2006.

IE48
A009
NMSS01

Enclosure 5 contains a complete copy of the entire ODCM, Revision 20, dated 10/20/06. In accordance with the requirements of TS 5.5.1.c., the changes are identified by markings in the margin of the affected pages. The manual also contains a Record of Revisions which includes a summary of the revision changes (refer to page 8 of the ODCM).

Summary of Commitments

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

Michael D. Wadley

Michael D. Wadley
Site Vice President, Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosures (4)

cc: Regional Administrator, USNRC, Region III
Project Manager, Prairie Island Nuclear Generating Plant, USNRC, NRR
NRC Resident Inspector – Prairie Island Nuclear Generating Plant
Tim Donakowski, State of Minnesota

ENCLOSURE 1

OFF-SITE RADIATION DOSE ASSESSMENT

January 01, 2006 – December 31, 2006

8 pages follow

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

January through December 2006

An Assessment of the radiation dose due to releases from Prairie Island Nuclear Generating Plant during 2006 was performed in accordance with the Offsite Dose Calculation 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 2006.

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. Gaseous release 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. Liquid release doses, both whole body and organ, are a small percentage of Appendix I Guidelines.

Doses to Individuals Due to Activities Inside the Site Boundary

Occasionally sportsmen enter the Prairie Island site 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 are reported in Table 1.

ABNORMAL RELEASES

There were a total of ten (10) abnormal releases for 2006. Of these, six (6) abnormal releases were due to containment openings as a result of the U1 Refueling Outage. These abnormal releases were combined into one abnormal release covering the U1 Refueling Outage. The resulting 2006 abnormal releases are summarized below:

1. Loss of Pressure in 129 Waste Gas Decay Tank (WGDT)

Operations noted an unexpected drop in 129 Waste Gas Decay Tank (WGDT) pressure. An evaluation of the waste gas system indicated that 129 WGDT pressure dropped from 79.0 psig to 37.0 psig between 2/8/2006 12:41 and 2/8/2006 18:05. During this time period 122/123 WGDTs were released per C21.3-10.10.

Cause: Leakage was identified at WG-3-13, the outlet of 129 WGDT to exhaust ducts. This valve is verified closed per C21.3-10.10, Waste Gas Decay Tank release procedure. WG-3-13 is a diaphragm valve.

The incident was due to a ruptured diaphragm or a misadjusted stem travel nut. Valve maintenance and preventive maintenance for diaphragm valves is the cause.

Corrective Action: 129 WGDT was sampled for nuclide mix.

Activity released based on sample taken from 129 GDT post release:

Nuclide	uCi/ml	uCi Released	Gamma Dose (mrad)	Beta Dose (mrad)
Kr-85	2.28E-03	8.66E+04	7.81E-08	8.86E-06
Xe-131M	3.85E-04	1.46E+04	1.20E-05	8.52E-05
Xe-133	3.08E-02	1.17E+06	1.79E-03	5.32E-03
Xe-133M	3.49E-04	1.33E+04	2.29E-05	1.03E-04
Xe-135	4.33E-05	1.65E+03	1.67E-05	2.13E-05
TOTAL			1.84E-03	5.54E-03

H3 9.57E-06 uCi/ml 3.64E+02 uCi 2.16E-06 mrem

CE 01014047-03, was written to "Evaluate the diaphragm valve PM program". This action is assigned to the system engineer for resolution.

Created release file RAB00010 to account for release.

Result: The dose from the activity released represented a small percentage of the total dose and was a very small percentage of limits. The dose did not impose upon the health and safety of the public.

The event was captured in the site's Action Request Process, CAP- 01014047.

The event was reported to the NRC Region 3 Radiation Protection (RP) Inspector, at the time of the event.

2. Release from Containment Openings during Unit 1 Refueling Outage

Unit One experienced a Fuel Element Defect during cycle 1R24. During refueling outage 1R24 (May 2006) routine air samples, taken at the containment openings (Equipment Hatch, airlocks), detected activity. It was determined that a release had occurred.

Procedural guidance was incomplete for the quantification and reporting of these releases, therefore these releases were determined to be abnormal and were documented in this report as such.

Cause: Typically, air samples taken at containment openings are negative. Due to the fuel element defect, an increased potential for elevated airborne levels in containment and increase probability of release at containment openings existed. Additional assessment should have occurred pre-outage and additional controls implemented.

Insufficient control of the containment openings was identified. An airborne limit, at which containment would be isolated, should have been established, to support effluent release goals, Definitive control of the weather curtain at the Equipment Hatch was lacking.

Additionally, the differential pressure across the containment, created when the Equipment Hatch and the Fuel Receipt Rollup Doors are opened simultaneously, was not sufficiently assessed or controlled. This led to airborne radioactivity escaping the Maintenance Airlock and being entrained in other ventilation pathways, not equipped with charcoal.

Even with sufficient controls the possibility of release exists. Guidance on quantifying and reporting was insufficient.

Corrective Actions: Each week the air samples taken at the various containment openings were assessed. Conservative assumptions were made with regards to flow and activity. Release files were created to quantify and report the activity released:

Nuclide	Activity Released (uCi)	Dose (mrem)
Co-58	3.36E+00	4.66E-06
Co-60	5.53E-01	2.01E-05
I-131	3.60E+02	2.33E-02
H3	1.78E+03	3.18-06
Total		2.33E-02

Nuclide	Activity Released (uCi)	Gamma Dose (mRad)	Beta Dose (mRad)
Xe-133	2.73E+06	1.03E-03	3.06E-03
Xe-135	1.26E+04	2.60E-05	3.32E-05
Total		1.06E-03	3.09E-03

The activity released was captured and reported in release files number RAC0102, RAC0110, RAC0132, RAC0133, RAC0145 and RAC0146.

Procedural guidance was generated to correct the deficiencies in monitoring and control. Training was conducted.

The procedure written also includes the conservative assumptions upon which such releases will be based on in the future.

Result: The dose from the activity released represented a small percentage of the total dose and was a very small percentage of limits. The dose did not impose upon the health and safety of the public.

The event was captured in the site's Action Request Process, CAP- 01027608 .

The event was reported to the NRC Region 3 Radiation Protection (RP) Inspector, at the time of the event.

3. Failed Primary Sample Valve Sprays Primary Coolant.

Various primary sample drag valves were removed, cleaned and reinstalled as part of pre-outage maintenance. When reinstalled, the Pressurizer Steam Space sample path was lined up as part of post maintenance testing. The cooler inlet, 2SM-5-4 sprayed steam. Elevated levels were noted on 2R30 and 2R37, Unit Two Auxiliary Building Stack monitors. The lineup was promptly secured.

Cause: The valve which failed was not one of the valves upon which maintenance was performed, however the Pressurizer Steam Space Sample path is typically only used during outage. This particular valve had not been operated in several months. Spray was determined to be from a packing leak.

Corrective Actions: Increase in radiation monitors readings for Unit Two Auxiliary Building Stack required integration and generation of an abnormal release report.

No samples were taken during this short duration release, therefore the nuclide mix was based on the Gaseous Source Term. Activity released and dose cosequences:

NUCLIDE	uCi	Beta (mRad)	Gamma (mRad)
Kr-85m	1.10E+03	1.45E-06	2.32E-06
Kr-85	6.90E+02	1.27E-08	1.44E-06
Kr-87	4.14E+02	2.74E-06	4.57E-06
Kr-88	1.79E+03	2.92E-05	5.62E-06
Xe-131m	6.90E+02	1.15E-07	8.21E-07
Xe-133m	1.79E+03	6.27E-07	2.84E-06
Xe-133	1.31E+05	4.95E-05	1.47E-04
Xe-135	2.76E+03	5.68E-06	7.27E-06
Xe-138	4.14E+02	4.09E-06	2.11E-06
TOTAL		9.34E-05	1.74E-04

The activity released was captured and reported as release file RAC0260.

Work Order #100264 was generated to implement repairs.

Result: The dose from the activity released represented a small percentage of the total dose and were a very small percentage of limits. The dose did not impose upon the health and safety of the public.

The event was captured in the site's Action Request Process, CAP-1055506.

The event was reported to the NRC Region 3 RP Inspector.

4. Unit Two Steam Generator Manway Removal Trips Containment Ventilation:

Unit Two experienced a Fuel Element Defect during cycle 2R24. During Unit Two refueling outage 2R24, in an effort to control containment airborne levels, HEPA hoses from Steam Generator Manways were directed to Containment Ventilation. Upon removal of the Steam Generator Inserts, Inservice Purge promptly tripped, due to increased radiation monitor count rate. Following the initial purge of the Steam Generator Bowls activity quickly decreased and purging of the Steam Generators directly to containment ventilation was recommenced without incident.

Cause: Unanticipated activity levels when initially venting Steam Generator Bowls.

Corrective Action: The duration of the release was determined to be the time that inserts were removed to the time that ventilation tripped. This was determined to be 5 minutes.

Radiation monitors were integrated to determine release activity. Total activity was ratioed to the nuclide mix previously identified:

Nuclide	Activity (uCi)	Gamma Dose (mRad)	Beta Dose (mRad)
Xe-131m	3.09E+04	3.68E-05	5.16E-06
Xe-133	1.78E+06	2.00E-03	6.73E-04
Xe-133m	1.61E+04	2.56E-05	5.64E-06
Total		2.06E-03	6.84E-04

Release file RAB0290 was created to document this release.

It was determined that additional constraints would be implemented to preclude this event, including continuous monitoring and throttling of Steam Generator Bowl purging until such time as the activity levels decreased to less than 50% of monitor set points.

Result: The dose from the activity released represented a small percentage of the total dose and was a very small percentage of limits. The dose did not impose upon the health and safety of the public.

The event was captured in the site's Action Request Process, CAP- 01062770.

The event was reported to the NRC Region 3 Radiation Protection (RP) Inspector, at the time of the event.

5. Leaking Steam Relief

On December 14, 2006, Following the Unit Two Refueling outage 2R24 it was noted that the down stream temperatures on CV-31104, STM GEN 2A STM DUMP TO ATM, were elevated indicating leak by and an on-going release.

Cause: CV-31104, STM GEN 2A STM DUMP TO ATM leaked by.

Corrective Action: Samples taken quantified no activity.

Release file RAB0088 was created to document this release.

Repairs we implemented to CV-31104.

Result: No activity was released and no there was no dose consequence to the general public. This release did not impose upon the health and safety of the public.

The event was captured in the site's Action Request Process, CAP-01068657.

The event was reported to the NRC Region 3 Radiation Protection (RP) Inspector, at the time of the event.

40CFR190 COMPLIANCE

The calculated dose from the release of radioactive materials in liquid or gaseous effluents **did not** exceed twice the limits of 10CFR50, Appendix I, therefore compliance with 40CFR190 **is not** required to be assessed, in this report.

SAMPLING, ANALYSIS AND LLD REQUIREMENTS

The minimum sampling frequency, minimum analysis frequency and lower limit of detection (LLD) requirements, as specified in ODCM Tables 2.1 and 3.1 **were not** exceeded in 2006.

MONITORING INSTRUMENTATION

There **were no** occurrences when less than the minimum required radioactive liquid and/or gaseous effluent monitoring instrumentation channels were operable as required by ODCM Tables 2.2 and 3.2.

Doses to Individuals Due to Effluent Releases from the Independent Spent Fuel Storage Facility (ISFSI)

Two (2) fuel casks were loaded and placed in the storage facility during the 2006 calendar year. The total number of casks in the ISFSI is twenty-two (22). There has been no release of radioactive effluents from the ISFSI.

CURRENT ODCM REVISION

The Offsite Dose Calculation Manual **was** revised in 2006. The current revision is 20. The revision date is October 20, 2006. A copy is submitted with this year's report.

PROCESS CONTROL PROGRAM

There **were no** changes made to the Process Control Program in 2006. Current manual is revision 8, August 25, 1999.

Table 1

OFF-SITE RADIATION DOSE ASSESSMENT - PRAIRIE ISLAND

PERIOD: JANUARY through DECEMBER 2006

		10 CFR Part 50 Appendix I Guidelines for a 2-unit site per year
Gaseous Releases		
Maximum Site Boundry Gamma Air Dose (mrad)	5.39E-02	20
Maximum Site Boundry Beta Air Dose (mrad)	1.62E-01	40
Maximum Off-site Dose to any organ (mrem)*	4.55E-01	30
Offshore Location		
Gamma Dose (mrad)	3.95E-03	
Total Body (mrem)*	9.66E-04	
Organ (mrad)*	9.54E-03	30
Liquid Releases		
Maximum Off-site Dose Total Body (mrem)	3.16E-03	6
Maximum Off-site Dose Organ - GI TRACT (mrem)	3.96E-03	20
Limiting Organ Dose Organ - TOTAL BODY (mrem)	3.16E-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 2006

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

ENCLOSURE 2

ANNUAL RADIOACTIVE EFFLUENT REPORT SUPPLEMENTAL INFORMATION

January 01, 2006 – December 31, 2006

9 pages follow

ANNUAL RADIOACTIVE EFFLUENT REPORT

01-JAN-06 THROUGH 31-DEC-06

SUPPLEMENTAL INFORMATION

Facility: Prairie Island Nuclear Generating Plant

Licensee: Northern States Power Company

License Numbers: DPR-42 & DPR-60

A. <u>Regulatory Limits</u>

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: ____20____

   Revision date : _10/20/06_

1.0 BATCH RELEASES (LIQUID)

	QTR: 01	QTR: 02	QTR: 03	QTR: 04
1.1 NUMBER OF BATCH RELEASES	2.60E+01	8.50E+01	3.50E+01	7.40E+01
1.2 TOTAL TIME PERIOD (HRS)	4.92E+01	1.59E+02	6.51E+01	1.30E+02
1.3 MAXIMUM TIME PERIOD (HRS)	2.40E+00	3.27E+00	2.32E+00	2.68E+00
1.4 AVERAGE TIME PERIOD (HRS)	1.89E+00	1.87E+00	1.86E+00	1.75E+00
1.5 MINIMUM TIME PERIOD (HRS)	1.60E+00	6.17E-01	1.43E+00	1.23E+00
1.6 AVERAGE MISSISSIPPI RIVER FLOW (CFS)	1.89E+04	3.79E+04	7.37E+03	7.06E+03

2.0 BATCH RELEASES (AIRBORNE)

	QTR: 01	QTR: 02	QTR: 03	QTR: 04
2.1 NUMBER OF BATCH RELEASES	1.00E+01	2.40E+01	0.00E+00	1.70E+01
2.2 TOTAL TIME PERIOD (HRS)	1.20E+02	3.56E+02	0.00E+00	1.92E+02
2.3 MAXIMUM TIME PERIOD (HRS)	2.40E+01	3.25E+01	0.00E+00	2.40E+01
2.4 AVERAGE TIME PERIOD (HRS)	1.20E+01	1.49E+01	0.00E+00	1.13E+01
2.5 MINIMUM TIME PERIOD (HRS)	2.83E-03	7.17E-03	0.00E+00	1.67E-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	1.00E+00	6.00E+00	0.00E+00	3.00E+00
4.2 TOTAL ACTIVITY RELEASED (CI)	1.29E+00	2.74E+00	0.00E+00	1.97E+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)	4.44E+00	3.83E+01	0.00E+00	9.56E+01
5.2 AVERAGE RELEASE RATE (UCI/SEC)	5.65E-01	4.87E+00	0.00E+00	1.22E+01
5.3 GAMMA DOSE (MRAD)	2.99E-03	1.48E-02	0.00E+00	3.62E-02
5.4 BETA DOSE (MRAD)	1.10E-02	4.34E-02	0.00E+00	1.08E-01
5.5 PERCENT OF GAMMA TECH SPEC (%)	2.99E-02	1.48E-01	0.00E+00	3.62E-01
5.6 PERCENT OF BETA TECH SPEC (%)	5.50E-02	2.17E-01	0.00E+00	5.39E-01
6.0 **IODINES**				
6.1 TOTAL I-131 (CI)	3.57E-06	6.27E-03	0.00E+00	3.68E-04
6.2 AVERAGE RELEASE RATE (UCI/SEC)	4.55E-07	7.98E-04	0.00E+00	4.68E-05
7.0 **PARTICULATES**				
7.1 TOTAL RELEASE (CI)	3.74E-06	2.93E-05	0.00E+00	1.36E-05
7.2 AVERAGE RELEASE RATE (UCI/SEC)	4.76E-07	3.73E-06	0.00E+00	1.73E-06
8.0 **TRITIUM**				
8.1 TOTAL RELEASE (CI)	3.03E+00	2.54E+00	2.11E+00	2.37E+00
8.2 AVERAGE RELEASE RATE (UCI/SEC)	3.86E-01	3.23E-01	2.68E-01	3.02E-01
9.0 TOTAL IODINE, PARTICULATE AND TRITIUM (UCI/SEC)	3.86E-01	3.24E-01	2.68E-01	3.02E-01
10.0 DOSE FROM IODINE, LLP, AND TRITIUM (MREM)	9.21E-03	4.11E-01	3.77E-03	3.11E-02
11.0 PERCENT OF TECH SPEC (%)	6.14E-02	2.74E+00	2.51E-02	2.07E-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
AR-41	CI					2.00E-04			
KR-85	CI				6.90E-04	2.65E-01			
KR-85M	CI				1.10E-03	1.41E-04			
KR-87	CI				4.14E-04				
KR-88	CI				1.79E-03				
XE-131M	CI				2.38E-01	1.46E-02			
XE-133	CI	2.93E+00	3.81E+01		9.52E+01	1.22E+00			
XE-133M	CI				2.04E-01	1.33E-02			
XE-135	CI		1.87E-01		7.45E-03	4.11E-03			
XE-138	CI				4.14E-04				
TOTALS	CI	2.93E+00	3.83E+01	0.00E+00	9.56E+01	1.51E+00	0.00E+00	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	3.57E-06	6.27E-03		3.68E-04		2.09E-07		6.02E-10
I-133	CI		1.55E-04		3.21E-05				
TOTALS	CI	3.57E-06	6.43E-03	0.00E+00	4.00E-04	0.00E+00	2.09E-07	0.00E+00	6.02E-10

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
BE-7	CI		4.11E-08						
CO-58	CI		3.67E-06		2.91E-07				1.86E-07
CO-60	CI		5.53E-07						
CS-137	CI		3.56E-08			3.74E-06			1.31E-05
RH-105	CI		2.50E-05						
TOTALS	CI	0.00E+00	2.93E-05	0.00E+00	2.91E-07	3.74E-06	0.00E+00	0.00E+00	1.33E-05

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)	3.85E+07	4.46E+07	1.88E+07	2.29E+07
17.0 VOLUME OF DILUTION WATER (LITERS)	1.62E+11	1.04E+11	2.66E+11	1.84E+11
18.0 FISSION AND ACTIVATION PRODUCTS				
18.1 TOTAL RELEASES W/O H-3, RADGAS, ALPHA (CI)	1.13E-02	2.85E-02	1.13E-02	3.23E-02
18.2 AVERAGE DILUTION CONCENTRATION (UCI/ML)	7.00E-11	2.73E-10	4.26E-11	1.76E-10
19.0 TRITIUM				
19.1 TOTAL RELEASE (CI)	3.40E+02	1.63E+02	1.29E+02	1.75E+02
19.2 AVERAGE DILUTION CONCENTRATION (UCI/ML)	2.11E-06	1.57E-06	4.87E-07	9.54E-07
20.0 DISSOLVED AND ENTRAINED GASES				
20.1 TOTAL RELEASE (CI)	1.35E-03	1.61E-02	1.75E-03	1.32E-02
20.2 AVERAGE DILUTION CONCENTRATION (UCI/ML)	8.34E-12	1.55E-10	6.59E-12	7.18E-11
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)	2.11E-06	1.57E-06	4.87E-07	9.55E-07
23.0 TOTAL BODY DOSE (MREM)	1.19E-03	7.16E-04	3.40E-04	9.10E-04
24.0 CRITICAL ORGAN				
24.1 DOSE (MREM)	1.19E-03	7.16E-04	3.40E-04	9.10E-04
24.2 ORGAN	TOT BODY	TOT BODY	TOT BODY	TOT BODY
25.0 PERCENT OF TECHNICAL SPECIFICATIONS LIMIT (%)	3.98E-02	2.39E-02	1.13E-02	3.03E-02
26.0 PERCENT OF CRITICAL ORGAN TECH SPEC LIMIT (%)	3.98E-02	2.39E-02	1.13E-02	3.03E-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-108M	CI						2.00E-05	9.67E-07	6.45E-06
AG-110M	CI					1.18E-03	2.90E-03	1.46E-03	5.81E-03
CO-57	CI					5.86E-05	6.58E-05	1.07E-05	2.77E-05
CO-58	CI					2.55E-03	5.81E-03	2.73E-03	8.75E-03
CO-60	CI					1.40E-03	8.07E-03	1.72E-03	4.39E-03
CR-51	CI					4.49E-04	2.07E-03	4.83E-04	1.95E-03
CS-134	CI	3.60E-05			2.46E-05		7.22E-06	1.19E-06	
CS-137	CI	6.28E-05			9.49E-05		6.55E-05	7.46E-06	3.24E-06
FE-55	CI					4.09E-03		2.90E-03	5.78E-03
FE-59	CI					2.13E-04	4.36E-04	1.31E-04	2.95E-04
I-131	CI				1.09E-06		5.04E-04		7.53E-05
LA-140	CI								5.89E-05
MN-54	CI					7.47E-05	4.10E-04	6.49E-05	1.78E-04
NB-95	CI					1.63E-04	2.43E-04	1.08E-04	1.72E-04
NB-97	CI					2.05E-06	7.88E-06	2.59E-06	8.28E-04
SB-124	CI					7.75E-05	1.33E-04	3.22E-06	2.63E-04
SB-125	CI					7.82E-04	2.66E-03	1.11E-03	2.70E-03
SN-113	CI					5.19E-05	1.01E-04	4.70E-05	1.01E-04
SR-85	CI					8.30E-07	1.24E-06		1.36E-06
SR-92	CI					1.37E-05	3.15E-05	1.62E-05	4.71E-05
TE-123M	CI						6.02E-05	2.37E-05	2.18E-04
TE-125M	CI						4.65E-03	4.32E-04	3.23E-04
TE-132	CI								3.22E-07
W-187	CI					6.20E-06	4.23E-05		5.52E-05
ZN-65	CI					1.56E-05	4.16E-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
ZR-95	CI					7.85E-05	1.25E-04	5.95E-05	1.76E-04
ZR-97	CI						3.56E-06	2.90E-06	1.91E-06
TOTALS	CI	9.89E-05	0.00E+00	0.00E+00	1.21E-04	1.12E-02	2.85E-02	1.13E-02	3.22E-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					1.91E-04	2.85E-04		3.14E-04
XE-131M	CI					5.24E-05	5.47E-05		1.14E-04
XE-133	CI	4.23E-05	2.52E-05		1.81E-05	1.06E-03	1.57E-02	1.75E-03	1.26E-02
XE-133M	CI						8.96E-06		7.61E-05
XE-135	CI								5.00E-05
TOTALS	CI	4.23E-05	2.52E-05	0.00E+00	1.81E-05	1.30E-03	1.61E-02	1.75E-03	1.32E-02

ENCLOSURE 3

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

January 01, 2006 – December 31, 2006

7 pages follow

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
NORTHERN STATES POWER

Period: 01/01/06-12/31/06
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	1.52E+01		179.4
	ft^3	5.38E+02		
	Ci	1.35E+02	2.50E+01	
B. Dry-Compacted	m^3			
	ft^3			
	Ci			
C. Non-Compacted	m^3	4.91E+02		1280
	ft^3	1.73E+04		260
	Ci	5.44E-01	2.50E+01	
D. Filter Media	m^3			
	ft^3			
	Ci			
S. Other (furnish description)	m^3	5.54E+01		1958
U1 Rx Head	ft^3	1.96E+03		
	Ci	1.39E+01	2.50E+01	

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: 01-01-06/12-31-06
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)
C	*Fe-55	6.43E+01
	Co-58	8.20E+00
	Co-60	8.20E+00
	*Ni-63	1.21E+01
	Zr-95	1.84E+00
	Nb-95	1.40E+00
	*C-14	1.00E+00
1% cutoff		
S	*Fe-55	3.50E+01
	Co-58	3.76E+01
	Co-60	5.56E+00
	Nb-95	2.39E+00
	Zr-95	1.35E+00
	Cr-51	2.67E+00
	*Ni-63	1.09E+01
	*C-14	3.07E+00
1% cutoff		

* = Inferred - Not Measured on Site

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
NORTHERN STATES POWER

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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)**
A	*H-3	1.52E+00
	*Fe-55	1.29E+01
	*Ni-63	4.61E+01
	Co-60	1.25E+01
	Cs-134	5.08E+00
	Cs-137	2.00E+01

1% cutoff

* = Inferred - Not Measured on Site

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
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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
6	Studsvik Logistics	RACE, LLC
3	Hittman	EnergySolutions, LLC (Bulk)
1	Perkins	EnergySolutions, LLC (Bulk)
3	Studsvik Logistics	Studsvik Processing Facility, LLC

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
NORTHERN STATES POWER

Period: 01-01-06/12-31-06
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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 (mCi)	Type of Waste	Container Code	Solidif. Code
	06-018	260/7.36	1.30E-01	C	L	N/A
	06-022	568.5/16.1	1.70E+01	C	L	N/A
	06-023	1958/55.44	1.39E+04	S	L	N/A
	06-021	1137/32.2	2.07E+01	C	L	N/A
	06-035	2560/72.5	5.01E+01	C	L	N/A
	06-037	2560/72.5	5.76E+01	C	L	N/A
	06-039	2560/72.5	3.36E+01	C	L	N/A
	06-040	2560/72.5	1.24E+01	C	L	N/A
	06-041	2560/72.5	1.64E+02	C	L	N/A
	06-042	2560/72.5	1.89E+02	C	L	N/A
	06-053	179.4/5.08	8.18E+03	A	L	N/A
	06-059	179.4/5.08	1.03E+05	A	A	N/A
	06-060	179.4/5.08	2.43E+04	A	L	N/A
TOTALS	13	19858/561.4	1.50E+05			

CONTAINER CODES:	L	=	LSA
(Shipment type)	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: 01-01-06/12-31-06
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: 01-01-06/12-31-06
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: 8/25/1999

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

ENCLOSURE 4

ATTACHMENT TO THE 2006 ANNUAL EFFLUENT REPORT
AMENDED LIQUID PATHWAY DOSE CALCULATION

Quarter 3, 2006

2 pages follow

ATTACHMENT TO THE 2006 ANNUAL EFFLUENT REPORT

Amended Liquid Pathway Dose Calculation
Quarter 3, 2006

Summary

The total liquid dose to the critical receptor for the third quarter of 2006 is 0.0198 mrem. This dose is reported in this attachment to the annual report.

Background

On August 4 and 5, 2006, approximately 168 gallons of secondary steam condensate leaked to the ground outside the northeast side of the turbine building. This water had a tritium concentration of 19,100 pCi/L. On August 5, 2006, the leak was contained and no further leakage to the environment occurred. It is assumed that the discharged water could potentially enter the groundwater and be incorporated in drinking water at the nearest resident who is also the critical receptor.

ODCM Considerations

The following calculation is independent of the ODCM. A revision to the ODCM to incorporate the Industry Initiative on Groundwater Protection is planned for 2007. Corrective actions have been taken to prevent a similar spill in the future.

Dose Calculation Assumptions

For the purpose of dose calculation, the dose-maximizing assumption was made that the receptor's concentration of tritium in body water and organic molecules is equal to the concentration of the released water diluted by a factor of 1000 (a dilution factor of approximately 1000 was calculated when tritium was discharged into the discharge canal versus sample results from a well 700 feet from the canal). (In this case, the critical receptor is 0.6 miles from the release point.) The tritium dose conversion factor is taken from page 9-3 of NUREG/CR-3332. Its value is 1.02E-4 mrem/year per pCi/liter of tritium in the body.

Discussion

The critical receptor is located 0.6 miles to the SSE of the Prairie Island site. The leaked water would have to travel in the groundwater under the recycle canal and discharge canal to reach the critical receptor. This assumed water flow maximizes the dose because the normal groundwater flow is towards the Vermillion River which would not carry the tritium toward the critical receptor.

The dose calculated for waterborne tritium is added to the critical receptor's fish pathway dose. This overestimates the dose because the tritium dose from eating fish is accounted for twice. It should be noted that the total airborne dose (99% plus of which is due to tritium) is greater than the total waterborne tritium dose. Even though the leaked tritium pathway delivers additional dose to the critical receptor, it is a lower concentration than that already in the body due to airborne exposure.

Dose Calculation

Quarter	Dose Conversion Factor (mrem/per pCi/L)	X	Diluted Tritium Concentration (pCi/L)	=	Whole Body Dose (mrem)	+	Fish Pathway Dose (mrem)	=	Total Liquid Dose (mrem)
3	1.04E-4		1.91E+2		1.95E-2		3.40E-4		1.98E-2

Dose Report

LIQUID EFFLUENTS - SUMMATION OF WATERBORNE TRITIUM AND FISH PATHWAYS

	QTR: 03
TOTAL BODY DOSE (MREM)	1.98E-02
CRITICAL ORGAN DOSE (MREM)	1.98E-02
ORGAN	TTL BODY
PERCENT OF TOTAL BODY TECH SPEC LIMIT (%)	6.50E-01
PERCENT OF CRITICAL ORGAN TECH SPEC LIMIT (%)	6.50E-01