March 7, 2001

Mr. J. Sorensen
Site General Manager
Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND - NRC INSPECTION REPORT 50-282/01-03(DRS); 50-306/01-03(DRS)

Dear Mr. Sorenson:

On February 15, 2001, the NRC completed a baseline inspection at your Prairie Island Plant. The enclosed report presents the results of that inspection which were discussed on February 15, 2001, with Mr. Werner and other members of your staff.

The inspection was an examination of activities conducted under your license as they relate to radiation safety and to compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel. Specifically, this inspection focused on occupational radiation safety, the radiological controls implemented for access to radiologically significant areas, and the ALARA (as-low-as-reasonably-achievable) program implementation for the Unit 1 refueling outage.

Based on the results of this inspection, one issue of very low safety significance (Green) was identified. This issue related to the failure of a radiation barrier during safety injection check valve work.

In accordance with 10 CFR Part 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

/RA by Steven Orth Acting For/

Gary L. Shear, Chief
Plant Support Branch
Division of Reactor Safety

Docket Nos. 50-282; 50-306
License Nos. DPR-42; DPR-60

Enclosure: Inspection Report No. 50-282/01-03(DRS);
50-306/01-03(DRS)

cc w/encl: Plant Manager, Prairie Island
M. Wadley, Chief Nuclear Officer
G. Eckholt, Site Licensing Manager
S. Northard, Nuclear Asset Manager
J. Malcolm, Commissioner, Minnesota
Department of Health
State Liaison Officer, State of Wisconsin
Tribal Council, Prairie Island Dakota Community
J. Silberg, Esquire
Shawn, Pittman, Potts, and Trowbridge
P. Tester, Assistant Attorney General
Minnesota Office of Attorney General
S. Bloom, Administrator
Goodhue County Courthouse
Commissioner, Minnesota Department
of Commerce

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-282 50-306
License No.	DPR 42 DPR 60
Report No:	50-282/01-03(DRS); 50-306/01-03(DRS)
Licensee:	Nuclear Management Company
Facility:	Prairie Island Nuclear Generating Plant
Location:	1717 Wakonade Drive East Welch, MN 55089
Dates:	January 29 to February 15, 2001
Inspector:	M. Mitchell, Radiation Specialist
Observer:	R. Schmitt, Radiation Specialist
Approved by:	Gary L. Shear, Chief Plant Support Branch Division of Reactor Safety

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

IR 05000282-01-03(DRS), IR 05000306-01-03(DRS), on 01/29-02/15/2001, Nuclear Management Company, LLC, Prairie Island Nuclear Generating Plant, Units 1 and 2. Refueling and outage occupational radiation safety.

The inspection was conducted by a regional radiation specialist. The inspection identified one Green finding. The significance of most/all findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "no color" or by the severity level of the applicable violation.

Cornerstone: Occupational Radiation Safety

Green: Radiation protection staff together with the NRC inspector identified that a radiation barrier failed when a radiation protection technician, providing job coverage, failed to immediately direct workers from an area of increasing radiation dose rates. The finding was of very low safety significance because the technician did direct the workers from the area after confirming the elevated dose rates and the delay did not cause significant unanticipated doses to the workers.

Report Details

Summary of Plant Status: Unit 1 was in a refueling outage, and Unit 2 was at 100 percent power throughout the inspection period.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

.1 Plant Walkdowns, Radiological Boundary Verifications, and Radiation Work Permit Reviews

a. Inspection Scope

The inspector conducted walkdowns of the radiologically restricted area (RRA) to verify the adequacy of radiological area boundaries and postings for high and locked high radiation areas in the Containment, Auxiliary, and Radwaste Buildings. Confirmatory radiation measurements were taken to verify that these areas and selected radiation areas were properly controlled in accordance with 10 CFR Part 20, licensee procedures, and Technical Specifications. Selected radiation work permits (RWPs) for tours and reactor containment activities were reviewed for protective clothing requirements, dosimetry requirements (including alarm set-points), and engineering controls. Walkdown of the Sump C posting and access control was conducted to assure that this Very High Radiation Area was appropriately controlled. The inspector also observed radiation workers performing the activities described in Section 2OS2.2, evaluated their awareness of radiological work conditions, and verified the implementation of radiological controls specified in applicable RWPs and ALARA (as-low-as-is-reasonably-achievable) plans. Additionally, the inspector reviewed the work planning, work performance, and dose assignment associated with a point source (particle) that became dislodged from the SR-9-2 Safety Injection Check Valve, during gasket repair and replacement, to assess licensee access controls and dose assessment.

b. Findings

One Green finding was identified. Specifically, a radiation barrier was compromised which could have resulted in a significant unintended or unplanned dose during SR-9-2 Check Valve gasket work.

On January 31, 2001, during the Unit One outage, a maintenance crew of two men arrived at containment to clean bolt holes and replace studs in preparation for closing the Safety Injection Check Valve, SR-9-2. The activity was performed as follow-up to gasket replacement and valve inspection. This valve had a hot particle lodged in the valve assembly that was not successfully removed from valve internals when the cover was removed on the evening of January 30, 2001. The unshielded source was measured in the range of 30 Roentgen/hour (R/hr) and was subsequently shielded by an inch thick lead block. The dose rate at the plane of the valve opening was 1.8 R/hr.

Lead blankets were used for shielding over the opening, and a Foreign Material Exclusion (FME) barrier was in place when work was not being conducted.

The maintenance crew was monitored by dedicated radiation protection technician (RPT) coverage, in addition to thermoluminescence dosimeters (TLD) and electronic dosimeters (ED) when conducting the work. The RPT conducted a survey of the work area prior to the men starting work. The survey confirmed that the radiological conditions had not changed, the particle source remained in the valve.

After the survey, the RPT allowed the crew access to the work platform and continued to monitor the men as they worked. During the work activities, the RPT noticed that the dose rate on the platform changed, and he identified dose rates on the platform of approximately 3 R/hr. The RPT confirmed the elevated dose rates by surveying his location and found that the dose rate had increased from approximately 25 mR/hr to 250 mR/hr. Not confident of the detector reading, the RPT stepped off the ladder where he was located and retrieved an adjacent Eberline RO-2 ion chamber to confirm the platform dose rate. It is at this point that the radiation barrier failed, when the RPT who was continuously monitoring the job failed to immediately remove the work crew from the area.

Just as the confirmatory readings were being made, a second RPT arrived at the site and was told that elevated dose rates were being identified on the platform. The second RPT confirmed the elevated dose rates and ordered the crew to move to the far end of the platform, as he attempted to identify the source of the increased dose rates. When the RPT confirmed the initial 3R/hr readings, he ordered the crew off the platform, to wait in a low dose waiting area. The two RPTs then identified the location of the particle source on the platform (directly under the valve) and removed it from the platform area to a lead container for transport to the waste storage building. Dose rate measurements of the particle source on contact and at one foot were approximately 600 R/hr and 3 R/hr, respectively.

Following the incident, the inspector raised several questions to the licensee concerning the workers' distance from the source and the delay in removing the workers from the platform. Based on those questions and some inconsistencies in the initial details of the incident, the licensee chose to perform a re-enactment of the event. The re-enactment showed that the crew was on the platform for approximately 1.5 minutes from the time of identified dose rate increase until they evacuated the platform. Based on the actual recovered source, the licensee conducted a dose reconstruction study and, in conjunction with the ED and TLD data, was able to calculate an extremity and whole body dose for the activity.

For the purpose of dose assignment, the licensee needed to make distance and time assumptions. The licensee used a conservative time estimate of 5 minutes of exposure. The actual dose rate from the source was measured. The licensee assumed that Worker A had his foot at six inches from the source for 5 minutes. The source strength at this distance was 12 R/hr. This resulted in assigning an extremity dose of 1 Rem. Further, the licensee calculated that when the individual was in a squatting position, the knee was a distance of 20 inches from the source for a duration of 5 minutes. The source strength at this distance was 1.1 Rem/hr, resulting in an additional exposure of 90 milliRem whole body dose.

Worker B was located farther from where the source was located and did not bend down during this job duty. His foot was estimated at 12 inches from the source for 5 minutes for a total extremity dose of 250 milliRem. His knee in relation to the source was estimated at 24 inches for 5 minutes. The assigned additional whole body exposure was 63 milliRem.

The inspector reviewed the licensee's dose reconstruction and dose assignment. The dose estimates were conservative. However, if left uncorrected, the root cause of this event could become a more significant concern and could cause a significant unintended and unplanned dose to workers. Therefore, this issue was reviewed using the NRC Significance Determination Process (SDP). The finding was not an ALARA (as-low-as-reasonably-achievable) finding, did not involve an overexposure, and the ability to assess the dose to occupationally exposed worker was not compromised. In terms of substantial potential for an overexposure, the conditions of the work activity (exposure time, source strength, and distance from the source) could not have resulted in an overexposure. Therefore, in accordance with the SDP, this issue is a Green finding. The issue is being reviewed by the licensee and can be located in the licensee's corrective action program as Condition Report (CR) Number 2001-1095. Further, the licensee has issued this CR a level 1 tracking code, placing it in the highest priority for follow-up. A Root Cause Team has been assigned to investigate the event and determine the factors leading to the barrier failure.

.2 Problem Identification and Resolution

a. Inspection Scope

The inspector reviewed the licensee's self-assessments and the CR database that related to radiation protection technician performance, radiation worker practices, and high radiation area access controls during the Unit 1 refueling outage. The inspector evaluated the effectiveness of the radiation protection self-assessment process to trend and identify problems and to implement corrective actions.

b. Findings

No findings of significance were identified.

2OS2 As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls

.1 ALARA Planning

a. Inspection Scope

The inspector reviewed the station's collective exposure histories for the year 2000, the current exposure trends for the ongoing Unit 1 refueling outage, and planned and completed radiological work activities for the outage to assess current performance and exposure challenges. The inspector evaluated the exposure data and the station's three-year rolling average exposure information and compared it with national pressurized water reactor industry data.

b. Findings

No findings of significance were identified.

.2 Job Site Inspections and ALARA Controls

a. Inspection Scope

The inspector selected the following potentially high exposure or high radiation area active or recently completed job activities and evaluated the licensee's use of ALARA controls:

- Scaffolding Placement and Removal/Steam Generator Laser Imaging;
- Steam Generator Nozzle Dam Installation and Removal;
- Reactor Coolant Pump Seal Replacement;
- Control Rod Drive Canopy Seal Overlay; and
- SI-9-2 Checkvalve Gasket Replacement and Inspection.

The inspector surveyed work areas to verify that radiation levels were consistent with the licensee's survey data and verified that low dose areas were designated and appropriately used by workers. The inspector evaluated the licensee's engineering controls at selected locations and verified that the controls were consistent with those specified in the ALARA plans. The inspector also observed and questioned workers at various job locations to determine that they had adequate knowledge of radiological work conditions and exposure controls.

b. Findings

No findings of significance were identified.

.3 Source Term Reduction and Control

a. Inspection Scope

The inspector reviewed the status of the licensee's source term reduction program, focusing on those initiatives taken for the outage which included hydrolazing, other decontamination work, and installation of temporary shielding. The inspector also evaluated other ongoing source term reduction strategies, such as hot spot reduction initiatives, and verified that a viable source term control program was in place.

b. Findings

No findings of significance were identified.

.4 Radiological Work Planning

a. Inspection Scope

The inspector selected the following outage job activities that were estimated to exceed five person-rem or were conducted in high radiation areas and assessed the adequacy of the radiological controls and work planning:

- Scaffolding Placement and Removal/Steam Generator Laser Imaging;
- Steam Generator Eddy Current Testing;
- Nozzle Dam Installation and Removal;
- Inservice Inspection of Various Piping, Equipment and Associated Work; and
- SI-9-2 Checkvalve Gasket Replacement and Inspection.

For each job activity, the inspector reviewed ALARA evaluations and associated dose mitigation techniques and evaluated the licensee's exposure estimates and performance. The inspector also assessed the integration of ALARA requirements into work packages and attended selected pre-job briefings, if the work was ongoing, to evaluate the licensee's communication of work plans and radiological safety precautions.

b. Findings

No findings of significance were identified.

.5 Verification of Exposure Estimate Goals and Exposure Tracking System

a. Inspection Scope

The inspector reviewed the methodology and assumptions used for the Unit 1 outage exposure estimates and exposure goals and compared job dose rate and man-hour estimates for accuracy. The inspector verified that job dose history files and dose reductions (anticipated through industry and corporate lessons learned) were appropriately used to forecast outage doses. The inspector also reviewed the licensee's exposure tracking system to determine if the level of exposure tracking detail, exposure report timeliness, and exposure report distribution was sufficient to support control of collective exposures. The inspector verified that the licensee's dose estimates for the outage were reasonably accurate and confirmed that no outage jobs greater than 5 person rem exceeded respective dose estimates by more than 50 percent.

b. Findings

No findings of significance were identified.

.6 Problem Identification and Resolution

a. Inspection Scope

The inspector reviewed the licensee's CR database for outage related radiological incidents involving personnel contamination events and RPT and radiation worker performance to evaluate the licensee's ability to identify and correct problems. The inspector verified that there were no radiation protection department licensee event reports or internal exposures in excess of 100 milliRem committed effective dose equivalent.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspector reviewed the licensee's corrective action program records for liquid and gaseous effluent releases that were reported to the NRC for the last four quarters to ensure that all Performance Indicator (PI) data was properly counted. The inspector also reviewed plant incidents to assess if there were any that involved radioactive liquids and gases that were not bounded by plant collection and monitoring systems and to assess the potential for unmonitored release paths.

A March 27, 2000, event and licensee follow-up were documented in NRC Inspection Report 50-282/2000004(DRP); 50-306/2000004(DRP). The event involved a resin sluice evolution that created a high radiation area greater than one rem/hr, which the licensee failed to timely identify and secure. For the first quarter of 2000 (January to March 2000), the licensee reported one PI occurrence based on concurrent nonconformance related to the March 27, 2000, event. The inspector reviewed the NRC and licensee conclusions regarding the number of PI occurrences resulting from this event and confirmed that the licensee amended its reported PI occurrences for the first quarter of 2000 to report two PI occurrences.

b. Findings

No findings of significance were identified. Based on the licensee's amended PI report for the first quarter of 2000, Unresolved Item (URI) 50-282/2000-007-01; 50-306/2000-007-01 was closed.

4OA6 Management Meetings

Exit Meeting Summary

The inspector presented the inspection results to Mr. Werner and other members of licensee management at the conclusion of the inspection on February 15, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

T. Barry, Maintenance
T. Beard, Training
M. Brossart, Engineering Superintendent
F. Englett, Radiation Protection Supervisor
L. Gard, Maintenance General Superintendent
S. Ginkle, Radiation Protection Technician
G. Gore, Engineering Superintendent
A. Johnson, General Supervisor, Radiation Protection and Chemistry
J. Kivi, Licensing Engineer
D. LaLone, System Engineer
S. Lappegaard, Chemistry
D. Larimer, Effluents and Environmental
G. Lenertz, Executive Engineer
G. Malnowski, ALARA Coordinator
C. Mundt, Engineering Superintendent
S. Munns, Contract Radiation Protection
S. Rupp, Contract Radiation Protection
D. Schuelke, Radiation Protection and Emergency Planning, NMC
R. Sloss, Quality Engineer
J. Sorenson, Site Vice President
M. Werner, Plant Manager
P. Wildenborg, Health Physics
J. Wright, Maintenance

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

50-282/2000-007-01 50-306/2000-007-01	URI	Occupational Radiation Safety PI Reporting Issues (4OA2)

Discussed

None

INSPECTION PROCEDURE USED

IP71121.01 Access Control to Radiologically Significant Areas
IP71121.02 ALARA Planning and Controls

LIST OF ACRONYMS USED

ALARA	As-Low-As-Is-Reasonably-Achievable
CR	Condition Report
DRS	Division of Reactor Safety
ED	Electronic Dosimeter
FME	Foreign Material Exclusion
HRA	High Radiation Area
PERR	Public Electronic Reading Room
PI	Performance Indicator
RP	Radiation Protection
RPT	Radiation Protection Technician
RRA	Radiologically Restricted Area
RWP	Radiation Work Permit
SDP	Significance Determination Process
TLD	Thermoluminescence Dosimeter

DOCUMENTS REVIEWED

Reports

Summary of DRD Exposure, January 31, 2001
Exposure Overview by Activity/Task, January 1 to December 31, 2001
Exposure Overview by Activity/Task, January 20 to February 1, 2001
Prairie Island Nuclear Plant Root Cause Investigation Report for CR 20005824, (Revision 0)
Weekly Radiation Exposure, January 20 to February 25, 2001 Outage
SI 9-2 Hot Particle Dose Assessment

Radiation Work Permits

Radiation Work Permit No. 01015 (Revision 0)
Radiation Work Permit No. 01027 (Revision 1)
Radiation Work Permit No. 01094 (Revision 0)
Radiation Work Permit No. 01123 (Revision 1)
Radiation Work Permit No. 01128 (Revision 1)
Radiation Work Permit No. 11015 (Revision 0)
Radiation Work Permit No. 11022 (Revision 4)
Radiation Work Permit No. 11024 (Revision 1)
Radiation Work Permit No. 11025 (Revision 1)
Radiation Work Permit No. 11027 (Revision 1)
Radiation Work Permit No. 11094 (Revision 0)
Radiation Work Permit No. 11122 (Revision 1)
Radiation Work Permit No. 11123 (Revision 1)
Radiation Work Permit No. 11126 (Revision 0)

ALARA Reviews

ALARA Review, 2001 Safety Injection Check Valve SI-9-2 Repair Seal

Condition Reports

20010232
20010258
20010325
20010342
20010405
20010455
20010474
20010488
20010537
20010644
20010788
20010812
20010986
20011095

Procedures

PINGP 258 (Revision 10), “Radiation Protection Survey Record”
PINGP RPIP 1004 (Revision 3), “Radiation Protection ALARA Program”
PINGP RPIP 1105 (Revision 13), “Extremity Monitoring”
PINGP RPIP 1106 (Revision 4), “ALARA Reviews”
PINGP RPIP 1108 (Revision 2), “Radiation Protection Key Control”
PINGP RPIP 1122 (Revision 7), “Hot Particle Program”
PINGP RPIP 1130 (Revision 10), “On the Job Dose Monitoring Procedures”
PINGP RPIP 1133 (Revision 7), “Multiple TLD Badging”
PINGP RPIP 1135 (Revision 10), “RWP Coverage”

Work Control Package

Work Control Package 9905374