Northern States Power Company

Prairie Island Nuclear Generating Plant

1717 Wakonade Dr. East
Welch, Minnesota 55089

July 2, 1997

10 CFR Part 50
Section 50.73

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

**PRAIRIE ISLAND NUCLEAR GENERATING PLANT**
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

**Unit 1 Reactor Trip Caused by Electrical Ground in Rod Control System**

The Licensee Event Report for this occurrence is attached. In the report, we made no new NRC commitments.

This event was reported via the Emergency Notification System in accordance with 10 CFR Part 50, Section 50.72, on June 2, 1997. Please contact us if you require additional information related to this event.

Joel P. Sorensen

Joel P Sorensen
Plant Manager
Prairie Island Nuclear Generating Plant

c: Regional Administrator - Region III, NRC
NRR Project Manager, NRC
Senior Resident Inspector, NRC
Kris Sanda, State of Minnesota

IE22 1/1

Attachment

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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TITLE (4)

Unit 1 Reactor Trip Caused by Electrical Ground in Rod Control System

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER |
| | | | | | | | | | | 05000 |
| 06 | 02 | 97 | 97 | -- 08 -- | 97 | 7 | 2 | 97 | FACILITY NAME | DOCKET NUMBER 05000 |

OPERATING MODE (9): N

POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

| | | | |
|---|---|---|---|
| 20.2201(b) | 20.2203(a)(2)(v) | 50.73(a)(2)(i) | 50.73(a)(2)(viii) |
| 20.2203(a)(1) | 20.2203(a)(3)(i) | 50.73(a)(2)(ii) | 50.73(a)(2)(x) |
| 20.2203(a)(2)(i) | 20.2203(a)(3)(ii) | 50.73(a)(2)(iii) | 73.71 |
| 20.2203(a)(2)(ii) | 20.2203(a)(4) | X 50.73(a)(2)(iv) | OTHER |
| 20.2203(a)(2)(iii) | 50.36(c)(1) | 50.73(a)(2)(v) | Specify in Abstract below or in NRC Form 366A |
| 20.2203(a)(2)(iv) | 50.36(c)(2) | 50.73(a)(2)(vii) | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Gene Eckholt | 612-388-1121 |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS |
|---|---|---|---|---|---|---|---|---|---|
| B | AA | CBL | W351 | No | | | | | |
| | | | | | | | | | |

| SUPPLEMENTAL REPORT EXPECTED (14) | | EXPECTED SUBMISSION | MONTH | DAY | YEAR |
|---|---|---|---|---|---|
| YES (If yes, complete EXPECTED SUBMISSION DATE). | X NO | | | | |


ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On June 2, 1997, Prairie Island Unit 1 was at steady-state, 100% power with the Rod Control System in automatic. At approximately 11:10 AM, Unit 1 control room operators received a "Rod Control System Non-Urgent Failure Alarm" followed seconds later by a "Rod Control System Urgent Failure Alarm". Approximately thirty seconds later, Unit 1 tripped from 100% power on a Negative Flux Rate Trip generated by Power Range Nuclear Instrumentation. The plant response to the reactor trip was as expected, except that following the trip, leakage past the normally closed Feedwater Regulation Bypass Valve to 11 Steam Generator (11 FRBV) resulted in a faster than expected cooldown rate. Operators took manual actions to stop the cooldown.

A short to ground was found in the (+) stationary gripper coil wire in a Control Rod Drive Mechanism (CRDM) cable. The cause of the shorted cable has been initially attributed to foreign material within the cable connector during original manufacture of cable assemblies. Corrective actions are being taken to ensure the cause of the failure is thoroughly understood and any actions necessary to prevent recurrence are identified. The CRDM cables on Unit 2 will be meggared during the next Unit 2 refueling outage.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

## EVENT DESCRIPTION

On June 2, 1997, Prairie Island Unit 1 was at steady-state, 100% power with the Rod Control System in automatic. At approximately 11:10 AM, Unit 1 control room operators received a "Rod Control System Non-Urgent Failure Alarm" followed seconds later by a "Rod Control System Urgent Failure Alarm". Approximately thirty seconds later, Unit 1 tripped from 100% power on a Negative Flux Rate Trip generated by Power Range Nuclear Instrumentation.

The plant response to the reactor trip was as expected, except that following the trip, leakage past the normally closed Feedwater Regulation Bypass Valve to 11 Steam Generator (11 FRBV) resulted in a faster than expected cooldown rate. Operators took manual actions to stop the cooldown. The plant response to the reactor trip also included the actuation of the auxiliary feedwater system as a result of low steam generator level. The actuation of the auxiliary feedwater system on steam generator low level is an expected part of the plant response to a reactor trip.

Upon initial examination of the rod control system[1] following the trip, both +24 $V_{DC}$ power supplies[2] in power cabinet 12AC were found tripped from Over Voltage Protection (OVP). The circuit cards in power cabinet 12AC were inspected. After re-seating all cards in power cabinet 12AC and resetting the +24 $V_{DC}$ power supplies, the Reactor Trip Breakers were closed. At this time, both +24 $V_{DC}$ power supplies in power cabinet 11BD and the Main +24 $V_{DC}$ power supply in power cabinet 11AC tripped due to OVP actuation (both +24 $V_{DC}$ supplies in 12AC remained operating). The reactor trip breakers were manually reopened. Resistance measurements taken between ground and the neutral bus were found to be ~12Ω. 50Ω is normal. A short to ground was subsequently found in the (+) stationary gripper coil wire in the H8 Control Rod Drive Mechanism (CRDM) cable[3]. Once the H8 CRDM loop was isolated from the rest of the rod control system, the neutral to ground resistance measured normal. The cable in question was disassembled. Physical evidence of the short (burn traces) were found in the connector[4] assembly The H8 CRDM was tested satisfactorily after it was isolated from the faulted cable.

## CAUSE OF THE EVENT

The shorted Rod H8 stationary gripper coil circuit placed an undesirable voltage on the rod control system common neutral bus which is used as a reference for the +24$V_{DC}$ power supplies and is common to all three power cabinets. The undesirable voltage caused both +24$V_{DC}$ power supplies in cabinet 12AC to trip on OVP. Without the +24$V_{DC}$ power, control power for the firing of thyristors that gate current to Control Rod Drive Mechanism (CRDM) stationary grippers was eliminated. The CRDM stationary grippers hold the rods in the withdrawn position. The loss of control power to the firing

[1] (EIIS System Identifier: AA)
[2] (EIIS System Identifier: AA; EIIS Component Identifier: JX)
[3] (EIIS System Identifier: AA; EIIS Component Identifier: CBL)
[4] (EIIS System Identifier: AA; EIIS Component Identifier: CON)

thyristors caused the stationary grippers to release and the cabinet 12AC control rods[5] to fall into the core. The rapid influx of negative reactivity caused by the falling rods was detected by the excore Nuclear Instrumentation System Power Range Detectors which actuated a reactor trip signal on Negative Flux Rate, as would be expected. Opening the reactor trip breakers removed the voltage from the neutral bus prior to tripping any power supplies on OVP that were as-of-yet unaffected in the remaining power cabinets. Troubleshooting activities reproduced the failure: when trip breakers were closed, power supplies in cabinets 11BD and 11AC tripped on OVP through the common neutral bus. As occurred in the actual trip, once current supply to rod H8 was terminated, further OVP actuation of the remaining power supplies was prevented.

As the plant was returned to service, the startup data for Rod H8 was compared to data from the last Unit 1 startup in April 1997. The startup data looked nearly identical, indicating that the CRDM cable failure did not cause the rod to misstep during the previous startup. Based on the results of the comparison of startup data, and operating experience since the previous startup, it was concluded that the CRDM cable degradation had no apparent impact on the operation of the rod control system prior to the failure.

The cause of the shorted coil cable has been initially attributed to foreign material within the cable connector during original manufacture of cable assemblies by Westinghouse. The cables were originally specified to have a 20 year useful life in the environment in which they are used. They have been installed approximately 7 years. The connector will be sent to an independent laboratory or Westinghouse, if possible, to confirm the failure mechanism.

ANALYSIS OF THE EVENT

The rod control system responded to the faulted cable as expected. The power supplies are designed to shut off in an overvoltage condition, and do not reset until power has been removed from the source. Reducing the neutral bus voltage is effectively the same as raising the power supply output, causing the OVP trip. Once the OVP tripped both power supplies, the rods were released to fall into the core.

The Power Range Nuclear Instrumentation System tripped the reactor on negative rate due to all cabinet 12AC rods dropping into the core (total of 9). The trip was appropriate.

The cooldown resulting from leakage past 11 FRBV did not exceed plant Technical Specification limitations on cooldown rates (100°F/hour). Operations personnel responded to the cooldown in a timely manner.

This event is reportable pursuant to 10 CFR Part 50, Section 50.73(a)(2)(iv) since this was an unplanned actuation of the reactor protection system. The health and safety of the public were

[5] (EIIS System Identifier: AA; EIIS Component Identifier: ROD)

unaffected since, with the exception of 11 FRBV, the plant systems responded as designed to the automatic trip. Because operator actions terminated the unexpected cooldown, the leakage past 11 FRBV had no impact on the health and safety of the public.

## CORRECTIVE ACTION

Troubleshooting activities were conducted to reasonably ensure no other failures occurred as a result of the undesirable voltage on the neutral bus. During troubleshooting, the CRDM timing test was performed: a failed phase control card and failed stationary gripper regulation card were found and replaced in cabinet 12AC, and the auxiliary $-24V_{DC}$ power supply in cabinet 11AC was found failed and was replaced. All rod control CRDM circuit loops were checked for grounds using specialized ground detection gear, meggared wire-to-wire and wire-to-ground, and coil resistance and inductance was measured. All rod control system power cabinet circuit board voltages were measured at test points and were compared against baseline data taken previously. Traces of various waveforms were also measured. The results of these tests were satisfactory, with nothing abnormal found. Finally, prior to leaving Hot Shutdown, the CRDM timing test was performed once again satisfactorily, and the rod drop test was performed satisfactorily.

All feedwater regulation valves and bypass valves were tested during shutdown and were found to be normal, except for 11 FRBV. A work request was initiated, and 11 FRBV was found to have a missing cotter pin on the travel stop nut, preventing it from full closure. 11 FRVB was tested following the repairs and was found to perform satisfactorily.

The following corrective actions are being taken to ensure the cause of the connector failure is thoroughly understood and any actions necessary to prevent recurrence are identified:

- The failed connector will be sent to an independent laboratory or Westinghouse, if possible, to further analyze the failure mechanism.

- CRDM and CRDM cable assembly preventative maintenance strategies will be reviewed to determine if any changes are necessary to reduce the possibility of this event from recurring.

The CRDM cables on Unit 2 will be meggared during the next Unit 2 refueling outage in an effort to determine if similar problems exist with the Unit 2 CRDM cables.

## FAILED COMPONENT IDENTIFICATION

Westinghouse CRDM cable assembly, Part Number 8249C08G03.

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## PREVIOUS SIMILAR EVENTS

There have been no similar previous events at Prairie Island.