Northern States Power Company

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March 23, 1992

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

Interruption of One Train of Residual Heat Removal
<u>During a Unit 2 Reactor Coolant System Draining Operation</u>

This letter is written in response to Items #1 and #5 of Confirmatory Action Letter RIII-92-008 dated February 21, 1992. We responded to other actions required by the letter on February 23, 1992. We have completed our investigation and determined the causes of the interruption of decay heat removal during reduced inventory operation at Prairie Island Unit 2 on February 20, 1992. A detailed report is contained in our Unit 2 Licensee Event Report 92-002, which is attached.

This event was reported via the Emergency Notification System in accordance with 10 CFR Part 50, Section 50.72, on February 20, 1992. Please contact us if you require additional information related to this event.

Thomas M Parker
Manager
Nuclear Support Services

c: Regional Administrator - Region III, NRC
NRR Project Manager, NRC
Senior Resident Inspector, NRC
Dr Raymond Thron, MDH

Attachment

JE22 1/1

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Prairie Island Nuclear Generating Plant Unit 2

DOCKET NUMBER (2): 0 5 0 0 0 3 0 6

PAGE (3): 1 OF 1 0

TITLE (4): Interruption of One Train of Residual Heat Removal During a Unit 2 Reactor Coolant System Draining Operation

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 2	2 0	9 2	9 2	— 0 0 2 —	0 0	0 3	2 3	9 2	PINGP Unit 1	0 5 0 0 0 2 8 2
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

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

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	XX OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	Voluntary Report
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME: Arne A Hunstad

TELEPHONE NUMBER: AREA CODE 6 1 2 — 3 8 8 - 1 1 2 1

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

SUPPLEMENTAL REPORT EXPECTED (14)

XX YES (If yes, complete EXPECTED SUBMISSION DATE) ☐ NO

EXPECTED SUBMISSION DATE (15): MONTH 0 5 DAY 1 5 YEAR 9 2

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

On February 20, 1992, Unit 2 was in the cold shutdown condition for a scheduled refueling and maintenance outage. Reactor Coolant System (RCS) temperature was being maintained at about 135 degrees F as indicated on core exit thermocouples. Water was being drained from the RCS to establish conditions for removing steam generator manways and installing steam generator nozzle dams in preparation for eddy current inspection of steam generator tubes. The RCS water level was allowed to decrease to too low a level and the inservice Residual Heat Removal (RHR) pump (Train B) began entraining air. The RHR pump was stopped, makeup water to the RCS was accomplished in accordance with procedures and the standby (Train A) RHR pump was placed in service for shutdown cooling. Although one core exit thermocouple reached 221.5 degrees F, the RCS average temperature remained below 200 degrees F. A Notification of Unusual Event was reported and immediately terminated because the event rapidly de-escalated to a non-reportable condition. Normal shutdown cooling flow was off for about 22 minutes. Train A RHR was available for cooling throughout this event, first via the Refueling Water Storage Tank (RWST), then in the normal shutdown cooling mode.

EVENT DESCRIPTION

On February 20, 1992, Unit 2 was in cold shutdown for refueling. Reactor Coolant System (RCS)(EIIS System Identifier AB) temperature was being maintained at about 135 degrees F, as indicated on core exit thermocouples, by Train B of the Residual Heat Removal System (RHR)(EIIS System Identifier BP). In preparation for inservice inspection of steam generators, plant operators were in the process of draining the Reactor Coolant System to the centerline of the Reactor Coolant System piping. This operation normally takes approximately 6 hours. The Reactor Coolant System was being drained from the loop drain to the Chemical and Volume Control System (CVCS) Holdup Tank (HUT) No. 121. In accordance with procedure D2, Reactor Coolant System Reduced Inventory Operation, the Reactor Coolant System pressure boundary was intact and vented to the Pressurizer Relief Tank (PRT) by way of the pressurizer power-operated relief valves. The Reactor Coolant System was being pressurized with nitrogen to aid in draining. See attached Figure. An engineer was assigned to provide assistance to the operating crew; this is customary for this evolution. However, the engineer assigned did not have as much experience with reduced inventory operations as engineering personnel assigned to this task in the past. The engineer also had an assignment to complete a functional test on a computer-based Reactor Coolant System level alarm and display system that had been installed during its last refueling outage. A similar system was tested and used during the last Unit 1 Reactor Coolant System draining and reduced inventory operation. The functional test procedure contained a note which stated that the Reactor Coolant System nitrogen overpressure should be minimized so the electronic level indicators would come on scale as early as possible. This note was missed by the engineer. The control room operators were not aware of the test note.

The procedure being used for draining, D2, in effect states in several places that pressure should be maintained at about 6 psig, and that the pressure should be allowed to decay as water is drained so that the pressure is less than 1 psig when the water level reaches Reactor Coolant System loop centerline. However, no guidance on how to accomplish this is given. The procedure did not contain guidance on reducing the draining rate as the end point was approached nor on pausing occasionally to verify conditions.

Draining of the Reactor Coolant System was begun at 1704 hours, February 20, 1992. The Reactor Coolant System was being pressurized to about 6 psig using nitrogen. Near the end of the day shift, the draining was stopped to allow for shift change and turnover to the night crew. The night crew resumed draining at 1934. Reactor Coolant System level was being continuously monitored locally (in the containment building) by an operator observing a clear tube, referred to as the "tygon tube". Since the top of this tube is open to containment atmosphere, any nitrogen overpressure in the Reactor Coolant System causes the level in the

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Prairie Island Unit 2	0 5 0 0 0 3 0 6	9 2	— 0 0 2 —	0 0	0 2 OF 1 0

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

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

tube to be higher than the actual water level in the Reactor Coolant System, making corrections to the indicated level necessary. This correction for overpressure was being calculated by personnel in the control room, first by licensed operators and the engineer, later only by the licensed operators. Similarly, the computer-based Reactor Coolant System level on the computer display (ERCS-D2) is referenced to containment atmosphere but is corrected for Reactor Coolant System pressure by the computer.

The draining procedure provides a table of level correction values for Reactor Coolant System overpressures up to 1.5 psig. Since Reactor Coolant System overpressure was being maintained above 1.5 psig, many sequential manual calculations to correct indicated level were necessary. Occasionally, errors were made in the calculations necessary to correct the tygon tube indication and convert it to the reference point used for the centerline of the Reactor Coolant System loops. These errors were caused by rounding off the pressure input to the calculations to expedite the calculation process. The sensitivity to rounding off the pressure input was not realized by the operators. The elevated Reactor Coolant System overpressure also over-ranged the new level transmitters, causing the computer to display "FAIL" for these points. However, the reason that the level was not available on the control room display (ERCS-D2) was not known to the operators nor to the assigned engineer. As concern over the unavailability of level on ERCS-D2 increased, the engineer left the control at the request of the shift manager to investigate and attempt to resolve the problem.

At about 2250 hours, the duty Shift Manager checked on the draining progress (as was done several times during the evolution by both the Shift Manager and Unit 2 Shift Supervisor) and calculated that it would take 32 minutes to reach Reactor Coolant System centerline at the current draining rate. This was announced to the operators. This determination was made by obtaining the level increase observed in the tank receiving the water (the Chemical and Volume Control Holdup Tank) and converting it from percent to gallons. In making this conversion, the Shift Manager used tank book data that has subsequently been determined to be in error. This calculation overestimated the volume to be drained.

Shortly after 2300 hours a corrected tygon tube level of about 723 feet was calculated. This corresponds to a level 4 inches below the loop centerline and raised operators' concern. In a short time frame, another tygon tube level reading was obtained from one operator in containment to confirm the level. A second operator in containment was directed to vent the RHR pump suction header and a control room operator began to depressurize the Reactor Coolant System by opening the pressurizer and reactor head vents. The operator venting the RHR suction reported the presence of air in the venting line. Also about this time

the ERCS-D2 display was noted to have the level on scale and that the level was below the reactor coolant piping centerline. The operator who was venting the RHR pump suction was directed to close the Reactor Coolant System drain valves. A third operator in containment heard this order and closed the drain valves.

The Unit 2 Shift Supervisor heard the order to close the drain valves and went to the control area. The time was about 2308 hours. The Shift Supervisor observed that the No. 22 RHR Pump motor current and flow were fluctuating. RHR trouble alarms were being received on the ERCS. The Shift Manager was in the control area also and saw RHR pump suction pressure low and fluctuating. The operators also saw these indications. The Shift Supervisor promptly ordered that No. 22 RHR Pump be stopped. No. 22 RHR Pump was stopped at 2310:05 hours. No. 21 charging pump was started. The Shift Supervisor ordered entry into contingency procedure D2 AOP1, Loss of Coolant While in a Reduced Inventory Condition, to respond to the condition (the draindown procedure referred the operator to this contingency procedure if suction problems occurred). No. 22 Charging pump was started in accordance with D2 AOP1. As core exit temperatures increased to 190 degrees F, a transition step to emergency procedure 2E-4, Core Cooling Following Loss of RHR Flow, was encountered and procedure 2E-4 was entered. In accordance with 2E-4, the Refueling Water Storage Tank (RWST) was lined up to supply water to the Reactor Coolant System via the unaffected RHR pump (No. 21 RHR Pump). This lineup does not include any of the common suction piping from the Reactor Coolant System; therefore, air entrained by the No. 22 RHR pump did not affect No 21 RHR Pump. No. 21 RHR Pump was started at 2325:57 hours. After the Reactor Coolant System water level was restored to approximately the reactor vessel flange level, makeup from the RWST was stopped. No. 21 RHR loop was then placed in service in the shutdown cooling mode at 2329:28 hours. The Reactor Coolant System was cooled down to a core exit temperature of about 135 degrees F at 2336 hours. A Notification of Unusual Event was reported and immediately terminated because the event rapidly de-escalated to a non-reportable condition.

<u>CAUSE OF THE EVENT</u>

PRIMARY CAUSES

1. Supervisory Methods - An appropriate level of in-task supervision was not properly determined prior to performing the task.

2. Work Organization/Planning - Sufficient engineering expertise was not continuously available to the control roo personnel, as it had been in the past.

3. Written Communications

   - Procedure D2 did not provide adequate guidance on what pressure to maintain related to the volume drained, did not provide guidance on pauses in the draining process to assess plant conditions, and did not provide information regarding the sensitivity of the level correction calculation to rounding off input values. Procedure D2 did not provide sufficient detail to be used without expert technical assistance and close supervisory oversight.

   - The tank book contained an incorrect conversion factor for Chemical and Volume Control Holdup Tank level percent to gallons.

4. Interface Design/Equipment Condition - A nitrogen overpressure required local instrumentation readings to be corrected to obtain actual level. It also caused the electronic level instrumentation to be out of range at elevated pressures.

5. Verbal Communications - There was inadequate pre-job briefing.

6. Work Practices

   - A note in a work request procedure was not observed.

   - Administrative procedure SWI-O-34, Conduct of Off-Normal Activities, which spells out requirements for management oversight of infrequently performed evolutions, was not used.

SECONDARY CAUSES

7. Training/Qualification - Training provided did not contain sufficient detail on the nuances required for the draining evolution.

8. Change management - Effectiveness of SWI-O-34 implementation had not been validated since its approval three weeks earlier.

9. Resource management - The tank book is not a controlled document.

ANALYSIS OF THE EVENT

For the draining and reduced inventory evolution, seven pumps were available to provide makeup water to the core and five sources of power were being maintained. Outage activities that could affect these sources had been curtailed prior to the evolution. At all times at least one steam generator and one auxiliary feedwater pump were being maintained available for heat removal contingencies by procedure. With one steam generator available for decay heat removal, loss of RHR flow can be tolerated for more than several hours.

The maximum recorded core exit temperature during the event was 221.5 degrees F. At this point, the combination of the nitrogen pressure (3.2 psig) and the head of water above the fuel pins at that time (4.43 feet) prevented boiling. A review of the data shows that the maximum recorded core exit temperature remained less than the saturation temperature throughout the transient. The lowest water level reached during the transient was 3.74 feet above the top of the fuel pins.

The average temperature of the water in the reactor vessel did not exceed 200 degrees F. Therefore, the unit remained in the cold shutdown mode throughout the event. This conclusion is based on 3 independent methods of estimating Reactor Coolant System temperature.

An evaluation of the effects of the event on the fuel in the core was performed. Based on the relative temperatures and heatup rates during the event compared to normal operational values, it was concluded that the event had no adverse effects on the fuel. Also, Reactor Coolant System samples after the event showed a slight decrease in activity, consistent with the addition of makeup water.

The heatup rate of some portions of the water in the Reactor Coolant System exceeded the Technical Specification value of 60 degrees F per hour stated in paragraph 3.1.B.1.a.1. The action required for this condition specified in 3.1.B.1.b is to perform an engineering evaluation. A conservative evaluation of the Reactor Coolant System pressure boundary was completed assuming a step heatup of 80 degrees F. This evaluation showed that the structural integrity of the Reactor Coolant System remains acceptable for continued operation.

As a precautionary measure, all unnecessary personnel were evacuated from the containment building in accordance with procedures. Prior to the evacuation, the operators in containment were notified of the reason for the upcoming evacuation and were instructed to remain at their posts.

The Reactor Coolant System was intact throughout the event. Effluent radiation monitors showed no increase in readings. Air samples from containment showed no increase in radioactivity level. Therefore, no release of radioactivity to the environment occurred as a result of this event.

Based on the above, there was no effect on public health and safety.

This event is not reportable under 10 CFR Part 50, Section 50.73. This report is being provided because of the generic implications of this event.

CORRECTIVE ACTIONS

Actions Taken Specifically for the Unit 2 February 1992 Refueling:

1. Immediately after the incident the pressurizer manway was removed to vent the Reactor Coolant System to containment atmosphere to bring all level indications into agreement.

2. The level compensation due to PRT pressure was deleted from the ERCS-D2 display to allow all instruments to be referenced to containment atmosphere.

3. For this draining operation, administrative controls were added for all instruments that could have an effect on Reactor Coolant System level indication.

4. Procedure D2.3, Reactor Coolant System Reduced Inventory Operation while Vented to Containment Atmosphere, was written for one-time use to allow draining of the Reactor Coolant System to install steam generator nozzle dams. Highlights of the differences in this procedure from the original draining procedure are as follows:

   - A senior engineer experienced in reduced inventory operation was required to be present during the draining.

   - Shift personnel with no concurrent duties were assigned to perform the draining.

   - Shift management personnel with no other concurrent duties were required to supervise the draining.

- D2.3 was reviewed in accordance with SWI-0-34, which spells out requirements for management oversight of infrequently performed evolutions.

- Steps in the procedure specifically required stopping the draining to allow refocusing of the team prior to draining below the top of the Reactor Coolant System hot leg.

- Pre-job briefing requirements were spelled out in detail.

- Containment closure times were made consistent with the time to boiling rather than the time to core uncovery.

- Precautions were added to specifically address work around electrical equipment that could affect RHR.

5. New procedure 2D2.1 was developed for draining the Reactor Coolant System to remove the nozzle dams. This procedure had many of the same enhancements as D2.3.

Actions Taken to Prevent Future Occurrences:

6. Draining procedures have been removed from the approved procedure list to assure they are not used again until revised with the recommendations of the task force.

7. A Prairie Island multidisciplinary task force was formed to assess the event in detail and to assure all areas for improvement are extracted from the event.

8. Emergency Procedure 2E-4, Core Cooling Following Loss of RHR Flow, was changed as follows:

   - Entry condition thermocouple temperature was changed from 190 to 150 degrees F to allow earlier entry.

   - An entry condition based strictly on the operators' judgement was added to allow quicker inventory makeup.

   - Initial response strategies were changed to use the safety injection pumps earlier in the scenario, rather than relying on the limited flow of the charging pumps.

9. Emergency plan procedures for RHR interruption events were clarified.

10. All operations crews using the above procedures were trained on them.

11. Requests have been initiated to review designs for draining the Reactor Coolant System which precludes going below mid-loop, and to review the thermo-hydraulics of the Reactor Coolant System during draining and during events when the Reactor Coolant System is not completely filled.

12. The Reactor Coolant System pressure boundary has been evaluated due to the heatup during the incident and found to be acceptable.

13. The Error Reduction Task Force has completed its event investigation.

14. Other procedures involving RHR manipulations were reviewed, and some changed, to be more conservative in the valving operations associated with RHR.

15. The tank book was removed from the control room and can only be used for information and not used for making safety related decisions.

16. Power Supply Quality Assurance has performed surveillances on both the Reactor Coolant System draining and the Modification package which installed the new Reduced Inventory equipment.

17. Operability assessments of No. 22 RHR Pump and surveillance testing have been performed to assure the pump was not damaged during the event.

A long-term action plan is being developed that will implement improvements in procedures, hardware, training, and management of draining operations and other critical functions. The actions incorporated from this task force will be documented in an update to this report.

FAILED COMPONENT IDENTIFICATION

None.

PREVIOUS SIMILAR EVENTS

There have been no previous similar events reported at Prairie Island.

NRC FORM 366A
(6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

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