



Northern States Power Company

Prairie Island Nuclear Generating Plant

1717 Wakonade Dr. East
Weich, Minnesota 55089

May 31, 1994

10 CFR Part 50
Section 50.73

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

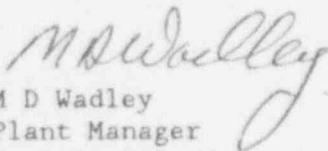
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Missed Surface Examinations of the Reactor Coolant Pump Flywheels

The Licensee Event Report for this occurrence is attached. In the report, we have made the following new NRC commitment:

A surface examination of the reactor coolant pump flywheel bore and keyway (in addition to the exposed surfaces, as required) will be done if a reactor coolant pump flywheel is removed from its shaft.

Please contact us if you require additional information related to this event.


M D Wadley
Plant Manager
Prairie Island

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

Attachment

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LICENSEE EVENT REPORT (LER)

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

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), 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.

FACILITY NAME (1) Prairie Island Nuclear Generating Plant U1	DOCKET NUMBER (2) 05000 282	PAGE (3) 1 OF 4
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TITLE (4) Missed Surface Examinations of the Reactor Coolant Pump Flywheels

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 NAME	DOCKET NUMBER
04	29	94	94	-- 03 --	00	05	31	94	Prairie Island U2	05000 306
									FACILITY NAME	DOCKET NUMBER
										05000

OPERATING MODE (9) N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)										
POWER LEVEL (10) 82%	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)			OTHER	
	20.405(a)(1)(iii)			X 50.73(a)(2)(i)			50.73(a)(2)(viii)(A)			(Specify in Abstract below and in Text, NRC Form 366A)	
	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 (Include Area Code) 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

SUPPLEMENTAL REPORT EXPECTED (14)						EXPECTED SUBMISSION DATE (15)		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)

During a 1993 internal audit of the inservice inspection activities at Prairie Island, the Nuclear Quality Department compared the Prairie Island Technical Specifications requirements for NDE examinations of the reactor coolant pump flywheels with the examinations which had been performed.

From this review and subsequent investigation it has been determined that the ten year interval surface examinations for both Units 1 and 2 have not been completed as required (at approximately 10-year intervals). A final determination of this was made on April 29, 1994 during a conference call with the NRC.

Since the determination that the examinations may not have been performed as frequently as required, all flywheels have been examined.

NRC FORM 366A (5-92)		U.S. NUCLEAR REGULATORY COMMISSION		APPROVED BY OMR NO. 3150-0104 EXPIRES 5/31/95			
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION				ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), 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.			
FACILITY NAME (1)		DOCKET NUMBER (2)		LER NUMBER (6)		PAGE (3)	
Prairie Island Unit 1		05000 282		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	2 OF 4
				94	-- 03 --	00	

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

EVENT DESCRIPTION

During a 1993 internal audit of the inservice inspection activities at Prairie Island, the Nuclear Quality Department compared the Prairie Island Technical Specifications requirements for NDE examinations of the reactor coolant pump flywheels with the examinations which had been performed. From Table TS.4.2-1, the requirements are:

An in-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and key way at approx. 3 year intervals, during the refueling or maintenance shutdown coinciding with the in-service inspection schedule as required by the ASME B & PV Code Section XI.

A surface examination of all exposed surfaces and complete ultrasonic volumetric examination at approx. 10 year intervals, during the plant shutdown coinciding with the in-service inspection schedule as required by the ASME B & PV Code Section XI. Removal of the flywheel is not required to perform these examinations.

From this review and subsequent investigation it has been determined that the ten year interval surface examinations for both Units 1 and 2 have not been completed as required. A final determination of this was made on April 29, 1994 during a conference call with the NRC.

The 10 year interval surface examinations had been last completed in 1976 during modifications to the pump lubrication systems at which time the flywheels were removed. The surface examination of all exposed surfaces was scheduled and completed for Unit 2 during the fall 1993 refueling outage and for Unit 1 during the spring 1994 refueling outage.

CAUSE OF THE EVENT

The primary cause of the event was an incorrect interpretation of the Technical Specification requirements. The Technical Specification requires that a surface examination of all exposed surfaces be performed at approximately 10 year intervals, during the plant shutdown coinciding with the inservice inspection schedule as required by the ASME B & PV code Section XI. This requirement was interpreted to mean that the Section XI requirements for scheduling and deferral would be followed. Section XI allows for the item to be deferred to the end of the current interval in some cases. Since more informative examination results could be obtained if the pump were to be disassembled for maintenance or repair, it was decided to perform the 10 year interval examinations at the time of such disassembly or, if the pumps were not disassembled for any purpose, the examinations would be scheduled and completed prior to the end of the

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

interval.

Our interpretation was in error in that the Technical Specification words "during the plant shutdown coinciding with the in-service inspection schedule as required by the ASME B & PV Code Section XI" did not mean that we should follow the ASME Section XI scheduling requirements. It rather was intended that we need not perform the required examinations at a plant shutdown simply because it was closest to being at a 10 year interval, but that we have the flexibility of choosing to perform them during a plant shutdown when ASME Section XI required in-service inspections are scheduled.

ANALYSIS OF THE EVENT

An engineering evaluation was performed to study the safety of continued operation until the next scheduled outage when the examinations could be completed. This evaluation concluded that there was little or no risk to safe operation until the next refueling outage. This conclusion was based on the following:

- The Prairie Island USAR 4.3.3.2.1 describes the Reactor Coolant Pump. The USAR notes that the critical flaw size for the flywheel is 17 inches. The assumed starting flaw is 1/2". The crack growth rate is 0.030" to 0.060" per 1000 cycles with an assumed 400 cycles in 40 years. Using the assumptions of the USAR the starting flaw would essentially never reach critical size.
- The examination results to date have detected no evidence of material degradation.

In addition, we have performed the complete volumetric examination each time we perform the volumetric examination of the bore and keyway. That is, we perform this examination at approximately 3 year intervals rather than the required 10 year intervals. We believe that this examination provides more meaningful information than a surface examination (of the exposed surfaces) and therefore, although we were not performing the examinations in strict accordance with the Technical Specifications, we were adequately monitoring the physical condition of the flywheel.

Therefore, the health and safety of the public are unaffected by this event. This event is reportable pursuant to 10 CFR Part 50, Section 50.73(a)(2)(i)(B).

CORRECTIVE ACTION

The examinations have been completed for both Units.

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

We have scheduled the required flywheel examinations in accordance with the correct interpretation of the Technical Specification requirements.

Additionally, as we had originally intended, a surface examination of the bore and keyway (in addition to the exposed surfaces, as required) will be done if a reactor coolant pump flywheel is removed from its shaft.

FAILED COMPONENT IDENTIFICATION

None.

PREVIOUS SIMILAR EVENTS

Deficiencies in the Inservice Inspection Program have been reported as Unit 1 LERS 80-024, 92-011, and 93-004 and Unit 2 LER 91-001.