NSP

Northern States Power Company

Prairie Island Nuclear Generating Plant

1717 Wakonade Dr. East
Welch, Minnesota 55089

March 8, 1996

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

Peak Clad Temperature Analysis not Performed for Core Reload Design

The Licensee Event Report for this occurrence is attached. In the report, we made new NRC commitments indicated in the report as the statements in italics.

This event was reported via the Emergency Notification System in accordance with 10 CFR Part 50, Section 50.72, on February 8, 1996. Please contact us if you require additional information related to this event.

Don Schuelke for

Michael D Wadley
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

Attachment

130051

IE22
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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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Prairie Island Nuclear Generating Plant Unit 1	05000 282	1 OF 4

TITLE (4)
Peak Clad Temperature Analysis not Performed for Core Reload Design

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
									Prairie Island Unit 2	05000 306
02	08	96	96	-- 05 --	0	3	8	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)		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)
POWER LEVEL (10)			20.2203(a)(1)		20.2203(a)(3)(i)	X	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)		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)
Jack Leveille	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	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)

When performing a safety analysis for a core, it is standard practice for the Nuclear Analysis Department (NAD) to begin by reviewing the analysis for a previous core. The core physics data of the new core is compared with that of a previous core(s) to determine if previous analysis will bound the new core. If a previous analysis bounds the new core, that analysis is not repeated for the new core. This limits the amount of analysis that must be performed for the new core. The acceptance criteria for the parameters of concern is given in the Topical Report and in a NAD procedure. Both the Topical Report and the NAD procedure give general guidance, but not specific instructions. It is standard practice that the total analysis package, called the Reload Safety Evaluation, is verified by another NAD person and is also design reviewed by a consultant. Nevertheless, a Peak Clad Temperature (PCT) analysis was not performed for Unit 2 cycle 17 (referred to as P217) core was not performed.

Subsequent analysis shows that both units currenlty meet the peak clad temperature design criterion.

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

EVENT DESCRIPTION

On February 8, 1996, while Unit 2 was operating at 100% power and Unit 1 was shutdown for refueling, it was determined that a reactor core design parameter had not been verified by analysis to meet its acceptance criterion for the Main Steam Line Break accident for the current Unit 2 cycle. When performing a safety analysis for a core, it is standard practice for the Nuclear Analysis Department (NAD) to begin by reviewing the analysis for a previous core. The core physics data of the new core is compared with that of a previous core(s) to determine if previous analysis will bound the new core. If a previous analysis bounds the new core, that analysis is not repeated for the new core. This limits the amount of analysis that must be performed for the new core. The acceptance criteria for the parameters of concern is given in the Topical Report and in a NAD procedure. Both the Topical Report and the NAD procedure give general guidance, but not specific instructions. It is standard practice that the total analysis package, called the Reload Safety Evaluation (RSE),is verified by another NAD person and is also design reviewed by a consultant. Nevertheless, a Peak Clad Temperature (PCT) analysis was not performed for the Unit 2 cycle 17 (P217) core or the Unit 1 cycle 17 (P117) core.

The safety analysis for Prairie Island Unit 1 cycle 17 (referred to as P117) was completed and verified in May of 1994. This was the first core reload for which the Main Steam Line Break accident (MSLB) analysis showed that DNBR would fall below the minimum that would assure no fuel failures (DNBR $\geq$ 1.45). The acceptance criteria of interest for the MSLB for this event are:

- DNBR $\geq$ 1.45;
- PCT $\leq$ 2750 $^{\circ}$F;
- if the DNBR limit is reached, then the number of fuel rods calculated to exceed the DNBR limit shall not exceed the maximum number calculated to maintain off site doses within 10CFR100 limits.

It is known that PCT cannot approach the 2750 ^{0}F limit unless DNBR falls below its limit. Therefore a specific analysis for PCT does not need to be performed unless the DNBR limit is reached. P117 was the first analysis that showed that the DNBR limit was reached, however no analysis for PCT was performed. Neither the verification process nor the review process detected this oversight in the P117 analysis package.

The RSE for core P217 was performed in accordance with the past practice of reviewing previous analysis folders to identify transients not bounded by previous analysis. Since the MSLB analysis of P217 produced results similar to P117, no analysis of PCT was performed for P217. Again, neither the verification process nor the review process detected the oversight of not performing the PCT analysis.

While evaluating Follow On Item 760 (a tracking form issued by the Design Standards Department when they identify a potential design discrepancy) , NAD discovered that the PCT analysis should have been performed for P217 and P117.

Analysis of P217 out to 13 Gwd/Mtu shows that the MSLB criteria for P217 are met. Core P118 is also bounded by this analysis out to 13 Gwd/Mtu. Further analysis must be done to determine if the criteria are met at burnups greater than 13 Gwd/Mtu. Unit 2 could reach 13 Gwd/Mtu as early as May 31, 1996. Unit 1, which has just restarted following the refueling shutdown, could reach 13 Gwd/Mtu as early as February 2, 1997.

CAUSE OF THE EVENT

The NAD procedure did not require documentation that each and every acceptance criteria was met and how it was met.

The attention to detail of the analyst, the verifier and the consultant was not sufficient to detect that the PCT calculation needed to be performed but had not been performed.

ANALYSIS OF THE EVENT

Analysis was performed that shows that all acceptance criteria for P217 are met through a core burnup of 13 Gwd/Mtu. This analysis was performed using the NRC approved methodology.

Additionally, analysis was performed using the methods in the Topical, but including water entrainment. This analysis shows sufficient margin to the DNBR acceptance criteria to assure that the PCT limit is not approached throughout core life. This conclusion applies to all cores to date for which a specific PCT analysis was not performed.

Therefore the health and safety of the public was not affected by this event.

This event is reportable pursuant to 10CFR50.73(a)(2)(ii)(B) since analysis using the approved Topical does not show that the design acceptance criterion of 2750 °F for the MSLB accident PCT is met for the entire core cycle.

CORRECTIVE ACTION

Analysis was performed that shows that all acceptance criteria for P217 are met through a core burnup of 13 Gwd/Mtu. This analysis was performed using the NRC approved methodology. The safety evaluation for modification 94L434, the modification that installed P217, has been revised to reflect this. Administrative controls have been issued that prohibit plant operation beyond 13 Gwd/Mtu. The administrative controls may be modified or removed as further analysis indicates.

Additionally, analysis was performed using the methods in the Topical, but including water entrainment. While the Topical does not include entrainment, this technique is used elsewhere in the industry for core

analysis and credit is taken for it in the Prairie Island USAR for containment analysis for the MSLB. The entrainment values used were obtained from Westinghouse document WCAP 8822. These analyses also used conservative uncertainties obtained from WCAP 8822. This analysis shows sufficient margin to the DNBR acceptance criteria to assure that the PCT limit is not approached throughout core life. This conclusion applies to all cores to date for which a specific PCT analysis was not performed. Because this analysis is outside the Topical, it cannot be used for final verification of core design parameters until it is formally reconciled with the Topical.

Prior to exceeding 13 Gwd/Mtu, an analysis will be performed to assure all design acceptance criteria are met beyond 13 Gwd/Mtu.

The Nuclear Analysis Department plans to:

- *develop a summary check sheet, identifying each and every acceptance criteria, the results of the analysis performed or the justification for not needing to do an analysis;*
- *develop more detailed procedures* (While the nature of the analysis process may preclude the use of procedures having considerable detail, guidelines should be developed that would enable less experienced personnel to perform or verify an RSE); *and*
- *improve the consultant design review process.*

<u>FAILED COMPONENT IDENTIFICATION</u>

There were no failed components.

<u>PREVIOUS SIMILAR EVENTS</u>

A previous event involving a core analysis error has been reported as Unit 1 LER 90-011.