Nebraska Public Power District

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NLS950243

December 21, 1995

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Dear Sir:

Cooper Nuclear Station Licensee Event Report 95-018 is forwarded as an attachment to this letter.

Sincerely,

J. H. Mueller
Site Manager

/nr

Attachment

cc: L. J. Callan
G. R. Horn
J. T. Herron
R. G. Jones
R. A. Sessoms
K. C. Walden
N. E. Champlin
INPO Records Center
NRC Resident Inspector
W. Turnbull
CNS Training
CNS Quality Assurance

200096

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

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.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
COOPER NUCLEAR STATION	05000298	1 OF 4

TITLE (4)

Maintenance Activity That Could Compromise a Steam Tunnel Blowout Panel Design Function During a High Energy Line Break Outside Containment.

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
11	21	95	95	-- 018 --	00	12	21	95	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N
POWER LEVEL (10)	000

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)	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)	X	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)
William R. Victor, Licensing and Compliance Specialist	(402) 825-3811

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
X	YES (If yes, complete EXPECTED SUBMISSION DATE).		NO		3	15	96

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

On November 21, 1995, it was recognized that the accident analysis for High Energy Line Break (HELB) outside Primary Containment contained a key assumption that had been rendered invalid. For a postulated HELB inside the Reactor Building Main Steam Tunnel (Steam Tunnel), the Steam Tunnel blowout panels had been credited with rupturing to allow a vent path to the Turbine Building, thereby preventing compartment over-pressurization. In an effort to reduce Secondary Containment leakage, maintenance was performed on the blowout panels in 1985 to cover them with fiberglass. As a result, the blowout panels would rupture at a higher pressure than credited in the HELB analysis.

The cause of this condition is due to management/quality assurance deficiencies [NUREG-1022 CAUSE CODE E]. The programmatic controls governing maintenance activities were insufficient, in this case, to prevent plant configurational changes that could challenge design basis assumptions. The corrective actions are to restore the blowout panels to their required configuration, validate that current work control processes would prevent this type of discrepancy today, and perform a selected maintenance history review to determine if other similar examples currently exist. A supplemental LER will be submitted to more fully describe the consequences of this condition following the completion of a more detailed study presently in progress.

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

PLANT STATUS

The plant was in Cold Shutdown during a scheduled refueling outage (RE16) when this reportable condition was identified.

EVENT DESCRIPTION

On November 9, 1995, as a result of inquiries by the NRC Senior Resident Inspector, it was recognized that the design bases of the Steam Tunnel blowout panels [EIIS: PL] were not well documented. A Condition Report was generated to fully document the blowout panels' design bases requirements and to explore the basis for a fiberglass installation that had been noted. As a result of this research, it was identified that the blowout panels had been modified in 1985 to address Secondary Containment [NG] leakage concerns. On November 21, 1995, it was determined that the addition of the fiberglass would increase the rupture pressure of the panels beyond what was considered in the HELB analysis. This was communicated to the NRC that day by a 4-hour ENS notification.

The Cooper Nuclear Station (CNS) Main Steam System [SB] includes piping from the Main Steam Isolation Valves (MSIVs) [ISV] in the Steam Tunnel (located in the Reactor Building outside Primary Containment [NH]) downstream to various power-operated valves [V] in the Turbine Building. A wall divides the Steam Tunnel from the Turbine Building [NM], through which the Main Steam lines pass (this wall is also a Secondary Containment boundary). The blockouts through which these and other lines penetrate this wall are sealed using light-weight cellular concrete applied to a specified thickness. These sealed areas constitute the blowout panels. Should a HELB occur in the Steam Tunnel downstream of the MSIVs, the blowout panels are credited with rupturing to create a relief path to the Turbine Building. In this manner, the design pressure of the Steam Tunnel will not be exceeded.

In 1985, a fiberglass cover was affixed to the wall, covering the blowout panel areas. This was performed as a maintenance activity requested by Plant Engineering to improve Secondary Containment leakage performance. However, this installation unknowingly increased the strength of the blowout area and, consequently, increased the pressure required to rupture the panels. Recent computations indicate that the blowout panels' rupture pressure would exceed the peak pressure calculated for a Main Steam Line break in the Steam Tunnel. This condition would result in additional unanalyzed loadings and environmental effects on affected safety-related equipment.

SAFETY SIGNIFICANCE

An Engineering study, with an estimated completion date of February 29, 1996, is being performed which will resolve the following safety implications of this unanalyzed condition:

1. The design pressure of the Steam Tunnel is 15 psi. It is not yet known what the peak pressure would have been in the Steam Tunnel given a HELB scenario with the fiberglass in place. However, if the peak pressure had exceeded the Steam Tunnel design pressure, this could have caused structural degradation of the Steam Tunnel.

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

SAFETY SIGNIFICANCE (continued)

2. Environmentally qualified electrical equipment is located outside the entrance to the Steam Tunnel. Their qualification is based on environmental conditions from a more limiting accident (HPCI line break in the Reactor Building) than the analyzed HELB resulting from a main steam line break in the Steam Tunnel. However, assuming an increased blowout panel rupture pressure, there would be an as yet unquantified increase in mass flow through the Steam Tunnel entrance. The higher ambient temperatures may challenge the qualification of this equipment. The performance of the safety-related functions of qualified components inside the Steam Tunnel has been judged to be unaffected by this condition.

3. An air passage connects the Steam Tunnel with the annulus between the Primary Containment liner and the outer concrete shell. Over-pressurization of the Steam Tunnel would cause the design pressure of the metal liner to be exceeded over a small localized area. Although Primary Containment Integrity is not credited in mitigating the radiological consequences of this accident, damage to the liner is contrary to the CNS licensing basis since the Atomic Energy Commission required license applicants to demonstrate that there were no adverse effects on the Primary Containment Structure due to a HELB outside Primary Containment. While further analyses will resolve this concern quantitatively, it currently appears more likely that local deformation would occur, rather than a rupture of the liner.

From a radiological release standpoint, the most limiting HELB outside Primary Containment is postulated to occur in the Turbine Building, rather than in the Steam Tunnel. Accordingly, evaluations are being made which will determine if the licensing basis HELB would still have been bounding. However, risk studies suggest that the probability of a more limiting Steam Tunnel HELB accident initiator is very low given that: a) a catastrophic failure must occur versus a more likely leak-before-break scenario, b) the four 24" Main Steam lines are the only ones with the potential of exceeding the Steam Tunnel design pressure upon a line break, c) the at-risk piping length is short (from the outboard MSIV to the blowout panels), and d) the initiator was possible only during those periods of Reactor at-power operation occurring since the 1985 fiberglass installation was completed. A supplemental LER will be submitted to describe more fully the consequences of this condition following the completion of a more detailed study which is currently in progress.

CAUSE

The condition was caused by the failure of the processes that were in effect in 1985 to ensure proper control of plant configurational changes that could affect the CNS design basis. This was manifested by: a) the ability to perform work such as the fiberglass installation as a maintenance activity rather than via the design change process and b) the lack of readily available design basis information regarding the safety functions of the blowout panels.

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

CORRECTIVE ACTION

The following corrective actions have been taken:

1. The bulk of the fiberglass has been removed from the blowout panels. This has restored the panels' ability to perform their design basis HELB function. The leakage that had been experienced in 1985 was corrected with an appropriate sealant under a Minor Modification and testing has confirmed that their Secondary Containment Integrity function has still been maintained. No other blowout panels with similar design characteristics have been identified.

2. The prior ability to perform de facto station modifications without using the design change process has been reviewed. The following points are salient to the barriers that are in place today:

   a. Management's expectations and the CNS safety culture have greatly improved since the 1994 forced outage. Proposed maintenance activities such as this would be reviewed with a more questioning attitude than existed in 1985.

   b. Improvements to the work control process have been put in place since 1985 when this condition occurred. These improvements have resulted in more structure with defined lines of responsibility, including inter-departmental review in the Maintenance Work Request generation and closure process.

   c. As part of the CNS Engineering reorganization that occurred in 1995, the Design Engineering Department (DED) has been moved to the site. As conservator of the CNS design basis, DED's direct onsite involvement has greatly improved the Station's ability to understand when a planned configurational change affects the design of the plant.

The following corrective actions will be taken:

1. Station procedures will be enhanced to provide a specific criterion in the review of MWRs whether the proposed maintenance activity constitutes a station modification, which requires implementation under a design document.

2. To provide assurance that other previous maintenance activities have not unknowingly compromised plant design basis assumptions, a review of representative MWRs will be performed in accordance with a pre-established sampling plan. Further corrective action will be assessed based on the results of this review.

SIMILAR EVENTS

LER 94-011 Primary Containment Penetration Design and Testing Deficiencies Discovered During Design Basis Reconstitution Activities.

LER 95-013 Plant Procedural Requirements Inconsistent with Station Blackout Assumptions.

Correspondence No: NLS950243

The following table identifies those actions committed to by the District in this document. Any other actions discussed in the submittal represent intended or planned actions by the District. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
Station procedures will be enhanced to provide a specific criterion in the review of MWRs whether the proposed maintenance activity constitutes a station modification, which requires implementation under a design document.	None
A review of representative MWRs will be performed in accordance with a pre-established sampling plan [to provide assurance that previous maintenance activities have not unknowingly compromised plant design basis assumptions].	None
Further corrective action will be assessed based on the results of this review.	None
A supplementary LER will be submitted to describe more fully the consequences of this condition following the completion of a more detailed study which is currently in progress.	None