Nebraska Public Power District

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NLS950107

May 15, 1995

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

Dear Sir:

Cooper Nuclear Station Licensee Event Report 95-010 is forwarded as an attachment to this letter.

Sincerely,

J. T. Herron
Plant Manager

/nr

Attachment

cc: L. J. Callan
G. R. Horn
J. H. Mueller
R. G. Jones
R. A. Sessoms
K. C. Walden
R. L. Koch
INPO Records Center
NRC Resident Inspector
R. J. Singer
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IE22 1/1

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

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.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) RHR Minimum Flow Valve Position vs. Design Basis Requirements

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	13	95	95	-- 010 --	00	05	15	95	FACILITY NAME	DOCKET NUMBER

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

OPERATING MODE (9)	N				
POWER LEVEL (10)	100				
		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 (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)	
		20.405(a)(1)(iv)	X 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	TELEPHONE NUMBER (Include Area Code)
Art Alford, Senior Staff Nuclear Licensing & Safety Eng.	(402) 825-3[illegible]

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)

Prior to the Residual Heat Removal (RHR) minimum flow valve design change (DC) 94-332, scenarios existed where the RHR system may not have met the Loss-of-Coolant Accident (LOCA) design basis of the plant. DC 94-332 was installed in December 1994 and changed the position of the RHR minimum flow valves (RHR-MOV-MO16A/B) from normally closed to normally open. During closeout of this DC, an unanalyzed condition was discovered on April 13, 1995.

During a LOCA, without a Loss-of-Offsite Power (LOOP) and with a single failure of one electrical distribution bus, the minimum flow valve in the non-LOCA affected RHR loop would have been prevented from opening. With the minimum flow valve closed, RHR pump failure due to dead-heading may have occurred before Reactor Coolant System (RCS) pressure decreased sufficiently to allow coolant injection. Additionally, the postulated loss of one electrical distribution bus would also result in the loss of one of the two Core Spray (CS) pumps.

The current Cooper Nuclear Station (CNS) Emergency Core Cooling System (ECCS) LOCA analysis for a Reactor Recirculation (RR) discharge line break (NEDO 24045) requires injection from two independent ECCS pumps (e.g., both CS pumps or one CS and one RHR pump). For this scenario, CNS did not meet the design basis requirements in that it is postulated that only one CS pump would be available for core cooling.

Per NUREG-1022, the cause of this event is attributed to Design, Manufacturing, Construction/Installation (cause code B), specifically failure to appropriately integrate the design basis of the plant in a previous design change.

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

Plant Status

The plant was at 100 percent power at the time of discovery of the event.

Event Description

DC 94-332 was installed in December 1994 and changed the position of the RHR minimum flow valves (RHR-MOV-MO16A/B) from normally closed to normally open. During closeout of the DC an unanalyzed condition (based on the pre-DC 94-332 design) was discovered on April 13, 1995.

During an 80% RR discharge line break LOCA, without a LOOP and a single failure of one electrical distribution bus the minimum flow valve in the non-LOCA affected RHR loop would have been prevented from opening. With the minimum flow valve closed, pump failure due to dead-heading may have occurred before RCS pressure decreased sufficiently to allow coolant injection. Additionally, the postulated loss of one electrical distribution bus would result in the loss of one of the two CS pumps. The current CNS ECCS LOCA analysis for RR discharge line break requires injection from two independent ECCS pumps (e.g., both CS pumps or one CS and one RHR pump). Therefore, CNS did not meet the design basis requirements in that it is postulated that only one CS pump would be available for core cooling.

The RHR pump manufacturer (Sulzer Bingham) has stated that running the pumps dead-headed for up to 20 seconds will not result in any immediate damage. However, the vendor will not qualify the pumps for greater than 20 seconds. Thus, if the minimum flow valve had failed to open, the RHR pumps would have needed to start injecting into the vessel within 20 seconds after the pumps were at rated speed to ensure they would not be damaged due to dead-heading.

Normally the worst case accident involves a LOCA with a concurrent LOOP. In the scenario described above, the worst case accident is a LOCA without a LOOP. The reason for this is that the RHR pumps start earlier in the event, therefore, less time is available for the RCS pressure to decrease sufficiently to allow injection to commence before pump damage occurs.

The RHR pump is up to speed in approximately 13 seconds. With the addition of the 20 second pump manufacturer's criteria, the earliest an adverse dead-heading condition could have occurred is at about 33 seconds. This is less than the GE calculated time of 43 seconds for when the RHR pump would be able to inject into the vessel. The availability of the RHR pump, prior to DC 94-332, for an 80% RR discharge line break with the above scenario could not be guaranteed. (NOTE: Smaller breaks (i.e., less than 80%) are expected to take longer than 43 seconds for RCS depressurization. These smaller breaks have not been specifically analyzed for CNS).

Cause

The design basis of the plant was not appropriately integrated into a design change. Originally, the position of the RHR minimum flow valves was not a concern because the Low Pressure Coolant Injection (LPCI) loop selection control logic ensured that at least two RHR pumps would be available for injection. DC 76-2, LPCI Modification removed the LPCI

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loop selection control logic and the position of the RHR minimum flow valves became significant.

The RHR minimum flow valve normal standby line-up had been closed since initial criticality. A proper design review was not performed on DC 76-2 to verify that CNS was still within its design basis requirements with the LPCI loop selection control logic removed and the minimum flow valve normally closed.

Safety Significance

Prior to DC 94-332, a single failure of one electrical distribution bus could prevent the minimum flow valve in the non-LOCA affected RHR loop from opening. This condition may have led to pump breakdown and loss of that RHR loop as well. In this scenario only one CS pump would have been left for adequate core cooling following the LOCA. GE document, G-HPC-6-176, Preliminary Safety Evaluation For RHR Pump Bypass Valves states in part:

> "... a realistic analysis would indicate that any low pressure pump (LPCI or core spray) would provide adequate core cooling for any accident events".

Although CNS has not been specifically analyzed for the scenario presented in this evaluation, GE has analyzed similar vintage plants as CNS using the SAFR/GESTR computer program and found that one CS pump is sufficient for this type of accident. Therefore, it has been concluded that the safety significance of this event is minimal.

Corrective Action

CNS implemented DC 94-332 which changed the RHR minimum flow valves from normally closed to normally open.

CNS is currently in the process of a Design Basis Reconstitution Project. This project includes the development of Design Criteria Documents (DCDs). The use of these DCDs have been incorporated into the design change procedures to assist in the enhancement of the design change process. This allows for readily accessible design basis information during the development of design changes.

New DCs are reviewed against the design basis criteria.

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Similar Events

Recent similar events involving conditions outside of the plant's design basis include:

LER 94-002 - Unexpected Cycling of CS Pump Minimum Flow Valves During MOV and System Operability Testing Potentially Resulting in Pump Degradation & Loss Of System Redundancy

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

LER 94-016 - Noncompliance with 10 CFR 50 Appendix R, Inadequate Isolation of Diesel Generator Control Circuits

LER 94-018 - Reactor Core Isolation Cooling System Trip and Throttle Valve Design Deficiency due to the Reset Motor Being Powered by AC Instead of DC

LER 94-021 - Design Error That Allows Spurious DG Room HVAC Isolation During a Fire or Seismic Event

LER 94-028 - Design Error That Places the Ultimate Heat Sink in an Unanalyzed Condition During Design Low River Level Conditions

LER 94-034 - Emergency Lighting System Cannot Be Assured of Meeting 8 Hour Operation Requirement due to Design and Maintenance Deficiencies

LER 95-002 - Failure to Modify 14 Primary Containment Isolation Valves to Prevent Auto-opening Upon Resetting a Group 2 Isolation Signal

LIST OF NRC COMMITMENTS	ATTACHMENT 3

Correspondence No: NLS950107

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
None	