U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Cooper Nuclear Station	0 5 0 0 0 2 9 8	1 OF 0 4

TITLE (4)
ECCS Pump Compartment Cooler Power Supply Design Deficiency Could Have Prevented Adequate Containment Heat Removal

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 3	2 6	9 3	9 3	— 0 0 7 —	0 0	0 4	2 2	9 3		0 5 0 0 0
										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)	X	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)		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 (AREA CODE)
John R. Myers	4 0 2 8 2 5 - 3 8 1 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)

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

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

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

Efforts associated with the Design Basis Reconstitution Program for Cooper Nuclear Station (CNS) have identified a design discrepancy in the compartment cooling for the Residual Heat Removal (RHR) pumps. This discrepancy could potentially affect the ability to maintain sufficient Low Pressure Coolant Injection flow in the event of a LOCA in the Core Spray (CS) System. In the event of a CS line break and a loss of offsite power concurrent with the failure of one division of 4160 VAC emergency power, two RHR pumps would be available to mitigate the consequences of the LOCA. The loss of 4160 VAC emergency power would result in the loss of compartment cooling for one of the RHR pumps, potentially resulting in premature failure. Thus, only one RHR pump would remain operable to mitigate the consequences of the CS line break, a condition which is not within the current licensing basis. On March 26, at 12:10 pm, this condition was determined to be reportable. With the plant in cold shutdown, a CS line break is not a credible accident and, therefore, Technical Specification Limiting Conditions for Operation were not applicable.

This condition is the result of design errors which occurred during the implementation of the Low Pressure Coolant Injection loop select modification in 1976. Prior to startup from the refueling outage, modifications will be made to resolve the identified concerns.

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

A. Event Description

During ongoing efforts associated with the Nebraska Public Power District's Design Basis Reconstitution Program for Cooper Nuclear Station (CNS), a design discrepancy in the compartment cooling for the Residual Heat Removal (RHR) pumps was identified. This discrepancy involved the ability of the RHR pumps to continue to operate if compartment cooling is lost due to the failure of one division of 4160 VAC emergency power upon a design basis accident (DBA) involving the Core Spray (CS) system. On March 26, at 12:10 pm, an evaluation of the discrepancy was conducted by the Station Operations Review Committee (SORC), which determined that this condition was reportable. At the time, the plant was in a refueling outage with the Reactor defueled. In this condition, neither RHR nor CS were required to be operable, and, since a CS line break is not postulated with the Reactor depressurized, there were no immediate compensatory measures required.

DBA conditions include a loss-of-coolant accident (LOCA) with a concurrent loss-of-offsite power (LOOP) and a worst case single failure. The postulated worst case single failure is a loss of one division of 4160 VAC emergency power, since this causes the failure of two of the four RHR pumps, one of the two CS pumps, and loss of the compartment cooler in one of the RHR pump compartments. Two RHR pumps are located in each compartment. Power to one RHR pump in each compartment is from the Division 1 emergency power source, and power to the other if from Division 2. One compartment cooler is also supplied from the Division 1 emergency power source, and the other from Division 2. The loss of the compartment cooler has been postulated to result in high temperature in the compartment leading to the failure of an operating RHR pump after approximately ten minutes. A CS system LOCA initially results in the two RHR pumps injecting to the Reactor vessel without any core spray. Upon failure of one of the operating RHR pumps due to the high temperature in the compartment, only one RHR pump would be available to mitigate the consequences of the DBA. The single RHR pump is adequate to provide core cooling, but may not be sufficient to provide both core and containment cooling. The General Electric ECCS Performance/LOCA Analysis previously assumed and analyzed configuration was one Core spray pump and one RHR pump, or two RHR pumps. Thus, this condition is outside the licensing basis of CNS.

B. Plant Status

Shutdown for the 1993 Refueling Outage, with the Reactor defueled.

C. Basis for Report

A condition alone that could have prevented the fulfillment of the safety function of systems needed to mitigate the consequences of an accident, reportable in accordance with 10CFR50.73(a)(2)(v).

D. Cause

This condition is the result of design errors which occurred during the implementation of the Low Pressure Coolant Injection (LPCI) loop select modification in 1976. In response to changes in the ECCS rule, 10CFR50.46, in the early 1970s, Cooper Nuclear Station was required to perform a plant modification, commonly called the LPCI loop select modification, to satisfy the peak clad temperature requirements under DBA conditions. Prior to the modification, the two RHR pumps in each Reactor Building compartment were powered from one division of emergency power, which also powered the compartment cooler. The major changes of the LPCI loop select modification included cross-powering two of the RHR pumps and converting the LPCI injection valves and the Reactor Recirculation System discharge valves to 250 VDC power. The dependence on AC power of the compartment coolers and the effect of a loss of cooling on the ability of the pumps to continue operating was apparently not recognized by the Architect/Engineer nor Nebraska Public Power District personnel responsible for the design change.

As a part of the modification and in response to the new ECCS performance rules, a Single Failure Analysis was also conducted to identify the limiting component failures for both the Reactor Recirculation System suction and discharge line breaks. The Single Failure Analysis was performed by the Architect/Engineer for Cooper Nuclear Station in 1976. This analysis also did not consider the potential consequences of the loss of AC power on the compartment coolers and the resultant effect on the pumps.

E. Safety Significance

The safety objective of the CS and RHR Systems is to provide a source of water to provide core and containment cooling in the event of a LOCA. These systems were intended to perform this task with a LOOP and worst possible single failure. For a CS system LOCA under these conditions, the loss of compartment cooling could potentially result in the subsequent failure of one of the two operational RHR pumps. Operating procedures direct that the RHR system be aligned to use the RHR Heat Exchanger to provide cooling for the water being injected into the Reactor vessel. Review of this condition by General Electric indicates the single RHR pump would be adequate to maintain the peak clad temperature within the requirements of 10CFR50.46. Both RHR pumps would operate for the initial 10 minutes of the transient, when peak clad temperature concerns are greatest. Additionally, the calculations for compartment heatup were conservative, not accounting for structural heat capacity and ventilation available due to piping penetrations allowing airflow to other areas within the Reactor Building, thus the 10 minute time limit is very conservative.

E. Safety Significance (Continued)

The analysis for Primary Containment is based on maintaining the flow from one RHR pump in Torus Cooling to maintain containment temperature below 200 degrees Fahrenheit. As part of the response to a DBA, the Emergency Operating Procedures require that operators maintain cognizance of Primary Containment temperature. LPCI flow is diverted to Torus cooling or spray as required to maintain containment temperature and pressure within an appropriate range. If containment integrity were threatened by high temperature or pressure, LPCI flow would be diverted to ensure containment integrity was maintained. A backup water source to supply the Reactor vessel with cold water is available from the Control Rod Drive and Service Water systems. Together with the RHR pump in Torus Cooling, this may have provided sufficient cooling to mitigate the consequences of the CS line break, although the condition is not within the licensing basis of the plant.

F. Safety Implications

The event which must occur in order for the compartment cooling to be of concern is a CS LOCA with a concurrent LOOP and a simultaneous failure of one 4160 VAC emergency power source. The probability of these three unrelated events occurring simultaneously is extremely remote (less than 10^{-8} per year of reactor operation).

G. Corrective Action

At the time this condition was identified, the plant was in a refueling outage with the Reactor defueled. In this condition, neither RHR nor CS were required to be operable, and, since a CS line break is not postulated with the Reactor depressurized, there were no immediate compensatory measures required. Prior to startup from the refueling outage, modifications are planned to provide natural convection cooling for the equipment in these compartments by replacing the solid equipment hatches with grating. Analyses are being performed to ensure that fire protection, environmental qualification, radiological, and flooding issues are addressed.

H. Similar Events

LER 92-013, Error in Limiting Single Failure Assumptions for the ECCS Performance Analysis, discusses another condition related to insufficient design review for the LPCI loop selection modification.