NRC FORM 366 (7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: [| | | | |] (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01 | N E F C S 1 (2) | 0 0 0 0 0 0 0 0 0 0 0 (3) | 4 1 1 1 1 (4) | (5)
LICENSEE CODE | LICENSE NUMBER | LICENSE TYPE | CAT

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01 | REPORT SOURCE L (6) | 0 5 0 0 0 2 8 5 (7) | 0 4 2 8 8 1 (8) | 0 5 0 3 8 1 (9)
DOCKET NUMBER | EVENT DATE | REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 During normal operations at approximately 45% power, all 125 volt DC power feeding bus
03 AI-41B was lost. The main steam isolation valves (HCV-1041A/HCV-1042A) failed closed
04 upon this loss of control power to AI-41B. These valves failing closed initiated an
05 EHC turbine trip, which in turn, caused a reactor trip. The loss of DC power would not
06 have affected the ability of the plant to safely shutdown should an accident have oc-
07 curred since redundant safeguards equipment was available and operable on DC panel
08 AI-41A.

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
09	E C (11)	X (12)	Z (13)	C K T B R K (14)	E (15)	Z (16)

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	8 1	—	0 0 3	/	0 1	T	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
A (18)	X (19)	A (20)	C (21)	0 0 5 3	Y (23)	N (24)	X (25)	I 2 0 3 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 One lead of the cable found on the distribution side of the 125 VDC, 100 amp switch (which feeds bus AI-41B) was found to be burnt loose from the switch completely. It
11 is believed that this cable had been separated due to the heat caused by loose con-
12 nection between the cable and the switch landing lug. The damaged section of cable
13 was replaced as was the 100 amp switch. All lugs on the entire DC bus breaker panels,
14 EE-8F and EE-8G were checked to ensure proper tightness.

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
15	E (28)	0 4 5 (29)	NA	A (31)	via turbine & reactor trips

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
16	Z (33)	Z (34)	NA	NA

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
17	0 0 0 (37)	Z (38)	NA

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
18	0 0 0 (40)	NA

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
19	Z (42)	NA

PUBLICITY

	ISSUED	DESCRIPTION (45)	NRC USE ONLY
20	N (44)	NA	

8105270214

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LER No. 81-003
Omaha Public Power District
Fort Calhoun Station Unit No. 1
Docket No. 05000285

ATTACHMENT NO. 1

Safety Analysis

The Fort Calhoun Station Unit No. 1 is so designed that no single failure can prevent the safe shutdown of the plant. The failure of the 125 volt DC power to panel AI41B would not have prevented the complete and proper loading or operation of essential safeguards equipment should the need have arisen. The loss of 125 volt DC power to panel AI-41B can be considered as a very conservative instance of the total loss of 125 VDC to DC bus #2. Total loss of 125 volt DC power to DC bus #2 has previously been analyzed per the Final Safety Analysis Report (refer to sections 7.3.5 and 8.3.4) as being: a single failure resulting in loss of control function redundancy but having no effect on safeguards systems performance.

The loss of 125 VDC power at AI-41B caused several significant events. These events, as well as an analysis on their relation to a safe shutdown of the plant in case of an accident, are listed below:

1. The component cooling water system valves feeding the SI tank leakage coolers, the Nuclear Detector Well Cooling units and the RC pump seals and bearing coolers closed as designed. In all cases where component cooling water feeding safeguards equipment was involved, all equipment (normally fed from AI-41B) acted as designed and attained the proper safety position. The Raw Water valves fed from AI-41B attained the fail-safe position and were properly aligned should Raw Water have been needed as a backup to the component cooling water system feeding the Safety Injection and Containment Spray Pump bearing coolers, the Containment Air Cooling and and Filtering Units, the Shutdown Cooling Heat Exchangers and the Control A/C units.
2. Although the Nuclear Detector Well Cooling System is not required to be operable should a shutdown be deemed necessary, Technical Specification 2.13 states that the "annulus exit temperature from the nuclear detector cooling system shall not exceed a temperature found to correlate to 150°F concrete temperature." Contrary to this technical specification, the loss of component cooling water to the nuclear detector well cooling system caused a rise in concrete temperature of part of the biological shield to 170°F. This rise to 170°F from the normal 120°F temperature was caused due to the loss of component cooling water corresponding to a period of 1.33 hours. Therefore, using two conservative assumptions, i.e.,
   a. The temperature was at 170°F for the full 1.33 hours rather than rising from 120°F to 170°F over the 1.33 hour period, and
   b. the thin film convection heat transfer coefficient (h) has been postulated in the analysis to tend to infinity (∞), whereas in actuality it is probably in the range of 10-20 even in rapid thermal transfer processes.

2. continued

An analysis was performed in an effort to assess possible concrete damage caused by the temperature rise to 170°F. The results of this analysis showed that the temperature of the biological shield concrete would drop to below 150°F (for a surface temperature of 170°F) at less than 1.5 inches into the concrete. It is OPPD's engineering judgement that the conditions described above could not create any detrimental effect or deterioration of the biological shield concrete. It is worthwhile to note that the duration time of 1.33 hours came about as the time it took the CCW system feeding the Nuclear Detector Well Coolers to be restored to fully operable status.

3. Although control power for certain safeguards equipment was lost for a period of approximately 20 minutes, all redundant equipment was available and operable as designed. Both diesel generators were verified/operated automatically as a result of the turbine/plant trip. The DC sequencer circuitry corresponding to Sequencer S2-1 was unavailable for a time period of 20 mintues (the length of time it took to return power to AI-41B via the manual transfer switch which allowed AI-41B to be fed from DC bus #1). However, during this 20 minute interval, the AC sequencer circuitry associated with the "B" safeguards train, Sequencer S2-2 as well as the AC and DC sequencer circuitry associated with the "A" safeguards train, sequencer S1-1 (DC) and S1-2 (AC), were available and capable of properly sequencing safeguards loads should the need have arisen.

4. Numerous containment isolation valves, radwaste system isolation valves, as well as safety injection actuation signal actuated valves, attained their fail-safe positions as designed upon de-energization of AI-41B.

It is important to note that the loss of 125 DC power to AI-41B lasted for a period of 20 mintues, at which time the manual transfer switch was operated and 125 volt DC power was returned to AI-41B via 125 volt DC bus #1. Upon re-energization all safeguards related circuits were restored to immediate operability with the exception of the CCW system which required an additional one hour to restore to the fully operable state due to lost inventory.

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ATTACHMENT NO. 2

Corrective Action

It was determined that a loose connection on one of the cables which extends from the 125 VDC panel switch to the transfer switch which feeds AI-41B, had caused the switch lug to be completely disconnected from the cable due to heat. As a result, both the switch itself (an ITE EH2-S100) as well as a section of damaged cable, were irreparable and were replaced with spares. All lugs on both DC distribution panels (EE-8F-fed from the 125 VDC bus #1 and EE-8G-fed from the 125 VDC bus #2) were checked for adequate tightness and adjusted if necessary to ensure a "good connection" between cables and lugs. The system was then returned to normal, i.e. the AI-41B bus feed was returned to DC bus #2.

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ATTACHMENT NO. 3

Failure Data

This is the first failure of this type to occur at Fort Calhoun Station Unit No. 1.