Nebraska Public Power District

COOPER NUCLEAR STATION
P.O. BOX 98, BROWNVILLE, NEBRASKA 68321
TELEPHONE (402) 825-3811

CNSS895810

July 7, 1989

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

Dear Sir:

Cooper Nuclear Station Licensee Event Report 89-021 is being forwarded as an attachment to this letter.

Sincerely,

G. R. Horn
Division Manager of
Nuclear Operations
Cooper Nuclear Station

GRH:sg

Attachment

cc: R. D. Martin
L. G. Kuncl
R. E. Wilbur
V. L. Wolstenholm
C. A. Trevors
INPO Records Center
ANI Library
NRC Resident Inspector
R. J. Singer
CNS Training
CNS Quality Assurance

IE22
1/1

NRC Form 366
(9-83)

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 8

TITLE (4) Undocumented Wiring Configurations Associated with Safety Related Equipment Discovered during Design Change Activities.

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 5 0 0 0
0 6	0 2	8 9	8 9	— 0 2 1	0 0	0 7	0 7	8 9		0 5 0 0 0

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 0 0	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)	X 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
D. L. Reeves, Jr.	AREA CODE 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)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

On April 25, 1989, during implementation of Design Change (DC) 87-190 associated with installation of limit switches to comply with Regulatory Guide (R.G.) 1.97 related requirements, an undocumented wire type was discovered on components that were previously believed to be fully qualified. As a result of additional preliminary inspections and evaluations, an inspection procedure was developed to identify the potential generic scope of this deficiency. On June 2, 1989, it was determined and confirmed that other deficiencies (non-documented splices on pigtails to internal rack wiring) existed. These deficiencies were associated only with racks provided by General Electric (GE), and involved Barton switches, their respective terminal blocks and short runs (less than 10 feet) of cable.

The root cause of the undocumented wiring configurations is the existence of unexpected and non-obvious GE supplied equipment configurations that were not discovered by the District during prior inspections.

Corrective action taken included removal of all wire with the undocumented wiring configurations and replacement with continuous lengths of fully qualified cable. Additionally, the District is pursuing, through GE, the potential reportability of this situation in accordance with 10CFR21.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Cooper Nuclear Station	0 5 0 0 0 2 9 8	8 9	— 0 2 1	— 0 0	0 2 OF 0 8

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

A. EVENT DESCRIPTION

Following the plant shutdown on April 8, 1989, for the 1989 Refueling/Maintenance Outage, Design Change (DC) 87-190, associated with the installation of limit switches to comply with Regulatory Guide (R.G.) 1.97 related requirements, was initiated. Nebraska Public Power District had previously committed to meeting all R.G. 1.97 requirements prior to startup from the 1989 Refueling Outage. On April 25, 1989, while replacing limit switches, an undocumented wiring type was discovered on components that were previously believed to be fully qualified.

As a result of further inspections and evaluations, a detailed inspection procedure was developed to identify the scope of the noted deficiency as it related to similar components which were considered to be fully qualified. Since most other configurations had previously been walked down, it was determined that this unexpected configuration was limited to local wiring from rack mounted equipment to its local terminal box/block. Accordingly, a walkdown of approximately 120 components was initiated on May 25, 1989. The walkdown included the inspection of each installation, including its terminal strip, the connecting conduit, and the interface wire within the conduit. In order to fully inspect this wire, all local (within 10 feet of component) condulets were opened.

As a result of the inspection of the 120 components, the scope of concern was narrowed to EQ classified Barton switches. Specifically, the walkdown resulted in the identification of a deficiency associated with rack-mounted Barton switch installations provided by General Electric (GE) (and its sub-contractors), wherein an instrument and terminal block with a short cable-run (less than 10 feet) were involved. These configurations, which were not identified in vendor drawings, were inconsistent with EQ qualified installation requirements at Cooper Nuclear Station. The configurations included the use of crimp style wirenuts (Blind Barrel), Scotch tape splices, and butt crimps. Of the EQ classified Barton switches, 28 were potentially in an unqualified configuration. Table 1 lists the as-found condition of the EQ classified Barton switches with non-documented configurations.

B. PLANT STATUS

At the time of the discovery of the condition, the plant was in a cold shutdown condition. The plant had been shut down for the 1989 Refueling/ Maintenance Outage on April 8, 1989.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Cooper Nuclear Station	0 5 0 0 0 2 9 8	8 9	0 2 1	0 0	0 3 OF 0 8

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

C. BASIS FOR REPORT

These conditions, documented on June 2, 1989, were assessed for operability and potential reportability considering the guidance provided in Generic Letter 88-07 of April 7, 1988. Accordingly, on June 5, 1989, the decision was made to initiate the notification and reporting processes required by 10CFR50.72 and 10CFR50.73. The conditions that were found are considered reportable in accordance with 10CFR50.73(a)(2)(i)(B) due to the indeterminate qualification status of certain instrumentation. This instrumentation was conservatively classified as inoperable, and as such, the plant may have previously been operated contrary to Technical Specifications.

Additionally, this condition is also considered to be reportable in accordance with 10CFR50.73(a)(2)(v) in that due to indeterminate qualification of certain switch wiring configurations during prior plant operation, fulfillment of a safety function could have been prevented. This determination, however, is conservative in that the District believes that notwithstanding qualification status, the intended safety function generally would have been performed (see Section F, Safety Implication, discussion).

D. CAUSE

During the plant walkdown of local instrument racks and associated cables referenced in Section A, potentially unqualified splices were found in conduit running from rack-mounted Barton switches to the associated terminal box mounted on the same rack. These conduits are all less than 10 feet in length, and most are less than 5 feet. Previous plant walkdowns had verified that qualified cables were present at the rack-mounted terminal box. Based on the short run of conduit to the instruments and the fact that the EQ integrity of the instrument would had to have been broken to inspect the cable termination at the instrument, a reasonable assumption had previously been made that qualified cable existed within the conduit from the terminal box to the switch or instrument.

These splices are not shown on any existing drawings for the instruments or the racks. In fact, the drawings for the instruments and racks reference a General Electric design specification for local instrument racks (209A4351). This specification states in Section 5.1.3.1. "Connections to external devices are to be on Terminal Blocks located in NEMA type 12 enclosed steel terminal cabinets." Therefore, there was no reason for the District to believe that splices existed inside the conduit.

*U.S. GPO: 1989-520-589/00070

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Cooper Nuclear Station	0 5 0 0 0 2 9 8	8 9	— 0 2 1	— 0 0	0 4 OF 0 8

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

E. SAFETY SIGNIFICANCE

None. At the time of discovery of these conditions, the plant was in a cold shutdown condition for scheduled refueling and maintenance. All instrumentation affected by the unqualified or non-documented connections was evaluated, considering potential accident scenarios for existing plant conditions. While shutdown, there are no scenarios that could result in this instrumentation being in a harsh temperature, radiation, pressure or humidity (including steam moisture) environment. Accordingly, the affected instruments need not be qualified for operation in harsh environments during shutdown conditions. Therefore, they were operable for all shutdown mode activities.

F. SAFETY IMPLICATION

All instrumentation affected by the potentially unqualified connectors was evaluated for operability during past plant operation. All of the connectors were internal to the conduit. As a result, based on qualitative engineering judgement, there is reasonable assurance of operability for harsh temperature, radiation and pressure environments. Accordingly, operability was specifically assessed for postulated harsh steam/moisture conditions. The following identifies the relevant switches and summarizes the safety implication associated with each.

- NBI-LIS-101A and B (Insulated Butt Crimps)
  NBI-LIS-101C and D (Crimp Style Wirenuts)

The function of this instrumentation is to monitor and provide trips associated with reactor water level (i.e., on low water level, initiates SCRAM and Group 2, 3 and 6 isolation functions; on high level, trips both High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems. Group 2, 3 and 6 Isolations include Primary Containment, Reactor Water Cleanup (RWCU) and Secondary Containment, including Standby Gas Treatment (SGT) System actuation.

The switches could be exposed to a steam environment upon a postulated HPCI steam line break. Upon a HPCI steam line break, HPCI would become inoperable and the instrument function for HPCI trip on high level would not be required. Assuming a HPCI steam line break, if a postulated Loss of Offsite Power (LOOP) condition were also assumed (the analyzed worse case scenario), a plant SCRAM would occur along with Group 2, 3 and 6 Isolations, regardless of operability of these switches. However, if no LOOP were assumed, the SCRAM function, along with the Group 2, 3 and 6 isolation functions, upon reaching a low Reactor Water level condition, could have been degraded. Likewise, the trip function for RCIC, on high water level, could also have been degraded.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Cooper Nuclear Station	0 5 0 0 0 2 9 8	8 9	— 0 2 1	— 0 0	0 5 OF 0 8

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

F. SAFETY IMPLICATION (Continued)

Approved alarm and emergency procedures exist which direct an operator to manually initiate required actions in the event that automatic functions do not occur as designed. Therefore, given this scenario, operator action would have been taken to achieve the desired action.

° NBI-DPIS-52B (Crimp Style Wirenuts)
NBI-DPIS-52D (Unidentified Scotch Tape Splices)

These are injection valve permissive switches for Residual Heat Removal (RHR) in the Low Pressure Coolant Injection (LPCI) mode of operation and Core Spray (CS). The switches could have potentially been exposed to a steam environment upon a HPCI steam line break. Assuming a worse case scenario in which both switches failed, the A and B loop injection valves for both LPCI and CS could potentially open too soon. If it is further assumed that one of the injection check valves had leaked (a passive failure), one loop of RHR or CS could have been overpressurized. Even if this overpressure condition had caused the loop to fail, it would not have resulted in any unevaluated flooding concerns. Even if such a scenario had occurred, three out of the four loops would have remained operable. The safety implications are, therefore, minimal. If no assumption is made regarding a check valve leak, then there would have been no effect on either the RHR/LPCI or CS systems. Injection would have occurred in this latter instance, when reactor pressure had decreased to the minimum flow pressure of the respective system.

° HPCI-FIS-78 (Crimp Style Wirenuts)

As with all of the Barton switches noted on Table 1, there were no implications regarding operability except from a harsh steam/moisture environment. This particular switch monitors flow from the HPCI pump discharge. There is no postulated steam environment to which this switch would be subjected except for a HPCI steam line break. In the event a HPCI steam line break had occurred, the steam line temperature switches would have functioned to cause HPCI System isolation. Therefore, there would have been no safety implications regarding the operability of this switch.

° MS-DPIS-116A, B and C (Crimp Style Wirenuts)
MS-DPIS-117A, C and D (Crimp Style Wirenuts)
MS-DPIS-118C (Crimp Style Wirenuts)
MS-DPIS-119A, C and D (Crimp Style Wirenuts)
MS-DPIS-116D (Non-documented Scotch Tape Splices)
MS-DPIS-117B (Insulated Butt Crimps)
MS-DPIS-118A, B and D (Insulated Butt Crimps)
MS-DPIS-119B (Insulated Butt Crimps)

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Cooper Nuclear Station	0 5 0 0 0 2 9 8	8 9	— 0 2 1	— 0 0	0 6 OF 0 8

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

F. SAFETY IMPLICATION (Continued)

These instruments all monitor Main Steam Line (MSL) high flow (line break). The safety function of these switches is to initiate closure of the Main Steam Isolation Valves (MSIVs) upon a main steam line break. This safety function would be performed before steam from the break could migrate to the switches located outside of, and at a substantial distance from, the steam tunnel. No other steam environment is postulated for these instruments. Additionally, in the event of failure, the safety function would have been fulfilled by completely qualified and redundant main steam temperature switches, installed to monitor for breaks and which are located adjacent to the MSLs. Therefore, there would have been no safety implications regarding operability of these switches.

° PC-DPIS-516A and B (Unidentified Scotch Tape Splices)

These switches monitor Drywell to Torus DP and control the Torus to Drywell vacuum breakers. The switches are required to respond only to a Loss of Coolant Accident. Because they are located in the RHR Heat Exchanger Room, they are isolated from the LOCA Steam environment. Therefore, the splices would have had no operational impact on these switches.

° PC-PS-16 (Unidentified Scotch Tape Splices)

This instrumentation monitors drywell pressure. The splices could have potentially been exposed to a harsh steam environment in the event of a HPCI steam line break. However, no safety concern would exist because this instrument is not required to function to mitigate a HPCI steam line break. In the line break scenario, failure could have led to a spurious alarm, indicating high drywell pressure. However, other indications of actual drywell pressure would have been available.

° RCIC-DPIS-83 and 84 (Crimp Style Wirenuts)

This instrumentation is provided to monitor for a RCIC steam line break. There are no safety implications arising from the as found configurations, since the safety function of the associated instruments is performed by alternate and fully qualified temperature switches which would have caused RCIC to isolate.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Cooper Nuclear Station	0 5 0 0 0 2 9 8	8 9	— 0 2 1	— 0 0	0 7 OF 0 8

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

G. CORRECTIVE ACTIONS

As noted previously in Section A, Event Description, initial actions included a thorough analysis of the deficiencies to ensure that the extent of the problem had been addressed. Initially, this consisted of a walkdown of all EQ program Air Operated Valves (AOVs) for which additional R.G. 1.97 related limit switch wiring deficiencies were noted and subsequently corrected. Initially believing the problem to be isolated to indication circuits for these AOVs, action was taken to confirm this through performance of a random sampling of other wiring installations between instruments and local terminal boxes. Unexpectedly, questionable wire, later identified as instrument pigtails, was found. This then led to development of the procedure noted in Section A, which involved a walkdown of all rack-mounted installations with similar configurations, and which included all equipment that utilized a local terminal block with field run cable, to ensure no other similar configurations existed.

Subsequently, all questionable rack-mounted Barton switch wire runs from the terminal block to the switch were replaced with continuous lengths of fully qualified cable.

To provide further assurance that all unqualified local splices had been identified, a sampling of items that had not been included in the inspection procedure was walked down. This sampling did not identify any additional unqualified configurations. As a result of the above efforts, the conclusion has been reached that all undocumented EQ wire and splices in the local wiring has been identified and replaced. These efforts were all completed prior to plant startup from the 1989 Refueling/Maintenance Outage on June 16, 1989.

Finally, the District contacted GE regarding the as-found conditions for their assessment as a defect reportable in accordance with 10CFR21. Follow-up efforts in this regard are continuing.

H. SIMILAR EVENTS

None.

TABLE 1

BARTON LOCAL SPLICE WALKDOWN RESULTS

COMPONENT	CONNECTION	LOCATION
HPCI-FIS-78	Wirenuts (4)	LB 2" above back of switch
MS-DPIS-116A,B,C -117A,C,D -118C -119A,C,D	Wirenuts (3)	LB 6" below and behind switch
MS-DPIS-116D	Scotch Tape Splices (3)	LB 6" below and behind switch
MS-DPIS-117B -118A,B,D -119B	Insulated Butt Crimps (3)	LB 6" below and behind switch
NBI-LIS-101A,B	Insulated Butt Crimps (6)	LB 3" below switch
NBI-LIS-101C,D	Wirenuts (6)	LB 6" below switch
NBI-DPIS-52B	Wirenuts (6)	LB 6" behind switch
NBI-DPIS-52D	Scotch Tape Splices (4)	LB 10" behind switch
PC-DPIS-516A,B	Scotch Tape Splices (3)	LB 6" below switch
PC-PS-16	Scotch Tape Splices (6)	LB 6" below switch
RCIC-PS-83,84	Wirenuts (6)	LB 6" behind switch