Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247
402/636-2000

March 14, 1994
LIC-94-0061

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

Gentlemen:

Subject: Licensee Event Report 94-001 for the Fort Calhoun Station

Please find attached Licensee Event Report 94-001 dated March 14, 1994. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(iv), 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(ii)(B). If you should have any questions, please contact me.

Sincerely,

W. G. Gates

W. G. Gates
Vice President

WGG/jrg

Attachment

c: LeBoeuf, Lamb, Greene & MacRae
L. J. Callan, NRC Regional Administrator, Region IV
S. D. Bloom, NRC Project Manager
R. P. Mullikin, NRC Senior Resident Inspector
INPO Records Center

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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)
Fort Calhoun Station Unit No. 1	05000285	1 OF 12

TITLE (4)
Safeguards Actuation and Subsequent Reactor Trip Due to Relay Failure

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
02	11	94	94	-- 001 --	00	03	14	94		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1
POWER LEVEL (10): 100

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

	20.402(b)		20.405(c)	X	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
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	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)
Keith A. Voss, Shift Technical Advisor	(402) 533-6931

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
X	JE	74	G080	Y						

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)

On February 11, 1994 at 0340, an invalid Containment Pressure High Signal (CPHS) occurred. The CPHS initiated Engineered Safeguards actuations including safety injection actuation, containment isolation actuation, steam generator isolation, and starting of emergency diesel generators. Steam generator isolation also resulted in a turbine/reactor trip. The cause of the event was a coil-shorting failure of a supervisory relay, resulting in actuation of the CPHS lockout relay.

Post-event investigation determined that if a coil-shorting failure of certain other supervisory relays were to occur coincident with certain design basis accidents, the result could involve premature actuation of lockout relays associated with the Recirculation Actuation Signal (RAS). It was concluded that these concerns represented conditions outside the design basis of the plant. The cause of the design basis concerns was determined to be failure of the original plant design to anticipate a potential failure mode.

The failed CPHS supervisory relay was replaced, and will be sent off-site for investigation of the failure mechanism. An engineering analysis was performed with respect to engineered safeguards supervisory relays. Eight supervisory relays have been disconnected to address the design basis concerns identified by the engineering analysis.

BACKGROUND

This LER discusses an event involving an invalid Containment Pressure High Signal (CPHS) (an Engineered Safeguards initiation signal), followed by a reactor trip. The Fort Calhoun Station (FCS) has two redundant trains ('A' and 'B') of Engineered Safeguards Controls and Instrumentation. Each train includes a panel that houses the relays, valve and pump controls, two sequencer panels, and an emergency diesel generator panel. When a measured parameter reaches the setpoint for initiation of a safeguards signal on the required number of instrument channels (ordinarily two out of four), the initiating signal lockout relay trips, actuating the associated safeguards functions. Examples of safeguards initiating signals include CPHS, Pressurizer Pressure Low Signal (PPLS), and Safety Injection and Refueling Water Tank Low Signal (STLS). The initiating signal lockout relay must be manually reset after the initiating logic has cleared.

Each Engineered Safeguards Controls and Instrumentation train includes prime initiation relays, prime actuation relays, derived initiation relays and derived actuation relays. To improve system reliability, the engineered safeguards prime initiation signals are designed to initiate a derived initiation signal on the opposite train. As a result, a prime initiation signal on either train results in actuation of both trains. The derived signal to the opposite train may be blocked by using the derived signal cutoff switch for the train that is generating the prime signal.

Means are provided to supervise the operability of the safeguards initiation/actuation circuits (see Figure 1). A supervisory relay is provided to monitor each initiation/actuation signal lockout relay circuit. The supervisory relay will become de-energized and actuate a trouble annunciator in the event of an open circuit, a loss of power, or completion of initiation logic without a corresponding lockout relay trip.

Safeguards actuation signals are generated from logical combinations of initiation signals. A CPHS results in generation of several safeguards actuation signals. Specifically, a CPHS initiates a Steam Generator Isolation Signal (SGIS), a Safety Injection Actuation Signal (SIAS), a Containment Isolation Actuation Signal (CIAS), auto-starting of the Emergency Diesel Generators, and sequential starting of engineered safeguards equipment.

SGIS closes valves to isolate main steam flow from, and main feedwater flow to the steam generators, to reduce an uncontrolled cooldown of the Reactor Coolant System (RCS). SGIS will close the main steam isolation valves (MSIVs), the main steam bypass valves, and the main feedwater isolation valves. SGIS can also indirectly initiate a turbine/reactor trip. When both MSIVs are on their closed seats, a turbine trip is initiated to protect the turbine from the effect of motoring the generator. When a turbine trip is initiated, a reactor trip is also initiated.

SIAS automatically actuates safety injection in the event of a Loss of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), or a Steam Generator Tube Rupture (SGTR). This signal will open the High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) loop injection valves, close the safety injection leakage cooler outlet pressure control valves, trip the Ventilation Isolation Actuation Signal (VIAS) lockout relays, and initiate load shedding of certain non-essential loads. SIAS also realigns certain Component Cooling Water (CCW) valves for the Safety Injection pumps and opens the CCW inlet and outlet valves for the Raw Water/CCW heat exchangers. SIAS provides additional signals that did not impact this event.

CIAS closes the containment isolation valves for flow paths that are not required to control or mitigate the event, opens valves to allow cooling water to the containment coolers to limit the pressure transient, and secures CCW flow to unnecessary heat loads. Some of the control functions CIAS performs include: isolating letdown, opening the CCW valves to the containment coolers, and isolating CCW flow to the Reactor Coolant Pump (RCP) seal and oil coolers (HCV-438A/B/C/D) if CCW pressure drops below 60 psig (which can occur during load sequencing). The CIAS signal can be overridden using a test switch on the CIAS panels.

The emergency diesel generator start signal will start the diesel generators to ensure a power supply to vital loads is available. The diesel generator breaker protection is overridden under emergency conditions.

When load sequencing is initiated, the sequencers start safeguards pumps and fans sequentially to allow acceleration of loads on the off-site power source or the diesel generators. The loads started by the sequencers include: the raw water pumps, the CCW pumps, the Auxiliary Feedwater (AFW) pumps, the LPSI pumps, and the HPSI pumps. The sequencers send a start signal to the components at the appropriate time. This emergency start signal will stay in place until the safeguards signal is reset or the appropriate control switch is taken to pull out.

VIAS initiates closing of the containment pressure relief, air sample and purge system valves, if open, to minimize the potential for release of radioactivity from the containment.

The safety injection leakage cooler outlet pressure control valves (PCV-2909, 2929, 2949 and 2969) function to control pressure due to RCS leakage to the leakage coolers. During normal operation the system is designed so that a PCV will open at approximately 400 psig, discharging RCS leakage from an accumulator to the Reactor Coolant Drain Tank (RCDT). However, when a SIAS is present, the leakage cooler PCVs remain shut.

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

FCS has two steam generators that provide steam to the turbine via a main steam header. Each header, one per steam generator, has five spring loaded safety valves located between the containment penetration and the MSIV. The safety valves are each set to progressively open between 1000 and 1050 psia, to relieve main steam header pressure. As header pressure increases, additional safety valves will lift to relieve the pressure and allow heat to be removed from the RCS. Each header has one safety valve (MS-291 and MS-292) that has a pneumatic operator that also allows the valve to be manually opened from the Control Room, in addition to its automatic function which is pressure controlled. This allows the steam generators to be used to control RCS temperature if the MSIVs are closed or the condensers are not available.

EVENT DESCRIPTION

At 0330 on February 11, 1994, with the plant at 100% power, Operations personnel in the Control Room smelled an acrid odor apparently originating in the vicinity of the engineered safeguards panels. A visual inspection of the panels was performed and no smoke or physical abnormalities were observed.

At 0340, CPHS lockout relay 86B/CPHS actuated. This, in turn, initiated SIAS, CIAS, VIAS, SGIS and starting of the emergency diesel generators. As a result of the SGIS, the MSIVs closed, initiating a turbine trip, which in turn generated a reactor trip.

The Control Room operators saw that containment pressure was indicated as approximately 0.6 psig (compared to the Technical Specification setting limit of 5 psig for CPHS initiation) and determined that the CPHS was not valid. The operating crew entered Emergency Operating Procedure EOP-00, 'Standard Post-Trip Actions.' At 0342, the Assistant Plant Manager-Operations was notified of the event. At 0343, CCW was restored to the Reactor Coolant Pump Lube Oil and Seal Coolers by opening valves HCV-438A/B/C/D. Since the SGIS had isolated main feedwater to the Steam Generators, AFW flow was manually initiated at 0347 using Motor Driven AFW Pump FW-6 which was already running at this time. (The minimum Steam Generator level reached during the event was approximately 61% of wide range level.) At 0349, with letdown isolated by the safeguards actuation and pressurizer level increasing, the operators took manual control of pressurizer level and secured two charging pumps, as allowed by EOP-00.

At 0350, the crew transitioned to EOP-01, 'Reactor Trip Recovery,' after diagnostics indicated that an uncomplicated reactor trip had occurred. The crew also entered Abnormal Operating Procedure AOP-23, 'Reset of Engineered Safeguards,' in order to reset engineered safeguards initiated by the invalid CPHS. The operators attempted to reset CPHS lockout relay 86B/CPHS in accordance with AOP-23, but it would not stay reset; therefore the invalid CPHS persisted.

At 0354, a low level alarm for the Emergency Feedwater Storage Tank (EFWST) was received due to AFW flow to the steam generators. Technical Specification (TS) 2.0.1 was entered when EFWST level dropped below the amount specified in TS 2.5(2) (i.e., 55,000 gallons). At 0359, with pressurizer level at approximately 75%, the last charging pump was secured.

At 0400, the Shift Supervisor declared a Notification of Unusual Event (NOUE) based on Emergency Plan Implementing Procedure EPIP-OSC-1, Emergency Action Level 11.6 (i.e., Plant Conditions Warrant Increased Awareness by Plant Staff or Government Authorities). At 0406, the NRC Senior Resident Inspector was notified of the event.

At approximately 0412, following discussions between the Senior Reactor Operator and the Shift Supervisor, the operators deviated from the guidance provided in AOP-23 by taking the derived signal cutoff switch (CS-B1/SP-B) to OFF and the CIAS Test Switch (01-AI-43B) to TEST. This action was taken in order to allow the operators to begin to reset 'A' train engineered safeguards and restore letdown flow, even though they were not able to reset the 'B' train CPHS lockout relay. The operators then began resetting 'A' train lockout relays. Based on operator discussions, it was decided that one particular relay, 86B1/VIAS, did not need to be reset at that time. However, the other 'A' train derived signal lockout relays were reset. Removal of the 'A' train SIAS allowed Safety Injection Leakage Cooler Outlet Pressure Control Valves PCV-2909 and PCV-2949, which receive a close signal from SIAS, to return to normal operation. These valves subsequently opened and began to discharge water from the HPSI header to the RCDT header.

At 0413, the Control Room Communicator notified the State of Iowa and the counties of Harrison, Pottawattamie and Washington. The State of Nebraska did not respond to the Conference Operations (COP) network call. (Post event investigation showed that a problem with the COP network did not allow the call to be completed. See Corrective Actions for the resolution of this issue.)

At 0414, letdown flow from the RCS was restored, and at 0415, a charging pump was restarted. Also at 0415, alarms were received indicating high water level in the containment sump. Initially the source of the water was thought to be lifting of a CCW relief valve. Further investigation ruled out CCW as the source of the water.

Following another unsuccessful attempt to notify the State of Nebraska via the COP network, at 0427, the State of Nebraska was successfully notified of the NOUE via the backup method, using a commercial telephone line. At 0428, the Assistant Plant Manager - Operations notified the NRC Operations Center of the NOUE, pursuant to 10 CFR 50.72(a)(1)(i).

At 0429, the sequencer mode switch (43-1/S2-1) was taken to OFF to allow various pumps which had been started by the sequencers, to be secured. Also the operators placed the 'B' channel 480V load shed switch in override and began resetting affected loads. At approximately 0440, valves PCV-2909 and PCV-2949 were closed from the Control Room and the control switch was placed in manual. At 0441, the Control Room crew began to secure loads which had been started by the sequencers. At 0503, two HPSI pumps were secured.

At 0510, the crew transitioned from EOP-01 to EOP-20, 'Functional Recovery Procedure.' This transition was performed because procedural guidance in EOP-01 did not address resetting of Engineered Safeguards, and the guidance provided in AOP-23 assumed that the lockout relay that caused the invalid actuation could be reset. Main Steam Safety Valves MS-291 and MS-292 were being used to provide a heat removal path with the MSIVs closed. The Shift Supervisor, Licensed Senior Operator and EOP Coordinator determined that guidance in EOP-20 could be used to establish a more controlled heat removal path using the Atmospheric Dump Valve and the Steam Dump and Bypass Valves.

At 0515, the last HPSI pump was secured. The containment Hydrogen Analyzers were placed in service at 0529, due to EOP-20 guidance associated with an abnormal increase in containment sump level.

Meanwhile, System Engineering personnel in the Technical Support Center were investigating possible causes of the tripping of the CPHS lockout relay and the inability to reset it. Their investigation indicated that the supervisory relay for the CPHS lockout relay may have failed. At 0530, after being briefed by System Engineering, the Shift Supervisor authorized the lifting of leads associated with the supervisory relay for 86B/CPHS.

At 0540, the main feedwater valves were switched to override so that the main feedwater regulating bypass valves could be opened to restore feedwater to the steam generators via the main feed ring. Feedwater flow through the main feed ring was re-established at 0608.

At 0613, the Shift Supervisor briefed Control Room personnel that the leads to the supervisory relay associated with 86B/CPHS had been lifted. Relay 86B/CPHS was then reset, and the remaining tripped lockout relays were subsequently reset. The MSIV bypass valves were opened at 0625, and the MSIVs were opened at 0648. At 0745, level in the EFWST had recovered enough to exit Technical Specification 2.0.1. The NOUE was terminated at 0747.

With respect to the water in the containment sump, post-event review determined that the water originated in the Safety Injection and Refueling Water Tank (SIRWT). When the 'A' train SIAS was removed, PCV-2909 and PCV-2949, which had remained shut with the SIAS signal present, were allowed to cycle to relieve pressure. With the HPSI header pressurized by the running HPSI pumps, these valves opened and passed sufficient flow to the RCDT header to result in water backing up through the RCP vapor seal drain line (which also discharges to the RCDT header). The water then came out the top of the annulus area above the RCP vapor seals, onto the floor, and then to the containment sump. The flow to the sump was stopped by closing PCV-2909 and PCV-2949.

During the event, operators noted an equipment issue involving Main Steam Safety Valves MS-291 and MS-292. These valves responded appropriately to the pressure transient resulting from closure of the MSIVs, but did not function as expected when operated manually later in the event. Post-event investigation indicated that O-rings and seals associated with the pneumatic operators had hardened with age. The safety function of MS-291 and MS-292 was not found to have been impacted by this condition.

CONCLUSIONS

The February 11, 1994 event was caused by the failure of relay 86B/CPHSS. This relay is a normally-energized supervisory relay for lockout relay 86B/CPHS. The supervisory relay experienced a short circuit across its coil which resulted in the invalid actuation of 86B/CPHS (see Figure 1). The actuation of 86B/CPHS initiated safeguards actuations and resulted in a turbine/reactor trip.

The relay that failed was a General Electric model number 12HGA17C52G, and is used in a number of applications within the Engineered Safeguards circuitry, as a power supply and continuity supervising relay. After the plant trip occurred, Design Engineering initiated a review to determine the potential consequences of invalid actuations of Reactor Protective System (RPS) or Engineered Safeguards initiation/actuation lockout relays, due to coil-shorting failure of an associated supervisory relay. This review considered such failures during normal plant operation and during design basis accidents. The intent of the review was to determine whether or not this type of failure was appropriately considered in the design of these circuits. This review is documented in Engineering Analysis EA-FC-94-008. Two design basis concerns were identified.

Initial investigation identified a concern involving a postulated premature actuation of either of two STLS lockout relays (i.e., 86A/STLS or 86B/STLS), due to a coil-shorting failure of a supervisory relay (see Figure 2). If such a failure were to occur coincident with certain accidents (i.e., a LOCA, a SGTR, or a MSLB), a premature Recirculation Actuation Signal (RAS) could be generated. Such a premature RAS could result in loss of water to the HPSI, LPSI and, if in service, the Containment Spray pumps, due to the realignment of the suction header from the SIRWT to the containment sump.

On February 18, 1994 at 1007, it was concluded that this concern represented a condition outside the design basis of the plant. NRC was notified of this determination on February 18, 1994 at .101, pursuant to 10 CFR 50.72(b)(1)(ii)(B). Immediate action was taken (via Temporary Modification 94-014), to disconnect the supervisory relays associated with 86A/STLS and 86B/STLS, in order to eliminate the possibility of such an occurrence. The safety function of the lockout relays was not compromised by disconnecting the associated supervisory relays. The alarm function provided by the supervisory relays is not available with the relays disconnected, however, indicator lights are available to provide an alternate method of monitoring the circuits (see Figure 1).

An additional concern was identified regarding the minimum recirculation header for the safety injection pumps. This header has two isolation valves in series (HCV-385 and HCV-386) that receive a signal to close in the event of a RAS. The concern related to a postulated premature actuation of any of eight STLS or RAS lockout relays (i.e., 86A/STLS, 86B/STLS, 86AX/RAS, 86BX/RAS, 86A1X/RAS, 86B1X/RAS, 86A1/STLS and 86B1/STLS) resulting in closure of HCV-385 and/or HCV-386 (see Figure 2). Two of these relays were also associated with the previously identified concern. Conservative action was taken on February 19, 1994 (via Temporary Modification 94-015), prior to completion of the engineering evaluation, to disconnect the six additional supervisory relays associated with this concern.

On March 3, 1994 at 0945, it was concluded that this concern had also represented a condition outside the design basis of the plant. Specifically, premature closure of HCV-385 and/or HCV-386 (due to a coil-shorting failure of a supervisory relay associated with any of the eight identified STLS/RAS lockout relays), coincident with certain MSLB, SGTR or small-break LOCA conditions, could result in dead-heading of HPSI pumps and the potential for pump damage. Depending on the specific relay failure, one, two or all three HPSI pumps may also not have suction available, but regardless of which relay failed, none of the pumps would have a minimum recirculation flow path available. The NRC was notified of this determination on March 3, 1994 at 1033, pursuant to 10 CFR 50.72(b)(1)(ii)(B).

The design of the STLS/RAS lockout relay supervisory circuits was part of the original plant design, and as such, was developed over twenty years ago. An investigation determined that the potential for a coil-shorting failure of a supervisory relay was considered at the time of the original design, but was apparently not found to be a credible failure mode. As a result, it has been concluded that the root cause of this problem was failure to anticipate a potential failure mode in the original design of Engineered Safeguards initiation/actuation circuitry.

This LER is being submitted pursuant to:

- 10 CFR 50.73(a)(2)(iv) due to the actuation of several engineered safety features and the RPS actuation that occurred on February 11, 1994,

- 10 CFR 50.73(a)(2)(i)(B) due to the entry into TS 2.0.1 that occurred during the February 11, 1994 event, and

- 10 CFR 50.73(a)(2)(ii)(B) due to the concerns identified regarding the potential for premature STLS/RAS lockout relay actuation coincident with certain accidents.

SAFETY ASSESSMENT

The plant response to the February 11, 1994 event was bounded by the Loss of Load analysis in Section 14.9 of the FCS Updated Safety Analysis Report (USAR). No safety injection flow into the RCS occurred during the event. Main Steam Safety Valves opened as required following the closure of the Main Steam Isolation Valves. This event did not result in releases of radioactive or hazardous material, or personnel injuries.

The safety significance of the identified supervisory relay/design basis concerns was also reviewed. The scenarios associated with both of the design basis issues require occurrence of certain design basis accidents coincident with a coil-shorting failure of one of the identified STLS/RAS supervisory relays. No scenario was identified in which a common cause relay failure would both initiate an event requiring safety injection and result in safety injection failure due to premature RAS. A review of the Probabilistic Risk Assessment (PRA) model for FCS has determined that the probability of a coil-shorting failure of any of the eight identified supervisory relays, coincident with an event requiring safety injection, is extremely low. As previously noted, the potential for such an occurrence has been eliminated by disconnecting the identified STLS/RAS supervisory relays.

CORRECTIVE ACTIONS

The following corrective actions have been or will be completed:

1. The supervisory relay that failed (86B/CPHSS) was replaced and the replacement relay was tested on February 12, 1994. The failed relay will be sent off-site for investigation of the failure mechanism. This investigation will be completed by August 31, 1994.

2. a. An Engineering Analysis was performed to assess the potential significance of short-circuit coil failures of supervisory relays in Engineered Safeguards and Reactor Protective System circuitry. Based on this evaluation, eight supervisory relays were identified as having the potential to cause unacceptable performance of plant safety systems during a design basis accident.

   b. Temporary modifications were implemented on February 18, 1994 and February 19, 1994 to eliminate the identified concerns, by disconnecting the eight supervisory relays.

   c. The temporary modifications will be removed and a permanent resolution of the design basis concerns implemented, by the end of the 1995 Refueling Outage.

3. The problem experienced during the event with the COP network has been addressed. A modem was replaced on February 11, 1994, and the COP network was successfully tested on February 14, 1994.

4. With respect to the difficulties with the use of the manual operators for MS-291 and MS-292 during the event, the operator seals and O-rings were replaced on February 11, 1994. Also, a Preventive Maintenance Order (PMO) will be developed to periodically replace the operator seals and O-rings. The development of this PMO will be completed by the start of the 1995 Refueling Outage.

PREVIOUS SIMILAR EVENTS

No previous similar events were identified involving initiation of engineered safeguards due to coil-shorting failure of a supervisory relay.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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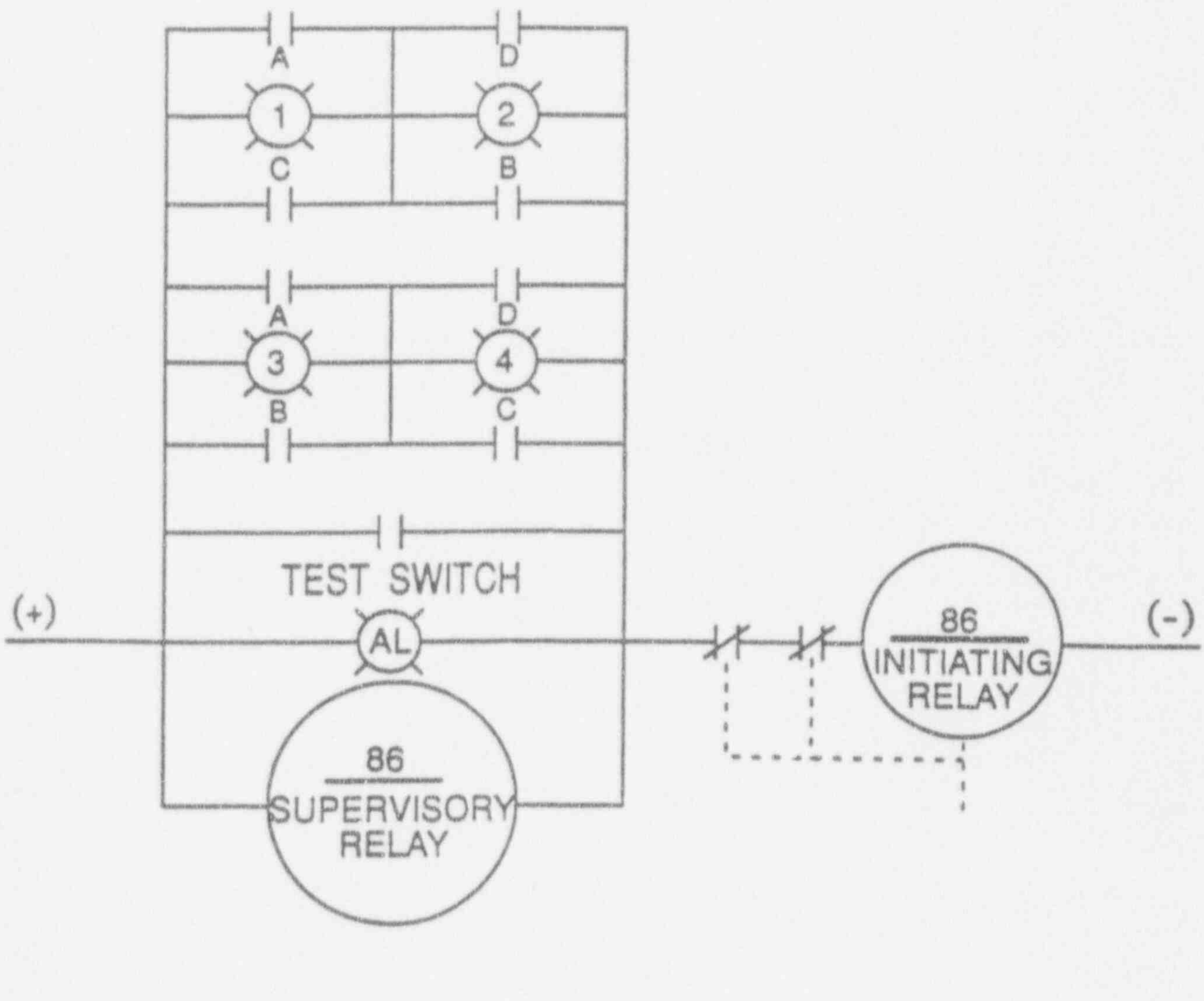
TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Figure 1

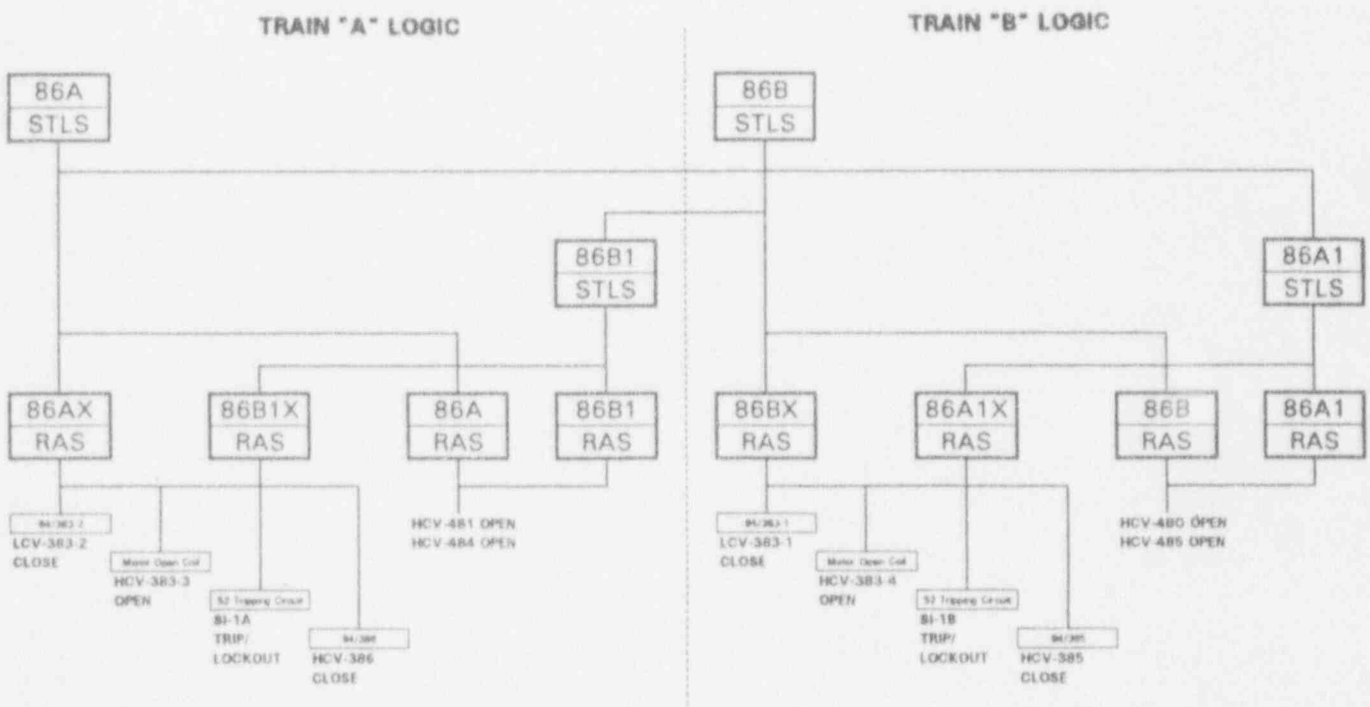
Supervisory Circuit Example



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

Figure 2

STLS/RAS Logic



NOTE: LOGIC ASSUMES A DBA (i.e. a PPLS or a CPHS signal present).