U. S. NUCLEAR REGULATORY COMMISSION
REGION 1

Report Nos.	92-13 92-12
Docket Nos.	50-334 50-412
License Nos.	DPR-66 NPF-73
Licensee:	Duquesne Light Company One Oxford Center 301 Grant Street Pittsburgh, PA 15279
Facility:	Beaver Valley Power Station, Units 1 and 2
Location:	Shippingport, Pennsylvania
Inspection Period:	April 19 - May 25, 1992
Inspectors:	Lawrence W. Rossbach, Senior Resident Inspector Peter P. Sena, Resident Inspector

Approved by: John F. Rogge   June 1, 1992
John F. Rogge, Chief   Date
Reactor Projects Section No. 4B

<u>Inspection Summary</u>

This inspection report documents the safety inspections conducted during day and backshift hours of station activities in the areas of: plant operations; radiological controls; surveillance and maintenance; emergency preparedness; security; engineering and technical support; and safety assessment/quality verification.

TABLE OF CONTENTS

EXECUTIVE SUMMARY

Beaver Valley Power Station
Report Nos. 50-334/92-13 & 50-412/92-12

Plant Operations

Overall, the units were both operated safely. A Unit 2 safety injection actuation occurred while in Mode 5. All plant equipment responded as designed. Although water was injected into the reactor coolant system, the safety significance was minor as the overpressure protection system was not challenged. However, a weakness was identified regarding the supervisory review of maintenance work instructions which led to this event. A self-identified, non-cited violation involving starting a main feedwater pump without meeting the initial conditions of a procedure was inspected. Operator error was the root cause.

Maintenance and Surveillance

Maintenance activities on the feedwater system were well planned and controlled. The 18 month emergency diesel generator surveillance test was properly conducted and demonstrated sequencer operability.

Emergency Preparedness

Operations personnel appropriately classified the safety injection as an Unusual Event and implemented the emergency preparedness plan implementing procedures in a timely fashion.

Engineering and Technical Support

Reliable decay heat removal was maintained during the outage. No safety concerns were identified.

Safety Assessment/Quality Verification

Several event reports were reviewed. The event descriptions, analysis, root cause determinations and corrective actions were of high quality. The licensee's followup of two information notices was reviewed and found to be responsive to the issues.

DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

Unit 1 operated at full power throughout this inspection period except for a planned power reduction to 30% power from May 15 to May 18. The purpose of the power reduction was to perform maintenance on the main feedwater regulating valves as discussed in section 4.3.

Unit 2 completed the Cycle III-IV refueling outage and returned to full power during this inspection period. On May 1, while still in Mode 5 (cold shutdown), a safety injection signal occurred which resulted in flow into the reactor coolant system. This Unusual Event is discussed in sections 2.2 and 5.1. The operators brought Unit 2 to Mode 4 (hot shutdown) on May 3 and to Mode 3 on May 5. Shortly after entering Mode 3 (hot standby), the motor driven auxiliary feedwater pumps, an engineered safety feature, started automatically due to a main feedwater pump trip. This event is discussed in section 2.3. The unit was brought critical at 7:05 p.m. on May 9. The refueling outage ended at 3:55 a.m. on May 12, the 60th day of the outage, when the main electrical generator output breakers were closed. This unit was at full power at the end of this inspection period.

2.0 PLANT OPERATIONS (71707, 93702)

2.1 Operational Safety Verification

Using applicable drawings and check-off lists, the inspectors independently verified safety system operability by performing control panel and field walkdowns of the following systems: low head safety injection; auxiliary feedwater; and emergency diesel generators. These system were properly aligned. The inspectors observed plant operation and verified that the plant was operated safely and in accordance with licensee procedures and regulatory requirements. Regular tours were conducted of the following plant areas:

- Control Room
- Auxiliary Buildings
- Switchgear Areas
- Access Control Points
- Protected Areas
- Spent Fuel Buildings
- Diesel Generator Buildings
- Safeguard Areas
- Service Buildings
- Turbine Buildings
- Intake Structures
- Yard Areas
- Containment Penetration Areas

During the course of the inspection, discussions were conducted with operators concerning knowledge of recent changes to procedures, facility configuration, and plant conditions. The inspectors verified adherence to approved procedures for ongoing activities observed. Shift turnovers were witnessed and staffing requirements confirmed. The inspectors found that control room access was properly controlled and a professional atmosphere was maintained. Inspectors' comments or questions resulting from these reviews were resolved by licensee personnel.

Control room instruments and plant computer indications were observed for correlation between channels and for conformance with Technical Specification (TS) requirements. Operability of engineered safety features, other safety related systems, and onsite and offsite power sources were verified. The inspectors observed various alarm conditions and confirmed that operator response was in accordance with plant operating procedures. Compliance with TS and implementation of appropriate action statements for equipment out of service was inspected. Logs and records were reviewed to determine if entries were accurate and identified equipment status or deficiencies. These records included operating logs, turnover sheets and system safety tags. The inspectors also examined the condition of various fire protection systems.

Plant housekeeping controls were monitored, including control and storage of flammable material and other potential safety hazards. The inspectors conducted detailed walkdowns of accessible areas of both Unit 1 and Unit 2. Housekeeping at both units was acceptable.

2.2 Unit 2 Safety Injection

On May 1, 1992, with the plant in Mode 5 (cold shutdown), a safety injection (SI) signal was initiated. The plant responded as designed to the SI signal. The inspector's review of the event identified several factors which contributed to the safety injection.

The safety injection resulted from a reset of the low pressure safety injection block permissive (P-11). This permissive functions to block the low pressure SI signal when pressurizer pressure is below 1845 psig. Protection channel I bistables for the SI block had been reset due to a surveillance test in progress. Protection channel III bistables for the SI block were reset due to the inadvertent deenergization of 120 Vac vital bus 2-3. With two of three bistables for low pressure safety injection block reset, an SI signal was initiated on low pressurizer pressure. Three pressurizer pressure bistables were already in a tripped condition as reactor coolant system (RCS) pressure was 343 psig.

All plant equipment responded as expected to the SI signal. Both emergency diesel generators auto-started but did not load onto the emergency buses since no undervoltage condition existed. The high head safety injection isolation valves (2SIS-MOV 867A-D) repositioned open and the 'A' charging pump injected about 2300 gallons of borated water into the RCS. RCS pressure increased to 375 psig during the injection. This pressure increase did not challenge the overpressure protection system since its setpoint was 458 psig. The duration of the injection signal was about 95 seconds before being reset by the operators. The operators then properly restored plant systems to their normal shutdown alignment and reenergized vital bus 2-3. The charging pump was secured within four minutes of the initial SI signal.

The 120 Vac vital bus 2-3 is normally energized by a rectifier/invertor assembly, specifically designated as an uninterruptible power supply (UPS). A 480/120 Vac voltage regulator is available as a backup or bypass power supply to the vital bus. Both power supplies were inadvertently removed from service in preparation for maintenance activities. Maintenance work request (MWR 09202) was authorized on May 1 for replacement of a fuse within the bypass voltage regulator. This was necessary in order to clear a "sync loss" alarm on the invertor. An operator, following the directions provided by the shift supervisor, first removed the UPS from service and aligned the bypass regulator to supply vital bus 2-3. The shift supervisor incorrectly believed that the UPS was to be removed from service for the intended maintenance activity. This was due, in part, to an inaccurate equipment mark (identification) number on the MWR which identified the equipment to be worked as "UPS-VITBS2-3." The inspector reviewed the licensee's master equipment list and identified that the correct equipment designation should have been "REG-VITBS2-3." After placing the voltage regulator in service, the operator proceeded to follow the MWR instructions which instructed operations to place the regulator out of service. The operator was not cognizant that the breaker manipulations being performed would remove the only remaining vital bus power supply from service. In addition, maintenance personnel knowledgeable of the system and system alignment were present in the switchgear room observing the operator's actions when both power supplies were removed from service.

The inspectors concluded that the licensee's actions taken in response to the safety injection were in accordance with procedures and that safety systems responded as designed. The actual safety significance of the event was minor. However, weaknesses regarding operation and maintenance personnel involvement in the work control process for this event were noted by the inspectors. Supervisory review of the MWR prior to authorizing work was found to be less than adequate. Although the mark number on the MWR was inaccurate and significantly contributed to the shift supervisor's misunderstanding, the inspectors considered the MWR instructions to take the regulator out of service to be clear and accurate. Maintenance personnel involved in the work planning were not aware that a separate mark number for the regulator existed. Additionally, maintenance personnel at the work site failed to recognize the consequences of the breaker manipulations being performed and failed to question the appropriateness of the operator's actions. Investigations by the licensee into the human performance factors of this event are continuing.

2.3 Auto Start of Unit 2 Auxiliary Feedwater Pumps

On May 5, 1992, operators started the first main feedwater pump, about an hour after entering Mode 3. The main feedwater pump tripped a few seconds later on low suction pressure. The auxiliary feedwater pumps started automatically, as designed, on the main feedwater pump trip. The operators secured the auxiliary feedwater pumps a few seconds later which prevented the steam generator from becoming overfilled. Since the feed system responded properly to this event and it did not lead to any undesirable plant conditions, the inspector concluded that it was of minor safety significance. The automatic start of the auxiliary feedwater pump was properly reported to NRC as an engineered safety feature actuation and an Licensee Event Report (LER) is being prepared.

Inadequate condensate flow was the cause of the main feedwater pump low suction pressure. Operating manual procedure 2.24.4.D, Revision 7, "Placing a Steam Generator Feed Pump in Service," states that the initial conditions for starting a main feedwater pump are that two condensate pumps are in operation. In violation of this procedural requirement, only one condensate pump was running at the time the main feedwater pump was started. The root cause of this engineered safety feature actuation was operator error in not meeting the initial condition. The operations manager counseled the supervisors and operator involved in this event on being attentive to procedure initial conditions. The utility is planning to include this issue in operator training materials. The inspectors concluded that adequate corrective actions had been taken by the licensee. This violation will not be subject to enforcement action because the licensee's efforts in identifying and correcting the violation met the criteria specified in Section VII.B of the revised Enforcement Policy dated February 18, 1992.

3.0 RADIOLOGICAL CONTROLS (71707)

Posting and control of radiation and high radiation areas were inspected. Radiation Work Permit compliance and use of personnel monitoring devices were checked. Conditions of step-off pads, disposal of protective clothing, radiation control job coverage, area monitor operability and calibration (portable and permanent), and personnel frisking were observed on a sampling basis.

Licensee personnel were observed to be properly implementing their radiological protection program.

4.0 MAINTENANCE AND SURVEILLANCE (61726, 62703, 71707)

4.1 Maintenance Observations

The inspectors reviewed selected maintenance work request (MWR) activities to assure that the activity did not violate Technical Specification Limiting Conditions for Operation and that redundant components were operable; required approvals and releases had been obtained prior to commencing work; procedures used for the task were adequate and work was within the skills of the trade; activities were accomplished by qualified personnel; radiological and fire preventive controls were adequate and implemented; QC hold points were established where required and observed; and equipment was properly tested and returned to service.

Maintenance activities reviewed included:

MWR 07784	Control Rod Drive Motor Generator Set No. 2 Bearing Replacement
MWR 07097	Steam Generator 1C Main Feedwater Regulating Valve Actuator Replacement

MWR 09140	Steam Generator 1A Main Feedwater Regulating Valve Inspection
MWR 08231 thru 08236	Inspect or Replace Solenoid Valves for Turbine Auto-Stop Trip and Overspeed Protection
MWR 08205	Refueling Cavity Low-Level Alarm Switch Reset Adjustment

Maintenance performed on the main feedwater system is discussed in section 4.3. The licensee's responsiveness to industry events involving turbine overspeed and auto-stop protection systems is discussed in section 8.3. There were no other notable observations.

4.2 Surveillance Observations

The inspectors witnessed/reviewed selected surveillance tests to determine whether properly approved procedures were in use, details were adequate, test instrumentation was properly calibrated and used, Technical Specifications were satisfied, testing was performed by qualified personnel, and test results satisfied acceptance criteria or were properly dispositioned. The following surveillance testing activities were reviewed:

EM 103216	Verify Turbine Auto-Stop Trip and Overspeed Protection Operability
OST 2.6.8	Pressurizer PORV Stroke Test
OST 2.24.3	Auxiliary Feed Pump 23B Test
OST 2.36.3	Emergency Diesel Generator 2-1 Automatic Load Test
BVT 1.1.1	Rod Drop Time Measurement and RPI Verification
1MSP 21.32-1	F-MS494 Loop 3 Steam flow Channel III Calibration
OST 1.36.19	Diesel Generator No. 1 Start-up
OST 1.30.2	Reactor Plant River Water Pump 1A Test

Operational Surveillance Test (OST) 2.36.3, "Emergency Diesel Generator (EDG) 2-1 Automatic Load Test," simulates a loss of offsite power in conjunction with a safety injection signal. The inspectors observed this 18 month surveillance following concerns of EDG sequencer relay operability as discussed in NRC inspection report 50-334/92-09 and 50-412/92-07. This surveillance verifies, in part, that the EDG starts from ambient conditions on an undervoltage auto-start signal, energizes the emergency buses, and energizes the auto-connected emergency loads through the load sequencer. This surveillance was an involved evolution requiring extensive coordination. The licensee appropriately designated a highly experienced senior reactor operator as the test coordinator. A detailed prebriefing was conducted to ensure all personnel involved were aware of their responsibilities. However, during the performance of the test, the SI signal was prematurely reset (prior to the time-out of a 75 second timer) upon direction by the test coordinator. Licensee personnel identified this error when an unsuccessful attempt was made to secure charging pump P-21C. The inspectors considered the consequences of this oversight to be of minor significance as operators correctly diagnosed the problem and took appropriate corrective action. The test results were not invalidated by the premature reset and the sequencer properly demonstrated the ability to auto-start the required emergency loads. Overall, the surveillance was properly performed and the reactor operators demonstrated good problem solving in identifying the premature SI reset.

4.3 Unit 1 Main Feedwater System Maintenance Activities

On May 15, 1992, the licensee reduced power from 100% to 30% in preparation for preplanned maintenance activities. The maintenance performed included inspection and cleaning of main condenser water boxes, replacement of three main feedwater regulating valve (MFRV) actuators, and inspection of the 'A' MFRV internals. The licensee decided to replace the MFRV actuators with refurbished spares in order to ensure the reliability of the valves during the upcoming peak demand season. Following the required post maintenance testing, power was successfully returned to 100% on May 18.

The inspectors observed the work activities and reviewed maintenance work requests 07097 and 09140 associated with the MFRVs. Corrective maintenance procedure 24FW-Feed Reg 1I was used as a guide by the maintenance personnel for the actuator removal and installation. The inspectors noted a high level of detail, as well as clear and precise instructions, within the procedure. The use of a formalized corrective maintenance procedure places less reliance on an individual's system expertise and helps to ensure repairs are performed consistently from one maintenance crew to another. At the job site, ample maintenance personnel and supervision were involved. The maintenance personnel demonstrated familiarity with the equipment being serviced and were aware of expected responses during valve testing. The inspectors concluded that the licensee demonstrated the ability to properly plan and control these maintenance activities.

5.0 EMERGENCY PREPAREDNESS (71707)

5.1 Notification of Unusual Event

On May 1, 1992, at 2:08 p.m., Beaver Valley declared an Unusual Event (UE) due to the emergency core cooling system discharge into the RCS (see section 2.2). The Nuclear Shift Supervisor correctly classified the event as required by Emergency Preparedness Plan (EPP), Implementing Procedure I-1. Event classification was timely as the UE was declared within 12 minutes of the event. Required notifications to federal, state, and local government agencies were satisfactorily completed within established time requirements. The event was appropriately declassified at 2:48 p.m. No deficiencies were noted by the inspectors.

6.0 SECURITY (71707)

Implementation of the Physical Security Plan was observed in various plant areas with regard to the following: Protected Area and Vital Area barriers were well maintained and not compromised; isolation zones were clear; personnel and vehicles entering and packages being delivered to the Protected Area were properly searched and access control was in accordance with approved licensee procedures; persons granted access to the site were badged to indicate whether they have unescorted access or escorted authorization; security access controls to Vital Areas were maintained and persons in Vital Areas were authorized; security posts were adequately staffed and equipped, security personnel were alert and knowledgeable regarding position requirements, and that written procedures were available; and adequate illumination was maintained. Licensee personnel were properly implementing the Physical Security Plan.

7.0 ENGINEERING AND TECHNICAL SUPPORT (2515/113, 37828)

The inspectors reviewed Unit 2 refueling outage activities that could contribute significantly to a loss of decay heat removal or that could contribute significantly to preventing a loss of decay heat removal as described in NRC Temporary Instruction 2515/113, "Reliable Decay Heat Removal During Outages." The inspectors noted that the licensee also reviewed their outage activities for the issues discussed in NRC Temporary Instruction 2515/113, and Nuclear Management Resource Council, "Guidelines for Industry Actions to Assess Shutdown Management."

No special tests that would contribute significantly to the loss of decay heat removal were identified. One such modification was identified. This temporary modification, 2-92-008, used the fire protection system as a source of cooling water to component cooling water heat exchanger 21B while service water cross-connect valves were repaired. The modification received close management involvement in the planning and performance of the modification activities in accordance with Nuclear Group Administrative Procedure 8.23, "Infrequently Performed Tests and Evolutions." Licensee management ensured that detailed safety assessments were performed, contingency plans were formalized and in-place personnel were properly briefed before implementing the modification. The temporary modification was fully tested prior to removing both service water trains from service.

Licensee operating manual procedures ensured that forced circulation decay heat removal was maintained when required. Natural circulation cooling was not planned or used but contingencies for natural circulation cooling exist in abnormal operating instructions and emergency operating procedures.

The Unit 2 safeguards equipment is divided between two electrical busses and is designated train 'A' or 'B'. Each of these two busses is powered from a separate offsite power supply through a station service transformer and each has an emergency onsite power supply powered by a diesel generator. During the refueling outage, the main generator disconnect links were removed and the main transformer was backfed to provide an additional source of offsite power to the safeguards busses. The licensee used approved procedures to establish and maintain this backfeed lineup. Throughout the outage, either train 'A' or 'B' was designated the priority train. The non-priority train and associated emergency power supply were released for maintenance consistent with Technical Specifications. This means that for the two residual heat removal (RHR) pumps both pumps were available when required by Technical Specifications; however, the emergency power supply for the non-priority train was removed from service for maintenance.

The inspectors noted that the station operators were properly trained with appropriate procedures to manually control the electric power system if automatic controls were disabled. Despite this training, the licensee declares the power system inoperable if automatic controls are disabled. Emergency diesel generators are also declared inoperable if its field flashing source is removed from service.

The inspectors concluded that reliable decay heat removal was maintained during the outage. No safety concerns were identified.

8.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION (71707, 92700, 90712)

8.1 Review of Written Reports

The inspectors reviewed Licensee Event Reports (LERs) and other reports submitted to the NRC to verify that the details of the events were clearly reported, including accuracy of the description of cause and adequacy of corrective action. The inspectors determined whether further information was required from the licensee, whether generic implications were indicated, and whether the event warranted further onsite followup. The following LERs were reviewed:

Unit 1:

91-19-01 Missed Examinations Resulting from a Programmatic Review of ISI Program

92-05 Missed Surveillance of River Water Valve for Component Cooling Water Heat Exchangers

These events were reviewed in NRC inspection report 92-09/07. The inspectors have no additional comments on these events.

91-26-01 Potentially Inoperable Charging Pump Due to Missing Nuts on High Speed Coupling

LER 91-26 was initially reviewed in NRC inspection report 91-23/22. This LER describes the identification of missing nuts to the 'B' charging pump. Specifically, ten nuts for the pump side spool piece coupling were not attached to the bolts, but were lying in the bottom of the coupling guard. The nuts have been reinstalled. The licensee's initial engineering evaluation indicated that if the charging pump was operating during a seismic event, then the coupling would remain intact and continue to operate. However, if the pump was idle during the seismic event, afterward the coupling might not function it the pump were started. The revised LER incorporates a second engineering analysis which concluded the coupling would remain intact during a seismic event, even if the pump was initially shut down during the event. This report was voluntarily submitted by the licensee.

92-04 Degraded Diesel Generator Ventilation System

The licensee identified a potentially degraded condition associated with the Unit 1 emergency diesel generator building exhaust ventilation system. Specifically, thermostats in the ventilation start circuitry were not Quality Assurance Category I and thus could not be relied upon under accident conditions. This design deficiency has been attributed to original plant construction. These thermostats normally start the diesel ventilation exhaust fans when the cubicle ambient temperature exceeds 90°F during extended diesel operation. In the event of a postulated thermostat failure, manual start of the fans would still be available. The licensee has since modified the fan start circuit to initiate exhaust fan operation whenever its associated diesel starts. The entire circuit is now Quality Assurance Category I.

Unit 2:

92-02 ESF Actuation - Control Rod Drive Fan Breaker Tripped

This event was reviewed in NRC inspection report 92-05/04. The inspectors have no additional comments on this event.

92-03 ESF Actuation - Feedwater Isolation Due to Hi-Hi Level in the 'A' Steam Generator

This event was reviewed in NRC inspection report 92-05/04. Since that inspection, the licensee determined that it is not necessary to maintain steam generator level between 60% and 70% for the steam generator chemistry soak. Operating procedure 20M-51.4D, "Station Shutdown - Cooldown from Hot Standby to Cold Shutdown," is therefore being revised to specify that steam generator level be maintained at the normal programmed level of 33% for the steam generator soak. This procedure change will provide additional margin to the feedwater isolation setpoint at 75% steam generator level and will make the high level deviation alarm at 38% level available to annunciate increasing levels. The inspectors consider this additional corrective action is adequate to prevent recurrence. The inspectors have no additional comments on this event.

92-05 Containment Penetration Improperly Sealed During Fuel Movement

This event was reviewed in NRC inspection report 92-09/07 and at an enforcement conference held on May 19, 1992. As discussed at the enforcement conference and in the LER, radiation monitor RMR-RQI301 would have detected radioactivity that could have resulted from release through these improper temporary containment penetration seals. Radiation above the RMR-RQI301 setpoint would have caused the exhaust from this area to automatically divert to the supplementary leak collection and release system for filtration prior to release. This automatic diversion capability was described to the inspector by the licensee after the completion of inspection 92-09/07 which only discusses a manual diversion capability based on the capability of radiation monitor HVS-RW101. The inspector concluded that this additional release detection and automatic diversion capability further reduced the potential for unfiltered release from a postulated fuel handling accident and therefore also reduced the safety significance of this event. The inspectors have no additional comments on this event. The enforcement conference which was held with the licensee in the Region I office on May 19, 1992, was attended by those listed in Attachment A. Duquesne Light Company presentations at that meeting are presented in Attachment B. The results of the enforcement conference will be documented separately.

The above LERs were reviewed with respect to the requirements of 10 CFR 50.73 and the guidance provided in NUREG 1022. Generally, the LERs were found to be of high quality with good documentation of event analyses, root cause determinations, and corrective actions.

8.2 Information Notice 92-30 Followup

In April, the inspectors discussed with the managers of nuclear operations, incidents at other sites where auxiliary operators had not properly completed their rounds. Information Notice 92-30, which was issued on April 23, describes this issue. The site quality services unit reviewed records of site operator performance from January 13 to March 14, 1992. The review verified that the operators had entered the areas required to be entered to complete their logs. The review found no problems with the performance of site tours by the assigned operators. The operations managers also issued night orders and shift supervisors held shift briefings to review information on these incidents and to remind operators of their responsibilities in this area. The inspectors concluded that site management had been responsive to this issue.

8.3 Information Notice 91-83 Followup

Information Notice 91-83, dated December 20, 1991, described solenoid-operated valve failures that resulted in turbine overspeed at other sites. Inadequate operational testing and preventive maintenance contributed to these events. In response to this issue, the licensee inspected and repaired or replaced turbine auto-stop trip solenoids and overspeed protection solenoids during the Unit 2 refueling outage. Operability of the system was verified through operational surveillance tests and special testing in accordance with Westinghouse corrective action letter 92-02. The preventive maintenance program for these components was upgraded to perform this work routinely. Similar work is planned for Unit 1 during the next outage. The inspectors concluded that the utility had taken thorough actions to improve the reliability of the turbine auto-stop and overspeed protection system.

9.0 MANAGEMENT MEETINGS AND NRC STAFF ACTIVITIES

9.1 Preliminary Inspection Findings Exit

At periodic intervals during this inspection, meetings were held with senior plant management to discuss licensee activities and inspector areas of concern. Following conclusion of the report period, the sident inspector staff conducted an exit meeting on May 29, 1992, with Beaver Valley management sur..marizing inspection activity and findings for this period.

9.2 Attendance at Exit Meetings Conducted by Region-Based Inspectors

Dates	Subject	Inspection Report No.	Reporting Inspector
April 24, 1992	MOV testing	92-30	M. Banerjee
April 24, 1992	ISI	92-12/08	P. Patnaik
May 8, 1992	Emergency Preparedness	92-09/07	L. Eckert

9.3 NRC Staff Activities

Inspections were conducted on both normal and backshift hours: 34 hours of direct inspection were conducted on backshift; 13 hours were conducted on deep backshift. The times of backshift hours were adjusted weekly to assure randomness.

A team inspection of safety-related motor-operated valve testing and surveillance was conducted from April 20 to 24, 1992. Richard Janati, Nuclear Engineer, Pennsylvania Department of Environmental Resources and P. K. Eapen, Section Chief, Nuclear Regulatory Commission, Region I, visited the site on April 23 and 24 in relation to this inspection (NRC inspection report 50-334/92-80 and 412/92-80).

An inspection of nuclear welding, inservice inspections, and steam generator eddy current inspections was conducted from April 20 to 24 and from April 27 to 30, 1992 (NRC inspection report 50-334/92-12 and 412/92-08).

Richard Janati, Nuclear Engineer, Pennsylvania Department of Environment Resources visited the site on May 4 and discussed radioactive waste packaging and shipping with the inspectors.

An inspection of the emergency preparedness program was conducted from May 4 to 8, 1992 (NRC inspection report 50-334/92-11 and 50-412/92-10).

An enforcement conference was held with the licensee in the Region I office on May 19, 1992. The conference was held to discuss the emergency diesel generator sequencing relay failures and temporary containment penetration seal issues described in NRC inspection report 50-334/92-09 and 50-412/92-07. Attachment A is the attendee list for the meeting. Attachment B contains the material presented by Duquesne Light Company representatives at that meeting.

ATTACHMENT A

Enforcement Conference Attendee List

Duquesne Light Company

J. Sieber	Vice President, Nuclear Group
N. Tonet	Manager, Nuclear Safety
G. Thomas	General Manager, Corporate Nuclear Services
K. Grada	Manager, Quality Services
T. Noonan	General Manager, Nuclear Operations
J. Turner	Nuclear Shift Supervisor
K. Halliday	Manager, Nuclear Engineering
S. LaVie	Senior Health Physics Specialist

US Nuclear Regulatory Commission (NRC)

C. Hehl	Director, Division of Reactor Projects (DRP)
A. Blough	Chief, Projects Branch 4, DRP
J. Rogge	Chief, Projects Section 4B, DRP
W. Ruland	Chief, Electrical Section, Division of Reactor
Safety	(DRS)
L. Rossbach	Senior Resident Inspector, Beaver Valley, DRP
P. Sena	Resident Inspector, Beaver Valley, DRP
T. Frye	Reactor Engineer, Projects Section 4B, DRP
J. Calvert	Reactor Engineer, Electrical Section, DRS
A. DeAgazio	Project Manager, Beaver Valley, Nuclear Reactor Regulation (NRR)
R. Fuhrmeister	Acting Enforcement Specialist, Region I

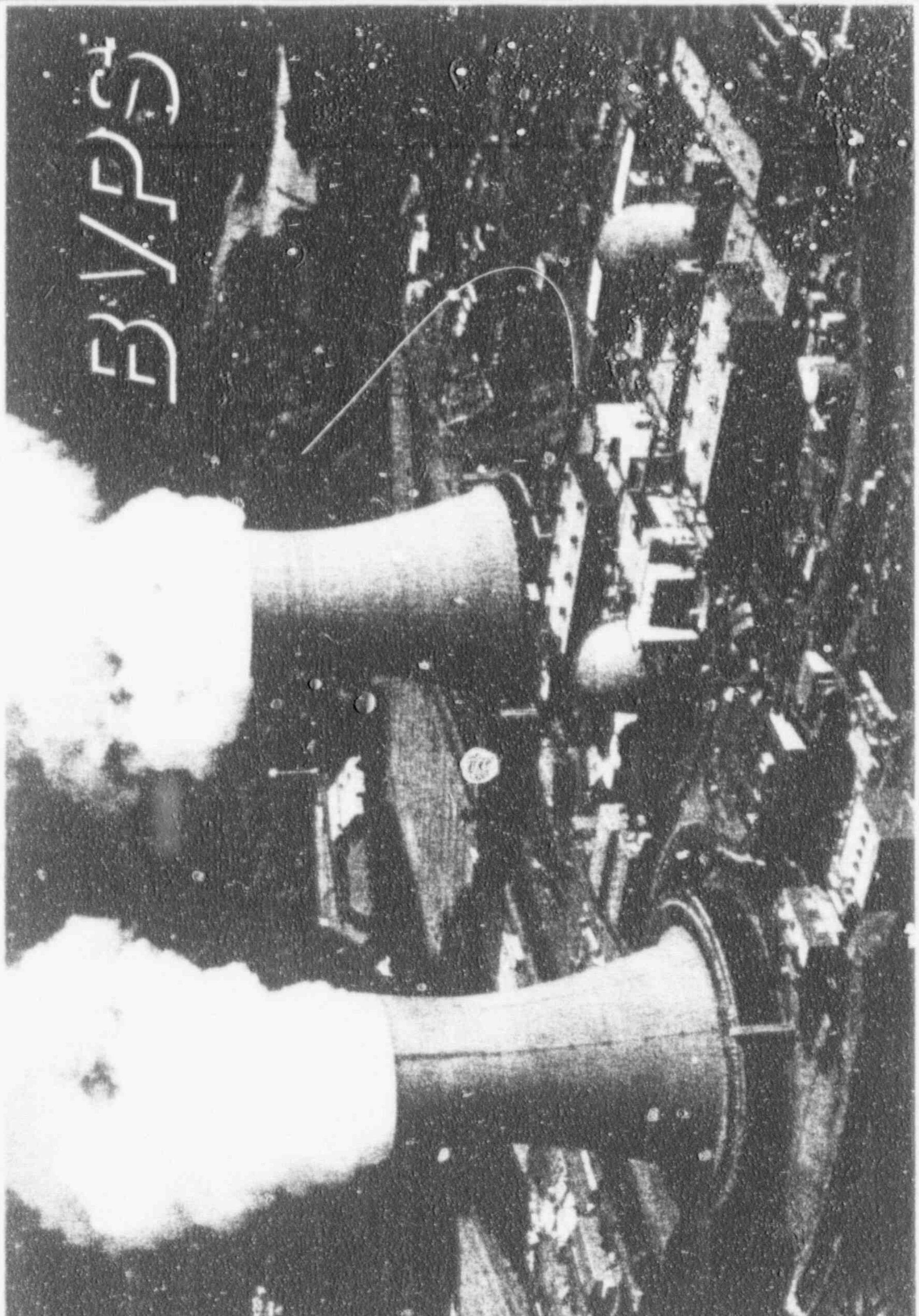
ATTACHMENT B

Duquesne Light Company's Enforcement Conference Presentation of May 19, 1992

NRC Enforcement Conference

King of Prussia, PA

May 19, 1992



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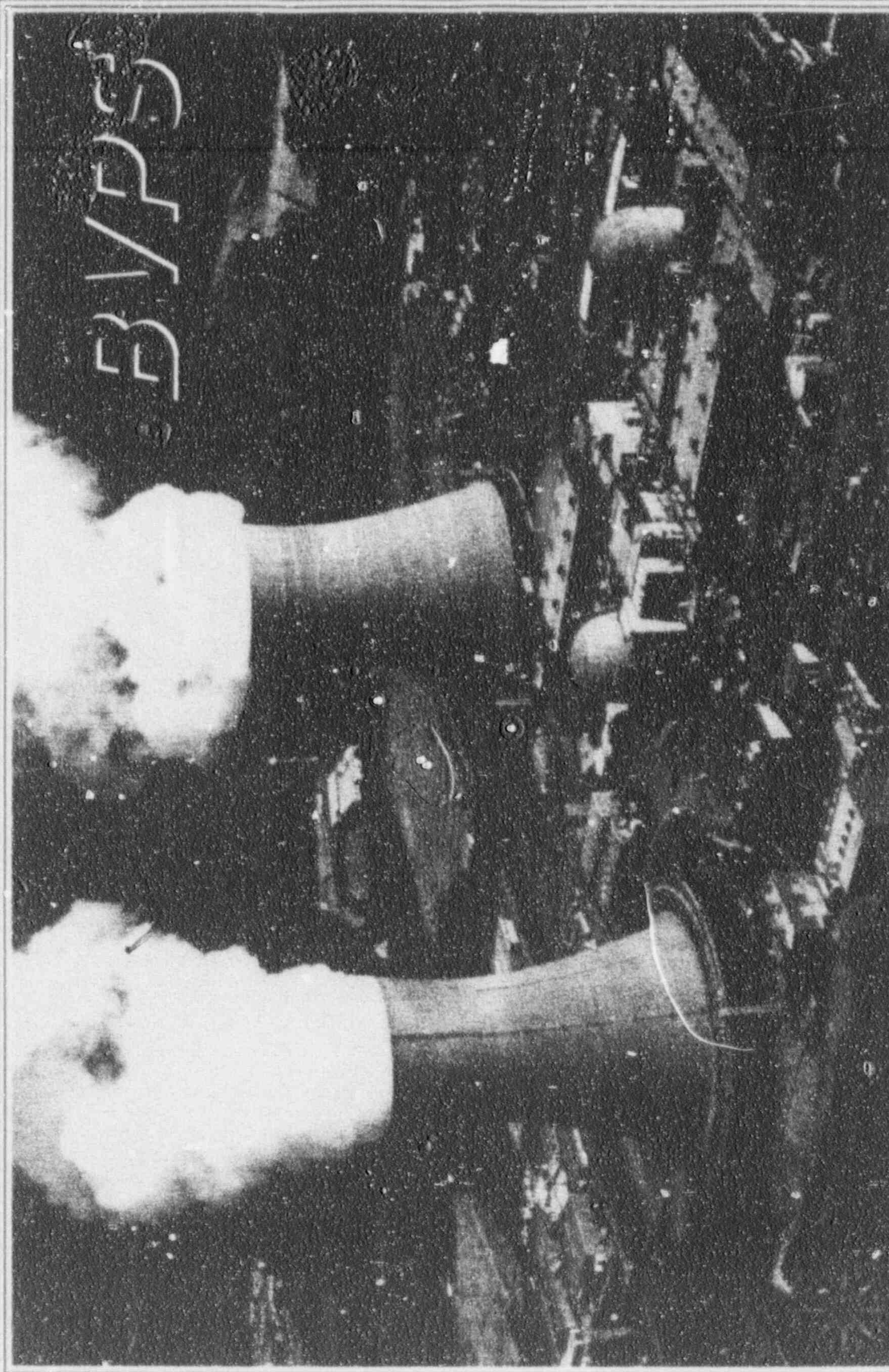
ATTACHMENT B

Duquesne Light Company's Enforcement Conference Presentation of May 19, 1992

NRC Enforcement Conference

King of Prussia, PA

May 19, 1992



BPENFRC R2

NRC Enforcement Conference
May 19, 1992
King of Prussia, PA

Duquesne Light Co.
Attendees

J. D. Sieber, Vice President, Nuclear
G. S. Thomas, Gen. Mgr., CNSU
T. P. Noonan, Gen. Mgr., NOU
K. D. Grada, Mgr., QSU
N. R. Tonet, Mgr., Nuclear Safety
K. E. Halliday, Mgr., Nuclear Engineering
S. F. LaVie, Sr. Health Physics Specialist
J. W. Turner, Nuclear Shift Supervisor

NRC ENFORCEMENT CONFERENCE
May 19, 1992
King of Prussia, PA

AGENDA

A. Containment Penetration Temporary Seal Air Leakage
   1. Sequence of Events
   2. Penetration Description
   3. Ventilation Configuration
   4. Event and Caus[illegible]actor Chart
   5. Technical Spec[illegible]on Review
   6. 10 CFR Part 2, App. C
      a. Identification
      b. Corrective Action
      c. Licensee performance
      d. Prior opportunity to identify
      e. Multiple occurrences
      f. Duration

B. Emergency Diesel Generator Sequencing Relay Failure
   1. Background Description of Relay Operation
   2. Sequence of Events
   3. Diesel Generator Relay Modification
   4. Failure Modes, Effects and Consequences
   5. Events and Causal Factors Chart
   6. 10 CFR Part 2, App. C.
      a. Identification
      b. Corrective Action
      c. Licensee performance
      d. Prior opportunity to identify
      e. Multiple occurrences
      f. Duration

Containment Penetration Temporary Seal Air Leakage

1. Sequence of Events
2. Penetration Description
3. Ventilation Configuration
4. Event and Causal Factor Chart
5. Technical Specification Review
6. 10 CFR Part 2 Appendix C Discussion

CHRONOLOGY of EVENTS

CONTAINMENT PENETRATION TEMPORARY SEAL AIR LEAKAGE

- 3-13-92 BVPS Unit 2 Shutdown for Refueling
- 3-16-92 Maintenance Work Request Issued for Installation of Temporary Cables
- 3-19-92 Work Began on Routing of Temporary Cables
- 3-22-92 Fire Protection Engineer Notified of Fire Seal Placement at Containment Penetration
- 3-23-92 Fuel Off-Load Containment Closure Verification Completed & Rx Head Lift Began (2005)
- 3-24-92 Quality Services Initiates Discussion of Dual Function Penetration Seals
- 3-25-92 Fuel Offload Began (1004)
- 3-27-92 Fuel Offload Completed (1358)

CHRONOLOGY of EVENTS

CONTAINMENT PENETRATION TEMPORARY SEAL AIR LEAKAGE

- 4-5-92 NCD Specialist Walkdown of Temporary Seal
- 4-7-92 Temporary Penetration for Eddy Current Restored to Normal Status
- 4-8-92 Engineering Memorandum Issued by Quality Services Requesting Technical Review of Temporary Fire Seals
- 4-8-92 Fuel Reload Containment Closure Verification Completed & Fuel Reload Began (2050)
- 4-9-92 Leaking Temporary Penetration Discovered (0356) & Fuel Reload Stopped. Fuel Element In-Transit Placed in Rx (0422)
- 4-9-92 Temporary Penetration Returned to Fully-Tested, Operational Status (0652) & Reload Restarted

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Containment Penetration Temporary Seal Air Leakage

Penetration Description

G.S. Thomas

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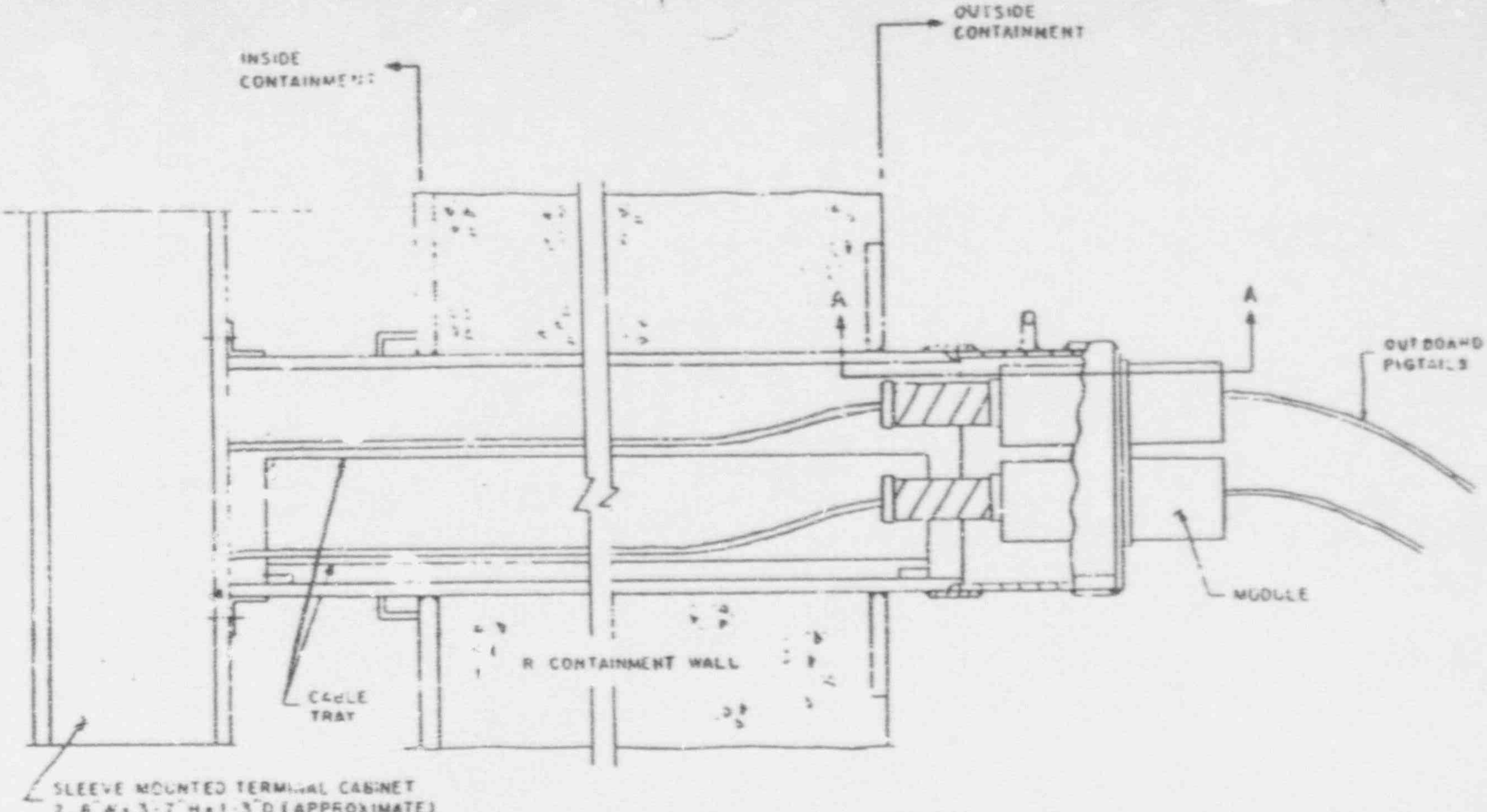
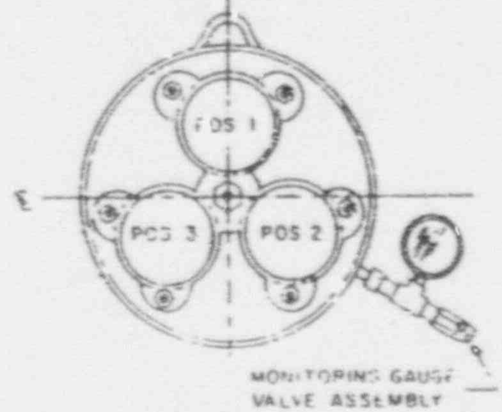


TYPICAL
ELECTRICAL PENETRATION ASSEMBLY



OUTBOARD END VIEW

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Containment Penetration Temporary Seal Air Leakage

Ventilation Configuration

S. F. LaVie

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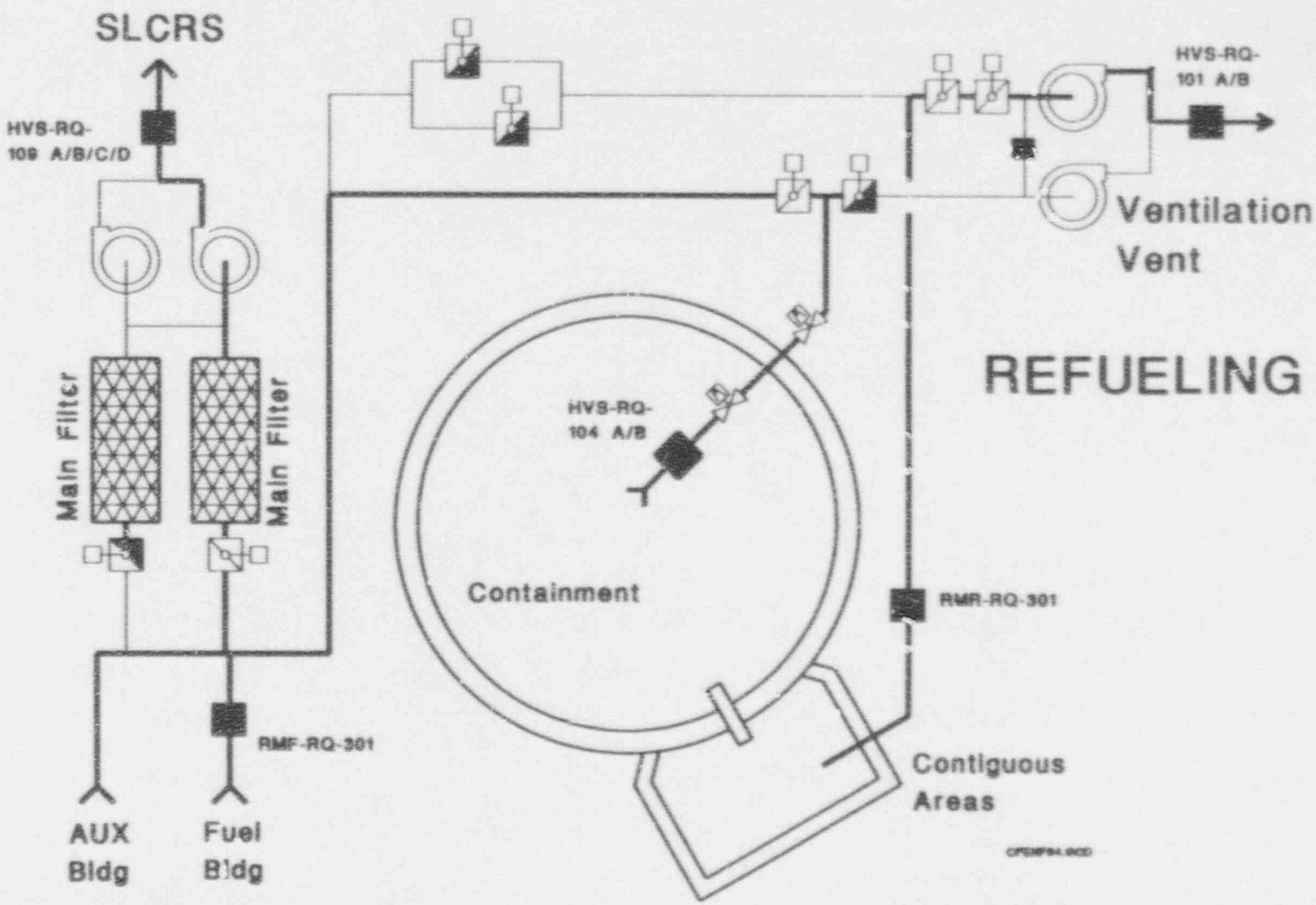
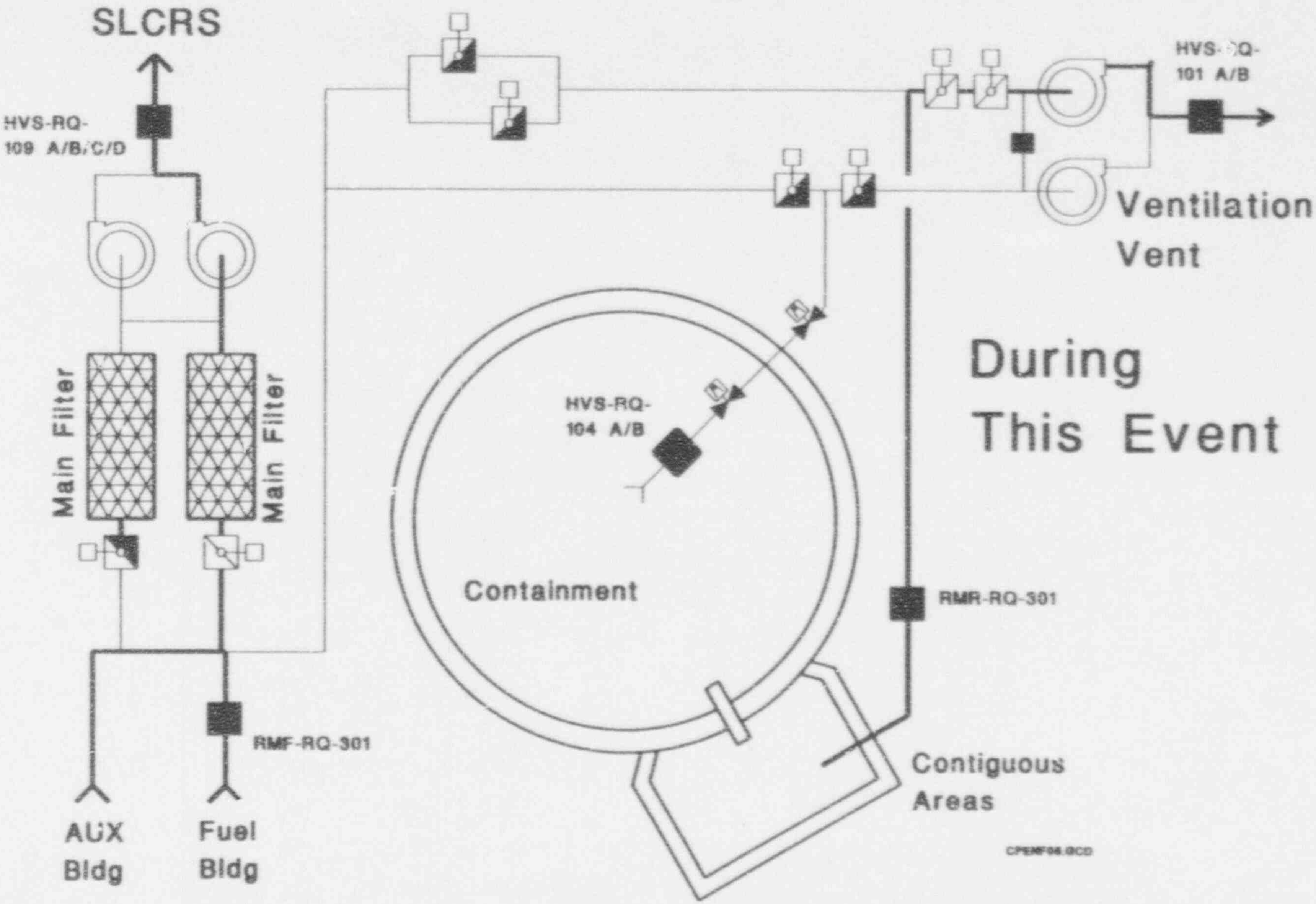
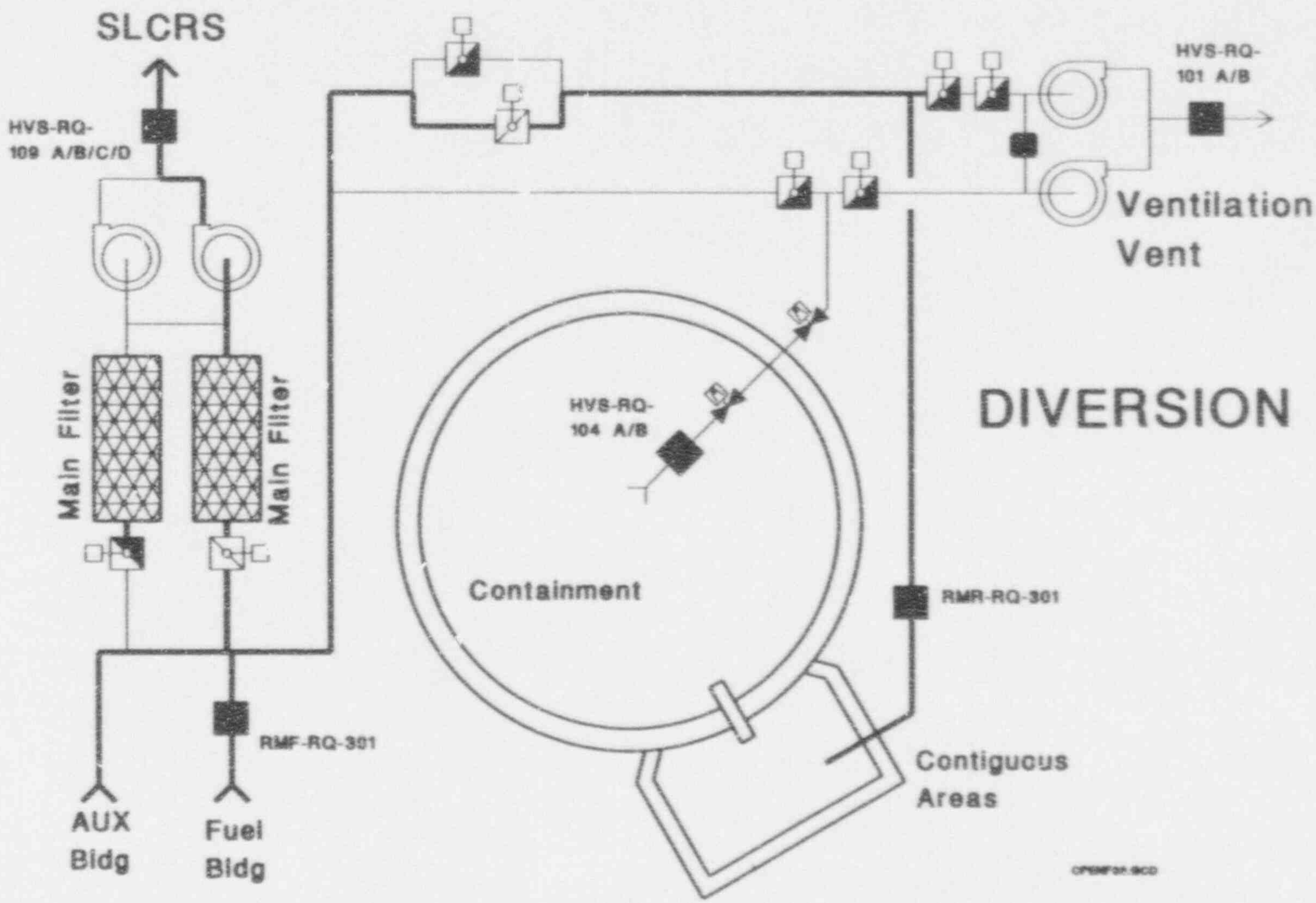

SLCRS
HVS-RQ-
109 A/B/C/D
Main Filter
Main Filter
AUX
Bldg
Fuel
Bldg
RMF-RQ-301
HVS-RQ-
104 A/B
Containment
RMR-RQ-301
Contiguous
Areas
HVS-RQ-
101 A/B
Ventilation
Vent
REFUELING

RMR-RQ301 Alarm

- *Basis:* 0.5 MPC at Site Boundary
- *Setpoint:* 3.11E-6 μCi/cc
- *Postulated Cnmt Activity*: 2.0E-2 μCi/cc
- *Other*
  - ★ Alarms in Control Room
  - ★ Initiates Diversion to SLCRS
  - ★ Response Procedures In Place
  - ★ Surveillances Performed on Monitor


SLCRS
HVS-RQ-
109 A/B/C/D
Main Filter
Main Filter
AUX
Bldg
Fuel
Bldg
RMF-RQ-301
HVS-RQ-
104 A/B
Containment
RMR-RQ-301
Contiguous
Areas
101 A/B
Ventilation
Vent
During
This Event


SLCRS
HVS-RQ-
109 A/B/C/D
Main Filter
Main Filter
AUX
Bldg
Fuel
Bldg
RMF-RQ-301
HVS-RQ-
104 A/B
Containment
RMR-RQ-301
Contiguous
Areas
HVS-RQ-
101 A/B
Ventilation
Vent
DIVERSION

UFSAR Fuel Handling Accident

- Containment Case not Specifically Addressed
- Fuel Bldg FHA is More Limiting

UFSAR FHA, *(REM)*

	Photon	Beta	Thyroid
0-2 Hours EAB	2.3	6.6	29.0
30 Day LPZ	0.11	0.32	1.4
30 day Control Room	0.056	4.0	3.2

Analysis Assumptions

- Based on RG 1.25, SRP 15.7.4
  (exc I-131 Gap Fraction is 12% Vicc 10%)
- 11 Day Decay Period
- Filtered/Unfiltered Release Cases
- 7500 cfm Release Rate
  (Expected 100-200 cfm)
- No Control Room Isolation Modeled
- Accident X/Qs

Radiological Consequences *(rem)*

	Beta	Photon	Thyroid
FILTERED RELEASE			
0-2 hr EAB	0.01	0.007	5.9
30 day LPZ	0.002	<0.001	0.7
30 day C.R.	0.002	<0.001	1.1
UNFILTERED RELEASE			
)-2 hr EAB	0.03	0.06	120.0
30 day LPZ	0.003	0.005	14.0
30 day C.R,	0.005	<0.001	22.0
10 CFR 100	*25*	*25*	*300*
SRP 15.7.4	*6*	*6*	*75*
GDC 19	*5*	*5*	*30*

Ref: ERS-SFL-92-022
Based on 7500 cfm Release Flow

Analysis Conservatisms

- ☞ Containment Mixing Assumption
- ☞ High Release Flow Rate
- ☞ No Credit for Plateout
- ☞ Low Pool Scrubbing Credit
- ☞ No Operator Action for 30 days

Consequence Summary

- ☞ If a FHA had occured in containment, releases greater than 10 CFR 20 limits would have caused exhaust diversion to the filter banks, thus mitigating the consequences of the accident

- ☞ Filtered release doses would have been less than DBA FHA

- ☞ Unfiltered release doses would have been less than 10 CFR 100 guidelines

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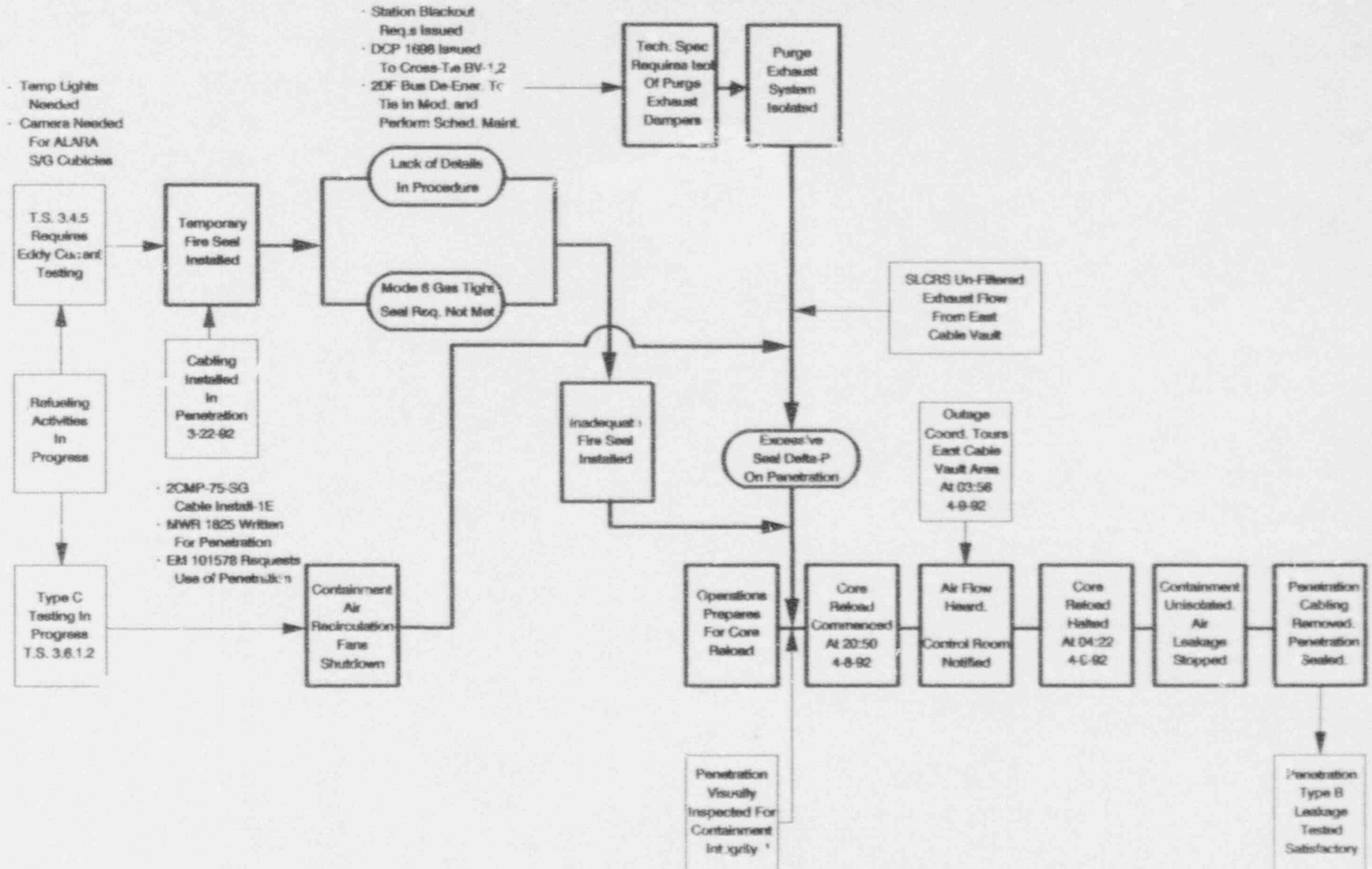
King of Prussia, PA
May 19, 1992

Containment Penetration Temporary Seal Air Leakage

Event and Causal Factor Chart

K.D. Grada

IMPROPERLY SEALED CONTAINMENT PENETRATION
(LER 92-005-00)



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May 19, 1992

Containment Penetration Temporary Seal Air Leakage

Technical Specification Review

K.D. Grada

Discussion on the Requirements of Technical Specification 3.9.4

Technical Specification 3.9.4 requires that each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either closed by an isolation valve, blind flange, manual valve or exhausting ≤ 7500 cfm through operable containment purge and exhaust isolation valves. The bases for technical specification 3.9.4 describes that the purpose of containment penetration closure is to limit leakage of radioactive material within containment to the environment to ensure compliance with 10 CFR 100 limits. These requirements are sufficient to restrict radioactive material release from a fuel element rupture, based upon the lack of containment pressurization potential while in the Refueling Mode.

The words "Containment Integrity" are not contained in technical specification 3.9.4 or its basis. No requirement exists to ensure that each containment penetration is operable and meets the stricter requirements imposed under the technical specification definition of "Containment Integrity" for Modes 1-4. While we characterized the problem in LER 92-005, as a containment integrity issue, the issue should have been reported as a degraded containment boundary.

The temporary installation of electrical penetration 2RCP-11E is acceptable for refueling and does not violate the intent of technical specification 3.9.4, provided that the penetration maintains a gas tight seal against the maximum differential pressure for Mode 6. This is consistent with other utilities' interpretations and certain NRC personnel. The penetration, if it had been installed to maintain a gas-tight seal, would not have provided a pathway from containment to the outside atmosphere. Once the penetration seal fails, then the requirements of technical specification 3.9.4 must be met. It should be noted that this leakage was not a "direct path" to the outside atmosphere and would have been filtered under a hypothetical radiological release.

CONCLUSIONS

1. Containment Integrity is not defined, or required, for Mode 6. T.S. 3.9.4 requires an adequate boundary on all containment penetrations during Mode 6.

2. A gas tight seal designed for the maximum differential pressure across the penetration would be in compliance with T.S. 3.9.4.

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May 19, 1992

Containment Penetration Temporary Seal Air Leakage

10 CFR Part 2, Appendix C Discussion

K.D. Grada

BPENFRC.PR2

10 CFR Part 2, Appendix C Review

Apparent Violation

One of the Unit 2 containment building penetrations did not conform to the technical specification 3.9.4 configuration requirements. Technical specification 3.9.4 requires that each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be closed by an isolation valve, blind flange, or manual valve, or exhausting at less than or equal to 7500 cfm through operable containment purge and exhaust isolation valves. Contrary to the above, spare electrical penetration 2RCP-11E was discovered to be leaking and therefore not meeting the requirements of technical specification 3.9.4.

The following is a review of the 10 CFR Part 2 Appendix C Criterion for Mitigation:

a. Identification

DLCo identified the problem on April 9, 1992 at 0356 hours. The on duty Shift outage Manager instructed fuel movement to be immediately halted once the fuel was secured in safe position. The NSS made a 10 CFR 50.72 four hour notification of the event on April 9, 1992 at 0615 hours. An incident report (IR 2-92-25) was prepared to address reportability. The event was determined to be reportable and a licensee event report (LER 92-005-00) was prepared. The LER was submitted to the NRC on May 11, 1992.

The condition, when electrical penetration 2RCP-11E did not conform to the requirements of technical specification 3.9.4 existed for 7.5 hours (best estimate) and less than 72 hours (worst case). The violation was not easily discovered due to the following factors:

1) The failure of electrical penetration 2RCP-11E could not be detected from any instrumentation or alarms which are available to control room personnel.

2) The location of the electrical penetration is not in a heavily traveled area.

b. Corrective Action

The following is a list of short-term corrective actions:

1) Immediate corrective action was taken by shift outage manager. After discovering the leaking electrical penetration, fuel movement was immediately halted once a fuel assembly was placed in a safe position.

2) Electrical penetration differential pressure was reduced.

3) Electrical penetration 2RCP-11E was cleaned, sealed, Q/C inspected and type B leak tested, to restore it to pre-outage condition.

4) Placed procedures that permit useage of temporary fire seals on hold.

5) Any usage of a temporary fire seal or penetration will be evaluated on a case-by-case basis (50.59 evaluation).

The following is a list of long-term corrective actions:

1) Provide permanent seals, to the extent possible, in locations whose temporary seals are installed.

2) Placed the use of temporary seals on hold until an evaluation of the Temporary Seal Program is complete.

3) Evaluate Tech Spec for potential revisions.

c. Licensee Performance

The installation of the temporary electrical penetration 2-RCP 1E was performed by plant construction personnel. Plant Maintenance was rated a SALP 1 during the last SALP period.

There have been no violations or LER's relating to failure to comply with technical specification 3.9.4 due to a failure of a temporary electrical penetration.

d. Prior Opportunity to Identify

On March 24, 1992, the Quality Services Department identified lack of adequate specifications and procedures to support the installation of temporary fire seals at BVPS. The follow up corrective action to resolve this concern was in progress at the time of this event. EM 102197 was issued just prior to the actual event. The installation of this temporary electrical penetration would have been identified as being deficient and corrected in a timely manner if this event would not have occurred.

A database search of NRC Bulletins, Circulars, Information Notices and Generic Letters has been performed to determine if the NRC has provided information relevant to this apparent violation. There have been no notifications related to failure to meet technical specification 3.9.4 as a result of a temporary electrical penetration (seal failure).

e. Multiple Occurrences

There have not been multiple occurrences of the violation cited.

f. Duration

The condition, where electrical penetration 2RCP-11E was not in a configuration which complied with technical specification 3.9.4, lasted 7.5 hours (best estimate) and less than 72 hours (worst case).

The failure of electrical penetration 2RCP-11E could not be detected from any instrumentation or alarms which are available in the control room.

The location of the electrical penetration is not in a heavily traveled area.

The failure of the electrical penetration is not easily detected without a follow-up investigation.

Emergency Diesel Generator Sequencing Relay Failure

1. Background Description of Relay Operation
2. Sequence of Events
3. Diesel Generator Relay Modification
4. Failure Modes, Effects and Consequences
5. Events and Causal Factor Chart
6. 10 CFR Part 2 Appendix C Discussion

BPENFRC.PR2

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Emergency Diesel Generator Sequencing Relay Failure

Background Description of Relay Operation

J.W. Turner

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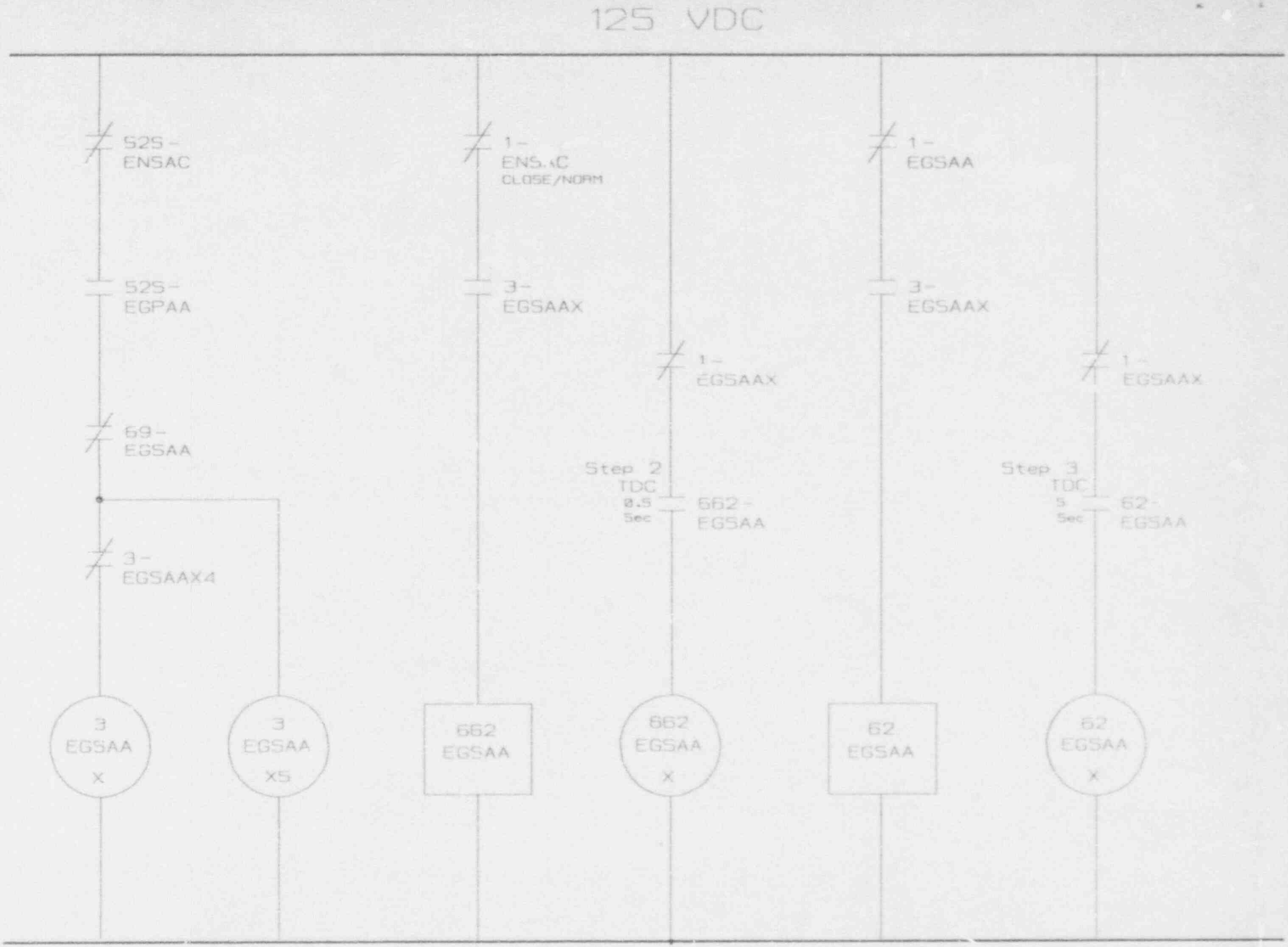
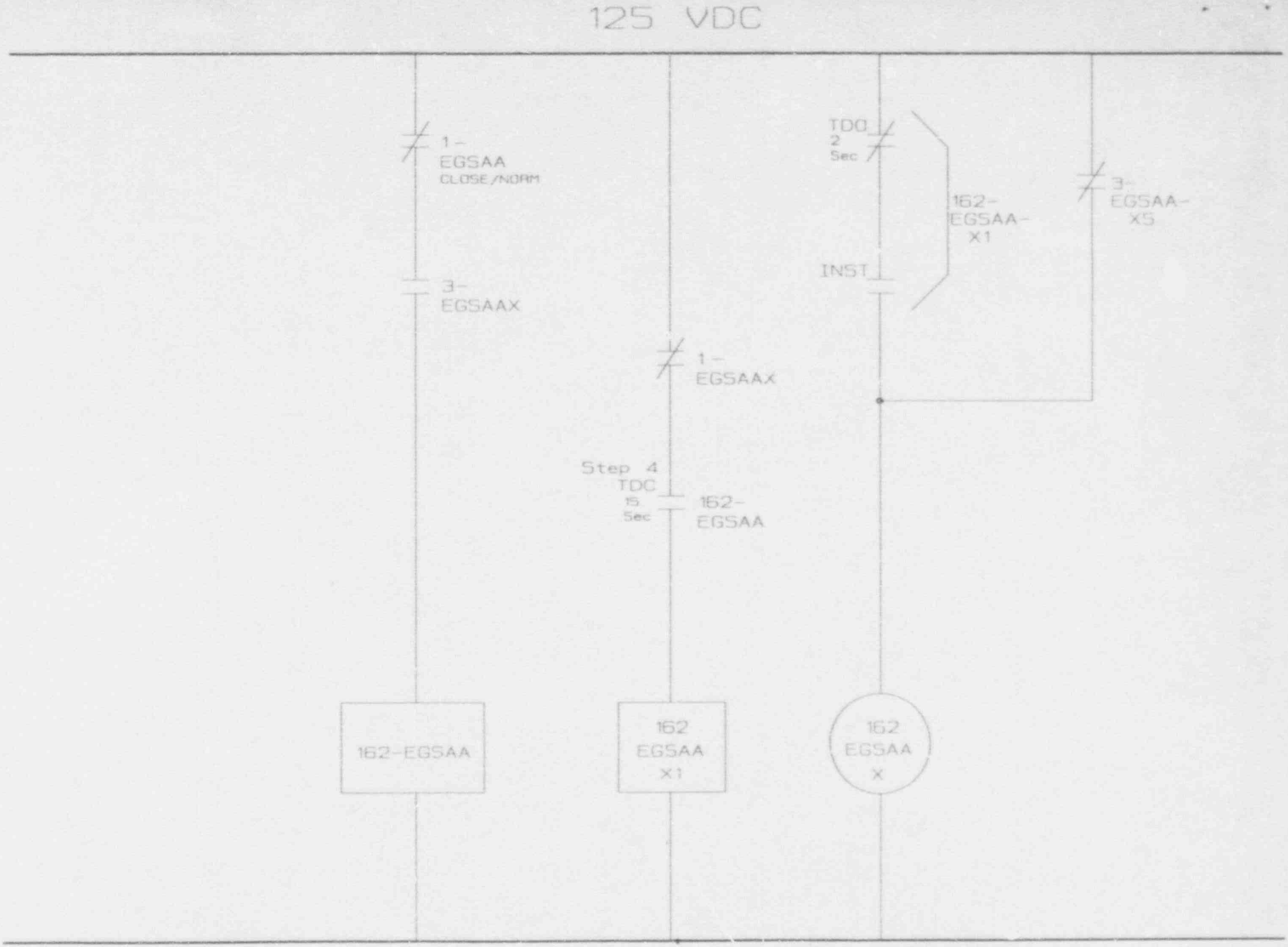
UNIT 2 SEQUENCER OPERATION

TIME (SEC)	STEP	RELAY	FUNCTIONS
-10	E7 Open	52-X1,X2,X3	BLOCK AUTO STARTS
0	1	3--X1,3--X5	START TIMERS, MORE BLOCKS
0.5	2	662	CHSpp, MCC E1,3,5,11,15
5	3	62	SWSpp, SISpp, MCC E7
15-17,60	4	162	FWE, QSS
20	5	362	HVS, RSS
40	6	462	CCP, HVZ, HVR, MCC E9,13
60	R	562	Lock in Step 4, Reset Timers, Enable SWE

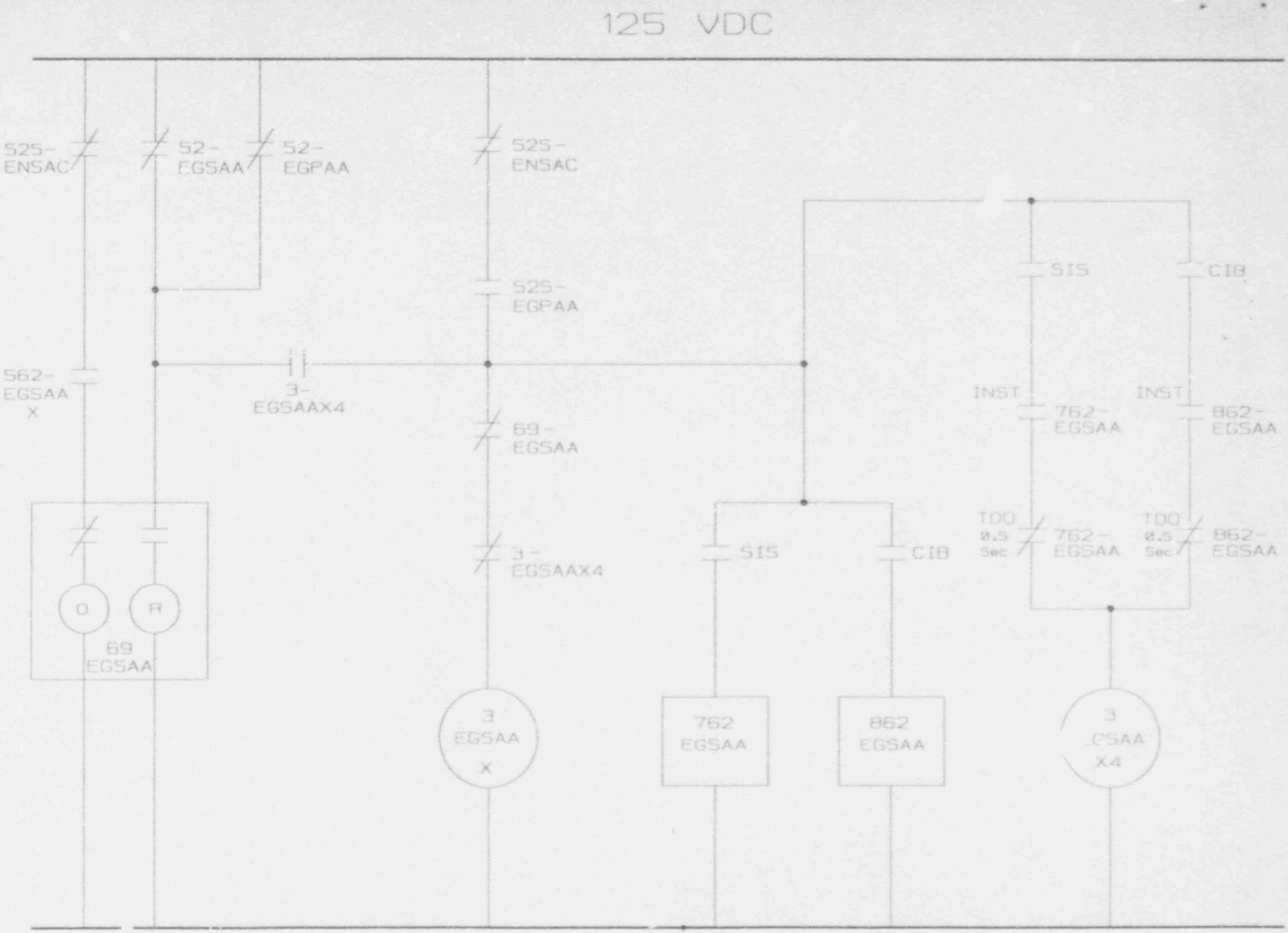


BASIC SEQUENCER CIRCUIT TYPICAL OF STEPS 2,3,5,6



BASIC CIRCUIT FOR SEQUENCER STEP 4

SK11



SEQUENCER RESET CIRCUITS

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Emergency Diesel Generator Sequencing Relay Failure

Sequence of Events

G.S. Thomas

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SEQUENCE OF EVENTS

4/21/89	EDG SEQUENCER RELAY TEST REPEATABLE TOLERANCE DIFFICULTY IDENTIFIED
8/1/89	EM 64656 ISSUED TO ENGINEERING TO ADDRESS EDG SEQUENCER TIMER CALIBRATION TOLERANCE PROBLEMS
5/3/90	DCP 1545 INITIATED TO REPLACE 16 EDG SEQUENCER TIMER RELAYS BASIS: EXISTING ELECTRO-MECHANICAL RELAYS WERE NOT ABLE TO BE CALIBRATED REPEATABLE TO WITHIN TECH SPEC TOLERANCE REQUIREMENTS
5/17/90	PERFORMED PRE-INSTALLATION TESTING OF NEW RELAY UNIT

6/1/90	ISSUED DCP 1545 DRAWINGS FOR CONSTRUCTION *10 UNITS WIRED WITH JUMPER *6 UNITS WIRED PER ATC TELECON RECOMMENDATION
8/03/90	PURCHASE ORDER D091625 ISSUED TO WYLE LABS FOR RELAYS AND UPGRADE/QUALIFICATION PROGRAM TO IEEE 323-1974 AND IEEE 344-1975
10/22/90	DCP 1545 INSTALLATION AND TESTING COMPLETED (2R)
3/30/92	TESTED ORANGE TRAIN (DG 2-1) RELAYS AND FOUND 3 UNITS INOPERABLE (ALL WIRED PER TELECON RECOMMENDATION)
3/30/92	ASSESSMENT OF IMPACT OF DG 2-2

3/31/92	ISSUED DESIGN CHANGE 1870 TO REVISE WIRING OF THE 6 RELAY UNITS TO JUMPER (PWR CKT TO CLOCK CKT) CONFIGURATION
4/2/92	REPLACED FAILED ORANGE TRAIN UNITS WITH REWIRED UNITS
4/6/92	RECEIVED PARTIAL CKT DESIGN INFORMATION (PROPRIETARY) FROM ATC FOR REVIEW OF PROBLEMS FOUND
4/8/92	TESTED PURPLE TRAIN (DG 2-2) RELAYS AND FOUND 3 UNITS INOPERABLE
4/8/92	REPLACED 3 PURPLE TRAIN FAILED UNITS WITH REWIRED UNITS

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May 19, 1992

Emergency Diesel Generator Sequencing Relay Failure

Diesel Generator Relay Modification

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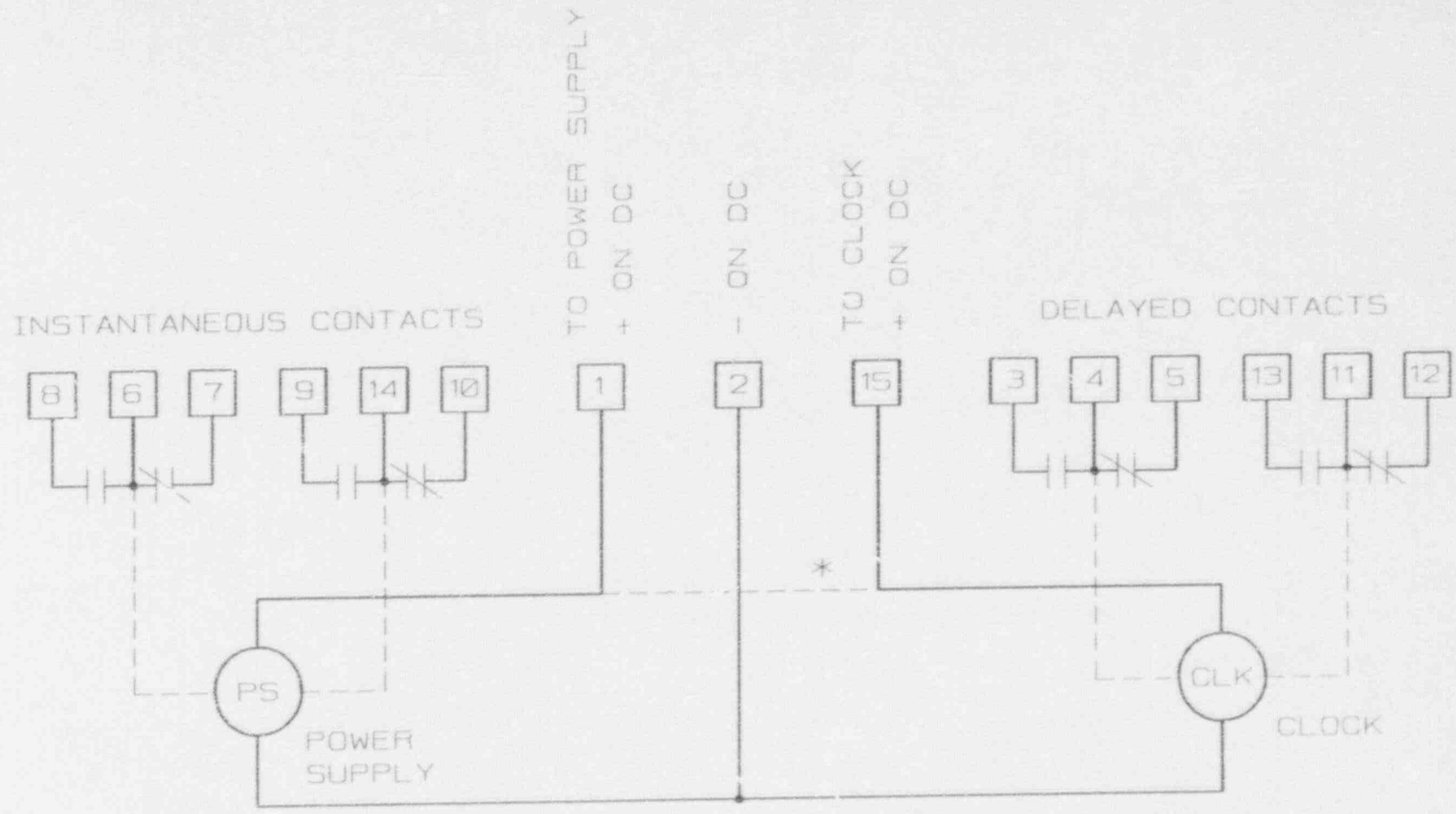
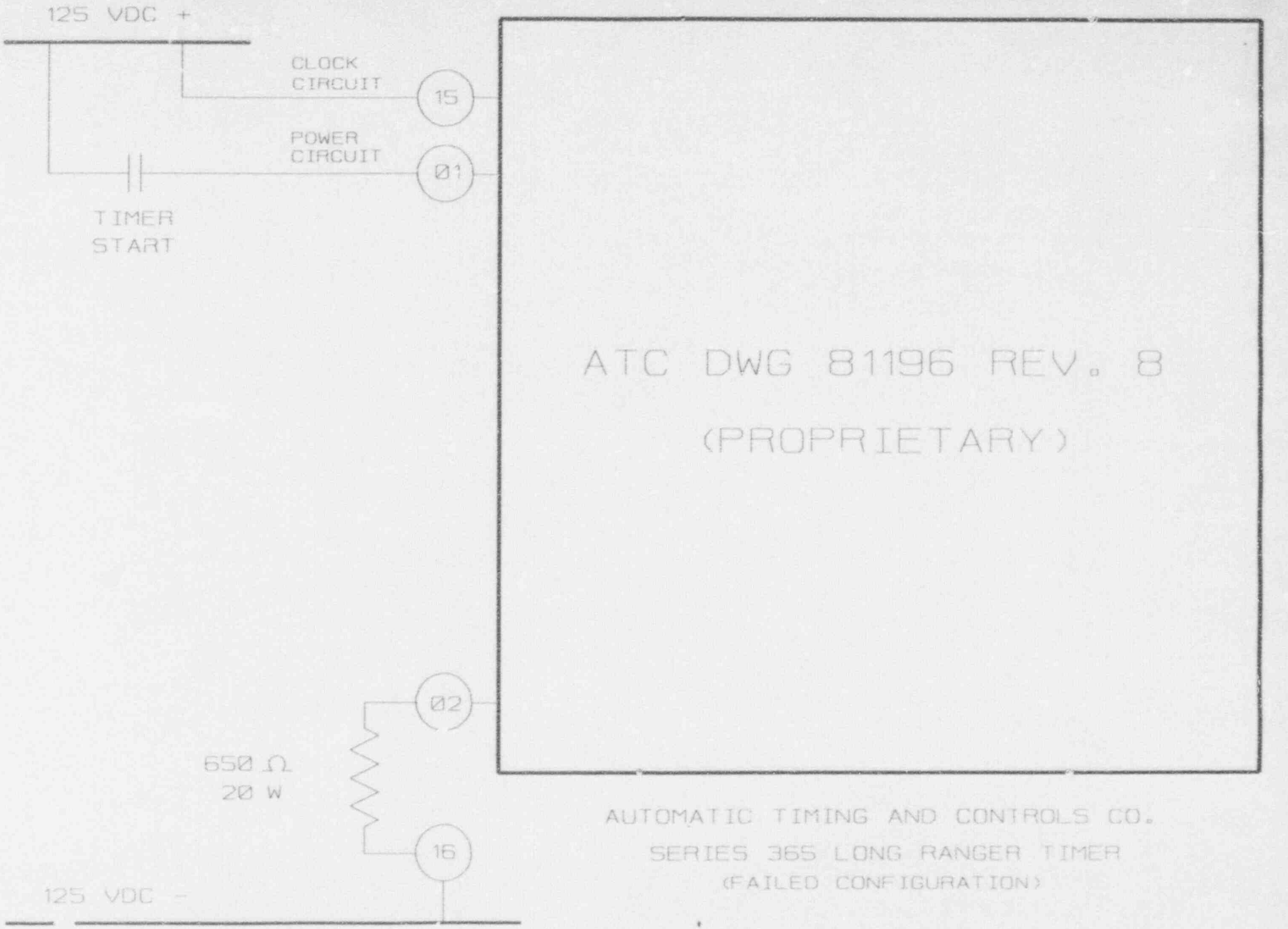
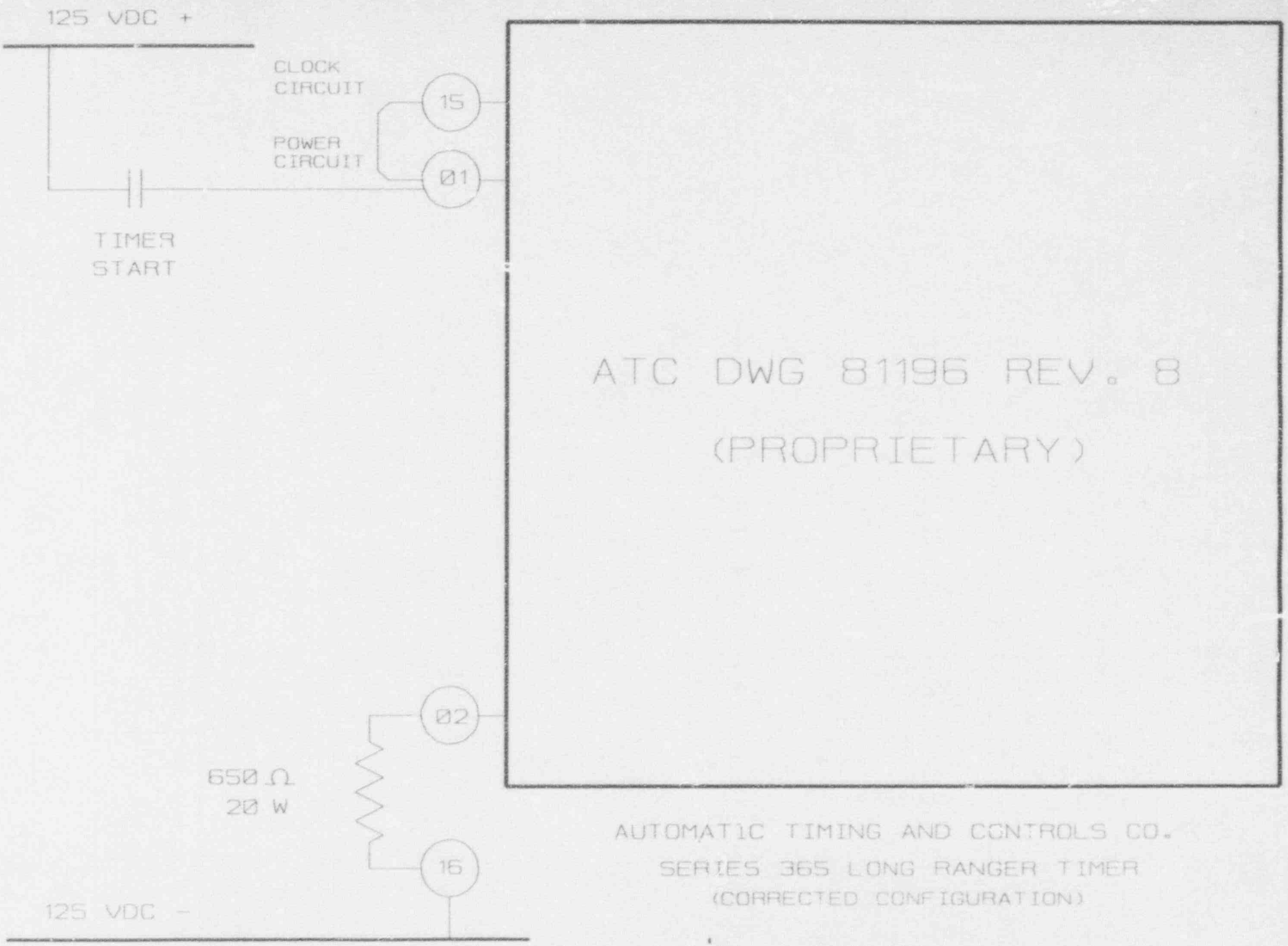


WIRING DIAGRAM

* DISPLAY MODEL ONLY. CLOCK PREWIRED TO TERMINAL ON NON DISPLAY MODEL


125 VDC +
CLOCK CIRCUIT
15
POWER CIRCUIT
01
TIMER START
ATC DWG 81196 REV. 8
(PROPRIETARY)
02
650 Ω
20 W
16
125 VDC -
AUTOMATIC TIMING AND CONTROLS CO.
SERIES 365 LONG RANGER TIMER
(FAILED CONFIGURATION)
SK14



AUTOMATIC TIMING AND CONTROLS CO.

SERIES 365 LONG RANGER TIMER

(CORRECTED CONFIGURATION)

SK35

CONCLUSION

A revision to the circuit was recommended by the original vendor and was accepted with minimal retesting of the proprietary circuitry. The qualified circuit had been tested in the 125 VDC application; and the original vendor indicated that the rewired configuration would not affect the operation.

The circuit was bench tested to verify improved and repeatable tolerances to the calibration requirements. The post installation testing and pre-operational functional testing exercised the circuit in the rewired configuration and provided no indication of a circuit deficiency.

Subsequently, this configuration has been bench tested since April 03 with 140VDC applied continuously to both the power and the clock circuits. At an inspection performed on May 18, only a slight discoloration around the area of the resister has been identified. The operation of the circuit has been regularly verified.

CORRECTIVE ACTIONS

COMPLETE CIRCUIT DIAGRAMS WILL BE REQUESTED FROM VENDORS THAT SUPPLY ELECTRICAL/ELECTRONIC EQUIPMENT WHICH HAS UNSPECIFIED INTERNAL CONFIGURATION (BLACK BOX) CIRCUITRY

PROCEDURES WILL BE ENHANCED TO INSURE THAT MODIFICATIONS TO BLACK BOX CIRCUITS ARE ADEQUATELY EVALUATED

DESIGN CHANGES PERFORMED OVER THE PAST FIVE YEARS WILL BE REVIEWED TO:

* IDENTIFY ANY MODIFICATIONS MADE TO BLACK BOX CIRCUITS
* VERIFY THAT THE MODIFICATIONS HAVE BEEN PROPERLY EVALUATED

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Emergency Diesel Generator Sequencing Relay Failure

Failure Modes, Effects and Consequences

J.W. Turner

BPENFRC.PR2

FAILURE MODE	EFFECT	CONSEQUENCE
ALL RELAYS FAIL	STEP 4 DELAY OF 45 SECONDS ON BOTH TRAINS	ALL ANALYSIS LIMITS CONTINUE TO BE MET WITH 45 SECOND DELAY ON INITIATION OF AFW AND QUENCH SPRAY
ALL RELAYS FAIL - 1 TRAIN RESET TIMER CLOCK FAILURE INSTANTANEOUS CONTACT OK ON OTHER TRAIN	STEP 4 DELAY OF 45 SECONDS 1 TRAIN, SEQUENCER FAILURE ON OTHER TRAIN	SAME AS ABOVE (ANALYSES ASSUME SINGLE FAILURE OF 1 TRAIN)
RESET TIMER CLOCK FAILURE INSTANTANEOUS CONTACT OK BOTH TRAINS	SEQUENCER FAILURE ON BOTH TRAINS	OPERATOR ACTION REQUIRED FOR MITIGATION

ALARMS/INDICATIONS OF SEQUENCER FAILURE

- 82% of the Lights on the BISI panel Light (Bypassed and Inoperable Status Indication System).
- 31 Components with RED SI marks have their position indicators extinguished.
- 39 Components with ORANGE CIA marks have their position indicators extinguished.
- 41 Components with BLUE CIB Marks have their position indicators extinguished.
- Many Loss-of-power alarms associated with the de-energized components.

Operators Response

- Manually start ESF equipment as they proceed through the E-0 procedure.

- Manually energize the 480V Motor Control Centers to verify the system alignments as directed by procedure E-0.

- Simulator Scenarios were run that confirmed that the above operator actions take place.

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Emergency Diesel Generator Sequencing Relay Failure

Event and Causal Factor Chart

K.D. Grada

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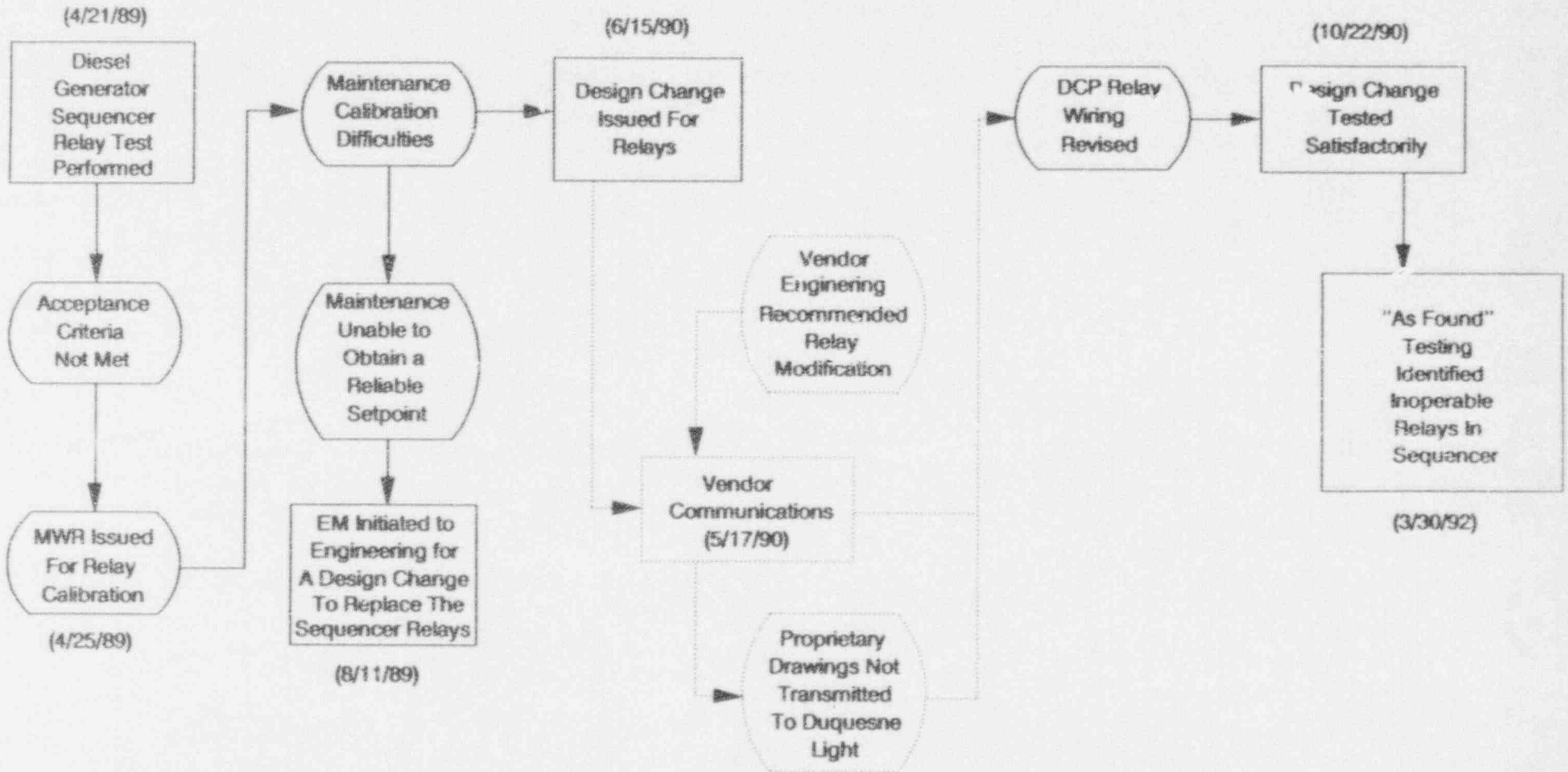
Diesel Generator Sequencer Relays
(LER 92-004-00)



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May 19, 1992

Emergency Diesel Generator Sequencing Relay Failure

10 CFR Part 2, Appendix C Discussion

K.D. Grada

10 CFR Part 2, Appendix C Review

Apparent Violation

Installation of ATC 365A relays for EDG 2-1 and 2-2 sequencers under an unsuitable configuration for its intended application.

a) Identification

DLC identified a potential problem with three Emergency Diesel Generator (EDG) sequencing relays for the 2-1 EDG on March 30, 1992 during periodic surveillance testing. At 1605 hours on March 30, 1992 these relays were determined to be inoperable. A four hour notification was made under 10 CFR 50.72(b)(2)(i) at 1636 hours on March 30, 1992. The plant was defueled at this time. A Licensee Event Report (LER 2-92-004) was issued on April 29, 1992.

b) Corr ctive action

Following identification of the relay problems and an engineering review of the failures, Design Change Package (DCP) 1870 was issued to modify the relay installation by replacing the failed relays and eliminating the possibility of further relay overheating failures. The relays for EDG 2-1 were replaced on April 2, 1992.

On April 8, 1992 the corresponding relays for EDG 2-2 were tested. These relays were also found either failed or degraded and were replaced per DCP 1870. This Emergency bus was out of service at this time.

Complete circuit diagrams will be requested from vendors that supply electrical/electronic equipment which has unspecified internal configuration (black-box) circuitry.

Procedures will be enhanced to insure that modifications to black box circuits are adequately evaluated.

Design changes performed over the past five years will be reviewed to

* identify any modifications made to black box circuits.
* verify that the modifications have been properly evaluated.

c) Licensee performance

There have not been any violations or LERs in the past two years relating to the installation of electrical equipment in an unsuitable configuration for its intended application.

The SALP category of Engineering/Technical Support which was rated as category 2 - Improving during the last SALP period. In addition no similar issues were identified during the recent NRC Electrical Distribution System Functional Inspection (EDSFI) in December 1991.

d) Prior opportunity to identify

The failure of the relays was identified during a surveillance test required by Technical Specification surveillance 4.8.1.1.2.[illegible] on an 18 month frequency and was the first opportunity to detect this condition. Post modification testing for DCP 1545 performed in September through October 1990 demonstrated the sequencer relays operated satisfactorily as installed.

A database search of NRC Bulletins, Circulars, Information Notices and Generic Letters was performed to determine if specific prior notification was given concerning this event. There has not been any notifications relative to installation of diesel generator sequencing relays in an unsuitable configuration.

e) Multiple occurrences

There has not been any multiple occurrences of this apparent violation.

f) Duration

The exact duration of the relay failures cannot be determined. However, testing of an identically modified relay in a continuously energized state began on April 3, 1992 and to date no failure has occurred.