U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 50-412/85-26

Docket No. 50-412

License No. CPPR-105 Priority -- Category B

Licensee: Duquense Light Company
Robinson Plaza Building No. 2
Suite #210, PA Route 60
Pittsburgh, Pennsylvania

Facility Name: Beaver Valley Power Station, Unit 2

Inspection At: Shippingport, Pennsylvania

Dates: December 3, 1985 - January 6, 1986

Inspectors:

G. A. Walton, Senior Resident Inspector — Jan. 10, 1986 date

L. J. Prividy, Resident Inspector — Jan. 10, 1986 date

Approved by:

for L. E. Tripp, Chief, Reactor Projects Section 3A — Jan. 16, 1986 date

Inspection Summary: Inspection on December 3, 1985 - January 6, 1986 (Report No. 50-412/85-26).

Areas Inspected: Routine, unannounced inspection by two resident inspectors (188 hours) of activities pertaining to previously identified unresolved items, 50.55(e) reports, deficient limitorque operators, lay-up of installed equipment, QA/QC program in proof testing activities, design/construction of main steam piping to auxiliary feedwater turbine, preservice examinations, fire protection supports and daily site tours.

Results: No violations were identified. This inspection identified a concern relative to the adequacy of Quality Control inspections on systems turned over to DLC Start-Up Group (Section 4). Previous unresolved items were further reviewed in the area of equipment qualification and testing of the Main Steam Isolation Valves (Section 3). Unresolved items were also identified concerning 50.55(e) reportability (Section 5) and the maintenance program for equipment lay-up (Section 6).

DETAILS

1. Persons Attending Exit Interview

Duquesne Light Company

L. E. Arch, Senior Project Engineer
R. Coupland, Director, Quality Control
C. R. Davis, Director, Quality Assurance
D. W. Denning, Assistant Director, Quality Control
C. E. Ewing, Manager, Quality Assurance
R. W. Fedin, Senior Project Engineer
E. J. Horvath, Senior Project Engineer
W. H. Sikorski, Director ISI
R. J. Swiderski, Manager, Startup

Stone and Webster Engineering Corporation

W. Baranowski, Assistant Project Manager
A. A. Dasenbrock, Senior Construction Manager
H. W. Durkin, Superintendent of Engineering
D. B. Lamson, Assistant Resident Engineer
R. C. Wittschen, Licensing Engineer

2. Construction Site Walk-Through Inspections

Daily tours of the construction site were made to observe work activities in progress, completed work, and plant status of the construction site. The presence of Quality Control inspectors and quality records were observed. No noncompliances were identified.

3. Licensee Action on Previous Inspection Findings

(Closed) Unresolved Item (85-14-02): Physical clearance around welds requiring preservice inspections (PSI). This unresolved item identified that the licensee had no method to assure adequate clearances were provided around PSI related welds. The following actions were taken by the licensee.

The welds requiring access for preservice inspection and subsequent inservice inspection have been identified by list to appropriate contractors. The welds are being physically identified with red tape to alert personnel of the need to maintain access in the area of these welds. The purpose of the tape was explained to appropriate installation disciplines (electrical, HVAC, fire protection, etc.) in training sessions. At present, the licensee anticipates that over 2,000 welds will be taped. These activities and existing Electrical/Mechanical specifications and ongoing Quality Control inspections will ensure weld accessibility is maintained. The inspector reviewed the licensee's program in this area and had no further questions.

(Closed) Unresolved Item (85-14-03): Physical identification of welds requiring preservice inspection (PSI). This unresolved item identified that several welds were found without weld number identification. It was found that these welds were identified on weld maps and process travelers. The licensee committed to hard mark all welds which required PSI. The licensee has taken the following actions on this item.

The DLC Quality Assurance Inservice Inspection Department has issued Procedure 3.6A, titled "Control of Preservice Inspection at Unit 2," which requires verification that the weld numbers are hard marked adjacent to each weld. This program has commenced.

The inspector reviewed the licensee's program in this area and witnessed the stamping and verification of correctness on two of the welds in the main steam and cable vault area. The inspector had no further questions on this item.

(Closed) Construction Deficiency Report (84-00-08): During factory tests of the first Colt-Pielstick PC-2.3 diesel generator set, constructed for Shoreham Nuclear Power Station, a broken boss was discovered on the engine-driven lube oil pump discharge nozzle. The engine-driven lube oil pump on the second Shoreham diesel generator set was examined by Colt, and the boss area was found to be cracked. A broken/cracked boss significantly reduces the sealing area between the pump and an adapter which screws into the boss. This condition, were it to occur on the Shoreham engines or the similarly designed BVPS-2 engines (model PC-2), could permit oil leakage sufficient to prevent operation of the affected engine. Duquesne Light Company (DLC) notified the NRC Region I Office on September 24, 1984, of this reportable deficiency.

DLC Site Quality Control issued Nonconformance and Disposition (N&D) Report 7020 on October 24, 1984, to document this problem and obtain an engineering evaluation using the site program. To correct this deficiency, the tapered threads on the lube oil pumps of the Colt diesel generator sets were replaced with adapters having straight threads with an undercut, thus eliminating the potential for overtorquing the threaded adapters. The threaded hole in the pump boss had straight threads and did not require modification.

The inspector reviewed N&D 7020 and the DLC Site Quality Control Inspection Report that was completed on June 14, 1985, documenting the satisfactory completion of the repairs. The inspector found these items acceptable. This item is closed.

(Open) Unresolved Item (84-10-04): Replacement of equipment with qualified life. This unresolved item dealt with the need of a maintenance program to assure Rc mount transmitters would be replaced within their 10 year qualified life. This inspection revealed that certain other equipment, with limited qualified life, could be outdated and require replacement before the plant goes operational. For example, the main steam isolation valve actuators contain limit switches which are qualified for five years. The instruction manual is not clear regarding the start date for this equipment. Certain equipment has a limited "shelf life" and could be beyond its qualified time before commencing operation. Other equipments' qualification date may begin on the day it goes operational.

In the case of the limit switches discussed above, a maintenance program has not been developed which reflects its expiration date. At present, the inspector was advised DLC Engineering was working with the vendors on each qualified component to establish a start date. After that is established, the DLC Maintenance Department plans to establish a maintenance program, order new components, and replace the outdated items based on their qualified life. Several items will be dated and require replacement prior to plant operation. The licensee is presently working on the above program.

This item continues open pending review of this program.

(Open) Unresolved Item (84-16-04): Repair of main steam isolation valves. After repairs were completed on the three main steam isolation valves (MSIV), the E&DCR required a hydrostatic test and a seat leakage test to be performed on the subject valves. The seat leakage test had previously failed and is discussed in Inspection Report 50-412/85-21. During this inspection period, the licensee performed both the hydrostatic test and seat leakage test on these valves. This is an update on these tests and discusses concerns raised by the inspector.

ASME Section III, Paragraph NB-6215, requires that during the hydrostatic test of valves, leaks such as from permanent seals, seats, and gasketed joints may be permitted when specifically allowed by the design specification. From discussions held with the Quality Control Inspector, who witnessed the test, leakage was observed coming from the bonnet seal and valve stem packing. However, the procedure did not require these leakage rates to be corrected and/or measured in accordance with the design specification allowances. Therefore, it is not known if the observed leakage was acceptable. Subsequent discussions with Quality Control, DLC Start-Up and Stone and Webster Engineering resulted in their issuance of a Nonconformance and Disposition Report (N&D 29244) to initiate corrective actions on this matter.

The hydrostatic test of the secondary side of the steam generators included the boundary up to the MSIV with the valves in the closed position. The leakage at the bonnet and valve stem, discussed above, was most notable when the valves were in the closed position. This indicates leakage occurred through the valve seat.

A response to the NRC Question 210.36 included in the FSAR stated: "ASME Class 2 and 3 valves are designed to ASME III requirements. The integrity of the Class 2 and 3 valve discs is assured through disc hydrostatic testing. The leak tightness for Class 1, 2, and 3 valve discs is assured through hydrostatic testing."

The inspector questioned the licensee regarding how the above statements were being achieved, particularily in light of the leakage on the MSIVs. No apparent method was used during the hydrostatic test to measure leakage across the valve seats.

After the hydrostatic test was completed, a seat leakage test using air at 47 psig was attempted. Valve C leakage was within the acceptance criteria of 4.8 SCFH (actual leakage 1.01 SCFH). Valves A and B had excessive leakage, 39.15 and 24.7 SCFH. All apparent external leakage was stopped by tightening the stem packing. It appears the excess leakage is through the valve seats. Nonconformance and Disposition Reports 8787, 8788 were issued by Quality Control to obtain engineering disposition. This item remains open pending resolution by the licensee of these concerns.

4. Lack of a QA/QC Program in Proof Testing Activities

The inspector reviewed some of the activities of various DLC personnel involved in performing proof testing. This review consisted of determining what requirements and commitments were presented in various parts of the DLC Startup Manual and how these requirements and commitments were being implemented in the proof testing activities. The inspection focused attention to current testing efforts being conducted to flush chemical and volume control system piping around the volume control tank.

The inspector received current information from the DLC Flushing Supervisor concerning the status of the performance of Test Procedure 2T-CHS-7-1.05, "Chemical and Volume Control System Flush". This flush procedure is developed from Test Procedure No. 2T-GFP-601, Revision 1, "Generic Flush Procedure Performance", which defines the documentation requirements and flushing methods to be used for the cleaning of piping, equipment, vents, drains and instrument lines. The flush procedure is the implementation of the Cleanness Verification Program as defined in DLC Startup Manual Chapter 5.11.

A key element of each flush procedure is the completion of the Cleanness Verification Report form which documents the water quality for the flush, the cleanness class of the system being flushed, and the acceptance of the flush. The inspector noted that the Cleanness Verification Report form has a signoff place in the Acceptance Block for the SQC Inspector. However, the inspector was advised by DLC - SUG personnel that DLC-SQC personnel do not participate in the acceptance of flushes. Accordingly, DLC-SUG personnel have been directed by their management to insert Not Applicable (N/A) in the signoff place of the Acceptance Block for the SQC Inspector. DLC-SUG personnel did point out to the inspector that two separate Level II personnel from DLC-SUG were required to accept and sign off acceptance of each flush. The inspector noted his concern in that this method of acceptance did not accomplish the same degree of QA/QC independence required and accomplished in other construction activities. Furthermore, it appears that this method of flush acceptance is not in accordance with the DLC QA Manual.

To further determine the extent of SQC involvement in the Cleanness Verification Program, the inspector reviewed FCP-954, Revision 1, "Field Construction Procedure for Cleanness Verification Program." The essence of FCP-954 is the Cleanness Verification Program as contained in the Startup Manual, Chapter 5.11, since FCP-954 simply refers to this chapter for the procedure. However, FCP-954 does define the responsibility for DLC-SQC to develop a program to verify that DLC-SUG and assigned craft personnel are in compliance with the requirements of FCP-954. Consistent with this responsibility, the inspector met with the Director of Quality Control and an Assistant Director to determine the program that SQC had in place to meet the requirements of FCP-954.

The inspector determined that SQC performs surveillances of the DLC-SUG flush program to fulfill their responsibilities of FCP-954. SQC Inspection Plan 11.1, "Surveillance of Cleanness Verification Program", describes the type of surveillance performed by SQC of the SUG flush activities. To date, twenty (20) surveillance reports have been generated per the guidelines of IP-11.1 with seventeen (17) of these surveillances performed on the Service Water System (7) and Primary Component Cooling Water System (10). No surveillances have been performed on the Chemical and Volume Control System yet. The inspector noted that the surveillance is procedure oriented in that it covers a procedure review, field walkdown and/or implementation of procedures. The inspector concluded that while these surveillances implemented the SQC responsibility in FCP-954 they did not constitute the level of SQC involvement as required and present in other construction activities.

The inspector met with DLC QA Manager and the Nuclear Group - Vice President, to explain this concern. The inspector stated that the proof testing activities conducted by DLC - SUG need to be reviewed from a QA/QC standpoint and QA/QC personnel need to define an acceptable inspection program so that QA/QC personnel have the opportunity to be involved in the acceptance of proof testing activities. This acceptable inspection program needs to be developed jointly by the QA and SUG groups with each department having an authoritative voice in decision-making and with the decisions being made within the overall guidance of the DLC-QA Manual. The DLC QA Manager and Nuclear Group - Vice President acknowledged that such a program does not now exist. Rather as previously stated, DLC-SUG personnel accept and sign off acceptance of each flush and accept/reject authority rests solely with DLC-SUG. This item is unresolved pending the development and implementation of an acceptable inspection program so that QA/QC personnel are involved with acceptance of proof testing activities (85-26-03).

5. Compliance with 50.55(e), Reportability Requirements for Rework of Limitorque Operators.

NRC Information Notices 82-52, 83-72 and 84-78 identified numerous deficiencies associated with environmental qualification test failures. One particular area identified by the notices was deficient limitorque actuators. To investigate this concern, on August 6, 1984, Engineering Field Action Report No. 216B was issued by Stone and Webster Engineering to perform sample inspections on thirteen (13) valves selectively picked to cover the four suppliers of the actuators. The results of this inspection, which was completed by March 19, 1985, disclosed deficiencies with each of the actuators. Some of the deficiencies were: cracked fingers on limit switches, loose wires on lugs, lugs incorrectly crimped, unqualified motor insulation, unqualified terminal strips, and electrical connections other than ring lug. A Nonconformance and Disposition Report (N&D) Number 4929A was issued on March 18, 1985, to obtain disposition. On September 19, 1985, the N&D was dispositioned.

Corrective action included: scrap damaged limit switch and replace with a new switch, scrap motors with unqualified insulation and replace with new motors, scrap unqualified terminal strips and replace with qualified terminal strips, and scrap electrical connectors other than ring lugs and replace with ring lugs. At that time, the licensee also initiated action to require inspection of all limitorque operators. The inspector performed a review of the associated documentation of these deficiencies to determine compliance with 10 CFR 50.55(e). As stated in 10 CFR 50.55(e)(3), the holder (Licensee) of a construction permit shall within 24 hours notify the NRC of each reportable deficiency and shall also submit a written report within thirty days to the NRC.

Stone and Webster's N&D program requires evaluation to determine if the item requires 10 CFR 50.55(e) evaluation. In the case of N&D 4929A, the appropriate block was marked "Not Required." From discussions held with Stone and Webster Engineering, another method of evaluation for reportability exists. A "Report Of A Problem" (ROAP) and "Initial Problem Report" (IPR) were generated and are evaluated for reportability. The ROAP was issued on March 22, 1985, and the IPR (51126) was issued April 22, 1985.

The IPR-51126, Supplement 1, and ROAP-MEC-077 are included as Attachment II. As shown on the IPR, an evaluation of similar problems at other nuclear facilities determined the condition is potentially reportable to the NRC under 10 CFR 50.55(e). From review of an August 20, 1985, interoffice memorandum, included as Attachment I, no specific evaluation of the Beaver Valley, Unit 2, deficiencies were required. The inspector was advised that evaluation had been performed and the item was determined to be "Not Reportable." However, no documentation was presented which supports this analysis.

The inspector advised the licensee that this item is unresolved pending further evaluation (85-26-01).

6. Maintenance Program for Lay-Up of Installed Equipment

The inspector reviewed the licensee's program for lay-up (storage) of equipment (valves, heat exchangers, etc.) turned over to the Start Up Group that are not being tested or operated. The equipment selected was the main steam isolation valves. Since the recent completion of hydrostatic testing, no further actions are planned for the valve internals until hot functional testing commences. The vendor maintenance manual for these valves requires specific lay-up controls if the valve will be inoperative for more than one month. The recommendations for lay-ups are either wet, by filling the valve cavity with demineralized water containing a rust corrosion inhibitor, or dry, with energized electric heaters and desiccant.

At present, the Start Up Group has no specific maintenance procedures to assure proper lay-up of this type of equipment. The Start Up Group advised the present method used to place equipment in lay-up is for the Systems Engineer to issue an Interoffice Correspondence (IOC) to the Maintenance Supervisor. Then, the Maintenance Supervisor would establish the necessary lay-up controls. However, no previously issued IOCs exist for lay-up of Category 1 components, including the MSIVs.

In subsequent discussions on this matter, the Start-Up Maintenance Supervisor advised the MSIVs would be properly layed-up. Further, a procedure would be established to control all items that require special lay-up that are installed and not being tested or operated. This item is unresolved pending review of this maintenance program. (85-26-02).

7. Acoustic Comparison of Preservice Calibration Blocks

The licensee commenced preservice examinations using ultrasonics on piping welds before the calibration blocks being fabricated for Beaver Valley, Unit 2 (BV-2) were completed. As permitted by ASME Section XI, Calibration, blocks of similar composition to the parent material must be used to establish calibration. Therefore, the licensee obtained calibration blocks that were similar in composition from other nuclear facilities to establish calibration sensitivity and allow examinations to be done before the permanent BV-2 blocks were completed. The decision to begin testing before the BV-2 calibration blocks were finished was made to allow examination of the welds on the feedwater and other associated pipe welds prior to performing the system hydrostatic test on these welds.

After the BV-2 calibration blocks were received on site, the licensee's inspection contractor performed attenuation checks between the block used for the test and the new blocks which will be used during future inservice examinations. Two calibration blocks identified as UT-16-2 (4" sch 60XS) and BV-1-60 (16" sch 80) were used to perform the test. They are being replaced with calibration blocks PP-18 (4" sch 60XS) and PP-12 (16" sch 80).

The acoustic comparison test performed on December 23, 1985, was witnessed by the inspector and the Authorized Nuclear Inspector. The test was made using 45 degree shear waves. The calibration was made in accordance with Procedure UT-302. A distance amplitude curve (DAC) was constructed from the V-notches in the blocks. Then, a comparison of amplitude (sensitivity) was made between the blocks. The correlation of amplitude between blocks must be within 2 db to be considered acoustically similar.

The licensee's inspection contractor concluded from the comparison that with the exception of a slight thickness variation between the blocks, the shape and slope of the DAC curves are the same and the db settings were within 2 dbs.

The inspector witnessed the comparison test, reviewed the documentation, and found all areas reviewed acceptable. No items of noncompliance were identified.

8. Preservice Examinations

On December 23, 1985, the inspector witnessed ultrasonic calibration and examination of weld number 2-SIS-071-F04 and liquid penetrant examination of weld number 2-SIS-071-F4A. The welds are ASME Class 1 located on the safety injection system in the Containment Building. The welds were examined to Procedure UT-303, Revision 0 and LP-101, Revision 0.

The inspector witnessed the ultrasonic examination to ascertain compliance with the Procedure, UT-303 Revision 0, in the following areas:

- The type of apparatus used, including frequency range as well as linearity and signal attenuation accuracy is within the limits specified in the approved procedure.

- The extent of coverage (beam angles, scanning, surface, scanning rate and directions) as well as the scanning technique is within specified limits.

- Calibration, methods and frequency including the type, size, geometry and material of identified calibration blocks as well as location and size of calibration reflectors within the block are clearly determined and recorded.

- The sizes and frequencies of search units are as specified.

- Beam angle or angles is as specified.

- Methods of compensation for the distance traversed by the ultrasonic beam as it passes through the material including distance is as specified.

- The reference level for monitoring discontinuities is as defined and the scanning gain setting is as specified.

- Levels or limits for evaluation and recording of indications is accomplished.

- Method of recording significant indications is acceptable.

- Acceptance limits are determined.

The inspector witnessed the liquid penetrant examination to assure the following areas were consistent with the approved procedure and ASME Section XI:

- The specified examination method is consistent with the procedure and consists of color contrast.

- The penetrant, penetrant remover, emulsifier and developer are identified and consistent with ASME Code Section V.

- Penetrant materials used for the examinations of austenitic stainless steel were analyzed for total halogens. The total residual halogen content does not exceed the established limits.

- The surface is dry prior to developing.

- The type of developer, method of developer application, and the time interval between penetrant removal and developer application.

- Examination technique and time interval between developer application and evaluation.

- Technique for evaluation of indications.

- Reporting of examination results.

All areas reviewed by the inspector were found acceptable. No items of noncompliance were identified.

9. Design and Installation of Seismic Supports

The inspector audited the records and drawings on three supports to determine compliance with Regulatory Guide 1.29 "Seismic Classification." RG 1.29 requires the first support beyond the boundaries of a safety-related isolation to be designed, constructed, and inspected as safety-related. The inspector selected supports PSR-550T, PSST-765 and PSR 719Y for audit. Each support listed is the first support beyond the safety boundary.

The inspector found all three supports were designed, constructed and Quality Control inspected as "Seismic Category II." Seismic Category II and Category I are controlled the same.

All areas reviewed were found acceptable and no items of noncompliance were identified.

10. DLC Start-Up Group Activities to Support Steam Generator Hydrostatic Test

The inspector reviewed some of the activities involving DLC-SUG personnel as they performed the steam generator hydrostatic test. This test was performed in accordance with Construction Proof Test Procedure No. 2T-MSS-21-2.01, "Hydrostatic Test of Steam Generators," during this inspection period. The inspector met with various DLC-SUG personnel who directed and performed this test and with Schneider Power Corporation (SPC) personnel who assisted DLC-SUG in the test performance. The inspector's review occurred after the hydrostatic test pressure had been attained and the associated inspection of the system had been conducted. The inspector's review was confined to the test personnel's activities related to disassembly/reassembly and operation of certain valves to support the test.

Various safety-related check valves had to be disassembled to remove the valve internals. This enabled test pressure to be applied to system piping which would not have been pressurized due to check valve action. For example, the internals were removed from the 16-inch check valves immediately downstream of the feedwater isolation valves. In discussions held with SPC personnel, the inspector was advised that such disassembly work was conducted in accordance with Field Construction Procedure 302, "Removal or Disassembly/ Reassembly of Permanent Plant Equipment." The inspector noted that SPC had properly executed valve disassembly and reassembly records for this work in accordance with FCP-302. The disassembly work was accomplished in accordance with instructions contained in the vendor manual and coordinated with DLC-SQC. The valves will be reassembled using this procedure.

The inspector determined from the DLC-SUG Test Director and other DLC-SUG personnel that temporary measures were needed to enable operation of the feedwater isolation valves, 2FWS-HYV 157A, B, C. The test procedure required that these valves be cycled during the procedure. Since the electrical control circuitry for the valve actuator was incomplete, DLC-SUG devised a temporary source of hydraulic power to cycle the valves. This installation consisted of high pressure tubing, valves, regulators and nitrogen bottles to pressurize the actuator hydraulic system. The DLC-SUG personnel consulted the valve instruction manual and drawings to determine a suitable installation and to provide an acceptable temporary operating procedure to cycle these valves. The inspector was advised by DLC-SUG personnel that the operating personnel were briefed properly on how to utilize this temporary installation for operating valves 2FWS-HYV 157A, B, C. No problems occurred with stroking the valves and stroking times were approximately 30 seconds.

The inspector found the review of these activities to be acceptable.

11. Fire Protection Supports

Although the fire protection system is not designed and constructed as Category 1 "Safety Related", the equipment is routed through safety-related buildings and around safety-related equipment and must be supported in accordance with seismic requirements. To ascertain this, the inspector selected supports 2FPD-PSA-037 and 2FPD-PSR-023 located in the Auxiliary Building and performed reviews for seismic design, construction and inspection.

Both supports were designed and constructed as seismic Category II. Quality Control inspections were performed on June 1, 1984, and June 19, 1984, to an approved procedure.

All areas reviewed were found acceptable and no items of noncompliance were identified.

12. Vertical Mounted Gould Pumps

Recent information from another nuclear site revealed a possible problem with Gould supplied vertical mounted pumps because of inadequate thread engagement in the motor hold down bolts. For corrective action, the vendor recommended the following:

> It is Goulds' recommendation that motor hold down bolts be changed to 2-1/2 inch length. Using the dimensions reported to be +field+ dimensions, the resulting thread engagement should be approximately 0.746 inches. The minimum thread engagement required for these bolts of this material is 0.551 inches. Therefore, there will be ample margin.

The inspector inquired if this condition existed with any Gould pumps supplied on site.

The licensee has advised they have no vertical mounted Category 1 Gould pumps. They do have two (2) Category 1 pumps mounted horizontal which they inspected to the above criteria and found acceptable.

The inspector had no further questions on this matter. No items of noncompliance were identified.

13. Design and Construction of Main Steam Piping to Auxiliary Feed Pump Turbine

The inspector expressed a concern to DLC Nuclear Construction Division personnel concerning the design and construction of the main steam piping to the auxiliary feed pump turbine. The inspector's concern was that unless specific design and construction measures were taken, there appeared to be a good possibility that water hammer could occur in the steam piping to the auxiliary feed pump turbine when it is called upon to emergency start and fulfill its safety function. There is an approximate 200 feet of 3-inch piping with associated valves from the point where main steam enters this piping at the 800' elevation in the Main Steam and Cable Vault Area to the basement (approximate 700' elevation) of the Safeguards Building where the auxiliary feed pump turbine is located. This large piping network is normally at ambient temperature since each 3-inch line off each 32-inch main steam lead has two normally closed, solenoid-operated valves which are located at the 800' elevation in the Main Steam and Cable Vault Area. If steam is rapidly discharged into this relatively cold piping, it would condense and possibly cause water hammer, vibrate the piping, and possibly damage the piping and turbine. The inspector noted that this is not a problem at Unit 1 since the arrangement is significantly different with much shorter piping runs.

After expressing this concern, the inspector was contacted by DLC Nuclear Construction Division and Stone and Webster Engineering personnel. These personnel told the inspector that drip pots were included in the 3-inch main steam piping arrangement to collect and discharge the condensate from the main steam piping so that water hammer would not be a problem. These drip pots are enlarged sections of piping (8-inch and 10-inch) located at about six different points along the 200-feet run of 3-inch main steam piping where elevation and other directional changes occur. An inlet to each drip pot comes from the 3-inch main steam piping and the drip pot outlet is directed to atmosphere or the main condenser. Stone and Webster indicated that these drip pots were sized to accommodate a 25% margin above the maximum amount of condensate to be expected during an emergency startup of the auxiliary feed pump turbine. Although detailed operating procedures have not been developed yet, the inspector was advised that during emergency startup of the auxiliary feed pump turbine, the drip pots would discharge to atmosphere. During routine monthly surveillance tests to check auxiliary feed pump operation, the drip pot discharge would be directed to the main condenser. After these routine tests, operational steps would be routinely taken to ensure that the 3-inch main steam piping leading to the turbine is drained free of water.

The inspector found the review of this item acceptable.

14. Exit Interview

A meeting was held with the licensee's representatives indicated in Paragraph 1 on January 6, 1986, to discuss the inspection scope and findings.

ATTACHMENT 1

INTEROFFICE MEMORANDUM

▲ 040.26

J.O. OR W.O. NO. 85/481

SUBJECT IPR-51126, SUPPLEMENT 1
VARIOUS PROBLEMS WITH LIMITORQUE ACTUATOR HARDWARE

DATE August 20, 1985

FROM WMEifert

TO See Distribution

CC ARJoyce
JGRosen
GMSchierberg
DPLopaus:jmm
Task # 2050
1940
1822

The attached BV-2 Report of a Problem (ROAP), NRC IE Information Notice No. 85-22, and River Bend EAP 16.2 First Reviewer evaluation have been reviewed by the Power Division, the Electrical and Control Systems Division and the Quality Assurance Department resulting in a recommendation that this material be issued for your information and use.

The Electrical Division Problem Report Coordinator has determined that all current nuclear projects have detailed inspection programs to address the various valve operator concerns.

Procurement Quality Assurance has indicated that the majority of the concerns would have been identified using attributes listed in applicable master and project inspection plans. Additionally, SWEC does not perform shop inspection for BV-2. PQA believes, since the concerns identified in the BV-2 ROAP were the results of an inspection of randomly selected installed actuators, that many of the problems may potentially have been caused during installation, handling, storage or protection during installation.

Advisory Operations Technical Guideline No. 2.15.1-1 (Motor Operated Valves) provides guidance for the testing of Motor Operator Valves and addresses the concerns cited in the BV-2 ROAP.

No response to this IPR is required.

W.M. Eifert
Chief Engineer
Engineering Assurance

Attachments

ATTACHMENT 2

REPORT OF A PROBLEM

MEC-077

TO: ~~P. RaySircar~~ W. EIFERT.

PAGE 1 OF 2

DIVISION/PROJECT REPORTING PROBLEM: Power - BVPS-2

JOB NO. 12241

CLIENT: Duquesne Light Co. STATION: BVPS UNIT: 2

DESCRIPTIVE TITLE FOR PROBLEM

Limitorque Actuator Deficiencies

REFERENCES

SYSTEM: N/A DRAWING/SPEC.: N/A

EQUIPMENT: Various Mot. Oper. Valves OTHER: See Attachment

QA CAT: 1 & 2

HOW PROBLEM WAS DISCOVERED AND DATE IDENTIFIED, PROBLEM SUMMARY, ACTION TAKEN TO DATE

Various problems related to Limitorque actuators have been identified in the industry which could affect the Beaver Valley Power Station - Unit 2. Findings reported by some of the utilities have been the subject of NRC notices and bulletins over several years.

The problems center around actuator subcomponents such as, but not limited to limit switches, internal wire, lugs, terminal boards, grease, etc. As a result of these problems, an inspection of 13 randomly selected actuators installed at BV-2 was initiated. This inspection was performed at the end of the year, 1984.

Below is a list of the deficiencies identified as a result of the inspection.

1. Cotter Pins incorrectly bent
2. Corrosion on Limitswitch
3. Oil on Limitswitch
4. Broken dividers on motor heater terminal strip
5. Cracked fingers on Limitswitch
6. Two wires on same lug
7. Loose wires on lugs
8. Lugs incorrectly crimped
9. Loose fasteners
10. Unqualified motor insulation
11. Unqualified terminal strips
12. Electrical connection other than ring lug
13. T drain not provided
14. Terminal barrel crushed
15. Wrong grease in Limitswitch intermitent gear box

Based on discrepancies noted above, an N&D is being prepared with an intended disposition of inspecting all Limitorque actuators for the BV-2 Project.

(ATTACH ADDITIONAL SHEETS IF NECESSARY)

DIVISION MOST INVOLVED WITH PROBLEM (RESPONSIBLE DISCIPLINE): Power

OTHER DIVISIONS NOW/MAY BE INVOLVED: Electrical and Controls

(PREPARED BY)	(DATE)	(LOCATION)	(DIV./DEPT.)	(EXT.)
G. R. Prunier	3/14/85	245/8	46	6631

REVIEWED BY: [signature] TITLE: PE DATE: 3-22-85

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REPORT OF A PROBLEM

MEC-077

Attachment to Report Of A Problem
Limitorque Actuator Deficiencies

IE Notice No. 83-72 (IPR-51148, ROAP-84002)
ENVIRONMENTAL QUALIFICATION TESTING EXPERIENCE

IE Notice No. 82-52 Equipment IPR 50772, Supp. 1
ENVIRONMENTAL TESTING EXPERIENCE

IE Notice No. 84-78 Underrated
Terminal Blocks on Limitorque Actuator IPR 51187
Model SMC-04

IPR-51126 (ROAP-84004) Damage to
Limitorque Motor Operated Limitswitch
Compartment Internals

SSINS No.: 6835
IN 85-22

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

March 21, 1985

IE INFORMATION NOTICE NO. 85-22: FAILURE OF LIMITORQUE MOTOR OPERATED VALVES RESULTING FROM INCORRECT INSTALLATION OF PINION GEAR

Addressees:

All nuclear power reactor facilities holding an operating license (OL) or a construction permit (CP).

Purpose:

This information notice is provided to alert recipients of a potentially significant problem pertaining to the incorrect installation of pinion gears in Limitorque motor-operated valves. It is expected that recipients will review the information for applicability to their facilities and consider actions, if appropriate, to preclude a similar problem occurring at their facilities. However, suggestions contained in this notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On December 21, 1984 and February 20, 1985, the Tennessee Valley Authority reported [Licensee Event Report (LER) 84-013] failure of the outboard high pressure coolant injection (HPCI) valve to open at the Browns Ferry Nuclear Power Station Unit 3. Operators observed the failure while attempting to perform an operability surveillance on the HPCI system.

An inspection of the Limitorque operator revealed that the pinion gear had been installed in a reversed position. This reversed installation resulted in only about one third of the normal gear mesh surface and the complete wearing away of the portion of the pinion gear teeth that were in contact with the shaft (drive) gear. With the pinion gear teeth worn away, the motor could no longer operate the valve. This inspection also revealed that the DC shunt field for the operator had failed. With the shunt field open, the valve travel speed was limited only by load. A review of completed surveillances since 1980 revealed that the travel time for the valve had been about 8 seconds. This is approximately one half of the normal travel time of 16 seconds and may have accelerated the gear tooth erosion.

Discussion:

A similar problem was found at Browns Ferry Nuclear Power Station Unit 1 (LER 79-035 reported January 2 and March 11, 1980, and February 12, 1981) and at Unit 2 (LER 80-2 reported March 11, 1980).

Several factors contribute to the potential for the reverse installation of the pinion gear, including the following: (1) the reverse installation is relatively easy and not readily detected by observation, (2) the reverse installation is not revealed in postmaintenance testing (except for Limitorque operator Types SMB 00 and SMB 000), and (3) the pinion is installed in one direction in certain types of Limitorque operators while in others it is properly reversed 180°. Because Limitorque valve operators are used for many safety-related valve applications, the unexpected failure of a valve to operate electrically could be very significant.

Corrective actions initiated by the licensee include:

1. Conducting a sampling of accessible safety-related Limitorque valve operators to ensure correct pinion gear installation. Should the sampling give a positive indication that other safety-related Limitorque valve operators are suspect, develop a program for their inspection to precede corrective action 4 below.

2. Adding a requirement for independent verification of the correct installation of pinion gear to applicable maintenance procedures.

3. Adding a caution statement to applicable maintenance and electrical procedures to ensure that personnel verify the correct Limitorque valve operator model, and to warn that incorrect installation cannot be detected in postmaintenance testing and can lead to unexpected failure of the valve.

4. Adding inspection of the pinion gear installation and gear tooth wear to the preventive maintenance program for Limitorque valve operators.

5. Adding the inspection of the shunt field for primary containment isolation valve dc operators to the Limitorque valve operator inspection program.

6. Providing training on proper pinion gear installation and the failure mode for Limitorque valve operators to all responsible crafts personnel, including electricians.

No specific action or written response is required by this information notice. If you have any question about this matter, please contact the Regional Administrator of the appropriate NRC regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

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Richard J. Kiessel, IE
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STONE & WEBSTER ENGINEERING CORPORATION - CHERRY HILL OPERATIONS CENTER

INTEROFFICE MEMORANDUM

J.O. OR W.O. NO 12210

SUBJECT EAP 16.2, FIRST REVIEWER EVALUATION OF LOOSE LUGS ON LIMITORQUE OPERATOR MOTOR LEADS ON CATEGORY I VALVES, GSU DR-205

DATE April 22, 1985

FROM JCBisti

TO CECronan

CC C4/12210/529/4YHWPC
General Files
JHGelston
KRFloyd
RJMcMorland
RBAvrich
BGSchultz
JAKirkebo
JMLord
WMEifert
DPBarry
RABerry
FACanuso
DDMartin G9.25.1.1
RBBradbury
LLDietrich
JWDempsey
RSTate:BP

This memorandum supersedes the memorandum from W. G. Culp to C. E. Cronan dated August 23, 1984.

Background and Description of Problem

During the replacement of terminal blocks, broken during operator installation checkout, on motor-operated valves 1SWP*MOV74A and 1SWP*MOV74B, it was discovered that the lugs on motor leads T1, T2, and T3 had been improperly installed by the motor operator supplier. This deficiency was documented on Nonconformance and Disposition Report (N&D) No. 5866. The following two conditions were identified on the N&D:

1. The lug crimp was not sufficiently tight, allowing the lug to be pulled off by hand with little effort.

2. The wire strands did not extend through the ring tongue end as required.

Motor leads T1, T2, and T3 are the power supply to the motor as shown on Limitorque Drawing No. 15-477-4071-3, SWEC Drawing File No. 0228.212-047-095C. In addition to the referenced N&D, Report of a Problem No. RB1-E-082 has been issued to document further this nonconformance. It has been determined that vibration during normal plant operation or a seismic event could cause a loose lug to separate from the motor lead. This would cause a loss of power to one phase with increased current in

the remaining two phases. The overcurrent condition would cause a motor overload condition followed by a deenergization of the motor.

In addition to the above, a similar problem was subsequently found to exist on valves 1CNS*MOV125 and 1HVK*MOV20A. These conditions are documented on N&D No. 6596 and 7446.

Underlying Cause

It must be conservatively assumed that the identified condition is a result of a breakdown in the vendor QA program. For motor-operated valves located inside the reactor containment, a comprehensive checklist has been issued. One of the requirements of this checklist is that terminals be checked to ensure that they are properly installed. Operators located outside the containment will be checked using generic test procedure No. 1-G-EE-18. These checks will be made prior to final system turnover to the utility.

Corrective Action

Limitorque Corporation (Limitorque) was notified of the problem by telex on July 10, 1984. This notification included a request to provide information relative to the action being taken by Limitorque to prevent recurrence of the problem. Limitorque's response dated August 15, 1984, basically stated that a wiring standard was now being used which ensured conformance with Limitorque wiring requirements. Limitorque's response further stated that the operator in question was probably furnished prior to this standard. Based on the Limitorque response, a call was made to determine when the Limitorque standard was placed into use. Limitorque stated that it became effective around mid-1981, after most River Bend Station - Unit 1 operators had been furnished.

This problem was corrected by replacing the terminal lugs furnished by the Seller with new lugs. Directions to complete this replacement were given in N&D No. 5866.

Safety Implication

Valves 1SWP*MOV74A and 1SWP*MOV74B are normally open valves which remain open on initiation of standby service water to maintain a flow of cooling water to auxiliary building unit cooler 1HVR*UC5. This unit cooler receives flow from each branch of the service water system, providing cooling to the high-pressure [illegible] pumproom. However, one division of the standby service water system would provide adequate cooling. Should any problem develop in either division of the standby service water system following standby service water system initiation, these valves must remain operable so that the malfunctioning division of the standby service water system can be taken out of service. If a pipe break should occur in combination with a loss of either valve, a resultant loss of water from the standby service water cooling tower basin would occur. This would potentially cause a loss of cooling to the residual heat

removal heat exchanger, among other safety-related equipment, causing a loss of cooling capacity to the reactor core.

Valve 1CNS*MOV125 is one of two long-term containment isolation valves located outside the reactor containment. The second isolation valve outside the reactor containment and in series with 1CNS*MOV125 is 1CNS*MOV130. The only isolation valve located inside the containment is valve 1CNS*V86, a swing check valve. A swing check valve may not be used for long-term containment isolation. It may be assumed that, during a postulated accident, valve 1CNS*MOV130 will fail to operate. Because of the problem with loose terminals on 1CNS*MOV125, 1CNS*MOV125 will not actuate for containment isolation. Since piping upstream of the containment isolation valves is ANS B31.1 piping, piping system integrity may not be assumed. As such, during a postulated accident, long-term containment isolation may be lost resulting in the release of radioactive nuclides to the environment.

Valve 1HVK*MOV20A is the discharge valve for pump 1HVK*P1A. This pump is in parallel with pump 1HVK*P1C. Should valve 1HVK*MOV20A fail to open, valve 1HVK*MOV20C would be opened and pump 1HVK*P1C would be brought into service. Should this valve fail to operate, the redundant train and pumps DFM*P1B and DFM*P1D would be put into service. Based on this redundancy, failure of 1HVK*MOV20A will not cause an adverse affect on the safety of operation.

Based on the above, it must be concluded that the loose terminal lugs could have adversely affected the safety of operation of the nuclear power plant had the condition gone uncorrected. Therefore, the condition is considered to be potentially reportable to the NRC under 10CFR50.55(e)(1)(i).

EAP 16.3 (10CFR21) Application

In addition, it has been determined that the identified condition is considered to be a potential defect, and an evaluation in accordance with EAP 16.3 will be conducted separately.

EAP 16.1 Application

By copy of this memorandum, this information is forwarded to Engineering Assurance for review in accordance with EAP 16.1.

JCBisti

J. C. Bisti
Project Engineer