U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No. 50-334/82-05

Docket No. 50-334

License No. DPR-66 Priority - Category C

Licensee: Duquesne Light Company

P. O. Box 4

Shippingport, Pennsylvania 15077

Facility Name: Beaver Valley Power Station, Unit 1

Inspections at: Shippingport and Pittsburgh, Pennsylvania

Inspections conducted: February 22 - 26, 1982

Inspectors: P. K. Eapen — 3/18/82
P. K. Eapen, Ph. D., Reactor Inspector — date signed

G. Napuda — 3/18/82
G. Napuda, Reactor Inspector — date signed

Approved by: D. L. Caphton — 3/19/82
D. L. Caphton, Chief, Management Programs Section, Engineering Inspection Branch, Division of Engineering and Technical Programs — date signed

Inspections Summary: Inspection on February 22-26, 1982 (Report No: 50-334/82-05)

Routine, unannounced inspection of previous inspection findings, facility modifications, and non-licensed employee training. The inspection involved 55 inspection hours on site and 16 inspection hours at the corporate offices by two region based inspectors.

Results: Of the three areas inspected, no items of noncompliance were observed in two areas, and one item of noncompliance was identified in the remaining one area. (Noncompliance - inadequate design controls for specifying design requirements, maintaining proper interfaces among participating organizations and verifying the adequacy of the design). (See paragraph 3.e).

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Details

1. Persons Contacted

G. Beatty, QA Engineer
F. Bissert, Manager, Nuclear Support Services
M. Coppula, Superintendent, Technical Services
F. Curl, Construction Department
* K. Grada, Superintendent, Licensing
* D. Hunkle, Director, QA Operations
* H. Jimenez, Director, Electrical Station Engineering
T. Jones, Manager, Nuclear Operations
N. Kerman, Project Engineer, Mechanical Engineering
C. Lamping, Station Engineer
* W. Laughlin, Senior Project Engineer
* R. Mafrice, Supervisor, Onsite Engineering Group
* R. Martin, Director, Nuclear Engineering
* J. McGee, Director, Administrative Services
A. Robosky, Staff Engineer, Mechanical Engineering
F. Salmon, Manager of Mechanical Engineering
S. Sero, Senior Engineer, DLC
* J. Sieber, Manager, Nuclear Safety and Licensing
P. Slifkin, Station Mechanical Engineer
* G. Sovick, Senior Compliance Engineer, DLC
* J. Starr, Supervisor, Station Engineering
* N. Tonet, Manager, Nuclear Engineering, DLC
* J. Turner, Nuclear Shift Supervisor
H. VanWassen,
* E. Vassello, Director, Nuclear Division Training
* H. Williams, Station Superintendent
B. Fini, Onsite Engineering Group

NRC

* D. Beckman, Senior Resident Inspector
W. Troskoski, Resident Inspector

The inspectors also held discussions with and interviewed other members of the power station and Duquesne Light Company Technical and Administrative Staff.

*Denotes those present at the exit interview conducted onsite, February 26, 1982.

2. Previously Identified Items

(Closed) Inspector Follow Item (334/81-18-10). Confirm completion of training for mitigating core damage per NUREG-0737, item TAP II.B.4. The inspector reviewed records that identified those requiring this training and the "Daily Training Roster(s)" for the course presentation.

The inspector noted that the course utilized a "Mitigating Core Damage Text Book", transparencies, slides, schematics, and other plant-related visual aids. Also, the Beaver Valley Power Station Training Manual, Issue 3, Section 2.0 and 9.4 reflected that such training was included in the program.

The inspector identified no violations. This item is closed.

(Closed) Inspector Follow Item (334/81-29-13). Plant specific training for the Senior Electrical Maintenance Engineer and the Maintenance Supervisor. Based on the discussion in paragraph 4 of this Report, this item is closed.

(Closed) Inspector Follow Item (334/81-29-12). A comprehensive written training program and replacement training program for all non-licensed personnel not provided or fully implemented. Based on the discussion in paragraph 4 of this Report, this item is closed.

3. Design Change/Modification Control

a. References:

-- NUREG-0737

-- Letter from J. J. Carey to Steven A. Varga, dated December 30, 1981

-- Letter from J. J. Carey to Steven A. Varga, dated November 9, 1981

-- Station Engineering Procedure 2.3, Revision No. 8

-- ANSI N 45.2.11 - 1974

-- ANSI N 45.2 - 1977

-- Operations Quality Assurance Procedure No. OP-4

-- Operations Quality Assurance Procedure No. OP-11

-- Operations Quality Assurance Procedure No. OP-15

b. Review

The design change packages listed in c., below, were reviewed on a sampling basis to verify that the following requirements have been met, as applicable:

-- Design Input Requirements, such as design bases, regulatory requirements, codes, and standards were identified, documented and their selection reviewed and approved.

-- Design activities shall be prescribed and accomplished in accordance with procedures that would assure the applicable design inputs are correctly translated into specifications, drawings, procedures, or instructions.

-- Interface controls were established to identify, control and maintain responsibilities, lines of communications, and documentation requirements for internal and external interfaces.

-- Design verification was established to determine the adequacy of the design to meet the requirements specified in design inputs.

-- Document control procedures were established to control the issuance of design documents and their changes.

-- Design change control procedures were established to control design changes.

-- Design documentation and records were maintained.

-- Audits were conducted to verify compliance with all aspects of QA programs for design and design changes.

-- New or modified systems were installed in accordance with the approved design.

-- New or revised procedures relating to the modified system were completed and approved for technical specifications.

-- As built drawings were revised to reflect modifications.

-- The operators were trained to use the modified system.

c. Document/Record Packages

The inspectors reviewed the following design change packages (DCP's):

-- DCP 265, Automatic Trip of Reactor Coolant Pumps

-- DCP 295, Reactor Coolant System Vents

-- DCP 298, Containment Water Level Monitor

-- DCP 299, Auxiliary Feed Water System Upgrade

-- DCP 303, Containment High Range Radiation Monitor

-- DCP 320, Post Accident Sampling

-- DCP 333, Reactor Vessel Water Level Indication

-- DCP 362, Recombiner Shielding

d. Detailed Interviews and Examination of Certain Documents

The inspector interviewed cognizant engineering personnel, both on site and at the corporate headquarters to determine the effectiveness of the licensee's design controls as they relate to the DCP's listed in section c. above. The following documents in each of the DCP's were reviewed to verify the adequacy of the licensee's engineering overview and administrative control for design modifications.

-- Design concept

-- Design Drawings

-- Design Verification Letter

-- Safety Analysis for 10 CFR 50.59 Evaluation

e. Findings and Conclusions

The inspector noted that there is an apparent lack of Interface Control in the area of design modifications, as evidenced by the following:

(1) Documentation on DCP-298 indicated that the level transmitters supplied by GEM-DELAVAL were qualified to meet the requirements of applicable 1971 IEEE standards. The Corporate Project Engineer informed the inspector that these transmitters would be replaced with those qualified to the requirements of NUREG-0737 as they become available and in accordance with NRC Bulletin 79-01 constraints. This information was not properly communicated to the Safety and Licensing Department. As a result, the licensee's commitment in letters dated December 30, 1981 and June 26, 1980 did not reflect the above exception from NUREG-0737 requirements. The inspector informed the licensee's representative that this lack of interface control among participating design organizations is an example of the violation discussed below.

(2) Neither the vendor nor the architect engineer kept the licensee apprised of the generic environmental qualification

problem associated with the Victoreen containment high range radiation monitors. As a result, the licensee's letter to the NRC dated December 30, 1981 did not identify these concerns. The licensee's representative told the inspector that the vendor and the architect engineer informed him of additional qualification problems recently. These additional problems are being addressed by the licensee, architect engineer, and the vendor. The corporate engineering personnel did not communicate this matter to the Nuclear Safety and Licensing Department in a timely manner. The inspector informed the licensee's representative that this lack of interface control is another example of the violation discussed below.

(3) The Corporate Cognizant Engineer informed the inspector that the Reactor Vessel Water Level Indication would not be installed by the end of the current refueling outage. This information was not communicated to the Safety and Licensing Department for initiating the required correspondence to the NRC. The inspector informed the licensee's representative that this lack of interface control was an additional example of the violation stated below.

(4) A number of DCP's did not specify the design input requirements in accordance with the guidelines of ANSI N 45.2.11 - 1979. The station personnel were aware of this concern; the licensee is compiling the above and other design documents in response to a finding in NRC Inspection No. 50-334/81-29.

(5) The design verification letters contained in DCP-298 and DCP-299 were found to be inadequate in that they did not meet the requirements of ANSI N 45.2.11-1974 (Section 6.3.1.4). This is an indication of an inadequate design verification program and is an example of the violation discussed below.

(6) The Safety Evaluation Reports generated by corporate personnel indicated that the present format forces a reviewer to limit the review to narrow perspectives. For example, the responsible safety reviewer failed to: (a) assess the impact of the design under DCP 362 on nearby safety systems, (b) recognize the importance of meeting 10 CFR 50, Appendix A, general design criterion 19 requirements to protect the plant personnel from radiation; and, (c) realize the importance of regulatory requirements as they apply to 10 CFR 50.59 reviews and the role they play to protect the health and safety of the public and plant worker. The inspector stated that this was another indication of inadequate design control. The licensee representative stated that DCP 362 would be further reviewed to address these concerns.

(7) The Station Engineering personnel informed the inspector that, under the new Nuclear Division Charter, the Station Coordinating Engineers (as defined in SEP No. 2.3) would perform preliminary safety evaluations. The inspector noted that the present coordinating engineers appear to be qualified and capable. However, from the discussions with the coordinating engineers, it was identified that the required training programs and refresher training programs in the area of Safety Evaluation are not in place (reference paragraph 4). A licensee representative stated to the inspector that he was aware of this need and a training program would be developed.

The inspector informed the licensee representatives that the examples stated above are contrary to 10 CFR, Part 50, Appendix B, Criterion III, and ANSI N 45.2.11 (as endorsed by Duquesne Light Company Quality Assurance Procedure No. OP-4) and collectively constitute a violation. (50-334/82-05-01).

4. Non-Licensed Training

a. References

-- Beaver Valley Power Station Training Manual, Issue II, Rev. 10

-- Engineering Management Procedure (EMP), 1, 6, Personnel Training, Rev. 4

-- Beaver Valley Training Status Report, Fourth Quarter (October - December, 1981) 1981, January 22, 1982

-- ANSI N 18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants

-- 10 CFR 50, Appendix B, Criterion II

b. Program Review

The inspector reviewed the Beaver Valley Power Station Training Manual, Issue 3, to determine that initial training and retraining requirements had been established and specified for personnel, such as auxiliary operators, craftsmen, engineers, and general employees.

The inspector determined that the established written training program did not clearly delineate the specific skill requirements for a given position, describe the method or responsibility for analysis of the particular individual's training needs, and detail the management/program controls to assure such training will be accomplished. Prior to departing from the site, the inspector reviewed the applicable portions of the licensee's response (dated February 26, 1982) to the Performance Appraisal Branch Inspection Report No. 81-29. This response did include actions that addressed the inspector's concerns. Pending review of the licensee's proposed actions and verification of implementation of these actions, this item is unresolved. (50-334/82-05-02).

c. Employee Training

The inspector randomly reviewed the licensee's training records to assure that programatically-required training had been given to plant personnel in the areas of administrative controls and procedures, radiological health and safety, controlled access and security, industrial safety, emergency plans and procedures, quality assurance indoctrination, and Regulatory Guide 8.13, Appendix A, contents for female employees. In addition, the inspector conducted interviews with certain individuals whose records were reviewed to verify that the scope of the training was similar to that contained in the licensee's records, the training as conducted was meaningful to those attending, and that the areas presented were covered accurately and sufficiently from the participant's point of view.

The employees interviewed were one new employee, an employee with more than one year service, one temporary employee, and a female. These employees represented the craft disciplines of Electrician, I&C Technician, and Mechanic.

The inspector noted that all then current electrical, mechanical and general labor personnel had received training in basic maintenance (Module 1) during 1980 - 1981. Training presented in progressive Modules becomes more advanced and/or specific. The inspector noted that approximately 60 percent of Module 1 attendees had completed Module 2 during 1981. Four electrical maintenance foremen and two engineers had taken training in "AC/DC Theory" and "Introduction to Electricity". The current outage forced the postponement of scheduled training for I&C Technicians. Special courses, such as "Reactor Coolant Pump Seal", are given on an as needed basis. The inspector verified that the applicable interviewed individuals did understand the information which had been presented (a sample).

The inspector also reviewed the training records of approximately ten engineers from corporate engineering. A formal "Engineering Management Procedures Course for Nuclear Projects" course was given to all engineers in 1979 and other formal training was provided to selected individuals. The inspector specifically reviewed the records of five principle engineers and noted that they either had attended a pressurized water reactor (PWR) course given by the NSSS or had extensive PWR background. In one instance, the engineer was previously employed by the NSSS for nine years.

The inspector determined that applicable training was being provided to employees. However, as discussed in paragraph b above, there was no detailed guidance for determining an individual's complete training needs or pre-planning to accomplish this training.

No violations were identified.

5. Exit Interview

The inspectors met with licensee representatives denoted in paragraph 1 at the conclusion of the inspection on February 26, 1982 to summarize the findings of the Inspection as detailed in this report. The actions committed to by the licensee representatives were restated by the inspectors at this meeting. They were:

(1) A detailed review of all NUREG-0737 related modifications will be performed.

(2) Differences between actual design and NUREG-0737 requirements will be identified and resolved with NRC.

(3) The QA department will conduct an audit to independently verify that this review was performed and appropriate actions initiated for any identified differences.

(4) Procedures will be developed to control interfaces required for NRC correspondence within the Nuclear division.

(5) A corporate-level procedure will be developed to control inputs from all divisions and outside organizations.

The Senior Resident Inspector advised the licensee representatives that the above licensee commitments should be fulfilled prior to start up to avoid unnecessary delays and further regulatory actions. (These licensee commitments were presented to NRC regional management, and a Regional confirmatory letter 82-06, dated March 4, 1982, addressing the pertinent commitments of understanding was sent to the licensee.)

The licensee further agreed that other existing commitments to the NRC will be reviewed to assure that they are implemented as stated.

The licensee representatives acknowledged the inspection findings and statements.