U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No. 50-334/82-24

Docket No. 50-334

License No. DPR-66 Priority -- Category C

Licensee: Duquesne Light Company

435 Sixth Avenue

Pittsburgh, Pennsylvania

Facility Name: Beaver Valley Power Station, Unit 1

Inspection at: Shippingport, Pennsylvania

Inspection conducted: September 7 - 16 and 27 - 30, 1982

Inspectors: for L. E. Tripp 10/14/82
W. J. Lazarus, Project Engineer — date signed

for L. E. Tripp 10/15/82
D. R. Haverkamp, Reactor Licensing Engineer — date signed

P. K. Eapen 10/14/82
P. K. Eapen, Reactor Inspector — date signed

Approved by: L. E. Tripp 10/14/82
L. E. Tripp, Chief, Reactor Projects Section No. 2A, Reactor Projects Branch No. 2 — date signed

Inspection Summary: Inspection on September 7 - 16 and 27 - 30, 1982, (Inspection No. 50-334/82-24).

Areas Inspected: Routine inspections by three region based inspectors (102 hours) of licensee actions taken to comply with selected NUREG-0737 TMI Task Action Plan items.

Results: No violations were identified.

DETAILS

1. Persons Contacted

   * D. Blair, Health Physics Specialist
   J. Carey, Vice President
   * K. Grada, Superintendent of Licensing and Compliance
   F. Lipchik, Senior Compliance Engineer
   * J. Maracek, Senior Licensing Engineer
   * S. Sovik, Senior Compliance Engineer
   J. Vassallo, Director, Nuclear Division Training

   The inspectors also interviewed several licensed operators and members of the technical staff.

   *Denotes those present at exit interview on September 30, 1982.

2. Implementation of TMI TAP Requirements (NUREG 0737)

   A.II.F.1 Additional Accident Monitoring Instrumentation

   The inspector reviewed the documentation and inspected selected, installed equipment associated with the following plant modifications to verify that the design changes had been properly reviewed, approved, and controlled in accordance with adequate procedures; that test results had been reviewed by appropriate personnel; that procedures and drawings had been changed as necessary; and that personnel had received appropriate training. A comparison of the design changes to NUREG-0737 criteria and licensee commitments was conducted to verify that the modification met these requirements.

   -- II.F.1.1 and II.F.1.2 Install Noble Gas, Iodine, and Particulate Effluent Monitors

   The following documentation was reviewed:

   -- DLC letters dated 9/17/80, 12/31/80, and 10/7/81

   -- DCP 303 Safety Evaluation Report

   -- Installation Procedures 251-0 and 333-0

   -- DCP 303 System releases and turnover checklists

   -- DCP 303 Cover Sheet (indicates operational acceptance 7/1/82)

   -- Initial Test/Calibration Procedures T-19-303-18 and T-303-10

-- Calibration Procedures 43.56, 43.57, 43.58, 43.59, 43.60, 44.08

-- Technical Specification Change Submittal 1A-70.

In addition to the documentation review, the inspector observed that the RM-CON-1 Control Terminal, which monitors the detectors, was operational in the Control Room. No inadequacies were identified.

-- II.F.1.4 Containment Pressure Monitor

The following documentation associated with this modification was reviewed:

-- DLC letters to NRR dated 12/30/81, 6/26/80, and 5/25/82

-- DCP 297 Design Concept

-- DCP 297 Final Safety Evaluation Report (OSC review: 36-81)

-- DCP 297 Cover Sheet (final operational acceptance 6/30/81)

-- Calibration Procedures MSP 12.05 and 12.06

The inspector also observed the containment pressure indicators and controllers installed in the Control Room. A technical evaluation of this design and installation is scheduled to be completed by NRR by November 1982. In addition, several items remain for resolution: (1) A Technical Specification change must be submitted following receipt of a "model" from NRR, (2) the pressure transmitters must be upgraded to IEEE 323-1974 environmental qualification standards when equipment is available, (3) audit findings identified to NRC for DCP 297 in a letter dated 5/25/82 must be corrected. (334/82-24-01).

Except as noted above, the inspector had no further questions in this area.

-- II.F.1.5 Containment Water Level

The following documentation associated with this modification was reviewed:

-- DLC letters dated 6/26/80, 12/31/81, 4/16/82, and 5/25/82

-- DCP 298 Design Concept

-- DCP 298 Final Safety Evaluation Report (OSC reviewed in meeting 6-81)

-- DCP 298 Cover Sheet (Indicates operational acceptance 4/3/81)

-- Installation Procedure BVPP 163-2

-- Calibration Procedure MSP 9.06

The inspector noted that due to the physical characteristics of the narrow range level transmitter, it does not start to indicate level until it reaches approximately three inches, instead of from the "bottom of the sump" as indicated in NUREG-0737. Due to the small size of the sump, there is apparently no safety significance in this discrepancy. The system design is presently receiving a technical evaluation by NRR. The licensee noted in their May 25, 1982, letter to NRR that the level transmitters are qualified to IEEE Standard 344-1971, rather than the 1975 edition, but are the best available. They will be qualified to the 1975 standard or replaced with qualified transmitters when they become available. The inspector had no further questions in this area.

-- II.F.1.6 Containment Hydrogen Monitor

The following documentation was reviewed:

-- DCP 294 Final Safety Evaluation Report

-- OSC Review of FSER in Meeting 69-82

-- Design Concept, EM 20237

-- DCP 294 Cover Sheet; indicates operational acceptance 6/16/82

-- Installation Procedures BVPP 223-0 and BVPP 313-0

-- Test Specification BVPP 372-0

Based on the results of this review and inspection of the instrumentation in the Control Room, no inadequacies were identified.

B. <u>Training and Requalification</u>

The inspector reviewed the Technical Evaluation Report of the Duquesne Light Company response (prepared by NRC contractor, Science Applications, Inc.) concerning requirements for upgrading training and requalification programs to meet the guidelines of NUREG-0737 I.A.2.1, and II.B.4. The Technical Evaluation Report (TER) concluded that the licensee's training and requalification programs satisfied the NRC guidelines, except for three areas which are discussed below. The inspector performed onsite follow-up in these areas to resolve the apparent inadequacies:

-- <u>I.A.2.1.C(3) Increased Operator Training for Transients</u>

The TER concluded that the licensee's training program did not contain increased emphasis in dealing with reactor transients. The inspector reviewed the licensee's Training Manual (Issue 3), lesson plans for plant response to various casualties, and the simulator training course. Based on this review, ample training in transient response is included in the licensee's operator training program to meet the NRC guidelines discussed above.

-- <u>I.A.2.1, Enclosure 1, Item C.1, Requalification Program Upgrading</u>

The guidelines state that requalification programs include instruction in areas of heat transfer, fluid flow, thermodynamics, and accident mitigation. The TER concluded that the requalification training program had no training in the area of accident mitigation. During a review of the Beaver Valley Operator Retraining Manual and the simulator training plan, the inspector identified several areas which are considered training in accident mitigation and concluded that the licensee's requalification training program meets the NRC guidelines in this area.

-- <u>I.A.2.1, Enclosure 1, Item C.3, Requalification Program to Include Certain Control Manipulations</u>

The TER concluded that the 27 control manipulations specified in enclosures 1 and 4 to NRR letter (Denton to All Licensees) dated 3/28/80 had not been included in the licensee's training/requalification program. A review of the training manual by the inspector verified that these control manipulations were being covered. The Director of Nuclear Division Training agreed to change the Training Manual to clearly specify that the 27 different control manipulations listed in ANS 3.1 (same as those specified in NUREG-0737) would be performed either at the plant or during annual simulator training.

The completion of this change to the Training Manual will be reviewed in a subsequent inspection. (334/82-24-02).

2. Plant Shielding Design Review

a. Background and Scope

As discussed in NUREG-0737, "Clarification of TMI Action Plan Requirements," each power reactor licensee was required to perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review was intended to identify the location of vital areas and equipment in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems. Additionally, each licensee was required to provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review was to determine which types of corrective actions were needed for vital areas throughout the facility.

These requirements were originally issued by NRC letters to all operating nuclear power plants, dated September 13 and October 30, 1979, and were incorporated into NUREG-0660, "TMI-2 Action Plan." Significant changes in requirements or guidance were described in NUREG-0737, Item II.B.2. In the case of Beaver Valley Unit 1, the shielding design review and corrective actions were discussed by the licensee in a letter to the NRC dated June 30, 1981. The licensee subsequently discussed the status and some design details for the modifications in letters to the NRC dated December 30, 1981, April 16, 1982, and April 28, 1982.

The licensee's plant shielding design review and corrective actions were reviewed during this inspection. The review included a sampling verification of the shielding design review, methodology, and representative calculations; a review of selected emergency procedures to determine if the vital areas where personnel must go are safety accessible; and a review of corrective actions taken or planned by the licensee, including plant modifications.

b. Shielding Design Review Verification

The licensee's shielding design review methods, including source terms, calculation of dose rates, calculation of doses to personnel during post-accident access to vital areas, and acceptance criteria, were described in QUAD-1-80-040, "Design Review of Plant Shielding of Spaces for Post-Accident Operation". The shielding design review, prepared for the licensee by QUADREX Corporation, was submitted to the NRC with the licensee's June 30, 1981, letter.

The inspector discussed the details of the Shielding Design Review with the licensee and his contractor. The contractor's representatives provided the assumptions and methodology used in shielding calculations and the results obtained from such calculations. The assumptions were consistent with the guidelines of NUREG-0737, and the methodology employed state-of-the-art mathematical models. The licensee's dose level estimates were compared and found consistent with the estimates made by NRC's consultants for similar configuration.

The bases for the number and locations of radioactive-fluid-carrying pipes used in the calculations were not available for review. The licensee stated that the required information would be obtained from the contractor and made available for NRC review. This item will be followed in future NRC inspections (334/82-24-03).

Except for the item noted above, the inspector had no further questions in this matter.

c. Vital Area Accessibility

The inspector reviewed Emergency Operating Procedure (EOP) E-1, "Loss of Reactor Coolant", which would be implemented by the licensee in the event of a loss of coolant accident. The review included a step-by-step comparison of the procedure with applicable piping and instrument drawings to verify that proper flow paths can be established, a plant walkdown of portions of the procedure to determine the ability to perform the procedure and the accessibility of manual valves that may require local operation, and an assessment of potential doses to plant personnel based on the results of the licensee's shielding design review.

Based on this review, the inspector determined that the procedure was technically correct. The procedure contained appropriate provisions to assure controlled access to vital areas for post-accident operations, and post-accident doses to plant personnel would be within the guidelines of NUREG-0737. In addition, the inspector reviewed Radcon Emergency Operating Procedure (REOP) 2.1, "Access and Dose Control for Vital Area Operations During Emergency Situations". The inspector noted the extensive licensee efforts taken with respect to implementation of post-accident procedural controls. EOP E-1 incorporated numerous notes and references to figures contained in Chapter 53, "Emergency Operations", of the Unit 1 Operating Manual. The figures provided primary and alternate routes to perform various post-accident operations and indicated the calculated dose rates in vital areas, as determined in the licensee's shielding design review. REOP 2.1 was developed specifically to set forth recommendations and provide general guidance to pertinent station personnel in performing post-accident operations in vital areas, based on the shielding design review. This procedure included the access route figures described above, as well as substantial radiological controls instructions for access parties that would perform vital area operations during emergency situations.

The inspector had no further questions regarding vital area accessibility.

d. Corrective Actions

Based on the results of the plant shielding design review, the licensee determined that the calculated doses would preclude post-accident access needed to perform certain operational actions without appropriate corrective actions. The modifications completed by the licensee included installation of shielding to permit access to the hydrogen recombiner control panels, installation of reach rods for hydrogen recombiner manual isolation valves, and installation of reach rod for the containment air manual cross-connect valve. These modifications were verified to be satisfactorily completed during NRC Region I Inspection 50-334/82-16. In addition, the licensee has implemented appropriate procedural controls to limit the entry times of operators with respect to the above and other post-accident operations, as discussed in paragraph 2.C.

The inspector had no further questions regarding the licensee's shielding design review corrective actions.

3. Fire Protection Technical Specification (Table 3.3 - 10) Change Review

a. References

1. Letter from J. J. Carey (Duquesne Light Company) to S. A. Varga (NRC) dated February 11, 1982; subject: Proposed Change Request No. 63 to Operating License.

2. NRC Fire Protection Safety Evaluation for Beaver Valley Power Station Unit 1, dated May 3, 1979.

3. Fire Protection Appendix R Review, Beaver Valley Power Station, dated June 1982.

b. Background and Scope

In Reference 1, the licensee proposed to amend Table 3.3 - 10 of the Unit's Appendix A Technical Specification. The proposed changes were to replace the smoke detectors in the station battery rooms with heat detectors and to add detectors at locations identified by the licensee and the NRC staff in Reference 2.

The scope of this review was to establish the bases for the changes proposed Technical Specifications. The information provided in Reference 3 for the locations identified in Table 3.3 - 10 of the Unit's Appendix A Technical Specification would be reviewed separately.

c. Evaluation

1. Item 10 - Table 3.3 - 10, Appendix A, Technical Specification Station Battery Rooms

   Licensee proposes a change to replace the smoke detectors at the station battery rooms with heat detectors. The reason for this change is the corrosion of the smoke chamber and associated electronics of the smoke detectors in the corrosive environment of the battery room. Licensee's representatives stated that the required change could be accomplished at the detector level. The remainder of the system would be unaffected by this change. Since the heat detector and the smoke detector come under the definition of the Automatic Fire Detectors of NFPA 72 E Standard, this change meets the intent of the Branch Technical Position CMEB 9.5-1 to have automatic fire detection in the station battery rooms. The present review did not address the status of the modification required in Section 5.9.6 of Reference 2. The licensee has provided the details of the modified battery rooms in Reference 3. Details of this modification will be reviewed at a later date.

2. Items 18 through 25 of Table 3.3 - 10 of Appendix A, Technical Specification

   The licensee proposes to add these locations to Table 3.3 - 10. Locations for Items 19, 21, 23, 24, and 25 were identified in Section 4.2 of Reference 2 as areas not having complete fire detection coverage. The remaining items were added by the licensee. The licensee has provided details of these areas in Reference 3. Review of the technical adequacy of these additional fire detectors will be conducted in conjunction with the review of Reference 3.

   A review of the licensee's records and a sample inspection of the areas indicate that the licensee has installed an adequate number of detectors to meet the minimum operable detector requirements of Table 3.3 - 10. The installed detection devices meet the requirements for such devices installed in similar areas previously reviewed and accepted by the NRC.

3. Conclusions

Based on the above, the bases for the request are technically sound and meet the intent of the guidelines provided in NRC Branch Technical Position CMEB 9.5-1.

A review to establish the technical adequacy of the fire protection system and compliance with the requirements of the Appendix R to 10 CFR Part 50 will be performed at a later date.

The inspector has no further questions. The techical review of this Technical Specification Change is complete.

4. Exit Interview

Meetings were held with senior facility management on September 16 and 30, 1982, to discuss the inspection scope and findings, as detailed in this report.