U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

Region I

Report No. 50-334/78-24

Docket No. 50-334

License No. DPR-66 Priority -- Category C

Licensee: Duquesne Light Company
435 Sixth Avenue
Pittsburgh, Pennsylvania 15219

Facility Name: Beaver Valley Power Station, Unit 1

Inspection at: Shippingport, Pennsylvania

Inspection conducted: September 12-15, 1978

Inspectors: William J Raymond 11/8/78
W. J. Raymond, Reactor Inspector date signed

date signed

date signed

Approved by: E.C. McCabe, Jr., for 11/9/78
R. R. Keimig, Chief date signed
Reactor Projects Section No. 1, RO&NS
Branch

Inspection Summary:
Inspection on September 12-15, 1978 (Report No. 50-334/78-24)
Areas Inspected: Routine, unannounced inspection by a regional based inspector of plant operations, including logs, records and plant status; conformance with Technical Specification LSSS, LCOs and SL; status of previous inspection findings; in-office review of LERs; onsite followup of licensee events; outage maintenance activities; outage restoration plans; followup on an event which occurred during the inspection; and, review of periodic reports. The inspection involved 36 inspector-hours onsite by one NRC regional based inspector.
Results: No items of noncompliance were identified.

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Region I Form 12
(Rev. April 77)

DETAILS

1. Persons Contacted

*Mr. R. Balcerek, Maintenance Supervisor
*Mr. C. Davis, Senior QA Engineer
*Mr. J. Hrivnak, Station QA
*Mr. E. Kurtz, Senior QA Engineer
Mr. F. Lipchick, Station QA
Mr. R. Prokopovich, Reactor Engineer
*Mr. L. Schad, Operations Supervisor
Mr. J. Werling, Station Superintendent
*Mr. H. Williams, Chief Engineer
*Mr. R. Woodling, Senior Engineer

The inspector also interviewed other licensee personnel, including members of the operations, maintenance and general office staff.

*denotes those present at exit interview.

2. Status of Previous Inspection Findings

(Closed) Noncompliance (334/78-07-01): Documentation of Post Maintenance Test Results. The inspector reviewed completed calibration records and/or calibration scheduling forms for five instruments identified in NRC:Region I Report 78-07 for which documented test data results were missing. Licensee memorandum BVPS:JRB:65, dated July 27, 1978, was issued to maintenance department engineers and foremen to reemphasize the requirements of the BVPS Maintenance Manual. Lesson plans and rosters for training conducted in August, 1978, were also reviewed to verify that instructions concerning test result documentation had been conducted as stated in the licensee's response letter to NRC:Region I.

(Closed) Noncompliance (334/78-13-01): Use of Unapproved Procedure Checklists. Procedure ISI 4.0, Steam Generator and Reactor Coolant Pump Support Hydraulic Snubber Examination, Rev. 3, dated July 20, 1978 contains references and inspection requirements consistent with Technical Specification 4.7.8.12.2, and incorporates the appropriate 31 day inspection data sheets.

(Closed) Noncompliance (334/78-14-01a and 1b): Drills and Hose Inspections. BVPS:LGS:22, dated July 13, 1978, was issued to all shift foremen and shift supervisors to require the operations logs to be annotated when monthly emergency drills are conducted, and to require shift supervisory review of completed OSTs. OMCN 78-124 was issued on September 24, 1978, to clarify procedural instructions for hose rack inspections.

(Closed) Noncompliance (334/78-17-04): Valve Status Board. BVPS:LGS:26, dated February 21, 1978, was issued to station operators to augment existing controls governing the use of the valve status board. The inspector reviewed the valve status board, on a sampling basis, on September 13, 1978 and identified no inadequacies.

(Closed) Unresolved Item (334/78-17-01): ORC Pump Operability and Event Report. LER 78-39 was submitted on June 23, 1978 to document the licensee's reviews, investigations and corrective actions regarding the construction dam discovered on June 22, 1978, in the ORS Pump 2B suction piping. The dam apparently was installed in the equalizing line between the two ORS pump suction lines during plant construction. The presence of the dam was not detected during pre-operational testing since each pump was able to draw suction from the containment sump through its own suction screen. The Onsite Safety Committee review of this item (documented in meeting minutes BV-OSC-32-78) determined that no other dams are installed in safety injection lines, as verified either by successful flow testing, visual examination or x-ray examination.

(Closed) Unresolved Item (334/77-29-01): Quality Aspects of Nitrogen Accumulators. The licensee provided documentation from the accumulator vendor (Southwest Fabricators) to show that the vessels were fabricated in accordance with the purchase specifications and ANSI B31.1, 1973 through 1976 Addenda. This information included the following: vendor letter to DLC dated June 13, 1978 regarding DLC purchase order no. CC-71; mill test reports for materials and filler metals; and certificate of conformance for materials and workmanship (vendor letter dated August 18, 1977). The inspector had no further comments on this item.

(Closed) Unresolved Item (334/77-29-02): Inspection Procedures. Procedures NSQC 4.1, Rev. 2, NSQC 7.2, Rev. 3 and NSQC 10.1, Rev. 4 were reviewed by the inspector and found to contain adequate guidance for determining quality level, as well as for determining the appropiate QC inspection requirements and criteria.

(Closed) Unresolved Item (334/76-19-05): Closeout of ENDCRs. The status of ENDCRs PS-2574-A, PS-2301, PS-2574, PS-2591 and P-1048 were reviewed with licensee personnel. All ENDCRs were either satisfactorily resolved (PS-2574-A, PS-2301, PS-2574) or being actively pursued. The inspector had no further comments on this item.

(Open) Unresolved Item (334/76-26-03): Allowable Cold Water Injection Cycles. The licensee stated that the DLC position was to assume 50 safety injection cycles as the basis for Unit 1 design due to similarities to Unit 2. This position was developed after consultation with the NSSS vendor and is based on the standard design basis for early W plants including BV2. The licensee stated that preliminary information from the A/E indicated that the materials employed in the two reactor vessels and interconnections were identical, i.e., type 304 and 316 stainless steel. The licensee's NSSS vendor indicated, however, that the required piping stress analysis to qualify BV1 to 50 SI transients has not been done, and further, that no SI transients were specified in the initial specifications for piping and fittings.

The inspector stated that the licensee's present position to assume 50 SI transients as the design basis was unacceptable without further supporting information, including a comparative listing of specific material similarities between the Units 1 and 2 system components. This item will remain unresolved pending further review by the NRC.

3. Review of Safety Limits (SL), Limiting Safety System Settings (LSSS) and Limiting Conditions for Operation (LCO)

A review was conducted to verify that plant operation was in conformance with Technical Specification requirements for safety limits, limiting conditions for operation and limiting safety system settings. The review consisted of monitoring plant instrumentation (refer to Paragraph 4), visual observation of equipment and components and an examination of surveillance records. The items reviewed are listed below and cover the operating period of January - June, 1978.

Tech Spec Reference	Item	Record Source
3.5.2	SI Pump Operability	OST 1.11.2
	ECCS Injection Flow Path	OST 1.11.6
	Accum Valve Bkr Alignment	OST 1.11.9
	Loop B Valve Checklist	OST 1.11.7
3.6.4.3	Hydrogen Post Accident Purge	OST 1.46.5
3.7.8.1	SLCRS Exhaust Fans and Dampers	OST 1.16.2

No items of noncompliance were identified.

4. Review of Plant Operations

a. Shift Logs and Operating Records

The following logs and records were reviewed for the periods indicated:

-- S1-1 to S1-9, L1-1, and L5-1 for the period of July 5-25, 1978.

-- Temporary Operating Procedures 78-29 through 78-34 covering the period from June 22, 1978 to September 6, 1978.

-- Jumper and Bypass log entries for the period from June 28, 1978 (#1970) to August 21, 1978 (#2020);

-- Clearance log for the period July 29, 1978 to September 2, 1978;

-- Plant Incident Reports IR1 78-40 (March 6, 1978) through IR1 78-60 (April 5, 1978).

The logs and records were reviewed to verify that:

-- log sheet entries are filled out and initialed;

-- log entries involving abnormal conditions are sufficiently detailed;

-- log book reviews are being conducted by the plant staff;

-- operating orders and temporary procedures do not conflict with the Technical Specifications; and,

-- jumper log entries do not conflict with the Technical Specifications.

Acceptance criteria for the above review included inspector judgement, the requirements of applicable Technical Specifications and the following procedures:

-- BVPS OM Chapter 48, Conduct of Operations;

-- OM 1.48.3, Section H, Temporary Procedures;

-- OM 1.48.5, Section D, Jumpers and Lifted Leads;

-- OM 1.48.6, Clearance Procedures;

-- OM 1.48,8, Records; and,

-- OM 1.48.9, Rules of Practice.

No items of noncompliance were identified and, except as noted below, the inspectors had no further questions on facility records.

(1) No discrepancies were identified with regard to the function of installed jumpers or in the control of jumpers and lifted leads. The inspector did note, however, a discrepancy between the detailed system listing of jumpers and the jumper index maintained in the log, in that twelve identical jumper sequence numbers were issued for tags on different plant systems. This item was brought to the attention of the shift supervisor who initiated corrective actions. The inspector had no further comments on this item.

(2) Plant incident report IR1 78-56, dated April 24, 1978, concerned the corrosion of raw (river) water supply lines to the River Water Pump seals and bearing coolers. After about 3 years of service, the carbon steel supply line for the 1A Reactor Plant River Water Pump was found almost plugged with rust and scale from corrosion of the

line. A similar condition was found on other pump supply lines but to a lesser extent. The supply lines for all pumps were cleaned to return the pumps to normal service. Design change DC 219 was issued to change the existing piping material to either stainless steel or copper; the design change is scheduled to be completed by the Spring of 1979. Normal operations surveillance will be adequate to detect system degradation during the interim period. The inspector had no further questions on the licensee's actions on this item.

b. Plant Tour

Inspection tours of the following plant areas were conducted at various times during this inspection: Turbine Building; Auxiliary Building, Control Room; Diesel Rooms; Tank Farm Area; Spent Fuel Pool and Decontamination Building; Switchgear rooms; Safeguard and Auxiliary Feedwater Pump Areas; Cable Tunnels; and, Containment. The following determinations/observations were made.

-- Control room and local monitoring instrumentation were reviewed to verify that instrumentation and systems required to support mode 5 operation were in conformance with Technical Specifications LCO requirements. This review included the radiation monitoring instrumentation, nuclear instrumentation, RWST, BAST, emergency boration system lineups, diesel generator lineup, offsite power system lineup and residual heat removal system lineup.

-- Radiation controls established by the licensee, including posting of radiation areas, the condition of step off pads and the disposal of protective clothing was observed. Several radiation work permits used for entry into radiation areas were reviewed.

-- Plant housekeeping conditions including general cleanliness conditions and storage of materials to prevent fire hazards were observed.

-- Systems and equipment in all areas toured were observed for the existence of fluid leaks and abnormal piping vibrations.

-- Mechanical snubbers and hangers installed on the Quench Spray, Service Water, Recirculation Spray, River Water, Safety Injection and Auxiliary Feedwater system piping were observed for proper settings and conditions.

-- Selected valve lineups on the Charging and Residual Heat Removal Systems were observed for proper positioning. The following were reviewed: RH-758, RH-700, RH-701, RH-720A, RH-720B, CH-115B, CH-115D and CH-350.

-- The control board was reviewed for annuciators that normally should not be lighted during the existing plant conditions. The reasons for existing annunciators were described by a plant operator.

-- The licensee's policy and practices regarding plant tours were reviewed and no changes from previous practices were noted.

-- Control room manning was observed on several occasions during the inspection and a shift turnover was observed on September 13, 1978.

Acceptance criteria for the above items included inspector judgement, and the requirements of the Technical Specifications, 10 CFR 50.54(k) and the following procedures:

-- BVPS Unit 1, Systems Valve Lists, and Mark-Ups

-- OM 1.48.5, Safety Related Systems Valves and Equipment

-- MM Chapter 1, Section J, Housekeeping

-- MM Chapter 1, Section H, Cleaning and Maintenance Cleaning

-- SAD-25, Housekeeping and Cleanliness Procedure

Except as noted below, the inspector had no further comments in this area.

(1) During the review of control room operations and a shift turnover on September 13, 1978, the inspector noted that the core RHR recirculation flow had been throttled back from a nominal value of 3200 gpm to about 2500 gpm. This had been done based on a recommendation from the NSSS vendor subsequent to consultation over possible reasons and solutions for problems which caused the RHR pumps to become air bound (reference LER 78-51). After discussions with license personnel, the inspector noted that the following considerations had not been accounted for in going to the off-normal operating configuration (per OM Chapters 1.10.2, 1.10.4, 1.48.7 and 1.48.3).

(a) The RHR flowmeter mounted on the control board has a nonlinear scale with unmarked graduations in the lower flow regions. Some confusion existed initially among operations personnel whether the lowest graduation represented 1000 gpm or some other value; it was subsequently determined to be 2500 gpm. To operate at the flow value recommended by the NSSS vendor (1500 gpm) would require operating below the graduated portion of the flow indicator.

(b) No formal consideration had been given to possible conflicts with Tech Specs 3.1.1.3 and 3.9.8.

(c) Operating below 3200 gpm was below the low RHR flow annunciator setpoint, causing this alarm to be in all the time. (No other provisions had been taken to provide surveillance on RHR flow, particularly in view of recent problems in losing both RHR pumps.)

The inspector discussed the above items with licensee personnel. The licensee immediately increased RHR flow to the nominal value pending further evaluations and reviews. This would include a change to the RHR flow meter scale and a revision to the appropriate OM procedures, with consideration given to the above items. The licensee intends to implement the reduced flow operation to preclude further problems with air in the RHR system, but will do so during a subsequent shutdown. During the interim period of lowered RCS loop operation, scheduled to last until September 22, 1978, the licensee will maintain RCS temperature at about 110 °F and continue to vent the RHR pumps daily. These actions appear to have been effective in preventing a loss of RHR pumps subsequent to the initial event described in LER 78-51. The inspector had no further comments on this item at the present.

5. In Office Review of Licensee Event Reports (LER's)

The inspector reviewed LER's received in the NRC:I office to verify that details of the event were clearly reported, including the accuracy of the description of cause and adequacy of corrective action, and the inspector determined whether further information was required from the licensee, whether generic implications were involved, and whether the event warranted on site followup. The following LER's were reviewed:

78-01/SP	Motor Driven AFP Inoperable for Testing
78-41/03L	BIT Surge Tank Recirculation Valve
78-46/03L	No. 4 Station Battery Charger
78-02/SP	Diesel Fire Pump Inoperable Due to Bearing Failure
*78-43/01T	Main Transformer Failure and Inadvertent SI
*78-49/01T	Loss of Both RHR Pumps
*78-50/01T	No. 1 Diesel Output Breaker Failure to Close
*78-51/01T	No. 1 Diesel Output Breaker Failure to Close

No items of noncompliance were identified.

6. On Site Licensee Event Followup

For those LER's selected for on site followup (denoted by an * in Paragraph 6), the inspector verified that reporting requirements of the Technical Specifications had been met, that appropriate corrective action had been taken, that the event was reviewed by the licensee as required by the licensee's procedures, and that continued operation of the facility was conducted in accordance with Technical Specification limits. The following LER's were reviewed on site:

a. <u>LER 78-43/01T: Main Transformer Failure and Inadvertent SI</u>

This event concerned an electrical fault that occurred in the station main transformer at 1536 hours on July 28, 1978 and resulted in a generator trip, turbine trip, reactor trip and a safety injection, followed four minutes later by a loss of offsite power (station blackout). The inspector interviewed licensee personnel and reviewed facility logs, records and strip charts of primary/secondary system parameters which document the licensee's evaluation of the incident.

The electrical fault in the main transformer resulted from a short circuit between the 345 KV high voltage winding and the 21.5 KV low voltage winding of the transformer The high voltage applied on the low voltage side of the main transformer resulted in a ground from insulation failure of the A phase low voltage termination within the transformer and a simultaneous failure of the B phase surge arrestor. The main transformer and its immediate area caught fire due to ignition of transformer oil sprayed from the main transformer sudden pressure relief valve. The oil fire was subsequently extinguished by the fire protection system. Coincident with the initiating event, the main generator tripped (on generator differential, ground overcurrent, etc.), the turbine tripped (caused by generator trip) and all three reactor coolant pumps (RCP) tripped on underfrequency. The reactor coolant pumps tripped within 250 msec of the initiating event for reasons unknown, since an underfrequency condition was not apparent from traces of bus voltage and frequency. Both the RCP underfrequency and the turbine trip caused the reactor to trip. The steam dump system operated for the 100% load rejection.

With the loss of power from the main generator, a fast bus transfer occurred to switch station power from the unit service transformers to the system service transformers which are fed from the Shippingport Atomic Power Station (SAPS) 138KV switch-yard and the 345KV/138KV auto transformers. At 15 seconds after the initiating event, with the turbine/generator coasting down and still supplying fault current to the short circuit, the main generator out-of-step relays (21-101 and 21-1101) operated. This faulty out-of-step relay operation caused an isolation of the BV 345KV switchyard (tripped all 345KV line breakers as well as the tie to the 138KV auto transformers). Additionally, three of the five 138KV breakers in the Shippingport switchyard tripped, leaving SAPS as the only source of power for BV and two other 138KV block loads. The isolation of the 345KV and 138KV switchyards occurred in accordance with the design bus-shedding scheme. However, the total load was in excess of the SAPS ability to supply and the frequency dropped to approximately 59 Hz. SAPS operators, unaware of the BV circumstances, tried to correct their underfrequency condition by backing off load with the turbine generator speed changer. SAPS remained on line for about four minutes.

At one minute after the initiating event, a SI actuation occurred from high steam flow coincident with low-low Tave, as determined from the first out annunciator in the control room. Conditions for the inadvertent SI were established from the high steam flow from the steam dump actuation, and from an apparent nonuniform cooldown of the RTD manifolds for two of the three RCS loops. Since steam dump operates on auctioneered high Tave, the low-low Tave interlock was reached on two loops prior to steam dump isolation. All safety systems functioned as designed including both diesel generators which started and came up to full speed. The diesels did not load the emergency buses at SI initiation since the buses were still being powered from the offsite supply.

At 1540 hours, four minutes after the initiating event, SAPS frequency had decayed to 56 Hz and the SAPS operators tripped the unit manually. This action removed the only source of power to the 138 KV bus sections and resulted in a station blackout at BV (along with the remaining two block loads as well). The No. 1 diesel generator sequentially loaded the 4KV AE emergency bus to provide power to one train of safety related equipment. However, the 4KV DF emergency bus did not energize since the No. 2 diesel generator failed to automatically flash its field. Maintenance personnel were dispatched to the diesel rooms to troubleshoot the No. 2 diesel.

By 1544 hours, control operators had stabilized the plant at a no-load, hot standby condition and, after verifying that pressurizer level was stable, high head safety injection was secured. At 1555 hours, station maintenance successfully flashed the No. 2 diesel generator field and the DF bus was energized. Offsite power was restored at 1557 hours and the 1A RCP was restarted at 1608 hours. Plant operators commenced an RCS cooldown to mode 5 at 1614 hours.

The response of the RCS during this transient was as expected and no adverse core conditions developed. During the period of no forced flow through the core, decay heat was removed by natural circulation. The RCS was not overpressurized. No damage to safety related equipment resulted from the incident. Subsequent radioanalysis of the reactor coolant verified that no core damage occurred.

The main transformer has been returned to the vendor's shop for rewinding. A replacement transformer is being installed, with a projected plant startup date of November 20, 1978.

The following items are under continued investigation by the licensee to identify the appropiate causes and corrective actions:

(1) The licensee is evaluating the protective relaying scheme associated with the out-of-steps relays. The relays are designed to detect a collapsed grid (offsite power) and to separate the station from the offsite electrical distribution system. Sufficient generator protection would be provided by tripping the main generator output breakers (331, 341). Under the existing relaying scheme, functioning of the out-of-step relavs also trips all tie breakers in the offsite switchyards.

The licensee is evaluating the acceptability of alternate bus shedding schemes from the out-of-step relays and will implement changes prior to plant startup. The licensee will document the corrective actions taken in this area in an update report to LER 78-43. NRC:Region I will follow the licensee's actions in this area.

(2) Investigation is continuing to determine why the RCPs tripped when an underfrequency condition did not exist. The corrective actions identified for this item will be completed prior to plant startup and described in the update report to LER 78-43. NRC:Region I will follow the licensee's actions for this item.

(3) The licensee will identify the cause and implement corrective actions for the No. 2 diesel generator failure to flash its field. The inspector noted that this was the second

occurrence of this type on the same diesel in about 12 months, in that a similar failure occurred during startup testing on July 12, 1977 (documented in LER 77-69). The licensee stated that, in addition to troubleshooting the diesel start control circuitry, consideration was being given to providing manual field flash capability while in the automatic mode via a control on the main control room benchboard.

The inspector stated that the NRC position is that diesel auto-field flash capability must be demonstrated prior to plant startup, including resolution of existing control circuitry problems, in accordance with design performance described in the FSAR and without reliance on manual intervention.

The licensee acknowledged the inspector's remarks and stated that the identified diesel problems and corrective actions would be addressed in an update report to LER 78-43. This item is unresolved. (50-334/78-24-01)

b. LER 78-49/01T: Loss of Both RHR Pumps

The events and corrective actions associated with this incident were reviewed, as described in Paragraph 4 of this report.

No items of noncompliance were identified.

c. LERs 78-50 and 78-51/01T: No. 1 DG Output Breaker Failure to Close

LERs 78-50 and 78-51 concern the failure of the No. 1 DG output breaker to close while in the manual exercise mode of operation during surveillance testing. The second event (LER 78-51) occurred during this inspection on September 12, 1978. The inspector interviewed licensee personnel, observed testing in progress and reviewed the following records:

- -- Diesel Generator Vendor Technical Manual
- -- BVPS Drawing No. 8700-1.30-41A
- -- TOP 78-33, Auto Start and Sequential Loading Test of the No. 1 Diesel Generator, inclusive of OM CN No. 1, September 12, 1978
- -- IR1 78-89 (for LER 78-50) dated September 5, 1978
- -- PMP 1-36SS-1E9-1E, Rev.1, June 30, 1976, Inspection and Test of I-T-E 5KV Air Circuit Breaker
- -- Master Maintenance Index for work completed on equipment mark nos. 1F9 and 1E9

These failures have occurred while in the manual excercise mode when the control room operator attempts to close the IE9 breaker using the bench board control switch. The failures have not been consistently repetative, in that after an initial close failure, subsequent attempts to close the breaker manually have proven successful. In all cases after the initial failure, the licensee has performed TOP 78-33 to demonstrate that the diesel capability to start up and sequentially load the emergency bus was not affected. The inspector determined that the 1E9 breaker has been inspected, tested using normal PMPs and locally exercised to demonstrate that no mechanical binding was occurring in the breaker. The licensee's position is that the problem is in the start circuitry associated with the manual exercise mode only and that the diesel's ability to auto-load the emergency bus is not affected. However, further testing is planned and will include the use of a specially fabricated test stand to monitor start circuitry relays and contacts.

The inspector had no further questions on this item at the present, but stated that this item will be reviewed by NRC:Region I during subsequent inspections.

7. <u>Outage Maintenance Activities</u>

During inspection tours of the containment, the inspector witnessed portions of the work activities in progress to verify:

- -- approved procedures were available and in use;
- -- work was being conducted in accordance with the procedures; and,
- -- health physics controls were established and observed.

The inspector witnessed the following activities in progress on September 13, 1978: repack RC 229; reinstall and tighten the A steam generator manway covers; and, perform steam generator sludge lancing operations.

No items of noncompliance were identified.

8. Startup and System Restoration

The inspector reviewed the licensee's plans and procedures to provide for an orderly startup following maintenance outage activities. The normal plant startup procedures are to be used, modified as required to accomodate changes necessitated by plant system modifications and/or special testing. The Operations Supervisor is responsible for coordinating input from all plant departments which may have an impact on the heatup/startup sequence. Items considered include: information from Engineering on any plant changes made and required testing or procedure changes; input from the test group on any TPs/OST required; power escalation limits from the Reactor Engineer; applicable committments from the OSC; a review conducted with the shift supervisors of the clearances issued for each plant system to make a determination as to which plant systems will require a complete valve checkoff to be done; a review of the jumper log and MWR status; a listing of closeout inspections to be completed by QC; and, a schedule of all OSTs required to be completed prior to entering each mode.

The inspector had no further comments on this item. No items of noncompliance were identified.

9. Review of Periodic and Special Reports

Upon receipt, periodic and special reports submitted by the licensee pursuant to Technical Specifications 6.9.1 and 6.9.2 were reviewed by the inspector.

This review included the following considerations.

-- The report included the information required to be reported by NRC requirements.

-- Test results and/or supporting information were consistent with design predictions and performance specifications.

-- Planned corrective action was adequate for resolution of identified problems.

-- Determination whether any information in the report should be classified as an abnormal occurrence.

Within the scope of the above, the following periodic reports were reviewed by the inspector.

-- Annual Operating Report - 1977
-- Monthly Operating Report - January through September, 1978

The inspector had no questions relative to this review.

10. Unresolved Items

Unresolved items are those items for which more information is required to determine whether the items are acceptable or items of noncompliance. An unresolved item is contained in Paragraph 6 of this report.

11. Exit Interview

A management meeting was held with licensee representatives (denoted in Paragraph 1) at the conclusion of the inspection on September 15, 1978. The purpose, scope and findings of this inspection were discussed as they appear in the details of this report. The licensee acknowledged the NRC position taken in regard to diesel generator operability.