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

Region I

Report No. 50-293/80-29

Docket No. 50-293

License No. DPR-35 Priority -- Category C

Licensee: Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Facility Name: Pilgrim Nuclear Power Station

Inspection at: Plymouth, Massachusetts

Inspection conducted: October 1-30, 1980

Inspectors: J. Johnson, Senior Resident Inspector — 1/5/81 date signed

T. Foley, Reactor Inspector (October 14-17, 1980) — 1/5/81 date signed

date signed

Approved by: T. T. Martin, Chief, Reactor Projects Section No. 3, RO&NS Branch — 1/5/81 date signed

Inspection Summary:

Inspection on October 1-30, 1980 (Report No. 50-293/80-29)

Areas Inspected: Routine unannounced inspection of plant operations including followup on previous inspection findings, an operational safety verification, followup on a plant trip of October 1, 1980, followup on the 'A' Safety Relief Valve (SRV) inadvertent opening on October 7, 1980, maintenance activities involving the SRV modifications, a survey of the location and status of the HPN and ENS phone circuits and the status of solid radiation waste storage facilities, and an implementation verification of the Category 'A' TMI Action Plan requirements. The inspection involved 114 inspector-hours by the resident inspector and one reactor inspector.

Results: Two items of noncompliance were identified (Infraction - Failure to lock a high radiation area door, Paragraph 3.B(1); Infraction - Failure to set the Main Steam Line High Radiation Monitors in accordance with Technical Specifications, Paragraph 3.b(2)).

Region I Form 12
(Rev. April 77)

8104030613

DETAILS

1. Persons Conta.ted

BECO

W. Armstrong, Deputy Nuclear Operations Manager
R. Cavalieri, TMI Project Engineer
E. Cobb, Chief Operating Engineer
*R. DeLoach, Group Leader - Mechanical
*J. Fulton, Senior Licensing Engineer
E. Graham, Compliance Engineer
E. Graham, Compliance Engineer
W. Hoey, Senior Radiation Protection Engineer
R. Machon, Nuclear Operations Manager - Pilgrim Station
*C. Mathis, Deputy Nuclear Operations Manager
*H. O'Conner, Principle Mechanical Engineer
*L. Scottor, Senior Marine Fishereies Biologist
R. Sevigny, Staff Assistant - Mechanical
R. Silva, Staff Engineer - Mechanical
P. Smith, Chief Technical Engineer
R. Smith, Chemistry Supervisor
R. Trudeau, Chief Radiological Engineer
J. Vorees, TMI Project Engineer
P. Williard, I&C Engineer
*E. Ziemanski, Management Services Group Leader

NRC

*S. Rubin, Senior Reactor Systems Engineer, AEOD
**T. Mart..., Chief Reactor Project Section No. 3, RO&NS Branch, Region I

*Denotes those present at a briefing at the station on October 14, 1980 to discuss recent problems with the safety relief valves and fouling of heat exhangers.

**Present between October 8-10, 1980, to discuss recent events with licensee management and the resident inspector.

The inspectors also interviewed other members of the Health Physics, Chemistry, Operations, Security, and Maintenance Staffs.

2. Followup on Previous Inspection Findings

(Closed) Inspector Follow Item (293/80-18-02): The inspector reviewed the maintenance activities following completion of the SRV Modifications (see Paragraph 7 for details).

(Closed) Unresolved Item (293/80-25-02): The specific cause of the 'D' SRV failure to open on the first attempt on July 26, 1980 following solenoid assembly replacement was not identified. Subsequent testing demonstrated operability. The cause of 'D' SRV failure to open on August 1, 1980, has not been specifically determined. Inspection of the solenoid assembly and topworks revealed no failed components and subsequent testing between August 3, 1980 and August 30, 1980 (about ten test cycles) showed satisfactory operation.

When 'D' SRV was manually opened on October 1, 1980, and it jammed in the partially open position, the entire valve including main internals was disassembled. Evidence of foreign material was suspected because of the identification of scoring marks on the main piston and guide. (See Paragraph 4.)

The entire 'D' SRV and solenoid assembly was replaced (with a spare) and tested with satisfactory results on October 5, 1980.

This item is closed, however, the performance of SRV's in the future will continue to be reviewed by the NRC.

3. Operational Safety Verification

a. Scope

The inspector observed control room operations, reviewed applicable logs and conducted discussions with control room operators during the month of October, 1980. The inspector verified the operability of selected emergency systems, and verified the proper return to service of affected components. Tours of the security perimeter, reactor building, turbine building and process buildings including the radwaste area, condenser bay, and vital switchgear room were conducted. The inspector's observation included a review of plant equipment conditions, potential fire hazzards, physical security, housekeeping, and the implementation of radiation protection controls.

These reviews and observations were conducted in order to verify conformance with the Code of Federal Regulations, the Technical Specifications and the licensee's administrative procedures.

b. Findings

(1) During a tour of the Radwaste Corridor on October 16, 1980, the inspector noted that door No. 41, an access to the Condenser Bay, was left ajar and unlocked. The inspector noted that the door was posted as a High Radiation Area, that an RWP was required for entry and that an additional sign required the door

to be shut and locked. A posted survey No. 215, dated October 10, 1980, indicated general area dose rates as high as 1,500 mrem/hr and contact readings as high as 5,000 mrem/hr.

The inspector immediately notified the health physics supervisor on duty who secured the door. Subsequently, the inspector and a health physics technician entered the Condenser Bay and measured general area radiation dose rates as high as 1100 mrem/hr.

The licensee's senior management was informed of this event and initiated an investigation into the cause. Preliminary results indicate that the door was not checked shut and locked following the last entry (about 15 minutes prior to the inspector's observations).

This failure to lock this High Radiation Area access door is considered an item of noncompliance at the infraction level (50-203/80-29-01).

(2) Following a tour of the control room on October 10, 1980 the inspector questioned the licensee concerning the method of calculating the trip setpoints for the Main Steam Line High Radiation monitors. The licensee had just recently raised the trip setpoints because of noting that they had been set overly conservative during the reactor trip on October 1, 1980 (see Paragraph 4).

On October 14, 1980, the inspector reviewed the Operations daily surveillance (OPER 09) for October 12-13, 1980, which includes the Main Steam Radiation monitor readings. These were compared with the current trip settings which were made on October 2, 1980.

Channel	Reading (mrem/hr) at Full Power	Trip setting (mrem/hr)
A	700	5600
C	600	5600
B	700	6400
D	690	5600

At 9:52 a.m. on October 14, 1980, the licensee compared the current full power monitor readings with the trip set points, determined that the trip settings were not in accordance with T.S. 3.1, (settings were approximately 7.7 times full power background) initiated rod insertion, and commenced recalibration.

At 10:48 a.m., October 14, 1980 all four monitor trip settings had been reset for the following values:

Channel	Setting (mrem/hr)
A	4500
C	4000
B	4500
D	4500

On October 15, 1980, the licensee issued a prompt report LER No. 80-72 describing this event.

The licensee stated that an error had been made in calculating the trip settings made on October 2, 1980 and would revise Procedure 7.4.14, Calibration of Main Steam Line Process Radiation Monitors, to prevent recurrence.

The failure to operate the reactor at power from October 2, 1980 to October 14, 1980 with Main Steam Line High Radiation trip settings in accordance with T.S. 3.1 is considered an item of noncompliance at the infraction level (50-293/80-29-02).

(3) During a tour of the Auxiliary Boiler Room on October 16, 1980 the inspector noted a fire door (between the boiler room and chemical treatment room) open and ajar. This was brought to the attention of the control room operator who immediately took action to secure the door.

The control of fire doors was discussed with the licensee's Fire Protection Engineer who stated that procedures were being prepared (in conjunction with recent changes in fire protection requirements) to address the control and testing of fire doors and would be implemented in accordance with schedules stated in the revised regulations.

The inspector also reviewed an entry in the Watch Engineer's Instruction log which reminded operators to check fire doors during their tours of the plant.

On subsequent tours of the facility the inspector has not observed improper positioning of fire doors.

The inspector had no further questions at this time.

c. Events

(1) On October 2, 1980, the licensee reported that continuing seismic analysis per IEB 79-14 identified the need to modify a RHR system piping support (No. H10 140-SA). The plant remained in cold shutdown while repairs were completed. The modification changed the support from an anchor to a restraint.

The inspector verified completion of this modification through discussions with the Construction Management Group Leader and a review of records including control room logs which indicated completion on October 5, 1980 prior to plant startup.

This event is described in LER 80-67.

No items of noncompliance were identified.

(2) On October 1, 1980, the inspector was notified by the Chief Radiological Engineer of a telephone call from another nuclear power plant concerning a possible overexposure of a contractor while he was at Pilgrim between February and June 1980.

The licensee stated that a review of records showed that no over-exposure had occurred.

The inspector reviewed the appropriate radiation exposure records (NRC Form No. 4, check-in and termination records) with the licensee representative and verified the licensee's conclusion.

No items of noncompliance were identified.

(3) At about 3:36 p.m. on October 10, 1980, the feeder breaker (No. 50-204) for motor control center (MCC) B-14 tripped. This caused a loss of loop 'B' RCBBW pumps and SSW pumps. Because the 'A' RBCCW heat exchanger was out of service for cleaning, the actions of T.S. 3.5.B.3 were followed and a power reduction immediately initiated.

At about 4:25 p.m. on October 10, 1980, after resetting the breaker, observing no faults in the switchgear, and observing normal operation of loop 'B' cooling components, normal power operations were resumed.

On October 11, 1980, following return of 'A' RBCCW heat exchanger to service, inspection and testing (breaker overcurrent trip device calibration. and insulation checks) of breaker 52-204 were performed in accordance with maintenance request (MR) No. 80-3237 and station electrical procedures.

The inspector verified that the actions of T.S. 3.5.B.3 were followed and that MR 80-3237 was properly followed and completed.

The licensee issued LER 80-70 which describes this event.

No items of noncompliance were identified.

(4) On October 23, 1980 the HPCI turbine tripped on overspeed during routine surveillance testing. Testing on redundant equipment was performed and the overspeed trip mechanism repaired.

The inspector reviewed control room logs, spot checked testing on redundant equipment and verified completion of the HPCI system overspeed trip test and system operability test on October 25, 1980.

No items of noncompliance were identified.

4. Reactor Trip and Stuck Open Safety Relief Valve on October 1, 1980

a. Description of Event

On October 1, 1980, at about 3:37 a.m., shortly after placing the 'B' condensate demineralizer in service following backwashing, the plant experienced a main steam line high radiation alarm, main steam isolation valve (MSIV) closure, and a reactor trip from about 96% power. After reactor vessel level was returned to normal and all feed pumps secured, it was noted by control room operators that the recirculation pumps had not run back to minimum speed. The recirculation pumps were manually decreased to minimum speed. Radiation surveys and a review of process instrumentation showed no abnormal offsite releases or radiation levels.

Following manual operation of the HPCI and RCIC systems, and 'C' Safety Relief Valve (SRV) to assist in controlling the reactor pressure rise, at about 8:00 a.m., (with reactor pressure about 600 psig) the 'D' SRV was manually opened and stuck in the partially open position until about 10:10 a.m. with RCS pressure at 20 psig.

The reactor was placed in cold shutdown at about 10:35 a.m. pending investigation and resolution of the problems.

b. Review of Events/Investigation/Conclusions

The inspector reviewed these events through discussions with operators and licensee management, and a review of logs, records, and control room instrumentation.

The following parameters were reviewed:

- RCS pressure/temperature
- RV level
- Feed water flow
- Recirculation loop flow
- Process radiation monitors
- SRV accoustic position indication

During a telephone conversation between NRC; IE; Region I management and Boston Edison Company management on October 1, 1980, the licensee agreed to maintain the plant in cold shutdown until the following actions had been completed:

- Determine and correct the specific problems which caused the 'D' SRV to stick open and the recirculation pumps to fail to runback.

The licensee also agreed to perform a thorough investigation to identify the cause of the main steam line high radiation trip and the inability to restart 'A' and 'C' reactor feed pumps following that trip.

These commitments were confirmed in a letter from the Director, NRC Region I to Boston Edison Company dated October 2, 1980.

The licensee's investigation revealed the following:

- Cause of M.S. Line Hi Rad Trip: The trip was attributed to a spike in N-16 radiation due to air being introduced following placing the 'B' condensate demineralizer in service. Operators transfer and backwashing of 'B' demineralizer was duplicated. Proper operation of all components was verified. (a problem with the demineralizer tank fill light indication was identified and corrected but not determined to indicate that the tank was not full because of other 100% indication on the gas scrubber.)

- Failure of Recirculation Pumps to Runback: The recirculation pumps did not run back to minimum speed because the feed water flow signal did not decrease to the range (<20%) required to initiate this run back. The erroneous feed water flow indication was due to an error in locating the ground wire connection in the temperature compensation network of the feedwater flow loop. This wiring error was corrected and a followup test of the 20% runback function verified.

- Reactor Feed Pumps not Starting: The 'A' feed pump control switch was identified to have pitted contacts. The 'C' feed pump control switch had loose connecting screws on the control panel. Both items were repaired and proper operation verified.

- 'D' Relief Valve Stuck Open: The licensee made a drywell entry on October 2, 1980 with the assistance of a technical representative of Target Rock Corp. Inplace (as found) testing revealed no abnormalties with the exterior of the bonnet, solenoid assembly and air operator assembly. The solenoid was cycled twice from the control room switch and proper operation was observed and heard locally.

  The complete SRV was removed for inspection and a spare was reinstalled.

  The solenoid assembly, air operator assembly, and pilot assembly were tested, disassembled, and inspected and showed normal attributes except deterioration of o-rings and evidence of loctite on the I.D. of the solenoid bonnet tube and plunger. The presence of loctite did not affect the solenoid operation for it's intended design conditions.

  The main valve internals showed indentations/scoring on the I.D. of the main guide and the O.D. of the main piston. Discoloration on the guide indicated that the valve was in the partially open position (~15/16") for a period of time.

  It was concluded that foreign material had been wedged between the main valve guide and piston, however, no foreign material was found during disassembly.

  The inspection of 'D' SRV and conclusions are described in LER 80-79.

  The inspector determined through discussion with licensee personnel and a review of records that repair were completed prior to plant startup and verified that operability testing of the replacement SRV was satisfactorily performed on October 5, 1980 (at about 182 psig) during plant startup.

c. Findings

The inspector had no further questions at this time, however, an item of noncompliance associated with the subsequent raising of the main steam line high radiation trip setpoints following this event is described in Paragraph 2.b.

5. Inadvertent Opening of 'A' Safety Relief Valve (SRV) at Power

a. Description of the Event

At about 4:03 a.m. on October 7, 1980, the 'A' SRV opened for no apparent cause at 96% reactor power. Operators were unable to close the valve by cycling the control switch, manually scrammed the reactor, and proceeded to cold shutdown pending an investigation into the event.

b. Review/Investigation/Resolution of Concerns

The inspector reviewed the events, held discussions with operators, licensee management, observed instrumentation and reviewed records. Parameters reviewed included recirculation loop temperatures, reactor vessel (R.V.) shell/flange ΔT, RCS pressure/Pressure, SRV position indication, and R.V. level.

The inspector observed actions during the cooldown and verified use of appropriate station procedures and completion of check lists.

During a telephone conversation between NRC, I.E. Region I management and Boston Edison Company management on October 7, 1980, the licensee agreed to take the following actions in relation to the two depressurization events (10/1/80, 10/7/80):

1. Determine the causes for the opening of the 'A' SRV and the inability to close both the 'A' and 'D' SRV's once stuck open.

2. Based on the findings of the investigation required by item 1 above, ensure all installed SRV's are not subject to similar failure mechanisms.

3. Maintain the plant in a cold shutdown condition until items 1 and 2 above are complete.

4. Provide by resumption of power operations, a date by which an evaluation will be documented of the effect on the reactor vessel of the cooldown experienced during the two depressurizations.

The licensee, accompanied by a technical representative from the Target Rock Corp., entered the drywell on October 7, 1980 to investigate and test the 'A' SRV. Testing revealed that the opening of the 'A' SRV was caused by excessive nitrogen supply pressure and that the SRV and solenoid assembly operated as designed.

No malfunction was identified.

On October 8, 1980 the inspector (accompanied by an NRC IE Region I management representative denoted in Paragraph 1) questioned the licensee to verify the commitments as described above. The following information was provided:

- The cause of the 'A' SRV to open was due to excessive nitrogen supply pressure (160-165 psig). (This may have been related to operation of the nitrogen regulators in conjunction with a recent delivery of liquid nitrogen, however, no maloperation of the regulators was identified). Supply pressures greater than 145 psig can initiate solenoid leakage and pressurization of the air operator which can initiate opening of the pilot and main valve when at normal operating pressure.

- The cause of the 'A' SRV to stay open was also due to the high supply pressure and the design of the solenoid valve. The design is such that once opened, solenoid closure is precluded against a supply pressure greater than approximately 135 psig.

- The cause of the previous (10/1/80) stuck open 'D' SRV was due to suspected foreign material in the main valve internals and was considered to be an isolated case specific to 'D' SRV.

- Assurance that all SRV's installed are not subject to similar failures was provided by administratively maintaining the nitrogen supply pressure at approximately 110-120 psig and requiring a check and logging this pressure once per shift. The 'D' SRV with suspected foreign material was replaced with a spare and was satisfactorily tested. An other installed SRV's had also been satisfactorily tested.

- The plant was maintained in cold shutdown until these corrective actions had been performed, and

- That the documented evaluation of the effect of the cooldowns on the reactor vessel should be available by October 10, 1980.

- The licensee further stated that previous concerns centered around keeping the nitrogen supply pressure greater than instrument air pressure when the drywell was inerted. An internal request was initiated to evaluate a modification including placing a relief valve in the nitrogen supply line.

The details of this event are described in LER 80-69.

c. Findings

The inspector reviewed the activities involved with startup and heatup on October 8, 1980, and verified completion of operability test on 'A' SRV at 150 psig during the startup as required by Technical Specifications.

The inspector reviewed documentation provided to the licensee from General Electric Company dated October 9, 1980, which stated that the actual blowdowns were less severe than the design basis blowdown and that the reactor could safely be returned to power.

The inspector also reviewed an entry in the Watch Engineer's Instruction log which required logging the nitrogen supply pressure once/shift and maintaining it $\leq$120 psig.

The inspector had no further questions at this time and forwarded the information concerning this event and the design of the SRV solenoid valve to NRC management for further review of a generic nature.

No items of noncompliance were identified.

6. Meeting on October 14, 1980 Concerning SRV Problems and Fouling of Heat Exchangers

On October 14, 1980 a meeting was held on site with the personnel denoted in Paragraph 1, concerning the following topics:

- Safety Relief Valve Modifications

- Recent problems with the nitrogen supply system/and interaction with other air systems

- Fouling of heat exchangers.

Information relative to item 2 and 3 above was provided separately to the NRC, AEOD representative, and information with respect to item 1, was provided to the inspectors as background information for a review of the licensee's activities surrounding the recent SRV modification (see Paragraph 7).

7. Maintenance Observation (SRV Modification)

a. Background

The inspector reviewed Plant Design Change Request (PDCR) 80-04, SRV Top Works Modification, dated March 7, 1980. This modification involved the removal of the bellows relief assembly, solenoid and internals of six (6) model 67F Target Rock Safety Relief Valves, and the installation of new internals, new pilot stage and electropneumatic system, known as the "Topworks". The modification additionally required some drilling of new vent ports, plugging of other ports and machining of mating surfaces of the old valve bodies to attain required tolerances.

The licensee contracted Target Rock Corporation to refurbish the valve bodies with a subcontract arrangement with Wyle Laboratories. Target Rock Corporation also provided technical direction for on-site reassembly of the valves.

b. Scope and Acceptance Criteria

The inspector reviewed the maintenance activities involved with this modification to determine whether they were conducted in accordance with the licensee's technical specifications, approved procedures and applicable codes and standards.

The following documents were reviewed:

- PDCR 80-04, SRV/Topworks Modification
- Safety Evaluation No. 785
- Material Certifications
- Target Rock Corporation - SRV Retrofit Instructions
- GE Field Disposition Instruction 172-78003
- Maintenance Request 80-02 and maintenance summary sheets
- Target Rock Corporation letter E-14607, dated May 7, 1980
- Target Rock Corporation, Summary of Observations and Task Accomplished, Project 80Z59-2
- Target Rock Corporation, Field Disposition Data - Project 80Z59
- Boston Edison Company, Pilgrim Station, Unit No. 1, QAD Audit Report No. 80-12
- Boston Edison Company internal memoranda Nos. NOD 80-350 of April 4, 1980, and NOD 80-448 of May 8, 1980.
- General Electric Company Service Information Letter (SIL) No. 196, Supplement 6

- BECO Purchase Order No. 68017
- General Electric Company Procurement Quality Control Reports 00834 and 00821
- Target Rock Corporation Letter E-14607 to BECO dated May 7, 1980
- General Electric Company letter G-HK-9-123 to BECO dated August 29, 1980

c. Findings

(1) The inspector questioned the licensee's representative to determine whether the solenoid portion of the Topworks on each valve was environmentally qualified. The licensee could not produce material certifications for each solenoid stating its qualifications, however, did produce documentation from General Electric Company stating a summary of the qualification requirements and the test results of the Target Rock Solenoid Valves used on the new 2-stage assembly. This documentation stated that the equipment is qualified for its intended service. The inspector had no further questions regarding this item.

(2) The inspector noted that formal instructions for electrically connecting the solenoids to the plant electrical system were not part of the installation instructions. When questioned regarding this matter the licensee's representative provided the inspector with a memorandum which provided instructions for the electrical connections and electrical diagrams marked up as required to appropriately install the solenoids.

These instructions were incorporated into the Design Change Package as a Field Revision Notice. The inspector had no further questions concerning this item.

The inspector noted that a Boston Edison Quality Assurance Audit (QAD 80-12) had been performed at Wyle Laboratories during the Target Rock modification of the Safety Relief Valves.

The QAD audit identified no discrepancies and summarized that all work performed complied with the referenced documents and all testing proved satisfactory. The inspector reviewed the results of this audit and had no further questions.

No items of noncompliance were identified during this review.

8. <u>Location and Status of ENS and HPN Phone Circuits</u>

The inspector was requested to review the current location and status of the ENS and HPN phone circuits at the station. On October 17, 1980, the inspector met with the licensee's representative and reviewed the following items:

- specific location of each ENS and HPN phone unit,

- whether any modifications had been made to these units, and

- the current and planned location for the various incident response centers.

This information has been forwarded to the NRC Region I Emergency Planning Coordinator.

The inspector had no further questions at this time.

9. <u>Solid Waste Processing and Storage</u>

On October 2, 1980, the inspector was requested to question the licensee concerning topics related to low-level waste generations, storage, processing and shipping. The specific questions were provided to the licensee's Chief Radiological Engineer who forwarded the answers separately to an NRC Region I representative.

The inspector had no further questions.

10. <u>TMI Task Action Plan Category 'A' Requirements</u>

A. <u>Scope and Acceptance Criteria</u>

The inspector reviewed the licensee's activities concerning the implementation of the short-term TMI Action Plan requirements. The criteria used as a basis for acceptance included the following documents:

<u>NRC</u>

- NUREG 0578

- Letter from NRR to All Operating Nuclear Power Plants dated September 13, 1979

- Letter from NRR to All Operating Nuclear Power Plants dated October 30, 1979
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- Letter from NRR to All Licensees and Applicants dated September 5, 1980

BECO

- Responses to NRR dated December 31, 1979 and April 4, 1980.

The inspector's review included the following items:

- tours of the facility including observations of modifications and equipment in place
- review of administrative, system operating, surveillance, and emergency procedures
- review of training records
- review of completed pre-op test procedures
- discussions with licensee representatives

The following topics/areas were reviewed:

NUREG 0578 Number	TAP 0660 Number	Subject
2.2.1.b	I.A.1.1	Shift Technical Advisor
2.2.1.a	I.A.1.2 I.C.3	Watch Engineer (shift sup) Responsibilities
---	I.C.1	Accident Procedures - SBLOCA
2.2.1.c	I.C.2	Shift Relief/Turnover
2.2.2.a	I.C.4	Control Room Access
2.1.8.a	II.B.3	Post Accident Sampling
2.1.3.a	II.D.3	SRV/SV Position Indication
2.1.1	II.G.1	Power Supplies to SRV's, RV Level Instruments
2.2.2.b	III.A.1.2	Technical Support Center
2.2.2.c	III.A.1.2	Operations Support Center
2.1.6.a	III.D.1.1	Primary Coolant Outside Containment
2.1.8.b	II.F.1	Increased Range of Accident Monitoring Instrumentation
2.1.8.C	III.D.3.3	Improved Inplant Iodine Monitoring
2.1.4	II.E.4.2	Containment Isolation

B. Findings

(1) The inspector determined that in several areas the licensee's actions may not have been completed in entirety with respect to either the licensee's commitments or the NRC requirements as referenced above. Specific items are discussed below.

NUREG 0578 Number	TAP Number	Comment
2.1.8.a	II.B.3	There was no procedure in place for taking/handling a containment atmospheric sample for subsequent radiological isotopic analysis.
2.1.8.b	II.F.1	There is not an approved procedure for quantifying release rates from radiation levels sensed by the licensee's new high range noble gas effluent monitors.
2.1.4	II.E.4.2	The licensee's April 4, 1980 letter to NRR stated that the control switches for the containment vent and nitrogen makeup valves (AO 5033 A&B, AO 5041 A&B, and AO 5043 A&B) would be replaced with new switches which would allow operation between the 'open' and 'close' positions without a key and would only require a key to get into the 'emergency open' position. (The open position is isolated by hi drywell pressure and low reactor vessel level; The emergency open position is isolated by low low reactor vessel level). On October 30, 1980 it was determined that the key was required for both the 'open' and 'emergency open' positions.

Pending evaluation and correction of these items and review by the NRC these three areas are unresolved 950-293/80-24-03).

(2) Additional findings/comments are described below.

NUREG 0578 Number	TAP Number	Comment
2.2.1.C	I.C.2	There was no system established to periodically evaluate the effectiveness of shift turnover procedures. The license stated that a system would be implemented by January 1, 1981.

NUREG 0578 Number	TAP Number	Comment
2.2.2.b	III.A.1.2	The NRC ENS phone has not been moved to the present Technical Support Center. Coordination between the NRC, Telephone Co., and the licensee is in progress and the ENS phone is expected to be installed within a few weeks.
2.1.6.a	III.D.1.2	The licensee's April 4, 1980 response included a leak test of the Standby Gas Treatment System (discharge). This was deleted after review determined it was not needed due to system operation during an accident. The April 4, 1980 response included a leak test of the sampling system. The test scheduling had been overlooked but was immediately scheduled and performed on October 28, 1980.
---	I.C.1	The inspector confirmed the licensee's commitment to conduct formal training during simulator requalification or classroom training on January 1, 1981.
2.1.1	II.6.1	The licensee is investigating whether or not the 3 or 4 original air compressors are automatically reconnected to the emergency power following a loss of offsite power. Also, a description of the current system configuration, including newly installed air compressors will be addressed in future response to the NRC on this item.

The inspector will review these areas during future routine inspections.

(3) The inspector also provided comments to the licensee concerning several items identified during this review. These included the following subjects:

- Procedure where STA's duties are described

- Level of management issuing periodic directive to personnel concerning Watch Engineer's (W.E.) responsibilities.

- More specifically addressing command relationships between the W.E. and other plant management - who can give orders to the W.E.

- Specifying when shift turnover sheet is signed.

- Pre-op test's (TP80-50, 54, 55, & 51) require addtional information.

- Procedural method of requiring manual isolation if necessary in precaution section of several procedures (2.2.30, 2.2.35, 2.2.36, 2.2.85, and 2.2.86).

- Condition of computer console phone in T.S.C.

- Revisions to procedures 5.1.1.3 reflecting recent station organization change and correct references to attachments.

The licensee acknowledged the inspector's comments, stated that some of these items had already been identified, that further review would be performed and the appropriate action taken.

No items of nonocompliance were identified during this review of the short term requirements however, the inspector stated that these areas will continue to be reviewed by the NRC in future routine inspections.

11. Unresolved Items

Areas for which more information is required to determine acceptability are considered unresolved. Unresolved items are discussed in Paragraph 10.B.1.

12. Exit Interview

At periodic intervals during the course of the inspection, meetings are held with senior facility management to discuss the inspection scope and findings.