U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 87-09

Docket No. 50-293

License No. DRP-35 Category C

Licensee: Boston Edison Company

800 Boylston Street

Boston, Massachusetts 02199

Facility Name: Pilgrim Nuclear Generating Station

Inspection At: Plymouth, Massachusetts

Inspection Conducted: February 9-13, 1987

Inspector: C J Anderson for 4/22/87
Frederick Paulitz, Reactor Engineer Date

Approved by: C J Anderson 4/22/87
Clifford J. Anderson, Chief Date
Plant Systems Section, EB

Inspection Summary: Routine unannounced inspection of electrical systems and components, protective action and control of engineered safety features, and followup of previous inspection findings by one Region based inspector.

Results: No violations or deviations were identified.

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DETAILS

1.0 Persons Contacted

1.1 Boston Edison Company

* K. Roberts, Nuclear Operation Manager
S. Hudson, Operation Section Manager
* P. Hamilton, Senior Compliance Engineer
* M. Bosee, Maintenance Section Manager
* J. Pawlak, Power Systems Group Leader, Nuclear Engineering Department (NED)
M. Maquire, Station Electrical Engineer
* R. Williams, Senior System & Safety Analysis Engineer,NED
* T. McLoughlin,Nuclear Operation Department(NOD)
F. Mogolesko, Principle, System & Safety Analysis Engineer, NED
* J. Poorbaugh, Quality Assurance Department
R. Titcomb, Quality Assurance Department

1.2 U.S. Nuclear Regulatory Commission

* J. Lyash, Resident Inspector

* Indicates those present at the exit meeting held on February 13,1987.

2.0 Licensee Action on Previous Inspection Findings

2.1 (Closed) Noncompliance Item (81-22-01) Inadvertent defeat of Engineered Safety Features due to control power removal

This item relates to a maintenance request requiring the D4 breaker to be opened.This action deenergized power to panel No. C930. This deenergization resulted in a loss of redundancy for the high temperature and high steam flow rate automatic isolation function of the primary containment, and a loss of the trip capability for the RCIC low Pressure automatic isolation function. The operations personnel were not aware of the defeat of the associated safety features that resulted from opening breaker D4.

This initial noncompliance was identified in combined inspection report 81-18 and 81-22. The licensee responded to the Notice of Violation by letter, BECo Ltr #82-87, dated March 19, 1982. An NRC followup of the licensee's corrective actions was conducted during May 28-31, 1985. The details of this followup are described in inspection report 85-14. As a result of that inspection the licensee specified a tentative schedule of September 30, 1986 to complete the long term corrective action program. The licensee submitted by letter, BECo LTR #86-154, dated September 30, 1986, a revised schedule of activities related to the resolution of the Item of Noncompliance. Completion of the long term corrective action program was scheduled for January 31, 1987.

The long term corrective action program included the following:

* Verify by E203 walkdown project that selected control panels wiring, fuses, and components were in agreement with the station drawings.
* Issue Discrepancy packages to the Nuclear Engineering Department for disposition.
* Perform a circuit analysis to determine the effect on components of opening circuit breakers or fuses.
* Revise the following Operating Procedures to document the above effects and identify in the procedure the applicable Technical Specification:

  - 2.2.14 "125 Volt DC Battery System" Panel D4, D5,D6, D19, D36 & D37.
  - 2.2.12 "Safeguard Power Supply" Panel Y3, Y4, Y13,Y14, Y31, & Y41.
  - 2.2.11 "120 Volt AC Instrument Power Supply" Panel Y1.
  - 2.2.16 "120/240 Volt Vital Services Instr. Power Supply" Panel Y2.

The inspector reviewed the above procedures, which were approved by the licensee on February 2, 1987. One deficiency was identified with regard to the corrective action. The above procedures do not reference specific Technical Specifications in all cases. Instead the term Technical Specification General was used since multiple Engineered Safety Features systems were affected by inoperable support systems.

The licensee has agreed to replace the term Technical Specification General with the Technical Specification of the affected ESF systems.

The licensee has resolved eighty percent of the discrepancies identified by reviews conducted as part of the completed E203 Project. The remaining discrepancies include forty one wiring and forty one fuse discrepancies. The resolution of these remaining discrepancies are dependent on the completion of two engineering department tasks. These tasks are the existing fuse coordination study and a new project referred to by the licensee as E203 Phase III. The E203 Phase III project includes verification of safety-related panels which were not included in Phase II. Tracking and close out of the above fuse and wiring discrepancies will be via the Potential Adverse to Quality (PCAQ) Report System. The preliminary schedule for completion of the PCAQ's is July 19, 1987.

The licensee has developed an Operation Procedure No. 3M.3-44 "Control of Fuses" to provide qualified Maintenance and Operations personnel with detailed instructions necessary for the control and replacement of "Q" fuses. This procedure was under review by the licensee and had not been approved at the time of this inspection.

The inspector concluded that the licensee had completed the long term corrective action. The above discrepancies are being tracked and are scheduled for completion July 19, 1987. This item is closed.

2.2 (Closed) Inspector Followup Item(IFI) (85-30-13) Battery Procedure Deficiencies

The NRC PAT Team identified the following deficiencies:

* The battery capacity in the periodic battery performance tests of the battery capacity was not calculated in accordance with Operating Procedure 8.9.8, "Battery Rated Load Discharge Test."

* The test procedure did not specify that the cell voltage measurement should be made from the terminal of one cell to the corresponding terminal of the next cell. This procedure would not detect excessive intercell contact resistance.

* The test procedure did not specifically state that the electrolyte temperature at the start of the discharge test should be used in the capacity calculation.

The inspector reviewed the 250 Volt Battery D3 test per OP 8.9.8 conducted on July 1, 1986,to establish if the battery capacity was calculated in accordance with the revised procedure. The inspector determined by independent calculations that the licensee had used the average end of test electrolyte temperature in the capacity calculation instead of the average electrolyte temperature at the beginning of the test. This was a repeat of one of the deficiencies listed above. However, there was a problem with the temperature measurements. As a result, the licensee had voided this test. A retest was conducted on July 2, 1986. In the retest the licensee used the correct temperature in the battery capacity calculation.

The inspector concluded that this item is closed based upon the following considerations:

* The licensee is calculating the battery capacity according to the procedure 8.9.8.

* The procedure had been revised to include the voltage drop of intercell connectors.

* The licensee will add a note to procedure page 8.9.8A-4, revision 13, that the temperature that should be used for the correction factor, K1, is the average electrolyte temperature at the beginning of the test.

2.3 <u>(Closed) IFI (85-30-14) Battery Technical Specification (T.S.) Adequacy of the Operability Definition</u>

Batteries were not considered inoperable in the T.S. Limiting Condition for Operation(LCO) 3.9 until the battery voltage dropped below 105 volts direct current(VDC) on the 125 VDC system and 210 VDC on the 250 VDC system. This same definition for battery operability is used in the battery procedures.

The inspector determined that the licensee proposes to revise the T.S. to agree with the following definition for operability:

* Minimum total battery voltage of 130V for the 125 VDC battery.

* Minimum total battery voltage of 260V for the 250 VDC battery.

The above definitions for operability were incorporated into the following procedures as acceptance criteria.

* O.P. 8.C.14 "Weekly Pilot Cell and Overall Battery Check and Weekly Battery Charger Test", revision 18, approved December 4, 1986.
* O.P. 8.C.16 "Quarterly Battery Cell Surveillance" revision 11, approved July 11, 1986.
* O.P. 8.9.8 "Battery Rated Load Discharge Test",revision 13, approved October 21, 1986. (After recharge)

Based upon the licensee's proposed T.S. revisions and the revisions made to the Operation Procedures listed above the inspector concluded that this IFI is closed.

2.4 <u>(Open) IFI (85-30-15) Motor Operator Valve (MOV) Overload Protection</u>

In a previous inspection, the NRC determined that the licensee selection of MOV thermal overload heaters was not consistent with the vendors recommendations nor was it consistent between MOVs having the same function and horsepower rating. Presently the heaters provide no MOV protective function but are used for alarm only during surveillance testing or accident conditions. The NRC had been concerned that the above inconsistencies might impact the alarm function resulting in MOV failure.

The licensee has conducted an industry survey to determine what other utilities are doing about MOV protection. The licensee is also reviewing the recommendations of the valve operator manufacture's, the nuclear steam supplier and the Architecture Engineer. The licensee had not decided what method should be used to provide MOV protection without also affecting MOV availability or presenting protection coordination problems. The present design precludes electrical breaker tripping coordination should a MOV motor have an (locked rotor) overload condition. This IFI remains open pending completion of the licensee's corrective action to provide MOV protection and electrical system protection coordination.

2.5 (Closed) IFI (85-30-16) Core Spray Inboard MOV 1400-25A(B) Open Signal Not Sealed In

During a previous inspection the following observations were made concerning the control of MOV 1400-25A(b) shown on elementary diagram MIK-16, Rev. E2:

* The circuit was not revised to note a wiring change on the circuit internal to the motor control center as a result of Field Revision Notice (FRN) 79-28A.1-03, dated March 3, 1980.

* A normally closed contact from relay 14A-K20A was used in this circuit. The description of this relay as given on drawing MIK 4-11, Rev. E3, was incorrect and did not agree with Operating Procedure 2.2.20 "Core Spray System." The description of this relay stated that an auto signal would be sealed-in, but instead it appeared that it would be bypassed. The first sheet of the core Spray system elementary diagram references IEEE 279. IEEE 279-1971, section 4.16, requires that once initiated, a protective action shall go to completion. The valve circuit does not contain a seal-in for the automatic safety signal so that the valve would complete its safety function by going to the full open position.

During this inspection, the inspector reviewed Elementary Diagram Core Spray System Sh 6 of 6 Drawing No. MIK16, revision E3. This drawing had been revised with a note on the motor control center terminal connections for MOV 1400-25A(B). This note states that the seal-in contact 42/C for valves 1400-25A(B) is disconnected internally from terminal point 4. The wires associated with terminal point 4 are for the MOV opening circuit. This action resolves the first deficiency related to FRN 79-28A.1-03.

The relay 14A-K20A(B) is a bypass relay which enables the operator to bypass (override) protective system actuation. The licensee stated that the relay description on drawing MIK -4-11 was in error. This error has apparently existed for some time. However, operating procedure 2.2.20 correctly states that the bypass function exists and the name plates on the Main Control Board at the controlling switches are correct. The licensee has determined the cause to be a drafting or checking error on the original drawing. The licensee noted that the original system Technical Manual carried the same error on the G.E.'s counterpart drawing to MIK-4-11. The inspector reviewed Elementary Drawing Core Spray System Sh. 2 of 6 Drawing No. MIK-4-11, revision E5. and determined that it had been revised to indicate that relay 14A-K20A(B) is the open logic by-pass for MOV 1400-25A(B). This drawing now agrees with operating procedure 2.2.20 and the actual control board name plates, which were correct.

The IEEE standard 279-1971, Section 4.16 Completion of Protective Action Once It Is Initiated states "The protection system shall be so designed that, once initiated, a protective action at the system level shall go to completion. Return to operation shall require subsequent deliberate operation action." The PAT Team concern was that a seal-in at the component (motor control center starter) level did not exist. The intent of the IEEE standard was to assure that the protective action occurred. The inspector noted that the protective action at the system level is provided by either the reactor vessel low level or high drywell pressure. The inspector reviewed the FSAR and determined the following:

* A minimum signal time of 10 seconds is needed. Section 7.4.3.4.4, Core Spray System Valve Control, indicates that the full stroke design operating time of the pump discharge valves is 10 seconds.

* Table 7.4-3 indicates one of the initiation process parameters for core spray system actuation is high drywell pressure, with a set point of 2.5 PSIG. Figure 14.7-10 Loss of Coolant Accident Primary Containment Pressure Response indicates that the containment pressure does not return below 2.5 PSIG until 16 hours after the accident.

* Table 7.4-3 indicates that the other initiation process parameter for core spray system actuation is reactor low-low water level, with a set point of 36 feet above vessel zero.

* Figure 6.5-9 Initial Core Response to flooding alone for a 4.4 square foot recirculation line break indicates that core spray is on 60 seconds after the accident with the swollen level reaching 36 feet at 180 seconds after the accident.

The fact that the core spray system is not sealed in at the system level nor the inlet valve sealed in at the component level should not prevent the valve from going to the full open position for the following reasons:

* The reactor vessel level does not return to the initiating low low level setpoint until 120 seconds after core spray initiation.

* The drywell pressure does not return to the initiation setpoint until after 16 hours.

The inspector concluded that the licensee has corrected the drawings discussed above and although the core spray system actuation does not "seal-in"at the system level or component level there is reasonable assurance that the core spray system valves, open actuation signal, will be maintained long enough for the valves to open fully. This IFI item is closed.

2.6 (Closed) IFI (86-37-07) Primary Containment Isolation System (PCIS) Design Discrepancy versus Technical Specification (TS)

The licensee had retained a technical consultant to assist the Quality Assurance Department's review of the TS and safety systems design to assure compatibility. During this TS review, Audit Number 86-52, on

October 31, 1986, the licensee identified that there was a possible design deficiency in the PCIS logic which did not agree with the TS. The TS Notes for Table No. 3.7.1 indicates that Group 1 containment isolation should occur for 5 conditions. The fifth of these conditions states that containment isolation should occur when "Main steam line has low pressure or the reactor vessel has high water level". However, the TS Table 3.2.A notes 5 and 8 state that the low main steam line pressure signal is bypassed when not in the "RUN" mode and the high level is not required in the "RUN" mode. The bases for Section 3.2 of the TS state that the protection provided by the main steam line low pressure in the "RUN" mode is replaced by the high water level trip when in the "REFUEL" or "STARTUP" mode.

The Elementary Diagram Primary Containment Isolation System Sh 7 of 17, Drawing No. MIN33-10,revision E1 shows the reactor vessel high level signal in series with the mode switch which are in parallel with the steam line low pressure signal. The note on the mode switch states "BYpass steam line low pressure trip, Open in "RUN" mode only. The circuit is normally energized and requires contacts to open to cause actuation. When the mode switch is placed in the "RUN" position the high water level contact has no circuit function. When the mode switch is in the "REFUEL" mode the low steam pressure contact has not circuit function since the pressure will be less than 880 psig and the contact open. However, when the mode switch is in the "STARTUP" position and the reactor pressure is above 880 psig <u>both</u> low steam pressure and high water level are required to actuate Group 1 isolation.

The licensee review of the intent of the isolation function indicated that only low steam line pressure trip was required in the "RUN" mode and only high reactor level trip was required in the "REFUEL" mode. The logic design need not be modified however the TS Notes for Table No. 3.7.1 would be revised as follows:

- * 5. Main steam line low pressure when mode switch is in the "RUN" mode.
- * 6. Reactor vessel high water level when main steam pressure is less than 880 psig and the mode switch is not in the "RUN" position.

The inspector concluded that the licensee corrective action to revise the TS notes instead of making a design change to the group 1 containment isolation logic was appropriate since the necessary containment isolation functions are provided. The inspector noted that the licensee has initiated efforts to change the Technical Specification to clarify the above. This IFI is closed.

3.0 <u>Core Spray System</u>

During the review of IFI(85-30-16), which is closed in paragraph 2.5 of this report, the inspector noted the following discrepancies between

Operation Procedure No. 2.2.20 "Core Spray", revision 29, approved January 31, 1987, and the following drawings:

* Core Spray System P&ID drawing M242.revision E11
* Core Spray System FSAR drawing figure 7.4-8.

The procedure indicates the following valve position:

* Page 8, VII Operating Procedures, A. Standby Status,
  5. MO-1400-25 A&B Switch in Auto. Valve position indication indicates valve open on Panel C903.

  6. MO-1400-24 A&B Switch in Auto. Valve position indication indicates valve closed on Panel C903

* Page 2.2.20B-3. Valve Check List
  - MO-1400-25A normal position open
  - MO-1400-24A normal position closed

* Page 2.2.20B-10 Valve Check List
  - MO-1400-25B normal position open
  - MO-1400-24B normal position closed

The drawing indicates that valves MO-1400-25A(B) are normally closed and valves MO-1400-24A(B) are normally open. However the procedure indicates that MO-1400-25A(B) are normally open and MO-1400-24A(B) are normally closed. Further support that the correct valve positions are as shown on the drawings is the Core Spray System Elementary Diagram, Drawing No. MIKI6, revision E3. The elementary diagram has a low reactor pressure (400 psig) permissive in the opening circuit of MO-1400-25A(B) whereas, MO-1400-24A(B) on elementary diagram MIK6-8,revision E4, has no pressure permissive in the opening circuit. Further evidence that MO-1400-25A(B) are normally closed is the OP 2.2.20 page 9 under response to auto initiation. The OP has the operator "observe that valves MO-1400-25A(B) open when the reactor pressure decreases to 400 psig."

The licensee's is explanation for the valve positions being different between the procedure and the drawings was based upon General Electric identification that auxiliary relay type HGA was not seismic qualified. Thus, a seismic event might prevent the indicated valves to be normally closed on the P&ID from opening upon demand.

The above discrepancy in valve position between the operation procedure and the drawings and the lack of seismic qualification of the HGA relay is an unresolved item (87-09-01).

4.0 Direct Current Electrical System

The 125 volt and 250 volt direct current (DC) system was inspected to determine the effectiveness of the licensee's preventative maintenance program.

The inspector observed the following condition of the station batteries:

* A gray material at the top of the electrolyte which also made it difficult to observe the electrolyte level.
* Material, 1-1/2 inch long by 1/4 inch diameter, lying across the top of the plates on the back side of cell # 49 , 250 V battery , Room A.
* Plate spacer material loose and floating in the electrolyte.
* A 1 inch space between the battery rack end rail and battery cell 1,40 & 41 and 0 inch space between the end rail and battery cell 20, 21 & 60 of the 125 V battery, Room A.
* Visible corrosion material on cell 59, 125 V battery, and cell 40, 250 V battery, Room B.
* Deficiency I.D.8504, B-125VDC -#15 low S.G., MR 87-46-18, dated 1-13-87, tag attached to battery room B door.

The licensee had contacted the battery vendor about the first four items listed above. The vendor,C&D Power Systems Inc., letter dated December 15, 1986, stated that the white substance found on the surface of the electrolyte has been identified as minute fragments of glass fiber. This fiber came loose during the handling of the mat and separator assembly during the manufacture of the cell. Further, it indicates an excess amount of the binder was used to adhere the glass mat assembly to the separator. Since all materials used in the manufacture of the cells are compatible with other components within the cell, no adverse conditions will develop regarding cell life or performance. This vendor letter referred to a previous battery inspection by the vendor. However, there is no indication that a sample of the floating material had been removed for analysis. Further, neither the licensee nor the vendor have addressed the significance of material described as item two above on battery service.

The vendor has advised the licensee that the space between the battery rack end rail should be between 1/4 to 1/2 inch in order for the batteries to conform to the seismic type test. The licensee took prompt corrective action to adjust the 125V battery A rack end rails to the vendor's space specification.

Regarding the low specific gravity (SG) condition of cell 15 of battery B, the licensee stated that a maintenance request(MR) had been withdrawn, since the low SG could be corrected by stirring the electrolyte. There was no indication, at the time of the inspection, if corrective action had been taken since the deficiency tag was still attached to the battery room door.

The licensee proposes to remove the battery intercell connection corrosion during some future preventative maintenance activity.

The above battery maintenance issues constitute an unresolved item (87-09-02) pending NRC review of the following activities by the licensee:

* Identify the materials listed in the first three items above and determine their effect upon battery operability.
* Corrective action for the low SG condition noted above.
* Corrective action for the intercell connection corrosion deficiency noted above.

5.0 Exit Interview

The inspector met with the licensee representatives, denoted in paragraph 1 on February 13, 1987. The inspector summarized the scope and findings of the inspection at that time.

No written material was provided to the licensee by the inspector.