BOSTON EDISON COMPANY
800 BOYLSTON STREET
BOSTON, MASSACHUSETTS 02199

WILLIAM D. HARRINGTON
SENIOR VICE PRESIDENT
NUCLEAR

June 16, 1986
BECo Ltr #86-079

Dr. Thomas E. Murley
Regional Administrator
U.S. Nuclear Regulatory Commission
631 Park Avenue - Region 1
King of Prussia, PA 19406

Subject: Second Response to NRC Confirmatory Action Letter #86-10, Regarding the Events Which Occurred on April 4, 11-12, 1986 at Pilgrim Nuclear Power Station

References:
(A) NRC CAL 86-10 dated April 12, 1986
(B) BECo Response to CAL 86-10 dated May 15, 1986
(C) NRC Inspection Report 86-17 dated May 16, 1986, Documenting the NRC's Augmented Inspection Team's Findings, Conclusions, and Recommendations
(D) NRC "Request for Additional Information" letter dated May 16, 1986

Dear Dr. Murley:

This letter provides the additional information requested in your letter dated May 16, 1986 and in a May 19th meeting held at the Region 1 office. Answers to the specific questions of the letter are included as Attachment (1) to this letter, and the enclosed Table of Contents will provide a cross-reference for other information requested by members of your staff.

We trust that the contents of this submittal combined with information provided in our preliminary response dated May 15, 1986 will provide information adequate to address the requirements of Confirmatory Action Letter 86-10.

Should you have further questions concerning these matters, please do not hesitate to contact me.

Sincerely,

W D Harrington

W. D. Harrington

JC/vep

Attachments

IE36
1/1

TABLE OF CONTENTS

ATTACHMENT 1

BECo Responses to NRC Questions As Presented

In NRC Letter Dated May 16, 1986

RHR

NRC Question: a. What was the rationale for not refurbishing both the 28B and 29B RHR valves and how will you control the normally closed injection valve in the "B" RHR loop in the future to isolate the RHR system?

BECo Response: We decided to refurbish only the 28B valve for the following reasons:

1. The "as found" leak rates were low and well within Type C LLRT limits.

2. Extensive maintenance experience with the 29B valve indicated that there was very little probability of improving upon its "as found" leak rate.

28B was not refurbished

The 28B valve was the "normally closed" injection valve when the RHR pressurization problem began; therefore, a change in this valve's leak rate had occurred.

4. The 28B valve is used to throttle flow and therefore, has the highest probability of experiencing seating surface wear.

The normally closed injection valve in the "B" RHR Loop will be controlled by system lineup procedures. It is expected that the MO 1001-28B valve will be the normally closed isolation valve, upon startup from the present outage.

NRC Question: b. What is the scope of the planned inspection for 28B valve?

BECo Response: An inspection of the disassembled MO 1001-28B valve internals was performed by BECo Maintenance and Quality Control Personnel. This consisted of a visual examination of the valve seat and disc and a bluing check of the disc seat. The NRC Resident Inspector witnessed the valve disassembly. The valve was local leak rate tested with air to 10CFR50 Appendix J criteria and with water at 950 psig as part of postwork testing. Inspection and testing results are presented in Attachment (2) to this letter.

NRC Question: c. Which recommendations in Attachment 4 to your letter apply to both RHR loops? For example, will pressure and temperature instrumentation be installed on both loops?

BECo Response: The following recommendations from our prior response letter (Reference B) apply to both RHR loops: B.1, C.1.b, C.1.c, C.2.a, C.3.a, C.3.b, C.4.a, C.4.b, C.5.a, and C.6. A table explaining these items is included later in this response(Page A1-24). More specifically. pressure and temperature monitoring devices will be installed on both loops of the RHR system.

NRC Question: d. Are baseline torque measurements on the RHR motor operated injection valves to be taken?

BECo Response: No. However, closure seating current as well as motor running current are recorded as part of post-maintenance testing of the RHR motor operated injection valves in accordance with Procedure 3.M.4-10. These current measurements are indicative of the operational condition of this motor operator.

NRC Question: e. What method will be used to control RHR system leakoff? How will the amount of leakage into the low pressure portion of the RHR system be measured? If a bypass valve is used, when will it be opened or closed? Will any administrative procedures be put in place to monitor and control letdown to the suppression pool and suppression pool level? A safety evaluation should be submitted to the NRC which evaluates the leakoff method.

BECo Response: RHR system leakoff will be controlled through throttling open a bypass valve around the RHR pump discharge check valve. Prior to opening this valve, the RHR intersystem leakage will be quantified by maintaining a pressure in the RHR system above the keep fill pressure (>125 psig) and measuring the unit volume of fluid leakage per time to maintain the set pressure. The RHR pump discharge check valve bypass valve will then be throttled open to maintain the same pressure as used in the measurement step. This valve will then be secured in that position with tie wraps.

The measured leakage will be controlled procedurally. Continued station oepration will not be permitted above 1.0 gpm withouth an additional Nuclear Engingeering Department evaluation. Once the throttled valve has been opened it will remain open as a controlled leak path. Subsequent RHR high discharge pressure alarm while in normal power operation will indicate that RHR intersystem leakage has exceeded the procedural limit and that additional corrective actions are to be initiated. Such corrective actions will include quantification of the leakage, temperature monitoring, and evaluation of acceptability of the leakage. A unit shutdown will take place if the 1.0 gpm flow rate is exceeded.

Since post-work testing of the MO 1001-28B valve did not exhibit an improvement in leak tightness the RHR leakoff will be used upon unit startup. In the event that RHR high pressure alarms recur, the controlled leakoff will be implemented. The alarm procedure has been revised to key the described actions and a procedure has been prepared to implement the controlled leakoff. The procedures are included as Attachments (3) and (4) to this response. Administrative procedures are already in place to monitor and control suppression pool level per technical specifications. A safety evaluation has been prepared for the controlled leakoff and is included as Attachment (6) to this response.

Training of the operating staff on the RHR intersystem leakage issue will be conducted on shift concurrent with plant startup.

NRC Question: f. What type of test (water or air) will be used to check the pressure drop across the 68 check valves? What test pressure will be used? What acceptance criteria will be used? Will this test be conducted at refueling outages or after each time the check valves are cycled?

BECo Response: Water testing of the injection check valve will be used as the preferred method for testing pressure drop capability of the 1001-68A and 1001-68B check valves. The test pressure will be approximately 925 to 975 psig with the system vented outboard of the check valve. Acceptance criteria are to be determined by our Nuclear Engineering Department to be consistent with respect to the mission of the check valve (to prevent reverse flow) and the downstream overpressure protection. The test described in this response will be performed during each refueling outage.

NRC Question: g. Item C.1.c in Attachment 4 to your letter indicates that a check valve position indication system will be designed. Will this indication system supplant more quantitative means of verifying valve position (e.g., leak rate testing)?

BECo. Response: The statement in our letter and in the RHR Task Force report is that the check valve design providing position indication would be based on an evaluation of need. Any decision on replacement or modification of the check valve to install position indication will be made after designs are studied and the benefits determined. Position indication devices, if added, will <u>not</u> replace the once-per-refueling outage leak test.

NRC Question: h. Will the RHR pressure gauges be used on a routine basis to check system pressure? If so, at what frequency will they be checked? Will they be located in both RHR loops? Will they be alarmed? Will they be used during valve operability testing to ensure valve closure? What will be their calibration frequency? Where will they be read out? What will be lost by the removal of the pressure gauge in item C.3.c?

BECo Response: RHR's system pressure gauges will be used on an as needed or on demand basis, i.e. upon alarm on RHR system high pressure or upon discovery of increased pipe wall/component temperature on the system. Pressure gauges will be installed on both RHR loops. The existing alarms from pressure switches PS 1001-74A and B will still be used. The pressure gauges will normally be isolated and valved out of service. Calibration of the pressure gauges will be once per refueling outage and upon request. The gauges in between the MO 1001-28 and MO 1001-29 valves will read out locally in the vicinity of the respective equipment rooms. The gauges outboard of the MO 1001-28 valves will read out locally in each RHR quadrant near the system instrument rack. The gauge removed by recommendation C.3.c will be replaced by the gauge between MO 1001-28B and 29B and the local vent for the injection line will be gained.

NRC Question: i. What specific action will be taken if the RHR system high pressure alarm annunciates in the control room? A copy of the revised alarm procedure (item C.5.a) should be submitted to Region 1 prior to restart. What are the maintenance and calibration histories of the RHR high pressure alarm switches?

BECo Response: Specific action to be taken per the revised alarm procedure 2.3.2.1 (Attachment 3) is to log the occurrence of the alarm and to let down pressure as directed by the alarm procedure. If the alarm occurs at a frequency greater than once per 12 hours, then action will be initiated to investigate and determine the leak rate. If a leak rate 1 gpm is exceeded, a unit shutdown will commence. A separate procedure, TP 86-85 (Attachment 4) has been prepared to accomplish the investigation/evaluation of the RHR leakage with respect to the controlled leakoff. After trial use as a temporary procedure, the method will be incorporated as part of the RHR system operating procedure.

The maintenance and calibration histories reveal that pressure switches PS 1001-74A and B have not required maintenance since initial plant startup (1972) and are not on a periodic calibration schedule. The actuation point of the switches as found was conservatively set low (approximately 360 psig as found versus 392 psig set point). The pressure switches will be placed on a once per cycle calibration schedule.

NRC Question: j. How will the RHR system be vented? Will the "A" RHR loop also require venting between the 28 and 68 valves?

BECo Response: The RHR system will be vented from four local high points. One vent is located on each loop of the LPCI injection lines and one on each containment spray line. This venting will be performed on both loops per TP 86-84 (Attachment 5). Venting on the LPCI injection lines will be performed between the MO 1001-29 and 28 valves when the reactor is at pressure.

NRC Question: k. How will RHR system temperature monitoring be conducted and at what frequency? What locations will be monitored (relative to injection valves and other RHR components)?

BECo Response: Temperature monitoring devices in the form of adhesive temperature sensitive markers will be placed on the valve bodies/bonnets and uninsulated metal surfaces in the RHR system from the containment wall to the MO 1001-28B and A valves. There is approximately 9" of uninsulated piping

outboard of the MO 1001-28B valve which will have a marker. The markers will be checked once per week for change. If a change in temperature is detected, or the RHR high pressure alarm (from PS 1001-74B or 74A) annunciates a portable temperature monitoring device will be used to measure local pipe wall temperature. This is part of the investigation procedure (TP 86-85 Attachment 4).

NRC Question: 1. What is the schedule for completion of items C.1.c, C.4.a, and C.4.b? With regard to the footnote in Attachment 4 to your letter, long term action items should be included in the next Long Term Plan revision submitted to NRR. However, the proposed schedule should also be submitted to Region 1 prior to restart of the plant.

BECo Response: Proposed schedules for RHR Action items are:

1. Item C.1.c "Evaluate the feasibility of replacing or redesigning check valves to provide positive position indication." (This is a long term item and should be based on a needs evaluation.)

   BECo will perform a review of industry experience with this application and a design review with GE, Bechtel, and the valve manufacturer (Rockwell). This design review started 5/15/86 and is expected to be completed by approximately 12/1/86. Any design modification recommendations which may result from this review will then be scheduled via the Long Term Plan.

2. Item C.4.a "IST program opens both 68A and 68B check valves once per quarter, if in cold shutdown change frequency to once per refuel. This action is long term because the IST program relief may not be possible."

   This recommendation has been reviewed by the coordinator of the Inservice Test (IST) program. The response to this proposal is that a lengthening of the test frequency for valve opening is <u>not</u> recommended. The ASME code Section XI of 1980 edition (Winter, 1980 addenda) and 10CFR50.55a(g) both indicate that since the RHR injection check valves (valves 1001-68A and B) are Category C valves and that a frequency of quarterly is appropriate. The test frequency of once per quarter if in cold shutdown is permitted by an approved relief request to our IST program. An increase of test frequency cannot be said to be impractical per 10CFR50.55a(g)(4) since it has been demonstrated that the test is practical as performed at its present frequency. Therefore, it is concluded that this recommendation will not be implemented .

3. Item C.4.b "Reduce frequency of test. 1001-28B and 29B were stroked 62 times between 11/84 and 2/86. This action is long term because it requires extensive Tech Spec changes."

The proposed changes to Technical Specification would include all ECCS system testing; the major change would be on the subject of test frequency. In simplified terms, Technical Specification requirements to test redundant components upon entry into an LCO will be replaced with a philosophy that if the redundant component is within specifications and its surveillance frequency is current, then additional testing would not be required. An anticipated schedule for these changes is that submittal would take place (9) months from now (or March, 1987). This activity will be included in our Long Term Plan.

MSIV

NRC Question: Discuss the design review and implementation process associated with the 1983 modification of the MSIV's.

BECo Response: Boston Edison recognized its responsibility for the design adequacy of the Main Steam Isolation Valve redesign and followed its standard design change process in designing and implementing the 1983 modification. The design change objectives were achieved by the redesign and with the setscrew problem resolved, the valves will perform better than the original design.

Boston Edison's standard design change process includes a number of controls to ensure the adequacy of the final product. The process begins with a requirements analysis to identify the main objectives of the change. This is followed by a scope, justification and approval phase where organizational agreement is reached relative to the modification. Next, engineering resources prepare a conceptual design package (if applicable) which includes more specific information regarding the planned modification. A multi-disciplined review of the conceptual design is performed including a constructability review. Upon receipt of organizational approval of the conceptual design, the engineering department proceeds with detail design. The detail design results in a Plant Design Change package (PDC). This package receives a multi-disciplined review, design review board review and approval, constructability review and then is forwarded to the station for review. At the station the package is routed for review to ORC members and once the package has been approved by those individuals, it is released to the Station Manager for implementation.

In this specific case, the design process was initiated in October of 1982, well in advance of refueling outage six. The requirements analysis, conducted over a four month period through January 1983, consisted of a review of plant-specific and industry-wide operating experience with the MSIV's. Industry problems related to valve leakage reported in the US NRC Inspection and Enforcement Branch's Information Notice 82-23 were to be addressed by the design change as well as plant-specific problems related to main valve poppet and guide wear and valve stem failures (Licensee Event Reports 78-019 and 82-36).

A number of candidate improvements to address these problems were suggested by maintenance, engineering, the valve manufacturer and the General Electric Boiling Water Reactor Owner's Group MSIV Subcommittee which was supported by the Institute of Nuclear Power Operations, Electric Power Research Institute and Nuclear Safety Analysis Center. The requirements analysis phase culminated in the selection by engineering, maintenance and the valve manufacturer of an optimum set of valve improvements.

A project scoping document justifying the set of improvements was prepared by engineering and maintenance, issued by both the Engineering and Operations Managers and was approved by the Vice President of Nuclear Operations. The scope agreed upon included provisions to resolve the problems identified in the requirements analysis phase.

MSIV leak-tightness was to be improved by a revision to the main poppet resulting in an elongated configuration for better seating of the main poppet. A self-aligning pilot poppet design was to be provided for improved pilot poppet seating. Guide and poppet wear was to be reduced through the use of a poppet anti-rotation design feature. Stem failures would be eliminated by increasing the stem diameter, eliminating a stress riser on the stem at the backseat area and by using the floating pilot poppet.

The detail design change process began in June of 1983 and followed standard practices and procedures. The valve manufacturer was contracted to design and manufacture parts to modify the MSIV's. Deliverables procured included assembly drawing, parts, installation instructions, machining drawings for modifying existing parts, instruction manual and design report. The Nuclear Engineering Department received and approved the valve manufacturer's design information and used it in preparing the PDC.

Following design review board review of the PDC, a pre-implementation meeting was held for the MSIV modification. The sole purpose of the meeting was to generate complete, clear and effective work instructions to implement the modification. The framework used was the manufacturer's installation instructions which had already been reviewed and approved by engineering. Participants included personnel from maintenance, design engineering, the valve manufacturer, quality control, field engineering and field craft supervision. The work instructions were agreed upon and the final assembly procedure was added to the PDC.

This set of documents specified the assembly steps to be followed, called out hold and witness points and provided for sign-off by quality control, field engineering, and construction supervision. A Maintenance Request provides control of the activity and is required for any work at the station. Implementation of the MSIV modification was governed by a Maintenance Request. The Maintenance Request approval sequence includes a statement of authorized work scope, review by Quality Control (and insertion of their quality requirements) preparation of plant systems for performance and the Watch Engineer's approval to start work.

After the PDC was issued for construction and the MR approved, Boston Edison implementation was assigned to the maintenance section with support to be provided by the site Resident Contractor. Then the Resident Contractor prepared detailed assembly documents called Work Process and Inspection Reports (WPIR).

The package was implemented in accordance with our standard work processes and procedures. Upon completion of the physical construction work, the documentation was submitted by the Resident Contractor to maintenance for post-work testing and final acceptance.

Mode Switch

NRC Question: Please submit a written assessment of the loose wires and drawing/wiring discrepancies that were identified during your investigation. The assessment should address what was found, the implications with regard to the unanticipated primary containment isolations, other safety implications, and your corrective actions. The assessment should provide a basis for concluding that similar conditions do not exist in other safety related systems.

BECo Response: A review of loose wiring found during the PCIS system walkdown showed that it was possible to come up with combinations that could lead to an inadvertent PCIS initiation. The evaluation team concluded that, although it is unlikely that these combinations would have repeated themselves during two successive shutdowns, at approximately the same time during the shutdown process, they could not be discounted as potential causes to the events of 4/4 and 4/12. It should be noted that terminal board terminations listed as being loose, were not excessively loose; a quarter of a turn on the terminal screw tightened the lugs to the acceptable point. The loose neutral terminations were felt to be more of a contributor than the terminal board connections.

The loose neutral terminations were found to be a result of improper compression lug size on the PCIS, RPS neutral busses. The existing lugs were meant for wire sizes larger than #12AWG, as is used in C915, C917, and C916. In most cases, only one wire was connected under each lug adding to the possibility of bad terminations .If a number of wires were landed under each lug , the problem may not have existed. In some cases, the wires were found to be improperly stripped; this problem was found to have existed from the day the plant was built and there probably would have been no reason to check these terminals until now. These loose terminations have been corrected using acceptable compression lugs. An investigation is being performed by the Nuclear Engineering Department to assess the safety significance of the findings.

A thorough walkdown of similar design safety related panels indicates that the use of incorrectly sized compression lugs are limited to the PCIS/RPS cabinets C915, C916, and C917. Therefore, no other safety systems are affected by this problem. All circuits identified as having loose terminations would have failed in the safe direction causing either a partial or full PCIS or scram condition. Therefore, at no time was the safety of the public or the safe operation of the plant compromised.

A review of the wiring discrepancies, noted during the investigative walkdowns, indicates two types of deviations, the first being typographical errors on the associated prints, and the second being electrically equivalent circuits. In no case, was a circuit found to not operate as designed or as intended by the elementary control diagrams. A contact off the high water level switch in the bypass circuit of the Main Steam line low pressure PCIS trip on Drawing MIN 33-10 was the only exception. This error is an omission from the elementary diagram and has since been corrected. The bypass circuit is wired and operates as designed as is shown on all other associated electrical prints.

The team concluded that there is no reason to believe improper safety system operation could result from drawing discrepancies because of the above findings. The MIN 33-10 discrepancy is being treated as an isolated case where it was captured on all other associated prints and no similar discrepancies were found.

Print discrepancies have been noted and forwarded to the Nuclear Engineering Department for corrections on 5/18/86. Current quality control practices should eliminate the recurrence of the improper use of compression lugs in any plant wiring in the future.

One troubleshooting technique used by the mode switch team was a "hand over hand" wire check. This involved physically and visually tracing wires from point to point in the suspect logic circuits. This method was employed to assure the team that the panels were indeed wired as shown and no new or extra wires had been added to the logic circuit incorrectly.

This check revealed several circuit problems.

The following is a brief listing of the logic sub channels (PCIS or RPS) affected and a brief description of the problem found. As can be seen by matching the subchannel combinations, it is possible to come up with combinations that could cause a full PCIS or a full reactor scram. The fact that this is possible, does not make it probable that two wires would become loose at exactly the same time in two different controlled reactor shutdowns eight days apart.

PCIS Subchannel
Affected

1. A1 Loose fuse - affected 16A-K7A contributed to a Group I isolation channel (MR-86-45-189 was issued to correct).

2. A1 Terminal block loose screw connection panel C2205A affects Yarway hi water level isolation signal - directly affects 16A-K7A (Group I isolation) when not in "RUN" mode (MR-86-45-193 was issued to correct).

3. A2 Panel C905 terminal block loose screw - directly affects main steam line low pressure bypass with the mode switch in "RUN" - could cause a 1/4 Group I isolation signal channel A2 by de-energizing relay 5A-K7C (MR-86-45-190 was issued to correct).

4. B1 Loose wire at RK2205A - affects bypass around main condenser low vacuum and main steam line valve closure - could cause 1/2 scram from subchannel B1 (MR-86-45-192 was issued to correct).

5. B2 Loose screw connection at RK2256B - affects relay 16A-K3D (steam line hi flow) - directly affects 16A-K7D (Group I isolation) (MR-86-45-191 was issued to correct).

6. Scram Reset A
and Channel A1 and A2

Neutral bus wire out of lug and resting on neutral bus - affects relays 5A-K18A&C, 5A-K19A&C and relay 5A-K22A - could cause auto scram channel A1 and A2 due to not bypassing scram discharge high water level - would not allow subchannel A1 and A2 to be reset (MR-86-300 was issued to correct).

RPS Channel

7. B1 Neutral wire loose under compression lug affects relays 5A-K3B, 5A-K3F, 5A-K4B and 5A and 5B - could cause subchannel B1 1/2 scram if reactor pressure was less than 600 psi also alarm typer alarm "Main Steam Isolation Valves Not Fully Open Scram Trip" for channel B (MR-86-301 was issued to correct).

8. A1 Neutral wire completely out of compression lug - affects relays 5A-K8A, 5A-K9A, 5A-K10A and 5A-K10E - could cause 1/2 scram subchannel A1 and annunciators "control valve fast closure scram trip," "generator load rejection - turbine stop valve scram bypass," turbine stop valve not fully open scram if first stage pressure greater than 45% (MR-86-302 was issued to correct).

9. A1 Panel C915 - Neutral bus terminal block EJ point #9 found loose wire under compression lug - affects relays 5A-K3A, 5A-K3E, 5A-K4A and 5A-K5A - could cause an 1/2 auto scram channel A1; annunciators "MSIV not fully open scram at RX pressure > 600 PSI", "Drywell high pressure scram trip", and "Reactor vessel high pressure scram trip"; and computer alarms C1509 - "Main steam line isolation valves not fully open scram trip", C1513 - "Drywell high pressure scram trip", and C1517 - "Reactor vessel high pressure scram trip"(MR-86-300 was issued to correct).

10. A1 Panel C915 - Neutral bus terminal block EJ point #10 found loose wire under compression lug - affects relays 5A-K2A, 5A-K11A, and 5A-K1A - could cause an 1/2 auto scram channel A1; annunciators "Main condenser low vacuum scram trip", "Discharge volume high water level CRD scram trip", and "Condenser low vacuum and main steam Isolation valve closure scram bypass"; and computer alarms C1505 - "Condenser low vacuum scram trip", and C1501 - "Discharge volume high water level scram trip"(MR-86-300 was issued to correct).

11. A2 Panel C915 - Neutral bus terminal block BJ point #1 found loose wire under compression log - affects relays 16A-K4C, 16A-K3C, 16A-K2C, and 16A-K1C - could cause an 1/2 Group I isolation from PCIS channel A2; annunciators "Main steam line low pressure", "Main steam line Hi flow", "Steam tunnel Hi Temp." and "Reactor vessel low low water level"; and computer alarms C1543 - "Main steam line High Flow" and C1547 - "Steam tunnel High Temp."(MR-86-300 was issued to correct).

12. A2 Panel C915 - Neutral bus terminal block BU point #2 found loose wire under compression lug - affects relays 16A-K44C, 16A-K7C, 16A-K6C, 16A-K5C, 16A-K58C, 16A-K59C, and 16A-K19C. This could cause an 1/2 Group I isolation from PCIS channel AZ; 1/2 isolation of T.I.P. withdrawal, RHR shutdown cooling, head spray and discharge to Radwaste valves; 1/2 isolation of RWCU system; a trip signal to the reactor building isolation and standby gas treatment initiation systems; annunciators "Exhaust vent high Rad", and "Vent exhaust monitor down scale"; and computer alarm C1168 - "Refueling floor vent exhaust Rad"(MR-86-300 was issued to correct).

13. A2 Panel C915 - Neutral bus terminal block BJ point #9 found loose wire under compression lug - affects relays 5A-K10C, 5A-K10G, 5A-K8C, and 5A-K9C. This could cause an 1/2 auto scram channel A2; annunciators "Turbine stop valve not fully open scram if first stage pressure > 45%", "Control valve fast closure scram trip", and "Generator load rejection & turbine stop valve scram bypass"; and computer alarms C1555 - "Turbine stop valve closure scram trip", and C1559 - "Control valve fast closure scram trip"(MR-86-300 was issued to correct).

14. A2 Panel C915 - Neutral bus terminal block BJ point #10 found loose wire under compression lug - affects relays 5A-K2C, 5A-K11C, and 5A-K1C. This could cause an 1/2 auto scram channel A2; annunciators - "Main condenser low vacuum scram trip", "Condenser low vacuum & main steam isolation valve closure scram bypass", and "Discharge volume high water level CRD scram trip"; and computer alarms C1503 = "Discharge volume high water level scram trip", and C1507 - "Condenser low vacuum scram trip"(MR 86-300 was issued to correct).

15. A2 Panel C915 - Neutral bus terminal block BJ point #12 found loose wire under compression lug - affects relays and contactors 5A-K14C, 5A-K14G, 5A-K19A, and 5A-K19C. This could cause an 1/2 auto scram channel A2; annunciator - "Reactor auto-scram channel "A"; and computer point C1539 - "Reactor auto scram channel "A"(MR-86-300 was issued to correct).

16. A3 Panel C915 - Neutral bus terminal block CS point #9 found loose wire under compression lug - affects relays and contactors 5A-K16A, 5A-K17A, 5A-K15A, 5A-K15C, 5A-K19C, and 5A-K19A. This could cause an 1/2 manual scram channel A3; annunciators - "Mode Switch shutdown scram bypass", and "Reactor manual scram channel "A"; and computer alarm C1537 - "Reactor Manual scram channel "A"(MR 86-300 was issued to correct).

17. B1 Panel C917 - Neutral bus terminal block EJ point #6 found loose wire under compression lug - affects relays 5A-K12F, 5A-K12B, 5a-K7B, 5A-K6B, 5A-K27B, and 5A-K28B. This could cause an 1/2 auto scram channel B1; annunciators - "Reactor neutron monitoring system scram trip", "Main steam line high radiation scram trip", "reactor vessel low level scram trip", and "Discharge volume high water level CRD scram trip"; and computer alarms C1530 - "Neutron monitoring system scram trip", C1526 - "Main steam line high radiation scram trip", C1522 - "Reactor vessel low water level scram trip", and C1586 - "Discharge volume high water level scram trip"(MR-86-301 was issued to correct).

18. B1 Panel C917 - Neutral bus terminal block EJ point #10 found loose wire under compression lug - affects relays 5A-K2B, 5A-K11B, and 5A-K1B. This could cause an 1/2 auto scram channel B1; annunciators - "Main condenser low vacuum scram trip", "Condenser low vacuum & main steam isolation valve closure scram bypass", and "Discharge volume high water level CRD scram trip"; and computer alarms C1506 - "Condenser low vacuum scram trip", and C1502 - "Discharge volume high water level scram trip"(MR-86-301 was issued to correct).

19. B2 Panel C917 - Neutral bus terminal block BJ point #1 found loose wire under compression lug - affects relays 16A-K6D, 16A-K5D, 16A-K58D, 16A-K59D, and 16A-K19D. This could cause an 1/2 Group I isolation from PCIS channel B2; and 1/2 isolation of T.I.P. withdrawal, RHR shutdown cooling, head spray and discharge into radwaste valves; 1/2 isolation of RWCU system; a trip signal to the reactor building isolation and standby gas treatment initiation systems; annunciators - "Exhaust vent high Rad", and "Vent exhaust monitor downscale"; and computer alarm C1168 - "Refueling floor vent exhaust Rad"(MR-86-301 was issued to correct).

20. B3 Panel C917 - Neutral bus terminal block CS point #9 found loose wire under compression lug - affects relays and contactors 5A-K16B, 5A-K17B, 5A-K15B, 5A-K15D, 5A-K19D, and 5A-K19B. This could cause an 1/2 manual Scram channel B3; annunciators - "Mode Switch shutdown scram bypass", and "Reactor Manual scram channel 'B'"; and computer alarm C1538 - "Reactor manual scram channel 'B'"(MR-86-301 was issued to correct).

21. B1 Panel C917 - Neutral bus terminal block CS point #12 found loose wire under compression lug - affects relay 5A-K13B. No effect with RPS non-coincident neutron monitoring shorting links installed(MR-86-301 was issued to correct).

Wiring & Print
<u>Discrepancies</u>

1. Print MIN 33-10 (Rev. E0) does not show 16A-K19A contact 1-2 in the mode switch bypass circuit.

   Error identified on PCAQ* to Engineering; panel wiring diagram shows circuit as designed. Schematic to be revised to reflect change.

2. Print MIP 426-16 (Rev. E5) shows EJ11 going to DZ-14 (5A-K3A) but in fact goes to DY-14 (5A-K18A).

   See Items 3 and 15 (Electrical Equivalent). PCAQ written to update print.

3. Print MIP 426-16 (Rev. E5) shows EJ9 going to DQ-14 (5A-K10A) but in fact goes to DZ-14 (5A-K3A).

   See Items 2 and 15 (Electrical Equivalent). PCAQ written to update print.

4. Print MIP 426-16 (Rev. E5) shows wire designation on DH-7 (16A-K7A) as DD-8 and it should be DD-18.

   Typo on print. PCAQ written to update print.

* PCAQ IS A "Potential Condition Adverse to Quality" report which is used to effect review and resolution of an identified condition which is potentially adverse to quality.

5. Print MIP 421-16 (Rev. E5) shows a wire on AF-14 (16A-K4D) going to BJ2 but in fact is not there. Also BJ2 shows wire going to AF-14 (16A-K4D) but is not there.

   Actual wiring is Electrical Equivalent. PCAQ written to update print or change wiring to agree with print.

6. Print MIP 421-16 (Rev. E5) shows only one wire on AK-14 from AJ-14 (16A-K7D) but in fact has a second wire on it going to DG-14 (16A-K44B).

   Similar to Item 5. This wire causes Electrical Equivalent circuit. PCAQ written to update print.

7. Print MIP 423-16 (Rev. E5) shows only one wire on DG-14 (16A-K44B) from DH-14 (16A-K7B) but has a second wire on it going to AK-14 (16A-K44D).

   Same as Item 6 but in opposite channel.
   PCAQ written to update print.

8. Print MIP 423-16 (Rev E5) shows terminal block point 7 going to DB-3. It should read DB-13.

   Typo error on print. PCAQ written to update typo.

9. Print MIP 423-16 and lamicoid marker in Panel C917 show fuse 16A-F64A going to C917 terminal block DD point 42. Print MIN 34-9 shows fuse 16A-F64B going to C917 terminal block DD point 42. MIP 426-16, MIN 33-10 and lamicoid marker in C915 show 16A-F64A going to DD point 42 in Panel 915.

   Drafting error on print MIP 423-16. F28(16A-F64A) should read 16A-F64B. PCAQ written to update print.

10. Print MIP 421-16 shows two F29's on terminal block BB - one should be F28 which is also designated as 5A-F27D. F29 is designated 5A-F28D.

    Typo error on print MIP 421-16. PCAQ written to update typo.

11. Print MIP 460-8 (Rev. E4) shows only one wire on device AN terminal 2 going to terminal block BB point 97. In fact, there is a second wire on device AN terminal 2 going to device AY terminal 9. Print shows AY-9 going to TB BB97 and TB BB97 to AY-9. This wire does not exist.

    Wiring is per elementary diagram. This is a drafting error on MIP 460-8. PCAQ written to update print or change wiring to agree with the print.

12. Print MIP 460-8 shows device BA terminal 10 going to device BO terminal 14 but this wire does not exist. Device BO terminal 14 has one wire on it going to device BP terminal B4. Device BP terminal B4 has two wires on it but print only shows one wire, the other wire goes to device BO terminal 14.

    Actual wiring is an Electrical Equivalent. PCAQ written to update print or change wiring to agree with the print.

13. Print MIP 459-9 (Rev. E4) device BF terminal 10 has two wires on it going to terminal block CC point 89. Print only shows one wire.

    This is a duplication of wiring. PCAQ written to remove wire or leave as is.

14. Print MIP 426-16 (Rev. E5) shows device DG terminal 13 going to device EG terminal 2; it should be terminal 12. Print MIP 426-16 (Rev. E5) shows device DH with two wires on terminal 3, the second wire going to device DC terminal 2 should be terminal 13.

    This is a typo error on MIP 426-16. PCAQ written to update print.

15. Print MIP 426-16 shows neutral bus EJ point 8 going to device DY terminal 14. EJ-8 goes to device DQ terminal 14.

    See Items 2 and 3. Electrical Equivalent PCAQ written to update print.

16. Print MIP 460-8 (Rev. E5) device BF terminal 10 has two wires on it going to terminal block CC point 89. Print only shows one wire.

    Similar to Item 13. PCAQ written to change wiring or leave as is.

17. Print MIN 27-16 (Rev. E5) relay tabulation does not show contacts 9-10 and 11-12 of relay 16A-K13 being used. In fact print MIN 36-7 (Rev. E6) does show them being used. Contact 9-10 is in 16A-D5159A circuit and contact 11-12 is in reset circuit.

    Contact usage was emitted from print MIN-27-16. Actual conditions were per print MIN 36-7. PCAQ written to update MIN 27-16.

18. Print MIP 426-16 (Rev. E5) shows terminal block DD point 9 going to DD-F3-1. It should show it going to DD-F5-1.

    Typo error on the print. PCAQ written to correct MIP 426-16.

ADDITIONAL INFORMATION REGARDING BECo's DISPOSITIONING OF THE WIRING DISCREPANCIES FOUND DURING THE MODE SWITCH REPLACEMENT PROJECT

TASK

Based on the miswiring found in the Reactor Mode Switch related circuitry (Reactor Manual Control System):

1. Evaluate the effect(s) on MSIV closure.

2. Determine if this discrepancy was more or less conservative than the design prints.

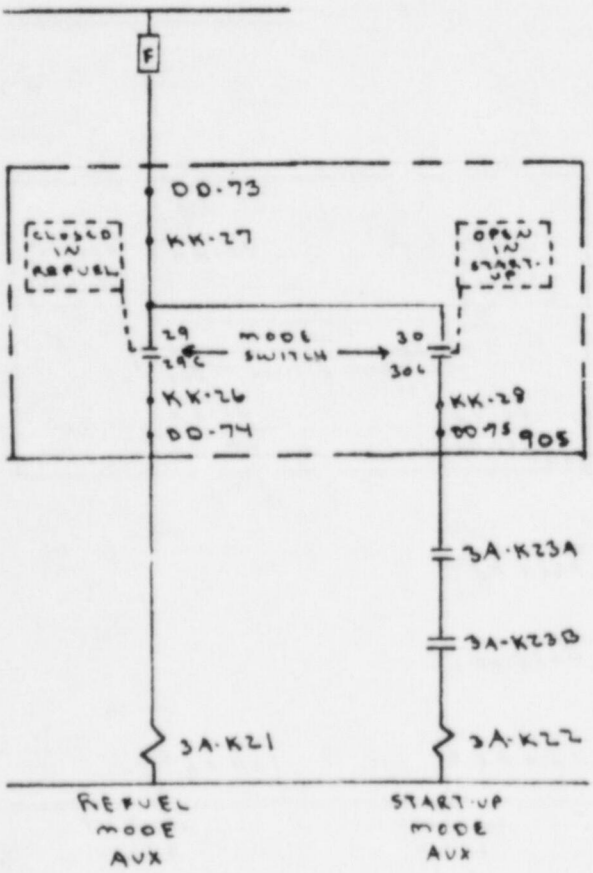
REFERENCES

MIV Series Drawings; Elementary Diagram Reactor Manual Control System
PNPS FSAR, Rev. 5.

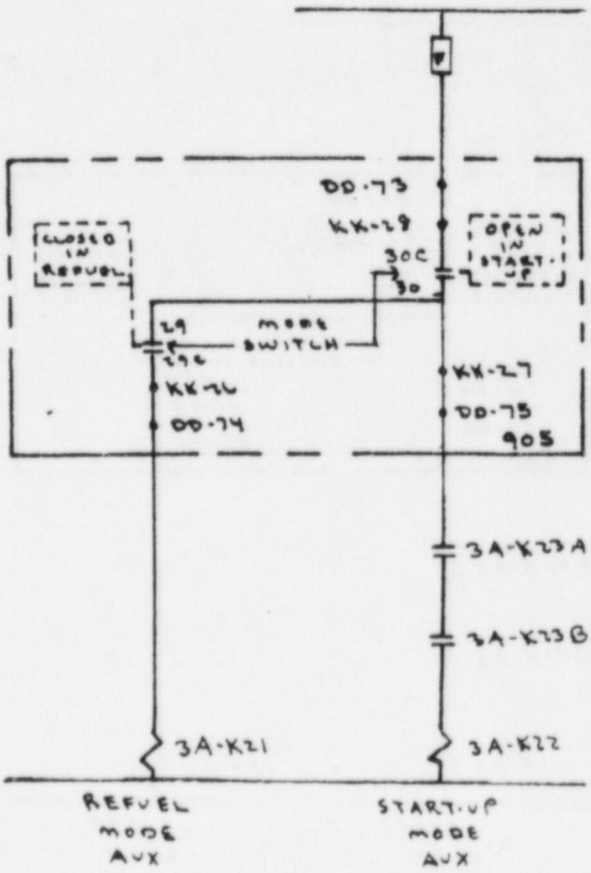
DISCUSSION OF MISWIRING

As a result of the performance of step 8 on page 11 of PNPS Procedure TP 86-64, Remove Existing SB-1 Type RMS, Replace with SB-9 Type RMS, a discrepancy was found between the Elementary Diagram MIV16-6 wiring and the "as found" wiring. The difference is shown below:

MIV16-6 (Should Be)



As found



From these drawings it can be seen that in the "should be" MIV16-6 configuration, relay 3A-K21 can be energized independent of the position of RMS contact 30-30C and is solely dependent on the position of RMS contact 29-29C, i.e., the contacts are in parallel, while in the "as found" configuration, relay 3A-K21 is dependent on the position of both RMS contacts 29-29C and 30-30C, i.e., the contacts are in series. As a result, 3A-K21 could never be energized with contact 30-30C open, which is in conflict with the design diagram. This concept is outlined in the following table where blank=open, X=closed, 0=de-energized, 1=energized:

POSSIBLE CONTACT POSITION COMBINATIONS		RESULTANT "SHOULD BE" RELAY CONDITION			RESULTANT "AS FOUND" RELAY CONDITION		
29-29C	30-30C	3A-K21	3A-K23A,B	3A-K22	3A-K21	3A-K23A,B	3A-K22
		0	0	0	0	0	0
	X	0	0	0	0	0	0
X		1	1	0	0	0	0
X	X	1	1*	1*	1	1*	1*

* Assuming contacts 3A-K33 A,B,C or D and 3A-K4 A&B are closed.

The following table shows the relationship between the RMS position and the 29-29C and 30-30C contact positions (taken from Elementary Diagram MIN14-4) where blank=open, X=closed:

RMS CONTACT	RUN	INTER	S/U & S/B	INTER	REFUEL	INTER	S/D
29-29C					X		
30-30C	X				X	X	X

As can be seen from this table, the positioning of the mode switch never requires contact 29-29C to be closed with contact 30-30C open. The only position in which contact 29-29C is required to be closed is the "Refuel" position in which contact 30-30C is coincidentally closed.

PURPOSE OF CIRCUIT

The circuit which was effected by the miswiring is part of the Refueling Interlock portion of the Reactor Manual Control System. As outlined in Section 7.6 of the FSAR, Refueling Interlocks:

Safety Design Basis

1. During fuel movements in or over the reactor core, all control rods shall be in their fully inserted positions.

2. No more than one control rod shall be withdrawn from its fully inserted position at any time when the reactor is in the refuel mode.

The refueling interlocks reinforce operational procedures that prohibit taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and operation of refueling equipment. The effected circuit is designed to allow movement of one control rod with the mode switch in the "Refuel" position and all other rods fully inserted provided all other interlocks are satisfied.

RESULTS

Using the relay tabulation on Elementary Diagram MIV8-5, all relays and contacts in the effected circuit were traced through drawings for their potential resultant action/ annunciation. The results of the analysis show the only interfaces to be with refuel and rod block interlocks. No annunciations or RPIS computer printouts could have resulted from the miswiring.

ANALYSIS OF POTENTIAL FAILURE MODES

Because of the nature of the miswiring found, the only failure mode of concern would be when the mode switch is in the "Refuel" position.

If contact 29-29C were to fail open in the refuel position while contact 30-30C were to close, the effect on the circuit would not be altered by the miswiring that was found. In either condition contact 3A-K21 would remain de-energized, and as a result, single rod withdrawal in the refuel mode could not be performed.

If contact 30-30C were to fail open in the refuel position while contact 29-29C were to close, relay 3A-K21 would again remain de-energized in the "as found" circuit while it would have energized in the "should be" circuit. As a result, single rod withdrawal in the refuel mode would not have been able to be performed as desired.

CONCLUSIONS

In response to Task 1, analysis of the Reactor Mode Switch circuitry and the Reactor Manual Control System Circuitry shows no potential effect on the MSIV closure. The RMS miswiring can in no way cause or prevent an MSIV closure.

In response to Task 2, since the miswiring could only result in the inability to withdraw a rod when desired, the resultant condition was more conservative than the design prints.

WIRING DISCREPANCIES DISCOVERED DURING REPLACEMENT OF REACTOR MODE SWITCH

MODE SWITCH EQUIPMENT	DEVIATION	EVALUATION	CORRECTIVE ACTION
Wire Marked KK24-28 (Contact 28)	Wire not landed at KK 24	Contact wires"swapped" but electrically equivalent	Wire marking changed to KK23 per MIN-18-8
Wire Marked JJ32-16 (Contact 16)	Wire not landed at JJ32	Contact wires "swapped" but electrically equivalent	Wire marking changed to JJ3 per MIN-17-8
Terminal screw at at contact 16 found loose	Potential intermittent contact condition in APRM setdown circuit. Failure of circuit would cause 1/2 scram if reactor power were above 15%	Functionally, the consequence of a loose terminal would be to de-energize the circuit and fail safe	None required, because wire harness was in the process of being removed from the old SB-1 switch
Wire marked KK23-28 (Contact 28C)	Wire not landed at KK23	Contact wires "swapped" but electrically equivalent	Wire marking changed to KK24 per MIN-18-8
Wire marked JJ3-16 (Contact 16C)	Wire not landed at JJ3	Contact wires "swapped" but electrically equivalent	Wire marking changed to JJ32 per MIN-17-8
Wire marked LL24-44 (Contact 44)	Wire not landed at LL24	Contact wires "swapped" but electrically equivalent	Wire marking changed to LL23 per MIN-17-8
Wire marked MM24-60 (Contact 60)	Wire not landed at MM60	Contact wires "swapped" but electrically equivalent	Wire marking changed to MM23 per MIN-18-8

MODE SWITCH EQUIPMENT	DEVIATION	EVALUATION	CORRECTIVE ACTION
Wire marked LL23-44 (Contact 44C)	Wire not landed at LL23	Contact wires "swapped" but electrically equivalent	Wire marking changed to LL24 per MIN-17-8
Wire marked MM23-60 (Contact 60C)	Wire not landed at MM23	Contact wires "swapped" but electrically equivalent	Wire marking changed to MM24 per MIN-18-8
Wire connected to switch contact 30C	Wire "nicked"	No functional deficiency however, potential ability to "short" to other circuits	Replaced and remarked existing wire with #12 GE "Vulkene Supreme" qualified wire. Removed damaged wire from harness.

Conclusion

Wiring discrepancies were electrical equivalents, damaged wire and loose connections. All electrical equivalents were returned to their designed condition. All loose wires were tightened and the damaged wire was replaced. These repairs were made through the replacement project with standard work practices employed and witnessed by the Quality Control Group.

The wiring errors are considered to be original plant construction; recurrence should be eliminated in the future by existing quality control practices.

NRC Question: How are channel separation of Q and non-Q circuits ensured between the monitored points and the GETARS system?

BECo Response: Channel separation between the GETARs Monitoring System and the actual monitored points are insured in three ways:

1. The points monitored with the use of dry contacts are separated from the PCIS logic by the relays themselves. The G.E. Century 100 relays provide mechanical and electrical isolation as a design function of the relay. All inter-panel wiring for C915 and C917 will be Class 1E. The GETARs System is fused at 1 amp 120V, and since the Century 100 relay contacts are rated at 12 amps continuous, there is no possibility of welding contacts closed, preventing the relay from opening.

2. Voltage divider circuits are used in the channels monitoring the actual PCIS logic channel. A 2 megohm resistor is used to provide the isolation. Any failure of the monitoring system could not affect the PCIS logic due to this high resistance value. The 2 megohm resistors were bought to Q requirements and are wired class 1E.

3. Both A and B logic systems are separated from each other by separate input multiplexors. These multiplexers are separated from the main GETARs Computer by fiber optics cable. This provides added insurance that a GETARs failure would not affect both logic channels.

In summary, the GETARs Monitoring System is separated from Q circuits by the physical constructions of the relays themselves and by the Q resistor voltage divider circuits used in the logic monitoring channels. As added insurance, the A and B PCIS logic channels are separated from each other by separate input multiplexers.

TABLE

Recommendation #	Summary
B.1	Mitigating the effect of (RHR) leakage by allowing a controlled amount of leakage to exit the system.
C.1.b	Perform once per/refueling outage leak test on 68's for pressure drop capability (to be initiated in RFO #7).
C.1.c	Evaluate the feasibility of replacing or redesigning check valves to provide positive position indication.
C.2.a	Install additional pressure gauges in system.
C.3.a	Develop system venting program.
C.3.b	Provide means for system temperature monitoring.
C.4.a	Change frequency of 68A and 68B check valve testing from once per quarter to once per refueling outage.
C.4.b	Reduce frequency of 28 & 29 isolation valve (stroke) testing.
C.5.a	Revise alarm response procedure to allow for control of repressurization and provide additional assessment and corrective action.
C.6	Trend surveillance history of 400 psig valve interlock for reliabiliy.

ATTACHMENT 2

BECo Responses to NRC Questions As Presented

at May 19, 1986 Meeting at Region 1 Office

Detailed Results of the 28B Valve Disassembly & Inspection

1. Inspection results of valve MO 1001-28B:

The disassembly of the MO 1001-28B valve showed no abnormal or significant wear on the mating surfaces of the valve disk or seat. A minor irregularity or mark was found at the 11 o'clock position on the valve seat. Also, the stem locknut to disk nub weld was cracked but remained structurally intact. The weld was repaired as a part of this refurbishment. The valve was lightly lapped and then blue-checked. The results of the blue-check were excellent.

The valve was inspected by VT-3 examination methods Data sheet 86-10-15 documents the inspection and is included in this attachment. The inspection results record evidence of erosion above the valve seating surfaces on the body of the main disk. The erosion varied in depth from approximately 3/16" to 5/16" for the full circumference of the disk. The Nuclear Engineering Department (NED) was requested to review and respond to the acceptability of this finding.

NED has responded that the root cause of the observed volumetric wear on the disk is attributed to cavitation (erosion) pitting followed by concentration cell corrosion attack. Specifically, cavitation pitting occurs initially while throttling during extended shutdown cooling service. Concentration cell corrosion attack in the resultant pits subsequently follows. The failure of the locking nut tack welds was attributed to poor weld technique during manufacture.

The wear noted is acceptable by analysis for interim operation as has been evaluated by NED to ASME III NB-3000 criteria. NED concludes that the remaining metal significantly exceeds the minimum acceptable disk thickness for postulated loads including design, emergency and faulted conditions. The wear rate will not result in an unacceptable reduction in disk thickness prior to refueling outage #7 (January 1987). This disk (for MO 1001-28B) will be removed and evaluated or restored during RFO-7 to confirm the wear rate. Disk deterioration in the form of a gross metal removal is not considered related to seat leakage. The MO 1001-28B valve will not have a motor operator torque setting in excess of 3 5/8 (as it is presently set). As a consequence of the findings on the MO 1001-28B valve, NED has concluded that similar wear may have occurred in the MO 1001-28A valve and recommends that the 1001-28A valve be disassembled and inspected at this time. This is to ascertain whether comparable disk deterioration has occurred. The valve disassembly is in progress in response to the recommendation.

2. Postwork test results of valve MO 1001-28B:

After reassembly, the MO 1001-28B valve was stroke tested, Local Leak Rate tested with air per Appendix J of 10CFR50, and hydrodynamically tested (water test). The table below summarizes the results.

LLRT Results MO 1001-28B (45 psig air)

As left RFO 6	As found April 1986	Postwork May 1986
1.9 SLM	1.5 SLM	2.0 SLM

Hydro dynamic Test Results 950 psig applied
Inboard of Check Valve 1001-68B

April 1986	May 1986	
MO 1001-29B and MO 1001-28B closed	MO 1001-29B and MO 1001-28B closed	MO 1001-28B closed
0.33 gpm	0.48 gpm	0.38 gpm

The post assembly air test results are 0.5 SLM higher than the April 1986 as-found results. The disassembly and lapping of MO 1001-28B did not result in an improvement in leakage rate. Test methods used were the same. The April results may have been lower because the valve was last closed in the warm condition prior to test. The post assembly high pressure water test was conducted after refill of the system, the system was vented and tested with both the MO 1001-29B and 28B valves closed. This leakage rate was higher than the April as found condition; however the test method uses the number of strokes that a positive displacement pump makes to maintain pressure. The April result was based on receiving an average of 4.75 strokes per minute where the May result was 7 strokes in a 1 minute period. The flow to the volume is not measured by a flow measurement device. The differences between the 0.33 gpm of April and the 0.48 gpm is accounted for by the method to count pump strokes (ie, one minute vs. ten minutes) and the isolation of valve 1001-33B, (the manual block valve that forms a test boundary) may not have been as tight in May as in April. The test conducted with only the 28B as boundary showed a lower leakage rate than with both valves as a boundary.

Based on the postwork test results, the controlled leakoff method discussed in the answer to NRC question (e), was selected as our course of action.

Instruction No. 50.81
Rev. 0
Attachment A

PILGRIM NUCLEAR POWER STATION

Data Sheet # 86-10-15
page 1 of 2

VISUAL EXAMINATION VT-3 AND VT-4

I have read and understood PNPS procedure 1.3.39 NLD (Examiner's Initials) 5-21-86 (Date)

ASME Sec XI Class 2 Category N/A System 10 - RHR

Isometric or Print ISI-I-10-4B Rev. A MR # 86-10-15

Component Description VALVE INTERNALS - MO-1001-28B RWP # _____

Location B RHR VALVE ROOM

Visual Exam: Direct X Remote ____

Observed Conditions	Yes	No	Accept	Reject	Requires Supplem Exam	Requires Further Eval
Cracks		✓	✓			
Corrosion		✓	✓			
Loose Parts		✓	✓			
Misalignments		✓	✓			
Mechanical Damage		✓	✓			
Erosion	✓	Valve	Disc			✓
Other (Describe)		✓	✓			

Sketch attached

Visual Aids Used FLASHLIGHT, MIRROR

Comments: SEVERE EROSION NOTED ON VALVE DISC - SEE ATTACHED SKETCH

ESR 86-192 Attached

Supplemental Examination Data Sheet# (if applicable) N/A

Examiner NL Desmond Level II

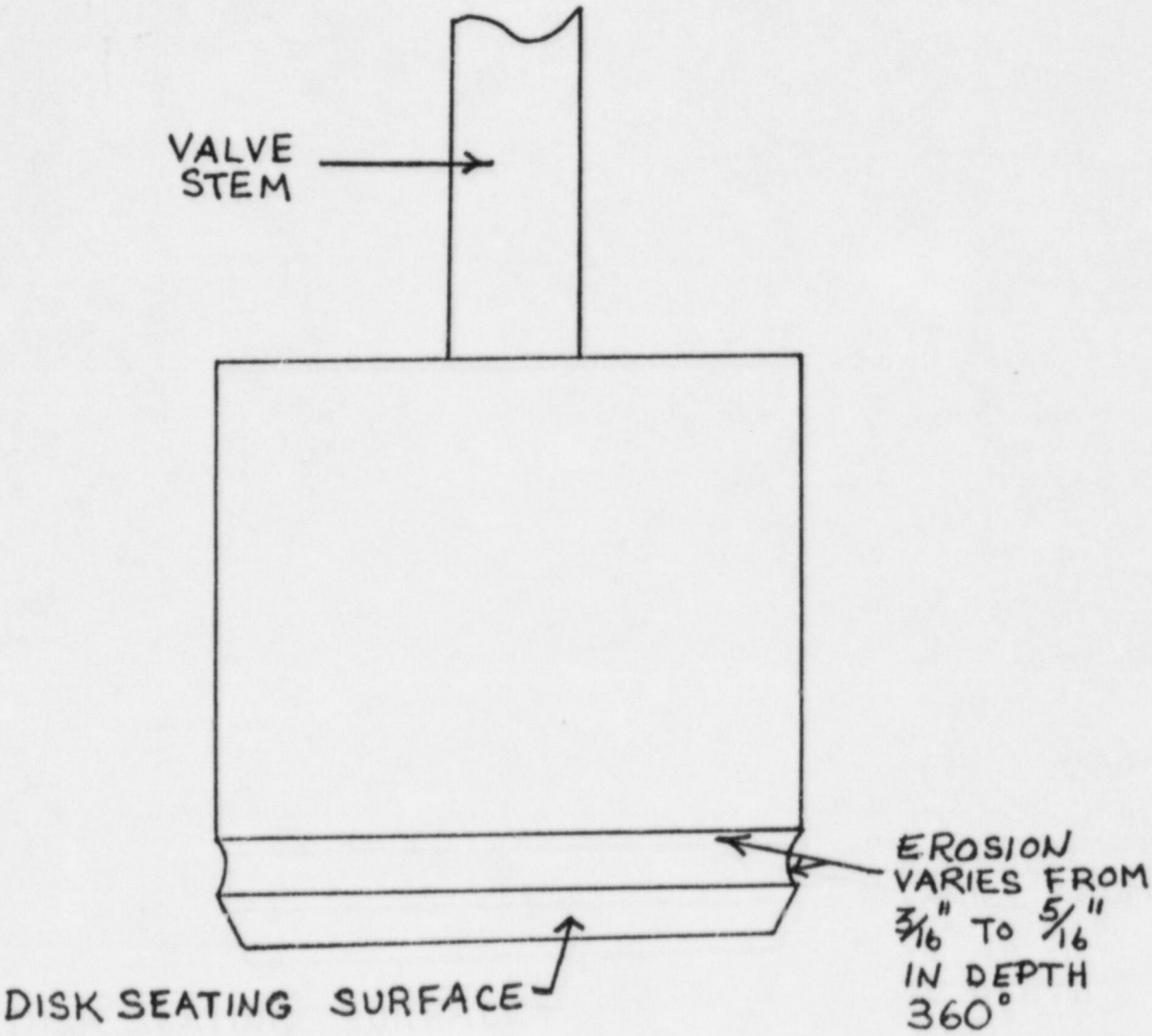
Reviewed By:

MW Green Level II Date 5-21-86
Sr. Mechanical Engineer (ISI) or Designee

N/A Date N/A
Authorized Inspector

Data Sheet # 86-10-15
Page 2 of 2

BOSTON EDISON



Description of Control Room Staffing (40 hour plan) and Details Regarding Plans to Comply with 10CFR55.32(e)

Due to the current labor action by Boston Edison's union employees, qualified exempt personnel have been assigned to fulfill the shift manning requirements established by our Technical Specifications. A three section watch bill is in effect. It has been structured such that individuals are scheduled to work not more than 60 hours per week. The limit established by our station procedures is 72 hours per week.

All personnel assigned as nuclear watch engineers, nuclear operations supervisors, and licensed reactor operators have been actively performing the functions of senior reactor operator as required by 10CFR55.31. This requirement has been met by independently standing watch in the Control Room during power operations. Additionally, each of the above individuals have previously held a reactor operator's license All of the above individuals completed simulator requalification training during the period October to December, 1985. This training includes hands-on manipulation of the reactor controls. Prior to start-up from the current outage, routine surveillance testing must be performed on the following safety related systems:

High Pressure Coolant Injection
Reactor Core Isolation Cooling
Low Pressure Coolant Injection
Emergency Diesel Generators
Core Spray
Neutron Monitoring
Reactor Manual Control System

This testing, which requires hands-on manipulation of the system's controls switches, will be performed such that each individual assigned as a licensed reactor operator receives hands-on refresher training with some of the above systems.

Personnel assigned as unlicensed nuclear plant operators (ie, auxiliary operators) are receiving on the job training in plant tours and operation of auxiliary equipment. This job-specific training will be documented and will be a function of the experience of the individual assigned as unlicensed operators.

BECo's Dispositioning of NRC Information Notice on "Set Screw" Problems Experienced Elsewhere in the Industry

During our presentation at the Region 1 office on May 19, 1986, and in subsequent discussions with your staff, you asked us to review seven (7) Inspection and Enforcement Information Notices (82-20, 83-70, 84-36, 84-53, 86-01, 86-09, and 86-29) regarding set screw issues to determine applicability to the MSIV Plant Design Change (PDC 86-28). The review has been completed and it has been determined that in all cases the issues raised in the information notices are either not applicable or properly addressed in PDC 86-28. The results of our review have been tabulated in the following Table.

TABLE

DOCUMENT	TOPIC	DISCUSSION	DISPOSITION
IEN-82-20	Check valve problem	Aloyco check valves mounted in vertical orientation sustained internal damage to disk stud serving as stop.	IEN does not include a set screw problem. Damage occurred to stud directly.
		Pacific valve also sustained similar damage and a material defect limited to manufacturers product line.	Note BECo does not use Aloyco or Pacific check valves in Pressure Boundary Applications. Velan, Rockwell and A/D valves used, either have integral disk stand off or are tilting disk.
IEN-83-70	Vibration induced valve failures	Threaded fasteners (Yoke to Bonnett Stem Clamp) had backed out in non-MSIV applications.	Applications were not MSIVs. Similar applications on PNPS utilize locking tabs to preclude this (note pilot/poppet set screw is staked vs. lock tab).
IEN-84-36	Loosening of locking nut on Limitorque Valve Operator	Set screw on the locking nut for the worm gear backed out.	Set screws were not staked. BECo went through and staked selected set screws on all Q MOV's. This IEN points out the need for staking as already considered in PDC 86-28.
IEN-84-53	Information concerning the use of Lock acorn nut fouled valve other anaerobic adhesive/ sealant	Excessive Lock-tite not wiped of a scram pilot solenoid valve	Lock-tite not used in MSIV due to this concern plus degradation induced by radiation. Use of Lock-tite is controlled to nuclear grades dedicated for use by Engineering and utilized under controlled process whereby administrative controls preclude excessive application (Ref RA&P-84-346)

TABLE

DOCUMENT	TOPIC	DISCUSIION	DISPOSITION
IEN-86-01	Failure of Main Feedwater check valves causes loss of FW system integrity and water hammer damage.	Generic failure of Pacific check valve disc nut retainers essentially identical to IEN-82-20 above	As per IEN-82-20, BECo doesn't use Pacific check valves in Pressure Boundary applications. Valves used have different integral disc stand off or tilting disc. Failure has no bearing on MSIV task. No apparent use or failure of set screws.
IEN-86-09	Failure of check valves subject to low flow conditions	Leakage past a normally closed motor operated valve in Auxiliary Feedwater turbine steam supply line fatigue cycled stop check valve disc & disc guide studs	No set screws involved, no record of similar occurrence at PNPS, no similar applications, except RCIC/HPCI exhaust lines which have not experienced this problem. Not applicable to MSIV PDC.
IEN-86-29	Effects of changing operator switch settings	Change to TS bypass limit switch settings for IEB-85-03 at Songs-3 resulted in unintended changes in closed position indications. Valves did not stroke full closed.	No set screws involved, PNPS does not use TS bypass, not applicable to MSIVs in particular or PNPS in general.

ATTACHMENT 3

Revised PNPS Procedure 2.3.2.1

Alarm Procedure Panel 903 Left