BOSTON EDISON

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James M. Lydon
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August 29, 1986
BECo Ltr. #86-128

Mr. J. Strosnider, Chief
Reactor Project Section 1B
U.S. Nuclear Regulatory Commission
631 Park Avenue - Region 1
King of Prussia, PA 19406

Docket No. 50-293
License No. DPR-35

Subject: Supplemental Response to NRC Confirmatory Action Letter #86-10

References:
(a) NRC CAL 86-10 Dated April 12, 1986
(b) BECo Response to CAL 86-10 Dated May 15, 1986
(c) NRC "Request for Additional Information" Letter Dated May 16, 1986
(d) BECo Response Dated June 16, 1986

Dear Mr. Strosnider:

This letter provides additional information requested by the NRC Region 1 through a list of comments and questions which we received on July 21, 1986. This list was a result of the NRC review of the BECo Supplemental Response Letter to CAL 86-10 dated June 16, 1986. A telephone conversation was held on July 31, 1986, between Mr. Strosnider, Dr. McBride and various Station management personnel to finalize the answers. Answers to the specific questions are included as Attachment 1 to this letter.

We trust the contents of this submittal combined with information provided in our preliminary response dated May 15, 1986, and the second response dated June 16, 1986, will provide information adequate to address the requirements of Confirmatory Action Letter 86-10.

SE35 1/1

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

Respectfully submitted,

Kenneth P Roberts for

James M. Lydon

JQ/ko

Attachments:
1. Supplemental Response to CAL 86-10
2. Pressure Switch History Data
3. Calculation M-269
4. Procedure 3.M.3-8 (Control Room)
5. Procedure 3.M.3-8 (Cable Spreading Room)

ATTACHMENT 1

SUPPLEMENTAL RESPONSE TO CAL 86-10

Boston Edison Company
Pilgrim Nuclear Power Station

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NRC Question 1

Has BECo evaluated the merits of periodic checks of the RHR system pressure and temperature after (1) startups, (2) any RHR system high pressure alarm or (3) after the leakoff system is initially placed in service? Also, has BECo considered periodic checks of RHR system pressure in the absence of high pressure alarms?

Response

Although the alarm response procedure is keyed to the RHR discharge pressure high pressure alarm, action such as obtaining local pressure readings takes place after a second annunciation. The alarm response Procedure 2.3.2.1 will be clarified prior to startup for action on any Panel C903-Left annunciator B-7 RHR high pressure alarm. This will cause local pressure and temperature data to be captured upon alarm annunciation. Additionally, a surveillance test for periodic data collection of RHR system pressure and temperature will be prepared prior to startup. Conceptually this surveillance test will record RHR system local pressures and temperatures from the temperature surveillance monitoring strips. A frequency of weekly surveillances during power operation is being considered, but frequency of the surveillance will be procedurally adjusted based upon the trend of the results. The resident inspectors will be appraised of the content of the proposed surveillance procedure.

NRC Question 2

The first paragraph of BECo's response to NRC question "e" describing the proposed leakoff method appears to differ from the method in Procedure TP86-85 and should be clarified. The paragraph infers that the bypass valve will be opened during a measurement step, closed, and reopened to establish the leakage path.

Response

BECo will follow the method described in TP86-85 for establishing the controlled leakoff. The response to NRC question "e" was presented as a concept of the method. The temporary procedures presents the actual method to be used.

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NRC Question 3

Will a reactor shutdown just be initiated or will the plant be shutdown (pending an engineering evaluation) if the 1.0 gpm limit is exceeded? Also, has BECo considered evaluating leakage rates above 1.0 gpm now; rather than waiting to conduct the evaluation until the leakage limit is exceeded?

Response

Yes, normal reactor shutdown will be initiated when the 1 gpm is reached with both valves shut. However we will be performing additional testing which may result in a change to the 1 gpm criteria.

Leakage greater than 1 gpm implies that the single closed MOV is at or beyond its local leak rate limit. Leakage rates greater than 1 gpm have been considered. The engineering evaluation has set a conservative limit of 1 gpm based upon measured Appendix J leakage, the Appendix J leakage limit, a conservative safety factor and high pressure water leakage. Steps required for additional engineering evaluation will involve retesting the boundary for Appendix J criteria and for high pressure water leakage which will be performed during this shut down.

NRC Question 4

The BECo response and the associated safety evaluation state that leakage in the RHR system will not be allowed to exceed 1.0 gpm. This is not strictly true since the measured parameter will be leakage through the bypass valve at 150 psig. Changes in RHR check valve leakage (such as after a pump has been operated) could make the leakoff measurements inaccurate and misleading. Has this been considered? Has BECo considered tracking the leakage rate into the torus over extended periods of time as a method of verifying stable RHR system in leakage?

The safety evaluation also states that all RHR pump flow will go into the RHR system. However, some flow could be diverted through the leakoff path if the "D" pump is idle (LPCI only requires 3 pumps operate). BECo should consider modifying the safety evaluation to address these two concerns.

Response

Leakage through the RHR pump discharge check valves was considered in BECo's evaluation of RHR system operation and isolation valve performance. The bypass valve addressed in SE 1959 R1 was part of a three-pronged approach which included the additional temperature and pressure monitoring equipment. The temperature monitoring equipment in particular will provide notice of large leakage rates past the isolation valves independent of RHR check valve condition. Tests performed on the RHR system using a hydro test pump on April 17, 1986, showed that the RHR discharge piping would become pressurized with

Response (Cont.)

only .33 gpm being added to the system. This test confirmed that the RHR pump discharge check valves are "tight". The valves have performed satisfactorily since plant startup and are expected to function satisfactorily in the future. Valve degradation, when it occurs, is expected to be gradual and will be identified through a change in the flow measured during engineering tests at the bypass line and possibly an increase in piping temperature upstream of the RHR injection valves if there is a concurrent increase in isolation valve leakage. BECo believes these measures along with our scheduled Appendix J LLRT's are sufficient to monitor the performance of the RHR system containment isolation valves.

The intent of the bypass valve modification and subsequent Safety Evaluation (S.E. 1959 Rev. 1) was to address the effects of an incremental increase in leak rate caused by the addition of the bypass valve. Large leak rate increases were to be monitored by other means.

BECo has considered trending torus water level but considers this impractical given the accuracy of the present and available instrumentation and the large volume in the torus.

The Safety Evaluation #1959 Rev. 1 did not address the reduction in RHR flow in the rare event during 3 pump LPCI injection concurrent with the condition when the fourth, idle pump, had the bypass line around the discharge check valve. This was omitted as the incremental flow rate if below the level of accuracy for pump curves or flow measuring devices (ie: less than 3 gpm out of a total flow of 14,400 gpm (ie: 3 pumps at 4800 gpm each).

BECo will revise SE #1959 Rev. 1 to clarify its intent and assumptions and forward it to you under separate cover by September 17, 1986.

NRC Question 5

The data sheet to procedure TP 86-85 requires that an RHR pump suction block valve (MO-1001-7D) be closed during the initiation of a controlled leakage path but not reopened (although the procedure does require this). Why has the latter step been left off the data sheet?

Response

The step to reopen the RHR suction valve MO 1001-7D was inadvertently deleted from signoff sheet A2. The procedure has been corrected to include this step.

NRC Question 6

The acceptance criteria for tests of the RHR injection check valves are not included in the response, as requested (NRC question "f"). The criteria should be determined and submitted to Region I.

Response

A 5 gpm leak rate criteria was selected for the following reasons:

a. The injection check valves are designed to limit loss of primary fluid while the motor operated isolation valves close. As such, rapid operation is more important than duplicating the leak tight seating capability of the containment isolation valves. The 5 gpm leak criteria is well within the makeup capacity of the plant under normal and emergency conditions.

b. 5 gpm leakage will not challenge the integrity of the low pressure RHR piping. The low pressure RHR pipe is protected by two PSVs with a combined relieving capacity of over 100 gpm.

c. 5 gpm leakage is well within the makeup capability of any of the CSCS Systems acting individually or in concert in the event of an accident event (i.e., HPCI makeup is nominally rated at over 4000 gpm).

d. 5 gpm has reportedly been used and accepted by the NRC as an acceptable leak rate limit for swing check valves at several other operating nuclear power plants.

NRC Question 7

The RHR pressure gauge calibration frequency was stated to be "once per refueling outage" in the response to question "h". This frequency is not defined in the technical specifications. How is it defined and what is the justification for the frequency?

Response

The once per refuel outage frequency is better stated as once per operating cycle which is defined by T.S. as "the interval between the end of one refueling outage and the end of the next subsequent refueling outage". The newly installed gauges in the RHR system are classified as supplemental instrumentation for which the owner (BECo) may determine the calibration frequency. BECo uses the once per cycle frequency for calibration on many instruments that support Technical Specification systems. Similar gauges are also located throughout the Station on various other systems and have not exhibited any major calibration drifting with once per cycle testing.

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NRC Question 8

The calibration histories of the RHR high system pressure alarm switches (PS1001-74A and B) should be submitted to Region 1. How is the proposed calibration frequency, "once per cycle", defined and justified?

Response

Calibration histories of PS1001-74A & B are included as Attachment 2. The once per cycle frequency is explained in the response to Item 7..

Additional information is needed to clarify a statement in the prior response to NRC question "i" concerning the setpoint for PS 1001-74B. This statement indicated that the actuation setpoint was found set low at 360 psig versus a setpoint of 392 psig. It should be clarified that 360 psig was obtained from the RHR pressurization tests and that the 360 psig was the point at which the RHR high pressure alarm was received in the control room. This reading was obtained from a test gauge mounted on the RHR system at approximate elevation +3'6". The PS 1001-74B is mounted on rack 2206 at an approximate elevation of -12'6". The pressure source used to obtain the 360 psig was a positive displacement pump which created pressure surges in the system. The 360 psig was not indicative of the actual pressure that actuated PS 1001-74B.

NRC Question 9

Will the 1001-28 and 29 valve on the "A" RHR loop be maintained normally closed? Procedure TP86-84 indicates that the 28A valve will be left closed, but the licensee has previously indicated that the 28A will be open and the 29A closed.

Response

The MO 1001-29A and B valves will be normally open and the MO 1001-28A and B valves will be normally closed. Procedure changes to establish this lineup were in progress but are being delayed because further safety evaluations are needed for the procedure change. If the valve lineup decision changes based on this pending safety evaluation, the resident inspector will be informed.

NRC Question 10

What is the accuracy and reliability of the temperature measuring markers? What specific change in temperature will require that additional measurements be taken with a portable measuring device? What is the accuracy and reliability of the portable temperature measuring device?

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Response

The specific change in pipewall temperature will be determined after ambient operating conditions are established. This may require the frequency of temperature checks mentioned in response to question 1 to be increased to daily or once per shift to establish baseline conditions.

The specific change in RHR pipe wall temperature that will require additional measurements is expected to be an increase in 20°F from the normal ambient temperature. This ambient temperature will be established as part of the surveillance test referenced in response to Question No. 1.

The temperature measuring markers, reversable and nonreversable have an accuracy of ± 1°F. Portable temperature monitoring can be accomplished with a Rochester Instruments System "Supercal" in the thermocouple mode, which has an accuracy of ± 1°F. Each of these instruments based on performance history is considered to have good reliability.

NRC Question 11

Will the licensee verify seating of the RHR check valves after operating? If not, why not?

Response

The Nuclear Engineering Department (NED) is currently investigating methods for confirming RHR check valve operability. After researching the implications of operationally testing these valves and assessing such concerns as opening leakage paths between the reactor and low pressure systems, the investigation is now concentrating on check valve disk position monitoring. The valves were originally equipped with position monitoring instrumentation, however, operational problems resulted and the instrumentation was inoperable and finally removed. NED is presently investigating new position monitoring equipment.

If the investigation shows that the new valve position monitoring equipment represents an improvement over the original design and will operate reliably in a power plant environment, NED will recommend the installation of the equipment on the RHR check valves. This investigation is expected to be complete by November 1, 1986.

An alternative previously considered to verify check valve seating by pressurizing the low pressure system with the reactor at approximately 250 psia has been reviewed by NED and not recommended. This recommendation is contained in a NED response and states that there is a concern that such a pressure test (i.e., opening the in-series valves MO 1001-29B, 28B, 34B and 36B and creating a direct pressure path to the torus) could over pressurize several portions of the 150 psig design piping in the RHR System. The event of a single operator error or equipment malfunction could cause the loss of a significant portion of the reactor vessel coolant inventory.

The check valves ability to prevent reverse flow will however be tested once per refueling outage as previously committed.

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NRC Question 12

Why will it take 9 months to submit a Technical Specification change to reduce compensatory surveillance testing in LCO's: considering that the issue was identified mid-1985 in connection with the on-line EQ modifications and also noted in the 1985 SALP report? Has the licensee considered contacting other facilities of a similar age to see if they have information that could be used to speed the evaluation and submittal process?

Response

As identified to the NRC in the NRC/BECo Management meeting held on June 12, 1986, one of the Management improvement items is the need to develop a Work Management Control Program which will identify and prioritize the entire NUORG workload. Until such time, it is not possible to accurately forecast completion dates that ultimately become commitments. The nine month time frame was chosen as a "Target" date only.

As for contacting other facilities, we contacted General Electric Company for background information on the Standard Technical Specification allowances. We had hope of being able to utilize whatever generic evaluations were available and apply them to our specific situation. This avenue did not prove fruitful, as no generic studies were available. We have, however developed a preliminary justification for satisfying the safety significance consideration which will be presented soon to the appropriate levels of management. As new scheduling information becomes available, we will update the NRC Project Manager for Pilgrim Station in accordance with the ongoing Licensing Action Report iterations.

NRC Question 13

What is the justification for limiting RHR pipe temperature to no less than 15 degrees of saturation temperature? Is this temperature margin adequate, considering that pipe wall temperature (rather than interior water temperature) is the measured parameter?

Response

The purpose of applying a 15°F limit to the RHR pipe wall temperature is to guard against steam void formation. This margin is deemed sufficient based upon a bounding calculation for the uninsulated pipe based on heat conduction through the pipe wall. Considering the air is stagnant in the area of the uninsulated RHR piping (6 inches from the floor) the maximum temperature drop through the steel wall was 0.15°F. When the effects of air velocity (400 ft/min) are considered, the additional temperature drop is approximately 0.5°F. Since the total temperature drop is less than 1°F the 15°F limit on saturation temperature is considered conservative.

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NRC Question 14

Calculation M-269 should be submitted to Region I for review.

Response

Please refer to Attachment 3 of this letter.

NRC Question 15

The following comments concern the draft procedure TP86-81 which will control the test for spurious group I primary containment isolations during the next reactor startup:

15a. Will reactor level instrument vibration be monitored? If not, why not?

Response

The reactor level instrument Racks C2205 and C2206 will be monitored for Vibration during the performance of TP86-81. A special Temporary Procedure TP86-82 has been written to control this activity (see attached copy). An outside vendor has been contracted by NED to perform the monitoring and analysis of the data taken. A change has been made to the startup procedure to include the vibration monitoring activity.

15b. Step VI.A indicates that the reactor mode switch will be placed in run for 24 hours. Why?

Response

The Draft TP86-81 has been redefined and approved, by the ORC for implementation. The 24 hours indicated in Step VI. a. of the draft has been lowered to 10-15 hours in the final version. This time period is required to stabilize the plant and should allow for any time dependent variables to be reproduced.

15c. Was a functional test conducted of the PCIS logic after GETARS modification was installed? What procedure was used for the functional test?

Response

A special Temporary Procedure (TP86-83) was written to satisfy the post installation testing of the GETARS equipment as well as the functional testing of the PCIS logic. This Procedure was completed at the completion of GETARS installation.

15d. Can the spurious isolation test be conducted at a power below the stated 30%?

Response

The isolation transient took place at a power level quite a bit lower than 30% but, the purpose of this startup test is to as completely as possible, reproduce that transient. Thirty percent is a comfortable power level for the plant to operate at during the stabilizing period. In addition most of the major operating equipment will be inservice at the point. This will help us to more accurately reproduce the shutdown sequence that led to the spurious isolation transient.

NRC Question 16

Does BECo plan to conduct a sampling review of other systems, given the large number of drawing/loose wire problems discovered during work on the reactor mode switch?

Response

An investigation was conducted on other Control Room and Cable spreading room panels, no compression connections similar to the type found loose in C915, 917 and 916 were identified. Therefore there is no further effort planned on that matter. The drawing discrepancies have all been dispositioned as electrical equivalents, duplication of wires, or typographical errors, with two exceptions. The first exception was a relay contact not shown on PCIS logic elementary diagram. This is considered an isolated case and measures have been taken to correct the drawing. The second exception was the miswiring of a reactor manual control circuit at the mode switch itself. Investigation showed the logic to be more restrictive to plant operations in its miswired form than it would be wired correctly. Regular surveillance testing would have identified a miswiring that would have resulted in a less restrictive mode of operations. This was the only case of a circuit performing other than was indicated by the elementary diagrams, therefore, no walk down of other elementary diagrams is planned. The miswiring was corrected by returning the circuit to its design condition.

The panel wiring diagram discrepancies dispositioned as electrical equivalents dealt with the neutral connections of relay coils. There is a chance such deviations exist in other panels but those discrepancies would, most likely, also be limited to neutral connections or typographical errors. Active relay contacts and the relay coil supply leads were portrayed accurately on the wiring diagrams. Drawing revisions/upgrades are a continuous process. For example, the NED Plant Design Change (PDC) closeout program consists of, but is not limited to, updating electrical wire diagrams to reflect changes made due to PDC's. Beyond that program, the Technical Group is included in the E-203 walk down program. This program involves the walkdown of the AC and DC power distribution systems. Print discrepancies found during this program are being corrected through the E-203 procedure developed by the Technical Group (reference Attachments 4 & 5).

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NRC Question 17

Why is local venting of the RHR system needed in addition to the keep fill system?

Response

Local venting in RHR is being used for assurance that there is no formation of air pockets in local high points. The venting Procedure TP86-84 schedules the venting to be performed once per week for four weeks (after maintenance is completed). This schedule is intended to detect the presence of air in local pipe runs and to be used as an input to regular venting practice. The results and locations will be incorporated into normal RHR System surveillance tests.

NRC Question 18

Is the RHR system always filled and vented after the 1001-34 and -36 valves are opened to depressurize the system?

Response

Since the keepfill system is continuously in service, the RHR system is always filled after the use of the 36 and 34 valves, but the system was not vented after it was used to depressurize the system. The results of the venting procedure discussed in Question 17 are intended to assess the presence of air and the severity of a potential problem in this area.

NRC Question 19

Has BECo considered the personnel safety aspects of the leakoff measurement process? At what location will system pressure be measured and what will be the expected water pressure at the measuring point? Will the measurement equipment withstand this water pressure?

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Response

The leakoff measurement process has not yet been tested in the field. Prior to actual use, a walkthrough and practice use of the procedure is planned to validate the method. This is to be performed prior to unit startup so that the pressure source would be controlled. This walkthrough has not yet been conducted because of the redirection of resources to Local Leak Rate Testing and the RHR pump wear ring inspection. Industrial safety concerns and appropriateness of equipment use would be additionally verified as part of this walkthrough. However, procedural steps to "slightly opening valve MO 1001-100D" and the fact that the vent bottle is vented (vent is equipped with a particulate filter for radiological concerns) reduces the possibility of over pressurizing the equipment. The system pressure of 150 psig is monitored in the RHR quadrant at PI 1001-80B. This gauge is at approximately the same elevation as the test connection (within 3' elev.). The components used for measurement equipment are Tygon tubing or rubber hose and a "Nalgene" reagent container (vent bottle) are judged to be sufficient to withstand 150 psig water pressure even if not vented.

NRC Question 20

Has BECo considered testing the leakage of injection check valves in ECCS systems other than LPCI?

Response

In June, 1985, the NRC designated Generic Issue No. 105 "Interfacing Systems LOCA at BWR's" as a high priority issue. Shortly afterward in September, 1985, the NRC's Office for Analysis and Evaluation of Operational Data (AEOD) issued a case study report that based on operational events indicated the likelihood of interfacing LOCA was higher than previously assessed and that this represents a trend with serious safety implications. As a result of the AEOD Report, a BWR Owners Group Committee of which BECo is a member was formed to assess the significance of this issue. This committee has worked closely with INPO and with the NRC Task Manager on this generic issue.

Among the possible fixes to address the root causes for these events is testing of isolation check valves and removal of the air operator on testable check valves. PNPS has removed the air operators from all ECCS injection check valves. The air operator was the major contributor to the events described in the AEOD report. At the present time, the significance and need for further corrective actions are being studied by the NRC, INPO and the BWR Owners Group. A decision on testing of these check valves will be based on the findings of the BWR Owners Group Committee.

ATTACHMENT 2

SUPPLEMENTAL RESPONSE TO CAL 86-10

Boston Edison Company
Pilgrim Nuclear Power Station

Docket No. 50-293
License No. DPR-35

Pressure Switch History Data Conclusion

PS 1001-74A

1974	399	Set Point
1986	392	Set Point

12 Year Span Lost 7 PSI

10-PS1001-74A
I&C INSTRUMENT NO.

SERVICE	MANUFACTURER
Discharge Header Pres.	Mercoid

MODEL NO.	SERIAL NO.	LOCATION
DA23-804	3D749811	El. 17'6" Rack 2259

P & ID	REV.	G. E. OR BECHTEL DATA SH. NO.	REV.
M241	4F	225A5750 DS53	0

MANUF. REF. DWG. NO.	INST. MANUAL NO.
	72

INSTRUMENT RANGE
25-600 psi

VERIFICATION CHECKS		INITIALS	REFERENCES
INSTALLATION	X	JMF	M1P-363-3 Rev. 1
PIPING	X	JMF	M1P-365-2 Rev. 0
WIRING	X	JMF	M1P-365-2 Rev. 0
VALVES	X	JMF	M1P-365-2 Rev. 0

CONTROL AND ALARM SETTINGS
Contact #1 closes @ 400 psi (inc) pres. (reset diff 25 psi)

SPECIAL PRECAUTIONS
Alarm discharge headers high press and shutdown suction headers.

CALIBRATION PROCEDURE NO.	VENDOR	G. E. SPEC. NO.	REV.
3C	Mercoid		

TEST INSTRUMENT USED
150C T.G. comparator

AS FOUND ← INITIAL CALIBRATION → AS LEFT

Contact close @ 400 psi inc.
Contact opens @ 366 psi dec.

INITIAL – POST CALIBRATION DATA

SET POINT	TRIP POINTS HIGH	TRIP POINTS LOW	ALARM POINTS HIGH	ALARM POINTS LOW	PROP. BAND	RESET	RATE

REMARKS

DATE	BY	APPROVED
12/7/71	JMF/GGL	WHD

SERVICE: Discharge Header Pres.

MANUFACTURER: Mercoid

10-PS1001-74A
I & C INSTRUMENT NO.

MODEL NO.	SERIAL NO.	LOCATION
DA23-804	3D749811	El. 17'6" Rack 2259

P & ID	REV.	G. E. OR BECHTEL DATA SH. NO.	REV.
M241	4F	225A5750 DS53	0

MANUF. REF. DWG. NO.	INST. MANUAL NO.
	72

VERIFICATION CHECKS		INITIALS	REFERENCES
INSTALLATION	X	JMF	MIP-363-3 Rev. 1
PIPING	X	JMF	MIP-365-2 Rev. 0
WIRING	X	JMF	MIP-365-2 Rev. 0
VALVES	X	JMF	MIP-365-2 Rev. 0

INSTRUMENT RANGE: 25-600 psi

CONTROL AND ALARM SETTINGS: Contact #1 closes @ 400 psi (inc) pres. (reset diff 25 psi)

SPECIAL PRECAUTIONS: Alarm discharge headers high press and shutdown suction headers.

CALIBRATION PROCEDURE NO.	VENDOR	G. E. SPEC. NO.	REV.
3C	Mercoid		

TEST INSTRUMENT USED: 150C T.G. comparator

AS FOUND ← INITIAL CALIBRATION → AS LEFT

Contact close @ 400 psi inc.
Contact opens @ 366 psi dec.

INITIAL – POST CALIBRATION DATA

SET POINT	TRIP POINTS HIGH	TRIP POINTS LOW	ALARM POINTS HIGH	ALARM POINTS LOW	PROP. BAND	RESET	RATE

REMARKS

DATE	BY
12/7/71	JMF/GCL

APPROVED: WHD

10-PS1001-74A
I & C INSTRUMENT NO.

SERVICE	MANUFACTURER
Discharge Header Pres.	Mercoid

MODEL NO.	SERIAL NO.	LOCATION
DA23-804	3D749811	El. 17'6" Rack 2259

P&ID NO.	REV.	G.E. OR BECHTEL DATA SHEET NO.	REV.	SURVEILLANCE FREQUENCY
M241	4F	225A5750 DS53	0	

MANUF. REF. DWG. NO.	INSTRUMENT MANUAL NO.	ROUTINE CALIBRATION FREQUENCY
	72	

INSTRUMENT RANGE

25-600 psi

CONTROL AND ALARM SETTINGS

Contact #1 closes @ 400 psi (inc) pres. (reset diff 25 psi)

SPECIAL PRECAUTIONS

Alarm discharge headers high press. and shutdown suction headers.

CALIBRATION PROCEDURE NO.	VENDOR	G.E. SPEC. NO.	REV.
3C	Mercoid		

SUGGESTED TEST INSTRUMENTS

150C, T.G. comparator

RECALIBRATION AND MALFUNCTION RECORD

DATE	BY/APPR'D	REASON FOR RECAL.	AS FOUND ALARM PT. HI	AS FOUND ALARM PT. LO	AS FOUND TRIP PT. HI	AS FOUND TRIP PT. LO	AS LEFT ALARM PT. HI	AS LEFT ALARM PT. LO	AS LEFT TRIP PT. HI	AS LEFT TRIP PT. LO	CONTROLLERS P.B.	CONTROLLERS RESET	CONTROLLERS RATE	CONTROLLERS SET POINT
9/12/74	DFC/PFW	ROUT. X			375				399					
	WHD	MAL.			340				360					
TEST INSTRUMENT USED: 77J		REMARKS												
2/25 86	RPL/BJL	ROUT.	Trip		392	PSI			No Adj.					
	DOL	MAL. X	Reset		345	PSI			Nec.					
TEST INSTRUMENT USED: I-6??H		REMARKS												
		ROUT.												
		MAL.												
TEST INSTRUMENT USED		REMARKS												
		ROUT.												
		MAL.												
TEST INSTRUMENT USED		REMARKS												
		ROUT.												
		MAL.												
TEST INSTRUMENT USED		REMARKS												
		ROUT.												
		MAL.												
TEST INSTRUMENT USED		REMARKS												
		ROUT.												
		MAL.												
TEST INSTRUMENT USED		REMARKS												

RED	BLUE
1	27
2	28
3	29
4	30
5	31
6	32
7	33
8	34
9	35
10	36
11	37
12	38
13	39
14	40
15	41
16	42
17	43
18	44
19	45
20	46
21	47
22	48
23	49
24	50
25	51
26	52

10-PSI 001 -74A
I&C INSTRUMENT NO.

SERVICE
Discharge Header Pres.

CALIBRATION PROCEDURE NO.
3C

MODEL NO.
DA23-804

SERIAL NO.
3D749811

INSTRUMENT MANUAL NO.
72

INSTRUMENT RANGE
25-600 psi

P&ID NO.
M241 REV. 4F

CONTROL AND ALARM SETTINGS
Contact #1 closes @ 400 psi (inc) pres. (reset diff 25 psi)

G.E. OR BECHTEL DATA SHEET NO.
225A5750 DS53 REV. 0

SPECIAL PRECAUTIONS
Alarm discharge headers high press. and shutdown suction headers.

DATE	BY	CALIBRATION ANA CAL.	CALIBRATION DIG. CAL.	P. M.	MAINTENANCE PERFORMED PARTS REPLACED	MAINTENANCE PERFORMED DESCRIBE MALFUNCTION
9/12/74	PFW/ DFC		X			Checked setpoint and adjusted
2/25/86	RPL BJL		X		NONE	Checked Setpoint Per MR86-140

ATTACHMENT 2 (Cont.)

SUPPLEMENTAL RESPONSE TO CAL 86-10

Pressure Switch History Data Conclusion

PS 1001-74B

1974	398	Set Point
1986	394	Set Point

12 Year Span Lost 4 PSI

10 PS 1001 74 B

I & C INSTRUMENT NO.

SERVICE	MANUFACTURER
Discharge Header Press	Mercoid

MODEL NO.	SERIAL NO.	LOCATION
DA 23 804	3D749812	E.L. 17' 6" Rack 2262

P & ID	G. E. OR BECHTEL DATA SH. NO.	REV.
241 4F REV.	GE DS 53	0

MANUF. REF. DWG. NO.	INST. MANUAL NO.
	72

INSTRUMENT RANGE

25 -600 psi

VERIFICATION CHECKS		INITIALS	REFERENCES	Q NO.
INSTALLATION	X	jmf	MIP 368 3 Rev 1	
PIPING	X	jmf	MIP 370 2 Rev0	
WIRING	X	jmf	MIP 370 2 Rev 0	26
VALVES	X	jmf	MIP 370 2 Rev 0	1345

CONTROL AND ALARM SETTINGS

Contact #1 close at 400 psig (inc) pres Reset diff. 25 psi

SPECIAL PRECAUTIONS

Alarm discharge headers high press and shut down suction headers

CALIBRATION PROCEDURE NO.	VENDOR	G. E. SPEC. NO. / REV.
3C	Mercoid	

TEST INSTRUMENT USED

TG #77D

AS FOUND ← INITIAL CALIBRATION → AS LEFT

AS FOUND	AS LEFT
Contact #1 closes at 265 psi inc	Contact closes (#1) at 400 psi inc.
Contact #1 closes at 241 psy dec	Contact #1 opens at 376 psi dec.

INITIAL – POST CALIBRATION DATA

SET POINT	TRIP POINTS HIGH	TRIP POINTS LOW	ALARM POINTS HIGH	ALARM POINTS LOW	PROP. BAND	RESET	RATE

REMARKS	DATE	BY
	12/17/78	JMF
		APPROVED
		WHT

10 PS 1001 74 B
I & C INSTRUMENT NO.

SERVICE	MANUFACTURER
Discharge Header Press	Mercoid

MODEL NO	SERIAL NO.	LOCATION
DA 23 804	3D749812	E.L. 17' 6" Rack 2262

P & ID	G. E. OR BECHTEL DATA SH. NO.
241 4F REV.	GE DS 53 REV. 0

MANUF. REF. DWG. NO.	INST. MANUAL NO.
	72

INSTRUMENT RANGE

25 -600 psi

VERIFICATION CHECKS		INITIALS	REFERENCES	Q NO.
INSTALLATION	X	jmf	MIP 368 3 Rev 1	
PIPING	X	jmf	MIP 370 2 Rev0	
WIRING	X	jmf	MIP 370 2 Rev 0	26
VALVES	X	jmf	MIP 370 2 Rev 0	1345

CONTROL AND ALARM SETTINGS

Contact #1 close at 400 psig (inc) pres Reset diff. 25 psi

SPECIAL PRECAUTIONS

Alarm discharge headers high press and shut down suction headers

CALIBRATION PROCEDURE NO.	VENDOR	G. E. SPEC. NO.
3C	Mercoid	REV.

TEST INSTRUMENT USED

TG #77D

AS FOUND ← INITIAL CALIBRATION → AS LEFT

AS FOUND	AS LEFT
Contact #1 closes at 265 psi inc	Contact closes (#1) at 400 psi inc.
Contact #1 closes at 241 psy dec	Contact #1 opens at 376 psi dec.

INITIAL – POST CALIBRATION DATA

SET POINT	TRIP POINTS HIGH	TRIP POINTS LOW	ALARM POINTS HIGH	ALARM POINTS LOW	PROP. BAND	RESET	RATE

REMARKS

DATE	BY	APPROVED
12/17/78	JMF	WHT

I & C INSTRUMENT NO. [illegible] 74 D

SERVICE: [illegible]	MANUFACTURER: [illegible]
MODEL NO.: [illegible]	SERIAL NO.: [illegible]
LOCATION: [illegible] 17' [illegible]	
P&ID NO.: [illegible] REV.	G.E. OR BECHTEL DATA SHEET NO.: [illegible] REV.
SURVEILLANCE FREQUENCY:	
MANUF. REF. DWG. NO.:	INSTRUMENT MANUAL NO.: 72
ROUTINE CALIBRATION FREQUENCY:	

INSTRUMENT RANGE
[illegible] - [illegible] psi

CONTROL AND ALARM SETTINGS
Contact #1 closes at 400 psig (inc) pres. Reset diff 25 psi

SPECIAL PRECAUTIONS
Alarm discharge headers high press and shut down suction headers

CALIBRATION PROCEDURE NO.	VENDOR	G.E. SPEC. NO.
[illegible]	Mercoid	REV.

SUGGESTED TEST INSTRUMENTS
[illegible] 77 D

RECALIBRATION AND MALFUNCTION RECORD

DATE	BY/APPR'D	REASON FOR RECAL.	AS FOUND ALARM PT. HI	AS FOUND ALARM PT. LO	AS FOUND TRIP PT. HI	AS FOUND TRIP PT. LO	AS LEFT ALARM PT. HI	AS LEFT ALARM PT. LO	AS LEFT TRIP PT. HI	AS LEFT TRIP PT. LO	CONTROLLERS P.B.	RESET	RATE	SET POINT
9/12/74	PFW/DFC	ROUT. X			385				398	inc				
	WHD	MAL.			362				370	reset				
TEST INSTRUMENT USED: 77J		REMARKS:												
2/25/86	RPL/JRP	ROUT.	Trip		394	PSI		No Adj.						
	DO'L	MAL. x	Reset		350	PSI		Nec.						
TEST INSTRUMENT USED: I-600H		REMARKS:												
		ROUT.												
		MAL.												
TEST INSTRUMENT USED		REMARKS												
		ROUT.												
		MAL.												
TEST INSTRUMENT USED		REMARKS												
		ROUT.												
		MAL.												
TEST INSTRUMENT USED		REMARKS												
		ROUT.												
		MAL.												
TEST INSTRUMENT USED		REMARKS												
		ROUT.												
		MAL.												
TEST INSTRUMENT USED		REMARKS												

RED	BLUE
1	27
2	28
3	29
4	30
5	31
6	32
7	33
8	34
9	35
10	36
11	37
12	38
13	39
14	40
15	41
16	42
17	43
18	44
19	45
20	46
21	47
22	48
23	49
24	50
25	51
26	52

SERVICE

CALIBRATION PROCEDURE NO.

MODEL NO.

SERIAL NO.

INSTRUMENT MANUAL NO.

I&C INSTRUMENT NO.

INSTRUMENT RANGE

P&ID NO.

REV.

CONTROL AND ALARM SETTINGS

[illegible] (abs) pres.

G. E. OR BECHTEL DATA SHEET NO.

[illegible]

REV.

SPECIAL PRECAUTIONS

[illegible] and shut down suction headers

		CALIBRATION			MAINTENANCE PERFORMED	
DATE	BY	ANA. CAL.	DIG. CAL.	P. M.	PARTS REPLACED	DESCRIBE MALFUNCTION
9/12/74	PFW/ DFC		X			Checked setpoint and adjusted
2/25/86	RPL JRP		X		None	Checked Setpoint Per MR86-141

ATTACHMENT 3

R5 Boston Edison Company

CALCULATION COVER SHEET

CAPITAL AUTHORIZATION NO. N/A

PILGRIM UNIT 1

Sheet 1 of 4

CALC NO. M-269 REV. 0 File No. ______

Subject WATER LEAKAGE PAST MO 1001-28B OR MO 1001-29B

Discipline Group Leader WS CLANCY

Approval /s/ W A Clancy Date 5/9/86

SR ☒
NSR ☐
Preliminary Calc. ☐
Finalization due date ______
Final Calc. ☒

Independent Verifier D Heard Statement Attached ☒

Page(s)		Date		Date	Agree
ALL	By PAUL J. CAFARELLA 1st Paul Cafarella	5/8/86	Chk'd D HEARD /s/ D Heard	5/8/86	Yes

PURPOSE: TO DETERMINE AN ACCEPTANCE CRITERIA FOR WATER LEAKAGE PAST THE MO 1001-28 B OR MO 1001-29B AND INTO THE RHR SYSTEM

Minor revision(s) made on page(s) _____ of this calculation. See next revision.

Replaces calc # ______ Voided by calc # ______ or attached memo ☐

☐ PRELIMINARY

REV ___ DATE ________

☒ FINAL

REV 0 DATE 5/8/86

CALCULATION SHEET

BOSTON Edison COMPANY

CAPITAL AUTHORIZATION NO. N/A

PREPARED BY [illegible] DATE 5/8/86

CHECKED BY [illegible] DATE [illegible]

APPVD BY [illegible] DATE 5/9/86

SUBJECT:

SR ☒

NSR ☐

GIVEN:

	PRESS. INBD CK VLV. 68B	SYSTEM PRESS. OUTBD 62B	FLOWRATE INTO SYSTEM	LLRT 29B	LLRT 28B	COMMENTS
RESULTS OF WATER TESTING	975 psig	365 psig	0.33 gpm	1.0 SLM (SEE COMMENTS*)	1.5 SLM	* AS FOUND LLRTs
EXPECTED TO SEE DURING NORMAL OPS.	1030 psig	150-200 psig	0.69 gpm (SEE BELOW)	7.89 SLM** (SEE COMMENTS)		** ACCEPTANCE CRITERIA FOR LLRTs

LR_1 (Leak Rate$_1$) $\propto (\Delta P_1)^2$ $LR_1 = .33$ gpm $\Delta P_1 = 610$ psid

LR_2 (Leak Rate$_2$) $\propto (\Delta P_2)^2$ $LR_2 = ?$ $\Delta P_2 = 880$ psid

$$LR_2 = \frac{LR_1 (\Delta P_2)^2}{(\Delta P_1)^2} = \frac{.33\,(880)^2}{(610)^2} = .69 \text{ gpm}$$ (equivalent leakage at 880 psid)

ASSUMPTIONS: 1.) Steady State Conditions are reached (Leakage rate is constant.

2) Temp. effects are negligible

3.) Air and water leakage are proportional

4) Leakage found during the water test was limited by the most limiting component (29B) at 1.0 SLM.

5.) The air to water correlation specific to this line & Valve configuration is 1.0 SLM at 45 psig = .69 gpm at 880 psid

M-269

☐ PRELIMINARY

REV ___ DATE ________

☒ FINAL

REV 0 DATE 5/8/86

CALCULATION SHEET

BOSTON Edison COMPANY

CAPITAL AUTHORIZATION NO. N/A

PREPARED BY [signature] DATE 5/8/86

CHECKED BY [signature] DATE 5/9/86

APPVD BY [signature] DATE 5/9/86

SHEET 3 OF 4

SUBJECT:

SR ☒

NSR ☐

CALCULATIONS:

The equivalent water leakage to the Appendix J acceptance criteria is:

as found LLRT (29B) = 1.0 SLM at 45 psig
Appendix J acceptance criteria = 7.89 SLM at 45 psig

Equivalent as found water leakage = 0.69 gpm (at 880 psid)
Equivalent as found water acceptance criteria = Y_1

$$\frac{1.0}{7.89} = \frac{0.69}{Y_1}$$

$$Y_1 = 5.44 \text{ gpm}$$

Using the conservative case where the 28B valve was limiting instead of the 29B = Y_2

the as found LLRT (28B) = 1.5 SLM at 45 psig

$$\frac{1.5}{7.89} = \frac{0.69}{Y_2}$$

$$Y_2 = 3.63 \text{ gpm}$$

CONCLUSION: ACCEPTABLE WATER LEAKAGE PAST THE MO1001-28B OR MO1001-29B IS LESS THAN 3.63 gpm.

NED - 1001

B.E.Co. FORM 3930

Attachment I - Independent Verification Statement M-269

Calculation #M-___, Revision # 0___ has been independently verified by the following method(s), as noted below:

Page 4 of 4

Design Review ☑ including verification that:

* ✓ Design inputs were correctly selected and included in the calculation.

* ✓ Assumptions are adequately described and are reasonable.

* ✓ Input or assumptions requiring confirmation are identified, and if any exist, the calculation has been identified as "Preliminary" and a "Finalization Due Date" has been specified.

* ✓ Design requirements from applicable codes, standards and regulatory documents are identified and reflected in the design.

* ✓ Applicable construction and operating experience was considered in the design.

* ✓ The calculation number has been properly obtained and entered.

* ✓ An appropriate design method or computer code was used.

o ✓ A mathematical check has been performed.

o ✓ The output is reasonable compared to the input.

Alternate Calculation ☐ including verification of asterisked items noted above. The alternate calculation (_____ pages) is attached.

Qualification Testing ☐ for design feature ______________________ including verification of asterisked items noted above and the following:

o The test was performed in accordance with written test procedures.

o Most adverse design conditions were used in the test.

o Scaling laws were established and verified and error analyses were performed, if applicable.

o Test acceptance criteria were clearly related to the design calculation.

o Test results (documented in ______________________) were reviewed by the calculation Preparer or other cognizant engineer.

Independent Verifier Comments: ______________________

____

____

See NED Procedure 3.05, Sec. 7.1.1 /s/ [signature] 5/8/86
Independent Verifier Date

Preparer concurrence with findings and comment resolution /s/ [signature] 5/8/86
Preparer or other Date

R5

BOSTON EDISON

CONTROL ROOM

NUCLEAR OPERATIONS DEPARTMENT

PILGRIM NUCLEAR POWER STATION

Procedure No. 3.M.3-8

INSPECTION/TROUBLE SHOOTING - ELECTRICAL CIRCUITS

List of Effective Pages

3.M.3-8-1
3.M.3-8-2
3.M.3-8-3

Attachments

3.M.3-8A-1
3.M.3-8A-2

Approved Charles J. Mathis
ORC Chairman

Date November 16, 1984

ATTACHMENT 4

I. PURPOSE

To provide guidelines for personnel inspecting and troubleshooting electrical circuits and to document that investigation and its results.

II. DISCUSSION

Inspecting/Troubleshooting may be performed due to any malfunction of an electrical circuit switch results in abnormal operation. The malfunction may take the form of electrical grounds, equipment failure, short circuits, open circuits, sequencing problems, or loose wires as examples. Troubleshooting electrical circuits may require equipment or circuit isolation. The momentary opening of circuits under controlled circumstances and with the Watch Engineer's understanding of those components affected and his permission to perform does not constitute making the systems inoperable.

III. REFERENCES

A. PNPS Single Line Diagram E-13: Single Line Relay & Meter Diagram 125V & 250V DC Systems

B. PNPS Single Line Diagram E-14: Single Line Diagram 120V Instrument AC, Vital & Reactor Protection AC Systems & $\pm$24 VDC Power System

C. Special Order 81-02 dated 10/8/81

IV. PREREQUISITES

A. Plant Conditions

1. Any condition permissible within Technical Specification Requirements.

B. System Conditions

1. Any condition permissible within Technical Specification Requirements.

C. System Isolation

1. Proper tagging procedure is to be observed.

D. Radiation Work Permit (RWP)

1. Check with Health Physics on conditions and requirements; initiate an RWP if required.

E. Signed Maintenance Request if required.

F. Any of the following in order to accomplish the return to normal operation.

1. Bechtel Prints
2. Vendor Prints
3. Vendor Manuals
4. BECo Prints

V. APPARATUS

1. Any test equipment necessary to perform the Inspection/Troubleshooting required.

VI. PRECAUTIONS

A. The approach and areas of investigation must be thoroughly discussed with the Watch Engineer and documented on the checklist.

B. Maintenance to be performed as a result of investigation shall be documented on a signed Maintenance Request.

C. Special note must be taken in those areas where redundant equipment cross interlocks are involved. There is a potential especially in AC distribution Panels Y1, Y2, Y3, and Y4 and DC distribution Panels D4, D5, and D6 breakers for enabling redundant safety related systems. Isolation of these loads other than momentary shall be on a component only basis, unless reviewed and approved by ORC.

VII. PROCEDURE

A. The procedure will comprise those steps as discussed and agreed to by the Watch Engineer and documented on Attachment A.

VIII. ACCEPTANCE CRITERIA

A. Inspection/Troubleshooting - Electrical Circuits shall be considered complete when Attachment A is signed off with no unexplained discrepancies and with second verification of Systems Returned to Normal.

IX. ATTACHMENTS

A. Inspection/Troubleshooting - Electrical Circuits Check List

ATTACHMENT A

INSPECTION/TROUBLESHOOTING - ELECTRICAL CIRCUITS

DATE: 5-17-86

CHECK LIST

MR# N/A

1. State Problems

VISUALLY INSPECT ALL NEUTRAL AND GROUND BUSSES FOR MISAPPLICATION OF COMPRESSION LUGS, IN ALL PANELS IN CONTROL ROOM ~~AND CABLE SPREADING ROOM~~

2. State Trouble Shooting Technique

VISUALLY INSPECT, NOTHING TOUCHED.

Potential Effects

NONE

Maintenance Technician H. S. SHIDNER J. NEVILLE

Watch Engineer KT/

3. Cause N/A

4. Corrective Action N/A

5. Associated Documentation (if applicable) MR# N/A

Jumper Log Entry # N/A

6. As Left Condition AS FOUND

a. Systems Returned to Normal N/A (Signed) N/A (Verified)

b. Other N/A

Discrepancies: — NONE

914 - OK
915 - OK
C-10 - OK
C08 - OK
C-05 - OK
C03 - OK
C01 - OK
C02 - OK
C905 - OK
C903 - OK
C927 - OK
C925 - OK
C07 - OK
C06 - OK
C-171 - CHANNEL "B" - OK
C-170 - CHANNEL "A" - OK
C-04 - OK

C-921 - OK
C-911 - OK
C-910 - OK
C-902 - OK
C-913 - OK
C-937 - OK
C-936 - OK
C-174 - OK
C-175 - OK

Date Performed 5-17-86 Plant Status OUTAGE

The Acceptance Criteria specified in Section VIII was met except as noted in discrepancies.

Test Equipment Used (Serial No./Calibration DATE DUE N/A

Maintenance Technician H. Sherman Skidmore Date 5-17-86

Maintenance Engineer J. Vander Date 5-17-86

Watch Engineer K Taylor /[illegible] Date 5/17/86

BOSTON EDISON

CABLE SPREADING ROOM

NUCLEAR OPERATIONS DEPARTMENT

PILGRIM NUCLEAR POWER STATION

Procedure No. 3.M.3-8

INSPECTION/TROUBLE SHOOTING - ELECTRICAL CIRCUITS

List of Effective Pages

3.M.3-8-1
3.M.3-8-2
3.M.3-8-3

Attachments

3.M.3-8A-1
3.M.3-8A-2

Approved Charles J. Mathis
ORC Chairman

Date November 16, 1984

ATTACHMENT 5

I. PURPOSE

To provide guidelines for personnel inspecting and troubleshooting electrical circuits and to document that investigation and its results.

II. DISCUSSION

Inspecting/Troubleshooting may be performed due to any malfunction of an electrical circuit switch results in abnormal operation. The malfunction may take the form of electrical grounds, equipment failure, short circuits, open circuits, sequencing problems, or loose wires as examples. Troubleshooting electrical circuits may require equipment or circuit isolation. The momentary opening of circuits under controlled circumstances and with the Watch Engineer's understanding of those components affected and his permission to perform does not constitute making the systems inoperable.

III. REFERENCES

A. PNPS Single Line Diagram E-13: Single Line Relay & Meter Diagram 125V & 250V DC Systems

B. PNPS Single Line Diagram E-14: Single Line Diagram 120V Instrument AC, Vital & Reactor Protection AC Systems & ±24 VDC Power System

C. Special Order 81-02 dated 10/8/81

IV. PREREQUISITES

A. Plant Conditions

1. Any condition permissible within Technical Specification Requirements.

B. System Conditions

1. Any condition permissible within Technical Specification Requirements.

C. System Isolation

1. Proper tagging procedure is to be observed.

D. Radiation Work Permit (RWP)

1. Check with Health Physics on conditions and requirements; initiate an RWP if required.

E. Signed Maintenance Request if required.

F. Any of the following in order to accomplish the return to normal operation.

1. Bechtel Prints
2. Vendor Prints
3. Vendor Manuals
4. BECo Prints

V. APPARATUS

1. Any test equipment necessary to perform the Inspection/Troubleshooting required.

VI. PRECAUTIONS

A. The approach and areas of investigation must be thoroughly discussed with the Watch Engineer and documented on the checklist.

B. Maintenance to be performed as a result of investigation shall be documented on a signed Maintenance Request.

C. Special note must be taken in those areas where redundant equipment cross interlocks are involved. There is a potential especially in AC distribution Panels Y1, Y2, Y3, and Y4 and DC distribution Panels D4, D5, and D6 breakers for enabling redundant safety related systems. Isolation of these loads other than momentary shall be on a component only basis, unless reviewed and approved by ORC.

VII. PROCEDURE

A. The procedure will comprise those steps as discussed and agreed to by the Watch Engineer and documented on Attachment A.

VIII. ACCEPTANCE CRITERIA

A. Inspection/Troubleshooting - Electrical Circuits shall be considered complete when Attachment A is signed off with no unexplained discrepancies and with second verification of Systems Returned to Normal.

IX. ATTACHMENTS

A. Inspection/Troubleshooting - Electrical Circuits Check List

ATTACHMENT A

INSPECTION/TROUBLESHOOTING - ELECTRICAL CIRCUITS — DATE: 5-17-86

CHECK LIST — MR# N/A

1. State Problems

VISUALLY INSPECT ALL NEUTRAL AND GROUND BUSSES FOR MISAPPLICATION OF COMPRESSION LUGS, IN ALL PANELS IN CABLE SPREADING ROOM. EL. 23 ~~T~~ TURBINE

2. State Trouble Shooting Technique — Potential Effects

State Trouble Shooting Technique	Potential Effects
VISUALLY INSPECT, NOTHING TOUCHED.	NONE

Maintenance Technician H. SHERMAN SHIDNER Watch Engineer K Taylor / [illegible]

3. Cause N/A

4. Corrective Action N/A

5. Associated Documentation (if applicable) MR# N/A

Jumper Log Entry # N/A

6. As Left Condition AS FOUND

a. Systems Returned to Normal N/A (Signed) N/A (Verified)

b. Other N/A

Discrepancies: NONE

C-11 - OK

C-16 - OK

C-47 - OK

C-935 - OK

C-932 - OK

C-930 - OK

C-933 - OK

C-939 - OK

C-941 - OK

C-942 - OK

C-1233B - OK

Date Performed 5-17-86 Plant Status OUTAGE

The Acceptance Criteria specified in Section VIII was met except as noted in discrepancies.

Test Equipment Used (Serial No./Calibration DATE DUE N/A

Maintenance Technician H. Sherman Sledrer Date 5-17-86

Maintenance Engineer Jack Vender Date 5-17-86

Watch Engineer R. N. Dazler Date 5-17-86