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FLORIDA POWER & LIGHT COMPANY

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JAN 3 1 1986
L-86-29

Dr. J. Nelson Grace
Regional Administrator, Region II
101 Marietta Street, N.W. Suite 2900
Atlanta, Georgia 30323

Dear Dr. Grace:

Re: Turkey Point Units 3 & 4
Docket Nos. 50-250, ~~50-251~~
Inspection Report 85-40

Florida Power & Light Company (FPL) hereby responds to NRC Inspection Report Nos. 50-250/85-40 and 50-251/85-40. As requested by NRC Region II's letter dated January 2, 1986 forwarding the subject Inspection Report, the attachment to this response includes FPL's plans for corrective action for each unresolved and inspector followup item identified in the Report, and describes the actions taken or planned to improve the effectiveness of FPL's management control systems for each such item.

While we recognize the need for prompt actions to improve the effectiveness of management control systems for the items addressed in the subject Inspection Report, we also request that NRC's deliberations on the attached response take account of the particular circumstances under which the inspection results were obtained. The subject Inspection Report describes the results of a follow-up inspection conducted within less than two months of NRC's intensive Safety Systems Functional Inspection (SSFI), which was itself unprecedented in its depth and approach. Without diminishing the importance of the matters raised in the subject Report, we believe that the results of any new inspection program, along with an immediate expansion of that program through regional followup inspection, should be interpreted with caution.

Moreover, the followup inspection was conducted before FPL had completed and submitted its response to the SSFI. Many of the responses to the SSFI were still in progress and there had been insufficient time for many of the corrective actions taken or planned in response to the SSFI to become effective and be reflected in the followup inspection results. We request in particular that our December 6, 1985 response to the SSFI and the description of corrective actions planned or taken therein, be given consideration in your deliberations. In addition, we ask that you give consideration to the fact that our ongoing corrective action activities have been adversely impacted by the demands imposed by the 2-week regional followup inspection.

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PEOPLE . . . SERVING PEOPLE

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We believe that the corrective actions described in our response to the SSFI and in the attached response have been timely and will be effective. We have implemented appropriate procedure revisions and demonstrated the adequacy of systems and components by analysis and/or test. Moreover, a significant number of unresolved items were already the subject of planned updating actions and at the time of the inspection were still in progress toward satisfactory resolution.

As we indicated in our December 6, 1985 response to the SSFI, the results of SSFI indicated a need to augment our ongoing, long-term Performance Enhancement Program (PEP) in regard to Maintenance and Configuration Control. Additionally, the AFW System Availability/Reliability Study and Safety System Reviews were viewed to be appropriate and have been commenced. These actions have been integrated into our overall management plan. We acknowledge that the burden is on FPL to assure effective implementation of that plan and we fully recognize and accept that responsibility. At the same time, we ask that you give consideration to the positive results of our longer term programs that are now beginning to emerge.

As stated in our response to the SSFI, FP&L committed (letter L-85-372, dated September 30, 1985) to apply and implement appropriate technical specification requirements to address the availability and surveillance testing of the two non-safety grade motor-drive standby feedwater pumps at Turkey Point. The Technical Specification amendment request has been submitted to the NRC (L-86-43 dated January 30, 1986). Since the inspection team concluded that there were no administrative controls or Technical Specification requirements in place to assure the availability of this system on demand, the team found it inappropriate to give credit for this system during analysis of the inspection findings. As stated in FP&L's letter L-85-372, however, these standby "pumps have been routinely run in accordance with plant procedures". In addition, backup power supply is obtainable from five non-safety grade diesel generators rated at nominal 2500 kw each. These diesels can supply power directly to nuclear side loads via internal (site) cable runs independent of the switchyard. Based upon the foregoing factors, FP&L submits that it would be appropriate for the capabilities of the standby feedwater system to be taken into consideration in the NRC's final analysis of its inspection findings.

A formal description of the detailed scope of work to be completed by the AFW Availability/Reliability study, discussed in our response to the SSFI, has been issued. A contractor has been selected; the AFW Availability-Reliability study is estimated to be completed within twelve (12) weeks. This study which includes reliability modeling and an evaluation of component failure contribution to reliability, will provide a real time assessment capability for AFW system readiness. A summary of this program is being presented to the Region II staff on January 31, 1986.

FPL has undertaken a two-phase Safety System review to assure that the concerns expressed in the inspection report do not apply to the operations and functions of other important safety systems, or that any appropriate corrective actions are promptly taken. The Phase I (Initial Assessment) review has been completed by the Safety Engineering Group. No system problems that might impede the functional performance of the systems selected for review were identified. With the Phase I results as input for prioritization, FPL will now undertake the more formal and detailed, in-depth Phase II (Comprehensive Assessment) review of the selected systems. This review will encompass and

ensure pertinent design bases are clearly specified, providing additional assurance that the systems will function as designed. Any necessary corrective actions will be tracked to implementation. Phase II has been scoped and scheduled and is estimated for completion within two years of the commencement of work activities.

Additionally, in order to more efficiently control and implement design related issues at Turkey Point, a Site Engineering Manager has been recently appointed to be responsible for all site design activities. As an indication of the level of importance assigned to this position, he will report directly to the Site Vice President.

In closing, we reemphasize our commitment to improving performance and assuring that the corrective actions described in the attached response are effective. We are confident that our corrective actions, when viewed in the context of our overall PEP, will continue to achieve improved performance.

Very truly yours,

C. O. Woody

C. O. Woody
Group Vice President
Nuclear Energy

COW/dh

Attachment

cc: V. Stello, NRC Executive Director for Operations (Acting)
H. R. Denton, Director NRR
J. M. Taylor, Director, NRC Office of Inspection and Enforcement
✓ S. E. Elrod, Section Chief, Region II
H. L. Thompson, Jr., Division Director, PWR Licensing Division A, NRR
S. A. Varga, Director, Project Directorate No. 3, PWR Licensing Division A, NRR
L. S. Rubenstein, Director, Project Directorate No. 2, PWR Licensing Division A, NRR
D. G. McDonald, Senior Project Manager, NRR
H. F. Reis, Esquire

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ATTACHMENT

Re: Turkey Point Units 3 & 4
Docket Nos. 50-250, 50-251
Inspection Report 85-40

FPL Response to Unresolved Items and Inspection Followup Items

URI 85-40-01:

Licensee Administrative Procedure ADM 701, Section 5.8.1.8 requires that the root cause of equipment failure be identified by the journeyman on the completed PWO. The licensee's failure to implement this procedural requirement is considered an unresolved item.

Response:

Journeymen, Supervisors and GEMS personnel have been directed to ensure that the "Analysis of the Cause or Reason" section of PWO's is completed. As stated in Inspection Report 85-40 (Page 4) "the inspector reviewed 15 safety related PWO's completed since the SSF inspection and noted that all 15 had the root cause section completed as required". Additionally, the Nuclear Job Planning System (NJPS), the development of which has been underway since the inception of PEP, requires an identical section to be completed on the CRT screen. These actions will enhance root cause identification and appropriate corrective action implementation. NJPS, when fully automated, will automatically datalog system equipment history. Field engineers are being added to all three maintenance disciplines to enhance corrective actions after root cause identifications. The inspection report also credited the Turkey Point Emergency Response Team (ERT) for its capability to identify root cause of failures which should "subsequently reduce the repetitive failures that have occurred at the Turkey Point Plant".

Finally, the inspection report recognizes that "the automated PWO program which will provide trending information, the automated PM program, and the performance based maintenance training program, should also contribute to a reduction in repetitive equipment failures on a long-term basis". (IR85-40, Page 4)

URI 85-40-02:

The post maintenance testing requirements of Administrative Procedure 0190.19 for instrument and controls and electrical areas were informally completed without specific direction or documentation on the associated PWO's. This situation appears to be another example of failure to implement or provide adequate procedures to control safety related activities.

Response:

The Procedure Update Program (PUP) has been writing post-maintenance testing requirements into all PEP maintenance procedures. AP 0190.28, "Post Maintenance Test Control" guidance, has been revised to cover I&C and electrical maintenance activities, as well as mechanical maintenance. PEP is being enhanced to incorporate formal post-maintenance testing criteria into the PWO and Maintenance Procedures. The SORP Post Maintenance Guidance Document (Rev. A), issued in September 1985, will be evaluated for incorporation and consolidation of formal post-maintenance testing criteria. This task is identified as Project 9, Task 6 and is scheduled to be completed by March 31, 1986.

URI 85-40-03:

The licensee failed to provide and implement adequate procedures to ensure that independent verification was performed and documented on the return to service of instrumentation vital to the operation of two safety related systems, namely, auxiliary feedwater and backup nitrogen as required by NUREG-0737 Item I.C.6, confirmed by a NRC order dated July 10, 1981. This appears to represent another example of a failure to provide and/or implement adequate procedures to control safety related activities.

Response:

Procedure 0-ADM-031 (Independent Verification) dated July 12, 1985, Step 5.3.1 requires independent verification of the removal and return to service of components controlled by equipment clearance orders. Procedure 0-ADM-107 dated October 25, 1985 (Writer's Guide for Maintenance Procedures), Step 5.8.4.C gives directions for independent verification for preparing maintenance procedure. As part of the PEP maintenance activities, maintenance and surveillance procedures are being revised to incorporate instrument alignment independent verification.

URI 85-40-04:

Licensee Procedure 0208.11, Off Normal Operating Procedure (ONOP) Annuciator Panel List - Panel I Station Service, contained erroneous operator action in the event of a low pressure alarm on the nitrogen backup system. The errors existed due to a failure to revise the procedure following modification to the system per plant change/modification (PC/M) 80-117.

Response:

Procedure (ONOP) 0208.11 was changed to clarify immediate operator actions in the event of an alarm. It was approved by the Plant Nuclear Safety Committee (PNSC) on September 25, 1985. Further, EOP's have been revised to require operators to shift FCV's to manual from automatic control within 3 minutes of AFW actuation.

Power Plant Engineering and Nuclear Energy Departments met in December 1985 to discuss implementation of the Standard Engineering Package. As part of these discussions, an agreement was reached with respect to inter-departmental coordination of PC/Ms. Prior to initiation of the design activity, Power Plant Engineering and Nuclear Energy will schedule an operability review meeting. This review will ensure that Engineering is provided with the necessary system operating information. This will facilitate Engineering providing more detailed guidance in the PC/M package concerning operating and maintenance procedures. As part of the total design effort, Engineering will review the plant procedures revised by Nuclear Energy with respect to integration with the design. This inter-department coordination will ensure that the pertinent plant procedures are identified, reviewed and modified to reflect the new system configuration. Guidance in this area is in the process of being formalized in Engineering and Nuclear Energy procedures.

URI 85-40-05:

The FP&L Q-List (Quality Instruction JPE-QI-2.3A) did not designate the nitrogen backup system electrical and instrumentation components as safety-related. Consequently, the requisite controls over maintenance activities were not applied to the component. PC/M 80-117 correctly designated the activities performed under the modification as safety-related; however, the document verification checklist contained in the PC/M failed to list the required changes to the Q-List. Although the checklist requires that changes to the Q-List be indicated, the entry under this item on the checklist was "Later." This apparent failure to adequately revise the Q-List may represent another example of an inadequate design control. This item is considered unresolved pending further NRC evaluation.

Response:

In regard to the design verification process for the Q-List, FPL has recognized that the current Q-List is a basic systems level document and is not intended to address individual system components. FPL previously discussed this issue with the NRC (refer to Inspection Report Nos. 50-250/84-33 and 50-250/84-34), and has committed to the development of an updated and more component specific Q-List. This new Q-List is data base effective as of November 15, 1985, and has been identified for NRC review as Inspection Followup Items 250/84-33-03 and 251/84-34-03. The classification of the nitrogen system components has been evaluated by FPL and is reflected in the updated Q-List.

Power Plant Engineering has developed draft Quality Instructions which define the requirements for modifications to and updating of the Turkey Point Q-List. These instructions require that a Q-List impact review be performed for all Turkey Point Plant Changes/Modifications (PC/Ms). They further provide the engineer with specific guidance on the mechanics of preparing changes to the computerized Q-List Data Base. These procedures will be in place by the end of February, 1986.

The current Q-List represents the plant as depicted on design documentation current as of May, 1985. FPL's contractor for the Q-List is being retained to update the Q-List to the now-current documentation. This effort is currently scheduled for completion by August 1986. Power Plant Engineering will then proceed to maintain the Q-List as a "living document" for future PC/Ms generated for Turkey Point.

URI 85-40-06:

The nitrogen backup system P&ID (Drawing 5610-M-339) incorrectly indicated that the system pressure regulators were set at 55 psig. Although this drawing was listed in PC/M 80-117 as a drawing requiring update, a change at some point in the implementation of the PC/M which modified the setpoint to 80 psig failed to ensure that the P&ID was again updated. This appears to be another example of inadequate design control. This item is considered unresolved pending further NRC evaluation.

Response:

The pressure control valves were originally set at 55 psig based on the original design of the plant which was not changed by the modifications made under PCM-80-117. Drawing 5610-M-339, Sheet 1 of 1 reflected this setpoint. Although this setpoint was acceptable based on vendor confirmation, the setpoint was adjusted to 80 psig after PCM 80-117 was implemented, to coincide with the normal air pressure operating range specified on the flow control valve data sheet. Due to an administrative oversight, this change was not incorporated on the referenced drawing.

The pressure setpoint shown on Drawing 5610-M-339 Sheet 1 of 1, Revision 15, is not a safety concern since the valve can operate at pressures significantly less than 55 psig based on previous discussions with the vendor and actual tests in the field. Therefore, the oversight did not affect the operability of the Auxiliary Feedwater System. Changes are currently being proposed by the Drawing Update Group to improve drawing accuracy. These changes are scheduled to be implemented by the end of 1986.

Revision 17 of Drawing 5610-M-339 and Revision 8 of the associated instrument index sheet 5610-M-311 Sht 155 have been issued to reflect the current setpoint of 80 psig.

URI 85-40-07:

The NRC Region II inspectors walked down additional portions of the licensee's AFW and related systems for Unit 4. The inspectors observed penetrations into the AFW headers which were identified as an abandoned in-place nitrogen blanket system. This system, at one time, provided a means of introducing nitrogen into the AFW system and subsequently into the Unit 4 steam generators. The inspectors reviewed Turkey Point Procedure 4-OP-075 to determine if the nitrogen system isolation valves (40-4-1610C, 40-4-1610B, 40-4-1283, and 40-4-1284) were identified in the valve alignment attachment. The aforementioned valves did not appear in this attachment.

Turkey Point Procedure O-ADM-031 dated December 14, 1984, Independent Verification, states that independent verification shall be applied to auxiliary feedwater system applicable procedures. This condition appears to be another example of an inadequate procedure. This item is considered unresolved pending further NRC evaluation.

Response:

Independent verification of the nitrogen blanket system valve alignment is not considered to be required by procedure O-ADM-031. Valve alignment of the nitrogen blanket system connected to the Train 1 AFW feedwater lines on Unit 4 is addressed in Operating Procedure O-OP-065.3 "Nitrogen Gas Supply System". This procedure identifies these valves to be in the closed position for normal operation. These valves are tagged and the tags are checked once per month.

This system has been reviewed for its necessity for tie-in to the auxiliary feedwater system. It was determined that this system is no longer required and is scheduled for removal by PC/M 85-181 during the current Unit 4 refueling outage.

URI 85-40-08:

Apparently no licensee evaluation had been or was intended to be performed on the scaffolding around the AFW FCV's. Housekeeping procedures AP 0103.11 and ASP-13 appeared to be inadequate.

Response:

A system is being established to provide a means to better control scaffolding. A scaffolding permit will be required (except in containment where other close out controls exist and on secondary system areas which do not directly impact safety related systems) prior to erecting a scaffold which will involve a review by operations personnel. This system will address the NRC concerns from page 18 and 19 of the Inspection Report. This scaffolding Control System will be proceduralized in Backfit Procedure ASP-26 by February 28, 1986 and in AP 0103.11 by March 31, 1986.

URI 85-40-09:

Emergency Operating Procedures 20004, revision dated August 23, 1985, and 20007, revision dated August 26, 1985, did not provide adequate guidance for the control room operators to assure the required 286 gpm of auxiliary feedwater is delivered to each unit within three minutes in the event of a two-unit trip with only one AFW pump available as specified by the Shared Auxiliary Feedwater System, System Description and Design Basis, Revision 1, dated January 31, 1985. This appears to be another example of failure to provide adequate procedures. This item is considered unresolved pending further NRC evaluation.

Response:

Emergency Operating Procedure 20004 (Loss Of Offsite Power) revision dated December 26, 1985, provides a note following step 5.3.1. This note provides guidance to assure the required AFW flow is provided in the event of a two unit trip. Emergency Operating Procedure 20007 (Loss Of All A.C. Power) revision dated October 30, 1985 provides a note following step 5.2.4. This note provides guidance to assure the required AFW flow is provided in the event of a two unit trip.

URI 85-40-10:

The licensee apparently failed to perform an adequate safety evaluation with regards to PC/M 80-117, in that at the time of PC/M implementation, the auxiliary feedwater system steam vent valves analysis was not performed for the condition of steam vent valve failure at low pressure conditions. This item is considered unresolved pending further NRC evaluation.

Response:

The classification of steam vent valves as non-safety related components is consistent with the original design basis for the plant and was not changed by PCM 80-117. Therefore, ANSI N45.2.11 was not applicable to either the original design of the vent valve or the subsequent modifications. A design analysis was performed prior to the modification as documented in Calculation MO8-162-02, dated November 20, 1981 to justify operation of the AFW system assuming failure of these non-safety related valves. The calculation was based on the scenario during the initial operating conditions of the Auxiliary Feedwater System with maximum steam pressure in the system. This case was considered bounding for the range of system operation during a transient.

In response to NRC concerns, a confirmatory analysis has been performed at the lowest steam operating conditions (at the time the RHR System is put into operation) which has confirmed previous engineering judgement. This analysis is documented in Calculation MOS-462-02, dated October 11, 1985. The analysis demonstrated and confirmed adequate steam supply to the Auxiliary Feedwater pump turbines is the event of a complete failure of the steam vent line.

The selection of the setpoint for the steam vent valve on the new steam supply header was based on the setpoint established under the original plant design for the vent valve in the existing header. At the time the new header was added, as an exact duplicate of the existing header, the setpoint of the original steam valve was specified for the new valve. There was no reason to question the validity of the original valve setpoint since the new valve was functionally identical. As stated previously, a design analysis has been performed which confirms the system's operability with the vent open at low steam pressure.

In any event the subject valves will be removed as discussed in the response to URI-85-40-11.

UFI 85-40-11:

FP&L is conducting a design review of the steam vent valve function. This review is expected to be complete by December 30, 1985. The review will address the following questions: (1) are the vent valves needed for the present auxiliary feedwater system design? (2) What was the reason for the 150 psi setpoint? (3) If the vent valves are required for system operability, can the 150 psi setpoint be reduced? The result of the design review and the licensee's actions will be inspected at a future date as an inspector followup item.

Response:

As part of our AFW system enhancement task team project (Item 8), we have reviewed the design basis for the original AFW steam header leakoff valves to determine their applicability to our present system. The results of this evaluation are documented in Bechtel letter SFB-2147 dated December 19, 1985.

The original Turkey Point Auxiliary Feedwater System contained pumps driven by low pressure steam turbines. In order to avoid over speeding and tripping of the turbines upon initial startup, the turbine pressure control valves were maintained closed. To start the pumps, the steam isolation valves (MOV-*-1403, MOV-*-1404, MOV-*-1405) were opened and the common steam supply line was pressurized. A pressure switch, located in the steam supply line then furnished a signal to open the pressure control valves at the turbine inlet to start the pumps. Normally open auxiliary feedwater steam vent valves (CV-*-6448, CV-*-2914) were provided downstream of the steam supply isolation valves to prevent pressurization of the steam supply line, due to isolation valve leakage. Pressurization of the steam line, as a result of isolation valve leakage, would cause undesirable cycling of the pressure control valves as well as the pumps.

The normally closed pressure control valves at the turbines were removed when the new high pressure turbines were installed. The new turbines were provided with motorized trip and throttle valves as well as governors with ramp bushings. The motorized trip and throttle valves are normally maintained open. The governor with ramp bushing will prevent overspeed tripping of the turbine upon initial starting. Therefore, the Auxiliary Steam Vent Valves are no longer required to prevent pressurization of the steam supply line as a result of isolation valve leakage, and undesirable cycling of the pressure control valve.

As a result of this evaluation, PC/M's 85-199 (Unit 3) and 85-200 (Unit 4) are being prepared to remove the vent valves from the system. This modification is scheduled for implementation in Unit 4 during the current refueling outage.

URI-85-40-12:

Procedure 3-OSP-075.1, dated August 7, 1985, did not adequately verify that MOVs 3-1404 and 3-1405 were independently capable of opening all their associated AFW flow control valves as designed. The upgrade of Procedures 3-OSP-075.1 and 4-OSP-075.1 dated August 1985 required opening both steam supply valves together which fails to ensure that either MOV 3-1404 or MOV 3-1405 could open all associated flow control valves in trains 1 and 2 thus ensuring a feedwater flowpath.

10 CFR 50, Appendix B, Criterion XVI, requires that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. The aforementioned procedure upgrade performed appears to represent an inadequate corrective action to a previously identified violation. This item is considered unresolved pending further NRC evaluation.

Response:

Technical Specification 3.8.4 provides LCO's for AFW trains. The procedure cited in the finding provided instructions which did in fact test the capability of the trains of AFW. No requirement was known to test the valves independently. Procedure 3/4 - OSP-075.1 now requires that each MOV is independently verified capable of opening all FCV's.

The steam supply MOV's (1403,1404 and 1405) are designed to trigger a switch upon opening to activate solenoids to allow operation of the flow control valves. Opening of each MOV will allow operation of all six flow control valves (three for each train).

The violation identified in Report Nos. 50-250/84-35 and 50-251/84-36 addressed the failure to visually verify the flow control valves would operate as designed. This previous violation did not explicitly require verification that the MOV's were independently capable of opening all associated AFW flow control valves. FPL conducted a procedure upgrade to include visual verification of valve operation.

Procedures 3-OSP-075.1 and 4-OSP-075.1 are followed to insure operability of each train of auxiliary feedwater. Operating valves 1404 and 1405 together verified operability of train 1. However, these procedures now require that each MOV is independently verified to be capable of opening all FCV's. This will insure more prompt identification of individual component failure.

URI 85-40-13:

Failure to meet a committment in a FPL letter dated December 20, 1979 to install adequate communications and DC lighting to support local AFW operations during testing and during control room inaccessibility is unresolved pending further NRC evaluation.

Response:

FPL's letter dated July 22, 1980 indicated that AFW system modificaitons had negated the need to add DC lighting or a sound powered phone link and thus FPL did not consider the December 20, 1979 committment to remain in effect. Nevertheless, DC lighting for the AFW pump area, feedwater platforms, and normal steam isolation valves has been completed, and a sound powered phone link between the AFW pump area and the control room will be complete by June 1986.

URI 85-40-14:

The control room inaccessibility procedure O-ONOP-103 dated August 7, 1985, did not address the local operation of train 2 of the AFW system. This apparent failure to provide an adequate procedure to cover a safety related activity is unresolved pending further NRC evaluation. The procedure also did not provide instructions on how to locally reset and restart a tripped AFW pump.

Response:

Procedure 0-ONOP-103, Control Room Inaccessibility, will be revised as follows:

1. Prior to taking any operator actions at Train 1 of the AFW System, the operator will be directed to check the AFW flow gauges to determine train operability and report the resulting finding to the Plant Supervisor - Nuclear.

2. A PC/M has been submitted to provide a means (reach rod) to make the instrument air isolation valves for Train 2 of the AFW System easier to operate. A procedure change will be made as necessary to incorporate any PC/M induced information into the procedure.

3. Instructions and setpoints will be incorporated to provide guidance to the operator of how to obtain proper flow to the steam generator utilizing a single AFW pump.

4. Instructions will be incorporated to address balancing of the AFW System flow to provide a flow rate of 286 gpm to each unit in the event only one pump is operable concurrent with a dual unit trip. This will be the same as presently delineated in EOP 20004 and 20007.

5. Instructions will be incorporated to provide instructions to the operator for locally resetting and restarting a tripped AFW System pump.

One item of disagreement with the report should be noted though. Isolation of the nitrogen valves at Train 1 of the AFW System does not isolate nitrogen to the Train 2 valves, therefore, the valve operation isolating nitrogen to Train 2 should not be deleted as stated in the report. No procedure change will be made.

IFI 85-40-15:

A review of the training and procedural improvements for assurance that the plant can be safely controlled from outside the control room will be made at a later date by an NRC evaluated walkthrough of the entire control room inaccessibility procedure.

Response:

Extensive training for operators and operator trainees on control room inaccessibility has been conducted. Each shift crew has conducted a walkthrough of the procedure. In addition this subject will be part of the next requalification cycle training scheduled to start in March 1986.

As indicated in the inspection report "The upgrade in lighting communications, procedural improvements, and additional training and walkthroughs to be conducted by the license should provide reasonable assurance that the plant can be safely controlled from outside the control room."

IFI 85-40-16:

Post-accident radiation zone map 5177-119-SK-M-1 dated April 27, 1981 indicates two of the AFW pumps would be in a very high radiation environment for a Unit 3 LOCA. In addition, although dose rate instruments are being kept in the control room, it appears that there is no guidance as to their use.

Response:

The AFW pumps are not required for mitigation of a large break LOCA on the affected unit. Should the pumps be required for the opposite unit, the personnel that would be required to take action on the two pumps would be staged in the onsite support center, where health physics review of tasks in the high radiation zone would be evaluated prior to dispatching individuals into the zones. All personnel required to work in radiation areas are trained to operate radiation detection instruments and are aware of actions to be taken for high readings on the instruments. In addition, FPL will conduct a review of the basis for the radiation map in question.

URI 85-40-17:

Administrative Procedure 0103.3, Control and Use of Temporary Systems Alteration, dated January 31, 1984, Section 5.8 states, the Plant Nuclear Safety Committee (PNSC) is responsible for reviewing applicable nuclear safety-related temporary system alterations within 14 days of the Plant Supervisor-Nuclear approval date.

Apparently, the licensee's PNSC failed to review and document TSA 3-84-11-75, TSA 3/4-85-8-75, TSA3/4-84-99-75 and TSA 3/4-84-100-75 within the prescribed time period. This appears to be another example of failure to implement approved procedures, and it is considered unresolved pending further NRC evaluation.

Response:

The TSA procedure (O-ADM-503) has been revised January 14, 1986, to require prior PNSC review of TSA's for equipment in service or component substitution. The 14 day review now applies only to TSA's implemented on equipment or systems out of service or covered by a clearance. This enhancement of management controls should reduce the likelihood of recurrence.

URI-85-40-18:

Apparently, the licensee failed to perform an adequate safety evaluation on TSAs 3-84-11-75, 3/4-84-99-75, 3/4-85-8-75 and 3/4-84-100-75, Removal of AFW Governor Speed Control Systems, in that the safety evaluations did not evaluate the mechanical reliability of the AFW pumps being operated under constant speed conditions. This item is considered unresolved pending further NRC evaluation.

Response:

A safety evaluation was conducted for these TSAs in accordance with AP 0103.3. However, the mechanical reliability of the pumps operating at a constant speed within the previously analyzed operating range was not addressed. To enhance management control of TSA evaluations, the Procedure (0-ADM-503) has now been changed so that TSAs for equipment in service will be reviewed by the PNSC prior to installation.

The AFW pumps and turbines are designed to operate up to a normal operating speed of 5900 RPM. The governor on the turbine driver is installed and set to maintain speed at a high speed setpoint of 5900 RPM. It is capable of receiving an external signal to vary turbine speed between 3200-5900 RPM.

Until recently, an external air signal was provided by a differential pressure controller which maintained a constant pressure drop across the flow control valves. Failure of this controller resulted in the turbine being required to operate at constant speed.

Removal of the differential pressure controller would also have resulted in the turbine operating at constant speed. This operating mode is within the previously analyzed operating range for both the turbine and pump. The designed mechanical reliability of the equipment is not considered to be reduced for that reason.

URI 85-40-19:

It appears that the licensee failed to take prompt and adequate corrective action to ensure that manual isolation valves 3-20-428 and 4-20-428, the common isolation valves for redundant safety-related condensate storage tank level instruments were properly administratively controlled. This item is considered unresolved pending further NRC evaluation.

Response:

The design bases for the addition of the redundant condensate storage tank level indication system, installed under PCM 80-77, was based on the connection to the tank being a passive portion of the system which allowed the redundant monitors to be on a common tap. This approach is considered acceptable since a single passive failure of this line or the isolation valve is not a design basis for Turkey Point. Prior to the implementation of PCM 80-77, the condensate storage tank was provided with a single level transmitter downstream of Isolation Valve 428. When the redundant level indication system was added downstream of Valve 428, it was assumed that the operation of the valve was adequately controlled by administrative procedure since PCM 80-77 did not modify either the tap off the tank or the valve. The need to control the isolation valve was not addressed in PCM 80-77 since the valve was existing and performing the same function, and Operating Procedure 7001.1 administratively controls Valve 428 in the open position.

In advertent closure of Valve 428 would create the potential for the operator to have misleading information concerning the condensate storage tank level. However, this is not considered to be a significant safety concern since there are other independent methods of determining tank level which would have alerted the operator to recognize that the level indication system was not functioning properly. Level Switch LS-3/4-1503, which is safety related, alerts the operator that the minimum Technical Specification volume of 185,000 gallons is remaining in the tank, would be available since it is not associated with this level tap. Should the condensate storage tank reach this level, the operator would have noted a discrepancy in the level readings and taken corrective action. Also, control room alarms are available to alert operators with respect to tank level.

The condensate storage tank Technical Specification also requires, by definition, a minimum volume of water for nineteen hours of Auxiliary Feedwater System operation. With the Auxiliary Feedwater System in operation and drawing on the inventory of the condensate storage tank, the operator would have noticed that the condensate storage tank level (as indicated by the redundant transmitters and checked by log readings) was not decreasing during this time period and questioned the validity of the level indication; appropriate corrective action could have been taken.

In addition, both condensate storage tanks are normally aligned to the Auxiliary Feedwater pump suction. Assuming Valve 428 was inadvertently closed on one tank, the level on the opposite tank would be operable. Since the levels in the

URI 85-40-19 (continued):

two tanks will decrease at approximately the same rate, a disparity between the levels in the two tanks would have been recognized by the operators and appropriate corrective action could have been taken.

It should also be noted that the design modification process has been substantially improved since the time this modification was implemented in recognition of the need to coordinate changes in the plant with operations and maintenance personnel. A program for the review of proposed plant modifications has recently been established to ensure that the effects on operating documents, procedures and administrative controls are accommodated in the design prior to approval of the PCM by the Plant Nuclear Safety Committee (PNSC). Engineering personnel are also currently on controlled distribution for the plant operating procedures, which provides the design engineer with a better insight into the actual operation of the system and the potential impact of modifications of the system. Also, utilization of Standard Engineering Packages should greatly aid this area.

As noted in the NRC report, Valve 428 has been locked open. In addition, the associated drawings have been revised to show this valve locked open by administrative control and the valve has been added to the locked valve list.

Administrative Procedure AP0103.5 (Administrative Control of Valves, Locks and Switches) provides instruction for placing valves, locks and switches under administrative control when it is necessary to lock the valve or switch to prevent inadvertent misoperation of the valve or switch. These valves were added to the procedure revision approved November 13, 1985.

URI-85-40-20:

The apparent failure to ensure that procedure 3/4-OP-018.1, Condensate Storage Tank, a safety-related procedure was approved for release by authorized personnel and appropriately distributed prior to the cancellation of OP-7001.1, is considered unresolved pending further NRC evaluation.

Response:

This is considered an isolated case which occurred due to the complexity of the specific change. In this case several procedures were being issued to replace one old procedure. Because one of the replacement procedures was undergoing review in a different section of the plant staff, the old procedure was inadvertently cancelled when the remaining replacement procedures were issued. Administrative Procedure 0109.7 provides guidance for the PUP group to cancel procedures as new replacement procedures are generated.

IFI 85-40-21:

The inspector noted several examples where cancelled procedures were referenced in other procedures still in use. The inspector was informed by the licensee that at present there is no method for cross-referencing procedures to ensure that a cancelled or changed procedure does not affect another procedure. The inspector informed the licensee that the development of a method to ensure all procedures are properly updated could be of benefit. This is an inspector followup item.

Response:

It is currently the procedure writer's responsibility to ensure that cancelled or changed procedures do not affect another procedure. FPL is considering a proposal for a computer cross referencing system. It is expected that a decision will be made on this potential enhancement by March 15, 1986.

URI 85-40-22:

The apparent failure to evaluate the impact of design changes on the AFW control system on the nitrogen consumption rate is considered an unresolved item pending further NRC evaluation.

Response:

The modification to split the nitrogen backup system into two headers was issued for implementation under the original scope of PCM 80-117 as shown on Drawing 5610-M-339/80-55, Revision 2, dated January 15, 1982. This design was established by engineering judgement (although not fully documented) based on a technical evaluation of the original design basis for the nitrogen backup system in consideration of the following factors:

- o The total number of flow control valves supplied by the on-line bottles in the split system was half of that in the original design. The original design had one bottle on line serving six flow control valves. The modified system resulted in one bottle on line serving three flow control valves.

- o The total nitrogen consumption for the modified system was significantly less than the original system. The air consumption rate of original flow control valves was 1.0 scfm per valve, as compared to the air consumption rate for the replacement valves of 0.26 scfm, based on the original regulator setpoint of 55 psig.

- o The pump differential pressure controllers were installed and operable, and maintaining the design pressure drop across the flow control valves. On this basis, valve oscillations were not an operational problem.

- o The design flow rate through each flow control valve was 200 gpm.

- o The low pressure alarm setpoint for the nitrogen bottle system at the time the PCM was issued was 1005 psig, which allowed 15 minutes for the operator to take the necessary action to valve-in a new bottle.

- o Air operated hand controllers for the flow control valves supplied by the backup nitrogen were removed from the system as defined in the scope of work under PCM 80-55.

The split of the nitrogen system into separate trains under the original scope of PCM 80-117 was considered acceptable within the parameters of the original design basis for this system. However, in response to NRC concerns with the engineering judgement used as a basis for this modification, a detailed analysis of the nitrogen backup system was performed, based on the original design parameters described above and worse case results of previous tests performed on the system. The results of this analysis are documented in Calculation MO8-462-05, Revision O, dated November 1, 1985. This analysis demonstrates that sufficient nitrogen should have been available from the valved-in bottles in the

split system to permit the system to operate for more than 20 minutes without operator action following receipt of a low level alarm at 1005 psi. Based on these results, the split of the nitrogen system was confirmed to be consistent with the original design basis and operating procedures for the system.

Responses to the specific NRC review team concerns are discussed below:

- The NRC review team indicated that a steady-state consumption rate of 0.26 scfm was utilized in evaluating the nitrogen consumption rate for the new flow control valves, instead of a rate of 0.36 scfm. The 0.26 scfm consumption rate was based on the original regulator setpoint of 55 psig. As discussed above, Calculation MO8-462-05 was prepared to analyze the available nitrogen from the split system. This analysis included a steady-state consumption rate greater than 0.36 scfm to account for system leakage and the change in the regulator setpoint to 80 psig, and confirmed the acceptability of the split system.

- The NRC review team concluded that the assumption of instantaneous steady-state operations was not consistent with the as-designed valve response. As discussed above, the analysis supporting the split nitrogen system was based on the original plant design features which limited flow control valve oscillations. This is consistent with the vendor's technical literature which indicates that the valves quickly reach their setpoint with virtually no overshoot.

However, several changes were made to this system after PCM 80-117 was released for implementation, which induced oscillations in the flow control valves and resulted in a subsequent increase in the nitrogen demand for the system. These changes were unrelated to the original scope of PCM 80-117 and included the following:

1. The differential pressure controllers on the Auxiliary Feedwater System were disconnected due to maintenance problems with these components and difficulties in obtaining spare parts, which resulted in an increased pressure differential across the flow control valves. This increased pressure differential resulted in oscillation in the flow control valves, and in increase in the nitrogen consumption rate under test conditions witnessed by the NRC review team.

   Removal of the pressure controllers was justified by analysis on the basis that the excessive differential pressure across since the steam generator pressure would rapidly rise to the safety valve setpoint. Under this condition, the pressure drop across the flow control valve would be essentially the same regardless of whether the differential pressure controller was installed.

2. Another change which induced oscillations in the flow control valves was the reduction in the auxiliary feedwater flow rate from 600 gpm to 373 gpm. This reduced flow rate resulted from a Westinghouse reanalysils of the feedwater flow requirements as documented in

Westinghouse letter W-P.TP-62, dated June 3, 1982. As a result of this change, the setpoint for the flow control valves were revised to 125 gpm.

Reduction in the setpoint compounded the oscillation problems in the flow control valves since the design basis for the system was established at 200 gpm. Subsequent to this change, a review of the valve oscillation problem was made during field testing of this system. These tests confirmed that the control valves performed satisfactorily under the original design condition of 200 gpm flow at 25 psi differential. However, oscillating control valve action was experienced when the system was tested at the reduced flow rate of 125 gpm and the auxiliary feedwater pumps running at maximum speed. As a result of these tests, modifications were recommended to eliminate the valve oscillation problems. It is anticipated that new valve trim will be installed by the upcoming refueling outages for each unit.

- The NRC review team noted that the reduction in the low pressure alarm setpoint from 1000 psi to 500 psi did not appear to be based on a documented analysis. This change was made based on a field performance test conducted on March 1, 1984. The criteria developed for the test was based on steady state operation of the valves on the understanding that valve oscillations were not a design basis for the system consistent with the vendor's technical literature which indicates that the valves quickly reach their setpoint with virtually no overshoot and the oscillations identified during testing would be eliminated by subsequent modifications to the valves. The amount of time required for valving in the next bottle upon actuation of the low level alarm was established at 10 minutes based on discussions with plant operations personnel. A total of six flow control valves were included in the test to add conservatism to the setpoint since only three valves are aligned to one open bottle in the new system arrangement. Also, the valves operated for approximately 15 minutes based on the 500 psig setpoint which was considered another safety factor margin for the setpoint. On this basis, the test is considered to be a satisfactory method of establishing the low pressure alarm setpoint, in lieu of a documented analysis.

  The NRC's concerns with the volume of available nitrogen to the flow control valves are directly related to the valve oscillation problems witnessed during the inspection. However, as discussed previously, valve oscillations are not considered a safety concern since the high differential pressure across the flow control valves not expected to exist when the system responds to a design basis accident. FP&L has pursued resolution of this problem in a systematic manner through coordination with its architect-engineer. NRC's final evaluation of its inspection findings should give due recognition to the fact that this problem was identified by FPL prior to the inspection and that design modifications were in progress.

When the NRC audit identified increased N_2 consumption due to the valve oscillations noted during testing, FPL reanalyzed the nitrogen system and revised the low pressure setpoint to 1350 psig, to allow a minimum of ten minutes for the operator to valve-in bottles based on a conservative assumption of valve oscillation and with no credit taken for placing the valves in the manual mode. The low pressure alarm has been temporarily reset in the field to this revised setpoint, and the appropriate design documents have been revised.

A surveillance procedure to dynamically test the nitrogen back-up system is currently being prepared and is scheduled for issuance by April 30, 1986. When implemented, this test should identify any detrimental effects on nitrogen consumption created by modifications to the AFW system.

IFI 85-40-23:

The AP 190.15, Document Verification Checklist, is still not being completed under Step 5 which requires that changes to the "Q" List be listed. The entry on PC/M 83-117 contains "to be determined by FP&L." Although the PC/M has been turned over and completely closed out, necessary changes to the "Q" List have not been evaluated.

Plant personnel stated that this was due to the development of the new, component level "Q"-List which will soon be issued. This check has been neglected on PC/Ms which have been processed during the "Q"-List development. Once the "Q"-List has been promulgated, engineering procedures will be developed for maintenance of the "Q"-List through evaluations of system modifications. This item will be left as an inspector followup item to verify appropriate procedures and responsibilities are developed.

Response:

Turkey Point Plant and Power Plant Engineering procedures are currently in the process of review and modification to incorporate the new component level Q-List.

At Turkey Point Plant, the Q-List Task Team has identified the Administrative Procedures which will require modification. The Procedures have been marked up with the team's recommended changes. A meeting will be scheduled for the end of January to discuss these proposed changes with Maintenance, Operations, QC, Procurement, and Procedure Update. At that time, an incorporation schedule will be developed.

Power Plant Engineering has developed draft Quality Instructions for Q-List use and maintenance. A meeting will be scheduled for the beginning of February to discuss comments on these procedures. These procedures are expected to be finalized and implemented the end of February.

Since the new Q-List reflects plant documentation current as of May, 1985, there exists a "delta" between the Q-List and up to date documentation. The Q-List contractor is being retained to update the Q-List for FPL and this effort is expected to be complete by August 1986. Power Plant Engineering will then proceed with future updates of the Q-List.

Engineering and plant personnel are now using the computerized Q list. Training is underway and for the interim period while the new Q list is being fully implemented, the plant is also checking the PNSC approved prior Q list in addition to the new list.

IFI 85-40-24:

The licensee is preparing an engineering evaluation addressing how long the AFW system must operate without operator action in the automatic flow control mode and subsequently in the remote-manual mode. This evaluation will necessarily impact on the required volume of stored nitrogen. It was also noted that the licensee is planning to add an additional five-bottle nitrogen supply so each train will have five bottles available. The results of the engineering evaluation and the additional nitrogen supplies will be tracked as an inspector followup item.

Response:

The original nitrogen backup system for the auxiliary feedwater flow control valves consisted of five nitrogen bottles to supply motive power for six flow control valves. Original calculations concluded that the five bottle station would provide sufficient capacity for two hours of valve operation, however this was not a design basis requirement for the system. The nitrogen station is designed to allow sufficient time for bottle change out while maintaining system operation.

A recent review of the present system has resulted in the preparation of PC/M's to install additional bottle stations for each unit to extend the presently sufficient time to change out bottles during AFW system operation with the N_2 station in use. The additional bottle station for Unit 4 is scheduled for installation during the current refueling outage. An evaluation to more clearly address the operator action requirements for the system is presently being prepared for incorporation into the AFW system design basis document.

URI-85-40-25:

The inspector noted that a portion of the nitrogen system that was outside of the scope of NCR 341-85 was supported at intervals greater than the 36 inches specified by the licensee's "Design Guide for Seismic Class I Instrument Tubing Installation" for stainless steel tubing. This document is FP&L's specification No. 5177-J711, Revision 2. The requirement for a maximum unsupported span of 36 inches is adopted from the ASME Code Section III. Contrary to this design requirement, the inspector measured two adjacent supports that were 41 inches and 43 inches apart. Not only do these exceed the licensee's seismic Class I requirements; but the instrument tubing was not attached to the central support between these two spans which creates a section of unsupported tubing approximately 85 inches in length. The inspector also found a section of unsupported stainless steel instrument tubing in Unit 4 of 41 inches in length. Pending further review of the circumstances which resulted in the nitrogen system not being seismically supported, this item is considered unresolved.

Response:

All of the specific support conditions identified during the inspection have been evaluated and the system has been determined operable based upon functionality criteria. To further identify potentially unsupported connections to the auxiliary feedwater system and other safety systems, FPL has initiated a program to walkdown and "as built" all 2 inch and under piping associated with these systems not evaluated under IE Bulletin 79-14.

URI 85-40-26

Consideration for NPSH was not documented in the FP&L calculation, Low Level Alarm on Condensate Storage Tank, dated November 15, 1979, and the calculation inadequately identified the necessary assumptions and design inputs. An informal, undated, untitled, annotated sketch was presented by the licensee as evidence of NPSH consideration, which had not been properly referenced in the November 15, 1979, calculation. The sketch lacked proper identification and detail to permit an understandable review.

It appears that the licensee failed to adequately document all assumptions and design inputs for FP&L's calculation of November 15, 1979, on the Low Level Alarm on Condenste Storage Tank. This item is considered unresolved pending further NRC evaluation.

Response:

The inspection report stated that "... the preparer appeared to have assumed that the minimum NPSH would be below the instrument tap, because the analysis calculated the height above the instrument tap which corresponds to 20 minutes of water at a usage rate of 600 gpm with a 10% factor for conservation. The team independently confirmed that the NPSH is well below the instrument tap and the design is not deficient".

Engineering has found evidence on the microfilm records which indicates that NPSH was considered in the original calculation. However, this consideration for NPSH was not documented in the FPL calculation dated November 15, 1979. Calculation MO8-462-01, dated October 1, 1985, was recently performed to confirm that the required NPSH water level for flows anticipated at the end of the cooldown transient is below the instrument tap level as assumed in the original calculation. The new calculation confirmed the results of the original calculation, and therefore the equipment and associated condensate tank level setpoints are acceptable. Enhanced documentation of calculation assumptions are being pursued for both internally and contractor-developed calculations. Quality Instruction revisions have been or are in the process of being issued to provide enhanced controls for documentation of calculational assumptions and inputs for both internal and contractor-developed calculations.

URI 85-40-27:

The inspector located two subsequent examples of OTSCs which were processed as a change of intent. On November 12, 1985, the PNSC reviewed and the plant manger approved OTSC No. 3734 to TOP 206, Reactor Protection System Periodic Test (Unit 4). The change of intent guidelines checklist indicated that this change did alter the intent of the original procedure. This procedure change was made in response to IE Bulletin 85-02. On November 12, 1985, the PNSC also reviewed and the plant manager approved OTSC No. 3733 to AP 103.12, Notification of Significant Events to the NRC. The change of intent guidelines checklist indicated that this change did alter the intent of the original procedure. This procedure change was also made in response to IE Bulletin 85-02. The conduct of these two temporary procedure changes is being further reviewed by the NRC and considered an unresolved item.

Response:

Administrative Procedure 0109.3 (On the Spot Changes to Procedures) is being revised to differentiate between temporary changes made in accordance with Technical Specification 6.8 and on the spot changes made with prior PNSC approval. The temporary change instructions will strengthen controls so that no changes to the intent of procedures will be made under this portion of the procedures. Because prior PNSC approved OTSC will be given all required safety reviews, change of intent will be allowed for this type procedure change.

IFI 85-40-28:

A review of the new maintenance training and the qualification tracking system will be an inspector followup item.

Response:

FPL personnel will be available to discuss the new maintenance training and the qualification tracking systems during the implementation and upon completion of the project.

URI 85-40-29:

There appears to be a difference in the philosophy between the manufacturer's recommended motor overload heater size and the size chosen by the licensee in order to agree with Regulatory Guide 1.106. This item will require further review by NRC and is identified as an unresolved item.

Response:

As stated in our previous response to this item contained in our letter L-85-439, Turkey Point has made no commitment to utilize the Limitorque sizing recommendations for overload heaters. Our current design meets the general intent of Reg. Guide 1.106. The overload heaters are sized by the heater manufacturer using his standard sizing criteria and the applicable plant motor data.

We have however, determined that we will re-evaluate our philosophy on overload heater sizing in accordance with the Limitorque sizing recommendations and Regulatory Guide 1.106, taking whatever action is required.

URI 85-40-30:

The safety system functional inspection team inspector expressed concern that the MOVs would not function under the conditions as stated in the manufacturer's letter and therefore requested a review of the manufacturer's calculations or that testing be performed at the low voltage conditions to verify acceptability. Additionally, the reduced voltage will result in an increased operating travel time. This added time should be reviewed to determine if any ISI requirements are affected.

This item will be unresolved until a review of the manufacturer's calculations or testing of these MOVs at reduced voltage is accomplished.

Response:

As stated in our previous response to this item contained in our letter L-85-439, calculation 5177-462-E01 was prepared utilizing a very conservative starting current of 53 amps. Subsequently this calculation has been reviewed utilizing the Limitorque data for starting current and the results indicate that the voltage at all of the subject valves will exceed 90V. This reduced voltage at MOV-4-1403 will result in a slightly increased stroke time which based upon a preliminary investigation should not adversely affect the design basis for the Auxiliary Feedwater System.

FPL is preceeding with preparation of a revision to calculation 5177-462-E01 utilizing the Limitorque data for starting current and also with a formal review to determine what effect, if any, increased MOV stroke time may have on the design basis for the Auxiliary Feedwater System. These items are expected to be completed by February 21, 1986.

URI 85-40-31:

The safety system functional inspection team expressed concern regarding the operation of a second steam vent valve that had been added between the steam admission valves and the AFW pump turbine.

Bechtel Power Corporation calculations No. MO8-462-02 approved September 30, 1985 (for low steam pressure conditions) and No. MO8-162-02, approved November 20, 1981 (for high steam pressure conditions) indicate that a 17% margin of steam is available with the 3/4" vent valve open which would be the condition for loss of AC.

This item will remain unresolved pending review of the calculations by the NRC.

Response:

The non-safety related classification for the steam vent valve added under PC/M 80-117 is consistent with the original design basis for the plant. As discussed previously in response to item 85-40-10, a detailed analysis was performed prior to equipment installation (Bechtel calculation MO8-162-02 approved November 20, 1981) which confirmed the capability of the AFW system to operate if the vent valve failed open. Bechtel calculation M-08-462-02 approved September 30, 1985 was performed to confirm previous engineering judgement and insure system operability with the failed vent for the full operating range of the system.

On this basis, there has no reason to power the steam vent from a safety related power source and powering the valve from a non safety AC source is considered acceptable. To support the NRC review, the referenced calculations are available at the Turkey Point Plant site.

URI 85-40-32:

It appears that the low nitrogen pressure switches were not reviewed during the Seismic Qualification of Auxiliary Feedwater System evaluation. The licensee advises that these switches are part of the original design and therefore, not designed or considered safety-related. However, current plant emergency operating procedures require action by the operator upon receipt of the nitrogen low pressure alarm. Additionally, consideration must be given to the fact that the nitrogen system is a backup system and the status of backup system should be known at all times. Since there appears to be confusion as to the safety-related application of these switches, this item is unresolved pending further NRC review.

Response:

As stated in our previous response to this item contained in our letter L-85-439, the design modifications for the Auxiliary Feedwater System utilized the pressure switches and annunciation system installed under the original plant construction, which were neither designed nor maintained as nuclear safety-related. As a result, the separation criteria for these components was not changed from the original design basis of the plant, and ANSI N45.2.11 does not apply.

However, we have determined that this system will be redesigned as part of the Auxiliary Feedwater System upgrade which involves the addition and relocation of the Auxiliary Feedwater Nitrogen Stations. This redesign will consist of the installation of new qualified pressure switches, indicators and wiring thereto. A pressure switch at each station will be alarmed in the Control Room with a trouble light and will be of a safety grade design.

URI 85-40-33:

The inspector reviewed Tech Spec. 4.8.2.b and determined that it requires the licensee to monthly perform an equalizing charge on each battery and to take specific gravity and the voltage readings of each cell. It is Plant Procedure 9604.1 which implements this requirement. However, it appears that this Technical Specification requirement is more applicable to lead antimony batteries which were the original type of batteries installed at Turkey Point. The Gould-GNB manual recommends that lead calcium batteries should be given an equalize charge only when needed. The inspector also observed that Plant Operating Procedure 9604.1 identifies two different float voltages for the Gould-GNB NCX type batteries. The inspector questioned the licensee about the acceptability of Tech Spec 4.8.2.b and about what effects this monthly equalize charge may have on the lead calcium batteries. The licensee indicated that they would contact the vendor to determine the acceptability of the current licensee requirements. This item is identified as unresolved item pending licensee and NRC evaluation.

Response:

We have contacted the battery manufacturer, Gould, with regards to the acceptability of Technical Specification 4.8.2.b and the effect the monthly equalizing charge may have on the lead calcium batteries. Gould has stated that the overcharging of the batteries at 140VDC once a month does not have any appreciable effect on the battery qualified life.

In order to clarify the battery charging requirements, a Technical Specification change to provide for an as-needed equalizing change will be submitted by March 31, 1986. This date revises that provided in our letter, L-85-439, dated December 6, 1985. This time extension allows for a more thorough review of the requirements and better coordination with the Standard Technical Specification Project.

URI 85-40-34:

Technical Specification 4.8.2.b states that monthly, each battery shall be given an equalizing charge, and afterwards specific gravity and voltage readings shall be taken and recorded for each cell. It appears that Plant Operating Procedure 9604.1 dated August 7, 1985, was inadequate in that it did not contain vendor recommendations for compensating cell specific gravity readings for electrolyte temperature and level. Furthermore, the procedure did not contain acceptance criteria for the specific gravity readings. This item is considered unresolved pending further NRC evaluation.

Response:

Plant Operating Procedure 9604.1 dated December 4, 1985 was revised to require specific gravity correction for electrolyte temperature and level. This procedure was also revised to contain acceptance criteria for the specific gravity readings. This procedure change had been planned prior to the inspection, but had not been completed.

URI 85-40-35:

The original battery specification and the specification for the new batteries required a capacity of supplying the specified loads for one hour without the battery terminal voltage falling below 105 VDC. It should be noted that the testing of the batteries did not meet the intent of the specification design requirements nor the acceptance criteria identified in the FSAR.

This appears to be another example of an inadequate procedure since test procedures for the initial testing of batteries 3A and 3B did not require load testing that verifies that the commitments of the FSAR or the design specifications were met. This item is considered unresolved pending further NRC evaluation.

Response:

FPL considers that the present one-half hour battery service test is adequate to demonstrate that the battery is capable of performing its intended safety function. This is based on the assurance described in the Bases for the Technical Specification, that considering any single failure, battery charging current should be supplied in one-half hour or less.

However, we are evaluating the discussion in the FSAR to determine if the present battery test should be modified.

URI 85-40-36:

Mechanical maintenance personnel were uncertain regarding the type of grease to be used in MOV gearboxes. This was considered a problem for two reasons. First, the mixing of different types of grease in the gearbox could cause hardening or separation of the lubricant. The potential for this exists at Turkey Point because its preventive maintenance instructions for Limitorque gearboxes specify the use of Texaco Marfac, while these same Limitorques have been supplied with either Exxon Nebula EPO or EPI or Sun 50 EP lubricants. Secondly, the only Limitorque lubricant that meets the environmental qualification requirements of 10 CFR 50.49 at Turkey Point is Exxon Nebula EPO or EPI.

Response:

Exxon Nebula is, for MOV's inside containment, the only Limitorque (MOV gearbox) lubricant that currently meets the environmental qualification requirements of 10 CFR 50.49. Preventive Maintenance instructions for Limitorque gearboxes, which specified the use of Texaco Marfac, have been pulled and are no longer in use at Turkey Point. The Lubrication Manual's current revision specifies usage of only environmentally qualified grease. This manual is now a PNSC reviewed document. MOV gearbox grease has been sampled and changed as appropriate. Additionally, an Engineering Evaluation has been performed and forwarded to the Region II Administrator on November 27, 1985 (L-85-448) documenting the justification for prior limited operation with various greases in Limitorque Valve Actuators.

In addition, instructions were in the process of being revised to address this issue when the inspection team reviewed this issue. The GEMS planners, who make up the work packages, were not uncertain about the type of grease to be used. Finally the grease guns used for maintenance are now in a controlled system.

URI 85-40-37:

During the AFW system walkdown, a Region II NRC inspector observed that the governor oil in the licensee's "A" AFW pump appeared to be different than that in the "B" and "C" turbine governors. The inspector requested any licensee controls or historical data along with vendor information which would identify the control oil in the governors. The licensee did produce the vendor manual with specific oil requirements; however, the licensee had to have performed an oil analysis to determine the oil in the governors. This analysis established that a Texaco 1OW oil was being used in the "A" AFW governor and Texaco 30W in the "B" and "C" governor.

The inspector expressed concern regarding whether the oil met vendor recommendations. It was later determined by the licensee that the Texaco 10W did not meet vendor temperature recommendations for the licensee's AFW operation. This item is considered unresolved pending further NRC evaluation.

Response:

To address the concern regarding apparent difference between the oil used in the "A" AFW turbine governor and that in the "B" and "C" turbine governors, we have sent samples for analysis to determine the specific oil used in each turbine governor. The results of the analyses identified that the "A" AFW turbine governor contained an SAE 10 weight oil (similar to Texaco Regal 32 used on site) and the "B" and "C" turbine governors contained an SAE 20 weight oil (similar to Texaco Regal 68 used on site).

To determine the correct oil to use in the AFW turbine governors, it is necessary to define the governor oil operating temperature range. Preliminary data was taken during turbine operation on November 25, 1985 to determine governor oil operating temperature. Comparing the data and specific oil weight to information provided in Woodward Governor Company Manual 25071C, "Oils for hydraulic controls" it was concluded that both oils are acceptable for use in the AFW turbine governors.

To determine the optimum oil for use in the AFW turbine governors, we have developed a governor testing program which will provide additional information to be used for oil selection.

IFI 85-40-38:

For the AFW flow control valves to meet the requirements of the present "FAIL-SAFE" test, they must be successfully exercised full open, then closed from the control room with visual verification at the valve. The power is not removed, nor is the air/nitrogen supplies isolated.

ASME Code Section XI, Division 1, IWV-3415, Fail-Safe Valves states, when practical, valves with fail-safe actuators shall be tested by observing the operation of the valves upon loss of actuator power. If these valves cannot be tested once every 3 months, they shall be tested during each cold shutdown; in case of frequent cold shutdowns, these valves need not be tested more often than once every 3 months.

As of November 22, 1985, a request for engineering assistance was being generated by the Turkey Point Nuclear Technical Department for the review of the present fail-safe testing method for these valves, identification of inadequacies in the fail-safe testing procedure, and development of a method for proper fail-safe testing of these valves. This will be an inspector followup item.

Response:

The auxiliary feedwater control values are designed to fail in the closed position by spring action on loss of air or power. For these valves to pass a fail-safe test in accordance with IWV-3415 of Section XI of ASME boiler and pressure vessel code, valves with fail-safe actuators shall be tested by observing operation of these valves upon loss of actuator power. Present plant procedures 3/4-OSP-075.1 and 3/4-OSP-075.2 require these valves to be exercised full open, then closed from the control room with visual verification at the valve to address the code requirement. Additionally, the valves are verified in the closed position when the testing is terminated and the steam supply MOV's (1403, 1404 and 1405) for that unit are closed which isolates power to the solenoids.

A review of the present fail-safe testing is being performed to assure compliance with Section XI and subsequent recommendation for any testing modifications will be provided by March 28, 1986.