U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

DIVISION OF REACTOR INSPECTION AND SAFEGUARDS

Report No.: 50-250/87-49 and 50-251/87-49

Docket No.: 50-250 and 50-251

Licensee: Florida Power and Light Company
P.O. Box 14000
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Inspection At: Turkey Point Power Plant
Florida City, Florida

Inspection Conducted: December 7-11, 1987

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1.0 Inspection Scope

This one-week team inspection evaluated safety review activities conducted by the licensee pursuant to Title 10 Code of Federal Regulations Part 50.59. Related functions of the Plant Nuclear Safety Committee (PNSC) and the Corporate Nuclear Review Board (CNRB) were also assessed. The inspection consisted of record reviews, interviews with cognizant personnel, and attendance at meetings. The purpose of the inspection was to determine whether appropriate issues were subjected to 10 CFR 50.59 reviews, whether correct determinations were made with respect to unreviewed safety questions, and whether documentation of these reviews were complete in describing the bases and rationale for conclusions.

2.0 Summary of Significant Findings

Steady improvement in the quality and completeness of 10 CFR 50.59 safety evaluation documentation was observed over the past year. Further improvement and the continuation of current skills in this area should be aided by licensee initiatives to issue comprehensive procedural guidelines and to provide training to evaluators.

In general, recent safety evaluations reviewed were sufficiently detailed to demonstrate, as a stand-alone document, the logic and bases for determinations regarding potential unreviewed safety questions. One notable strength was the use of prior 10 CFR 50.59 review and Plant Nuclear Safety Committee (PNSC) authorization for temporary modifications to the plant, including jumpers and lifted leads.

Some areas were identified as weaknesses which should receive attention by the licensee. Despite attempts at mitigation, the volume of material requiring PNSC review has resulted in long and frequent meetings, some brief reviews, and diversion of management from normal duties. Some safety evaluations reviewed were no longer valid, but no system was in place to identify these. Licensee attention should be directed to insuring that Request for Engineering Assistance (REA) evaluations were completed in a timely manner. Finally, the licensee should reevaluate its position regarding the treatment of check valve active failures and the retention of design calculations.

3.0 Design Changes

3.1 Design Change Process

Licensee procedures for all design and safety analyses performed by the Power Plant Engineering Department were contained in Quality Instruction JPE-QI-3.2, Revision 3, dated July 30, 1982. The guidance provided in this procedure was minimal with respect to 10 CFR 50.59 evaluations in that the definition of unreviewed safety question was quoted from the rule, and an additional statement was made to the effect that the discussion should clearly indicate the bases for the conclusions reached.

Station Administrative Procedure 0190.15, Plant Changes and Modifications (PC/M), dated October 26, 1987, provided essentially the same information as JPE-QI-3.2. Station Administrative Procedure 0190.22, Changes, Tests, and Experiments, dated December 12, 1986, offered some additional guidance to the extent of providing a suggested format for safety evaluations in an appendix. This suggested format included specific references to the basis for each conclusion when documenting responses to the questions in evaluating the potential for an unreviewed safety question or reduction in the Technical Specification (TS) margin of safety.

In view of the observed recent improvement in documenting safety evaluations, the inspector expressed concern that procedures had not been upgraded to provide more complete guidance. Additionally, licensee representatives stated that no specific training had been provided to those individuals conducting the reviews. As part of the continuing improvement process, copies of various industry documents relating to 10 CFR 50.59 had been distributed. Finally, in response to the above concerns, the licensee provided a draft copy of procedure JPE-QI-3.8, 10 CFR 50.59 Evaluations Performed By Power Plant Engineering.

To the extent reviewed, QI-3.8 will provide considerably more guidance and direction to the 10 CFR 50.59 review and documentation process than that which is contained in existing procedures. While the initiative to issue such guidance is seen as a strength, the inspectors identified some potential problem areas in the procedure as discussed below. The licensee also stated that training will be conducted in the spring of 1988 after JPE-QI-3.2 is issued.

3.2 Design Changes Reviewed

The inspectors reviewed several issued design change packages which are described in detail below. One overall strength was noted during this review. Safety evaluations were completed for all Plant Change/Modification (PC/M) without regard for classification (nonsafety-related, quality related, safety-related). In addition, even though the proposed change may have been determined not to be discussed in the FSAR, the remainder of the evaluation was completed to address any potential for an unreviewed safety question.

PC/M 87-194, Revision 1, issued June 17, 1987, added a restricting orifice in each Unit 3 Containment Spray Pump discharge pipe to maintain pump NPSH and prevent pump runout. The change was initiated after a design basis reconstitution review found that the original design calculation included the orifice but it had not been installed in the plant. The associated design analysis and safety evaluation classified the change as nuclear safety related and considered the following factors: ALARA, other changes in progress, orifice material, effects on environmental qualification, human factors effects, ASME Section XI testing, ANSI B31.1 requirements, masonry wall interactions, fire protection, and diesel generator loading. The safety evaluation concluded that no unreviewed safety question was involved and provided a detailed basis for the response to each of the questions in 10 CFR 50.59 (probability of occurrence, change in consequences, accident of different type, probability of equipment failure, margin of safety in TS bases, TS change).

One of the conclusions reached in the design analysis dealt with the Refueling Water Storage Tank (RWST) low level alarm. To ensure sufficient pump NPSH, the RWST low level alarm setpoint was to be raised to 40', resulting in a minimum tank volume of 155,000 gallons. To confirm that the design change process was complete, the inspector reviewed procedure 3-PMI-062.1, Refueling Water Storage Tank Level Instrumentation Channels L-3-6583 A/B Calibration, dated June 16, 1987. The instrument low level alarm setpoint was found to be consistent with the value stated in the design change. A spot-check of Emergency Operating Procedure 3-EOP-E-1 also found that the new value had been inserted.

PC/M 87-296, Revision 0, issued October 12, 1987, installed additional raceway and cable in order to allow the use of nonsafety-related battery 4C to supply DC bus 3A loads while battery 3A was load tested. The PC/M was issued to cover only the installation and did not address termination of the new cable. The design change was classified as quality related (i.e., important to safety) due to its proximity to safety-related components. The associated design analysis and safety evaluation considered: seismic supports, 3-hour fire seals at penetrations, related design changes in progress, ALARA, human factors, environmental qualification, cable type and ampacity, and a masonry wall penetration. In addition, the implementation instructions required that the unterminated cable coil be seismically restrained in DC Equipment Room 3A. The safety evaluation provided explicit bases for the conclusion that no unreviewed safety question was involved. The broad scope of interface review, considering the classification of this PC/M as nonsafety-related, was seen as a strength in the licensee's program.

Safety-related PC/M 87-346, which was issued as Design Equivalent Engineering Package (DEEP) 87-346 on October 28, 1987, provided a temporary support for the Unit 4 "C" Safety Injection Accumulator line so that the permanent support (SR-450) could be disassembled due to its interference with MOVATS testing of valve 4-865-C.

A DEEP was described in procedure JPE-QI-3.7, Design Equivalent Changes Performed by JPE, dated October 2, 1987. As defined in the procedure, a design equivalent change is a design change to a structure, system, component, part, design-related document, or design output document, that does not alter the plant's design basis, (i.e., functional, performance, operating and regulatory requirements), and is bounded by the existing design analyses.

Design equivalence was to be demonstrated by evaluating, as appropriate: form (the physical characteristics, design ratings, safety-related classification, code applicability, QA requirements, and NRC requirements associated with an item); fit (the mounting attachments, and the space that is occupied or required to support operation of an item); and function (the performance characteristics and range of operation of an item). The DEEP was a simplified, checklist-oriented PC/M, which appeared to be intended primarily for component substitutions, especially those resulting from lack of current availability of original equipment.

With regard to the temporary support, DEEP 87-346 contained a checklist for 10 CFR 50.59 considerations and an attached engineering evaluation. While the engineering evaluation did not specifically relate to the 10 CFR 50.59 concerns, the bases for conclusions can be inferred from the engineering information provided. Specifically, dead weight and seismic loading were considered and, most significant was a requirement that the original support be restored prior to heating up the plant from Mode 5. With regard to this restriction, the inspector confirmed, by review of work process sheets, that the temporary support had been removed and the permanent support restored prior to heatup.

The inspector also reviewed DEEP 87-327, which was issued October 1, 1987, to repair or replace twelve leaking seal table reducing unions. The modification consisted of replacing the unions with a different type that was one-half inch longer. The 10 CFR 50.59 safety evaluation was documented on a checklist and determined that no unreviewed safety question was involved. The attached engineering evaluation did cover such aspects as: pressure boundary integrity; seismic effect of the increased weight; design temperature, pressure, and material; and, potential need to shorten thimble tubes by one-half inch.

The design changes reviewed contained sufficient information to support conclusions made in the safety evaluations and the recent changes were found to be the more detailed of the evaluations reviewed.

3.3 Temporary Modifications

The inspectors reviewed licensee programs to control temporary system modifications such as jumpers and lifted leads to ensure that adequate controls were applied for review and approval consistent with 10 CFR 50.59. Procedure 0-ADM-503, Control and use of Temporary System Alterations, dated September 15, 1987, provided controls for Temporary System Alterations (TSAs) and Fire Protection Impairments (FPIs) to ensure Operations Department awareness of all changes and, to ensure that TSAs made to plant equipment did not unacceptably degrade the design intent. The procedure designated the following as TSAs: temporary wire, line, or hose used specifically to jumper, bypass, or otherwise alter normal system functions or equipment, and temporary connection such as hoses, tubing, or piping, which joins systems together, bypasses a component within a system, or removes components within a system, thus altering the system's design or configuration.

For each TSA, a multi-page form was completed, which identified the alteration and which provided a 50.59 safety evaluation. In many cases, a detailed narrative safety evaluation was completed by Power Plant Engineering (PPE) and attached to the TSA. For TSAs affecting in-service equipment, review by the Shift Technical Advisor (STA), Technical Department Supervisor, Plant Supervisor - Nuclear, and the Plant Nuclear Safety Committee (PNSC), was required prior to implementation of the TSA. The procedure allowed some specific exceptions to the process, such as tubing to floor drains and modifications installed and removed as part of a PNSC-approved procedure. TSA files were maintained by the STA in the control room. At the time of this inspection, approximately 30 TSAs per unit were in effect. Each STA was assigned a group of active TSAs in an effort to ensure that final resolution (e.g., design change) was actively pursued so that the TSA could be cleared. In addition, each STA conducted a quarterly audit of assigned TSAs to verify continued necessity, to confirm continued validity of the safety evaluation, and to ensure that identifying tags were still installed in the field.

The inspector reviewed the log of active TSAs and conducted a detailed review of five TSA packages. In each case, the inspector verified that the TSA had received prior PNSC approval and that the accompanying safety evaluation provided an adequate basis for the conclusion that no unreviewed safety question was involved as a result of the temporary modification.

The TSAs reviewed were:

TSA Number	Description
3-86-13-36	Replace instrument air compressors with diesel-driven units employing air-cooled after coolers.
3-86-13-50	Relocate diesel air compressors for instrument air and provide less restrictive flow path
3-87-71-67	Addition of pipe cap on Steam Generator 3A tube sheet drain line.
4-86-71-58	Install valve in Steam Generator 4C tube sheet drain line.
4-87-94-14	Install temporary isolation valve in post-accident hydrogen monitoring system train A to allow heat-up of plant during down stream piping repair.

TSAs 3-86-13-36 and 3-86-13-50 related to the installation of diesel driven instrument air compressors to reduce the electrical loads on station emergency diesel generators resulting from the installed motor driven compressors. The instrument air compressors were classified as non-nuclear, safety-related, and air-operated valves which were required to operate in an accident were equipped with local compressed gas bottles as a backup for instrument air. For each TSA, the attached safety evaluations considered interfacing aspects with safety-related and quality related (important to safety) systems, such as: seismic interactions, Appendix R, redundancy, loss of off-site power, service pressure, and capacity. Both safety evaluations concluded that no unreviewed safety question was involved and both provided a stated basis for this conclusion. The team noted that the more recent evaluation (TSA-86-13-50) was more comprehensive in providing this basis.

TSAs 3-87-71-67 and 4-86-71-58 both dealt with seat leakage through the two valves in the steam generator tube sheet drain lines. In order to arrest the leakage pending repair of the valves, TSA 3-87-71-67 added a pipe cap on the drain and TSA 4-86-71-58 installed an additional isolation valve in the drain line. The associated safety evaluations addressed the piping classification (non-safety related), the ISI code boundary, seismic calculations, service pressure, and materials. Both evaluations provided the basis for the conclusion that no unreviewed safety question was involved.

TSA 4-87-94-14 added a temporary manual globe valve to the Post Accident Hydrogen Monitoring System (PAHMS). The purpose of this valve was to retain containment integrity and allow plant heatup while repairing leaks in downstream portions of the system. An effect of this TSA was to render Train A of the PAHMS inoperable. The associated safety evaluation considered seismic effects, ASME Code, materials, and containment integrity requirements. A stated basis was provided for the conclusion that no unreviewed safety question was involved. However, the conclusion was constrained by additional restrictions, one of which was the requirement that the system be operable prior to entry into Mode 2 (Plant Startup). The inspector then reviewed licensee mechanisms to ensure that constraints included within safety evaluations were satisfied prior to making plant mode changes.

Plant Procedure 4-GOP-503, Cold Shutdown to Hot Standby, dated November 12, 1987 contained prerequisite step 3.1.3.9, which required the Plant Supervisor - Nuclear to review the TSA log and verify that outstanding alterations would not interfere with the safe operation of the plant. In addition, the licensee established a computer listing (Critical Path Report) which identified those activities required prior to key events such as heatup. Safety evaluation constraints were also included in this listing. Based on the records reviewed, the PAHMS was restored to operable status prior to unit startup.

The inspector also reviewed the following safety evaluations associated with temporary lead shielding:

Safety Evaluation Number	Date Issued
JPE-M-86-010 Rev. 0	2/10/86
JPE-M-87-021 Rev. 0	2/27/87
JPE-M-87-117 Rev. 0	11/5/87
JPE-M-87-117 Rev. 1	11/7/87

The temporary lead shielding involved was to be applied to the following pipe sections: RHR loop A suction piping, RHR and SI loop C suction piping, and pressurizer spray piping. The first of these evaluations (JPE-M-86-010) declared no basis for the conclusion that no unreviewed safety question was involved. However, a restriction was included, which stated that the conclusion was valid only while the reactor was defueled.

The second evaluation (JPE-M-87-021) provided the basis that no unreviewed safety question was involved because the system was not required while the reactor was defueled and added this plant condition as a restriction.

The third evaluation (JPE-M-87-117) provided the most information with regard to a basis for the conclusion. In addition to considering potential pipe failure and possible drain down of the pressurizer, the evaluation also addressed the potential effects of the temporary shielding becoming loose in a seismic event. The safety evaluation concluded that, since the shielding could not cause a pipe rupture with adverse impact on the plant, and since its loosening would not cause a problem, no unreviewed safety question was involved. The safety evaluation further contained a restriction that the conclusions were valid only while the plant remained in Mode 5.

These three essentially similar safety evaluations, which covered a 19 month period of time, demonstrated a steadily improving review and documentation process, which now provided a more coherent stand alone evaluation to provide the basis for conclusions relative to 10 CFR 50.59.

Overall, licensee control, evaluation, and approval of temporary modifications was found to be a strength.

4.0 Non-Design Change Safety Evaluations

4.1 Engineering Responses to Request for Assistance

On occasions, Power Plant Engineering was asked to perform safety evaluations in response to Requests for Engineering Assistance (REAs). A review of the REAs, requiring a safety evaluation, was performed by the inspector and three were selected for detailed review (REA TPN-86-058-012, REA TPN-86-047, and REA TPN-87-021).

A weakness was identified in the response to REA TPN-87-21. This REA was issued to reevaluate eight containment penetrations with respect to containment isolation and leak rate testing concerns. This reevaluation was to be completed prior to unit startup and the REA cited operability concerns. Because of other needs for engineering resources, a formal reevaluation was not accomplished prior to restart and still had not been performed before the team's departure from the site. Prior to changing modes, Power Plant Engineering performed an initial review of the penetrations in question and documented that review in a June 3, 1987 Inter-Office Correspondence. Instead of evaluating the individual concerns, Power Plant Engineering relied upon prior NRC acceptance of the design and the lack of indication of problems during NRC scrutiny of the Florida Power and Light Company (FP&L) Integrated Leak Rate Test program as the basis for delaying evaluation until after restart.

The containment isolation issues, raised in the REA, appear to be substantive issues identified by an experienced team with a specific goal and acceptance criteria. The team, known as the Plant Isolation Project, was composed of representatives from the following Turkey Point organizations: Technical Department, Operations, Safety Engineering Group, Site Power Plant Engineering, Procedure Upgrade, and Bechtel Design Basis Group. Since Power Plant Engineering's upgrade and review deferred action until after restart, it was essentially a justification for continued operation. The team noted that typically justifications for continued operations are prepared only after detailed reviews had been conducted to positively identify discrepant conditions with possible associated administrative limitations on operations. The treatment of this reevaluation by Power Plant Engineering appeared to be too superficial given the technical capability of the team that found the apparent discrepant conditions and the importance of containment isolation.

During the inspection, FP&L indicated that their review was more intensive but has not been documented. The NRC team was informed that the containment isolation concerns were not substantially different from the concerns reviewed previously for another reason and cited past, reviewed, and approved safety evaluations and Inter-Office correspondence as an acceptable basis for concluding that the plant could restart without completing a detailed reevaluation. The team was concerned that safety evaluations were not maintained current when the plant configuration changed and that no mechanism existed to identify which safety evaluations were current, which ones had been superseded, and which ones may have been accurate in the past but were now incorrect because of some subsequent change. For example, safety evaluation B-SFB-3033 [Operability of AFW Pump A at Reduced Performance, Revision 0, August 1, 1986] contained an assumption of a pump versus train configuration which was not consistent with the current Technical Specification (TS).

Because safety evaluations were not maintained current, like a design output document, they could be out-of-date and should not be relied upon as justification for current plant activities unless verified to be valid.

4.2 Review of Proposed Quality Instruction for Performance.of 10 CFR 50.59 Evaluations

Proposed Quality Instruction JPE-QI 3.8, 10 CFR Evaluations Performed By Power Plant Engineering, was found to be a worthwhile instruction and, when implemented, should improve safety evaluations and ensure consistency between evaluations. A weakness was found with the instruction's application of the single failure criterion to check valves. The same application was described in an approved quality instruction - Supplement QI 2.3-1, Safety Classification Reference Guide, Revision 0, February 19, 1986.

The weakness was the description that failure of a check valve to move to its correct position, when required, was a passive failure. This interpretation was contrary to industry practice, as first described in 1976 by ANS 51.7 - 1976, Single Failure Criteria for PWR Fluid Systems, and restated in 1981 by ANS 58.9 - 1981, Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems. Industry practice has been that check valves can fail in either an active or passive manner. The failure of a simple check valve to move to its correct position, when required, is an active failure. The failure of a check valve disk under pressure is an example of a passive failure.

The interpretation in the proposed quality standard was contrary to industry experience as described by the NRC in various Bulletins and Information Notices. Therefore, the team conducted a limited examination of containment isolation design provisions to determine if this interpretation was reflected in the plant's design. No single failure concerns were identified and the limited examination suggested that check valves had been treated as active devices in accordance with industry practice and experience. Therefore, this concern seems to be related to procedures and instructions rather than hardware. FP&L's current interpretation described in quality instructions may result in inadequate design changes, safety evaluations, or failure modes and effects analyses.

4.2 Engineering Procedures and Controls

During a limited review of design change packages and associated safety evaluations, a weakness was observed in FP&L's treatment of calculations and analyses. Specifically, Attachment A to Supplement QI 3.1-3, Revision 0, June 23, 1987 states that:

> Calculations need not be made an integral part of the engineering package but shall be retained by Power Plant Engineering for informational use only. Sufficient information shall, however, be included in the package to document the methods, assumptions, and results so that a competent engineer could, at a later date, independently verify the results, if required.

Safety Systems Functional Inspection (SSFI) and Safety Systems Outage Modifications Inspection (SSOMI) experience suggests that the lack of calculations filed in a central location are readily accessible to the design engineers and operators can result in incomplete and sometimes inadequate plant activities. A similar concern was experessed in NRC Inspection Report 85-32, the Turkey Point Unit 3 and 4 SSFI inspection report. During that inspection, the SSFI team found that design calculations were not being updated to reflect current modifications. At that time, the SSFI team was informed that design inputs were maintained so that, if required, the calculations could be recreated. Many of the design issues identified by the SSFI team were attributed to the lack of calculations. To continue the past practice of not maintaining design analyses appears to be unwise when compared to the efforts that are currently being used to capture and document the design bases of plant systems. The lack of readily available or retrievable calculations appears to be incompatible with a good configuration control program.

5.0 Review Committees

5.1 Company Nuclear Review Board (CNRB)

Due to time constraints and the interest in focusing on the activities conducted by the onsite review committee, the team abbreviated its review of the CNRB. The TS contained standard offsite review committee provisions for FP&L's CNRB, which conducted independent review activities for both the Turkey Point and St. Lucie nuclear stations. FP&L described the internal requirements for CNRB activities in a four page charter, Revision 8, dated August 1986. The charter was supplemented by a six page administrative guideline issued on the same date. The CNRB also instituted a standing subcommittee, its requirements described in a four page charter, Revision 1, dated June 1985, and supplemented by a 12 page guideline issued in September 1985. Both the CNRB and its standing subcommittee published minutes of their meetings, and the CNRB formally approved the standing subcommittee's minutes.

The team compared the requirements of the CNRB and standing subcommittee charters and guidelines and found the requirements to be essentially identical to each other and to the review responsibilities in TS section 6.5.2.7. Following reviews of CNRB and subcommittee minutes and interviews, the team concluded that the standing subcommittee performed virtually all the TS 6.2.5.7 review functions. When questioned, the licensee representative stated that the only review activity retained for the exclusive domain of the CNRB were TS amendments. The team noted that TS section 6.5.2.2 specified the CNRB composition by job title, and that these titles were those for very senior officials in FP&L's nuclear organization. The licensee representative stated that he understood that this TS provision had been imposed to force corporate nuclear managers to become more involved in station activities. Evidently, this approach was not entirely successful; and in the team's view, should not have been expected to be effective. In the team's experience, offsite review committees must principally perform their functions by reviewing documents. FP&L's use of a standing subcommittee to perform these functions is not prohibited by the TS, and their action has placed these review activities at a level in the organization where they can be thoroughly performed. In the team's view, there can be other means to ensure periodic onsite plant involvement by senior nuclear managers.

Because of the brief amount of time the team devoted to examining CNRB activities, the team was unable to make any overall conclusions about the performance of the CNRB.

5.2 Plant Nuclear Safety Committee (PNSC)

Inspection of PNSC consisted of discussions with licensee personnel, including PNSC members, review of documents, and attendance at the PNSC meeting held on December 8, 1987.

One of the events that prompted this inspection was PNSC Meeting No. 86-232 held on August 31, 1986. That meeting was conducted by means of individual telephone calls from the Shift Technical Advisor to each of the PNSC members. PNSC Meeting 86-232 was conducted to get concurrence from the PNSC to start up Unit 4 without first repairing a small, identified reactor coolant leak. The leak was through an instrumentation port column assembly conoseal fitting located on the reactor vessel head. The plant was subsequently restarted and operated at power. During an outage in March 1987, the licensee determined that a significant amount of boric acid from the leak had accumulated on the reactor vessel head. At that time, the licensee also determined that corrosion rates of materials, in contact with the boric acid, may have been greater than those on which the PNSC based the decision to startup the plant in August 1986. This event was documented in licensee Letter No. L-87-186, dated April 27, 1987, to the U.S. Nuclear Regulatory Commission. The consequences of this event indicated that the practice of obtaining PNSC concurrence by walking-around items or by serial telephone calls can result in less than adequate reviews.

The licensee revised Administrative Procedure (AP) 0110.4, "Plant Nuclear Safety Committee General Procedure," dated October 6, 1987, to clarify and tighten the requirements for holding PNSC meetings. The October 6, 1987 revision of AP 0110.4, did not allow walk-around PNSC concurrences. The procedure also required that telcon PNSC meetings were to be conference call type, with the members talking to each other. The following items could not be approved utilizing the telcon PNSC meeting:

- Any item which required a written safety evaluation for approval;
- Any item which involved a change in the FSAR or Technical Specifications;
- Any plant changes or modifications, controlled plant work orders, and process sheets.

One of the licensee personnel interviewed by the team was the Chairman of the PNSC, the Plant Manager-Nuclear. The Plant Manager-Nuclear did not chair many of the PNSC meetings; instead, the majority of the PNSC meetings were chaired by the vice chairman, the Operations Superintendent-Nuclear. The plant manager stated that he intentionally did not chair many of the meetings because by having the vice chairman conduct the meetings, a more independent review of PNSC items was obtained. He felt this practice was necessary because, as plant manager, he had to review each item and give final approval for it to be issued.

The team verified by review of licensee quality assurance documents and discussions with licensee quality assurance personnel, that the licensee's quality assurance department was monitoring PNSC activities. Two of the QA documents reviewed were QA audits QAO-PTN-86-723 and QAO-PTN-87-818. These annual audits were conducted to verify that the PNSC was meeting the requirements of TS 6.5.1 and the related plant implementing procedures. A third QA document reviewed was Corrective Action Request (CAR), Unit 4 Conoseal Leak, CAR-87-019. This

CAR reported noncompliances identified in a QA performance monitoring activity of events related to the Unit 4 conoseal leak. CAR-87-019 reported that PNSC Meeting 86-232 was conducted on August 31, 1986, by the shift technical advisor making individual telephone calls to each PNSC member, and that it was common practice to obtain PNSC concurrences in this way rather than by conference calls, as now specified by AP 0110.4.

The team made the following two observations during the inspection:

1. The licensee did not have a formal training program for PNSC members or their alternates. In 1984, the licensee held a one-day training session, which was taught by a contractor, for the PNSC members. In June 1987, the PNSC coordinator prepared a PNSC training guideline manual which contained material relative to the operation of the PNSC. This manual was a required reading type training course. These training manuals were sent to each of the PNSC members with a letter stating that the manual was to provide training for the PNSC members and their alternates. At the time of this NRC inspection, the PNSC coordinator had received documentation back from four of the members showing that they and their alternates had reviewed the material referenced in the training manual. At the exit meeting, the licensee stated that they were developing training programs for PNSC members and their alternates.

2. The second observation has to do with the large volume of material that is reviewed by the PNSC. The PNSC was meeting a minimum of twice a week with each meeting lasting over two hours. In addition to the two scheduled meetings every week, call meetings were frequently held. Through December 13, 1987, 334 PNSC meetings had been held during 1987. Discussions with licensee personnel indicated that the minimum of twice-weekly meetings had been held for a number of years. Because of the volume of material and the frequency of the meetings, the meeting agenda and package of items to be reviewed were not provided to the members until they arrived at the meeting. This practice did not give the members an opportunity to familiarize themselves with the items on the agenda. To help alleviate the problem of the large number of items to be reviewed, the PNSC has required that a sponsor be present at the meeting to present each item. The sponsor provided expertise to the PNSC on each item presented. Any item that did not have a sponsor present at the meeting was tabled. In addition, any item that any member questioned was sent back to the preparer to resolve these questions prior to PNSC approval.

This observation describes a condition that is common to many facilities with the traditional standard TS for their onsite review committee. This TS requirement, which applies to Turkey Point, requires that the PNSC review the types of procedures contained in Appendix A to Regulatory Guide 1.33. This appendix lists most of the types of procedures contained in a nuclear plant site's procedure file, and, as a result, the number of procedures subject to review, can be several thousands. This large review task is further compounded by the fact that many of these procedures could undergo changes several times a year. Interviews with PNSC members revealed the not surprising statistic that the vast majority of PNSC meeting time was devoted to procedure and procedure change review.

In addition to the requirement to review procedures and procedure changes, TS 6.5.1.6.f requires a review of facility operations to detect potential safety hazards. When PNSC members were asked how they conducted such a review they were initially unable to respond. After some thought, one member suggested that such a review was accomplished by the review of Licensee Event Reports (LERs). However, the team pointed out that review of LERs was more explicitly covered by TS 6.1.5.6.k, which required review of all reportable events. The team also asked PNSC members whether they had ever been requested to perform special reviews and investigations by the CNRB. These reviews are a TS requirement in 6.5.1.6.g. None of the PNSC members interviewed could recall conducting such a review. Admittedly, both of these TS requirements are very broad and the absence of their reviews by the PNSC would not, in and of itself, constitute a violation of TS. However, the team questioned PNSC members as to whether they had ever made those review requirements meeting agenda items. In the team's view, the PNSC is so burdened with the extensive review of routine procedure changes that it is effectively unable to devote time to reflect upon broad safety issues. This issue is compounded by the fact that PNSC members are key station managers who must divide their time among PNSC activities, plant tours, and supervision of their departments. The team considers that these key people could spend their time more effectively if they could develop alternative means for independent review of many of the station procedures. Certain types of key procedures, such as emergency operating procedures and administrative procedures, could be left for full PNSC review, while other procedures could be reviewed by another process. During the exit meeting the licensee indicated that they would consider requesting a TS amendment and instituting appropriate administrative control changes that would allow them to reduce the procedure review workload on the PNSC.

Based on this limited inspection, the team determined the following:

- The PNSC was conducting adequate reviews.
- Adequate administrative procedures had been implemented to control PNSC activities so that TS requirements were being met.
- The QA department had performed monitoring activities and audits of PNSC activities.

6.0 Exit Meeting

The inspection team conducted an exit meeting on December 11, 1987, to provide a summary of issues identified during the inspection. The licensee's representatives at the exit meeting are identified in Attachment A. The scope of the inspection was discussed, the observations were presented for each area inspected, and team members responded to questions from the licensee representatives.

ATTACHMENT A - PERSONS CONTACTED

T. Abbatiello	QA Supervisor
J. Anderson	QA Supervisor
*J. Arias, Jr.	Regulatory and Compliance Supervisor
*C. Baker	Plant Manager
*W. Bladow	QA Superintendent
*M. Bowskill	STA Lead Engineer
*J. Brannin	CNRB Standing Subcommittee Chairman
*J. Donis	Site Engineering Supervisor
*R. Earl	QC Supervisor
J. Evans	Document Control Supervisor
F. Flugger	Manager, Plant Engineering Licensing
*S. Franzone	Licensing Engineer
*R. Hart	Licensing Engineer
*P. Higgins	Site Engineering - Licensing Lead
*F. Houtz	PNSC Coordinator
*J. Kappes	Maintenance Superintendent - Nuclear
*J. Labarraque	Technical Department Supervisor
*S. Mathavan	Fuel Resources
*W. Miller	Senior Technical Advisor
*K. Needham	CNRB - Executive Administrator
J. Odom	Vice President - Turkey Point
L. Pabst	Manager, Mechanical Nuclear Section
*L. Pearce	Operations Superintendent
*F. Southworth	Senior Technical Advisor
*D. Taylor	Operations Systems Enhancement Coordinator
*D. Tseng	Senior Plant Engineer
D. Tomaszewski	Instrument and Control Supervisor

*Attended Exit Meeting on December 11, 1987