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Licensee: Florida Power and Light Company

Facility: Turkey Point Nuclear Plant, Units 3 & 4

Location: 9760 S. W. 344 Street
Florida City, FL 33035

Dates: July 25 - September 4, 1999

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Enclosure

EXECUTIVE SUMMARY

Turkey Point Nuclear Plant, Units 3 & 4
NRC Inspection Report 50-250/99-05, 50-251/99-05

This integrated inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 6-week period of resident inspection; in addition, it includes the results of inspections by a regional radiation specialist and a regional engineering inspector.

Operations

- An operator opened the wrong circuit breaker during execution of a clearance order and subsequently a second operator incorrectly independently verified the breaker position. A non-cited violation was identified for an inadequate clearance execution and independent verification (Section O1.2).

- The licensee maintained an operator aids program and procedure. Several operator aids had been changed to permanent information postings. No permanent information postings were found outdated or incorrect (Section O1.3).

- The control room operators responded to a loss of normal control room lighting and other indications in accordance with procedures and the recovery was well coordinated (Section O2.1).

Maintenance

- The licensee was satisfactorily pursuing corrective actions and addressing radiator coolant leakage on the Unit 3 diesel generators. A proposed license amendment for an allowed outage time extension to replace both diesel generator radiators has been submitted to the NRC (Section M1.1).

- There were very few primary coolant leaks outside containment and most were inactive. All active leaks in the plant were aggressively tracked, prioritized and scheduled for work. In general, plant work orders were initiated for identified leaks, however several minor, inactive leaks did not have a work order. Overall, the licensee was effective in minimizing leakage on the involved systems. Integrated system leak tests were being performed as required and measured leakage was well within the safety analysis assumptions (Section M1.2).

- The FIN team was accomplishing its activities effectively and in compliance with procedural requirements (Section M6.1).

Engineering

- The Chemical Volume and Control system was well maintained. Valve and breaker line-ups were verified to meet Technical Specification requirements. Strong engineering field support during maintenance activities was observed (Section E1.1).

- The 25th containment tendon surveillance was performed in accordance with NRC requirements. The containment post-tensioning system meets design requirements (Section E2.1).

- The licensee has maintained the containment temperature within the Technical Specification limits and design criteria. An inspector follow-up item was identified to further evaluate the acceptability of the concrete temperature at the interface between the reactor structural supports and the shield wall and the long term effects on the concrete (Section E2.2).

Plant Support

- The licensee was effectively labeling, controlling, and storing radioactive material as required by regulations (Section R1.1).

- The licensee had effectively implemented a program for shipping radioactive materials required by NRC and DOT regulations (Section R1.2).

- The environmental air sampling equipment was being maintained in an operational condition to comply with TS requirements and UFSAR commitments (Section R2.1).

- The inspectors determined the licensee's most recent formal Quality Assurance Audit effectively assessed the areas of environmental sampling and transportation of radioactive material (Section R7.1).

Report Details

Summary of Plant Status

Both units operated at full power this period. Unit 3 has been online since June 24, 1999. Unit 4 has been online since April 18, 1999.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety-conscious; specific events and noteworthy observations are detailed in the sections below. Operators were knowledgeable and attentive. Traffic in the control room was appropriately controlled with only essential personnel permitted access. Main control board deficiencies and alarms were minimal.

O1.2 Clearance Execution and Independent Verification

a. Inspection Scope (71707)

The inspectors reviewed a clearance execution and independent verification issue on the #1 Service Air Compressor.

b. Observations and Findings

On August 24, 1999, during execution of a clearance for preventive maintenance of the #1 Service Air Compressor (SAC), an operator opened the circuit breaker for the #2 SAC instead of the specified breaker for the #1 SAC. Subsequently, a second operator performing the independent verification of the breaker position did not detect this error. Prior to starting work on the #1 SAC and unrelated to that clearance, the work controls supervisors inquired whether the #2 SAC was operating properly because it had been earlier released on a separate clearance. Electrical maintenance found that the #2 SAC was not running and found the #2 SAC breaker open with the #1 SAC tag attached. Operations ordered a stop work and condition report (CR) 99-1233 was initiated with a severity level one and required a root cause analysis. Operations concluded the root cause was due to inadequate self-checking by both the operator executing the clearance and the operator providing the independent verification.

The licensee found that during initial development of the clearance order, operators discussed the location of the power supply to the #1 SAC and mistakenly concluded that it was located in the 3E load center. The #1 SAC is actually powered from the 3G load center. These two load centers are not located near each other and are at different locations in the plant. In reviewing how both barriers failed, the operators added that the clearance executer and the independent verifier traveled together to the load center. The independent verifier waited outside the 3E load center room while the clearance

executer opened the breaker. The independent verifier subsequently entered the load center room and incorrectly verified the breaker position for the #1 SAC.

The inspectors reviewed the completed CR and discussed the results with the licensee. The inspectors reviewed the independent verification procedure and the plant equipment clearance orders procedures. The inspectors interviewed the operators that executed the clearance and the independent verification. Additionally, the inspectors walked down the location of the #1 SAC breaker and the #2 SAC breaker. Inspection of the 3E load center indicated the breaker for the #2 SAC was correctly labeled. The inspectors verified that the clearance order for the #1 SAC specified the correct breaker number.

Technical Specification (TS) 6.8.1 requires that written procedures shall be established, implemented, and maintained covering the activities referenced in Appendix A of Regulatory Guide 1.33, Revision 2, dated February 1978. Section 1.c of Regulatory Guide 1.33 specifies procedures for equipment control (e.g., locking and tagging). Section 5.11 of 0-ADM-212, In-Plant Equipment Clearance Orders, specifies the requirements for execution of clearances and specifies the clearance alignment and clearance tag placement shall be performed in a step by step manner as specified on the clearance order. Section 5.0 of 0-ADM-031, Independent Verification, specifies the method to verify electrical alignments by visual inspection of the specific breaker position. Contrary to these requirements, an operator failed to adequately execute a clearance order and a second operator performed an inadequate independent verification. This Severity Level IV violation is being treated as a non-cited violation, consistent with Appendix C of the NRC Enforcement Policy. This is identified as NCV 50-250, 251/99-05-01, Inadequate Execution of a Clearance and Independent Verification. This violation is in the licensee's corrective action program as CR 99-1233.

c. Conclusions

An operator opened the wrong circuit breaker during execution of a clearance order and a second operator incorrectly independently verified the breaker position. A non-cited violation was identified for inadequate clearance execution and independent verification.

O1.3 Operator Aids (71707)

a. Inspection Scope

The inspectors reviewed the use and control of operator aids using administrative procedure 0-ADM-209, Equipment Tagging and Labeling.

b. Observations and Findings

The definition of an operator aid, according to procedure 0-ADM-209, is "information including sketches, notes, graphs, instructions, drawings, and other documents used to assist operators in performing assigned duties." The inspectors reviewed the operator aid index and noted that it did not identify any current operator aids being in use. When walking down the main control room (MCR) panels and other areas of the plant, the

inspectors had noted several instances where a placard was in place to provide the operators with information or instructions. Examples included pressure-temperature limits curve in the MCR and alternate shutdown panel (ASP) room, flow diagram for reactor head vent system in the MCR, and DB50 reactor trip breakers operation. The Operations supervisor advised the inspectors that these operator aids had been changed to permanent information postings and therefore now fell under the definition of permanent information. Permanent information is also addressed in 0-ADM-209. The definition of permanent information was "information that appears on a medium not suitable to change and determined by the Operations Supervisor to be applicable indefinitely." Additionally, as described in the procedure, a temporary information tag or an operator aid may become permanent information after being in place for more than six months. The inspectors noted that there is no audit or review process to address permanent information and questioned the licensee how the information is assured to be current. Operations management indicated that any changes required to permanent information placards would be captured via the plant change modification process. The inspectors reviewed selected portions of the permanent information and did not identify any outdated or incorrect information.

c. Conclusion

The licensee maintained an operator aids program and procedure. Several operator aids had been changed to permanent information postings. No permanent information postings were found outdated or incorrect.

O2 Operational Status of Facilities and Equipment

O2.1 Loss of Normal Control Room Lighting (71707)

On July 28, 1999, the inspectors observed a loss of normal plant lighting while in the plant computer room and responded to the control room. A loss of D non-vital 480 volt Motor Control Center (MCC) had occurred. A welding machine located in the Unit 4 charging pump room shorted out causing a trip of a receptacle breaker and feeder breaker to the MCC. This resulted in a loss of normal control room lighting and other lighting in the plant. Control room lighting switched over to the direct current backup source. Indications of digital megawatts and turbine valve position were lost. The plant operators observed that the plant was stable and no transient had occurred.

Plant management responded to the control room along with additional personnel. Normal lighting was restored within an hour. CR 99-1119 was written to enter this problem into the corrective action program. A recent industry experience discussed a problem with a similar type welding machine. The licensee has restricted use of this type of welder until the concern is resolved.

The inspectors reviewed the plant response with the operators and engineering. Applicable drawings and procedures were reviewed. The plant response was as expected for the loss of a non-vital MCC and the immediate corrective actions were well coordinated.

O2.2 Periodic Verification of Containment Isolation Lineup (71707)

The Unit 4 containment isolation status was inspected during a walkdown of accessible containment isolation valves (CIVs), pipe penetrations, and applicable power supplies. Plant procedures and the Updated Final Safety Analysis Report (UFSAR) were reviewed and compared against control room indications to verify that the containment isolation valves were properly aligned. The material condition of CIVs and pipe penetrations was acceptable, associated pipe hangers were in place, and proper labeling and locking of valves was evident. Power supplies were properly aligned and in good condition. Several minor discrepancies were identified as a result of this review which have been addressed through the licensee's corrective action program.

O4 Operator Knowledge and Performance

O4.1 Senior Nuclear Plant Operator (SNPO) Tour (71707)

On August 5, 1999, an inspector accompanied the SNPO responsible for the Reactor Auxiliary Building on a routine plant tour. The SNPO was knowledgeable of the systems and component operation in his areas of responsibility. The operator took the time to thoroughly examine conditions in order to detect abnormalities or changes in equipment operation. When equipment deficiencies were identified, appropriate actions were taken by the SNPO to address the problem. Overall, the SNPO conducted a thorough tour, assessing equipment status and identifying equipment and housekeeping deficiencies to implement corrective actions.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Work Order and Surveillance Observations

a. Inspection Scope (61726) (62707)

The inspectors observed surveillance and maintenance activities as follows:

3-OSP-023.1	Diesel Generator Operability Test
WO 99013506	Repair Radiator Leak
3-OSP-075.6	Auxiliary Feedwater Train 1 Backup Nitrogen Test

b. Observations and Findings

On July 23, 1999, during observation of the 3B Diesel Generator (DG) operability test, the inspectors identified a small leak on the DG radiator. The leak rate was approximately five drops per minute. Operations immediately initiated actions to address the operability of the DG. Engineering wrote a 3-day operability assessment and

concluded that the diesel was operable. CR 99-1102 was written to enter this issue into the corrective action program. The licensee determined that the leakage was coming from the radiator gasket. The inspectors observed that once the radiator heated up to temperature, the leakage stopped. The licensee wrote a work order to replace and tighten the bolts in the area of the radiator leak. The inspectors later verified that the radiator was not leaking in a standby condition.

Similar radiator gasket leakage had occurred on the 3A and 3B DGs during surveillance runs in April of 1998. CR 98-415 had been written to address that issue. Additionally, as described in NRC Inspection Reports 50-250, 251/98-11 and 98-12, the Unit 3 DG radiators had also experienced through-wall radiator tube leakage. The licensee had subsequently planned to replace the Unit 3A and 3B diesel radiators. However, the 3-day action statement for DG inoperability is not sufficient time to allow for the replacement of each radiator. Engineering had concluded that a seven-day outage was required for each radiator replacement. The licensee subsequently submitted a proposal to the NRC for an allowed outage time extension. The licensee proposed to replace the Unit 3 radiators after the 1999 hurricane season and prior to the 2000 spring refueling outage.

The inspectors observed testing of the 3B DG on August 18, 1999, following replacement of radiator bolts in the area that leaked on July 23, 1999. Independent verification steps were properly completed. The inspectors verified post maintenance testing requirements were completed.

c. Conclusions

The licensee was satisfactorily pursuing corrective actions and addressing radiator coolant leakage on the Unit 3 diesel generators. A proposed license amendment for an allowed outage time extension to replace both diesel generator radiators has been submitted to the NRC.

M1.2 Integrated Leak Testing Of Systems Outside Containment

a. Inspection Scope (61726)

The inspector reviewed implementation of the licensee's program to reduce leakage from primary coolant sources outside containment as required by TS 6.8.4.a. The inspector reviewed applicable test results, interviewed personnel responsible for the program, and verified the system testing conformed with the TS requirements. The inspectors also inspected portions of affected plant systems to assess program effectiveness.

b. Observations and Findings

The TS 6.8.4.a program applies to systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The systems specifically listed in TS 6.8.4.a are safety injection, chemical volume and control, and containment spray. The licensee's program includes not only the required systems but also several other

systems that could contain highly radioactive fluids, including the normal and post-accident sampling systems.

TS 6.8.4.a requires reducing applicable system leaks to "as low as practical levels." During tours of the affected systems, the inspector observed very few leaks, and almost all of them appeared to be inactive, with the only indication being accumulation of dry boric acid. The inspector noted that all active leaks were aggressively tracked, prioritized, and scheduled for work. These were also regularly discussed in the morning work planning meetings. Although plant work orders (PWO) appeared to be routinely written for the majority of leaks, the inspector identified several minor inactive leaks that had not been addressed by a PWO. The most notable was an inactive leak on the 3B residual heat removal pump recirculating valve. The inspector expressed concern that this could be an active leak during system operation and should be assessed. All of the observed leaks were reported to the Nuclear Plant Supervisor, who promptly wrote PWOs for each one. Overall, the licensee was effective in minimizing leakage on a routine basis per TS 6.8.4.a(1).

Integrated leak tests were required to be performed "for each system at refueling cycle intervals or less" as prescribed by TS 6.8.4.a(2). Based on a review of 0-OSP-207.2, Visual Leak Inspection of Systems Outside Containment, and the results documented on applicable procedural attachments, the inspector concluded that the integrated leak tests were being performed and measured leakage was well within UFSAR safety analysis assumptions.

The inspector also reviewed historical records for tracking system leak test performance frequency. The licensee was controlling the scheduling of the leak testing by establishing an 18 month periodicity and allowing the TS 4.02 extension of 25 percent of the interval to be applied. The inspectors noted that the extension was applied to a significant number of the completed tests. Discussion with Nuclear Reactor Regulation staff indicated that it was not intended that licensees apply TS 4.02 to testing intervals contained in the administrative section of the TS. After a detailed review of completed testing records, the inspector concluded that, in general, the testing had been completed within the TS required periodicity of each refueling cycle interval. Out of 36 completed tests, only two were not completed within the refueling cycle interval. Those tests were completed within several days of the required interval with acceptable results. This failure to meet TS 6.8.4 due to procedural deficiencies constitutes a violation of minor significance and is not subject to formal enforcement action. The licensee entered the test scheduling issues into the corrective action program as Condition Report 99-1217.

c. Conclusions

There were very few primary coolant leaks outside containment and most were inactive. All active leaks in the plant were aggressively tracked, prioritized and scheduled for work. In general, plant work orders were initiated for identified leaks, however several minor, inactive leaks did not have a work order. Overall, the licensee was effective in minimizing leakage on the involved systems. Integrated system leak tests were being

performed as required and measured leakage was well within the safety analysis assumptions.

M6 Maintenance Organization and Administration

M6.1 Fix-It-Now Team

a. Inspection Scope (62707)

The inspectors reviewed activities of a recently formed Fix-It-Now (FIN) Team.

b. Observations and Findings

Earlier this year, the licensee established a multi-disciplined maintenance team that was tasked with controlling and correcting emergent work activities with a rapid response. This was intended to provide schedule protection, increase production, and improve maintenance efficiency. The FIN Team's first priority has been to deal with items identified as emergent work including, but not limited to, industrial safety conditions, control room green tags, and load threatening conditions. Other priorities include validating new work orders and work order backlog reduction. Work scope limitations have been placed on the FIN Team such as; work should normally be completed within one shift; FIN Team backlog should be maintained to a maximum volume of one week's worth of work; no ASME Section XI Repair and Replacement Plan welding should be performed by the Team; work should be accomplished with the resources assigned to the Team, and no environmental qualification work should be performed by the Team. The inspectors also interviewed the FIN Team leader and several FIN Team members. The Team was well integrated into the work control process. Single points of contact were established within organizations, such as Operations and Engineering, for interactions with the FIN team.

The inspectors reviewed several maintenance work items and work orders with the FIN team supervisor and members. The supervisor was at all times well informed with the ongoing daily FIN team work. Interviews with several FIN team members indicated that members were well versed with prerequisites and requirements of the maintenance work orders.

c. Conclusions

The FIN team was accomplishing its activities effectively and in compliance with procedural requirements.

III. Engineering

E1 Conduct of Engineering

E1.1 Chemical and Volume Control System

a. Inspection Scope (71707, 37551)

The inspectors performed a detailed system walk down on the Chemical Volume Control System.

b. Observations and Findings

According to the licensee's Probabilistic Safety Assessment, the Chemical and Volume Control System (CVCS) is the second most risk significant system at Turkey Point. Portions of the systems are shared and other portions can be cross connected between the units. The inspectors performed a detailed system walk down of the CVCS on Unit 3 and Unit 4. Recently completed surveillance procedures were reviewed for completeness and TS compliance. The inspectors independently reviewed valve and breaker positions for the boration system flow path and verified it was consistent with surveillance procedures and system prints. One discrepancy was identified. Valve 379, located on the discharge from the 3B boric acid transfer pump, was identified on the print as being locked open. The inspectors found there was no lock on the valve. Engineering subsequently reviewed the issue and concluded that the print was in error. A change to the print was initiated to remove the requirement to lock the valve. Procedural requirements correctly depicted the valve position. The inspectors walked down the Unit 3 and Unit 4 charging pump system with the responsible system engineer. Additionally, ongoing maintenance work performed on the CVCS during the inspection period was reviewed with several system engineers. Good system engineering field support was noted during execution of maintenance activities. Equipment operability, material condition, and housekeeping were acceptable. The inspectors identified no substantial concerns.

c. Conclusions

The Chemical Volume and Control system was well maintained. Valve and breaker line-ups were verified to meet Technical Specification requirements. Strong engineering field support during maintenance activities was observed.

E2 Engineering Support of Facilities and Equipment

E2.1 Containment Tendon Surveillance Inspection

a. Inspection Scope (37550)

The inspectors reviewed the results of the 25th year containment (reactor building) tendon surveillance inspection.

b. Observations and Findings

The inspectors reviewed Engineering Evaluation PTN-ENG-SECS-97-011, Unit 3 and Unit 4 25th Year Containment Tendon Surveillance Final Report. This report summarizes the results of the tendon surveillance inspections performed in accordance with Technical Specification 3/4.6.1.6, the Updated Final Safety Analysis Report (UFSAR), and Regulatory Guides 1.35 and 1.35.1. The tendon surveillance was performed on 12 randomly selected tendons from each unit and included: visual inspection of the tendon anchors and buttonheads, sampling and testing of the tendon corrosion protection material (grease), and determination of the prestress force in the tendons. The results showed the anchors were acceptable, the grease was free of contaminants, and the tendon prestress forces exceeded the predicted lower limits (PLL). The PLL was calculated using the results of the containment structure reanalysis which was completed after the 20th year tendon surveillance. In addition, three tendons (one horizontal, one vertical, and one dome) were also detensioned. A sample wire was removed from the detensioned tendons, visually inspected for presence of corrosion, and subjected to tensile testing. The tendons were then retensioned. Elongation of the tendons was measured during retensioning. The elongation measurements for two tendons exceeded the original installation elongation of the five percent limit. This condition was investigated and determined to be acceptable. The tendon surveillance results demonstrated that the containment post-tensioning system is performing in accordance with design requirements.

c. Conclusions

The 25th containment tendon surveillance was performed in accordance with NRC requirements. The containment post-tensioning system meets design requirements.

E2.2 Containment Temperature Issues

a. Inspection Scope

The inspectors reviewed containment temperature data and design criteria and regulatory requirements which address temperatures in the containment building.

b. Observations and Findings

Temperatures in the containment are monitored by 24 thermocouples installed at various locations in the containment building and by three resistance temperature detectors (RTDs) installed at azimuths 0, 120, and 360 degrees at elevation 58 on the containment wall. The RTDs are used as the primary instruments to monitor containment temperature as required by Technical Specification 3/4.6.1.5. The TS requires that the containment temperature be determined at least once per 24 hours using the arithmetical average of the three RTD temperatures. The TS acceptance criteria is a maximum temperature of 125°F, not exceeding 120° F for more than 336 hours per year. Data from these instruments are displayed on the Emergency Response Data Acquisition and Display System (ERDADS). In the event one or more RTDs are out of service, thermocouples installed near each RTD location provide a backup to the RTDs. Data from the thermocouples are registered on a recorder. The licensee has a program to periodically check the operation of the RTDs through their instrument calibration program. The thermocouples used to monitor containment temperature are not included in a calibration program, although the recorder, R-1413, is periodically calibrated.

The inspectors reviewed the following documents which addressed containment temperature problems:

- Bechtel letter number SFB-485, dated July 22, 1982, Subject: Evaluation of Increased Ambient Temperatures, Unit 3 Containment.

- Bechtel letter number SFB-607, dated October 29, 1982, Subject: Evaluation of Increased Containment Temperatures, Unit 3.

- Condition Reports (CRs) 96-1536, 96-1539, and CR 99-1178

Review of the Bechtel letters disclosed that higher temperatures had been noted inside the Unit 3 containment in 1982 following replacement of the steam generators. A detailed study was performed by Bechtel which identified several causes for the temperature increases. These included improper installation of the new reflective insulation, poor maintenance and operation of the component cooling water (CCW) and containment ventilation systems, and missing insulation on some secondary piping systems. These problems were addressed by correcting the installation problems with the insulation and improvements to maintenance and operation of the CCW and containment ventilation systems. The inspectors reviewed containment temperature data for 1974, 1980, 1982, and 1999. This data showed that containment temperatures had increased in 1982 by approximately 15°F above the temperatures previously recorded. It was estimated that the higher temperatures occurred over approximately a one year period. Review of the 1999 temperature data show that the current containment temperatures are approximately the same as they were in 1974 and 1980, and lower than the temperatures observed in 1982. In 1991, a modification to the normal containment ventilation system was also implemented which improved efficiency of the system. This modification resulted in removing additional heat from the containment and further decreased the containment air temperature.

Review of licensee data indicated that the effect of the previously elevated temperatures on equipment qualification were evaluated. This evaluation was documented in a report titled "Evaluation of Increased Containment Temperatures," which was an attachment to Bechtel letter SFB-607. The conclusions of the evaluation were that the increased containment temperatures had no near term effect on the equipment, but that equipment replacement schedules would require revision due to the higher temperatures.

Discussions with licensee engineers in the License Renewal Project disclosed that instrumentation was installed in Unit 3 during the last refueling outage to record temperatures in the proximity of electrical equipment and instrumentation. This data will be used to evaluate the qualified life of the equipment, and to establish appropriate equipment replacement schedules. Similar instrumentation is scheduled to be installed in Unit 4 during the next Unit 4 refueling outage.

In cases where high concrete temperatures occur, the aging effects on the concrete include reduction in compressive strength, changes in Poisson's ratio, modulus of elasticity, and bulk modulus. The requirements for consideration of effects of temperature on concrete are addressed in ACI 349, Code Requirements for Nuclear Safety Related Concrete Structures. Temperature limitations specified for normal operation are 150°F except for local areas, such as around a penetration which are allowed to have increased temperatures not to exceed 200°F. For accident conditions or other short term periods, temperatures are permitted up to 350°F for concrete surfaces. Local areas are allowed to reach 650°F from steam or water jets in the event of a pipe failure.

Bechtel Design Criteria 5177-C-000, Appendix A, Revision 0, considered the effects of temperature on concrete under Section 7.0, titled Heat Transfer. The criteria stated that loss of strength and stresses are of concern whenever the temperature of concrete is higher than 150°F. Section 7.1 of the criteria stated the following: In all areas where steel bearing or other embedded items have a contact temperature with the concrete above 150°F, high strength concrete will be specified to allow for both loss of strength and high stresses. Section 7.2 of the criteria addressed thermal stresses resulting from temperature gradients across the primary shield wall due to heat from neutron and gamma radiation, and heat convection from the reactor vessel. Heat generated by attenuation of neutron and gamma radiation was calculated to peak at 150°F at a distance of 16 inches from the inner face of the wall. The gradient was shown in Figure V of Appendix A of Criteria 5177-C-000. Cooling coils were installed in the reactor shield wall, at a depth of approximately 16 inches from the inside surface (reactor side) of the wall. The cooling coils were cooled by component cooling water. The gradient was calculated based on the assumption that the cooling coils would not be operating.

The design criteria also stated that heat resulting from convection from the reactor vessel would result in a linear temperature gradient. However the criteria also stated that a gradient does not exist because this heat is continuously removed by the containment ventilation system. Review of containment temperature data showed that the maximum air temperatures measured in containment occur just above the reactor vessel at the

intake to the control rod drive mechanism (CRDM) ventilation coolers. The purpose of the CRDM ventilation system is to remove heat radiated and conducted from the reactor vessel from around the CRDMs and provide cooling for the CRDMs. The normal recorded temperatures for the CRDM inlet was generally a maximum of 150 to 152°F. With the exception for the year 1982, the air temperatures measured in other areas in the containment generally were a maximum of 115 to 118°F. The temperature data were measured at several locations including around the reactor coolant pumps. The data indicated that the containment ventilation provided good air circulation and maintained air temperatures well below 150°F. The inspectors concluded that the containment ventilation system effectively removes the heat radiated by the reactor from the reactor cavity.

The cooling coils in the reactor primary shield wall had been isolated due to leaks in the component cooling system. Removal of the connection of the CCW piping to the reactor shield cooling coils was completed in the late 1980's under plant modifications PCM 86-068 for Unit 4 and PCM 86-162 for Unit 3. Review of the safety evaluations for these modifications disclosed that the justification for abandonment of the shield wall cooling system was as stated in Design Criteria 5177-C-000. That is: (1) the cooling was not required to maintain concrete temperature below 150° F due to heat radiated from the reactor vessel; (2) cooling coils were not required to maintain concrete temperature below 150° F due to heat generated by attenuation of radiation; and (3) in areas where steel bearing plates or other embedded item have a contact temperature above 150° F, high strength concrete was specified to allow for loss of strength. Review of the containment temperature data and design criteria showed that justification numbers (1) and (2) were sufficiently documented and substantiated. Review of Bechtel drawing number 5610-C-191, Revision 2, Containment Structure Reactor Shield Wall, showed that the specified concrete strength was 3000 psi unconfined compressive strength. High strength concrete was not specified to be used around the embedded reactor vessel supports. The licensee was questioned regarding whether they had any measured temperature data or design evaluations to show that the temperatures around the embedded supports were less than 150° F. These discussions indicated that the temperatures of the reactor supports at their point of contact with the concrete had not been specifically measured or evaluated. The inspectors questioned the adequacy of the safety evaluation since high strength concrete was not placed at contact points between the steel supports and shield wall concrete, and the licensee had no data to demonstrate the temperature was less than 150° F. An inspector follow-up item (IFI) was identified by the NRC to further evaluate the acceptability of the concrete temperature at the interface between the reactor structural steel supports and the shield wall and the long term effects. This IFI was identified to the licensee as IFI 50-250, 251/99-05-02, Evaluate Acceptability of Concrete Temperature at Interface Between Reactor Structural Steel Supports and Shield Wall and Long-Term Effects on Concrete.

As stated above, the thermocouples used to measure containment air temperatures are not included in a routine instrument calibration program. Review of some records provided indications of discrepancies with recorded temperatures. Some examples were as follows:

- Bechtel letter numbers SFB-485 and SFB-607 documented the increase in Unit 3 containment temperatures. During evaluation of the increased temperatures, actual temperatures were measured in the proximity of thermocouples using calibrated thermometers. The thermometer data showed that some of the points reading out on the recorder were transposed due to incorrect wiring and that the recorder needed to be adjusted due to temperature data registered on the recorders reading 18° F higher than actual field measurements indicated.

- CR numbers 96-1536 and 96-1539 documented a containment temperature reading issue. A thermocouple was being utilized to measure containment air temperature in place of an RTD which was out of service. Further review of this issue resulted in licensee concluding that thermocouple temperature had a larger degree of uncertainty in measured temperatures. An allowance for the larger degree of inaccuracy, minus 2° F, was incorporated into FP&L procedure numbers 3-OSP-201.1 and 4-OSP-201.1, RCO Daily Logs.

- CR 99-1178 documented a discrepancy with temperatures recorded at the air intake to the normal containment coolers A through D in Unit 3 which indicated the air temperatures in these areas were approximately 9 to 10° F higher than the temperatures recorded by the RTDs or by other thermocouples at other locations in the proximity of the coolers. Discussions with I&C personnel disclosed that most likely cause of this problem was poor connections in junction boxes due to dissimilar materials between thermocouple cables, terminal blocks, and other lead wires.

Based on review of the above problems, the inspectors concluded that the discrepancies were not the result of thermocouple calibration errors, but were caused by either the result of wiring problems or errors in the instruments (recorders) which provide the temperature output data from the thermocouples. The inspectors determined that the thermocouples, which operate at the low temperature ranges such as those measuring containment air temperatures, do not require periodic calibration. These type of thermocouples either function properly, or they fail.

Another issue occurred with the feedwater thermocouples (temperature elements) after completion of the power uprate project. This was the result of increased steam temperatures which damaged the insulation on the thermocouple lead wires. The problem was corrected by replacing the wires. The data from these instruments are utilized in plant calorimetric calculations and the tolerance for these thermocouples is plus or minus 1°F. The instruments are controlled under FP&L procedure PMI-074.19, Calorimetric Instrumentation Periodic Calibration, with a normal temperature range of up to 500°F. The calibration is performed by comparison with a controlled platinum RTD. This is similar to comparisons which are made between containment air temperature thermocouple and RTDs.

c. Conclusions

The licensee has maintained the containment temperature within the Technical Specification limits and design criteria. An inspector follow-up item was identified to further evaluate the acceptability of the concrete temperature at the interface between the reactor structural supports and the shield wall and the long term effects on the concrete.

E8 Miscellaneous Engineering Issues (92902)

E8.1 Year 2000 (Y2K) Readiness of Health Physics Information System (TI 2515/141)

The inspector conducted a review of Y2K documentation associated with the recently installed health physics administrative computer system using Temporary Instruction (TI) 2515/141, Review of Y2K Readiness of Computer Systems at Nuclear Power Plants. Documents reviewed include the Software Verification and Validation Plan and the Software Certificate of Compliance for Y2K Operation. Tests performed and software certifications were adequate to ensure the Health Physics Information System is Y2K compliant.

IV. Plant Support

P1 Conduct of Emergency Planning Activities

P1.1 Strike Contingency Plans (92709)

The inspectors performed Inspection Procedure 92709 to review the licensee's strike contingency plan. The inspectors discussed operations shift manning with Operations Management. The licensee had adequate staffing for four shifts without any problems. Plans for staffing the fire brigade were discussed. The corporate plan was discussed including offsite contracts. The emergency plan has in place actions to respond to a civil disturbance. Emergency plan staffing response personnel were designated. Contingency plans were in place in the event of a strike or other civil disturbance if needed.

F4 Fire Protection Staff Knowledge and Performance

F4.1 Fire Drill (71750)

An announced fire brigade drill on August 12, 1999, was challenging and realistic, including a fatality and medical injury. Good fire fighting and emergency rescue practices and techniques were used. A thorough critique was performed afterwards.

R1 Radiological Protection and Chemistry Controls

R1.1 Tour of Radiological Protected Areas (86750)

a. Inspection Scope

The inspectors reviewed implementation of selected elements of the licensee's radiation protection program as required by 10 Code of Federal Regulations (CFR) Parts 20.1902, and 1904.

b. Observations and Findings

During tours of the Turbine Building, Auxiliary Building, Radioactive Waste (Radwaste) Building, and storage and handling facilities, the inspectors reviewed survey data and observed activities in progress. The inspectors determined the licensee was processing radioactive waste to maintain exposures As Low As Reasonably Achievable (ALARA) and to minimize quantities of radioactive waste stored on site. Based on observations and independent radiation survey results, the inspectors determined that the licensee was effectively labeling, controlling, and storing radioactive material as required by regulations.

c. Conclusions

The licensee was effectively labeling, controlling, and storing radioactive material as required by regulations.

R1.2 Transportation of Radioactive Materials (86750)

a. Inspection Scope

The inspectors evaluated the licensee's transportation of radioactive materials programs for implementing the revised Department of Transportation (DOT) and Nuclear Regulatory Commission (NRC) transportation regulations for shipment of radioactive materials as required by 10 CFR 71.5 and 49 CFR Parts 100 through 177.

b. Observations and Findings

Licensee's records for five shipments of radioactive material performed since the last inspection of this area were reviewed, including a shipment of radioactive waste oil being prepared at the time of the inspection. The shipping papers reviewed contained the required information. The licensee had maintained records of shipments of licensed material for a period of three years after shipment as required by 10 CFR 71.91(a). In addition, the licensee personnel qualified to perform shipments of radioactive material had met the recurrent training requirements of 49 CFR 172.704.

c. Conclusions

The licensee had effectively implemented a program for shipping radioactive materials required by NRC and DOT regulations.

R2 Status of Radiation Protection and Chemistry Facilities and Equipment

R2.1 Environmental Samplers (84750)

a. Inspection Scope

The inspection scope was to determine if environmental monitors were being maintained in an operational condition as required by Technical Specifications (TS) and Updated Final Safety Analysis Report (UFSAR) to monitor the radiation and radionuclides in the environs.

b. Observations and Findings

The inspectors observed environmental samplers at five air sampling stations and discussed sampling procedures with laboratory personnel. The inspectors determined that the environmental sampling equipment was calibrated and functional at the time of the inspection. The inspectors also verified locations were consistent with their descriptions in the Offsite Dose Calculation Manual (ODCM) and UFSAR and that the samples performed were in accordance with procedures. The inspectors concluded environmental air samplers were being maintained in an operational condition to comply with TS requirements and UFSAR commitments.

The inspectors reviewed the licensee's 1998 Annual Radiological Environmental Operating Report issued prior to May 1, 1999. Based on the report data, the inspectors determined that the environmental sampling and analysis frequencies specified in the ODCM had been met. The inspectors also reviewed the licensee's 1998 Annual Effluent Release Report issued prior to May 1, 1999. Based on the report data, the amounts of activity released from the plant in liquid and gaseous effluents has remained stable over the last several years and the radiation doses resulting from those releases were well within regulatory limits.

c. Conclusion

The environmental air sampling equipment was being maintained in an operational condition to comply with TS requirements and UFSAR commitments.

R7 Quality Assurance in Radiation Protection and Chemistry

R7.1 Quality Assurance Audits

a. Inspection Scope

Licensee periodic reviews of the Radiation Protection program were reviewed to determine the adequacy of problem identification and corrective actions as required by 10 CFR 20.1101.

b. Observations and Findings

The inspector reviewed the licensee's most recent quality assurance audit report and checklist used to conduct audits in the areas of environmental monitoring and transportation of radioactive materials. The reports demonstrated the licensee was effectively assessing the environmental sampling program and meeting requirements for conducting quality assurance audits in these areas.

c. Conclusion

The inspectors determined the licensee's most recent formal Quality Assurance Audit effectively assessed the areas of environmental sampling and transportation of radioactive material.

V. Management Meetings and Other Areas

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on September 10, 1999. Interim exit meetings were held on August 13, and August 26, 1999 to discuss the findings of Region based inspection. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Lowens, Quality Assurance Manager
S. Franzone, Licensing Manager
R. Hovey, Site Vice-President
D. Jernigan, Plant General Manager
T. Jones, Operations Manager
J. Kirkpatrick, Protection Services Manager
M. Lacal, Training Manager

G. Hollinger, Work Control Manager
R. Rose, Maintenance Manager
E. Thompson, License Renewal Project Manager
D. Tomaszewski, Site Engineering Manager
J. Trejo, Health Physics/Chemistry Supervisor
A. Zielonka, System Engineering Project Manager

Other licensee employees contacted included office, operations, engineering, maintenance, chemistry/radiation, and corporate personnel.

NRC

C. Patterson, Senior Resident Inspector
R. Reyes, Resident Inspector
T. Ross, Senior Resident Inspector (St. Lucie)
G. Warnick, Resident Inspector (St. Lucie)
J. Lenahan, RII
D. Forbes, RII
A. Boland, RII

INSPECTION PROCEDURES USED

IP 37550:	Engineering
IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 84750:	Radioactive Waste Treatment, and Effluent and Environmental Monitoring
IP 86750:	Solid Radioactive Waste Management and Transport of Radioactive Materials
IP 92709:	Licensee Strike Contingency Plans
IP 92902:	Followup - Engineering

ITEMS OPENED AND CLOSED

Opened

50-250,251/99-05-01	NCV	Inadequate Execution of a Clearance and Independent Verification (Section O1.2).
50-250,251/99-05-02	IFI	Evaluate Acceptability of Concrete Temperature at Interface Between Reactor Structural Steel Supports and Shield Wall and Long-Term Effects on Concrete (Section E2.2)

Closed

50-250,251/99-05-01	NCV	Inadequate Execution of a Clearance and Independent Verification (Section O1.2).