UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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April 18, 2000

Florida Power and Light Company
ATTN: Mr. T. F. Plunkett
President - Nuclear Division
P. O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: NRC INTEGRATED INSPECTION REPORT 50-250/00-02, 50-251/00-02

Dear Mr. Plunkett:

This refers to the inspection conducted on February 20 through April 1, 2000, at the Turkey Point 3 and 4 reactor facilities. The enclosed report presents the results of that inspection.

During the inspection period, your conduct of activities was generally characterized by safety conscious operations.

Based on the results of the inspection, the NRC has determined that two violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII B.1.a of the Enforcement Policy. The NCVs are described in the subject inspection report. If you contest a violation or severity level of an NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II, the Resident inspector at your facility, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR).

Sincerely,

/RA/

Leonard D. Wert, Chief
Reactor Projects Branch 3
Division of Reactor Projects

Docket Nos. 50-250, 50-251
License Nos. DPR-31, DPR-41

Enclosure: Inspection Report Nos. 50-250/00-02, 50-251/00-02

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos: 50-250, 50-251
License Nos: DPR-31, DPR-41

Report Nos: 50-250/00-02, 50-251/00-02

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: February 20 - April 1, 2000

Inspectors: C. Patterson, Senior Resident Inspector
R. Reyes, Resident Inspector
J. Coley, Reactor Inspector (Sections M1.2, M7.1)

Approved by: L. Wert, Chief
Reactor Projects Branch 3
Division of Reactor Projects

Enclosure

EXECUTIVE SUMMARY

Turkey Point Nuclear Plant, Units 3 & 4
NRC Inspection Report 50-250/00-02, 50-251/00-02

This integrated inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 6-week period of resident inspection and includes the results of an inspection by a regional specialist of the inservice inspection program.

Operations

- A non-cited violation was identified for failure to follow procedures associated with pressurizer level instrumentation during a reactor coolant system inventory reduction early in the refueling outage. Application of industry experience concerning this issue was not effective (Section O1.2).

Maintenance

- Overall, outage activities were well controlled. When problems were encountered, the involved activities were suspended and the issues were thoroughly reviewed. Emphasis was placed on industrial safety and material readiness (Section M1.1).

- Inservice examination activities observed were performed using approved procedures by certified examiners. The inspection results were properly recorded and evaluated in accordance with the appropriate test procedures. The Code repair package reviewed was detailed and complete (Section M1.2).

Engineering

- During fuel movement in the containment, a rod control cluster assembly (RCCA) inside a fuel assembly was damaged as a result of interference between the manipulator crane mast and the RCCA. A modification to the mast had resulted in inadequate clearance between the mast and a RCCA in an assembly seated in the fuel transfer equipment. The fuel handling crew completed appropriate actions in response to the incident and the licensee's review was self-critical. A non-cited violation was identified for failure to verify adequacy of the in-mast design modification (Section E2.2).

Report Details

Summary of Plant Status

Unit 3 operated until February 28, 2000, when the unit was taken offline for a refueling outage. The unit had been online since November 18, 1999. The unit was returned to power generation on March 26, 2000.

Unit 4 operated continuously during this report period and has been online since January 27, 2000. Power was reduced to 60 percent on March 28, 2000, to repair a condenser tube leak and was returned to full power the next day.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. Overall, the conduct of operations was safety-conscious; specific events and noteworthy observations are detailed in the sections below.

O1.2 Pressurizer Level Indication Issues During Reactor Coolant System (RCS) Inventory Reduction

a. Inspection Scope (71707)

During the Unit 3 refueling outage, the inspectors reviewed the initial draindown of the RCS and details of a problem with some of the pressurizer level indicators. The problem occurred during an planned inventory reduction.

b. Observations and Findings

On March 1, 2000, the inspectors responded to the control room after Operations management identified indications that more water was drained from the RCS than was expected based on pressurizer level indication. Unit 3 was in cold shutdown. The inspectors observed that the cold calibrated pressurizer level indication, LT-462, was at 34% and two of the hot calibrated level indicators, LT-459 and LT-460, were indicating zero. Earlier that morning, operators had depressurized the RCS from 340 to 30 pounds per square inch gauge and commenced a draindown from a solid plant condition. The draining was stopped by operations management due to an apparent disparity between indicated pressurizer level and the volume of water drained. Plant management ordered an Event Response Team (ERT) formed to determine the cause of the disparity and placed a hold on additional draindown activities.

There are four pressurizer levels indicators in the control room. These are the cold calibrated indication, LT-462, and three protective channels, LT-459, 460, 461, which are calibrated for hot plant conditions. LT-462 and LT-461 share a common reference leg as do LT-459 and LT-460. Since LT-461 was indicating 44% and the other two hot calibrated indications (off a different reference leg) were indicating zero, a reference leg

problem was suspected. The licensee backfilled the reference leg connected to LT-462 and LT-461 and the indicated levels changed from 33% and 44% to -6% and 8% respectively. This confirmed that there was a reference leg problem, and that LT-462 and LT461 had been indicating higher than the actual pressurizer level.

The ERT determined that the cause of the reference leg problem was gases coming out of solution when the RCS was depressurized. The root cause was that the maintenance practice used to fill the reference leg could trap significant amounts of air. Two level transmitters and three pressure transmitters are connected to this reference leg. Corrective actions for this problem were to change the filling procedure to fill the instruments in a specific order and also to fill them following all maintenance at the end of the outage. There were a few occasions over the last several years when the reference leg required filling. This past history was documented in Condition Report (CR) 95-1126.

The ERT also determined that a contributing or potential cause was human performance by plant operators. There were several indications that could have been used to determine that LT-462 was not reading correctly. Operators did not fully understand the relationship between hot calibration indication level and cold calibration level. For example, when LT-462 indicated 33%, the hot calibration instruments LT-459, 460, 461, should have all indicated near 50%. In this case LT-461 indicated 44% but LT-459 and 460 indicated zero. This disparity existed during the draindown and information was available hours earlier. Corrective actions for this problem were to revise the procedure to provide more specifics and additional training for the operators.

Also, the ERT noted that there was considerable industry experience related to this event. A similar event occurred at Sequoyah Unit 1 in March 1997, that resulted in an industry notification. An action item was assigned to review the licensee's actions in response to the communication.

The inspectors performed an independent review of this problem. The outage risk assessment schedule time line indicated this evolution as a transition from green to red due to reduced inventory in the core and shorter time to boil once drained down. The inspectors reviewed procedure 3-OP-041.7, Draining the Reactor Coolant System, and noted several cautions on page 18 and 23. These cautions state that reliance on one level indicator may lead to a loss of control of RCS inventory. The procedure clearly indicated that the three pressurizer level protection channels and PT-3-402 should be monitored and trended along with pressurizer cold calibration level to verify proper response of pressurizer cold calibration level. The inspectors noted that, during normal power operation, a maximum of 8% level deviation is allowed between the hot calibrated instruments. A label underneath the instruments on the control board indicates this allowed deviation. During the draindown, the deviation varied from 30% to 60%. From discussions with plant operators, the inspectors noted that the relationship between hot and cold calibration level instruments was not fully understood. A chart of this relationship was available in the control room in the plant curve book, but was not used.

The inspectors concluded that a violation of plant procedures had occurred. Technical Specification (TS) 6.8.1 requires that written procedure shall be established, implemented, and maintained covering the applicable procedures recommended in

Appendix A of Regulatory Guides (RG) 1.33, Revision 2, February 1978. Section 3 of Appendix A to RG 1.33 recommends procedures for draining the RCS. Plant procedure 3-OP-041.7, Draining the Reactor Coolant System, implements this requirement. Cautions in the procedure were not followed to verify proper response of pressurizer cold calibration level during draindown. This Severity Level IV violation is being treated as a Non-Cited violation (NCV), consistent with Section VII B.1.a of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 00-0391. The safety significance of this incident is limited. The draindown was stopped due to questioning by an Operations manager. Actual RCS level was at about the bottom of the pressurizer when the draindown was stopped. Other level indications were available if the draindown had continued. Shutdown cooling system operability was not affected by the incident. This is identified as NCV 50-250/00-02-01, Failure to Verify Proper Response of Pressurizer Cold Calibration Level During Draindown.

c. Conclusion

A noncited violation was identified for failure to follow procedures associated with pressurizer level instrumentation during a reactor coolant system inventory reduction early in the refueling outage. Application of industry experience concerning this issue was not effective.

O1.3 Control Room Observations(71707)

The inspectors observed control room activities during major operational evolutions during the Unit 3 outage. The inspection included backshift and deep backshift observations. The inspectors observed the control room activities during reactor power decrease to hot standby, Residual Heat Removal (RHR) cooling, fuel movement in the containment, Emergency Core Cooling System (ECCS) Motor Operated Valve (MOV) testing, filling and venting the RCS, boration to criticality, and reactor power decrease to perform turbine trip test. Prior to several major evolutions, the inspectors reviewed the applicable procedures with the control room operators and discussed the expected annunciator and system response, risk significance of evolutions, and applicable technical specification action statements. The inspectors found that the Reactor Operators and Senior Reactor Operators were very knowledgeable with all of these items. Additionally, the inspectors noted that the tailboard briefings were detailed and thorough. During the evolutions, the inspectors followed the procedures and independently verified selected instrumentation readings as described in the procedures, and verified compliance with TS requirements. Additionally, the inspectors verified the minimum control room staffing to perform specific evolutions, which included Reactor Engineering and Operations Management oversight, as described in the licensee's Operational Procedures. Control room communications amongst the Operators in the Control Room and with the field were frequent and detailed. Control room supervisors maintained good command and control during the evolutions to ensure appropriate focus on the evolution being performed. Reactor Engineering provided good oversight during reactor reactivity changes and during fuel movement in the containment.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Work Order and Surveillance Observations

a. Inspection Scope (61726) (62707)

The inspectors observed the following surveillance and maintenance activities:

OSP-023.2	3B Emergency Diesel Generator 24 Hour Full Load Test and Load Rejection
3-OSP-041.8	Filling & Venting The Reactor Coolant System
TP-99-049	Differential Pressure testing of Component Cooling Water Valve MOV -3-1417
0-ADM-035	Limitations and Precautions For Handling Fuel Assemblies
3-OP-040.2	Refueling Core Shuffle
OTSC 00-0117	Manipulator Crane -Operating Instructions
4-OP-064	Safety Injection Accumulators
0-OP-046	Chemical Volume Control System Boration Control
3-GOP-103	Power Operations to Hot Standby
3-OSP-089.1	Turbine Generator Overspeed Test
3-OSP-064.2	Accumulator Outlet Check Valves Leak Test
0-PME-028.2	Rod Position Indicator Inverter Maintenance

b. Observations and Findings

Most observed activities were properly performed and no problems identified. Problems were encountered with pressurizer level indications during draindown and a Rod Control Cluster Assembly (RCCA) was damaged during fuel movement. In each case, management stopped the evolution and directed an Event Response Team (ERT) to identify the cause and corrective actions. In addition, the scope of the steam generator tube eddy current testing was expanded beyond that originally planned for the outage. Strong attention was given to industrial safety during the outage resulting in a reduced injury rate. Material readiness of the plant for restart was emphasized following the outage. The inspectors accompanied the Site Vice President on a final containment walkdown after reaching mode four. Only minor discrepancies were noted on the detailed walkthrough of containment.

c. Conclusions

Overall, outage activities were well controlled. When problems were encountered, the involved activities were suspended and the issues were thoroughly reviewed. Emphasis was placed on industrial safety and material readiness.

M1.2 Inservice Inspection (ISI) - Observation of (73753) (73753)

a. Inspection Scope

The inspectors observed three methods of examination of components which included: visual examination of an installed support component (mechanical snubber) and the functional test of the same snubber, surface examination of a weld using magnetic particle examination, and volumetric examination of a feedwater weld using manual ultrasonic examination. In addition, the inspectors observed augmented ultrasonic examinations of the B loop and C loop steam generator feedwater piping, reviewed a completed Unit 3 Code repair package for replacement of auxiliary steam system piping (repair packages which included weld radiographic film were not yet complete for the Unit 3 Spring 2000 refueling outage). These observations were performed to determine whether the ISI, repair, and replacement of Class 1, 2, & 3 pressure retaining components at the Turkey Point facility were performed in accordance with Technical Specifications, the American Society of Mechanical Engineers (ASME) Code (1989 Edition, no Addenda, Sections XI & V), and correspondence between NRC staff and the licensee.

b. Observations and Findings

The inspectors observed the visual examinations of Safety Injection System Snubber No. 3-PRWH-2 and observed the functional test of the same snubber, observed magnetic particle examination of Feedwater Weld No. 14"-FWB-2303-17, observed volumetric ultrasonic examination of Feedwater Weld No.14"-FWB-2303-17, and observed augmented ultrasonic examinations of the B loop and C loop steam generator feedwater piping from the nozzle ramp to one diameter upstream of the horizontal elbow weld. No defects were observed during these examinations. In addition, the inspectors reviewed a previously completed (1999) Unit 3 Code repair package for replacement of auxiliary steam system piping Work Order No. 98010491-01, “Piping Down Stream of Valve AFSS-4-004 on Drawing 5610-P-809-S SHT 8". No findings were identified.

c. Conclusions

Inservice examination activities observed were performed using approved procedures by certified examiners. The inspection results were properly recorded and evaluated in accordance with the appropriate test procedures. The Code repair package reviewed was detailed and complete.

M5 Maintenance Staff Training and Qualification (62707, 71707)

M5.1 Contract Worker Qualifications

The inspectors reviewed the implementation of the licensee’s procedures used to determine the qualifications of employees contracted to work on safety or quality related equipment. Procedure 0-ADM-005, Control Of On-Site Services, is used to screen contractors to perform work in the plant and is applicable to personnel that perform work on safety or quality related systems and components. Attachment One of the procedure is used to evaluate prospective contract personnel to ensure they are adequately trained and qualified, prior to commencing work. However, craft personnel brought onsite by the Site Labor Broker are exempt from these requirements by step 1.1.1 of 0-ADM-005. Generally, these are members of craft unions. Qualifications are established by their

specific trade and all work performed is under the supervision of the licensee and performed to the licensee's QA program.

The inspectors reviewed Attachment One of procedure 0-ADM-005 for selected personnel that performed work during the Unit 3 outage on Fuel Handling Operations, Instrumentation and Control Reactor Vessel Cables, and electrical Reactor Coolant Pump Motor work. The inspectors determined that the licensee maintained complete records describing the qualifications of the contractors and the basis for permitting specific work to be performed by selected personnel. The inspectors reviewed the procedure with Maintenance supervisors and discussed training and qualification requirements for performing specific duties. Additionally, the inspectors interviewed personnel involved in fuel handling operations and discussed specific qualifications as described in Attachment One of the 0-ADM-005 procedure. As a result of this review, no issues were identified with the qualification for these contract employees.

M7 Quality Assurance in Maintenance Activities

M7.1 Licensee Assessments of Inservice Inspection (ISI) Activities (73753)

The inspectors evaluated the effectiveness of licensee's controls for identifying, resolving and preventing problems in ISI by reviewing the corrective actions taken for items identified in Audit No. QAS-CSI-99-01, "Functional Area Audit of Component, Support and Inspection," and Self Assessment No. QAO-PTN-99-007, "Review of ISI/IST Program Functional Area Audit." After thorough examination of the identified problems, the inspectors concluded that the licensee's controls were effectively identifying and resolving issues within the corrective action program.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Modifications (37551)

The inspectors reviewed the following plant change modifications during the Unit 3 refueling outage:

PC/M 99-015	Fuel Reload
PC/M 99-047	Fuel Sipping / Fuel Reconstitution
PC/M 99-060	Inservice Inspection Steam Generator Feed Ring
PC/M 99-029	High Head Safety Injection Pump Gas Binding

With the exception of the fuel sipping modification discussed in the next section, these modifications were performed without significant problems. The inspectors observed the in-mast sipping system test being performed in containment during core off-load, and observed the identified failed fuel rod inspection and replacement which were performed in the spent fuel pool. The inspectors found that personnel performing the reconstitution were very knowledgeable with the overall process and followed all safety

precautions as described in the fuel reconstitution procedures. The inspectors noted that engineering's description, review and documentation of this fuel failure issue, and subsequent generic implications review and corrective actions to address this issue were comprehensive and timely.

E2.2 In-Mast Sipping Modification and Fuel Handling Incident

a. Inspection Scope (37551, 71707)

The inspectors reviewed a fuel handling incident that occurred in the containment during core off-load that resulted in a damaged Rod Control Cluster Assembly (RCCA) inside a fuel assembly.

b. Observations and Findings

Just prior to the refueling outage, the licensee implemented a modification to perform in-mast sipping during core off-load to identify leaking fuel rod(s). The modification package also included the performance of Ultrasonic Inspection and Fuel Assembly Reconstitution. The in-mast sipping system allows a means of performing on-line quantitative leak testing of fuel assemblies in the refueling mast during normal fuel handling operations. To perform the in-mast testing, a mechanical design change was made to the bottom of the manipulator crane mast. A sipping fixture was added to the bottom of the mast. This caused the total length of the manipulator mast to increase. The in-mast sipping system was designed, evaluated, fabricated and installed by a vendor. The modification was approved by the licensee under Plant Change/Modification (PC/M) 99-047.

On March 5, 2000, during core off-load in the containment, a RCCA inside a fuel assembly was damaged due to mechanical interference between the fuel assembly and the manipulator crane in-mast sipping fixture. Seven fuel assemblies had already been transferred from the core. This was the first fuel assembly containing a RCCA. The fuel assembly had just been inserted into the upender and the manipulator mast gripper had been verified to be up and disengaged. As the mast moved away from the upender, simultaneously the upender operator initiated lowering of the upender assembly. The RCCA hub caught on the bottom of the mast and was pulled along the travel path of the mast. This caused the RCCA hub to bend until the interference was cleared. The inspectors responded to the site upon notification by the licensee of this problem. The inspector viewed the damaged RCCA and fuel assembly by monitoring an underwater camera display. There were no indications of damaged fuel. The licensee formed an ERT to handle the issues. The inspectors monitored all recovery activities associated with this problem.

The inspectors reviewed the fuel handling issue and modification PC/M 99-047, Unit 3 In-Mast Sipping, Ultrasonic Inspection and Fuel Rod Reconstitution; root cause; and planned corrective actions with engineering management. Additionally, the inspectors reviewed the licensee's fuel handling procedures and discussed the issue individually with the fuel handling crew that was performing the fuel movement during the incident. No issues were identified with performance of the fuel movement as described in the licensee's procedures. The crew took appropriate actions after the incident.

The root cause of the mechanical interference between the RCCA hub and the in-mast fixture was due to insufficient verification of the vertical clearance between the modified mast and the fuel assemblies containing RCCAs. A contributing cause was that the licensee did not adequately review this critical attribute of the design provided by the contracted design organization. Corrective actions included modifying the in-mast fixture to provide adequate clearance. The licensee subsequently completed full core off-load and no other issues were encountered.

10 CFR Part 50 Appendix B, Criterion III, Design Control, requires in part that "design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods or by performance of a suitable testing program." The licensee did not adequately verify clearance between the modified mast and the fuel assemblies containing RCCAs. Consequently, during fuel movement in the Unit 3 containment, a fuel assembly containing a RCCA was damaged as a result of mechanical interference. The fuel remained intact. The operating crew responded correctly after the incident. The licensee's review of the event was self-critical and identified the deficiencies that caused the problem. This Severity Level IV violation is being treated as a Non-Cited violation, consistent with Section VII B.1.a of the NRC Enforcement Policy. This is identified as NCV 50-250/00-02-02, Failure To Verify Adequacy of A Fuel Sipping Design Modification. This violation is in the licensee's corrective action program as CR 00-0452.

c. Conclusions

During fuel movement in the containment, a rod control cluster assembly (RCCA) inside a fuel assembly was damaged as a result of interference between the manipulator crane mast and the RCCA. A modification to the mast had resulted in inadequate clearance between the mast and a RCCA in an assembly seated in the fuel transfer equipment. The fuel handling crew completed appropriate actions in response to the incident and the licensee's review was self-critical. A non-cited violation was identified for failure to verify adequacy of the in-mast design modification.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Temporary Instruction (TI) 2515/142, Draindown During Shutdown And Common-Mode Failure (NRC Generic Letter 98-02)

The licensee's engineering evaluation and assessment relating to this generic issue was documented in PTN-ENG-SENS-98-046. The licensee concluded that since the Emergency Core Cooling Systems are designed with a common pump suction header, Turkey Point was susceptible to common cause failure. That conclusion assumed that several barriers must first have failed, i.e., procedure adherence, compliance with Technical Specifications, and loss of command and control in the control room. The report also described that the Turkey Point Residual heat Removal (RHR) system design includes Motor Operated Valve (MOV) open permissive and pressure switch interlocks. These prevent opening MOVs which would allow pressurized water to enter the Refueling Water Storage Tank (RWST) when MOVs from the RCS hot leg were opened. Additionally, the licensee described training and administrative controls that were in place to preclude RCS draindown and voiding in the RWST suction header.

The inspectors reviewed the RHR system drawings, MOV and pressure interlock prints, Operating procedures, and Shutdown Off-Normal Operating Procedures, and verified implementation of selected activities during the Unit 3 outage. During RHR cooling, the inspectors independently verified valve and MOV breaker positions in the field, and valve and breaker position indication in the control room.

A review of recent Reactor Operator (RO) training records verified that the licensee includes training on Outage Risk Assessment relating to this generic issue. The training includes review of NRC Information Notice 95-03 which described the Wolf Creek event. During the Unit 3 shutdown when RHR cooling was in effect, the inspectors reviewed this generic issue with two recently licensed ROs which were in the control room overseeing the cool down activities on the Unit. The inspectors found that the Operators were knowledgeable with the generic issue. They understood the barriers and the general MOV open permissive and pressure switch interlocks. The inspectors reviewed scheduling of MOV testing and found that the Outage Risk Assessment team recommended scheduling of RHR MOV work only during the time when there was no fuel in the core. This further reduced the risk of RCS draindown and RWST voiding. The inspectors verified no RHR MOV work was performed during hot shutdown. No issues were identified as a result of the review. This item is closed.

E8.2 (Closed) LER 50-251/2000-001-00; Unit 4 Manual Reactor Trip Due To Main Feed Water Flow Control Valve Cage Disengagement. This LER addressed the Unit 4 trip, root cause, and corrective actions. The trip, the root cause analysis, and several corrective actions were previously reviewed in NRC Inspection Report 50-250,251/ 00-01. During this report period, the inspectors verified completion of selected corrective actions applicable to Unit 3 which were to be performed during the outage. Specifically, a torque specification was added to the Steam Generator feed regulator valve preventive maintenance procedure, and the torque was applied to the valve cages. Based on the licensee's thorough review of this issue, and the progress completed on implementing the corrective actions, this LER is closed.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Radiological Posting (71750)

The inspectors reviewed radiological postings during the Unit 3 refueling outage. Emphasis was placed on posting of high radiation areas and locked high radiation areas. The inspectors observed the licensee exercise care while performing a crud burst at the beginning of the outage. Several inspections were conducted in the containment, spent fuel pool room, and volume control tank room during the outage. The inspectors verified the radiological conditions in the volume control tank room were as specified on a current survey map. This area had recently been changed from a locked high radiation area to a radiation area. No problems were identified.

R1.2 Dose Reduction

The licensee carefully monitored personnel exposure during the outage, focusing on dose reduction. A successful crud burst was implemented at the beginning of the outage. Several major jobs were accomplished with less dose than pervious outages. Despite the expanded scope of some jobs such as the steam generator tube inspection, the overall outage dose was less than that of previous outages.

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 April 6, 2000. Interim exit meetings were held on March 10, 2000 to discuss the findings of Region based inspection. The licensee acknowledged the findings presented.

Proprietary information was reviewed during this inspection but was not discussed in this report.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

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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 Manager

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

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 73753	Inservice Inspection, Observation of ISI Work Activities
IP 92903:	Follow-up - Engineering

ITEMS OPENED AND CLOSED

Opened

50-250/00-02-01	NCV	Failure to Verify Proper Response of Pressurizer Cold Calibration Level During Draindown (Section O1.2).
50-250/00-02-02	NCV	Failure to Verify Adequacy of Fuel Sipping Design Modification (Section E1.1).

Closed

50-250/00-02-01	NCV	Failure to Verify Proper Response of Pressurizer Cold Calibration Level During Draindown (Section O1.2).
50-250/00-02-01	NCV	Failure to Verify Adequacy of Fuel Sipping Design Modification (Section E1.1).
50-251/2000-001-00	LER	Manual Reactor Trip Due To Main Feed Water Flow Control Valve Cage Disengagement (Section E8.2).
Temporary Instruction	TI	TI 2515/142, Draindown During Shutdown And Common-Mode Failure (NRC Generic Letter 98-02) (Section E8.1)