U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos: 50-335, 50-389
License Nos: DPR-67, NPF-16

Report No: 50-335/96-12, 50-389/96-12

Licensee: Florida Power & Light Co.

Facility: St. Lucie Nuclear Plant, Units 1 & 2

Location: 9250 West Flagler Street
Miami, FL 33102

Date: July 12, 1996

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EXECUTIVE SUMMARY

St. Lucie Nuclear Plant, Units 1 & 2
NRC Inspection Report 50-335/96-12, 50-389/96-12

This special inspection included aspects of licensee's configuration management and 10 CFR 50.59 programs. Specifically, the inspection examined the extent to which plant changes were appropriately incorporated into procedures and drawings and the performance of 10 CFR 50.59 safety evaluations. Conclusions included the following:

- A review of a number of screenings and evaluations performed pursuant to 10 CFR 50.59 resulted in the identification of four apparent violations:
    - One example of an apparent failure to perform a safety evaluation due to a failure to employ engineering controls in the construction of the Unit 2 Control Element Drive Mechanism Control System room and a continuing failure to recognize the nondocumented nature of the room (paragraph E1.1.b.1).
    - One example of an apparent failure to identify that the installation of a temporary fire pump represented a change to the plant as described in the Update Final Safety Analysis Report, resulting in a failure to perform a safety evaluation (paragraph E1.1.b.2).
    - One example of an apparent failure to recognize that refueling equipment setpoints were included in the Updated Final Safety Analysis Report while performing a safety evaluation screening, leading to a failure to perform a safety evaluation (paragraph E1.1.b.3).
    - One example of an apparent failure to recognize an unreviewed safety question in the development of a safety evaluation for an Emergency Diesel Generator fuel oil transfer line valve lineup change (paragraph E1.1.b.4).
- A review of off-normal operating procedures relating to safety-related annunciators identified a number of inaccuracies (paragraph E7.1).
- Five apparent failures to properly incorporate Plant Change/Modification packages into drawings and procedures were identified (paragraph E7.2).

Report Details

E1 Conduct of Engineering

E1.1 Safety Evaluations/10 CFR 50.59 Issues (37550, 71707)

a. Inspection Scope

The inspectors reviewed a sample of the licensee's safety evaluations (SEs) performed pursuant to 10 CFR 50.59. The evaluations were reviewed for threshold for determining if an unreviewed safety question (USQ) existed because of an increase in the probability of a design basis accident occurring, an increase in equipment malfunction, a reduction in the margin of safety, or an increase in radiation dose consequences. These evaluations were also reviewed for adequacy of screening and assumptions used for the safety evaluations.

b. Observations and Findings

The inspectors reviewed twelve SEs or issues which might require SEs. The issues were:

- Cracking of Westinghouse Alloy 600 Mechanical Steam Generator Plugs.
- Temporary Relocation of Class Break on Intake Cooling Water.
- Installation of Temporary Fire Penetration Seals in Pipe Barrier BW064.
- Temporary Installation of Strain Measuring Devices on the Pressurizer Relief Valve Discharge Piping.
- Safety Injection Tank (SIT) Discharge/Loop Check Valve Stroke Test-Unit 1.
- Freeze Seal Application for V3651 and V3652 on the 1B Shutdown Cooling Return Line.
- Safety Evaluation For Boraflex Blackness Testing Results.
- Wide Range Nuclear Instrumentation Temporary System Alteration.
- Temporary Configuration for Control Element Drive Mechanism Control System (CEDMCS) Cooling System and Enclosure, Unit 2.
- Safety Evaluation for Inoperable Fire Pump
- St. Lucie Unit 1 Refueling Equipment Underload and Overload Settings.

- The Isolation of Fuel Oil Supply Line to the 2B Emergency Diesel Generator.

Problems were identified with the last four items and the details are discussed in the following paragraphs.

1) Temporary Configuration for CEDMCS Cooling System and Enclosure

On June 4, 1996, a control room annunciator indicated that an undervoltage condition existed on the CEDMCS. Operations responded to the CEDMCS equipment and noted that the CEDMCS enclosure was approximately 11 degrees warmer than normal. This enclosure is located in the cable spreading room on the 43 foot elevation of the reactor auxiliary building.

Following this event, an In-House Event Report and Condition Reports (CRs) 96-1238, 96-1245 and 96-1325 were issued. The following items with appropriate plant corrective action tracking numbers were identified by these reports:

- CEDMCS enclosure and air conditioning units did not appear on the plant's controlled drawings. (STAR 951320)

- CEDMCS enclosure air conditioning units were not seismic qualified. Final design was in process to provide seismic restraints for the air condition units. (PM 96-06-208)

As part of the action for CR 96-1325, a 10 CFR 50.59 safety evaluation was performed on the CEDMCS enclosure. The evaluation found that this air conditioned enclosure was erected in the early 1980's during the pre-operational testing phase. Testing performed at that time found that the CEDMCS enclosure required an air conditioned environment to prevent overheating of the four CEDMCS cabinets. The licensee's current review determined that the design of the enclosure was acceptable, except that the air conditioning units and one air conditioning duct presented a hazard to safety related equipment in a seismic event. Therefore, seismic supports and restraints were provided for the air conditioning units and duct prior to the unit's restart on June 13.

The inspector reviewed the 10 CFR 50.59 SE prepared for the design and installation of the seismic restraints and justification of the installation of the CEDMCS enclosure. A 10 CFR 50.59 review was apparently not performed when the enclosure was originally erected. The CEDMCS was described in the Updated Final Safety Evaluation Report (UFSAR) but the cooling system and enclosure for the CEDMCS were not described in the UFSAR. This was identified as another example of Unresolved Item (URI) 50-335,389/96-04-09, "Failure to Update UFSAR."

The failure to perform an evaluation as required by 10 CFR 50.59 prior to, or at any time subsequent to, making a change to the plant as described by the UFSAR is an apparent violation (EEI 50-389/96-12-01, "Failure to Perform a 10 CFR 50.59 Safety Evaluation for CEDMCS Enclosure," EA 96-236).

2) Safety Evaluation for Inoperable Fire Pump

During the Spring 1996 Unit 1 refueling outage, one of the two Unit 1 Emergency Diesel Generators (EDGs) had been placed out of service to perform maintenance and modification work activities. Only one EDG was in service to provide power in the event of a loss of offsite power event. To prevent a possible overload on the single EDG unit, a number of breakers to various components were opened and the units 480V electrical busses were crosstied in accordance with OP 1-0910024, Rev 6, "Crosstying/Removal of 480V Buses." One of the components removed from service was Fire Pump 1B. The breaker to this fire pump was opened on May 21, and this pump was removed from service and remained out of service on June 8.

AP 1800022, Rev 16, "Fire Protection Plan," Appendix A, Sections 2.2 and 2.3 required two fire pumps rated at a capacity of 2300 gpm to be operable at all times. Appendix A, Section 4.1.A, stated that with one of the two fire pumps inoperable, the inoperable equipment was to be restored to service within seven days or an alternate backup pump was to be provided within the next 30 days.

Fire Pump 1B had been out of service for 18 days. The compensatory measure established for this pump being out of service was the installation of a portable gasoline engine drive pump rated at 750 gpm. This pump had been connected to take suction from the fire protection water storage tank for Fire Pump 1A. This alternate pump was not of the same capacity as one of the two required pumps and a justification was not provided to demonstrate that this pump was of adequate capacity to meet the maximum fire flow requirement for the safety related areas of the plant. The licensee initiated a CR to review this item.

The licensee informed the inspector that the out of service pump could be restored to operability by restoring the existing open breaker to the closed position. Also, the 30 day time to provide an alternate backup pump had not been exceeded. This met the requirements of AP 1800022 for one pump being inoperable.

Resolution of CR 96-1356 indicated that the installation of the portable fire pump as the compensatory measure with one of the permanently installed fire pumps out of service was performed without an engineering evaluation to ensure adequate capacity and without a review under 10 CFR 50.59. The inspector found that the installation of the temporary pump resulted in a change to the plant as described in the UFSAR, Figure 9.2-5, "Flow Diagram Fire Water, Domestic & Makeup Systems." The inspector concluded that a safety evaluation should have been prepared to justify and document the temporary configuration. The licensee stated that no 10 CFR 50.59 screening (and hence, no evaluation) was performed for this installation because the temporary pump, and its associated piping, was installed via Work Order, with no pre-approved procedure and outside the licensee's Temporary System Alteration process (which, if exercised, would have required a safety screening/evaluation). This is an apparent violation (EEI 50-335,389/96-12-02, "Failure to Perform a 10 CFR 50.59 Safety Evaluation For Use of a Temporary Fire Pump," EA 96-236).

3) Refueling Equipment Overload and Underload Settings

CR 96-812 was issued on the SE SEFJ-96-020 by the licensee. The report stated that an engineering evaluation had been written to modify the overload and underload setpoints described in the UFSAR without performing a 50.59 safety analysis/evaluation. These overload and underload load cell setpoints provide a margin to account for resistance encountered while lifting or lowering fuel assemblies and prevent exceeding the fuel assembly and refueling equipment design loads.

The licensee had obtained information from the vendor for use in this Unit 1 refueling outage which would allow an increase in hoist interrupt from 10 percent of the weight of a fuel assembly to 18 percent (approximately 200 pounds). The original engineering analysis did not take into account that these changes in setpoint values would affect the UFSAR and thus the CR was written.

St. Lucie Quality Instruction (QI) 2.0, "Engineering Evaluations," Rev 1 dated January 31, 1996, provides general requirements and guidance for the development and processing of engineering evaluations. This procedure references QI 2.1, "10 CFR 50.59 Screening/Evaluation," Rev 1 dated March 30, 1996, which stated, in part, that the screening process was designed to determine whether an activity required a complete 10 CFR 50.59 by asking a series of four questions. One question, "Does the change represent a change to procedures as described in the SAR?" should have been answered "yes" in the case of the original engineering analysis. The procedure also stated that, "A positive response to any of the first four...questions requires a 10 CFR 50.59 evaluation."

The Facility Review Group (FRG), the site safety committee, noted that a safety evaluation was not present with the requested procedure change and returned the procedure to the engineering group for correction and the CR was written to identify the problem. This failure to perform an evaluation as required by 10 CFR 50.59 prior to making a change to plant procedures described in the UFSAR is an apparent violation (EEI 50-335/96-12-03, "Failure to Perform a 10 CFR 50.59 Safety Evaluation For Change in Setpoints Listed in UFSAR," EA 96-136).

4) Safety Evaluation for Closing Manual Valve to EDG Fuel Supply

In July, 1995, the inspector reviewed SE JPN-PSL-SENS-95-013, which was prepared to allow operation with a manual isolation valve closed in the 2B EDG fuel oil (FO) line from the Diesel Fuel Oil Storage Tank (DFOST) to the day tanks. The configuration was proposed when a leak was determined to exist in the underground line between the two tanks. The action was designed to minimize the amount of FO released to the environment until the leak could be identified and corrected.

As a compensatory measure, the licensee proposed dedicating a Non-Licensed Operator (NLO) to the task of opening the closed valve in the event of an EDG start. The licensee calculated that the EDG day tanks contained enough FO to allow 126 minutes of EDG operation at full load before a transfer of FO was required. The licensee then specified that the NLO would be required to open the valve within 20 minutes of an EDG start. Procedures were revised to include direction to open the valve on an EDG start, and administrative controls were put in place to ensure that the NLO would not be required to perform any other immediate response duties. Additionally, the licensee performed a response time test, placing the operator at the G-2 warehouse (as far away from the EDG as he could credibly be in the protected area) and requiring the NLO to proceed to the valve and open it. The NLO performed this task in approximately seven minutes.

In considering the issue, the licensee employed Probabilistic Risk Assessment (PRA) techniques to estimate the increase in the risk of the loss of the 2B3 bus due to a failure of either the operator to open the valve or a failure of the valve to be able to be opened. The licensee concluded that the increase in probability was approximately 6 percent. However, in considering 10 CFR 50.59 criteria, the licensee concluded that no increase in the probability of failure of a component important to safety was created by the proposed action. The inspector questioned the licensee on this issue. The licensee explained that a deterministic conclusion of no increased probability was reached when the existence of procedural guidance and heightened awareness was balanced against the approximate 6 percent increase in failure probability presented by the two new failure modes.

The inspector noted that 10 CFR 50.59 was written in terms of absolute increases in the probabilities of failure represented by a proposed change. The inspector continued to question whether 10 CFR 50.59 criteria could ever be satisfied when new failure modes are imposed on a previously reviewed system (i.e whether added risk, once qualitatively established, could be completely mitigated). The inspector concluded that insufficient guidance existed from a regulatory perspective to take immediate issue with the licensee's rationale. Further, the inspector concluded that the licensee had taken prudent measures to ensure the continued operability of the 2B EDG while minimizing the FO leak's effect on the environment. The inspector referred the question to the Office of Nuclear Reactor Regulation for resolution.

After consideration of the issue, the NRC determined that the actions taken by the licensee in this instance introduced two new failure modes to the EDG system; failure of the operator to unisolate the fuel oil line and failure of the manual isolation valve to cycle. As a result, the NRC has concluded that the licensee's actions necessarily increased the probability of a failure of a component important to safety and, as such, represented a USQ, as defined in 10 CFR 50.59. Consequently, this action is identified as an apparent violation (EEI 50-389/96-12-04, "Unreviewed Safety Question Involving EDG 2B," EA 96-236).

c. Conclusions on Conduct of Engineering

The inspectors concluded that four apparent violations relating to CFR 50.59 safety evaluations existed. The inspectors noted that these issues varied both in vintage and in individual detail. Summarizing, the examples were the result of:

1) One example of a failure to perform a safety evaluation due to a failure to employ engineering controls in the construction of the Unit 2 CEDMCS room and a continuing failure to recognize the nondocumented nature of the room.

2) One example of a failure to identify that the installation of a temporary fire pump represented a change to the plant as described in the UFSAR, resulting in a failure to perform a safety evaluation.

3) One example of a failure to recognize that refueling equipment setpoints were included in the UFSAR while performing a safety evaluation screening, leading to a failure to perform a safety evaluation. This example was identified by the licensee and corrected before any actual change took place.

4) One example of a failure to recognize an unreviewed safety question in the development of a safety evaluation for an EDG fuel oil transfer line valve lineup change.

E7 Quality Assurance In Engineering Activities

a. Inspection Scope

During the week of May 20, the inspector performed a walkdown of the Unit 1 Plant Auxiliary Control Board (PACB) safety-related annunciators

LA and LB to verify the accuracy of annunciator response procedures. This consisted of a review of the following procedures and engineering drawings, including:

- ONOP 2-0030131, Rev 51, "Plant Annunciator Summary"
- Other Procedures
- Applicable Engineering Drawings
- UFSAR Section 7.5

b. Observations and Findings

As a result of the walkdowns, the following discrepancies were noted:

Procedure	Attribute	Erroneous Attribute	Correct Attribute
ONOP 2-0030131, Rev 51, "Plant Annunciator Summary"	Annunciator LA-6 "ATMOS STM DUMP ISOL VALVES MV-08-15, MV-08-17 MOTOR OVERLOAD VALVES CLOSED"	Indicated Condition "C" "Feeder breaker open to MV-08-15 or 16"	Indicated Condition "C" "Feeder breaker open to MV-08-15 or 17"
	Annunciator LA-9 "DIESEL OIL DAY TANKS 2A1, 2A2 LOW-LOW LEVEL"	Sensing Elements listed as LS-59-006A and 10A	LS-59-9A and 14A
	Annunciator LA-12 "ATM STM DUMP MV-08-18A/18B OVERLOAD/SS ISOL"	Indicated conditions, CWD reference and sensing element	This indicated condition and contacts were removed by PC/M 275-290, closed 10/28/92
	Annunciator LB-9 "DIESEL OIL DAY TANKS 2B1, 2B2 LOW-LOW LEVEL"	Sensing Elements listed as LS-59-018B and 024B	LS-59-021B and 028B
	Annunciator LB-14 "FUEL POOL HIGH/LOW LEVEL HIGH TEMP"	Sensing Element TA-4421 not listed	
	LB-10 "COMPONENT COOLING WTR SURGE TANK HIGH LEVEL COMPARTMENT B LOW LEVEL"	Sensing Element does not specify contact 71X	
	Annunciator LB-11 "PRESSURIZER LO-LO LEVEL CHANNEL Y"	Sensing Element listed as LC-1110X	Sensing Element should be LA-1110X
	Annunciator LB-12 "ATM STM DUMP MV-08-19A/19B OVERLOAD/SS ISOL"	Indicated conditions, CWD reference and sensing element	This indicated condition and contacts were removed by PC/M 275-290, closed 10/28/92
Drawing 2998-B-327 Sheet 211, Rev 14, "Component Cooling Water Shutdown Heat Exch & Surge Tank Fill Valves"		Does not show which LA annunciator alarms from LS-14-1A	

2998-B-327 Sheet 1142, Rev 7, "Plant Auxiliaries Control Board Annunciator - LA"	Annunciator LA-9	Sensing Element specified as LS-17-552A, 553A	Sensing Element should be LS-59-009A, 14A
2998-B-327 Sheet 1143, Rev 7, "Plant Auxiliaries Control Board Annunciator - LB"	Annunciator LB-9	Sensing Element specified as LS-17-552B, 553B	Sensing Element should be LS-59-021B, 028B

The inspector noted that the errors above were additional examples of errors identified in previous inspection reports which had been documented under URI 96-04-05, "Configuration Control Management." The inaccuracies noted were consistent with inaccuracies identified in previous, similar, walkdowns. The inspector noted that two inaccuracies (annunciators LA-12 and LB-12) were clearly the result of the inadequate implementation of the design change process. These inaccuracies are discussed in the context of other, similar, inaccuracies in paragraph E7.1, below.

c. Conclusions

The inspectors concluded the following with respect to annunciator panels LA and LB for the PACB:

- Annunciator response procedure inaccuracies existed of the same types identified in previous, similar, walkdowns.

- In the cases of two annunciator windows, the inaccuracies were identified to be the result of inadequate implementation of the design change process.

E7.2 PC/M Execution Issues (71707, 37551, 92901, 92903)

a. Inspection Scope

Inspection Report (IR) 96-04 identified several potential configuration control weaknesses involving inaccuracies in control room annunciator response summaries and engineering drawings. Of the deficiencies noted, one was tied to an inadequacy in the implementation of a PC/M. URI 96-04-05, "Configuration Control Management," was opened to track the issue while the inspection scope was expanded. IR 96-06 documented additional deficiencies, identified during system walkdowns, which were the result of PC/M implementation inadequacies. During the current inspection period, two additional PC/M implementation issues were identified; one, involving inaccuracies in annunciator response summaries, is described in paragraph E7.1, above; one, involving licensee-identified procedural inadequacies, is described below. The inspectors performed a review of the relevant inspection findings in an attempt to characterize the identified issues.

b. Findings

The inspectors reviewed issues identified under URI 96-04-05, "Configuration Control Management." IR 96-06 summarized recent NRC findings in the area of inaccuracies in plant procedures and drawings and stated that ten examples of alarm setpoint inaccuracies and 18 other (e.g. wrong sensing element, wrong action directed) inaccuracies in the Annunciator Response Summaries had been identified in both units' ICW and CS systems. The inspectors reviewed findings generated in IRs 96-04, 96-06, and the current reporting period to identify examples which demonstrated that design changes made to the plant resulted, through inadequate implementation, in such inaccuracies. As a result, the inspectors identified the following items:

1) IR 96-04 documented the fact that, on January 6, 1995, the licensee closed out PC/M 109-294 [Setpoint change to the Hydrazine Low Level Alarm (LIS-07-9)] without assuring that affected procedure ONOP 2-0030131, "Plant Annunciator Summary," was revised. This resulted in annunciator S-10, "HYDRAZINE TK LEVEL LO," showing an incorrect setpoint of 35.5 inches.

2) IR 96-06 documented the fact that, on May 16, 1994, the licensee closed out PC/M 341-192 [ICW Lube Water Piping Removal and CW Lube Water Piping Renovation]. The as-built Dwg. No. JPN-341-192-008 was not incorporated in Dwg. No. 8770-G-082, "Flow Diagram Circulating and Intake Cooling Water System," Rev 11, sheet 2, issued May 9, 1995, for PC/M 341-192. This resulted in Dwg. No 8770-G-082 erroneously showing valves I-FCV-21-3A & 3B and associated piping still installed.

3) IR 96-06 documented the fact that, on February 14, 1994, the licensee closed out PC/M 268-292 [ICW Lube Water Piping Removal and CW Lube Water Piping Renovation] without assuring that affected procedure ONOP 2-0030131, "Plant Annunciator Summary," was revised. This resulted in annunciator E-16, "CIRC WTR PP LUBE WTR SPLY BACKUP IN SERVICE," incorrectly requiring operators to verify the position of valves MV-21-4A & 4B following a Safety Injection Actuation System (SIAS) signal using control room indication. These valves no longer received a SIAS signal, were deenergized and had no control room position indication.

4) This inspection report (paragraph E7.1) documents the fact that, on October 28, 1992, the licensee closed out PC/M 275-290 [FIS-14-6 Low Flow Alarm and "Manual" Annunciator Deletions] without assuring that affected procedure ONOP 2-0030131, "Plant Annunciator Summary," was revised. This resulted in safety-related annunciators LA-12, "ATM STM DUMP MV-08-18A/18B OVERLOAD/SS ISOL," and "LB-12 ATM STM DUMP MV-08-19A/19B OVERLOAD/SS ISOL," incorrectly requiring operators to check the Auto/Manual switch or switches at RTGB-202 and PACB for the MANUAL position. The relay contacts which energized these annunciators based on switch position were removed to eliminate nuisance

alarms.

In addition to these findings, the licensee identified one example of a failure to include operational limitations imposed by a calculation in a plant procedure:

5) During the current inspection period, the licensee identified the fact that assumptions made in the heat load calculation supporting the Unit 1 full core offload were not appropriately factored into the applicable procedure. Specifically, PC/M 054-196, supplement 0, "St. Lucie Unit 1 Cycle 14 Reload," included, in Attachment 8, operational limitations which resulted from the heat load calculation performed to support the full core offload. These included:

   - Ensuring that initial Spent Fuel Pool (SFP) temperature was less than or equal to 106°F.

   - Ensuring that the reactor was subcritical for at least 168 hours prior to commencing the offload.

   - Verifying that the SFP high temperature alarm, which annunciated in the control room, was operable.

   - Verifying that two SFP cooling pumps were in operation.

   - Verifying that Component Cooling Water (CCW) flow to the fuel pool heat exchangers was maintained at approximately 3560 gpm when two SFP cooling pumps were operating.

   On May 12, the licensee's Quality Assurance (QA) organization identified the fact that these limitations were not included in OP 1-1600023, "Refueling Sequencing Guidelines." The offload of seven fuel assemblies had occurred by the time the deficiencies were identified. The defueling evolution was subsequently stopped, and the prerequisites were added to OP 1-1600023, "Refueling Sequencing Guidelines," as revision 62 to the procedure.

10 CFR 50 Appendix B, Criterion III, "Design Control," requires, in part, that measures be established to ensure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The licensee's Topical Quality Assurance Report, TQR 3.0, Rev 11, "Design Control," included the following provisions:

- Section 3.2.2, "Design Change Control," stated, in part, "Design changes shall be reviewed to ensure that implementation of the design change is coordinated with any necessary changes to operating procedures..."

- Section 3.2.4, "Design Verification," stated, in part, that "Design control measures shall be established to independently verify that design inputs, design process, and that the design inputs are correctly incorporated into design output."

The inspectors concluded that the examples cited above failed to meet the criteria of 10 CFR 50 Appendix B and the licensee's QA program. The inspectors found that the number of examples identified indicated that a programmatic flaw existed in the licensee's program for ensuring that material changes to the plant were reflected properly in engineering drawings and plant procedures. As such, the issues above were found to constitute five examples of one apparent violation (EEI 50-335,389/96-12-05, "Failure to Ensure Configuration Control," EA 96-249).

The licensee's QA organization performed an audit of this area and documented their findings in QSL-PCM-96-11, "PC/M Design Control." The licensee found the following with regard to the process:

- Plant procedures and instructions did not adequately define the review and comment process by plant departments impacted by PC/Ms or the resolution to those comments.

- Plant procedures and instructions did not adequately address the identification of plant procedures impacted by PC/Ms.

- Plant procedures and instructions did not adequately address the review of Safety Evaluations for impact on plant procedures and instructions (this applied to Safety Evaluations which included conditions to ensure that the assumptions in the evaluations were maintained valid).

The inspectors found the licensee's findings to be in general agreement with observations made by the NRC.

In response to the issue, the licensee adopted corrective actions which included:

- Implementing design control processes from Turkey Point, which provided more positive control over the initial reviews and documentation of required actions for PC/Ms.

- Performing reviews of all Unit 1 outage related PC/Ms to ensure that required procedural changes were identified.

- Requiring that all PC/M paperwork for modifications installed during the current Unit 1 outage be closed out prior to returning the affected system to service.

- Revalidating open items from previous PC/Ms on both units and establishing timelines for closure of the open items.

- Initiating a vertical slice inspection of selected, PRA-significant systems to ensure that the systems were properly installed and that procedures were adequate.

The inspector reviewed the results of the vertical slice inspections referenced above, performed on the EDG, High Pressure Safety Injection (HPSI), and CCW systems. The results were documented in CRs 96-1588 (Unit 1 items for Operations disposition), 96-1589 (Unit 1 items for Engineering disposition), 96-1360 (Unit 2 items for Operations disposition) and 96-1361 (Unit 2 items for Engineering disposition). In general, the licensee's findings were consistent with NRC findings in this area and included cases in which procedure-to-drawing deviations existed in valve position, cases of annunciator response summary errors existed, cases of instrument range differences between the UFSAR and design documents, and cases of configuration differences between the plant and design documents.

The inspectors found that the licensee had initiated actions to address the PC/M issues discussed above and to ensure that the as-built configuration of the plant was adequate. The overall adequacy of the licensee's actions will be determined in followup inspections to the apparent violations described above.

URI 96-04-05, "Configuration Control Management," is closed.

c. Conclusions

The inspectors concluded the following with respect to configuration controls:

That programmatic flaws resulted in one apparent violation involving the issue of configuration management and the licensee's ability to correctly translate design changes into drawings and procedures. The apparent violation included five examples:

1) One example of a failure to update an annunciator response summary when a hydrazine tank low level alarm setpoint was changed via PC/M.

2) One example of a failure to update an engineering drawing to reflect the deletion, via PC/M, of valves and piping for the Intake Cooling Water System.

3) One example of a failure to update an annunciator response summary to reflect a change, made via PC/M, which removed automatic and control room operation capability from a pair of valves.

4) One example of a failure to update an annunciator response procedure to reflect a change, made via PC/M, which removed the alarm function from an annunciator.

5) One licensee-identified example of a failure to update an operating procedure to include operational limitations imposed by a PC/M-transmitted spent fuel pool heat load calculation.

The licensee's QA organization was identifying specific areas of concern in the configuration management area. The licensee had initiated actions to address the configuration management deficiencies identified by both the NRC and the licensee's QA organization.

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 July 12. 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

Bladow, W., Site Quality Manager
Bohlke, W., Vice President, Engineering
Burton, C., Site Services Manager
Dawson, R., Business Manager
Denver, D., Site Engineering Manager
Fulford, P., Operations Support and Testing Supervisor
Holt, J., Information Services Supervisor
Johnson, H., Operations Manager
Scarola, J., St. Lucie Plant General Manager
Weinkam, E., Licensing Manager

Other licensee employees contacted included operations, engineering, maintenance, and corporate personnel.

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 64704:	Fire Protection Program
IP 71707:	Plant Operations
IP 92901:	Followup - Plant Operations
IP 92903:	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened</u>

50-389/96-12-01	EEI	Failure to Perform a 10 CFR 50.59 Safety Evaluation for CEDMCS Enclosure
50-335,389/96-12-02	EEI	Failure to Perform a 10 CFR 50.59 Safety Evaluation For Use of a Temporary Fire Pump
50-335/96-12-03	EEI	Failure to Perform a 10 CFR 50.59 Safety Evaluation For Change in Setpoints Listed in UFSAR
50-389/96-12-04	EEI	Unreviewed Safety Question Involving EDG 2B
50-335,389/96-12-05	EEI	Failure to Ensure Configuration Control

<u>Closed</u>

50-335,389/96-04-05	URI	Configuration Control Management

<u>Discussed</u>

50-335,389/96-04-09	URI	Failure to Update UFSAR

LIST OF ACRONYMS USED

ATTN	Attention
CCW	Component Cooling Water
CEDMCS	Control Element Drive Mechanism Control System
CFR	Code of Federal Regulations
CR	Condition Report
CW	Circulatory Water
DFOST	Diesel Fuel Oil Storage Tank
DPR	Demonstration Power Reactor (A type of operating license)
DWG	Drawing
EA	Enforcement Action
EDG	Emergency Diesel Generator
EEI	Escalated Enforcement Item
FIS	Flow Indicator/Switch
FO	Fuel Oil
FPL	The Florida Power & Light Company
FRG	Facility Review Group
gpm	Gallon(s) Per Minute (flow rate)
HPSI	High Pressure Safety Injection (system)
ICW	Intake Cooling Water
IR	[NRC] Inspection Report
JPN	(Juno Beach) Nuclear Engineering
LIS	Level Indicating Switch
MV	Motorized Valve
NLO	Non-Licensed Operator
No.	Number
NPF	Nuclear Production Facility (a type of operating license)
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory (NRC Headquarters Publication)
ONOP	Off Normal Operating Procedure
OP	Operating Procedure
PACB	Plant Auxiliary Control Board
PC/M	Plant Change/Modification
PDR	NRC Public Document Room
PM	Preventive Maintenance
PRA	Probabilistic Risk Assessment
PSL	Plant St. Lucie
QA	Quality Assurance
QI	Quality Instruction
QSL	Quality Surveillance Letter
SAR	Safety Analysis Report
SE	Safety Evaluation
SFP	Spent Fuel Pool
SIAS	Safety Injection Actuation System
SIT	Safety Injection Tank
St.	Saint
TQR	Topical Quality Requirement
UFSAR	Updated Final Safety Analysis Report
URI	[NRC] Unresolved Item
USNRC	Unite States Nuclear Regulatory Commission
USQ	Unreviewed Safety Question