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

Report No: 50-335/96-22, 50-389/96-22

Licensee: Florida Power & Light Co.

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

Location: 9250 West Flagler Street
Miami, FL 33102

Date: November 14, 1996

Inspectors: J. York, Reactor Inspector
C. Rapp, Reactor Inspector
M. Miller, Senior Resident Inspector

Approved by: C. Casto
Chief, Engineering Branch
Division of Reactor Safety

REWRITE THIS SECTION !!!

EXECUTIVE SUMMARY

St. Lucie Nuclear Plant, Units 1 & 2
NRC Inspection Report 50-335/96-22, 50-389/96-22

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

a. Inspection Scope

The inspectors reviewed activities associated with the plant modification used to replace the Nuclear Instrumentation drawers located in the Unit 1 control room. This modification reulted in NI channels A, B, C, and D being wired backwards due to a design error. This review included evaluations of the root causes and safety significance from a core physics view point.

b. Observations and Findings

On July 30, 1996, St. Lucie Unit 1 was operating at approximately 100 % power when reactor engineering was analyzing the data taken during power ascension and noted an anomaly in the results. The data indicated three of the four excore linear detectors measured core power moving to the top of the core during power ascension. This was an unexpected phenomena and did not agree with the trend of the power moving to the bottom of the core indicated by RPS Channel B Linear Range Detector, Control Channel #9 Linear Range Detector, and the BEACON Core Power Distribution Monitoring System. Evaluation of the data collected indicated that RPS Channels A,C,and D could have reversed (rolled) leads of the top and bottom chambers input to the RPS drawers.

The modification performed during the outage associated with this problem was No. PC/M 009-195. During the outage, the licensee replaced the power range NI drawers for the Reactor Protection System (RPS) with new Gamma Metrics drawers. This modification combined the linear power range input to the RPS and the logarithmic wide range channel into a single drawer, i.e. reduced the number of drawers on Unit 1 from eight to four. This modification increased the limits of the instruments range and replaced aging equipment.

1) Evaluation of Root Cause

A design error was responsible for the reverse connection (rolled leads) on four NI safety related drawers on Unit 1. The Controlled Wiring Diagram (CWD), no. JPN-009-195-001/002 depicted the upper Uncompensated Ion Chamber (UIC) connected to the lower UIC input at the NI drawer. The root cause noted that the designer and the lead engineer interpreted conflicting information on the existing CWDs and made an assumption.

The independent verification may have caught this error had the process been properly performed. The drawings were prepared by the lead designer with input from the lead engineer. The drawings were then checked by a second designer who had no special knowledge of the NI design. This check was essentially a drafting check. The drawings were then reviewed by the lead designer and then by the engineering supervisor.

Engineering Quality Instructions (QI) 1.7, Design Input/Verification, dated July 5, 1995, states in part that "Design verification is the process whereby a competent individual, who has remained independent of the design process, reviews the design inputs, ... and design output to verify design adequacy. This independent review is provided to minimize the likelihood of design errors in items that are important to nuclear safety." Contrary to this requirement the first reviewer could not be considered as competent because he was not an engineer as required by QI 1.7 and the lead engineer as the third reviewer could not be considered to have remained independent of this design project. One of the action items to prevent recurrence was to check all the I&C and electrical PC/M to see if all the drawing approval signatures could qualify as independent verifiers. The licensee found three out of eight open modifications where this was a potential problem, two of these modifications were electrical and one was I&C. This therefore is not an isolated case. This failure to perform independent verification according to procedure is identified as example one of an apparent violation (EEI 50-335/96-22-01, "Failure to Control the Design Process According to the Requirements of 10 CFR 50, Appendix B, Criterion III," EA 96-457).

2) Evaluation of Safety Consequences

The licensee had installed BEACON during this refueling outage to replace the older IMPAX code used for in-core flux monitoring. BEACON provided several significant improvements over IMPAX one being real-time flux profile monitoring. This improvement permitted reactor engineering to identify the NIS problem quickly and initiate prompt corrective actions.

During power operations, reactor engineering used BEACON to obtain the actual in-core flux profile. The actual in-core flux profile was then used to verify compliance with Technical Specifications and provide calibration information for the excore NIS drawers. As part of these routine surveillance, reactor engineering compares actual in-core flux profile to the in-core flux profile predicted by the core design code. Reactor engineering noted larger than normal errors between actual and predicted in-core flux profile. Because BEACON used the same neutronics engine as used in the core design code, reactor engineering could not explain the error and notified the corporate core design engineers. As part of the process to resolve these errors, it was discovered that a simplifying assumption, used to overcome limitations of the IMPAX, was not accounted for in the original design of BEACON. This simplifying assumption was used because the licensee had changed the fuel design to incorporate a longer end cap to prevent debris induced fuel failures. This longer end cap raised the overall core height by 2.64" causing an offset between detector midplane and actual core midplane. The IMPAX code assumed detector midplane was along core midplane and could not accommodate the 2.64" offset. Therefore, the licensee, after discussion with the fuel vendor (Siemans), used this simplifying assumption to essentially lower the core midplane by 2.64" so that final design output would be referenced to detector midplane; not core midplane. However, the engineer preparing the design input for BEACON was not aware of this simplifying assumption consequently BEACON was referenced to core midplane resulting in an increased error between the core design predicted in-core flux profile and actual in-core flux profile.

The licensee's root cause evaluation identified lack of cross-discipline review as the significant contributor to this design error. The inspector concurred with the licensee's evaluation. Engineering Quality Instructions (QI) 1.7, Design Input/Verification, dated July 5, 1995, states in part that "Design verification is the process whereby a competent individual, who has remained independent of the design process, reviews the design inputs, ... and design output to verify design adequacy. This independent review is provided to minimize the likelihood of design errors in items that are important to nuclear safety." Contrary to this requirement, the design inputs were not adequately reviewed by a competent individual in that the core midplane offset was not identified as a design input for BEACON. This failure to perform an adequate independent design review for the BEACON system is identified as example two of an apparent violation (EEI 50-335/96-22-01, "Failure to Control the Design Process According to the Requirements of 10 CFR 50, Appendix B, Criterion III," EA 96-457).

The safety significance of reversing the detector inputs to the NIS drawers substantially reduced the safety margin between the TM/LP trip setpoint and the analysis limit even considering the increased TM/LP margin to the trip setpoint due to actual core operating conditions. The safety impact of the failure to identify the core and detector midplane offset on TM/LP or LPD safety limits was minimal.

The licensee also identified that BEACON was placed into service on Unit 1 without any benchmarking against IMPAX, the on-line core performance monitoring code BEACON was replacing. Instead, BEACON was installed on Unit 2 and benchmarked against CECORE, which did not require any modifications to accommodate the core midplane offset. Technical Specification 6.8, Procedures and Programs, paragraph 6.8.1 requires in part that written procedures recommended in Appendix A of Regulatory Guide 1.33 revision 2, February 1978, shall be established, implemented... Engineering Quality Instruction (QI) 3.7, Computer Software Control, revision 1, Section 5.4. requires that SQA1 software shall be validated and verified (V&V'ed) in accordance with Section 5.6. Section 5.6 states that new software shall be V&V'ed prior to use. V&V includes the use of test cases to ensure the new software produces correct results. Item 4 of Section 5.6 states that technical adequacy shall be determined by comparing the test case to results from alternative methods such as functionally equivalent and previously validated software. In the case of BEACON, IMPAX would have been functionally equivalent software. Benchmarking BEACON against IMPAX may have identified the design error concerning core midplane offset because the two codes would not have yielded the same results. Contrary to this requirement, BEACON was placed into service on Unit 1 without benchmarking against IMPAX. This is identified as an apparent violation (EEI 50-335/96-22-02, "Failure to Follow Procedure for Placing the BEACON System in Service," EA 96-457).

3) Corrective Actions

The inspectors reviewed the possibilities that the licensee had to determine that a design control problem existed by reviewing information from QA audits and from information in the root cause analysis. The QA monitoring report, QSL-PM-96-17, had a finding concerning problems with the NI modification. This finding discussed the large number of Change Request Notices (CRN), a number of scope changes, and the writing of four separate work orders for troubleshooting purposes. Fourteen Condition Reports (CR), four by the QA organization, were written during the implementation of this modification. Besides this being a qualitative indicator of a problem with implementing the modification, the QA report noted that "workers on the job frequently complained about the unmanageability of the implementation documentation".

The NRC SRI had a discussion with one o. the Reactor Operators (RO) who had noted and questioned an anomonly in the reading of the linear NI values at or below five percent power. In a review of the licensee's root cause report in the area of personnel interviews, it was noted that ROs questioned the readings below five percent, and on increasing power above five percent. In addition, this report states that a Reactor Engineering representative also questioned the difference when in the increasing power range of 70 to 90. Therefore, the results of the areas mentioned in the QA monitoring report and opportunities afforded by the questions from the ROs/Reactor Engineers should have allowed the licensee to identify design problems and to have taken effective corrective action. The failure to take effective corrective action is identified as an apparent violation (EEI 50-335/96-22-03, "Failure to Take Effective Corrective Action to Prevent a Design Error," EA 96-457).

c. Conclusions on Conduct of Engineering

As a result of this inspection three violations were identified for the engineering area. The first violation had two examples of problems with the licensee's design control process. One example involved the failure to independently verify Controlled Wiring Diagrams which were in error and resulted in wiring the NI drawers backwards. Another example resulted in the core midplane offset not being identified as a design input to BEACON, the computer program used for real time flux profile monitoring. The second violation involved the failure to validate and verify a new computer program before placing it in use. The third violation involved the failure to take effective corrective action to prevent a design error from being implemented.

M.1 Conduct of Maintenance

a. Inspection Scope

The inspectors reviewed the maintenance activity for replacing the no. 6 detector for channel B of the Linear Range Detector. The maintenance activity allowed reversal of the field cables.

b. Observations and Findings

All four of the RPS Linear Range Detectors had the connectors reversed as previously discussed but the B channel unlike the other three channels was giving the correct data. At the same time that the drawers were being replaced on Unit 1, the detector for channel B (detector no. 6) was being replaced as a maintenance activity. During connection of the field cables, the connections were reversed for the upper and lower detection chambers, thereby causing the B channel to record properly.

The root cause for the swap of the cables was that the new detector had different labeling than the existing cables. The existing cables were labeled TOP SIG and BOT SIG, and the new detector had A and B. The inspectors discussed this maintenance job with the I&C supervision who had supervised the latter part of this maintenance project. Several opportunities were presented to the maintenance personnel, one when the detectors were checked out in the warehouse and a second time when this condition was noted in the field.

Maintenance personnel should have resolved the labeling problem by writing a Condition Report (CR) and having a formal resolution. Technical Specification 6.8, Procedures and Programs, paragraph 6.8.1 requires in part that written procedures recommended in Appendix A of Regulatory Guide 1.33 revision 2, February 1978, shall be established, implemented...Administrative Procedure No. 0006130, Condition Reports, rev. 4, dated March 22, 1996, Par. 8.1.1.A states in part that "Any individual who becomes aware of a problem or discrepant condition ... should initiate a CR. If doubt exists, a CR form should be initiated". This failure to comply with the requirements of the administrative procedure is identified as an apparent violation (EEI 50-335/96-22-04, "Failure to Initiate a Condition Report for Labeling on Safety Related Detectors," EA-457).

c. <u>Conclusions on the Conduct of Maintenance</u>

The I&C maintenance personnel reversed the field cables for the no. 6 channel B detector. The cables were labeled differently than the existing ones and the maintenance personnel had two occasions to question this condition by initiating a Condition Report. An apparent violation for failure to conform to administrative procedure for writing a CR was identified.

REWRITE LIST!!!!!!!!!!!!!!!!!!!!!!!!!!!!!!!!!!!!!!!

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 37550: Engineering

REWRITE THIS !!

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

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

REWRITE LIST!!

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

R-1216320B

RUN DATE: 11/13/96

SAFETY ISSUE MANAGEMENT SYSTEM

GENERIC ISSUES WITH TI GUIDANCE AVAILABLE - (OPEN)

PLANT	DOCKET	TYPE	NSSS SUPPLIER	THERMAL CAPACITY	LICENSEE	STATE	REGION	OL ISSUE	OL EXPIRATION
ST LUCIE 1	05000335	PWR	COMB	0802 MWT	FLORIDA POWER & LIGHT CO.	FL	2	03/76C	03/16

APPLICABLE ISSUES		TAC #	LIC ACT CMP DATE	LICENSEE IMPL DATE	VERIFY COMPLETE	TIREF NUMBER	INSPECTION REPORT NUMBERS	VERIFY ACCESSION
A-46	B105	M69483	04/96	04/96		2515/124		

SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS

JUST GOT RESOLVED. I'LL BE DOING INSP TO CLOSE OUT ISSUE PER CLOSURE LETTER

This is supposed to be the printout from IFS showing SIMS stuff open, URIs greater than 6 months old, and IFIs greater than 2 years old. Please check for accuracy, mark up, and return to Julian when you can. Thanks

Caudle

FF16

R-1216320B

RUN DATE: 11/13/9

SAFETY ISSUE MANAGEMENT SYSTEM

GENERIC ISSUES WITH TI GUIDANCE AVAILABLE - (READY)

PLANT	DOCKET	TYPE	NSSS SUPPLIER	THERMAL CAPACITY	LICENSEE	STATE	REGION	OL ISSUE	OL EXPIRATIO
ST LUCIE 1	05000335	PWR	COMB	0802 MWT	FLORIDA POWER & LIGHT CO.	FL	2	03/76C	03/16

APPLICABLE ISSUES		TAC #	LIC ACT CMP DATE	LICENSEE IMPL DATE	VERIFY COMPLETE	TIREF NUMBER	INSPECTION REPORT NUMBERS	VERIFY ACCESSION
GL-89-10	B110	M75721	06/90C	02/95C	03/97	2515/109	91-18 94-11	
SAFETY-RELATED MOTOR-OPERATED VALVE TESTING AND SURVEILLANCE								
MPA-B041	B041	M43853	05/84C	02/85C		2515/062		
FIRE PROTECTION - FINAL TECH SPECS (INCLUDES SER SUPPLEMENTS)								

Z ?

89-10 INSPECTION TO CLOSE GL
GOT DEFFERRED AT PPR TILL
NEXT PPR PERIOD (AFTER APRIL)

R-1216320B

RUN DATE: 11/13/96

SAFETY ISSUE MANAGEMENT SYSTEM

GENERIC ISSUES WITH TI GUIDANCE AVAILABLE - (READY)

PLANT	DOCKET	TYPE	NSSS SUPPLIER	THERMAL CAPACITY	LICENSEE	STATE	REGION	OL ISSUE	OL EXPIRATION
ST LUCIE 2	05000389	PWR	COMB	0000 MWT	FLORIDA POWER & LIGHT CO.	FL	2	03/83C	04/23

APPLICABLE ISSUES		TAC #	LIC ACT CMP DATE	LICENSEE IMPL DATE	VERIFY COMPLETE	TIREF NUMBER	INSPECTION REPORT NUMBERS	VERIFY ACCESSION
GL-89-10	B110	M75722	06/90C	03/96C	03/97	2515/109	91-18 94-11	

SAFETY-RELATED MOTOR-OPERATED VALVE TESTING AND SURVEILLANCE

SITE: STL ST LUCIE

ST R	RPT/IFS/ EA/ NBR	SEQ-NO NOV-ID	ITEM TYPE	REF NBR / EA-NBR	SEVERITY SUPLMNT	SALP AREA	REPORT / EVENT DT	STS	CREATE DATE	CLOSEOUT PRJ/ACT*	CLSOUT ORG NO	CLSOUT EMP	UPDATING INSPECTION REPORTS
STL1	I 94-008	03	URI			ENG	04/08/94	0	04/13/94		2320	JVL	
TITLE:	QUALITY LEVEL OF PORV AND SRV DISCHARGE PIPING									PROC NUMBER: 37700			
STL1	I 96-001	02	URI			MAINT	03/18/96	0	03/22/96		2320		96-001 CLOSED
TITLE:	IMPROPER HEALTH PHYSICS PRACTICES.									PROC NUMBER: 62703			
STL1	I 96-004	05	URI			OPS	04/29/96	0	05/08/96		2230	CLOSED	
TITLE:	CONFIGURATION CONTROL MANAGEMENT									PROC NUMBER: 71707			
STL2	I 96-004	05	URI			OPS	04/29/96	0	05/08/96		2230		

NOTE: DEFINITION FOR CHARACTER PRECEEDING REPORT NO: I = INSPECTION REPORT NUMBER, E = EA NUMBER (ENFORCEMENT / NOV ITEM)
N = IFS NUMBER, NUMBER USED TO IDENTIFY NON-INSPECTION ITEMS

TOTAL OPEN ITEMS ==>> 4
TOTAL OPEN REPORT SEQUENCES ==>> 3

*IF ITEM IS OPEN, PROJECTED CLOSEOUT DATE IS REPORTED
IF ITEM IS CLOSED, ACTUAL CLOSEOUT DATE IS REPORTED

SITE: STL ST LUCIE

UNIT ABBR	RPT/IFS/ EA/ NBR	SEQ-NO NOV-ID	ITEM TYPE	REF NBR / EA-NBR	SEVERITY SUPLMNT	SALP AREA	REPORT / EVENT DT	STS	CREATE DATE	CLOSEOUT PRJ/ACT*	CLSOUT ORG NO	CLSOUT EMP	UPDATING INSPECTION REPORTS
STL2	I 93-025	01	IFI				12/01/93	O	12/07/93		2350		
TITLE:	REVIEW OPERABILITY OF UNIT 2 MOV MV-08-13 DURING THE									PROC NUMBER: 37700			
STL1	I 94-008	03	URI			ENG	04/08/94	O	04/13/94		2320	JVL	
TITLE:	QUALITY LEVEL OF PORV AND SRV DISCHARGE PIPING									PROC NUMBER: 37700			
STL1	I 94-011	01	VIO		4/1	ENG	06/13/94	O	06/08/94		2350		
TITLE:	INADEQUATE CORRECTIVE ACTION FOR MOVS WHICH STALLED									PROC NUMBER: 2515/109			
STL1	I 94-011	02	IFI			ENG	06/13/94	O	06/08/94		2350		
TITLE:	INADEQUATE RECOGNITION OF MOV TEST PRESSURE AND FLOW									PROC NUMBER: 2515/109			
STL1	I 94-011	03	IFI			ENG	06/13/94	O	06/08/94		2350		
TITLE:	LACK OF INSTRUCTIONS OR GUIDANCE FOR TRENDING									PROC NUMBER: 2515/109			
STL2	N 94-332		LER	94-006-01			07/14/94	O	10/11/94		2230	✓ CLOSED	
TITLE:	TRIP CIRCUIT BREAKER FAILURE DUE TO A BROKEN PIECE OF									PROC NUMBER:			
STL1	N 95-005		LER	94-009-00			11/22/94	O	01/04/95		2230	✓ SHOULD BE CLOSED	
TITLE:	INADVERTENT SAFETY INJECTION ACTUATION SIGNAL/CONTAINME									PROC NUMBER:			

NOTE: DEFINITION FOR CHARACTER PRECEEDING REPORT NO: I = INSPECTION REPORT NUMBER, E = EA NUMBER (ENFORCEMENT / NOV ITEM)
N = IFS NUMBER, NUMBER USED TO IDENTIFY NON-INSPECTION ITEMS

TOTAL OPEN ITEMS ==========>> 7
TOTAL OPEN REPORT SEQUENCES ====>> 7

*IF ITEM IS OPEN, PROJECTED CLOSEOUT DATE IS REPORTED
IF ITEM IS CLOSED, ACTUAL CLOSEOUT DATE IS REPORTED

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SUPERCEDED
BY REV 1
REV 1 CLOSED
IN 95-09

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

November 14, 1996

MEMORANDUM TO: Roy P. Zimmerman, Associate Director
Associated Director for Projects

FROM: Ellis W. Merschoff, Director
Division Reactor Projects

SUBJECT: INCONSISTENCIES BETWEEN NRC POSITIONS IN RESPONSE TO TIA 95-013 AND NRC INSPECTION MANUAL CHAPTER, PART 9900 INTERIM GUIDANCE ON 10 CFR 50.59, ISSUED IN APRIL 1996.

Attached is a copy of Florida Power and Light (FPL) response (L-96-254 dated 10/21/96) to a violation involving the 2B Emergency Diesel Generator oil line unreviewed safety question (NRC Special Inspection Report Nos. 50-335, 389/96-12 (EA 96-236 and 96-249)). While FPL agreed to the violation, FPL identified inconsistencies between NRC positions in response TIA 95-013 and NRC Inspection Manual Chapter, Part 9900 Interim Guidance on 10 CFR 50.59 issued in April 1996. NRC Inspection Manual Chapter, Part 9900 (Page 3, Paragraph 4) specifically states that "... the staff has found compensating effects such as changes in administrative controls acceptable in offsetting uncertainties and increases in the probability of occurrence or consequences of an accident previously evaluated in the safety analysis report or reductions in margin of safety, provided the potential increase or reductions in margins are negligible." On the other hand, NRC positions in response to TIA 95-013 suggests that compensatory measures can no longer be credited to offset small potential increases in probability. Therefore, the licensee requests that NRC resolve the differences in interpretation and apparent inconsistencies.

Since this issue is related to the current efforts to review the implementation of 10 CFR 50.59, it is provided to you for appropriate action.

Docket Nos. 50-335 and 50-389

Attachment: As stated

CONTACT: Kerry D. Landis, DRP/Branch 3
404 331-5509

cc: S. Ebneter
L. Reyes, RII
J. Johnson, RII
F. Hebdon, NRR
F. Gillespie, NRR
L. Wiens, NRR
K. Landis, RII
C. Julian, RII
A. Gibson, RII
C. Casto, RII
M. Miller, RII

FF/5

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Florida Power & Light Company, P.O. Box 14000, Juno Beach, FL 33408-0420

FPL

OCT 2 1 1996

L-96-254
10 CFR 2.201

'96 OCT 28 AM 1:58

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Re: St. Lucie Units 1 and 2
Docket No. 50-335 and 50-389
Reply to a Notice of Violation
NRC Special Inspection Report 96-12 (EA 96-236 and 96-249)

Florida Power and Light Company has reviewed the subject Notice of Violation and, pursuant to 10 CFR 2.201, the responses to the violations are attached.

Very truly yours,

J R Gattle for

T. F. Plunkett
President
Nuclear Division

TFP/JAS/EJW

Attachment

cc: Stewart D. Ebneter, Regional Administrator, USNRC Region II
Senior Resident Inspector, USNRC, St. Lucie Plant

Attachment

9610300138

STATE OF FLORIDA)
) ss.
COUNTY OF PALM BEACH)

J. A. Stall being first duly sworn, deposes and says:

That he is Vice President, St. Lucie Plant, of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

[illegible signature]

J. A. Stall

STATE OF FLORIDA

COUNTY OF Palm Beach

Sworn to and subscribed before me

this 21 day of October, 19 96

by J. A. Stall, who is personally known to me.

Roberta S Economy

Name of Notary Public - State of Florida

ROBERTA S. ECONOMY
MY COMMISSION # [illegible] EXPIRES
June 1, 1997
BONDED THRU TROY FAIN INSURANCE, INC.

(Print, type or stamp Commisioned Name of Notary Public)

VIOLATION I:

10 CFR 50.59, "Changes, Tests and Experiments," provides, in part, that the licensee may make changes in the facility as described in the Safety Analysis Report (SAR) without prior Commission approval, unless the proposed change involves an unreviewed safety question. A proposed change shall be deemed to involve an unreviewed safety question if the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR may be increased, if a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR may be created, or if the margin of safety as defined in the basis for any technical specification is reduced.

Contrary to the above, in July 1995, the licensee made a change to the facility which involved an unreviewed safety question without prior Commission approval. Specifically, the 2B Emergency Diesel Generator (EDG) fuel oil line was manually isolated to secure a through-wall fuel oil leak. In taking this action, the licensee introduced two new failure modes for the 2B EDG, which both increased the probability of occurrence of a malfunction of the EDG above that previously evaluated in the SAR and the possibility for malfunction of a different type than any evaluated previously in the SAR, resulting in an unreviewed safety question. (01013)

This is a Severity Level III violation (Supplement I)

RESPONSE I:

1. FPL concurs with the violation.

2. REASON FOR THE VIOLATION

The cause of the violation was that FPL procedural guidance for performing 10 CFR §50.59 evaluations in place at the time of the violation was not consistent with the NRC's interpretation of the regulation.

The subject evaluation was prepared in accordance with the "Nuclear Engineering Department Guidance for Performing 10 CFR 50.59 Safety Evaluations." The primary basis for this engineering procedure was NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluations," which is the generally accepted industry standard on the subject. In July 1995, both the FPL procedure and NSAC-125 allowed a conclusion of no increase in probability if the increase was determined to be insignificant.

The subject 10 CFR §50.59 safety evaluation allowed plant operation with the 2B diesel fuel oil transfer pump discharge isolation valve (normally locked open) in the closed position in order to isolate a leak in the underground portion of the piping. Compensatory actions were required by the evaluation to ensure the valve would be opened in the event of an EDG start. These compensatory actions were consistent with the guidelines of NRC Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability" for the use of manual action in place of automatic action. As noted in the FPL evaluation, two new failure modes and a slight increase in the probability of a component failure were created. Pursuant to the FPL procedure, the evaluation concluded that this slight increase in probability was insignificant and that no unreviewed safety question existed as a result of the proposed plant configuration.

3. CORRECTIVE STEPS TAKEN AND THE RESULTS ACHIEVED

The unreviewed safety question identified in NRC Inspection Report 96-12 was associated with the closing of the diesel fuel oil transfer pump discharge isolation valve. Compliance was re-established upon restoring the valve to its normally open position following replacement of the leaking underground piping downstream of the valve. The replacement of the leaking piping, restoration of the valve to the open position, and return of the 2B EDG to OPERABLE status were completed by November 25, 1995.

4. CORRECTIVE STEPS TO AVOID FURTHER VIOLATIONS

A. FPL Engineering issued a Technical Alert to engineering personnel on March 6, 1996, informing them that when performing 10 CFR §50.59 evaluations, any quantified increase in the probability of occurrence of accidents or any quantified increase in the probability of occurrence of a malfunction of equipment important to safety must be considered an unreviewed safety question.

B. Revision 1 to the "Nuclear Engineering Guidance for Performing 10 CFR 50.59 Safety Evaluations" was issued on May 17, 1996. This revision proceduralized the requirement stated in the Technical Alert discussed in 4.A. above.

C. FPL will evaluate the need for further procedural revisions taking into account the NRC's position documented in TIA 95-013 and the latest industry guidance on performing 10 CFR §50.59 evaluations. This action will be completed within three months following the issuance of NRC and industry guidance on performing 10 CFR §50.59 evaluations.

5. Full compliance was achieved by November 25, 1995, with the completion of Item 3 above.

6. ADDITIONAL INFORMATION

While FPL concurs with the violation as cited, it was not until August 19, 1996, that FPL had the benefit of reviewing the NRC memorandum dated July 30, 1996, which responded to a Technical Assistance Request (TIA 95-013). The response to TIA 95-013 illustrates the existing confusion surrounding interpretation of 10 CFR §50.59.

The issue of concern deals with the Staff's position on the introduction of new failure modes as they relate to permitted compensatory actions. The NRR response to TIA 95-013 contains a narrower interpretation of the permissible use of administrative controls, specifically compensatory actions, when compared to a previous NRC position. Specifically, NRC Inspection Manual, Part 9900 interim guidance on 10 CFR §50.59, issued in April 1996. Part 9900 (pg. 3, paragraph 4) states that "...the staff has found compensating effects such as changes in administrative controls acceptable in offsetting uncertainties and increases in the probability of occurrence or consequences of an accident previously evaluated in the SAR or reductions in margin of safety, provided the potential increase or reductions in margin are negligible."

On the other hand, the response to TIA 95-013 suggests that compensatory measures can no longer be credited to offset small potential increases in probability. In the case of the FPL 10 CFR §50.59 evaluation, a compensatory operator action was used in place of an automatic function. Specifically, the response to the TIA asserts that "an unreviewed safety question exists because the proposed change introduces a new procedure and associated malfunction of a different type (operator error)..." The response to the TIA further asserts that "[T]n general, the introduction of compensatory measures suggests that there is an unreviewed safety question for which compensation is needed, hence, a 50.90 submittal should be prepared by the licensee and evaluated by the staff to determine whether the compensation is adequate." This position conflicts with the position set forth in the April 1996 Part 9900 guidance.

The new position also has implications for the NRC's operability guidance in Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability." Under this guidance, NRC recognizes that substitution of manual action for automatic action may be acceptable under certain circumstances. Additionally, in an NRC letter to Northeast Nuclear Energy Company dated October 21, 1994 (John F. Stolz to Richard M. Kacich), it is stated that, "[I]f an operability conclusion is made based upon implementing compensatory actions resulting in a change to the facility or procedures as described in the FSAR, an evaluation pursuant to §50.59 must be performed..." However, as stated above, the new position taken in the response to the TIA, with regard to compensatory measures, would appear to foreclose the possibility that such a change could be made pursuant to 10 CFR §50.59.

The inconsistencies between the positions set forth in the TIA response, NRC Inspection Manual Part 9900 guidance, and docketed correspondence illustrate the current state of confusion regarding the interpretation of 10 CFR §50.59. In the response to the TIA, the Staff states that it "...is in the process of better defining what constitutes appropriate use of compensatory measures in 10 CFR 50.59 safety evaluations." Further attesting to the currently evolving state of 10 CFR §50.59 interpretation is NRC's "Action Plan for Improvements to 10 CFR §50.59 Implementation and Oversight," from James M. Taylor to Chairman Jackson, dated April 15, 1996. The action plan recognizes several issues in need of clarification and that a final paper to the Commission on the action plan is not scheduled for issuance until February 1997.

FPL respectfully requests that, in light of the position on the introduction of new failure modes through manual operator action expressed in the response to the TIA, the Staff resolve, in a timely manner, the differences in interpretation and apparent inconsistencies that exist. The Staff's resolution of these differences will permit FPL and other licensees to properly implement the requirements of 10 CFR §50.59 and 10 CFR §50.90 in day-to-day plant operations.

VIOLATION II A:

10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criterion III requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis for safety-related structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions.

Florida Power and Light Company Topical Quality Assurance Report, TQR 3.0, Revision 11 implements these requirements. Section 3.2, "Design Change Control," provides, in part, that design changes shall be reviewed to ensure their implementation is in each case, coordinated with any necessary changes to operating procedures. In addition, Section 3.2.4, "Design Verification," provides, in part, that design control measures shall be established to verify the design inputs, design process, and that the design inputs are correctly incorporated into the design output.

Contrary to the above, the licensee failed to coordinate design changes with the necessary changes to operating procedures as evidenced by the following examples:

1. Plant Change/Modification (PC/M) 109-294, "Setpoint Change to the Hydrazine Low Level Alarm (LIS-07-9)," was completed on January 6, 1995, without ensuring that affected Procedure ONOP 2-0030121, "Plant Annunciator Summary," was revised. This resulted in Annunciator S-10, "HYDRAZINE TK LEVEL LO," showing an incorrect setpoint of 35.5 inches in the procedure.

2. PC/M 268-292, "Intake Cooling Water Lube Water Piping Removal and Circulatory Water Lube Water Piping Renovation," was completed on February 14, 1994, without ensuring that affected Procedure ONOP 2-0020131, "Plant Annunciator Summary," was revised. This resulted in the instructions for Annunciator E-16, "CIRC WTR PP LUBE SPLY BACKUP IN SERVICE," incorrectly requiring operators to verify the position of valves MV 21-4A and 4B following a safety injection actuation system signal to ensure they were de-energized and had no control room position indication.

3. PC/M 275-290, "Flow Indicator/Switch Low Flow Alarm and Manual Annunciator Deletions," was completed on October 28, 1992, without ensuring that affected Procedure ONOP 2-0030131, "Plant Annunciator

Summary," was revised. This resulted in the instructions for safety-related Annunciators LA-12, "ATM STM DUMP MV-08-18A/18B OVERLOAD/SS ISOL," and LB-12, "ATM STM DMP MV-08-19A/19B OVERLOAD/SS ISOL," incorrectly requiring operators to check Auto/Manual switch or switches for the manual position. (02014)

This is a Severity Level IV violation (Supplement I).

RESPONSE II A:

1. FPL concurs with the violation.

2. REASON FOR THE VIOLATION

The cause of the violation was an inadequate configuration control process which failed to ensure that procedures and processes affected by plant modifications were identified and updated in a timely manner as required to accurately reflect the modifications made in the plant.

A contributing factor to this violation was that, at the time of implementation of the plant modifications in the examples to the violation, there was a general acceptance by plant management of routine backlogs for open items related to plant modifications.

3. CORRECTIVE STEPS TAKEN AND THE RESULTS ACHIEVED

The St. Lucie Plant Annunciator Summary Procedure, ONOP 2-0030131, was revised to correct the three discrepancies identified in this violation. This action was completed on July 5, 1996.

4. CORRECTIVE STEPS TO AVOID FURTHER VIOLATIONS

A. A self-assessment of the plant modification front-end review process was conducted, which included benchmarking with FPL's Turkey Point Plant. In addition, a Quality Assurance audit of the design control process was conducted, which supported the results of the self-assessment.

B. St. Lucie Design Control Procedure QI 3-PR/PSL-1 was revised to incorporate a positive check for completion of procedure updates prior to system turnover and to ensure updating and tracking of affected procedures prior to restoring a modified system to service. The revised QI 3-PR/PSL-1 process was first implemented at St. Lucie Plant during the Summer 1996 St. Lucie Unit 1 refueling outage. Procedure QI 3-PR/PSL-1 was further revised, post-Unit 1 refueling outage, to incorporate the results of the self-assessment discussed in 4.A. above.

C. The Configuration Management Group at St. Lucie Plant was reorganized and restructured, with the addition of staffing and supervision to support the new plant modification review process and to continue to optimize process control and design/procedure integration.

D. Plant management expectations and requirements for thorough review, processing, and closeout of modification-related action items, and the documentation of these actions prior to restoring a modified system to service, were communicated to plant personnel via Items 4.B. and 4.C., above. The plant's acceptance of backlogged modification-related action items was replaced with an unambiguous requirement to fully process modification documentation to closure prior to declaring restoration of operability.

5. Full compliance was achieved on July 5, 1996, with the completion of Item 3 above.

VIOLATION II B:

10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criterion III requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis for safety-related structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions.

Florida Power and Light Company Topical Quality Assurance Report, TQR 3.0, Revision 11 implements these requirements. Section 3.2, "Design Change Control," provides, in part, that design changes shall be reviewed to ensure their implementation is in each case, coordinated with any necessary changes to operating procedures. In addition, Section 3.2.4, "Design Verification," provides, in part, that design control measures shall be established to verify the design inputs, design process, and that the design inputs are correctly incorporated into the design output.

Contrary to the above, the licensee failed to assure that the design of the Circulating and Intake Cooling Water System was correctly translated into plant drawings. Specifically, during implementation of PC/M 341-192, "Intake Cooling Water Lube Water Piping Removal and Circulatory Water Lube Water Piping Renovation," the as-built Drawing No. JPN-341-192-008 was not incorporated into Drawing No. 8770-G-082, "Flow Diagram Circulating and Intake Cooling Water System," Revision 11, Sheet 2, issued May 9, 1995, for PC/M 341-192. This resulted in Drawing No. 8770-G-082 erroneously showing valves 1-FCV-21-3A and 3B and associated piping as still installed. (03014)

This is a Severity Level IV violation (Supplement I).

RESPONSE II B:

1. FPL concurs with the violation.

2. REASON FOR THE VIOLATION

The cause of the violation was cognitive personnel error by utility drawing update personnel who failed to incorporate as-built drawing number JPN-341-192-008 into drawing number 8770-G-082, Revision 11, Sheet 2, which resulted in an inaccurate documentation of design changes made to the plant.

Several additional factors contributed to the event:

A. The drawing update discrepancy noted above was not identified during subsequent independent review by utility personnel prior to release.

B. For the drawing that was not updated, a discrepancy was observed in the computer based drawing update tracking program in place at the time, regarding the date of transmittal of the subject drawing to document control. This discrepancy in transmittal date provided a source of confusion regarding actual drawing status at the time of transmittal, and contributed as a causal factor to the violation.

C. The missed drawing update in this event was associated with a plant modification which was implemented over a long period of time, which also contributed to the event.

D. Accountabilities and responsibilities associated with the Drawing Update checker and verifier roles were informally communicated at the time drawing verification was initially performed.

3. CORRECTIVE STEPS TAKEN AND THE RESULTS ACHIEVED

Drawing number 8770-G-082, Sheet 2, was revised to incorporate as-built drawing number JPN-341-192-008, which deleted valves 1-FCV-21-3A and 3B. This revision was issued on April 24, 1996.

4. CORRECTIVE STEPS TO AVOID FURTHER VIOLATIONS

A. A complete review of plant change/modification (PC/M) 341-192 was conducted. Three additional drawing errors were discovered and all drawings have been corrected and reissued.

B. This event was reviewed with drawing update personnel to provide training and ensure understanding of the responsibilities, accountabilities, and expectations of personnel involved in the process of drawing update.

C. To ensure proper updating, and to generically assess the potential for other errors resulting from other causes, a sample of updated drawings from previously implemented PC/Ms is being reviewed. This sample includes drawings exhibiting a potential discrepancy between status transmittal date, as described above, and also drawings associated with PC/Ms which were implemented over an extended period of time. This action will be complete by December 30, 1996.

D. The computer based system used to track drawing updates was converted to a new system in 1996. The new system allows personnel checking drawing updates to print a complete list of all drawing updates required for a given modification. The accountabilities and responsibilities associated with the Drawing Update checker and verifier are better defined in that drawing update personnel are required to use this list when verifying the drawing updates performed by the drafter for a given modification package. The use of this list aids personnel in verifying that all applicable revision requirements for a given drawing have been incorporated.

5. Full compliance was achieved on April 24, 1996, with the completion of Item 3 above.