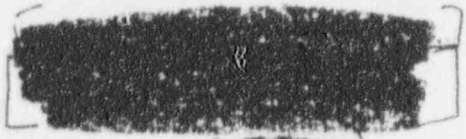
From: Beverly Michael, R2
To: ATP2(LJW2) L WATSON, R2
Date: 3/18/96 9:44am
Subject: st. Lucie operator -Reply



Docket No([redacted])

License No([redacted])

E46



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01-08-97 05:25p Directory H:\196OPEN.ENF\96040STL.DIR*.*
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| FINALSIG | SDE | 40,801 | 04-01-96 03:06p | NAMEADDR | OPR | 1,010 | 03-21-96 04:28p |
| RABREIF | . | 96,672 | 03-11-96 03:42p | REPORT | . | 142,495 | 03-11-96 03:23p |

## 15.2.4 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION - BORON DILUTION EVENT

### 15.2.4.1 Identification of Causes

The chemical and volume control system (CVCS) described in Section 9.3.4 regulates both the chemistry and the quantity of coolant in the reactor coolant system. Changing the boron concentration in the reactor coolant system is a part of normal plant operation, compensating for long-term reactivity effects, such as fuel burnup, xenon buildup and decay, and plant startup and cooldown. For refueling operations, borated water is supplied from the refueling water tank, which assures adequate shutdown margin. An inadvertent boron dilution in any operational mode adds positive reactivity, produces power and possibly temperature increases, and, in Modes 1 and 2 (startup and power operations) can cause an approach to both the DNBR and CTM limits.

Boron dilution is conducted under strict administrative procedures which specify permissible limits on the rate and magnitude of any required change in boron concentration. Boron concentration in the reactor coolant system can be decreased either by controlled addition of unborated makeup water with a corresponding removal of reactor coolant (feed and bleed) or by using the deborating ion exchanger. The deborating ion exchanger is normally used for boron removal when the boron concentration is low (<ppm) and the feed-and-bleed method becomes inefficient. A boronometer is located in a line upstream of the deborating and purification ion exchangers in the CVCS. This instrument provides a continuous measure of boron concentration and high-low boron concentration alarms.

During normal operation, concentrated boric acid solution is mixed with demineralized makeup water to the concentration required for proper plant operation and is automatically introduced into the volume control tank in response to a low water level signal from the volume control. To effect boron dilution, the makeup controller mode selector switch must be set to "Dilute" and the demineralized water batch quantity selector set to the desired quantity. When the specific amount has been injected, the demineralizer water control valve is shut automatically.

Dilution of the reactor coolant can be terminated by isolation of the makeup water system, by stopping either the makeup water pumps or the charging pumps, or by closing the charging isolation valves. A charging pump must be running in addition to a makeup water pump for boron dilution to take place.

The CVCS is equipped with the following indications and alarm functions, which will inform the reactor operator when a change in boron concentration in the reactor coolant system may be occurring:

a) Boronometer high and low alarms and concentration indication

b) Volume control tank level indication and high and low alarms

c) Makeup flow indication and alarms

d) Volume control tank isolation.

Changes in boron concentration while the reactor is on automatic control at full power are compensated for by repositioning the CEA's. However, to assist the reactor operator in maintaining an adequate shutdown margin, CEA insertion below a position that would provide a minimum of one percent shutdown margin (assuming one stuck CEA) is accompanied by control room alarms. Because of the procedures involved and the numerous alarms and indications available to the operator, the probability of a sustained or erroneous dilution is very low.

15.2.4.2 Analysis of Effects and Consequences

15.2.4.2.1 Method of Analysis

The time required to achieve criticality from a subcritical condition due to boron dilution is based on the initial and critical boron concentrations, the boron reactivity worth, and the rate of dilution. Reactivity increase rates due to boron dilution are based on the boron worth and the dilution rate.

Cases have been analyzed for all six operational modes, i.e., power operation, startup, hot standby, hot shutdown, cold shutdown, and refueling.* In each case, it is assumed that the boron dilution results from pumping unborated demineralized water into the reactor coolant system at the maximum possible rate of 132 gpm (3 x 44 gpm per charging pump) and that the boron concentrations are uniform at all times.

The boron dilution rate is calculated by CESEC for all cases except dilution during refueling. CESEC described in Section 15.1.4-1 divides the reactor coolant system into 15 control volumes with the continuity equation being satisfied by all nodes. The charging rate of non-borated water and the boron content of the system are inputs to CESEC. The maximum dilution rate (10.5 ppm/minute) occurs at the initiation of the transient. For dilution during refueling the reactor coolant system is assumed to be one control volume with the boron concentration calculated by: the time rate of change of boron equals flow in times the boron concentration minus flow out times boron concentration.

The uniformity of the boron concentration can be assured for the different modes of operation as follows:

a) During refueling

Prior to cooldown, the reactor coolant system boron concentration is increased to a minimum of 1720 ppm. The boron is mixed by the reactor coolant system pumps. Because the boron is chemically dissolved in the reactor coolant, it will not precipitate. The only possible means of obtaining a nonuniform solution is by the addition of demineralized water via the charging pumps. However, because the maximum water

* An additional boron dilution event would be via the Iodine Removal System (NaOH spray additive). This event is not governing, however. See Reference 42.

# CORRECTIVE ACTIONS
## PORV TESTING

IMMEDIATE ACTIONS TAKEN

- UPGRADED POST MAINTENANCE TEST PROCEDURE M-0037, "PORV MAINTENANCE" TO VERIFY VALVE OPERATION IN ADDITION TO SEAT LEAKAGE TEST

- TESTED BOTH PORVs ABILITY TO LIFT PRIOR TO INSTALLATION

- UPGRADED INSERVICE TEST PROCEDURE AP 0010125A, DATA SHEET #24, "VALVE TESTING"

  - IN ADDITION TO ACOUSTIC MONITORS' RESPONSE, VERIFY RCS PRESSURE DECREASE >5 PSIG TO CONFIRM MAIN DISC OPENING

  - OTHER CONFIRMING SYSTEM PARAMETERS ARE RECORDED AND EVALUATED:

    - QUENCH TANK TEMPERATURE
    - QUENCH TANK LEVEL
    - QUENCH TANK PRESSURE
    - PORV TAILPIPE TEMPERATURE

# CORRECTIVE ACTIONS
## PORV TESTING
(continued)

| ACTIONS TO PREVENT RECURRENCE | EXPECTED COMPLETION |
|---|---|
| • REVISE THE UNIT 2 PORV POST MAINTENANCE TEST PROCEDURE TO VERIFY THAT THE MAIN DISC ACTUATES | 11/15/95 |
| • REVIEW POST MAINTENANCE TESTING OF OTHER SAFETY RELATED EQUIPMENT TO ENSURE THE TESTING ADEQUATELY DEMONSTRATES COMPONENT OPERABILITY: | |
| - CONSOLIDATE TEST GROUPS UNDER A SINGLE MANAGER REPORTING TO THE OPERATIONS MANAGER | COMPLETE |
| - REVIEW UNIT 2 OUTAGE SCOPE POST MAINTENANCE TEST PROCEDURES TO ENSURE CRITICAL COMPONENT FUNCTIONS ARE ADDRESSED | 11/9/95 |
| - REVISE PROCESS FOR POST MAINTENANCE TESTING TO IMPROVE COORDINATION AMONG OUTAGE MANAGEMENT, OPERATIONS AND MAINTENANCE | 1/1/96 |
| - REVIEW UNIT 1 OUTAGE SCOPE POST MAINTENANCE TEST PROCEDURES TO ENSURE CRITICAL COMPONENT FUNCTIONS ARE ADDRESSED | 4/96 |

# EVALUATION OF SAFETY SIGNIFICANCE

- FUNCTIONS OF PORVs

- POWER AND SHUTDOWN OPERATION ASSESSED

- PROBABILISTIC SAFETY ASSESSMENT

- CONCLUSIONS

# EVALUATION OF SAFETY SIGNIFICANCE

(continued)

## FUNCTIONS OF PORVs

- POWER OPERATION
  - PORVs PREVENT LIFTING CODE SAFETIES DURING OPERATING TRANSIENTS AND ARE NOT RELIED UPON FOR ANY SAFETY RELATED OPERATING FUNCTION
  - EMERGENCY OPERATING PROCEDURES (EOPs) USE PORVs AS CONTINGENCY FOR BEYOND DESIGN BASIS EVENTS THAT INVOLVE MULTIPLE SINGLE FAILURES SUCH AS COMPLETE LOSS OF SECONDARY HEAT REMOVAL

- SHUTDOWN OPERATION
  - PORVs ARE REQUIRED FOR LTOP MITIGATION AND ARE INCLUDED IN THE TECHNICAL SPECIFICATIONS:
    - MODE 4: RCS COLD LEG ≤ 304°F
    - MODE 5 & 6: VESSEL HEAD ON AND RCS NOT VENTED

# EVALUATION OF SAFETY SIGNIFICANCE

(continued)

## POWER OPERATION ASSESSMENT

- VARIOUS METHODS FOR DEPRESSURIZATION
  - PRESSURIZER SPRAY SYSTEM
  - ATMOSPHERIC DUMP VALVES
  - STEAM BYPASS CONTROL SYSTEM

- SOME PRESSURIZED WATER REACTORS DO NOT HAVE PORVs
  - TECH SPECS ALLOW BLOCK VALVE CLOSURE

- PORVs ARE NOT REQUIRED FOR UFSAR CHAPTER 15 ACCIDENT MITIGATION
  - CODE SAFETY RELIEF VALVES PROVIDE PROTECTION

# EVALUATION OF SAFETY SIGNIFICANCE

(continued)

## PROBABILISTIC SAFETY ASSESSMENT (PSA)

- PORVs MODELED IN BASE PSA

- EFFECT OF PORVs ON CDF

  - PSA WITH PORVs: $2.1 \times 10^{-5}$/YEAR

  - PSA WITHOUT PORVs: $7.6 \times 10^{-5}$/YEAR
    (i.e., LOSS OF SECONDARY HEAT REMOVAL)

- CDF REMAINS LESS THAN NRC SAFETY GOAL OF $10^{-4}$/YEAR

- ADDITIONAL CREDIT FOR OPERATOR ACTIONS TO RESTORE FEEDWATER TO S/Gs WOULD FURTHER REDUCE CDF

# EVALUATION OF SAFETY SIGNIFICANCE

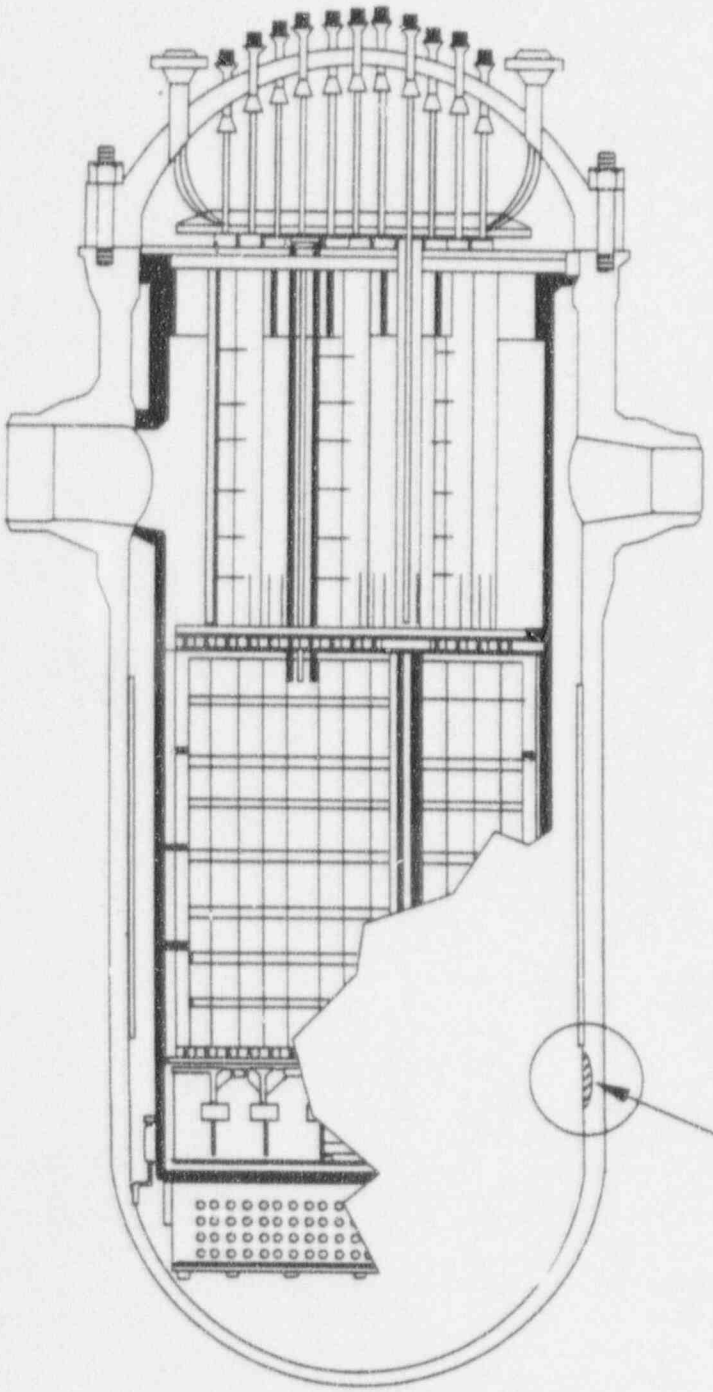
(continued)

## SHUTDOWN OPERATION ASSESSMENT

- REVIEWED UNIT 1 OPERATING HISTORY
  - HAVE NEVER EXPOSED THE REACTOR VESSEL/RCS TO OVER PRESSURE EVENT (LTOP RANGE)
- REVIEWED THE ANALYZED LTOP TRANSIENTS
- CONSIDERED MASS AND ENERGY ADDITION TRANSIENTS
- MASS ADDITION TRANSIENTS ARE MOST LIMITING
  - HPSI AND/OR CHARGING PUMPS

# SHUTDOWN OPERATIONS ASSESSMENT
# EFFECTS ON REACTOR VESSEL

## REACTOR VESSEL



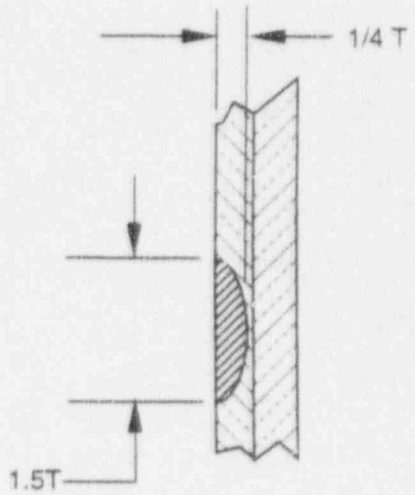
(PRES/950352-F4-R0)

## ASSUMPTIONS

- ISOTHERMAL EVENT
- 1/4 T FLAW (2.2" X 13")
- LOCATION LIMITING WELD (LOWER AXIAL WELD)
- RT ndt CORRECTED FOR IRRADIATION
- BOUNDING CRACK INITATION CURVE -K1c (ASME APP. A)

## ACCEPTANCE CRITERIA

- NO CRACK INITIATION




(DETAIL "A")

ASSUMED FLAW
( SEE DETAIL "A")

# EVALUATION OF SAFETY SIGNIFICANCE

(continued)

## SHUTDOWN OPERATION ASSESSMENT (continued)

| LTOP TRANSIENTS ANALYTICAL RESULTS | | | |
|---|---|---|---|
| RCS TEMPERATURE (°F) | ASSUMED SOURCE OF PRESSURIZATION | MAXIMUM RCS (PSIA) | VESSEL ALLOWABLE (PSIA) |
| ≤ 200 | HPSI PUMP | ≤ 1300 | ≤ 1510 |
| > 200 | 2 HPSI AND 3 CH PUMPS | ≤ 2575[1] | ≤ 2750 |

[1] CODE SAFETY RELIEF PRESSURE (2500 PSIA) PLUS ACCUMULATION (75 PSIA)

- IMPOSED STRESSES ARE LESS THAN REQUIRED FOR CRACK INITIATION

# EVALUATION OF SAFETY SIGNIFICANCE

(continued)

## CONCLUSIONS

- NO EFFECT ON CHAPTER 15 SAFETY ANALYSES

- CDF REMAINS LESS THAN NRC SAFETY GOAL

- UNDER LTOP, IMPOSED STRESSES LESS THAN THOSE REQUIRED FOR CRACK INITIATION

- NO UNDUE RISK TO PUBLIC HEALTH AND SAFETY

# ENFORCEMENT AND INVESTIGATION

# COORDINATION STAFF

# REGION II

## REFERENCE PACKAGE

## EA 95-180

## SEVERITY LEVEL III, $50,000 CP
## FLORIDA POWER AND LIGHT COMPANY
## ST. LUCIE NUCLEAR PLANT

## REFERENCE DOCUMENT CHECKLIST

## REGION II ENFORCEMENT AND INVESTIGATIONS COORDINATION STAFF

## ENCLOSURES

**EA 95-180**
**FLORIDA POWER AND LIGHT COMPANY**
**ST. LUCIE NUCLEAR PLANT**

[1] NRC Inspection Reports and other documentation of the case:

a. NRC Inspection Report 50-335/95-16 AND 50-389/95-16

[2] Licensee reports (Submitted at the Conference):

a. Engineering Evaluation - Evaluation of PORV Unavailability on Plant Operations
b. LER 95-238, PORVs Inoperable Due to Personnel Error, August 22, 1995
c. Topical Quality Assurance Report

[3] Applicable license conditions:
Technical Specification 3.4.13
Technical Specification 4.0.5
Article IWV-3000, Test Requirements

[4] Applicable licensee procedures or extracts

a. General Maintenance Procedure No. 1-M-0037
b. Second Ten-Year Inservice Inspection Interval Inservice Testing Program

[_] Copy of discrepant licensee documentation referred to in citations such as NCR, inspection record, or test results

[5] Enforcement Panel Questionnaire/PEC Briefing Paper

[6] Licensee PEC Presentation Materials

[ ] Referenced ORDERS or Confirmation of Action Letters (CALs)

[ ] Other miscellaneous documents:

**ENFORCEMENT COORDINATOR: LINDA WATSON (404) 331-4192**

## REFERENCE DOCUMENT CHECKLIST

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**ENFORCEMENT COORDINATOR: LINDA WATSON (404) 331-4192**

REFERENCE PACKAGE RII
EA NO: 95-180
RII/EICS NO: 95-E-026

ENCLOSURE NO: 1

REFERENCE PACKAGE RII
EA NO: 95-180
RII/EICS NO: 95-E-026

ENCLOSURE NO: 2a

REVISION 5
01/6/95

FLORIDA POWER and LIGHT COMPANY
NUCLEAR ENERGY SERVICES
700 Universe Boulevard
Juno Beach, Florida 33408

# SECOND TEN-YEAR INSERVICE INSPECTION INTERVAL INSERVICE TESTING PROGRAM

FOR

# PUMPS AND VALVES

ST. LUCIE NUCLEAR POWER PLANT
UNIT NO. 1

NRC DOCKET NUMBER: 50-335

DOCUMENT NUMBER: ENS-PSL-703 REVISION 5



ST. LUCIE PLANT REVIEWS AND APPROVALS:

PREPARED BY: Jim Hallam DATE: 1/5/95
PSL PLANT IST ENGINEER

APPROVED BY: R Rich DATE: 1/6/95
PSL TEST & CODE SUPERVISOR

[NREG 14.82]

REVISION 5
01/6/95

# INSERVICE TESTING (IST) PROGRAM PLAN
# ST. LUCIE UNIT 1

## 1.0 INTRODUCTION

Revision 5 of the St. Lucie Unit 1 ASME Inservice Inspection (IST) Program will be in effect through the end of the second 120-month (10-year) interval unless revised and reissued for reasons other than the routine update required at the start of the third interval per 10 CFR 50.55a(g). The second inspection interval is defined as follows:

| Begins | Ends |
|---|---|
| February 11, 1988 | February 10, 1998 |

This document outlines the IST Program for St. Lucie Plant, Unit 1, based on the requirements of the ASME Boiler and Pressure Vessel Code (the Code), Section XI, 1983 Edition through summer 1983 Addenda. References in this document to "IWP" or "IWV" correspond to Subsections IWP and IWV, respectively, of the ASME Section XI, 1983 Edition, unless otherwise noted.



The IST program incorporates the requirements of ASME/ANSI OM-1987, Part 1, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices" for the testing of safety and relief valves. The use of ASME/ANSI OM-1987, Part 1 requirements as an alternative to ASME Section XI, 1983 Edition, Subsection IWV-3510 requirements was approved by the NRC Safety Evaluation of St. Lucie Unit 1 Inservice Testing Program Relief Requests dated September 27, 1994.

The inservice testing requirements identified in this Plan were prepared to verify the operational readiness of pumps and valves which have a specific function in mitigating the consequences of an accident or in bringing the reactor to a safe shutdown.

In this regard, the general requirements of Paragraphs IWP-1100 and IWV-1100 form the following basic scope document as it applies to ISI Class 1, 2, and 3. Specifically components to be included are:



9/26/95 JA

REFERENCE PACKAGE RII
EA NO: 95-180
RII/EICS NO: 95-E-026

ENCLOSURE NO: 5

# NRC CLOSED PREDECISIONAL ENFORCEMENT CONFERENCE

## ST. LUCIE NUCLEAR PLANT

SEPTEMBER 25, 1995

NRC CLOSED PREDECISIONAL ENFORCEMENT CONFERENCE

ST. LUCIE NUCLEAR PLANT

SEPTEMBER 25, 1995

| TAB | TITLE |
|---|---|
| 1 | Predecisional Enforcement Conference Agenda |
| 2 | Expected Attendees, Meeting Announcement |
| 3 | Opening Remarks and Introductions |
| 4 | NRC Enforcement Policy |
| 5 | Summary of the Issues |
| 6 | Statement of Concerns / Apparent Violations |
| 7 | Inspection Report No. 50-335/398/95-16 |
| 8 | Enforcement Pre-Panel Questionnaire |
| 9 | 50.72 Report, LER 95-242 |
| 10 | Closing Remarks: |

PREDECISIONAL ENFORCEMENT CONFERENCE AGENDA

ST. LUCIE
SEPTEMBER 25, 1995, AT 10:00 A.M.
NRC REGION II OFFICE, ATLANTA, GEORGIA

I. OPENING REMARKS AND INTRODUCTIONS
S. Ebneter, Regional Administrator

II. NRC ENFORCEMENT POLICY
B. Uryc, Director
Enforcement and Investigation Coordination Staff

III. SUMMARY OF THE ISSUES
S. Ebneter, Regional Administrator

IV. STATEMENT OF CONCERNS / APPARENT VIOLATIONS
E. Merschoff, Director
Division of Reactor Projects

V. LICENSEE PRESENTATION
W. Goldberg, President
St. Lucie Nuclear Plant

VI. BREAK / NRC CAUCUS

VII. NRC FOLLOWUP QUESTIONS

VIII. CLOSING REMARKS
S. Ebneter, Regional Administrator

## EXPECTED ATTENDEES

Licensee

J. Goldberg, President, Nuclear Division
D. Sager, Vice President, St. Lucie Site
W. Bohlke, Vice President, Engineering
L. Bladow, Nuclear Assurance Manager
D. Denver, Site Engineering Manager
L. Rogers, Systems and Component Engineering Manager
J. Marchese, Maintenance Manager
J. West, Operations Manger

NRC

Stew Ebneter, Regional Administrator, Region II (RII)
Ellis Merschoff, Director, Division of Reactor Projects (DRP), RII
Al Gibson, Director, Division of Reactor Safety (DRS), RII
Bruno Uryc, Director, Enforcement and Investigation Coordination Staff (EICS), RII
Charles Casto, Chief, Engineering Branch, DRS, RII
Kerry Landis, Chief, Reactor Projects Branch 2, DRP, RII
Linda Watson, Senior Enforcement Specialist, EICS, RII
Carolyn Evans, Regional Counsel, RII
Richard Prevatte, Senior Resident Inspector, St. Lucie, DRP, RII
Robert Schin, Project Engineer, Reactor Projects Section 2B, DRP, RII
Edwin Lea, Project Engineer, Reactor Projects Section 2B, DRP, RII
George Hopper, Operator Licensing Examiner, DRS, RII

## OPENING REMARKS AND INTRODUCTIONS
(S. Ebneter)

Good morning. I am Stew Ebneter, Regional Administrator for the Nuclear Regulatory Commission's Region II Office. This morning we will conduct a predecisional enforcement conference between the NRC and **St. Lucie** which is **CLOSED** to public observation.

The agenda for the conference is shown in the viewgraph. Following my brief opening remarks, Mr. Bruno Uryc, the Director of the Region II Enforcement Staff, will discuss the Agency's Enforcement Policy. I will then provide introductory remarks concerning my perspective on the events to be addressed today. Mr. Ellis Merschoff, Director of the Division of Reactor Projects, will then discuss the apparent violations. You will then be given an opportunity to respond to the apparent violations. In this regard, I wish to reiterate to you that the decision to hold this conference does not mean that the NRC has determined that violations have occurred or that enforcement action will be taken. This conference is an important step in arriving at that decision.

Following your presentation, I plan to take about a 10-minute break so that the NRC can briefly review what it has heard and determine if we have follow-up questions. Lastly, I will provide concluding remarks.

At this point, I would like to have the NRC staff introduce themselves and then ask you to introduce your participants.

[INTRODUCTIONS]

Thank you.

Mr. Uryc will now discuss the Agency's Enforcement Policy.

**NRC ENFORCEMENT POLICY**
(B. Uryc)

NRC Enforcement Policy and Procedure

After an apparent violation is identified, it is assessed in accordance with the Commission's Enforcement Policy, which was recently revised and became effective on June 30, 1995. The Enforcement Policy has been published as NUREG-1600.

The assessment of an apparent violation involves categorizing the apparent violation into one of four severity levels based on safety and regulatory significance. For cases where there is a potential for escalated enforcement action, that is, where the severity level of the apparent violation is categorized at Severity Level I, II, or III, a predecisional enforcement conference is held.

There are three primary enforcement sanctions available to the NRC and they are Notices of Violation, civil penalties, and orders. Notices of Violation and civil penalties are issued based on identified violations. Orders may be issued for violations, or, in the absence of a violation, because of a significant public health or safety issue.

This predecisional enforcement conference is essentially the last step of the inspection or investigation process before the staff makes its final enforcement decision.

The purpose of this conference is not to negotiate a sanction. Our purpose here today is to obtain information that will assist us in determining the appropriate enforcement action, such as: (1) a common understanding of the facts, root causes and missed opportunities associated with the violations, (2) a common understanding of corrective action taken or planned, and (3) a common understanding of the significance of issues and the need for lasting comprehensive action.

The apparent violations discussed at this conference are subject to further review and they may be subject to change prior to any resulting enforcement action. It is important to note that the decision to conduct this conference does not mean that NRC has determined that a violation has occurred or that enforcement action will be taken.

I should also note at this time that statement of views or the expression of opinion made by the NRC staff at this conference, *or the lack thereof*, are not intended to represent final determinations or beliefs.

Following the conference, the Regional Administrator in conjunction with the NRC Office of Enforcement and other NRC Headquarters offices will reach an enforcement decision. This process should take about four weeks to accomplish.

Predecisional enforcement conferences are normally closed to the public as is this conference. However, the Commission implemented a trial program in July 1992 to allow certain enforcement conferences to be open for public observation. [July 10, 1992 - *Federal Register*] This trial program was recently extended for additional evaluation.

Finally, if the final enforcement action involves a proposed civil penalty or an order, the NRC will issue a press release 24 hours after the enforcement action is issued.

## SUMMARY OF THE ISSUES
(S. Ebneter)

Issues: St. Lucie Power Operated Relief Valves Inoperable

Power Operated Relief Valves V-1404 and V-1402 were reassembled incorrectly and did not receive adequate post-maintenance testing. The PORVs were reinstalled in the Reactor Coolant System on November 5, 1994, without adequate surveillance testing sufficient to provide reasonable assurance that the valves would perform their intended function while in service.

quality interface
engl/ops interface
use of diverse indications

Defect:

As a result of the inadequate reassembly, post-maintenance testing, and surveillance testing the PORVs were inoperable from the time they were installed in the RCS during the 1994 refueling outage until they were removed and reworked in August of 1995. Plant data indicated that the valves could not have performed their intended safety function. Plant data also indicated several

instances in which an operable PORV was required by TS, but was not available.

Consequences:

Inoperable equipment was installed in the Reactor Coolant System which was unable to perform intended safety function.

## STATEMENT OF CONCERNS / APPARENT VIOLATIONS
(E. Merschoff)

This is a Predecisional Enforcement Conference to discuss three apparent violations associated with PORV maintenance and operability. The first and second apparent violations involve the adequacy of the procedure used to perform post-maintenance testing and the procedure used to perform surveillance testing. The third apparent violation addresses the operability of the PORV as required by TS.

We are concerned with those activities which resulted in the PORVs being rendered inoperable. Encompassed in our concerns are the facts that post-maintenance and surveillance testing should be of sufficient scope, and acceptance criteria of sufficient technical rigor, to ensure component operability. We are also concerned with the consequences of operating outside TS limits, due to inoperable PORVs.

From clear opportunities to discover
11/25
2/27
7/11 trip

Our findings are documented in NRC Inspection Report 50-335/389/95-16, which were transmitted to you on September 8, 1995. At this conference, we are affording you the opportunity to provide information relative to:

--- Any errors the inspection reports

--- The severity of the violations

--- Any escalation or mitigation considerations

--- Any other application of the Enforcement Policy relevant to this issue.

# ISSUES TO BE DISCUSSED

1. 10 CFR 50, Appendix B, Criterion XI required, in part, that a test program be established to ensure that all testing required to demonstrate that components will perform satisfactorily in service be performed and that the program include proof tests prior to installation. FPL Topical Quality Assurance Report TQR 11.0, revision 4, "Test Control," stated, in part, that a test program shall be established to assure that testing required to demonstrate that structures, systems and components will perform satisfactorily in service and that the program shall include proof tests prior to installation.

   In November, 1994, valve maintenance was performed under a work package, which directed the rebuilding of Power Operated Relief Valves V-1404 and V-1402 per licensee procedure 1-M0037, Revision 6, "Power Operated Valve Relief Valve Maintenance. The post-maintenance testing was limited to a bubble test for seat leakage prior to reinstallation. The procedure contained a note explaining that lift set point testing was not required, as the valve was lifted based upon solenoid valve input. The procedure did not require a verification that the valve would change state under pressure prior to installation.

**NOTE:** The apparent violations discussed in this predecisional enforcement conference are subject to further review and are subject to change prior to any resulting enforcement decision.

# ISSUES TO BE DISCUSSED

2. 10 CFR 50, Appendix B, Criterion XI required, in part, that a test program be established to ensure that adequate test instrumentation is available and used. FPL Topical Quality Assurance Report TQR 11.0, revision 4, "Test Control," stated, in part, that a test program shall be established to assure that testing required to demonstrate that structures, systems and components will perform satisfactorily in service is performed and that the program shall include operational tests. TQR 11.0 further states that test procedures shall incorporate requirements and acceptance limits in the applicable design and procurement documentation.

   On November 25, 1994, and on February 27, 1995, operational surveillance testing, performed under Administrative Procedure 1-0010125A, revision 39, Data Sheet 24, did not employ adequate test instrumentation to detect the inoperability of both valves and did not employ test acceptance limits derived from the valves' design documentation. Specifically, the use of acoustic data, as opposed to system pressure reduction derived from valve capacity, to indicate valve position was insufficient to discern the difference between bypass flow through the PORV pilot valves and actual changes in main valve position.

**NOTE:** The apparent violations discussed in this predecisional enforcement conference are subject to further review and are subject to change prior to any resulting enforcement decision.

# ISSUES TO BE DISCUSSED

3. Technical Specification 3.4.13 requires, in part, that two Power Operated Relief Valves be operable in "Mode 4 when the temperature of any RCS cold leg is less than or equal to 304°F, Mode 5 and Mode 6 when the head is on the reactor vessel; and the RCS is not vented through a greater than 1.75 square inch vent." TS 3.4.13 AS (c) required that, "with two inoperable PORVs, at least one PORV be returned to an operable status or that the RCS be completely depressurized and vented through a minimum 1.75 square inch opening within 24 hours."

   From November 22 through 27, 1994, and from February 27 through March 6, 1995, St. Lucie Unit 1 was in conditions requiring operable Power Operated Relief Valves but no operable relief valves were in service. The inoperability of the Power Operated Relief Valves resulted from a combination of personnel error during maintenance and inadequate post-maintenance and surveillance testing.

**NOTE:** The apparent violations discussed in this predecisional enforcement conference are subject to further review and are subject to change prior to any resulting enforcement decision.

## CLOSING REMARKS
(S. Ebneter)

In closing this predecisional enforcement conference, I remind the Licensee of two things:

First, the apparent violations discussed at this predecisional enforcement conference are subject to further review and may be subject to change prior to any resulting enforcement action.

Second, the statements of views or expressions of opinion made by NRC employees at this predecisional enforcement conference, or the lack thereof, are not intended to represent final agency determinations or beliefs.

ENFORCEMENT PRE-PANEL - ST. LUCIE

8/23/95 - 2:00 pm

E42

BRIDGE #: (301) 415 - 7605 [redacted]

SENIOR DRP MANAGEMENT REPRESENTATIVE - ELLIS MERSCHOFF

SECTION CHIEF - KERRY LANDIS

✓EICS REPRESENTATIVE - LINDA WATSON

REGIONAL COUNSEL - BRAD FEWELL

SENIOR PROGRAM OFFICE REPRESENTATIVE - JAN NORRIS

DRS MANAGEMENT REPRESENTATIVE - AL GIBSON

DRS REPRESENTATIVE - CHUCK CASTO

DRS REPRESENTATIVE - TOM PEEBLES

ESCALATED ENFORCEMENT
PANEL QUESTIONNAIRE

INFORMATION REQUIRED TO BE AVAILABLE FOR ENFORCEMENT PRE-PANEL

PREPARED BY: Mark S. Miller

NOTE: The Section Chief is responsible for preparation of this questionnaire and its distribution to attendees prior to an Enforcement Panel. (This information will be used by EICS to prepare the enforcement letter and Notice, as well as the transmittal memo to the Office of Enforcement explaining and justifying the Region's proposed escalated enforcement action.)

1. Facility: St. Lucie

   Unit: 1

   Docket Nos: 50-335

   License Nos: DPR-67

   Inspection Dates: July 2 - August 28, 1995

   Lead Inspector: Richard L. Prevatte

2. Check appropriate boxes:

   [X] A Notice of Violation (without "boilerplate") which includes the recommended severity level for the violation is enclosed.

   [] This Notice has been reviewed by the Branch Chief or Division Director and each violation includes the appropriate level of specificity as to how and when the requirement was violated.

   [] Copies of applicable Technical Specifications or license conditions cited in the Notice are enclosed.

3. Identify the reference to the Enforcement Policy Supplement(s) that best fits the violation(s) (e.g., Supplement I.C.2)

   I.B.1* I.C.2.(b)*

   I.C.6*

   * As stated in new enforcement policy

4. What is the apparent root cause of the violation or problem?

   Personnel error in maintenance combined with a failure to perform adequate post-maintenance testing and a failure to provide adequate acceptance criteria for surveillance testing of PORVs.

5. State the message that should be given to the licensee (and industry) through this enforcement action.

Post-maintenance and surveillance testing should be of sufficient scope, and acceptance criteria of sufficient technical rigor, to ensure component operability.

6. Factual information related to the following civil penalty escalation or mitigation factors (see attached matrix and 10 CFR Part 2, Appendix C, Section VI.B.2.):

a. IDENTIFICATION: (Who identified the violation? What were the facts and circumstances related to the discovery of the violation? Was it self-disclosing? Was it identified as a result of a generic notification?)

The condition was identified by control room operators when it was noted that operation of the PORVs during surveillance testing did not result in expected changes in RCS and quench tank parameters. The identification of the misaligned main disk guide was identified by the licensee when the valves were removed from the system for inspection. Failure to perform adequate post-maintenance and surveillance testing was identified by the licensee in course of a rigorous root cause effort.

b. CORRECTIVE ACTION: Although we expect to learn more information regarding corrective action at the enforcement conference, describe preliminary information obtained during the inspection and exit interview.

The discrepant conditions were corrected by the licensee. See attachment.

What were the immediate corrective actions taken upon discovery of the violation, the development and implementation of long-term corrective action and the timeliness of corrective actions?

The PORVs were declared inoperable, the unit was placed in a condition not requiring their operability and then was cooled down and depressurized. Immediate corrective actions were appropriate and the subject failures were investigated aggressively. Short term corrective actions included properly assembling and testing the subject PORVs, adding a QC hold point to the PORV maintenance procedure to insure proper disc guide installation and to require a bench lift test under air pressure. Additionally, the licensee changed the acceptance criteria in the PORV surveillance testing procedure requiring documentation of RCS pressure change, PORV tailpipe temperature change, and quench tank pressure, level and temperature changes.

What was the degree of licensee initiative to address the violation and the adequacy of root cause analysis?

Root cause determination was well-coordinated, timely, and comprehensive.

c. LICENSEE PERFORMANCE: This factor takes into account the last two years or the period within the last two inspections, whichever is longer.

List past violations that may be related to the current violation (include specific requirement cited and the date issued):

No violations involving the adequacy of post-maintenance testing have been issued in the last two years.

Identify the applicable SALP category, the rating for this category and the overall rating for the last two SALP periods, as well as any trend indicated:

The subject functional area is Maintenance and Surveillance, which was most recently rated SALP 1 (January 94). The previous SALP period, the area was rated a SALP 1.

d. PRIOR OPPORTUNITY TO IDENTIFY: Were there opportunities for the licensee to discover the violation sooner such as through normal surveillances, audits, QA activities, specific NRC or industry notification, or reports by employees?

Proper post-maintenance testing would have been effective in identifying the inoperable status of the PORVs. Additionally, adequate surveillance testing would have detected the inoperability. The prior opportunities to identifiy the subject conditions is at the heart of this enforcement action.

e. MULTIPLE OCCURRENCES: Were there multiple examples of the violation identified during this inspection? If there were, identify t' number of examples and briefly describe each one.

Multiple occurrences have not been identified, except to the extent that the noted test failures applied to two PORVs.

f. DURATION: How long did the violation exist?

Since November, 1994.

ADDITIONAL COMMENTS/NOTES: See attached description of the subject events. Note that, in addition to violations relating to post-maintenance and surveillance testing adequacy, a violation of TS 3.4.13, regarding PORV operability for LTOP, existed. The information in this document is current as of August 22, 1995.

ESCALATION AND MITIGATION FACTORS (57 FR 5791, February 18, 1992)

| IDENTIFICATION | CORRECTIVE ACTION | LICENSEE PERFORMANCE | PRIOR OPPORTUNITY TO IDENTIFY | MULTIPLE OCCURRENCES | DURATION |
|---|---|---|---|---|---|
| +/- 50% | +/- 50% | +/- 100% | + 100% | + 100% | + 100% |
| Licensee identified (M) [To be applied even if licensee could have identified the violation sooner] | Timeliness of corrective action (M) [Did NRC have to intervene to accomplish satisfactory short-term or remedial action (E)] | Current violation is an isolated failure that is inconsistent with licensee's good performance (M) | Licensee should have identified violation sooner as a result of prior opportunities such as audits (E) | Multiple examples of violation identified during inspection (only for SL I, II or III violations) (E) | Used for significant regula y message to licensee. (E) |
| NRC identified (E) | Promptly developed schedule for long term corrective action (M) | Violation is reflective of licensee's poor or declining performance (E) | Opportunities available to discover violation such as through prior notification (E) | | |
| Self-disclosing (M-25% if there was initiative to identify root cause) | Degree of licensee initiative (M) [To develop corrective actions and root cause] | Prior performance and effectiveness of previous corrective action for similar violations | Ease of earlier discovery (E) | | |
| Licensee identified as a result of generic notification (M) | Adequacy of the root cause analysis for the violation (M) | SALP - Consider: SALP 1 - (M) SALP 2 - (O) SALP 3 - (E) | Period of time between violation and notification received by licensee (E) | | |
| | Comprehensive corrective action to prevent occurrence of similar violation (M) | Prior enforcement history including escalated and non-escalated enforcement | Similarity between the violation and notification (E) | | |
| | Immediate corrective action not taken to restore safety and compliance (E) | | Level of management review the notification received (E) | | |

OTHER CONSIDERATIONS

1. Legal aspects and potential litigation risks
2. Negligence, careless disregard, willfulness and management involvement
3. Economic, personal or corporate gain
4. Any other regulatory framework factors that need to be considered: pending action with regard to licensing, commission meeting, or press conference.
5. What is the intended message for the licensee and the industry?

---------- NOTES----------

**SAFETY SIGNIFICANCE:** In determining the safety significance of a violation in conjunction with the enforcement process, the evaluation should consider the technical safety significance of the violation as well as the regulatory significance. Consideration should be given to the matter as a whole in light of the circumstances surrounding the violation. There may be cases in which the technical safety significance of the matter is low while the process control failure(s) may be significant, and, therefore, the severity level determination should be based more on the process control failure(s) than on the technical safety issue. The following factors should also be considered: 1) Did the violation actually or potentially impact public health and safety? 2) What was the root cause of the violation? 3) Is the violation an isolated incident or is it indicative of a programmatic breakdown? 4) Was management aware of or involved in the violation? 5) Did the violation involve willfulness?

Proposed Violation A

10 CFR 50, Appendix B, Criterion XI states, in part, that a test program shall be established to ensure that all testing required to demonstrate that components will perform satisfactorily in service and that the program shall include proof tests prior to installation and operational testing. The criterion further states that test procedures shall include provisions to assuring that adequate test instrumentation is available and used. FPL Topical Quality Assurance Report TQR 11.0, revision 4, "Test Control," states, in part, that a test program shall be established to assure that testing required to demonstrate that structures, systems and components will perform satisfactorily in service and that the program shall include proof tests prior to installation, operational tests, and retest after repair. TQR 11.0 further states that test procedures shall incorporate requirements and acceptance limits in the applicable design and procurement documentation.

Contrary to the above, on November 5, 1994, and on November 6, 1994, Power Operated Relief Valves V-1404 and V-1402, respectively, were installed in the Unit 1 Reactor Coolant System, placed into operation on November 22, and relied upon to be operable for approximately nine months without adequate post-maintenance and surviellance testing sufficient to provide reasonable assurance that the valves would perform satisfactorily in service. As a result of these failures, both Unit 1 PORVs were inoperable from the time of their reinstallation after maintenance until August 11, 1995, when the reactor coolant system was depressurized and vented. Examples are:

1. No post-maintenance bench test was performed to ensure that the valves' main discs would change state in a pressurized environment.

2. On November 25, 1994, and on February 27, 1995, operational surveillance testing, performed under Administrative Procedure 1-0010125A, revision 39, Data Sheet 24, did not employ adequate test instrumentation to detect the inoperability of both valves and did not employ test acceptance limits derived from the the valves' design documentation. Specifically, the use of acoustic data, as opposed to system pressure reduction derived from valve capacity, to indicate valve position was insufficient to discern the difference between bypass flow through the PORV pilot valves and actual changes in main valve position.

This is a Severity Level II violation (supplement I).

Proposed Violation B

Technical Specification 3.4.13 requires, in part, that two Power Operated Relief Valves be operable in "Mode 4 when the temperature of any RCS cold leg is less than or equal to 304°F, Mode 5 and Mode 6 when the head is on the reactor vessel; and the RCS is not vented through a greater than 1.75 square inch vent."

Contrary to the above, from November 22 through 27, 1994, and from February 27 through March 6, 1995, St. Lucie Unit 1 was in conditions requiring operable Power

Operated Relief Valves but no operable releif valves in service. The inoperability of the Power Operated Relief Valves resulted from a combination of personnel error during maintenance and inadequate post-maintenance and surveillance testing.

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

## St. Lucie Unit 1 PORV Inoperability

Operational Events

On August 9, the licensee performed ASME Section XI stroke testing on V-1402 and V-1404 (Unit 1 PORVs) per AP 1-0010125A, revision 39, "Surveillance Data Sheets," Data Sheet 24. The test was performed with RCS pressure controlled at 257 to 268 psig. The methodology of the test involved placing the PORV control switches in "override," (which ensured that the valves would not open) removing High Pressurizer Pressure bistables from the RPS cabinets (which would send an "open" signal to the PORVs which would be blocked by the status of the control switches), and then, for each PORV, placing the control switch in "normal," which would send the open signal to the PORV. The stroke time for each PORV was to be measured from the time the control switch was taken to "normal" to the time that acoustic monitors indicated that the subject valve had opened. Once a valve stroke time had been obtained, the subject valve's control switch was to be returned to override to close the valve.

The results of the subject testing indicated that the valves did not stroke open. No acoustic signal was received in the control room. The licensee then returned the valves to service while questions of acoustic monitoring calibration and threshold levels were considered. The test was reperformed approximately one hour later with temporary acoustic monitors and the resulting acoustic signals indicated that both valves stroked in under one second. LTOP was placed back in service, but Operations personnel began to question the validity of the test results, as no changes were noted in either RCS or Quench Tank parameters. While evaluations were being conducted, the unit was taken to Mode 4. At 7:03 p.m. on August 9, the valves were retested and found to be inoperable based, in part, on observations of RCS and Quench Tank parameters. Each was declared out of service and the licensee entered TS 3.4.13 Action (c), which required depressurization of the RCS and venting through a 1.75 square inch or greater opening within 24 hours.

At 9:37 p.m., operators were directed by management to perform a cooldown of the RCS. When placing the SDC system in service, the SDC discharge relief valve lifted and would not reseat without securing the SDC pumps to reduce SDC system pressure at the relief valve. This issue will be discussed in a separate enforcement action; however, the inoperability of the SDC system (due to the relief valve issue) precluded the licensee from cooling down and depressurizing Unit 1. Consequently, the licensee entered TS 3.0.3 at 10:45 a.m. on August 10 and began a heatup to greater than 304°F, a plant condition for which TS 3.4.13 did not apply. 305°F was achieved at 11:53 a.m.

The SDC system was returned to service on August 11. A cooldown was commenced at 6:25 a.m. the same day. The licensee made plans to re-enter TS 3.4.13 AS (c) during the cooldown, and to create the required vent path by removing the bonnet of PCV-1100F, one of two pressurizer spray valves, which would create the required vent path to RCS cold leg 1B1. The subject AS was entered at 7:15 a.m. on August 11 and exited at 8:40 p.m. the same day, when the system was vented.

PORV Operation

The PORV design in question is a Dresser Industries Model 31533VX-30. The valve is a 2.5" inlet by 4" outlet pilot operated valve with a relief capacity of 153,000 lbm/hr. The internals of the valves are displayed in Figure 1.

The valve, as installed in Unit 1, is actuated by a solenoid valve, which acts on the pilot valve operating lever to open the pilot valve. The open pilot valve creates a vent path from below the main disc of the main valve, through holes machined in the main disc guide, to the quench tank. The reduction in pressure below the main valve main disc allows the disc to move open (down, in Figure 1) under the force of system pressure acting on the main disc.

When pressure has been reduced below the applicable reseat pressure (depending upon PORV mode - normal, LTOP low range, or LTOP high range), the solenoid valve is deenergized, which closes the pilot valve and isolates the vent path from below the main valve disc to the quench tank. Once the vent path is isolated, pressure builds up below the main valve disc as system pressure is admitted to the space through an orifice in the main valve retainer plug. When pressure has been built up below the main valve disc in this manner, the disc is moved into a closed position under pressure aided by spring force.

Diagnostic Maintenance

The subject PORVs were removed on August 12 and placed on a test bench for lift tests to be conducted under air pressure. Both valves were tested at a number of pressures within the LTOP range and were found to be inoperable. Disassembly and inspection revealed that the main disc guide was installed upside down, with the holes (required to vent the space below the main disc) located at the upper extreme of the main disc cavity such that proper venting below the main valve disc could not take place.

As a function of diagnosing the root cause, the licensee reversed the main disc guide orientations (to the proper orientation) and retested the valves under air pressure. Both valves tested satisfactorily. The licensee also sent a spare valve to Wylie Laboroatories for testing under water and steam pressure, as these conditions could not be established at the site. The spare PORV was tested under water and steam with the main disc guide misoriented (the as-found condition of the Unit 1 PORVs) at pressures ranging from LTOP pressures to NOP ranges. The PORV failed to open under any condition with the main disc guide misoriented. Additionally, it was found that:

- Under water pressure at 335 and 450 psig, 10-15 psig was developed at the discharge of the pilot valve, indicating that some leakage around the main disc guide was possible, but not enough to provide venting sufficient to open the PORV.

- Under steam pressure from 50 psig to 450 psig in 50 psig increments, 20-60 psig was developed at the pilot valve discharge.

- Under steam pressure at 2400 psig, 1500-1800 psig was developed at the pilot valve discharge.

The pressures and media flow detected at the pilot valve discharge indicated that acoustic data may be received during PORV testing without being indicative of a PORV changing state.

Maintenance

The subject PORVs were last reworked in November, 1994, as part of the Unit 1 refueling outage. The rework was conducted by employees of Furmanite, which were used by the licensee for outage-related valve work. The PORVs were each worked by the same two workers. The work package which directed the rebuild invoked the licensee's procedure 1-M-0037, revision 6, "Power Operated Valve Relief Valve Maintenance." The licensee determined that step 9.8 "reassembly of Main Valve," step 7, which directs the installation of the main disc guide, did not include a QC hold point to verify proper installation. It was noted that this is the only component which can be installed improperly and result in undetected inoperability. The procedure was revised to include a QC Hold Point prior to the valves' reassembly.

The inspector questioned the licensee as to post-maintenance testing requirements as applied to the PORVs. The licensee stated that post-maintenance testing was limited to a bubble test for seat leakage prior to reinstallation. The inspector noted that 1-M-0037 only required the noted bubble test as post-maintenance testing and, in fact, contained a note explaining that lift set point testing was not required, as the valve was lifted based upon solenoid valve input. The procedure did not require a proof that the valve would change state under presseure prior to installation, but did include a check for main disc mechanical freedom.

In discussing post-maintenance testing with the licensee, it was stated that, while no documented lift test existed, air lifts were typically performed as a function of preparing for seat leakage tests. It was explained that, upon initial reassembly, the PORVs rarely, if ever, satisfied seat leakage criteria due to relative misalignment between the main valve disc and its seat. As a result, the licensee stated that lifts under air pressure were performed as a matter of course to allow the main disc to orient itself properly against its seat. The inspector noted that the governing procedure included a note to this affect, but no evidence existed to indicate that lifts had occurred on the test bench.

The licensee stated that, in discussions with the Furmanite Supervisor who oversaw the rebuilding of the PORVs during the 1994 outage, the Supervisor stated that he recalled at least 6 lifts under air pressure per PORV in attempts to obtain satisfactory seat leakage tests. No documentation existed to validate the claim. The licensee stated that, when testing of the spare PORV at Wylie was complete, and the PORV was returned to the site, Furmanite was going to be allowed to rebuild the valve and demonstrate that lifts could be achieved with the main disc guide installed backwards. The inspector discussed the plausibility of such lifts with the valve vendor representative on site, who stated that, in principle, such lifts were possible it sufficient gaps existed between the main disc guide and the gasket below the guide. The results of the test are pending.

The inspector concluded that post-maintenance testing, described in 1-M-0037, was inadequate to verify that maintenance had been satisfactorily performed on the PORVs. As described below, surveillance testing was performed on the PORVs during unit heatup and repressurization

following the Unit 1 outage. However, the inspector concluded that insufficient testing had been performed on the PORVs, prior to installation, to obtain a reasonable assurance that the PORVs would perform satisfactorily in the LTOP conditions which would exist prior to the subject surveillance test.

The inspector discussed the issue of post-maintenance testing with Operations personnel. It was confirmed that Operations had accepted the subject PORVs from maintenance with the assumption that they had been properly tested and, as such, considered them operable upon installation. The inspector found this assumption to be counter to the understanding of Maintenance personnel, as in-situ surveillance testing was considered to be the post-maintenance test of the valve overhaul. The inspector reviewed PWOs 63/8104 and 63/8105, which directed I&C to perform PORV solenoid valve inspection and testing on V-1402 and V-1404, respectively. Included in the task description of each PWO was a requirement to "Notify Ops Dept to perform valve stroke time verification test per QI 11-4/AP 1-0010125A DS-10." The referenced procedure and data sheet were the same as that described below for PORV stroke time testing. The inspector concluded that the Maintenance and Operations Departments were under completely different impressions of the status of the valves.

Surveillance Testing

The inspector questioned the licensee as to whether any in-situ testing had been performed on the PORVs since their installation during the 1994 outage. The licensee stated that two tests had been performed; one on November 25, 1994, with RCS pressure at 245 psia, and one on February 27, 1995, with RCS pressure at 1750 psia. Both tests were documented as satisfactory. The satisfactory results were, by procedure, based upon acoustic data, as opposed to system parameter changes (e.g. RCS pressure, quench tank conditions). As stated above, results from testing at Wylie indicated that sufficient flow could be developed through bypass around a misinstalled main disc guide (and then out an open pilot valve) to provide acoustic data without actual main valve movement.

The inspector concluded that the acceptance criteria provided for verifying PORV operability in OP 1-0010125A was insufficient to demonstrate valve operability in that tests performed on November 25, 1994, and February 27, 1995, did not detect the inoperability of the subject PORVs.

PORV Operability

The inspector reviewed the licensee's activities with regard to root cause determination for the subject PORV conditions. In particular, the inspector noted the following:

- Bench testing of both PORVs, once removed from the system and prior to individual valve disassembly, indicated that the valves would not lift under air pressure at any process pressure from the LTOP range to the NOP range.

- Disassembly of each PORV resulted in the discovery of incorrectly installed main valve disc guides.

- Upon correction of main valve disc guide orientation alone (i.e. no other piece part changes or replacement) for each PORV, bench testing under air pressure resulted in satisfactory lifts.

- Wylie Laboratory testing of a spare PORV, under water and steam, under pressure conditions ranging from below LTOP setpoints to above NOP, indicated that no lift was possible with the main valve disc guide installed backwards.

As a result, the inspector concluded that the PORVs were inoperable from the time they were installed in the RCS during the 1994 refueling outage until they were removed and reworked in August, 1995.

Impact on Unit 2

Unit 2 is not susceptible to the same failure, as Unit 2 employs Garrett/Crosby PORVs, which are of distinctly different design. Additionally, the Unit 2 PORVs provide direct main valve position indication, provided by a indexing rod attached to the main valve disc which activates a reed switch. The inspector reviewed AP 2-0010125A, revision 43, "Surveillance Data Sheets," Data Sheet 24, which directed surveillance testing for Unit 2 PORVs. The inspector found that the procedure directed that stroke time be based upon indicated valve position change, as opposed to acoustic data. [FIND OUT IF UNIT 2 PROCEDURE HAS BEEN REVVED TOO]

Impact on Unit 1

The St. Lucie Unit 1 design employs two PORVs, which provide overpressure protection both during normal operation and for LTOP concerns. Additionally, the PORVs are employed in EOPs for once through cooling in the event of a loss of other core heat removal options. During power operations, PORVs are designed to open only in the event of a high pressure reactor trip, and are sized to allow the unit to suffer a loss of load trip from full power without lifting a pressurizer code safety valve. Accident analyses do not credit PORV operation. During low-mode conditions, the PORVs operate on one of two selectable LTOP setpoints, depending upon cold leg temperature and whether a heatup or cooldown is in progress.

Following the Unit 1 refueling, the unit was filled solid on November 22. The RCS was pressurized and in a condition requiring LTOP from November 22 through November 27. The unit was subsequently at NOP until a Short Notice Outage (SNO) in February, 1995. During the SNO, the unit was in conditions requiring LTOP protection from February 27 through March 6. Notably, on March 4, 1994, Unit 1 experienced a loss of shutdown cooling event with the unit in a solid water condition. The condition was corrected by operators, but not before RCS pressure had exceeded the LTOP anticipatory alarm setpoint. No LTOP lift of PORVs was demanded or experienced (peak pressure was 343 psia, LTOP setpoint at the time was 350 psia).

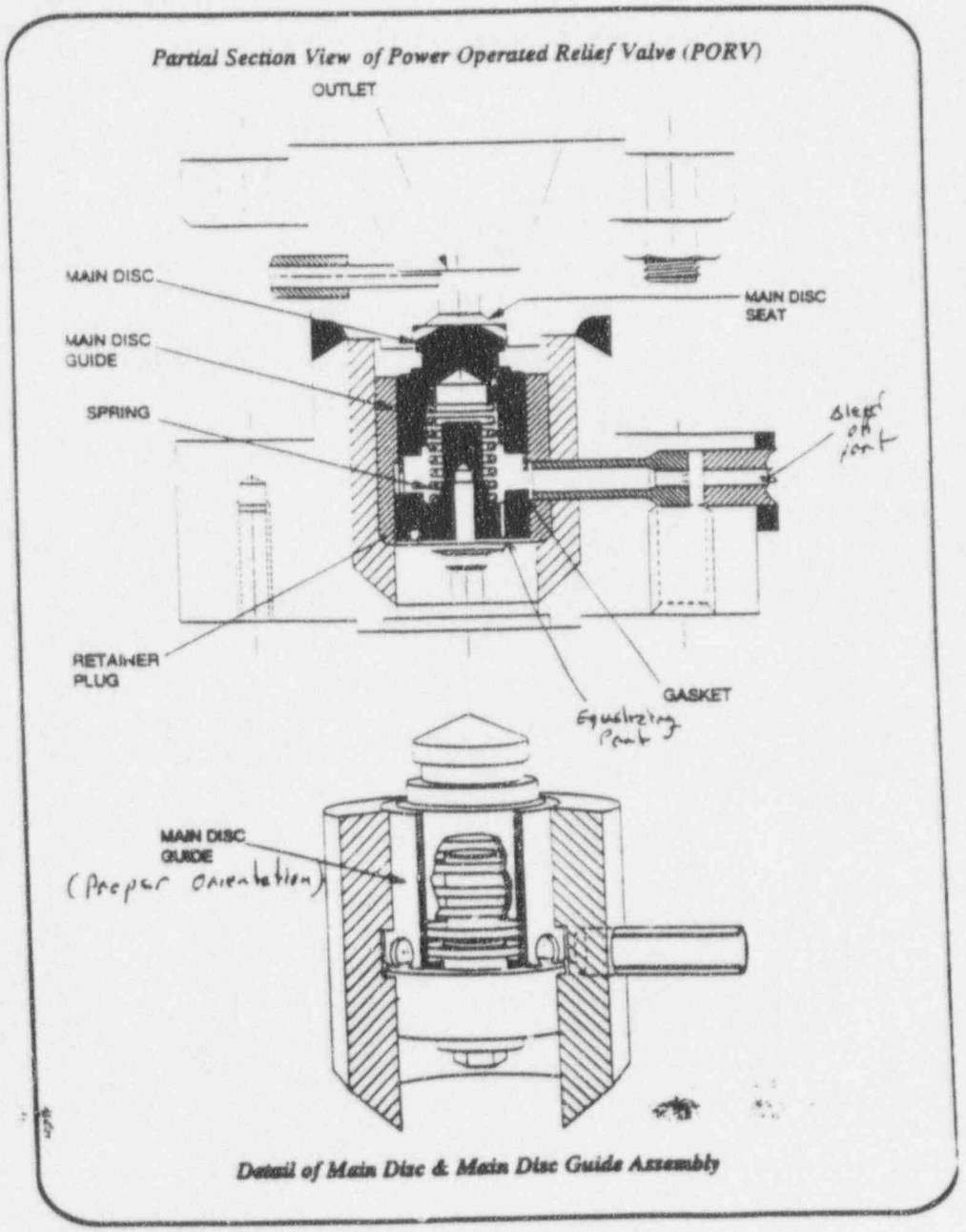
On July 11, 1994, Unit 1 suffered a high pressure trip (see IR 95-14) which, according to the licensee at the time of the trip, included a lifting of both PORVs. The conclusion was supported at the time by the inherent design of the system, the fact that acoustic data indicated that the

PORVs lifted, and noted increases in Quench Tank temperature. The licensee is now doubtful that the PORVs lifted during the trip, based upon a review of data (which suggested that pressure drifted above the PORV setpoint, as opposed to plateauing) and of analyses which showed that the post-trip loss of heat source acts, in conjunction with steam reliefs to limit pressure increases.

As regards once through cooling functions of the PORVs, the St. Lucie IPE includes PORV use in early post-accident heat removal for reactor trips, loss of pressure control events, loss of offsite power events, main steam line break accidents, and steam generator tube ruptures. The licensee performed a PRA analysis which quantified the change in CDF for common-mode failures in PORVs. The licensee determined that the Unit 1 CDF had increased by an approximate factor of 3 for the period of inoperability of the subject PORVs.

With regard to LTOP concerns, the licensee analyzed the impact of a loss of LTOP PORV function for the energy and mass addition events in the original LTOP design basis. The licensee determined that, based upon current levels of Unit 1 fluence, the maiximum allowable vessel stress would not be exceeded for any of the previously-analyzed LTOP events. Pressure relief by pressurizer code safety valves, or shutdown cooling relief valves (depending upon the event considered), were found to be sufficient to limit peak pressures to below maximum allowable values.

Figure 1


Partial Section View of Power Operated Relief Valve (PORV)
OUTLET
MAIN DISC
MAIN DISC SEAT
MAIN DISC GUIDE
SPRING
Sleeve on port
RETAINER PLUG
GASKET
Equalizing Port
MAIN DISC GUIDE
(Proper Orientation)
Detail of Main Disc & Main Disc Guide Assembly

REFERENCE PACKAGE RII
EA NO: 95-180
RII/EICS NO: 95-E-026

ENCLOSURE NO: 6

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified applicability condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

4.0.5 (Continued)

| ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities | Required frequencies for performing inservice inspection and testing activities |
|---|---|
| Weekly | At least once per 7 days |
| Monthly | At least once per 31 days |
| Quarterly or every 3 months | At least once per 92 days |
| Semiannually or every 6 months | At least once per 184 days |
| Yearly or annually | At least once per 366 days |

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

REACTOR COOLANT SYSTEM

POWER OPERATED RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.13 Two power operated relief valves (PORVs) shall be OPERABLE, with their setpoints selected t the low temperature mode of operation as follows:

a. A setpoint of less than or equal to 350 psia shall be selected:

1. During cooldown when the temperature of any RCS cold leg is less than or equal to 215°F and

2. During heatup and isothermal conditions when the temperature of any RCS cold leg is less than or equal to 193°F.

b. A setpoint of less than or equal to 530 psia shall be selected:

1. During cooldown when the temperature of any RCS cold leg is greater than 215°F and less than or equal to 281°F.

2. During heatup and isothermal conditions when the temperature of any RCS cold leg is greater than or equal to 193°F and less than or equal to 304°F.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 304°F, MODE 5, and MODE 6 when the head is on the reactor vessel; and the RCS is not vented through greater than a 1.75 square inch vent.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days; or depressurize and vent the RCS through greater than a 1.75 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through greater than a 1.75 square inch vent within a total of 32 hours.

c. With both PORVs inoperable, restore at least one PORV to operable status or complete depressurization and venting of the RCS through greater than a 1.75 square inch vent within 24 hours.

d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway every 12 hours.

e. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.13 Each PORV shall be demonstrated OPERABLE by:

a. Verifying the PORV isolation valve is open at least once per 72 hours; and

b. Performance of a CHANNEL FUNCTION TEST, but excluding valve operation, at least once per 31 days; and

c. Performance of a CHANNEL CALIBRATION at least once per 18 months.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified applicability condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

TABLE 5.5-5

PRESSURIZER POWER-OPERATED RELIEF VALVE PARAMETERS

| | |
|---|---|
| Tag Numbers | V-1404, V-1402 |
| Design Pressure, psig | 2485 |
| Design Temperature, F | 675 |
| Fluid | Saturated Steam, 0.1% (wt) Boric Acid |
| Capacity, lb/hr | 153,000 |
| Type | Solenoid Operated Voltage 125 vdc |
| Size | 2 1/2" - 2500 lb by 4" - 300 lb |
| Sat Pressure, psig | 2385 |
| Body Material | ASTM-A-182 F-316 |
| Nozzle Material | ASTM-A-182 F-316 (Stellited) |
| Disc | Inconel 718 |
| Code | ASME Code for Pumps and Valves for Nuclear Power Class I, Nov. 1968 Draft |

TIMD030 COMPONENT/ASSOCIATE DATA 10/24/95 12:33

Help Data Print Roadmap Exit Options

Facility = PSL Unit= 01 Comp. Status: OK As Of: 07/23/90
Component = V1404 Associate = NPRDS: Y (Y/N)
System = 01 REACTOR COOLANT SYSTEM
Name : POWER OPERATED RELIEF VALVE (PORV) FOR PRESSURIZER TO QUENCH TANK
Comp. Type = * Sub-Type = Tech Spec : (Y/N)
Train : Safety Channel: IST Reqd. : Y (Y/N)
Loc. Code : RCB/80/S-31/E-36 RWP Reqd. : Y (Y/N)
Loc. Desc. :
Comp. Group = EQ Related: Safety Cls:
EQ Doc Pac : Critical Comp.: Q Group : A
Seismic : I Startup Sys. : 034B P Class :
Orig. P.O. : CE-1900(9903305) Work Group : Q Basis :
Reg. 1.97 : Reg. Type: EQ Remarks :
Reg. Catg. : EQ Tab : EQ Comp. Tag : Rev.:

Mfr Code : Installed: Eng. Ver :
Model No. : P.O. No. :
Serial No : UTC: Stock Cd:

F9 TO VIEW ADDITIONAL COMPONENT DATA. F10 TO ACCESS CONFIGURATION X-REF.
F1=Help F4=Prompt F5=Search F6=Refresh F10=Perform Msg F12=Cancel

TOTAL P.02

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Help Data Print Roadmap Exit Options

Facility = PSL Unit= 01 Comp. Status: OK As Of: 07/23/90
Component = V1405 Associate = NPRDS: Y (Y/N)
System = 01 REACTOR COOLANT SYSTEM
Name : MOTOR OPERATED ISOLATION VALVE UPSTRM OF PZR RELIEF VALVE V1404
Comp. Type = * Sub-Type = Tech Spec : (Y/N)
Train : Safety Channel: IST Reqd. : Y (Y/N)
Loc. Code : RCB/79/S-29/E-29 RWP Reqd. : Y (Y/N)
Loc. Desc. :
Comp. Group = EQ Related: Safety Cls:
EQ Doc Pac : Critical Comp.: Q Group : A
Seismic : I Startup Sys. : 034B P Class :
Orig. P.O. : CE-1900(9002051) Work Group : Q Basis :
Reg. 1.97 : Reg. Type: EQ Remarks :
Reg. Catg. : EQ Tab : EQ Comp. Tag : Rev.:

Mfr Code : Installed: Eng. Ver :
Model No. : P.C. No. :
Serial No : UTC: Stock Cd:

F9 TO VIEW ADDITIONAL COMPONENT DATA. F10 TO ACCESS CONFIGURATION X-REF.
F1=Help F4=Prompt F5=Search F6=Refresh F10=Perform Msg F12=Cancel

## 6.0 ADMINISTRATIVE CONTROLS

### 6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor, or during his absence from the control room a designated individual, shall be responsible for the control room command function. A management directive to this effect, signed by the President - Nuclear Division, shall be reissued to all station personnel on an annual basis.

### 6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for unit operation and corporate management. This onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Topical Quality Assurance Report and updated in accordance with 10 CFR 50.54(a)(3).

b. The President - Nuclear Division shall be responsible for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the health physics manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

REACTOR COOLANT SYSTEM

POWER OPERATED RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.13 Two power operated relief valves (PORVs) shall be OPERABLE, with their setpoints selected to the low temperature mode of operation as follows:

a. A setpoint of less than or equal to 350 psia shall be selected:

1. During cooldown when the temperature of any RCS cold leg is less than or equal to 215°F and

2. During heatup and isothermal conditions when the temperature of any RCS cold leg is less than or equal to 193°F.

b. A setpoint of less than or equal to 530 psia shall be selected:

1. During cooldown when the temperature of any RCS cold leg is greater than 215°F and less than or equal to 281°F.

2. During heatup and isothermal conditions when the temperature of any RCS cold leg is greater than or equal to 193°F and less than or equal to 304°F.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 304°F, MODE 5, and MODE 6 when the head is on the reactor vessel; and the RCS is not vented through greater than a 1.75 square inch vent.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days; or depressurize and vent the RCS through greater than a 1.75 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through greater than a 1.75 square inch vent within a total of 32 hours.

c. With both PORVs inoperable, restore at least one PORV to operable status or complete depressurization and venting of the RCS through greater than a 1.75 square inch vent within 24 hours.

d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position, otherwise, verify the vent pathway every 12 hours.

e. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.13 Each PORV shall be demonstrated OPERABLE by:

a. Verifying the PORV isolation valve is open at least once per 72 hours; and

b. Performance of a CHANNEL FUNCTION TEST, but excluding valve operation, at least once per 31 days; and

c. Performance of a CHANNEL CALIBRATION at least once per 18 months.

# ARTICLE IWV-2000
# DEFINITIONS

## IWV-2100 ACTIVE AND PASSIVE VALVES

*(a) active valves* — valves which are required to change position to accomplish a specific function

*(b) passive valves* — valves which are not required to change position to accomplish a specific function

## IWV-2200 CATEGORIES OF VALVES

Valves within the scope of this Subsection shall be placed in one or more of the following categories. When more than one distinguishing category characteristic is applicable, all requirements of each of the individual categories are applicable, although duplication or repetition of common testing requirements is not necessary.

*(a) Category A* — valves for which seat leakage is limited to a specific maximum amount in the closed position of fulfillment of their function

*(b) Category B* — valves for which seat leakage in the closed position is inconsequential for fulfillment of their function

*(c) Category C* — valves which are self-actuating in response to some system characteristic, such as pressure (relief valves) or flow direction (check valves)

*(d) Category D* — valves which are actuated by an energy source capable of only one operation, such as rupture disks or explosively actuated valves

## IWV-2300 OTHER DEFINITIONS

*(a) exercising* — the demonstration based on direct or indirect visual or other positive indication that the moving parts of a valve function satisfactorily

*(b) inservice life* — the period of time from installation and acceptance until retired from service

*(c) inservice test* — a special test procedure for obtaining information through measurement or observation to determine the operational readiness of a valve

*(d) maintenance* — routine valve servicing or work on a valve undertaken to correct or prevent an abnormal or unsatisfactory condition

*(e) operational readiness* — the capability of a valve to fulfill its function

# ARTICLE IWV-3000
# TEST REQUIREMENTS

## IWV-3100 PRESERVICE TESTS

Each valve, after installation and prior to service, shall be tested as required by this Subsection. These tests shall be conducted under conditions similar to those to be experienced during subsequent inservice tests. Safety and relief valves which will be removed and bench tested during subsequent inservice tests need not be installed prior to the preservice test.

## IWV-3200 VALVE REPLACEMENT, REPAIR, AND MAINTENANCE

When a valve or its control system has been replaced or repaired or has undergone maintenance[1] that could affect its performance, and prior to the time it is returned to service, it shall be tested to demonstrate that the performance parameters, which could be affected by the replacement, repair, or maintenance, are within acceptable limits.

## IWV-3300 VALVE POSITION INDICATOR VERIFICATION

Valves with remote position indicators shall be observed at least once every 2 years to verify that valve operation is accurately indicated.

[1]Examples of maintenance that could affect valve performance parameters are: adjustment of stem packing; removal of the bonnet, stem assembly, or actuator; and disconnection of hydraulic or electrical lines.

## IWV-3400 INSERVICE TESTS — CATEGORY A AND B VALVES

### IWV-3410 VALVE EXERCISING TEST

#### IWV-3411 Test Frequency

Category A and B valves shall be exercised at least once every 3 months, except as provided by IWV-3412(a), IWV-3415, and IWV-3416.

#### IWV-3412 Exercising Procedure

*(a)* Valves shall be exercised to the position required to fulfill their function unless such operation is not practical during plant operation. If only limited operation is practical during plant operation, the valve shall be part-stroke exercised during plant operation and full-stroke exercised during cold shutdowns. Valves that cannot be exercised during plant operation shall be specifically identified by the Owner and shall be full-stroke exercised during cold shutdowns. Full-stroke exercising during cold shutdowns for all valves not full-stroke exercised during plant operation shall be on a frequency determined by the intervals between shutdowns as follows:

For intervals of 3 months or longer — exercise during each shutdown.

For intervals of less than 3 months — full-stroke exercise is not required unless 3 months have passed since last shutdown exercise.

*(b)* The necessary valve disk movement shall be determined by exercising the valve while observing an appropriate indicator, which signals the required change of disk position, or observing indirect evidence (such as changes in system pressure, flow rate, level, or temperature), which reflect stem or disk position.

#### IWV-3413 Power Operated Valves

*(a)* The limiting value of full-stroke time of each power operated valve shall be specified by the Owner.

*Full-stroke time* is that time interval from initiation of the actuating signal to the end of the actuating cycle.

*(b)* The stroke time of all power operated valves shall be measured to the nearest second, for stroke times 10 sec or less, or 10% of the specified limiting stroke time for full-stroke times longer than 10 sec, whenever such a valve is full-stroke tested.

### IWV-3414 Valves in Regular Use

Valves that operate in the course of plant operation at a frequency which would satisfy the exercising requirements of this Subsection need not be additionally exercised, provided the observations otherwise required for testing are made and analyzed during such operation, and are recorded in the plant record at intervals no greater than specified in IWV-3411.

### IWV-3415 Fail-Safe Valves

When practical, valves with fail-safe actuators shall be tested by observing the operation of the valves upon loss of actuator power. If these valves cannot be tested once every 3 months, they shall be tested during each cold shutdown; in case of frequent cold shutdowns, these valves need not be tested more often than once every 3 months.

### IWV-3416 Valves in Systems Out of Service

For a valve in a system declared inoperable or not required to be operable, the exercising test schedule need not be followed. Within 30 days prior to return of the system to operable status, the valves shall be exercised and the schedule resumed in accordance with requirements of this Article.

### IWV-3417 Corrective Action

*(a)* If, for power operated valves, an increase in stroke time of 25% or more from the previous test for valves with full-stroke times greater than 10 sec, or 50% or more for valves with full-stroke times less than or equal to 10 sec is observed, test frequency shall be increased to once each month until corrective action is taken, at which time the original test frequency shall be resumed. In any case, any abnormality or erratic action shall be reported.

*(b)* If a valve fails to exhibit the required change of valve stem or disk position or exceeds its specified limiting value of full-stroke time by this testing, then corrective action shall be initiated immediately. If the condition is not, or cannot be, corrected within 24 hr, the valve shall be declared inoperative. When corrective action is required as a result of tests made during cold shutdown, the condition shall be corrected before startup. A retest showing acceptable operation shall be run following any required corrective action before the valve is returned to service.

## IWV-3420 VALVE LEAK RATE TEST

### IWV-3421 Scope

Category A valves shall be leak tested, except that valves which function in the course of plant operation in a manner that demonstrates functionally adequate seat tightness need not be leak tested. In such cases, the valve record shall provide the basis for the conclusion that operational observations constitute satisfactory demonstration.

### IWV-3422 Frequency

Tests shall be conducted at least once every 2 years.

### IWV-3423 Differential Test Pressure

Valve seat leakage tests shall be made with the pressure differential in the same direction as when the valve is performing its function, with the following exceptions.

*(a)* Globe-type valves may be tested with pressure under the seat.

*(b)* Butterfly valves may be tested in either direction, provided their seat construction is designed for sealing against pressure on either side.

*(c)* Gate valves with two-piece disks may be tested by pressurizing them between the seats.

*(d)* Valves (except check valves) may be tested in either direction if the function differential pressure is 15 psi or less.

*(e)* Leakage tests involving pressure differentials lower than function pressure differentials are permitted in those types of valves in which service pressure will tend to diminish the overall leakage channel opening, as by pressing the disk into or onto the seat with greater force. Gate valves, check valves, and globe-type valves, having function pressure differential applied over the seat, are examples of valve applications satisfying this requirement. When leakage tests are made in such cases using pressures lower than function maximum pressure differential, the observed

# NRC CLOSED PREDECISIONAL ENFORCEMENT CONFERENCE

## ST LUCIE NUCLEAR PLANT

## MARCH 8, 1996

| TAB | TITLE |
|---|---|
| 1 | Predecisional Enforcement Conference Agenda |
| 2 | Expected Attendees, Meeting Announcement |
| 3 | Opening Remarks and Introductions |
| 4 | NRC Enforcement Policy |
| 5 | Summary of the Issues |
| 6 | Statement of Concerns / Apparent Violations |
| 7 | Inspection Report No. 50-335,389/96-03 |
| 8 | Unit 1 Control Room Arrangement, CVCS Charging System Flow Diagram, Enforcement Pre-Panel Questionnaire |
| 9 | Licensee Procedure OP 1-0250020, Boron Concentration Control - Normal Operation; and TC 1-96-017 to OP 1-0250020 of 1/23/96 |
| 10 | Licensee Procedure QI5-PR/PSL-1, Preparation, Revision, Review/Approval of Procedures |
| 11 | Licensee Procedure AP 0010120, Conduct of Operations; and TC 0-96-014 to AP 0010120 of 1/29/96 |



1-11-16

| | |
|---|---|
| 12 | Memo from E. Jordan on Licensed Power Level of 8/22/80 |
| 13 | St. Lucie Unit 1 FSAR |
| 14 | Closing Remarks |

# PREDECISIONAL ENFORCEMENT CONFERENCE AGENDA

## ST LUCIE
## MARCH 8, 1996, AT 10:30 A.M.
## NRC REGION II OFFICE, ATLANTA, GEORGIA

I. OPENING REMARKS AND INTRODUCTIONS
S. Ebneter, Regional Administrator

II. NRC ENFORCEMENT POLICY
B. Uryc, Director
Enforcement and Investigation Coordination Staff

III. SUMMARY OF THE ISSUES
S. Ebneter, Regional Administrator

IV. STATEMENT OF CONCERNS / APPARENT VIOLATIONS
A. Gibson, Director
Division of Reactor Safety

V. LICENSEE PRESENTATION
T. Plunkett, President - Nuclear Division
Florida Power & Light Company

VI. BREAK / NRC CAUCUS

VII. NRC FOLLOWUP QUESTIONS

VIII. CLOSING REMARKS
S. Ebneter, Regional Administrator

## EXPECTED ATTENDEES

Licensee

Tom Plunkett, President - Nuclear Division, FPL
Bill Bohlke, Vice President, St. Lucie Nuclear Plant
Jim Scarola, Plant General Manager, St. Lucie
Dan Denver, Engineering Manager, St. Lucie
Ed Weinkam, Licensing Manager, St. Lucie
Peter Honeysett, Nuclear Plant Supervisor, St. Lucie
Frank Cone, Reactor Controls Operator, St. Lucie
Hank Holzmacher, Reactor Controls Operator, St. Lucie

NRC

Stev: Ebneter, Regional Administrator, Region II (RII)
Luis Reyes, Deputy Regional Administrator, RII
Al Gibson, Director, Division of Reactor Safety (DRS), RII
Ellis Merschoff, Director, Division of Reactor Projects (DRP), RII
Gene Imbro, Director. Project Directorate II-2, NRR
James Beall, Enforcement Coordinator, Office of Enforcement (OE)
Johns Jaudon, Deputy Director, DRS, RII
Jon Johnson, Deputy Director, DRP, RII
Bruno Uryc, Director, Enforcement and Investigation Coordination Staff (EICS), RII
Charles Casto, Chief, Engineering Branch, DRS, RII
Tom Peebles, Chief, Operations Branch, DRS, RII
Kerry Landis, Reactor Projects Branch 3, DRP, RII
Jan Norris, Project Manager, NRR
Linda Watson, Senior Enforcement Specialist, EICS, RII
Carolyn Evans, Regional Counsel, RII
Mark Miller, Senior Resident Inspector, St. Lucie, DRP, RII
Robert Schin, Reactor Inspector, Engineering Branch, DRS, RII
Edwin Lea, Project Engineer, Reactor Projects Branch 3, DRP, RII

**From:** Anne Boland, R2
**To:** ATP1.JWY J York, R2
**Date:** 7/24/96 11:15am
**Subject:** END DATE FOR ST. LUCIE INSPECTION

WHAT IS THE END DATE FOR THE ST. LUCIE INSPECTION WHICH IDENTIFIED THE 50.59 VIOLATIONS?



III/8