**From:** Linda J. Watson (LJW2)
**To:** ATB, BXU
**Date:** Tuesday, October 10, 1995 10:42 am
**Subject:** St Lucie relief vlv

K. Landis indicated that he was going to re-exit the St. Lucie relief vlv issue as an apparent violation on 10/11 and would proceed to set up enf. conf. He will be issuing separate report on this issue approx. 10/20.

Need E File

# EA NUMBER REQUEST FORM

| | | | |
|---|---|---|---|
| TO: OEMAIL OR FAX TO OE | | | |
| FROM: | ANNE T. BOLAND | REGIONAL CONTACT | |
| DATE OF REQUEST | SEPTEMBER 29, 1995 | REGION | II |
| LICENSEE | FLORIDA POWER AND LIGHT COMPANY | | |
| FACILITY/LOCATION | ST. LUCIE / JENSEN BEACH, FLORIDA | UNIT | 1 |
| LICENSE/DOCKET NO(S). | DPR-67, 50-335 | | |
| LAST DAY OF INSPECTION | SEPTEMBER 16, 1995 | | |
| OI REPORT NO. | NONE | DATE OF OI REPORT | N/A |

**SUMMARY OF FACTS OF CASE (ANNUAL REPORT FORMAT FOR EATS ENTRY) (MAXIMUM OF 300 CHARACTERS)**

FAILURE TO TAKE PROMPT CORRECTIVE ACTION RESULTING IN THE FAILURE OF A UNIT 1 RELIEF VALVE TO RESEAT WITHOUT OPERATOR INTERVENTION. THE EVENT RESULTED IN APPROXIMATELY 4000 GALLONS OF REACTOR COOLANT ACCUMULATING IN THE UNIT 1 PIPE TUNNEL.

**BRIEF SUMMARY OF INSPECTION FINDINGS (IF NOT SUFFICIENTLY DESCRIBED ABOVE)**

ON AUGUST 10, 1995, WHILE PLACING THE UNIT 1 SHUTDOWN COOLING SYSTEM IN SERVICE, THE A LPSI HEADER THERMAL RELIEF, LIFTED RESULTING IN THE LOSS OF APPOX. 3500-4000 GALLONS OF COOLANT INTO THE UNIT 1 PIPE TUNNEL. THE ROOT CAUSE OF THE PROBLEM WAS A DESIGN ISSUED ASSOCIATED WITH HIGH BLOWDOWN VALUES; HOWEVER, THE LICENSEE FAILED TO EVALUATE AND CORRECT ANOMALOUS RELIEF VALVE BEHAVIOR ON FEBRUARY 20, MARCH 2, AND MARCH 10, 1995, WHICH MAY HAVE PREVENT THE AUGUST EVENT. OVER 100 VALVES WERE EVALUATED FOR THIS CONDUCTION AND 15 REQUIRED REPAIR AND/OR REPLACEMENT. NO SAFETY SYSTEMS WERE DETERMINED TO BE INOPERABLE AS A RESULT OF THIS PHENOMENON.

**REASON FOR POTENTIAL ESCALATED ACTION**

SUPPLEMENT I.C.2.B, A SYSTEM DESIGNED TO PREVENT OR MITIGATE A SERIOUS SAFETY EVENT BEING DEGRADED TO THE EXTENT THAT A DETAILED EVALUATION IS REQUIRED TO DETERMINE ITS OPERABILITY.

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| DELEGATED CASE | | | YES | X | NO | | | |
| | MED INST | | PHYSICIAN | | NUC PHARM | | RADIOG | IRRAD |
| | WELL LOGGERS | | ACADEMIC | | GAUGE | | MOISTURE DENSITY | |
| | OTHER TYPE: | | | | | | | |

| | |
|---|---|
| CITE SIMILAR CASE: EA NO. | SUPPLEMENT 1.C.2.B |

| | | | | |
|---|---|---|---|---|
| SHOULD OE ATTEND ENF CONF | X | YES | | NO |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| NONDELEGATED CASE | | X | YES | | NO | | |
| X | NONDELEGABLE TYPE | | OI REPORT/WILLFUL | | COMPLEX/NOVEL | | |
| | DISCRETION | | COMM APPROVAL | | OI INTEREST | | SL 1 OR 2 |
| | OTHER REASON: | | | | | | |

IS THERE A BASIS TO CLOSE ENFORCEMENT CONFERENCE? NO IF YES, EXPLAIN:

ENFORCEMENT CONFERENCE TO BE SCHEDULED.

| | | | | | |
|---|---|---|---|---|---|
| EA # ASSIGNED BY OE | 95-227 | DATE: | | ES ASSIGNED | |

EICS STAFF NOTES

FACILITY: ST. LUCIE

[X] ENF PANEL
[ ] ENF CONF
[ ] CAUCUS
[ ] OTHER

DATE: 09-29-95
EICS NO:
IR NO:
DUE TO OE:
EA NO:

INSPECTION END DATE:
INSPECTOR:
RESPONSIBLE DIVISION:
RESPONSIBLE SECTION:
START: 0900 TERM:

ATTENDEES

| | |
|---|---|
| MERSCHOFF | HAYES * |
| URYC | BEALL * |
| EVANS | GRAY * |
| LEE | |
| LANDIS | |
| BOLAND | |
| PEEBLES | |
| PREVATTE | |
| LIEBERMAN * | |

SUBJECT: TS violation re valves

prompt review for extent of condition
review 5 months too long to take
corrective action

Criterion 16 VIO - OK -

JL - was it scheduled to be done
at first available outage -

relevant

because 3 events - we expected that
action should have been taken sooner
to correct

vulnerability that there could have
been a significant radiological event
with bypass

problem should have been addressed
after Feb/March events -

OE NOTIFIED: SL: SUP: CP:
ENF CONF SCHEDULED: OPEN/CLOSED:

C:\ADMINIS.DIR\ENFNOTE.95 (December 30, 1994)

95E035

St. Lucie

HIGH STAFF NOTES ... FACILITY: St. Lucie

[✓] ENF PANEL
[ ] ENF CONF
[ ] CAUCUS
[ ] Other

DATE: 9/29/95
CASE NO:
IR NO:
DUE TO OE:
EA NO:

ATTENDEES:
Bagavan, Lieberman, Merschoff, Landis, Evans, Lea, Trice, Boland
Beall, Burdette, Fuller, Gray

LAST DATE OF INSPECTION:
INSPECTOR:
START: TERM:
RESPONSIBLE DIVISION:
RESPONSIBLE SECTION:

SUBJECT:

Failure to take C.A.

NOTES:

3500-4000 gal. blowdown into pipe tunnel

Failure to take C.A.
similar problems with blowdown earlier in the year
licensee knew of problem
< 100 valves be affected
required NRC pressing & questions

After NRC intervention → good follow-up

Leak outside of containment

Lieberman - what is the real weakness
reset point < operating pressure
initiated non-conformance reports
NRC identified 21 valves which needed to be evaluated
15 safety valves needed to be replaced or setpoints reset

OE NOTIFIED: SL: SUP: CP:
EC SCHEDULED: LOCATION:

Specification issue - expect vendor (Crosby) to issue Part 21
valves installed @ construction → not a post maintenance test problem
design basis reverification work should have pick-up? Don't know
After 2/95 failure → STAR → 3/2 → STAR 3/10 → All sent to engineering for review - WAS the time reasonable 30-60 days seems reasonable (?)
ENM - after 3rd event should have been prompt
→ After transient - did not go & evaluate other valves of other systems - until NRC prompted
JXL Prevatte to determine → Current outage fixing → before 8/95 event was this planned to be worked during outage.
Criterion XVI Violation

Consequences / Erosion / Loss of S/D cooling & loss of inventory outside containment

LPCI valve lift → artifact of attempting to open S/D cooling

If happened during a LOCA - release to environment & modes of S/D cooling would not be available

Operating - could have lived / evaluated / tested

Potential release path → dose could potentially exceed Part 100 dose - highly dependent on the nature of liquid be released - Small Break LOCA

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Enforcement and Investigation
Coordination Staff

DATE: 9/29/95

PAGE: 3

dependent on [I°] and temperature

The release that actually occurred →
contamination in pipe tunnel → some exposure
~~0 license dose. 160 less than NRC for workers~~
~~4000 gal. →~~
minimal impact

Loss of Safety Function for 15 Valves? No
2PCI would have been able to perform
safety function in accident
Other 5 systems would have worked
NRC did not fully appreciate in March

~~15 Not met for SDr Cooling~~

Pervatte → Program for Testing Valves - Periodic
at least the program would possibly
be inadequate

Significant Regulatory Concern -
Significant ~~exposure~~ or potential
release

Consensus — III

Need PE: - want to hear how
they address extent of problems -
- training

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Enforcement and Investigation
Coordination Staff

DATE: ______ PAGE: ______

Enforcement history → No prior to inspection (+ one ext program) (doesn't count)

C.A. Root cause – how have they improved C.A. program

ID missed opportunities – N/A

Discretion – ? Base or NO/Penalty

considered

Long run reliability time – big problem if accident – BASIS FOR DISCRETION

EWM wants from NRR

Need Core Damage risk assessment

PORVs

Stretch – issue – cumulative effect of errors drive to higher penalty

– missed opportunities

– multiple barriers

– Reported –

BRUNO URYC, Jr.
Enforcement and Investigation
Coordination Staff

ESCALATED ENFORCEMENT
PANEL QUESTIONNAIRE

INFORMATION REQUIRED TO BE AVAILABLE FOR ENFORCEMENT PANEL

PREPARED BY: R. Prevatte

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. ucie

   Unit(s): 1

   Docket Nos: 50-335

   License Nos: DPR-67

   Inspection Dates: July 30 - September 16, 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.C.2.B

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

   Engineering evaluation and prioritization of potential equipment problem was not timely.

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

   Improve prioritization and timeliness of response to plant problems.

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?)

Licensee identified anomalous behavior of safety related thermal relief valves on February 20, March 2, and March 10, 1995, but did not take action until a failure also occurred on August 10, 1995 and NRC questioned corrective action.

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.

See item A.

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?

Initial problem was under engineering review for several months. After questioning by NRC, the problem was thoroughly researched and corrected.

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

Initial - not timely.

Final - good investigation and broadened scope led to review of over 100 relief valves.

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):

NCV 94-25-01, Inadequate design control of NAOH suction relief valves.

VIO 94-11-01, Inadequate corrective action for MOV which stalled during surveillance.

VIO 94-12-01, 1E swing bus would not strip on undervoltage due to wiring problem

94-08-01, Inadequate corrective action on waterhammer event.

Inoperable snubbers and SRV PORV tailpipes.

94-08-02, Failure to document above non-conformance.

94-06-02, Inadequate design control on Unit 2 charging pump sequence.

94-06-01, Failure to report DG failure.

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:

Eng. Support 1 - 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?

Problem known but not pursued.

e. MULTIPLE OCCURRENCES: Were there multiple examples of the violation identified during this inspection? If there were, identify the number of examples and briefly describe each one.

No.

f. DURATION: How long did the violation exist?

Problem has existed on thermal relief valves since initial installation.

ADDITIONAL COMMENTS/NOTES:

5) Shutdown Cooling Relief Valve Lift

A. Background

On February 28, while placing the 1A SDC train in service, the licensee experienced a lift of 1A LPSI pump suction relief valve V-3483 (see IR 95-04). The valve did not reseat, and the loss of RCS inventory was abated by closing LPSI hot leg suction isolation valves V-3480 and V-3481, which isolated the valve from RCS pressure. The root cause of the lift was determined to be water hammer, which resulted from passing relatively hot RCS fluid through the suction line at high velocity as the LPSI pump was started. As corrective action, the licensee revised OP 1-0410022, "Shutdown Cooling," to change the methodology of starting the LPSI pump to the following:

- Shut LPSI pump discharge isolation and LPSI header isolation valves
- Start the LPSI pump
- Immediately open the LPSI pump isolation valve
- Throttle open two LPSI header isolations to 150 gpm per header
- Run for 15 minutes
- Start the second pump
- Throttle open the remaining LPSI header isolation valves to 150 gpm per header
- Wait 5 minutes
- Incrementally open header isolation valves to obtain full flow.

The licensee reasoned that this methodology would result in a slow increase in flow, allowing controlled system heatup and minimizing the potential for water hammer.

B. LPSI Discharge Isolation Valve Lift

On August 10, while placing the Unit 1 SDC system in service to support a cooldown required due to inoperable PORVs (see IR 335/95-16), V-3439, the A LPSI header thermal relief, lifted resulting in a loss of approximately 3500-4000 gallons of RCS coolant in the Unit 1 Pipe tunnel. The following timeline was developed from operator interviews, logs and instrumentation data:

0018 A LPSI pump start (ANPS, NWE, Logs)
Pressurizer level begins to drop (strip chart data)

0025 ANPS directs SNPO to tour pipe tunnel due to minor reduction in pressurizer level (ANPS)
No increases in HUT, RWT, etc noted (ANPS)
SNPO reports no unusual conditions in pipe tunnel

0105 B LPSI pump start (ANPS, NWE, Log)
Pressurizer level recovers and oscillates (strip chart)

0140 Cooldown flow established (ANPS, NWE)

0210 Fire watch calls control room, reports water issuing from watertight door isolating pipe tunnel from RAB (ANPS, NWE)

0215 SDC secured (ANPS, NWE)
Pressurizer level increases and stabilizes (strip chart)

0226 Floor drain isolation valves (FCV 25-1 through 7) noted to be closed on control panel (ANPS, NWE)
Drain valves subsequently opened (ANPS, NWE)
Flooding in RAB ONOP entered (ANPS)
Water levels in pipe tunnel weren't dropping due to clogged floor drains (NWE)

0345 Water in pipe tunnel pumped by maintenance personnel to floor drains in RAB (ANPS)
Operators cycle various isolation valves looking for leak

0611 1A LPSI pump started with NWE observing in pipe tunnel (ANPS)

0612 NWE identifies V-3439 as passing water (ANPS)

The licensee concluded that the cause of the relief valve lift was a pressure surge while LPSI pumps were operating in a low-flow condition. The combination of RCS pressure (a maximum of 267 psia at the time) and LPSI pump discharge head at essentially no flow (approximately 182 psid) combined with possible perturbations (when starting the pump) was considered enough to challenge the relief valve setpoint (485-515). This condition existed from the time the 1A LPSI pump discharge isolation valve was opened until operators initiated flow through the LPSI header isolation valves.

V-3439 was designed to provide a 10 percent blowdown, which, if applied to the lower acceptable lift setpoint of the valve (485 psig), would require a 48.5 psia reduction in pressure to allow reseat. Given these parameters, should V-3439 open, RCS pressure would have to drop to 436.5 psia to allow valve reseat (assuming only a 10 percent blowdown). The volume of the RCS and pressurizer would preclude such a reseat until significant volumes of coolant were lost.

The volume of coolant lost during the event was

estimated by the inspector, based upon floor layouts as displayed on drawings. Given water depths reported by the NWE (up to approximately 14" in some areas), the inspector estimated that approximately 3500 gallons were lost. The CVCS makeup integrator, measuring volume added to the VCT in maintaining pressurizer level on setpoint, indicated that 4000 gallons were added to the VCT.

The licensee concluded that the closed floor drain isolation valves, HCV-25-1 through 7 (a set of 7 ganged valves) were the result of valve stroke testing in preparation for Hurricane Erin. During testing conducted by control room operators, some of the valves had failed to stroke properly. As a result, the valves were left closed for troubleshooting and were never reopened. OP 1-0010123, Rev 99, "Administrative Control of Valves, Locks, and Switches," required, in step 8.1.6, that "All valve or switch position deviations or lock openings shall be documented in Appendix C, Valve Switch Deviation Log..." The inspector reviewed archived Appendix C logs completed in July and August and control room open Appendix C logs and found no evidence that HCV-25-1 through 7 were logged as being out of position. The failure to enter the valves' closed status into the valve deviation log is an example of a violation (VIO 335/95-15-01, "Failure to Follow Procedures," Example 4). STAR 950917 was initiated to develop a PM for verifying that floor drains were unclogged.

The licensee prepared an evaluation of the effects of the subject setpoint/blowdown values on plant operation. JPN-PSL-SENP-95-101, Rev 1, "Assessment of the Effects on Plant Operation of Lifting the LPSI Pump Discharge Header Thermal Relief Valve," concluded that the subject condition would not have a significant effect on safe plant operation during normal, shutdown, and design basis accident conditions. In reaching this conclusion, the evaluation noted the following:

- As flowrate through the relief valve (at lift setpoint pressure) was approximately 40 gpm, the loss of inventory was within charging system capacity (44 gpm per pump).

- During the injection phase of an accident, the LPSI pumps would draw suction from the RWT, thus pressure developed by the pump would not compound a high pressure suction source and the relief valve's lift setpoint would not be challenged.

- The relief valve in question discharged to a floor drain which directed flow to the safeguards room sump. The sump was designed to be pumped down in level to the EDT automatically when offsite power is available. Thus, with offsite power available, no flooding hazard would exist. Under conditions with no offsite power available, the water level in the safeguards room (after the sump overfilled) would not rise to the level of the HPSI pump motors until approximately 7 hours after the lift. Before this time elapsed, the licensee reasoned that sump high level alarms would alert operators to the event, allowing operator intervention prior to the loss of the HPSI pump.

- The licensee noted that, while SDC was assumed to be placed in service during postulated small break LOCAs, ESDEs, and SGTRs (when RCS pressure may have been high enough to have led to a relief valve lift), the FSAR analysis demonstrated that fuel damage (and thus the release of significant amounts of radioactive material to the RCS) was involved only because of extremely conservative assumptions. The evaluation went on to state that "A review of FSAR analysis of small break LOCAs, ESDEs and SGTRs demonstrates that these accidents will not result in fuel damage if assumptions that reflect the actual operating history of the plant are applied. Therefore, the radiological consequences of these FSAR accidents will not be increased and the FSAR offsite doses remain bounding."

The inspector took exception to the licensee's conclusion. The subject passage was included in Section 4 of the evaluation, "Analysis of Effects of Lifting V3439," in a section entitled "Increases in Radiological Consequences of Design Basis Accidents." The inspector found that, in choosing to neglect design basis assumptions in their analysis of the event (specifically, a return to power and fuel failure resulting from the most reactive rod failing to insert), the licensee did not evaluate the increases in the radiological consequences of design basis accidents. Rather, the licensee evaluated the radiological consequences of a less significant set of accidents and concluded, without providing quantitative results, that the radiological consequences of design basis accidents bounded the noted relief valve lift. While the inspector agreed with the licensee's position that the circumstances assumed in design basis accidents were,

probablistically, of low likelihood, the inspector pointed out that the assumptions were the approved licensing basis of the plant and, as such, provided the appropriate common ground upon which to evaluate the event's significance. The inspector brought this to the attention of the licensee, who stated that they would consider the issue. At the close of the inspection period, the licensee had not presented a final position on the issue. As a result, this issue will be tracked as an unresolved item (URI 95-15-04, "Adequacy of Engineering Evaluation Regarding Unit 1 Loss of Inventory via V-3439").

On August 12, the inspector requested data on approximately 25 relief valves on both units which communicated with the RCS in some way. The requested data included lift and blowdown setpoints, tolerances, relief capacity, and normal operating pressures experienced by the valves. Shortly after requesting the information, the licensee informed the inspector that a team had been formed to evaluate all safety-related relief valve data. The team included members from Engineering, Maintenance, Operations, Tech Staff, and Licensing.

The team's review was documented in JPN-SPSL-95-0334, "St. Lucie Units 1 and 2 Design Review of Safety Related Relief Valves," transmitted to the site by letter dated August 30. The inspector found the methodology of the study to be sound, considering worst case combinations of system operating pressures, relief valve setpoint, and blowdown. Relief valves were evaluated for their margin to lift and blowdown margin (the difference between reseat pressure and normal system operating pressure). The document reported that, of 114 relief valves reviewed, 8 valves on Unit 1 and 5 valves on Unit 2 required further review due to unacceptable margins of lift or blowdown. Corrective Actions were specified for the following valves:

Unit 1 Valves

- V2324, V2325, and V2326 - Charging Pump Discharge Relief Valves - MEP 107-195M was issued to reduce the design superimposed backpressure from 165 psig to 115 psig.
- V2345 - Letdown Relief Valve - PC/M 108-195 issued to reduce letdown backpressure to 430 psig and to reduce the valve's blowdown from 25 percent to 15 percent.
- V3412 - HPSI 1B Discharge Header Relief Valve -

EP 115-95 was issued to increase the design setpoint from 1735 psig to 1750 psig and to reduce blowdown from 25 percent to 10 percent.

- V3417 - HPSI Pump 1A Discharge High Pressure Header Relief Valve -design setpoint increased from 2400 psig to 2485 psig and blowdown reduced from 25 percent to 15 percent.

- V3468 and V3483 - SDC Suction Relief Valves - STAR 950430 was issued to evaluate new setpoints and blowdown values.

Unit 2 Valves

- V2345 - Letdown Relief Valve - At the close of the inspection period, an EP was being prepared to implement actions similar to those implemented on Unit 1 for this valve.

- V3412 - HPSI 2B Discharge High Pressure Header Relief Valve - At the close of the inspection period, an EP was being prepared to reduce blowdown from 25 percent to 10 percent.

- V3417 - HPSI Pump 2A Discharge High Pressure Header Relief Valve - At the close of the inspection period, an EP was being prepared to increase the valve's setpoint from 2400 psig to 2485 psig and to reduce blowdown from 25 percent to 10 percent.

- V3439 and V3507 - Low Pressure A and B Discharge Relief Valves - At the close of the inspection period, an EP was being prepared to increase the valve's setpoint from 500 psig to 535 psig.

As a result of the licensee's investigation, and through discussions with vendors, the licensee determined that some relief valves had been provided with unacceptably high blowdown values. This was, apparently, due to procedural problems at the vendor's test facility. At the close of the inspection period, the vendor (Crosby) was considering the 10 CFR 21 ramifications of the issue. The licensee documented the conditions on STAR 951024. The inspector reviewed the STAR and noted that it had not been identified as an "N" STAR (indicating a nonconforming condition). The inspector brought this to the attention of QC, and the condition was corrected. The licensee identified the affected relief valves and segregated them appropriately.

The inspector reviewed the licensee's STAR database

for events similar to the subject event and found the following:

- STAR 2-950167, initiated February 20, documented the lifting of SDC heat exchanger CCW relief valve SR-14350 when stroking CCW "N" header isolation valves closed. Once open, the relief valve had to be isolated (by closing an upstream valve in the process line) to bring about a reseat.

- STAR 0-950234, initiated March 2, documented the fact that relief valves had lifted and that blowdown values placed the reseat pressure of the valves in the operating ranges of the systems they protected.

- STAR 1-950269, initiated March 10, documented relief valve lifts on the Unit 1 CVCS letdown line during evolutions which should not have challenged the valve's setpoint.

- STAR 0-950917, initiated August 18, documented the subject SDC relief valve lift.

In addition to the STARs referenced above, IR 95-05-01 discussed work performed on the Unit 2 CVCS system to prevent letdown line relief valve lifts. The IR also described the failure of the relief valve to reseat (once lifted) due to a blowdown value which placed the reseat pressure below the system's normal operating pressure.

The inspector reviewed the status of the STARs listed above and found them all to be open. The inspector discussed the timeliness of the resolutions to the subject STARs with the licensee. The licensee stated that their focus had been on the methodologies for setting blowdown values on the valves in question, rather than on the appropriateness of the setpoints themselves. The licensee offered STAR 950234 as being illustrative of this point. The proposed corrective actions included:

- Completion of SRV test benches, which would allow the licensee to better set and test SRVs for lift setpoint and accumulation. It was noted that the bench had only limited blowdown test capability.

- Performing an engineering design basis review of all safety related SRVs to validate or correct setpoints and issue a design document that summarizes the design data.

- Enhancing journeyman training on SRVs.

While the inspector found the licensee's proposed activities prudent, it was noted that nothing precluded engineering from addressing the setpoint issue earlier in the process. The licensee stated that the STAR was addressed in stepwise fashion and that the maintenance-related items were addressed prior to forwarding the STAR to engineering.

The inspector found that the licensee's corrective actions for the subject event were comprehensive and sound. However, the inspector concluded that the actions could have reasonably been expected to be performed in a much more timely fashion. The subject phenomenon was identified as early as February, 1994, and repeated itself on no less than 3 separate systems, and on both units, prior to the most recent event. Once focused on the issue, an engineering product of high quality was developed, and corrective actions initiated, in approximately 2 weeks and identified valves requiring attention in a comprehensive action. 10 CFR 50, Appendix B required that, for conditions adverse to quality, prompt corrective action be taken to prevent recurrence. The licensee's failure to take prompt corrective action to the February/March events is a violation (VIO 335/95-15-02, "Failure to Take Prompt Corrective Actions for Repeated Relief Valve Lifts").

St. Lucie

# EVALUATION OF SAFETY SIGNIFICANCE

**EVENT**

- Estimated 4000 gallons reactor coolant released into Unit 1 RAB following lifting of LPSI discharge thermal relief V3439 on 8/10/95 during shutdown cooling operation.

**OBJECTIVE**

- Assess safety significance of event to identify reporting and/or corrective actions.

**APPROACH**

- Evaluated effects for alternate operating conditions (normal & accident):

  a. Flooding on safety related equipment

  b. Loss of RCS inventory

  c. Increases in radiological consequences

     i. Susceptible design basis accidents

     ii. Considered operating history of Unit 1

- Credible but conservative "backward look" is consistent with regulatory guidance and other utilities.

**RESULTS**

- Flooding detected prior to loss of ECCS pumps.

- Loss of RCS inventory (≈40 gpm) within makeup capability of charging and/or HPSI pumps.

- No increases in FSAR radiological consequences if operating history of the plant is considered.

  a. ESDEs: Essentially no fuel failure since all rods have always inserted on reactor trip.

  b. SBLOCAs: Essentially no fuel failure since very small break and inventory loss within capability of HPSI and/or charging.

c. SGTREs: No additional fuel failures caused by event; limited offsite releases (including leakage to RAB) since no operation with primary-to-secondary leakage and DEQ 131 at T/S limits.

d. In addition to the above event evaluations, a realistic dose calculation assuming some fuel failure shows no increase in FSAR radiological consequences.

- LER filed describing/analyzing event and corrective actions.

# EVALUATION OF SAFETY SIGNIFICANCE

(additional supporting information for engineering evaluation)

## APPROACH

An evaluation was performed to determine safety significance following the release of reactor coolant into the RAB from the lifting of V3439 during shutdown cooling operation on 8/10/95. The evaluation involved a realistic backward look at the potential consequences of FSAR design basis accidents assuming that V3439 was part of the accident scenario. Considering the operating history of the plant, the evaluation concluded that consequences of such accidents would be bounded by the offsite doses currently in the FSAR.

The evaluation focused on design basis accidents where RCS conditions necessary to lift the relief valve may be present. Excess steam demand events (ESDEs), small break loss of coolant accidents (SBLOCAs) and steam generator tube rupture events (SGTREs) were identified as the types of accidents that could involve partial depressurization of the RCS, and therefore, may involve shutdown cooling operation at a sufficiently high RCS pressure to cause V3439 to lift.

## ESDEs

ESDEs are characterized by rapid and significant cooling of the RCS due the faulted S/G. In these events, the most reactive control rod is normally assumed to be stuck out of the core. The positive reactivity addition from the stuck rod and the cooler RCS temperatures causes the reactor to experience an overpower transient which results in fuel exceeding minimum DNBR. St Lucie Plant has never had a failure of its control rods to insert upon reactor trip. With all rods inserted during an ESDE, the reactor would neither return to power, nor would it experience power distribution anomalies, and very limited fuel failure would be expected. The lifting of V3439 during shutdown cooling operation following such an event would not involve consequences exceeding FSAR offsite doses.

## SBLOCAs

The FSAR analyzed SBLOCA includes depressurization of the RCS, high pressure safety injection at 1275 psia, dumping of the safety injection tanks at 215 psia and low pressure safety injection at 200 psia. Shutdown cooling operation would occur at a very low RCS pressure after securing high pressure safety injection, and therefore, would not involve the lifting of V3439. Only small SBLOCAs, involving partial depressurization of the RCS, provide the

conditions necessary to lift V3439. For small SBLOCAs, either high pressure safety injection or charging will provide sufficient makeup, and fuel failure, if any, would be extremely limited. The lifting of V3439 during shutdown cooling operation following such an event would not involve consequences exceeding FSAR offsite doses.

SGTREs

Since the FSAR SGTRE involves shutdown cooling operation following a controlled reduction in RCS pressure, it could result in lifting of V3439. The FSAR analysis of this event provides offsite doses resulting from plant operation at the Technical Specification allowed maximum DEQ 131 concentration coupled with a maximum 1 gpm primary to secondary leak. No additional fuel failures occur as a result of this event. St Lucie Plant has neither operated with primary to secondary leakage nor with DEQ 131 concentrations at the Technical Specification limit. The lifting of V3439 during shutdown cooling operation following such an event would not involve consequences exceeding FSAR offsite doses.

# LER: ASSESSMENT OF SAFETY CONSEQUENCES

## SOURCES OF INFORMATION

**10 CFR 50.73(b)(3)** – The LER shall contain: "An assessment of the safety consequences and implication of the event. This assessment must include the availability of other systems or components that could have performed the same function as the components and systems that failed during the event."

**Statements of Consideration for 10 CFR 50.73 (b)(3) (48FR33857, dated July 26, 1983)** -- Paragraph (b)(3) requires that the LER include a summary assessment of the actual and potential safety consequences and implications of the event. This assessment may be based on the conditions existing at the time of the event. The evaluation must be carried out to the extent necessary to fully assess the safety consequences and safety margins associated with the event. An assessment of the event under alternative conditions must be included if the incident would have been more severe (e.g., the plant would have been in a condition not analyzed in the Safety Analysis Report) under reasonable and credible alternative conditions, such as power level or operating mode....

**NUREG -1022, Rev. 1 Second Draft, "Event Reporting Guidelines 10 CFR 50.72 and 50.73, pgs. 114 and 115** -- "Give a summary assessment of the actual and potential safety consequences and implications of the event, including the basis for submitting the report. Evaluate the event to the extent necessary to fully assess the safety consequences and safety margins associated with the event.

Include an assessment of the event under alternative conditions if the incident would have been more severe (e.g., the plant would have been in a condition not analyzed in its latest SAR) under reasonable and credible alternative conditions, such as a different operating mode. For example, if an event occurred while the plant was at low power and the same event could have occurred at full power, which would have resulted in considerably more serious consequences, this alternative condition should be assessed and the consequences reported.

Reasonable and credible alternative conditions may include normal plant operating conditions, potential accident conditions, or additional component failures, depending on the event. Normal alternative operating conditions and off-normal conditions expected to occur during the life of the plant should be considered. The intent of this section is to obtain the result of the considerations that are typical in the conduct of routine operations, such as event reviews, *not to require extraordinary studies.*"

**Other Versions of NUREG 1022, Sept. 1983; Supp. 1, Feb. 1984; Supp. 2, Sept. 1985; Rev. 1, Sept. 1991** – The other versions of NUREG 1022 are similar in wording; however, several notable differences are provided below:

NUREG 1022, Rev. 1, September 1991 – "A conclusion about the actual or implied effect on public health and safety of the event may be included as part of the assessment, but is not required."

NUREG 1022, Supplement 2, September 1985 – "A discussion of the safety consequences and implications is required in at least a few sentences or a paragraph that is clearly identifiable as a safety assessment. This discussion should indicate: (a) all of the safety consequences of the event including an assessment of the consequences had it been possible for the event to occur under a more severe set of initial conditions, or (b) if there were no safety consequences or implications, it should explicitly state why there were none."

NUREG 1022, Supplement 1, February 1984 – "*Question 12.5* Does the term "reasonable and credible" conditions really refer to normal plant operating conditions or to potential accident conditions? In addition, do we have to consider additional component failures as "reasonable and credible" alternative conditions?

*Answer*: "Reasonable and credible" alternative conditions may include either normal plant operating conditions, additional component failures, or potential accident conditions depending on the event. Each licensee is required to assess its operating experience. In order to determine the safety significance and implications of operating events, consideration will normally be given to the implications of the event under normal alternative operating conditions such as reactor power conditions expected to actually occur during the life of the plant. The intent of this section is to obtain the results of such routine reviews. [redacted]

# CALCULATION COVER SHEET

Calculation No: PSL-1FJN-95-001

Title: Estimate of the Offsite Dose Consequences from Leakage of LPSI Header Relief Valve, V3439

D R A F T

| | | | | | | | |
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| 0 | INITIAL ISSUE | | | | | | |
| No. | Description | By | Date | Chk/Ver | Date | Appr | Date |

REVISIONS

## LIST OF EFFECTIVE PAGES

CALCULATION NUMBER PEL-1FJN-95-001 REV. 0

| PAGE | SECTION | REV | PAGE | SECTION | REV | PAGE | SECTION | REV |
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| i | -- | 0 | | | | | | |
| ii | -- | 0 | | | | | | |
| iii | -- | 0 | | | | | | |
| 1 | -- | 0 | | | | | | |
| 2 | -- | 0 | | | | | | |
| 3 | -- | 0 | | | | | | |
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CALCULATION NUMBER PSL-1FJN-95-001 REV. 0

CALCULATION NO. PSL-1FJN-95-001 REV. 0 SHEET NO. 1

REF.

## I Purpose/Scope

The purpose of this calculation is to estimate the Thyroid Dose (CDE) at the Exclusion Area Boundary (EAB) from the valve leakage during a postulated accident.

The EAB serves as a reference distance for comparison to other accident analyses and or Part 100 limits.

The Thyroid Dose is of concern because the leaked fluid would be from the containment sump and, therfore, contain very little noble gas.

## II References

1. St. Lucie Unit 1 FSAR Amendment 14.
2. Reg-Guide 1.4, Rev 2, "Assumptions used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors"
3. EPA-400, October 1991, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents"
4. NUREG/CR - 1465 "Accident Source Terms for Light-Water Nuclear Power Plants"; Draft for Comment

## III Methodology

The Thyroid Dose at the EAB will be determined using FSAR styled calculations and assumptions. Then, more realistic factors and assumptions will be applied. This will establish the boundaries of the potential doses.

The process involves determining the amount of iodine released from the leakage to the Reactor Auxillary Building (RAB), then to the environment, estimating the 'downwind' concentrations and finally conversion to a dose. The remainder of the process will be applying, sequentially, realistic factors and assumptions, and demonstrating their effect on the doses.

CALCULATION NO. PSL-1FJM-95-001 REV. 0 SHEET NO. 2

| REV. | |
|---|---|
| | **IV Assumptions/Bases** |
| 1 | 1. Accident type : Small Break Loca (SBLOCA) or ESDE<br>This accident type is needed to maintain primary pressure sufficiently high to cause the relief valve to open upon initiation of shut-down cooling (SDC). |
| | 2. The SBLOCA or ESDE involves 1% cladding failure |
| | 3. Primary system make-up during the SBLOCA is from HPSI at a flow of 300 gpm; HPSI flow is constant until entering SDC, at which time HSPI is shut off. |
| | 4. Enter SDC at 12, or 18, or 24 hours after start of SBLOCA. Upon initiating SDC, HPSI is secured, plant switches to recirculation mode and LPSI pump starts. |
| | 5. Relief valve leak rate is 40 gpm; duration of 8 hours, then the valve reseats - leak stops. Volume of sump water leaked = 19200 gallons. |
| | 6. 4000 gallons of the leaked water is 'trapped' in the pipe tunnel and associated drains; the remainder (15200 gallons) leaks to the RAB floor. |
| | 7. Radioiodines do not decay or deposit once they've been released from the plant; this is a conservative simplification. |

CALCULATION NO. PSL-1FJN-95-001 REV. 0 SHEET NO. 3

| REF. | |
|---|---|
| | **V Calculation** |
| | A) Curies of $^{131}$I Deq available From Gap for release (1% Gap) (see Attachment 1): |
| | 9.4 E+4 @ 12 hr. 9.04 E+4 @ 18 hr 8.72 E+4 @ 24 hr. |
| 2 | RG 1.4 assumes that 50 % become airborne; therefore, 50% is in the sump water - the curies available to be leaked are : |
| | 4.7 E+4 @ 12 hr. 4.52 E+4 @ 18 hr 4.36 E+4 @ 24 hr. |
| | B) Volume of water in sump and primary system : |
| | Initial RCS volume (65000 gallons) + HSPI injected (300 gpm for 12,18,24 hr) |
| | Water volume (*thousands of gallons*) |
| | 281 @ 12 hr 389 @ 18 hr 432 @ 24 hr |
| | C) Concentration, Ci $^{131}$I Deq / gallon |
| | 0.167 @ 12 hr 0.116 @ 18 hr 0.088 @ 24 hr |
| | D) Curies leaked to RAB floor (conc, Ci/gal * 15200 gal) |
| | 2.54 E+3 Ci @ 12 hr 1.76 E+3 Ci @ 18 hr 1.33 E+3 Ci @ 24 hr |
| | F) Fraction of leaked Iodine that becomes airborne (curies leaked * partition factor); partition factor = 0.1 from the SGTR analysis. This becomes the curies released to the environment. |
| | 254 Ci @ 12 hr 176 Ci @ 18 hr 133 Ci @ 24 hr |
| | G) Downwind concentration (release rate, Ci/sec * X/Q, sec/m3). X/Q at the EAB = 8.55 E-5 sec/m$^3$ Assume release duration = 1 hour; exposure duration equals release duration, so the time will null in the next step. (remember that the time, in hours, is hours post trip when the release began) |
| | 6.03E-6 Ci/m$^3$ @ 12 hr 4.18E-6 Ci/m$^3$ @ 18 hr 3.16E-6 Ci/m$^3$ @ 24 hr |
| | H) Convert concentration to Dose (Ci/m$^3$ * 1.3E+6 Rem/hr per Ci/m$^3$) |
| | 7.84 Rem @ 12 hr 5.43 Rem @ 18 hr 4.11 Rem @ 24 hr |
| | At this point, following an FSAR styled analysis, the thyroid dose consequences range from 7.8 to 4.1 Rem for the event. The following page of calculations apply the more realistic assumptions and factors. |

CALCULATION NO. PSL-1FJM-95-001 REV. 0 SHEET NO. 4

REF.

V Calculation continued

I) Using the concept of RG 1.4, in that some (50%) of the Iodines released to the containment atmosphere plate-out or condense onto surfaces, and applying the same plate-out fraction to this scenario reduces the iodines released, and the resultant dose, to one half that calculated in step H. The range would then be 3.9 to 2.06 Rem for the event.

J) The FSAR discussion of ESF leakage indicates that for water temperatures less than 212 °F, a partition factor of 0.01 is more appropriate. The water, having passed through the shutdown heat exchanges would most likely be less than 212 °F. This leads to a ten-fold decrease in the iodines released, and the resultant dose, to one half that calculated in step H. The range would then be 0.78 to 0.41 Rem for the event.

4 K) NUREG-1465 discusses the sequestration of iodines by chemically binding with cesium. Although this would increase the activity in the sump water & water leaked to the RAB floor, the reduced mobility of the iodine leads to a lesser (1/10th) release than that calculated in step H. Holding the partition factor constant the effect of CsI formation would be:

95% iodines in water * 5% available for partitioning = 0.0475 net fraction of iodines available for partitioning. This is, about 1/10th of the 0.50 fraction of iodines available for partitioning assumed in RG 1.4.

| REF. | |
|---|---|
| | VI Results |

A) The doses calculated following the FSAR styled factors and assumptions range from 7.8 Rem, for a release occuring 12 hours after reactor trip, to 4.1 Rem for a release occuring 24 hours after reactor trip.

B) The doses calculated using the 'more realistic' assumptions and factors range from 0.04 Rem, for a release occuring 12 hours after reactor trip, to 0.02 Rem for a release occuring 24 hours after reactor trip.

C) The doses calculated following the FSAR styled factors and assumptions are a small fraction of Part 100 limits.

CALCULATION NO. PSL-1FJN-95-001 REV. 0 SHEET NO. 6

REF.

## ATTACHMENT 1

### Determination of Available Curies of $^{131}$I Deq

1,3

| Nuclide | T1/2, Hr | @ T=0 Gap Ci | EPA-400 DCFs | Weighting Factor (Wf)[1] |
|---|---|---|---|---|
| I-131 | 1.93 E+02 | 8.86 E+06 | 1.30 E+06 | 1.00 E+00 |
| I-132 | 2.26 E+00 | 1.82 E+06 | 7.70 E+03 | 5.92 E-03 |
| I-133 | 2.03 E+01 | 7.83 E+06 | 2.20 E+05 | 1.69 E-01 |
| I-134 | 8.67 E-01 | 2.10 E+06 | 1.30 E+03 | 1.00 E-03 |
| I-135 | 6.68 E+00 | 4.34 E+06 | 3.80 E+04 | 2.92 E-02 |

[1] Wf = DCF Iodine-xx + DCF Iodine-131

12 Hours post trip :

| | Gap Ci | Gap x Wf | |
|---|---|---|---|
| I-131 | [illegible].49 E+06 | 8.49 E+06 | |
| I-132 | 4.59 E+04 | 2.72 E+02 | |
| I-133 | 5.20 E+06 | 8.80 E+05 | DEQ I-131 |
| I-134 | 1.43 E+02 | 1.43 E-01 | From 1% Gap |
| I-135 | 1.25 E+06 | 3.65 E+04 | |
| | | 9.40 E+06 | *.01 = 9.40 E+04 |

18 Hours post trip :

| | Gap Ci | Gap x Wf | |
|---|---|---|---|
| I-131 | 8.31 E+06 | 8.31 E+06 | |
| I-132 | 7.29 E+03 | 4.32 E+01 | |
| I-133 | 4.23 E+06 | 7.17 E+05 | DEQ I-131 |
| I-134 | 1.18 E+00 | 1.18 E-03 | From 1% Gap |
| I-135 | 6.70 E+05 | 1.96 E+04 | |
| | | 9.04 E+06 | *.01 = 9.04 E+04 |

24 Hours post trip :

| | Gap CiGap x Wf | | |
|---|---|---|---|
| I-131 | 8.13 E+06 | 8.13 E+06 | |
| I-132 | 1.16 E+03 | 6.85 E+00 | |
| I-133 | 3.45 E+06 | 5.84 E+05 | DEQ I-131 |
| I-134 | 9.75 E-03 | 9.75 E-06 | From 1% Gap |
| I-135 | 3.60 E+05 | 1.05 E+04 | |
| | | 8.72 E+06 | *.01 = 8.72 E+04 |

## Proposed Violation B

10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected.

Contrary to the above, prompt corrective action was not taken in the case of St. Lucie Action Requests which reported anomalous relief valve behavior and which were initiated on February 20, March 2, and March 10, 1995. The failure to take prompt corrective action for these conditions led to a repetition of the anomalous behavior on August 10, 1995, when Unit 1 relief valve V-3439 lifted and failed to reseat without operator intervention. The subject event resulted in approximately 4000 gallons of reactor coolant accumulating in the Unit 1 pipe tunnel.

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

02-07-96

ENF Panel

CH: Gibson
Costo
Peebles
Lourdis
Schining
Benoit
*Beall
OE

*Sabrino
*Gray
*Mendiola
*LEWIS/OE
*Mark Miller
Crenshaw
*NORRIS

St. Lucie Overdilution Event

2nd vio 50.54(m)(2)(iii) -
general consensus that this
cite is not valid - and
should not be pursued -
general consensus! } Will

nominal power

pull togeth

JAG: Should in CL that
SRO was not particularly
aware of what was going
on } OK

---

Combine 3 vios for 1 SL3 problem
I.C.T.

---

Panel

Wrongdoing discussed for vios and no action.

VII.A.1.d = discretion

EEE/121

letter of severe disappointment

Re licensed operators

VIO for operator who went to kitchen?

OLB -

proposed letters w/o NOV

Strong letter to RO who left controls

Shift supervisor letter to SRO

Closed PEC

EA 96-040

-need not be transcribed

EWM, AFG, CAC
RCB, KDL, BAM

Post-panel brief: 1615 St. Lucie

1) 3 vios SL3 problem

a) failure to follow procedure

b) Crit 3 design control - procedure for dilution different than dilution in FSAR

c) 50.59 - changed dilution procedure even more different than ~~[illegible]~~ FSAR

**EICS STAFF NOTES**

**FACILITY:** St. Lucie

[✓] ENF PANEL
[ ] ENF CONF
[ ] CAUCUS
[ ] Other

**DATE:** 2/7/96
**CASE NO:**
**IR NO:**
**DUE TO OE:**
**EA NO:** 95-[illegible]

**LAST DATE OF INSPECTION:** [illegible]
**INSPECTOR:** [illegible]
**START:** **TERM:**
**RESPONSIBLE DIVISION:** DRS
**RESPONSIBLE SECTION:** [illegible]

**ATTENDEES**

| | |
|---|---|
| T. Mendiola | J. Gray |
| T. Morris | M. Satorius |
| S. Lewis | J. Beall |
| M. Miller | B. Croteau |
| S. Sandin | J. Jaudon |
| B. Salyers | A. Gibson |
| K. Landis | D. Desai |
| E. Costo | C. Evans |
| C. Kapp | B. Uryc |

**SUBJECT:** Dilution Event

**NOTES:**

Generally add 25-40 gal/day - as a routine (once/hour)
Operator at the controls requested another RO to relieve him - failed to turnover dilution
SRO responsible for turnover [illegible]
Other SRO (asst) - [illegible] controlled area - would normally [illegible] ROs
Alarm - sounded
RO returned & immediately recognized error
Rx power went to 101.18%
Promptly returned to 100% [illegible]

3 Violations Against Part 50 Licensee
(1) Procedures
(2) Power Limit } Breakdown in Controls
(3) 50.59 } SL III Problem

RO - NOV
SS - Letter

**OE NOTIFIED:** **SL:** **SUP:** **CP:**

**EC SCHEDULED:** **LOCATION:**

DATE: ________ PAGE: 2

Licensee

(A) 4 Procedure Requirements Violation

Keep consensus

1 - Failed to stop diluted when required
intended to add ≈30 actually add ≈400
2 - Inadequate watch turnover RO-Desk RO
3 - Lack of Verbatim Compliance
(Straight $H_2O$ instead of blended water as req'd)
+ Friend notification ?

Licensee 10CFR 50.4(m) — no violation
of presences

delete consensus

SSO did not meet NRC expectations; needs a general level of cognizance
OE would support Conduct of Operations violations
Human Factors Issue — Pueblo/Mendula
CANPS Can't See What's Happening

Concern that SRO was not aware of activities - hope they would do in future

(B) 1980 Jordan Memo - acknowledges >100% operation + provides guidance

delete consensus

Do not cite power separately - precedent
(*) Fold into Contrary to Statement in VIO A
- establishes severity level

2 VIOs into problem merged into 1 violation

(C) (?) 50.59 Violation
Procedure - Manual Dilution w/ option to VCT or direct to Charging Pumps
Procedure has never ~~???~~ been consistent w/ FSAR
(?) Criterion III - Failure to translate requirements (FSAR) into procedures

BRUNO URYC, Jr.
Enforcement and Investigation
Coordination Staff

Deviation ?

DATE: ____________ 

one problem

Failure to follow procedures
Inadequate procedural controls
(1) 50.59
(2) Criterion III

(1)

possibly
~~relieved~~ for duty
to be considered
disciplined

Status of Operators — (a) RO who left in training
no licensed activities
Less than attentive too long
(b) Desk RO - back on shift
(c) NPS - " " & also
returning from leave

No NRC action as I see it
No wrongdoing on behalf of operator

Operators — only the operator who left

NOV for Operator @ controls — Proposal
Against condition in license - not license
T/S Compliance

Letter to U.S.S. -

generally cite for careless disregard & deliberate
- consider actions by licensee to discipline

Letters (not proper attention to duties
not consistent with performance standards

to Stringent - RO
SRO

Individuals come in with Part 50
all 3

SEC

ID - (No) proposed
missed opportunities
NRC ID'd Crit III/50.59

CA - (Yes) proposed
In-house event report
Removed RO
Night Order
Training for Ops.
Rev 1 Crit 1 procedure
etc...

need added info

Discretion - Yes
(1) Prior Operator problems
(2) Left out unit 1
when revised
unit 2

BRUNO URYC, Jr.
Enforcement and Investigation
Coordination Staff

DATE: ______

PAGE: ______

Closed - No transcription

EA 96-040 Licensee

Final discussion

Licensee

3 VIOS (procedures) ⇒ SL III w/-

50.59

Part III

No ID

CA - Yes - [illegible]

Maybe discretion

Exc

[illegible] to [illegible], [illegible] + [illegible]

[illegible]

[illegible] ID + [illegible]

[illegible] 50.54(f) [illegible] + [illegible] message in [illegible]

[illegible]

BRUNO URYC, Jr.
Enforcement and Investigation
Coordination Staff

**Notice of Violation**

10 CFR 50.54(m)(2)(iii) requires, in part, that, for a nuclear power unit in an operational mode other than cold shutdown or refueling, a person holding a senior reactor operator license shall be present in the control room at all times.

10 CFR 50.54(m)(2)(iii) requires, in part, that, for a nuclear power unit in an operational mode other than cold shutdown or refueling, a licensed operator or senior operator shall be present at the controls at all times.

Contrary to the above:

1. On January 22, 1996, a boron dilution event occurred at St. Lucie Unit 1 which demonstrated that the senior reactor operator in the control room was not aware of the dilution in progress and, therefore, was incapable of providing the oversight function required by the subject rule.

2. On January 22, 1996, a boron dilution event occurred at St. Lucie Unit 1 which demonstrated that the licensed operator at the controls was not aware of the dilution in progress and, therefore, could not and did not continuously monitor plant instrumentation or properly manipulate controls associated with the dilution to prevent the reactor from exceeding 100% power, thus failing to satisfy the requirements of the subject rule.

This is a Severity Level III Violation (Supplement I)

T/S Procedural Adherence
Procedure - Operations

**Background**

On July 11, 1983, the NRC published in Federal Register (48 FR 31611) the final rule 10 CFR 50.54(m)(2)(iii) which amended NRC regulations to require licensees of nuclear power units to provide a minimum number of licensed operators and senior operators on shift at all times to respond to normal and emergency conditions.

10 CFR 50.54(m)(2)(iii) states "When a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times.. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times."

In the Statements of Consideration:

The requirement for a senior operator's continuous presence in the control room would assure that:

(1) A person is available who can provide the oversight function of the supervisor so that the probability of correctly detecting abnormal events early enough to mitigate potential adverse consequences might be increased;
(2) the senior operator in the control room is aware of plant conditions prior to and resulting from an abnormal event so that the senior operator will be able to use extra experience, training and knowledge to act promptly to mitigate that event; and
(3) the reactor operator is able to direct attention to performing the immediate actions necessary to mitigate that event rather than having to brief the senior operator about the background of that event if that person were absent from the control room.

The presence of a senior operator...will also increase the probability of correctly detecting abnormal events and human error early enough to mitigate potential consequences of any accident.

For the Licensed Operator:

The requirement that an operator be at the controls...will assure that plant instrumentation is continuously monitored and that controls are properly manipulated.

# ENFORCEMENT ACTION WORKSHEET

[ST LUCIE OVERDILUTION EVENT]

PREPARED BY: R. Schin DATE: February 5, 1996

**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.**

Signature

**Facility: St. Lucie**
**Unit(s): 1**
**Docket Nos: 50-335**
**License Nos: DPR-67**
**Inspection Report No: 50-335,389/96-01**
**Inspection Dates: January 26-30, 1996**
**Lead Inspector: R. Schin**

1. **Brief Summary of Inspection Findings:**

   Concern with operator attentiveness related to a reactivity addition event, and related operator violations of procedures:

   a. Operators failed to stop dilution when the proper amount had been added.

   b. There was inadequate watch turnover for the operator at the controls during dilution.

   c. Operators failed to follow the Conduct of Operations procedure in performing the dilution procedure (lack of strict/verbatim compliance).

   d. Operators failed to adequately report the event to licensee management.

   Also, operators exceeded the steady state licensed power limit of 2700 megawatts thermal (100% power).

   In addition, the licensee nade a change to the procedures as described in the SAR without a 10 CFR 50.59 safety evaluation.

   See the attached draft NOV, General Description of Event, Detailed Sequence of Events, Summary of Draft Preliminary Inspection Findings, Control Room Diagram, CVCS Charging System Diagram, Procedures, and FSAR.

2. **Analysis of Root Cause:**

   Operator inattentiveness to reactivity addition.

3. **Basis for Severity Level (Safety Significance):**

   I.C.3 Inattentiveness to duty on the part of licensee personnel, while adding reactivity to the reactor, and

   I.C.7 A breakdown in the control of licensed activities involving a number of violations that are related that collectively represent a significant lack of attention or carelessness toward licensed responsibilities.

4. **Identify Previous Escalated Action Within 2 Years or 2 Inspections?**

   EA 95-180 (EEI 95-16-01); LTOP inoperability due to PORV failure
   Event date 8/9/95

5. **Identification Credit?** No

   Identified through an event. The licensee initiated an In-House Event Report and gave a copy to the NRC resident inspector promptly after the event. The event occurred at approximately 0220 on January 22, 1996.

   Missed opportunities:

   a. In response to SOER 94-02, dated September 1994, which described a similar Turkey Point overdilution event and several inadvertent dilution events at other utilities, the licensee reviewed the St. Lucie operating procedures related to dilution and concluded that no changes were needed. This was a missed opportunity to strengthen operating procedures to prevent the 1/22/96 overdilution event.

   b. The Unit 2 dilution procedure had been changed in December 1995, but not the Unit 1 procedure, to more accurately describe dilution the way the operators had performed it for years (in manual and direct to the charging pumps). During the event, manual dilution could not be accomplished by using the Unit 1 procedure in compliance with the Conduct of Operations (strict/verbatim compliance).

6. **Corrective Action Credit?** Yes

   The licensee initiated an In-House Event Report summarizing the event and began distribution of that report within about four hours after the event. The licensee also immediately removed the reactor operator who had initiated the event from licensed duties, promptly issued a Night Order and conducted training on the event with operators on each shift; revised the Unit 1 procedure for dilution so that manual dilution could be performed by strict compliance to the procedure steps; revised the Conduct of Operations procedure to require the RO to get prior approval

from the SRO for dilution/boration, the SRO to directly supervise dilution/boration, no RO or SRO turnover during dilution/boration, and RTGB walkdown prior to RO or SRO short term relief; and initiated further review of the event.

Weaknesses in the licensee's corrective actions included:

a. Potential VIO of 10 CFR 50.59: The revised procedure (after the event) did not support the FSAR Chapter 15 accident analysis assumptions on how dilution was performed. The revised procedure described dilution in manual (with no automatic shutoff) and directly to the suction of the charging pumps. The FSAR assumed dilution in automatic (with an automatic shutoff) and to the VCT (where the demineralized water would mix with boric acid solution before going to the suction of the charging pumps and result in a lower rate of reactivity addition). The licensee had not performed a safety analysis of this difference and had not revised the procedure and/or FSAR to make them agree.

N/A because non VIO

b. The revised procedure for manual dilution (after the event) did not require the operator at the controls to remain by the dilution controls and to closely monitor the dilution during a manual dilution with no automatic shutoff.

minor

c. The licensee initial investigation of the event was not thorough in that it concluded that maximum reactor power was 100.2%. Subsequent review by the NRC and licensee found that maximum reactor power was approximately 101.18%.

7. **Candidate For Discretion? [See attached list]** Yes - potential escalation.

During the last year, the licensee's performance in Operations has declined from SALP 1 to SALP 2 (predecisional). There have been several operator violations of procedures that are, in part, related to the current violation:

1) VIO 335/94-22-02, "Improper Modification of Control Room Logs", November 25, 1994

2) NCV 335/95-07-01, "Failure to Follow Shutdown Cooling Operating Procedures", April 19, 1995

3) VIO 335/95-15-01, "Failure to Follow Procedures and Block MSIS Actuation", October 16, 1995

4) VIO 335/95-15-02, "Failure to Follow Procedures during RCP Seal restaging", October 16, 1995

5) VIO 335/95-15-03, "Failure to Follow Procedure and Document abnormal valve position in the Valve Switch Deviation Log", October 16, 1995

6) VIO 335/95-15-04, "Failure to Follow Procedures during Alignment of Shutdown Cooling System", October 16, 1995

7) VIO 389/95-18-01, "Failure to Follow Procedures and Maintain Current and Valid Log Entries in the Rack Key Log and Valve Switch Deviation Log", November 27, 1995

8) VIO 389/95-21-02, "Failure to Follow the Equipment Clearance Order Procedure and Require Independent Verification of a TS Related Component", December 8, 1995

All of the above VIO/NCVs involved licensed operators with a licensee corrective action commitment to strict adherence to procedures.

8. **Is A Predecisional Enforcement Conference Necessary?**

Yes

Why: There is substantial interest in this event and in the NRC message to the licensee and to the industry. The message for this enforcement action should be that operators must treat Dilution/Boration as seriously as control rod manipulations. Also, that unusual operations events must be transmitted promptly to management.

If yes, should OE or OGC attend? Yes
Should conference be closed? No

9. **Non-Routine Issues/Additional Information:**

10. **This Action is Consistent With the Following Action (or Enforcement Guidance) Previously Issued:** I.C.3

Basis for Inconsistency With Previously Issued Actions (Guidance)

11. **Regulatory Message:** The message for this enforcement action should be that operators must treat Dilution/Boration as seriously as control rod manipulations. Also, that unusual operations events must be transmitted promptly to management.

12. **Recommended Enforcement Action:** SLIII with CP

13. **This Case Meets the Criteria for a Delegated Case.** No

14. **Should This Action Be Sent to OE For Full Review?** No, informal review.

15. **Regional Counsel Review** To be determined at a later date.

No Legal Objection Dated:

16. **Exempt from Timeliness: No**
**Basis for Exemption:**

Enforcement Coordinator:
DATE:

## ENFORCEMENT ACTION WORKSHEET - ISSUES TO CONSIDER FOR DISCRETION

☐ Problems categorized at Severity Level I or II.

☐ Case involves overexposure or release of radiological material in excess of NRC requirements.

☐ Case involves particularly poor licensee performance.

☐ Case (may) involve willfulness. Information should be included to address whether or not the region has had discussions with OI regarding the case, whether or not the matter has been formally referred to OI, and whether or not OI intends to initiate an investigation. A description, as applicable, of the facts and circumstances that address the aspects of negligence, careless disregard, willfulness, and/or management involvement should also be included.

☒ Current violation is directly repetitive of an earlier violation (in part).

☐ Excessive duration of a problem resulted in a substantial increase in risk.

☐ Licensee made a conscious decision to be in noncompliance in order to obtain an economic benefit.

☐ Cases involves the loss of a source. (Note whether the licensee self-identified and reported the loss to the NRC.)

☐ Licensee's sustained performance has been particularly good.

☐ Discretion should be exercised by escalating or mitigating to ensure that the proposed civil penalty reflects the NRC's concern regarding the violation at issue and that it conveys the appropriate message to the licensee. Explain.

Enclosure 3

**REFERENCE DOCUMENT CHECKLIST**

[ ] NRC Inspection Report or other documentation of the case:

NRC Inspection Report Nos.:

[ ] Licensee reports:

[ ] Applicable Tech Specs along with bases:

[x] Applicable license conditions

[x] Applicable licensee procedures or extracts

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

[x] Extracts of pertinent FSAR or Updated FSAR sections for citations involving 10 CFR 50.59 or systems operability

[ ] Referenced ORDERS or Confirmation of Action Letters

[ ] Current SALP report summary and applicable report sections

[ ] Other miscellaneous documents (List):

## PROPOSED VIOLATION

A. Technical Specification (TS) 6.8.1.a required that written procedures be established, implemented, and maintained covering the activities recommended in Appendix A of Regulatory Guide 1.33, Rev 2, February 1978. Appendix A includes operating procedures for the chemical and volume control system and administrative procedures for relief turnover, procedural adherence, and authorities and responsibilities for safe operation.

Operating Procedure No. 1-0250020, Boron Concentration Control - Normal Control, Rev. 35, step 8.5.14 required that operators monitor the water flow totalizer and close valve V2525 after the desired volume was added during a boron concentration dilution using the direct path to the charging pump suction.

Administrative Procedure No. 0010120, Conduct of Operations, Rev 79, Appendix D, Crew Relief/Shift Turnover, required that, for short term watchstander relief, a turnover be conducted including: general watchstation status, off-normal conditions, and tests in progress.

Administrative Procedure No. 0010120, Appendix M, Procedural Compliance and Implementation, required that controlled procedures be implemented and complied with in accordance with the instructions provided in QI 5-PR/PSL-1. Procedure QI 5-PR/PSL-1, Preparation, Revision, Review/Approval of Procedures, Rev 67, Section 5.13.2, stated that all procedures shall be strictly adhered to and identified that Operating Procedure 1-0250020 was not considered "skill of the trade" and was not to be performed from memory without referring to the procedure.

Administrative Procedure No. 0010120, Appendix E, Notification of Operations Supervisor/FPL Management, required prompt verbal notification of the Operations Supervisor for unplanned reactivity changes.

Contrary to the above:

1. On January 22, 1996, at approximately 2:30 a.m., Unit 1 operators failed to close valve V2525 after the desired volume was added during a boron concentration dilution using the direct path to the charging pump. Operators had desired to add between 25 and 40 gallons of primary makeup water, but failed to stop the dilution until approximately 400 gallons were added. During this time, the temporary relief operator at the controls was unaware that a boron concentration dilution was in progress, which resulted in an unmonitored reactivity addition. The SRO and other operators in the control room were also unaware that a reactivity addition was in progress.

2. On January 22, 1996, at approximately 2:30 a.m., the Unit 1 operator at the controls conducted a short term watchstander relief with an inadequate turnover in that it failed to include general watchstation status and conditions including that a boron

concentration dilution was in progress. As a result, the relief operator at the controls was unaware that a boron concentration dilution was in progress and failed to adequately monitor and control the dilution.

3. On January 22, 1996, at approximately 2:30 a.m., operators performed Operating Procedure 1-0250020 from memory, without referring to the procedure, and without strictly adhering to the procedure. At the time, the procedure was written such that the boron concentration dilution that was performed could not have been performed by strictly adhering to the procedure.

4. On January 22, 1996, between 2:30 a.m. and 7:20 a.m., operators failed to give prompt verbal notification to the Operations Supervisor for unplanned reactivity changes that had occurred.

B. The Facility Operating License for St. Lucie Unit 1 authorizes the licensee to operate the facility at a steady state reactor core power level not in excess on 2700 megawatts thermal (MWt). TS 1.25 defines rated thermal power to be a total reactor core heat transfer rate to the reactor coolant of 2700 MWt. TS 1.33 defines thermal power to be the total reactor heat transfer rate to the reactor coolant.

Contrary to the above, on January 22, 1996, between approximately 2:20 and 3:30 a.m., the reactor core thermal power level limit of 2700 MWt (100%) was exceeded, due to operator inattentiveness. 100% reactor power was exceeded for approximately 70 minutes. Also, 101% reactor power was exceeded for approximately 4 minutes and a peak reactor power of approximately 101.18% was reached.

C. 10 CFR 50.59 allows the licensee to make changes to the procedures as described in the Safety Analysis Report (SAR), without prior Commission approval, unless the change involves, in part, an unreviewed safety question. A proposed change shall be deemed to involve an unreviewed safety question if, in part, the probability of occurrence of an accident important to safety previously evaluated in the SAR may be increased. The licensee shall maintain records of changes in procedures made pursuant to this section, to the extent that they constitute changes in procedures as described in the SAR. These records must include a written safety evaluation which provides a basis for the determination that the change does not involve an unreviewed safety question.

Contrary to the above, on January 23, 1996, the licensee made a change in Unit 1 procedures as described in the SAR and the records for that change did not include a written safety evaluation. Temporary Change 1-96-017 to procedure 1-0250020, Boron Concentration Control - Normal Operation, Rev. 35, added instructions for dilution in manual and directly to the suction of the charging pumps. However, the SAR, paragraph 15.2.4.1, states that boron dilution is conducted under strict administrative procedures which limit the rate and magnitude of any required change in boron concentration. Further, the SAR states that boron dilution must be conducted in automatic (such that when the

specific amount has been injected, the demineralized water control valve is shut automatically) and describes introduction into the volume control tank (VCT). The SAR concludes that, in part, because of the procedures involved, the probability of a sustained or erroneous dilution is very low. The licensee implemented Temporary Change 1-96-017 on January 23, 1996, without a written safety evaluation.

## General Description of the Event

At approximately 0225 on January 22, 1996, the Unit 1 control board Reactor Controls Operator (RCO) began a manual dilution to the RCS by aligning primary makeup water (demineralized water) directly to the suction of the 1B Charging Pump. Moments after beginning the dilution, the board RCO responded to a secondary plant annunciator and then saw the desk RCO return from the kitchen. He requested that the desk RCO relieve him so that he could prepare his lunch. During the turnover, there was no discussion of the dilution in progress. Following the turnover, the relief operator at the controls and the Nuclear Plant Supervisor (NPS), who was at the desk RCO station, were not aware that a dilution was in progress. The original board RCO returned between 5-10 minutes later and immediately recognized his error. He informed the other RCO of the overdilution, which was overheard by the NPS, and stopped the dilution. The NPS directed the ANPS take charge and begin a manual boration. Unit 1 entered 2-hour TS LCO Action Statement 3.2.5 for $T_c$ greater than 549°F. The maximum $T_c$ obtained was 549.9°F and the maximum reactor power was 101.18%. $T_c$ was above the TS limit of 549°F for approximately 50 minutes and reactor power was above 100% for approximately 70 minutes. The TS LCO Action Statement for $T_c$ was not exceeded and the guidance of the Jordan memorandum on maximum reactor power was not exceeded. The operators did not verbally notify plant management or the NRC of this event.

## Detailed Sequence of Events

(Note that the times for the sequence of events are approximate and only relevant events are mentioned)

1/21/96

11:00 p.m. Incoming mid shift assumed Unit 1 responsibility with the Unit at 100% power, 870 MWe, Tavg at 575 degrees F, Thot at 600 degrees F, Tcold at 548.9 degrees F, RCS Boron concentration at 376 ppm, Xe worth at -2722 pcm, all CEAs fully withdrawn and manual, and no Technical Specification action statements in effect. Major evolution planned for the shift was to place the waste gas system in service. Further, there was an annunciator alarm E-9 associated with circulating water pump lube water supply strainer delta P high that was intermittently coming in due to a failed pressure switch.

11:45 p.m. Board RCO reset to zero the primary water (to VCT or charging pump) flow totalizer in preparation for inventory balance (RCS

leak rate calculation)

11:00 p.m.-
2:00 a.m The board RCO recalled performing at least two dilutions of approximately 35 gallons each between 11:00 p.m. and 2:20 a.m. without resetting the totalizer.

1/22/96

xx:xx a.m NPS arrived in Unit 1 control room to gather data for morning report meeting and sat near desk behind control boards. STA was also present near NPS

xx:xx a.m. ANPS turned over control room senior reactor operator responsibility to NPS and proceeded to the kitchen to prepare breakfast

xx:xx a.m. Desk RCO left control room to go to the kitchen

2:20 a.m. Normal continued fuel burnup resulted in indicated Tc of 548.7 degrees F on RTGB-104 (digital meter). At this point the board RCO decided to restore Tc to maximum allowable program value of 549.0 degrees F.

xx:xx a.m. Desk RCO arrived in the control room with his meal

2:25 a.m. The board operator began a manual dilution by aligning primary water to the suction of the charging pumps by opening FCV-2210X and AOV-2525. The flow rate was approximately 44 gpm.

2:26 a.m. Annunciator E-9 associated with circulating water lube water supply strainer high delta P was received. The board RCO walked to the panel and acknowledged the annunciator.

2:27 a.m. After acknowledging the annunciator, the board operator decided to proceed to the kitchen to prepare his meal. The board operator conveyed this to the desk operator and requested that he take over the board operator responsibilities. However, he did not mention the ongoing dilution. The desk operator got up and proceed to the board in the vicinity of panel 103. The original board operator proceeded to the kitchen and started preparing his meal on a skillet that had been kept warm. At this time the NPS and the STA were in the control room at the desk area. The NWE had been in and out of the control room throughout the shift. The relief operator at the controls, NPS, STA, and NWE were not aware of the ongoing dilution.

2:35 a.m. The original board operator returned from the kitchen with his meal. Upon approaching the board, he realized that he had left the control room with an ongoing manual dilution. He exclaimed that he had overdiluted and immediately began securing the dilution. The desk operator questioned how much water was added and the board operator noted from the totalizer that approximately

400 gallons was added.

2:35 a.m. Soon after, annunciator M-16 associated with RCP controlled bleedoff pressure high was received. At this point the Tc was noted by the desk operator to be 549.6 degrees F. Entry into two hour action statement associated with Technical Specification 3.2.5, DNB paramenters was recognized and later logged.

2:36 a.m. The desk operator directed the board operator to initiate boration to restore Tc to program. The NWE calculated the amount of borated water to be added to the RCS. The NPS asked the desk operator to notify the unit ANPS to come to the control room.

x:xx a.m. ANPS walked into the control room.

2:41 a.m. Tc reached the highest noted value of 549.9 degrees F. MWe reached 875 and indicated reactor power was approximately 101.2%

x:xx a.m. Operator secured boration.

3:14 a.m. Tc noted below 549.0 degrees F. Technical Specification action statement was exited.

x:xx a.m. STA initiated an In-House Event Report and notified HPES personnel by telephone.

5:45 a.m.-
6:00 a.m. Shift turnover occurred. It appears that the dilution event was not discussed with the oncoming shift.

6:25 a.m. In-House Event Report was E-mailed to standard distribution, which included plant management, by the STA.

6:30 a.m. The Operations Manager toured the control room but was not informed of the over dilution event.

7:20 a.m. The Operations Manager read the control room logs (in his office by computer) and questioned the log entry associated with the overdilution event.

7:30 a.m. Licensee initiated a detailed investigation associated with the event.

7:45 a.m. Senior Plant management was notified of the event during the morning meeting.

10:00 a.m. NRC resident inspector was given the event report that was initiated associated with the event.

ST. LUCIE ONSITE EVENT FOLLOWUP INSPECTION
OVERDILUTION EVENT of 1/22/96
(Exit was at 10:00 a.m. on 1/30/96)

Inspectors: R. Schin, S. Sandin, B. Desai

Summary of draft preliminary findings:

1. Magnitude of power and temperature excursion

   a. Reactor power
      - Peak reactor power was approximately 101.18%
      - 100% power was exceeded for approximately 70 minutes
      - 101% power was exceeded for approximately 4 minutes
      - The event was within the accident analysis
      - The guidelines of the Jordan memo were not exceeded

   b. Cold leg temperature
      - Peak Tc was approximately 549.9 degrees F
      - TS limit of 549 was exceeded for approximately 50 minutes
      - TS 2-hr. action statement was properly entered and was not exceeded

2. Concern with operator attentiveness - Potential/Apparent VIO of procedures (Enforcement panel form completed on this issue):

   a. Operators failed to stop dilution when the proper amount had been added.

   b. There was inadequate watch turnover for the operator at the controls during dilution.

   c. Operators failed to follow the Conduct of Operations procedure in performing the dilution procedure.

   d. Operators failed to adequately report the event to licensee management.

3. Concern with control room command and control - Weakness

   a. The SRO in the control room was not aware of the dilution in progress.

   b. The board operator did not inform the SRO of dilution - this was a general practice at the site and not required by procedures.

   c. The watchstander board was not maintained (on Saturday).

   d. The SRO in the control room was allowed to be in the ANPS office for unlimited time, out of sight of control room activities and out of hearing range of almost all control room activities except annunciator alarms (not applicable during this event).

4. Weaknesses in procedures

   a. The Unit 2 dilution procedure had been changed, but not the Unit 1 procedure, to more accurately describe dilution the way the operators had performed it for years (in manual and direct to the charging pumps). During the event, manual dilution could not be accomplished by using the Unit 1 procedure in compliance with the Conduct of Operations.

   b. Procedures and practices for dilution (before and during the event) did not support the FSAR accident analysis assumptions on how dilution was performed. The FSAR assumed dilution in automatic and to the VCT.

   c. Procedures for dilution (before and during the event) did not require the operator at the controls to remain by the dilution controls and to closely monitor the dilution during a manual dilution with no automatic shutoff.

5. Weaknesses in corrective action

   a. Potential VIO of 10 CFR 50.59: Revised procedure (after the event) did not support the FSAR Chapter 15 accident analysis assumptions on how dilution was performed. The FSAR assumed dilution in automatic and to the VCT.

   b. The revised procedure for manual dilution (after the event) did not require the operator at the controls to remain by the dilution controls and to closely monitor the dilution during a manual dilution with no automatic shutoff.

   c. The licensee initial investigation of the event was not thorough in that it concluded that maximum reactor power was 100.2%. Subsequent review by the NRC and licensee found that maximum reactor power was approximately 101.18%.

6. Weakness in Operational Experience Feedback

   a. In response to SOER 94-02, dated September 1994, which described a similar Turkey Point overdilution event and several inadvertent dilution events at other utilities, the licensee reviewed the St. Lucie operating procedures related to dilution and concluded that no changes were needed. This was a missed opportunity to strengthen operating procedures to prevent the 1/22/96 overdilution event.

7. Other comments

   a. There was no clearly noticeable indication of dilution in progress. The dilution clicker was quiet (might not be heard from the desk area) and sounded identical to the nearby clickers that routinely made noise.

b. Operators routinely did not log reactivity additions; however, the licensee's Conduct of Operations procedure stated that operators should log reactivity changes.

LICENSEE DISSENTING COMMENTS

1. The licensee had dissenting comments on item 5.a. above, the potential violation of 10 CFR 50.59. The inspectors told the licensee at the exit that those dissenting comments would be included in the inspection report, for further review by NRC management. The dissenting comments, from the engineering manager (Dan Denver) and the licensing manager (Ed Weinkam), included:

   a. The previous procedure allowed diluting in manual and directly to the suction of the charging pumps, and that had been the practice for many years. Therefore, the temporary change on 1/23/96 (after the event) did not change the method of dilution, but only clarified a previously existing procedure and made it conform to "verbatim compliance" rules. The inspectors did not disagree. In fact, further review, as requested by the inspectors, found that the first time the dilution procedure was changed to allow opening of valve 2525 (directly to the suction of the charging pumps) was in a change to rev. 2 of the procedure, in 1976, before the operating license was issued.

   b. The design of the plant (piping, valves) always was such that dilution in manual and directly to the suction of the charging pumps was possible. The inspectors did not disagree.

   c. The accident analysis assumed a worst case dilution event with demineralized water going directly to the suction of the charging pumps and three charging pumps running. That would be three times the flowrate of this event and therefore that analysis bounds this event. The inspectors did not disagree.

   d. The FSAR Chapter 9 description of the Chemical and Volume Control System did not prohibit dilution in manual and directly to the suction of the charging pumps. The inspectors did not disagree.

   e. The automatic mode of dilution is less safe than the manual mode, in that there is more opportunity for a malfunction that could result in a maximum flowrate approaching the design limit. The inspectors did not comment on that position.

   f. The procedure change that first allowed dilution directly to the suction of the charging pumps was made before the operating license was issued, therefore 10 CFR 50.59 did not apply to that change. The inspectors did not comment on that position.

   g. Since the operating procedure that was in effect at the time the operating license was issued allowed dilution in manual and directly to the suction of the charging pumps, that method was

included in the original licensing basis of the plant. The inspectors did not agree with that position.

h. After receiving these licensee comments, the inspectors' concern remained unchanged: The Temporary Change of 1/23/96 (after the event) described procedure steps for dilution in manual and directly to the suction of the charging pumps. That procedure was different from the one described in the FSAR. The licensee's procedure differed from the FSAR in that it allowed a faster rate of reactivity addition and without an automatic shutoff. The licensee had not performed a safety analysis of this difference and had not revised the procedure and/or FSAR to make them agree.

2. The licensee also had a dissenting comment on item 5.c. above, the weakness in the licensee's initial investigation. The dissenting comment, from the Plant Manager (Jim Scarola), was:

   a. The initial investigation, for the In-House Event Summary, was done by the STA. Timeliness was more important than quality at that time. Subsequent more thorough review would be performed by the licensee. The inspectors acknowledged the licensee's comment.

Proposed Operator
NOTICE OF VIOLATION

Docket No. 55-
License No. OP-
EA(s) TBD

During an NRC inspection conducted on January 26-30, 1996, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violations are listed below:

Technical Specification 6.8.1.a required that written procedures be established, implemented, and maintained covering the activities recommended in Appendix A of Regulatory Guide 1.33, Rev 2, February 1978. Appendix A includes operating procedures for the chemical and volume control system and administrative procedures for relief turnover, procedural adherence, and authorities and responsibilities for safe operation.

Operating Procedure No. 1-0250020, Boron Concentration Control - Normal Control, Rev. 35, step 8.5.14 required that operators monitor the water flow totalizer and close valve V2525 after the desired volume was added during a boron concentration dilution using the direct path to the charging pump suction.

Administrative Procedure No. 0010120, Conduct of Operations, Rev 79, Appendix D, Crew Relief/Shift Turnover, required that, for short term watchstander relief, a turnover be conducted including: general watchstation status, off-normal conditions, and tests in progress.

Administrative Procedure No. 0010120, Appendix M, Procedural Compliance and Implementation, required that controlled procedures be implemented and complied with in accordance with the instructions provided in QI 5-PR/PSL-1, Preparation, Revision, Review/Approval of Procedures, Rev 67. Procedure QI 5-PR/PSL-1 Section 5.13.2, stated that all procedures shall be strictly adhered to and specifically identified that Operating Procedure 1-0250020 was not considered "skill of the trade" and was not to be performed from memory without referring to the procedure.

Contrary to the above:

1. On January 22, 1996, at approximately 2:30 a.m., the Unit 1 operator failed to close valve V2525 after the desired volume was added during a boron concentration dilution using the direct path to the charging pump. The operator had desired to add between 25 and 40 gallons of primary makeup water, but failed to stop the dilution until approximately 400 gallons were added. During this time, the temporary relief operator at the controls was unaware that a boron concentration dilution was in progress, which resulted in an unmonitored reactivity addition. The SRO and other operators in the control room were also unaware that a reactivity addition was in progress.

2. On January 22, 1996, at approximately 2:30 a.m., the Unit 1 operator at the controls conducted a short term watchstander relief with an inadequate turnover in that he failed to include general watchstation status and conditions including that a boron concentration dilution was in progress. As a result, the relief operator at the controls was unaware that a boron concentration dilution was in progress and failed to adequately monitor and control the dilution.

3. On January 22, 1996, at approximately 2:30 a.m., the Unit 1 operator performed Operating Procedure 1-0250020 from memory, without referring to the procedure, and without strictly adhering to the procedure. At the time, the procedure was written such that the boron dilution that was performed could not have been performed by strictly adhering to the procedure.

These violations represent a Severity Level III ~~problem~~ violation (Supplement___).

Pursuant to the provisions of 10 CFR 2.201, ************* is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region II, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. However, if you find it necessary to include such information, you should clearly indicate the specific information that you desire not to be placed in the PDR, and provide the legal basis to support your request for withholding the information from the public.

Dated at Atlanta, Georgia
this ____ day of Februay 1996

ANPS only required by ...

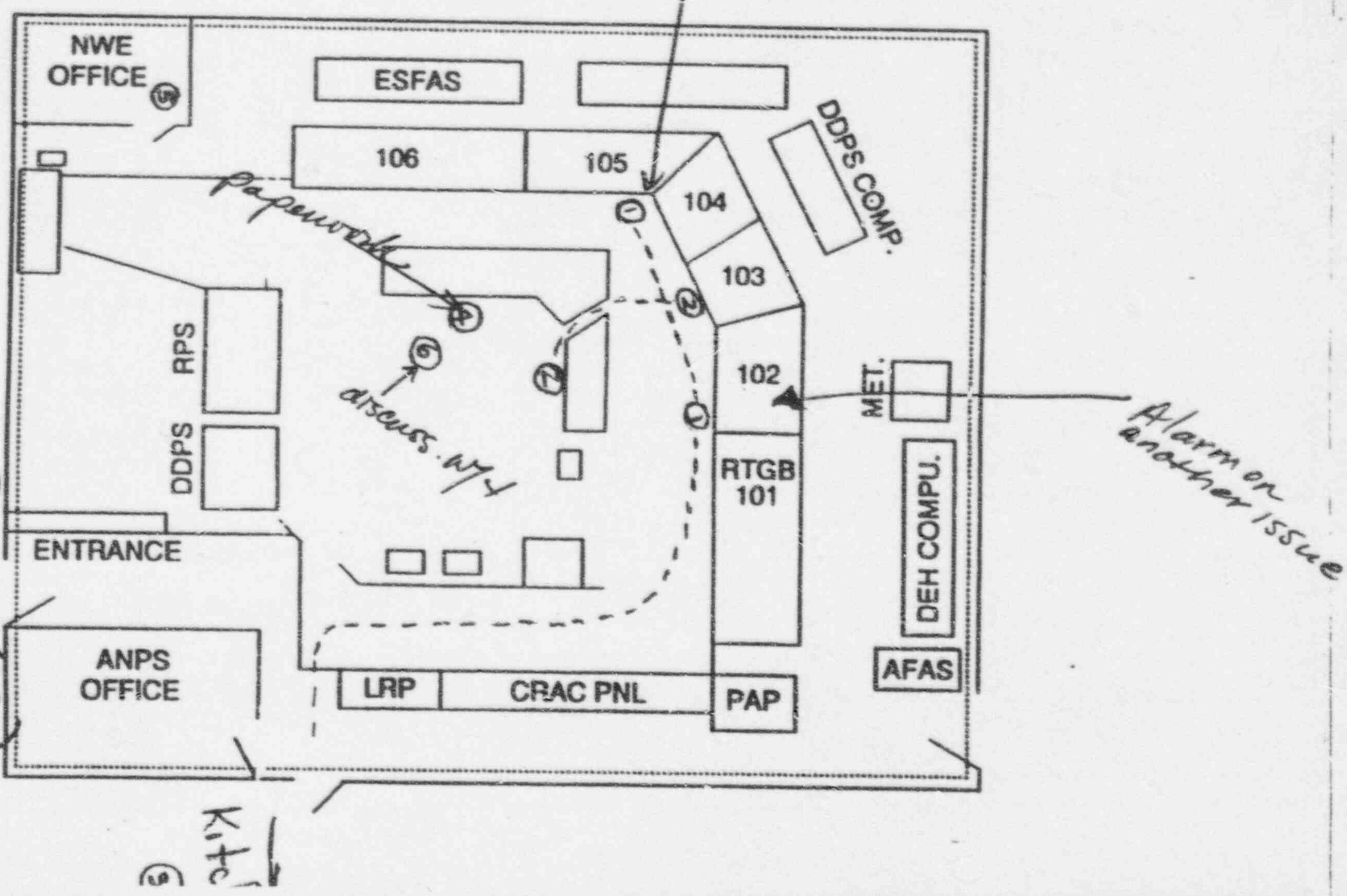


ST. LUCIE PLANT
ADMINISTRATIVE PROCEDURE NO. 0010120, REVISION 79
<u>CONDUCT OF OPERATIONS</u>

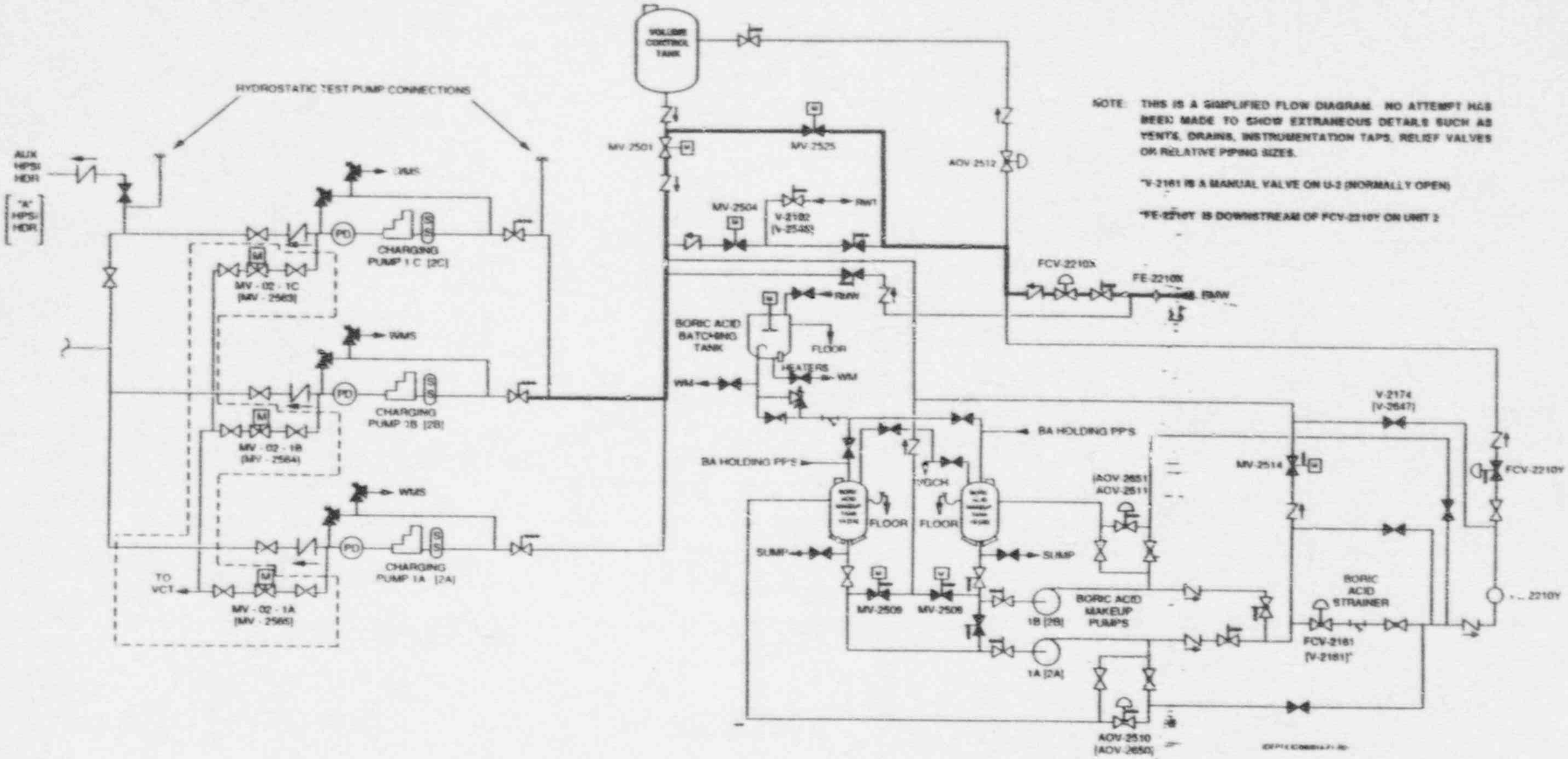
## FIGURE 3

controls for Boron dilution

① Board RCO
② Desk RCO
③ ANPS (SRO + Sup.)
④ NPS (SRO)
⑤ NWE
⑥ STA - not licensed



# CVCS CHARGING




NOTE: THIS IS A SIMPLIFIED FLOW DIAGRAM. NO ATTEMPT HAS BEEN MADE TO SHOW EXTRANEOUS DETAILS SUCH AS VENTS, DRAINS, INSTRUMENTATION TAPS, RELIEF VALVES OR RELATIVE PIPING SIZES.

*V-2161 IS A MANUAL VALVE ON U-2 (NORMALLY OPEN)

**FE-2210Y IS DOWNSTREAM OF FCV-2210Y ON UNIT 2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE UNIT 1
OPERATING PROCEDURE NO. 1-0250020
REVISION 35



1.0 TITLE:

BORON CONCENTRATION CONTROL - NORMAL OPERATION

2.0 REVIEW AND APPROVAL:

Reviewed by Plant Nuclear Safety Committee__________ 5/30 1974

Approved by ___K.N. Harris___ Plant General Manager__________ 6/3 1974

Revision 35 Reviewed by Facility Review Group__________ 8/10 & 8/17 1995

Approved by ___C. L. Burton___ Plant General Manager__________ 8/17 1995

3.0 PURPOSE:

This procedure establishes a method of operation to supply makeup water to the Reactor Coolant System (RCS), Safety Injection System and Refueling Water Tank (RWT) at a desired boron concentration and provides instructions for the following modes of control:

3.1 BORATE

3.2 DILUTE

3.3 MANUAL

3.4 AUTOMATIC

3.5 Shutdown Cooling (SDC) Boron Concentration Control

8.0 INSTRUCTIONS: (continued)

8.4 (continued)

3. Enter the number of gallons to be added into the PMW Batch Integrator and set desired flow rate on FRC-2210X (Makeup Water Flow).

4. Start one Primary Water Pump if not running.

5. Place V2512 in the OPEN position.

6. Place Mode Selector switch in DILUTE and observe flow indication of FRC-2210X.

7. Monitor VCT level to ensure tank does not fill up to high level alarm. For extended dilutions, match makeup flow with charging flow using the PMW makeup flow controller to prevent over-filling the VCT while diverting letdown.

8. Upon completion of dilution, return V2512 control switch to AUTO or CLOSED position.

9. Return Mode Selector Switch to AUTO or MANUAL.

10. Ensure that the desired reactivity change occurs.

8.5 Manual Mode of Operation

1. Determine the desired volume to be added to the VCT and calculate the proper blend ratio using the most recent chemistry boron samples of the 1A or 1B BAMT and the RCS. If the chemistry sample for the RCS is not available then use the boronometer reading.

8.0 INSTRUCTIONS: (continued)

8.5 (continued)

> **NOTE**
>
> The following formulas can be used to determine volume and blend ratio. Remember to make note of the current totalizer readings.
>
> Volume to be added = desired VCT level % - actual VCT level % X 33.8 gal/%.
>
> Blend ratio = BAMT Concentration divided by RCS Concentration minus one
>
> $$\frac{BAMT}{RCS} - 1$$

2. Ensure Mode Select switch is selected to MANUAL.
3. Place FRC-2210Y and FRC-2210X to manual and close FCV-2210Y and FCV-2210X by taking the controller output to zero.
4. Ensure 1A or 1B primary water pump is running.
5. Ensure the BAM pump recirc valves V2510 and V2511 are open.
6. Start either the 1A or 1B BAM pump.
7. Open the Boric Acid Makeup isolation valve FCV-2161.
8. Ensure FCV-2210X, Reactor Makeup valve, selector switch is in AUTO.
9. Ensure FCV-2210Y, Boric Acid valve, selector is in AUTO.
10. If blending directly to the VCT, then open V2512, Reactor Makeup Water stop valve.
11. If direct path to the charging pump suction is desired, then open valve V2525, Boron Load Control Valve.

8.0 INSTRUCTIONS: (continued)

8.5 (continued)

**CAUTION**

To preclude lifting the VCT relief valve while using V2525, do not allow the combined PMW and boric acid flowrates to exceed the running charging pump(s) capacity.

12. Adjust FRC-2210X and FRC-2210Y to the desired flow rates.

**NOTE**

Monitor VCT level for increase.

**NOTE**

The addition of Boric Acid should be completed before the PMW, such that, the total blend volume remaining allows for at least 30 gallons of primary makeup water alone, to flow through the lines and flush out any remaining boric acid.

13. When the desired amount of Boric Acid has been added, place the selector switch for FCV-2210Y to CLOSE.
14. When the Boric Acid and water flow totalizers show that the proper amounts have been added to the VCT, then close V2512 or V2525, which ever was used.
15. Place the running BAM pump switch to AUTO and ensure pump stops.
16. Close FCV-2161.
17. Close FCV-2210X.
18. Monitor for any abnormal change in temperature. Check Boronometer for undesirable change in Boron Concentration.

# FIGURE 4
## TEMPORARY CHANGE REQUEST
(Page 1 of 3)

**A**

Reference Information: (**Originator** to complete)

St. Lucie Unit # PSL 1 TC # 1-96-017

Procedure Title: BORON CONCENTRATION CONTROL - NORMAL OPERATION

Procedure Number: OP 1-0250020 Rev. 35

Reason for change: ADD PROCEDURAL GUIDANCE FOR CVCS OPERATION, MANUAL DILUTION AND BORATION OF THE RCS. THIS INFORMATION IS IN THE SAME FORMAT IN OP 2-0250020, REV 23.

Originator: R PENNINGA Phone: X 7059 Date: JAN / 23 / 1996

**B**

Procedural Controls: (**Originator** to complete)

| Yes | No | |
|---|---|---|
| ☐ | ☒ | Is the intent of the procedure altered? (Tech. Spec. 6.8.3.A) If yes, a TC is NOT applicable. A PCR is required. |
| ☐ | ☒ | Is this Temporary Change for a one-time use? If yes, this TC can be executed one time only. If no, this TC may be used up to 90 days, and the originator of the TC shall submit a procedure change request incorporating this TC at the same time the TC is approved. |

Department Head or Designee [signature] for C. H. Wood 1/23/96

| Yes | No | |
|---|---|---|
| ☐ | ☒ | Is this T.C. for a Q.I.? If yes, the Quality Manager or designee and the Dept. Head or designee who is jurisdictionally responsible for the Q.I. shall sign. |

Quality Manager or Designee ____________ ___/___/___

Department Head or Designee ____________ ___/___/___

**C**

Temporary Change Contents: (**Originator** to complete)

Does this Change:

| Yes | No | |
|---|---|---|
| ☐ | ☒ | Incorporate complex or extensive changes? If Yes, Subcommittee required. ________ Subcommittee Initials |
| ☐ | ☒ | Modify instrument setpoints? |
| ☐ | ☒ | Delete an independent verification? |
| ☐ | ☒ | Alter a QC holdpoint? |
| ☐ | ☒ | Modify a procedural step which alters a regulatory requirement as identified in the procedure? |
| ☐ | ☒ | Alter the first execution of a procedure? (Preop, LOI) |
| ☐ | ☒ | Addition of any chemicals? |

> **NOTE**
> If any of the above criteria are marked yes, prior FRG review is required.

# FIGURE 4
## TEMPORARY CHANGE REQUEST
(Page 2 of 3)

TC # 1-96-017

**D** 10 CFR 50.59 Screening

| | Yes | No |
|---|---|---|
| 1. Does the change represent a change to the facility as described in the SAR? | | X |
| 2. Does the change represent a change to procedures as described in the SAR? | | X |
| 3. Is the change associated with a test or experiment not described in the SAR? | | X |
| 4. Could the change affect nuclear safety in a way not previously evaluated in the SAR? | | X |
| 5. Does the change require a change to the Technical Specifications? | | X |

**NOTE**
If the answer to ALL the above 10 CFR 50.59 screening questions are no, (Questions 1 - 5), then a safety evaluation is not required.

STA review (signature) ____________ Date 1/23/96

**E** Does this change: (NPS to complete)

| | Yes | No |
|---|---|---|
| 1. Compromise the separation of redundant trains of equipment? | | ✓ |
| 2. Potentially isolate pressure reliefs? | | ✓ |
| 3. Defeat automatic signals? | | ✓ |
| 4. Defeat mechanical or electrical interlocks? | | ✓ |
| 5. Alter the completion of an evolution due to an operator work around. | | ✓ |

If yes to No. 5, authorization from the Plant General Manager or Site Vice President shall be obtained.

____________ Date ___/___/___

Yes ☐ No ☒ Prior FRG review required?

**NOTE**
If any of the above criteria are marked yes, discuss possible alternatives with the originator.

NPS Signature ____________ Date 1/23/96

**F** FRG Review:

Plant General Manager Approval ____________ Date ___/___/___

FRG Number ______-______ This change shall be reviewed (if prior FRG review is not required) by the Facility Review Group and approved by the Plant General Manager within 14 days of the authorization date. (Tech. Spec. 6.8.3.C)

REJECTED by FRG/Plant General Manager ____________ Date ___/___/___

Reason: ____________

Return to Originator ____________

It is the responsibility of the originator of the rejected temporary change to cancel the change in the appropriate Control Room, destroy all field copies and halt all subsequent evolutions using this temporary change.

**FIGURE 4**
**TEMPORARY CHANGE REQUEST**
(Page 3 of 3)

G

TC # 1-96-017

Approval: (This change shall have prior approval by a NPS and one member of the **plant management staff.**) (Tech. Spec. 6.8.3.B)

Plant Management Staff Signature ____ Date 1/23/96

NPS Signature ____ Authorization Date 1/23/96

H

Cancellation Authorization ____ (NPS/ANPS) Date __/__/__

Reason: ____

8.0 INSTRUCTIONS: (continued)

8.4 (continued)

3. Enter the number of gallons to be added into the PMW Batch Integrator and set desired flow rate on FRC-2210X (Makeup Water Flow).

4. Start one Primary Water Pump if not running.

5. Place V2512 in the OPEN position.

6. Place Mode Selector switch in DILUTE and observe flow indication of FRC-2210X.

7. Monitor VCT level to ensure tank does not fill up to high level alarm. For extended dilutions, match makeup flow with charging flow using the PMW makeup flow controller to prevent over-filling the VCT while diverting letdown.

8. Upon completion of dilution, return V2512 control switch to AUTO or CLOSED position.

9. Return Mode Selector Switch to AUTO or MANUAL.

10. Ensure that the desired reactivity change occurs.

1-96-017

8.5 Manual Mode of Operation

1. Manual Blend

A. Determine the desired volume to be added to the VCT and calculate the proper blend ratio using the most recent chemistry boron samples of the 1A or 1B BAMT and the RCS. If the chemistry sample for the RCS is not available then use the boronometer reading.

ST. LUCIE UNIT 1
OPERATING PROCEDURE NO. 1-0250020, REVISION 35
BORON CONCENTRATION CONTROL - NORMAL OPERATION

8.0 INSTRUCTIONS: (continued)

8.5 (continued)

> **NOTE**
>
> The following formulas can be used to determine volume and blend ratio. Remember to make note of the current totalizer readings.
>
> Volume to be added = desired VCT level % - actual VCT level % X 33.8 gal/%.
>
> Blend ratio = BAMT Concentration divided by RCS Concentration minus one
>
> $$\frac{BAMT}{RCS} - 1$$

B. Ensure Mode Select switch is selected to MANUAL.

C. Place FRC-2210Y and FRC-2210X to manual and close FCV-2210Y and FCV-2210X by taking the controller output to zero.

D. Ensure 1A or 1B primary water pump is running.

E. Ensure the BAM pump recirc valves V2510 and V2511 are open.

F. Start either the 1A or 1B BAM pump.

G. Open the Boric Acid Makeup isolation valve FCV-2161.

H. Ensure FCV-2210X, Reactor Makeup valve, selector switch is in AUTO.

I. Ensure FCV-2210Y, Boric Acid valve, selector is in AUTO.

J. If blending directly to the VCT, then open V2512, Reactor Makeup Water stop valve.

K. If direct path to the charging pump suction is desired, then open valve V2525, Boron Load Control Valve.

Add new section 8.5.2 and 8.5.3 per attached sheets

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8.0 INSTRUCTIONS: (continued)

8.5 (continued)

> **CAUTION**
> To preclude lifting the VCT relief valve while using V2525, do not allow the combined PMW and boric acid flowrates to exceed the running charging pump(s) capacity.

L. 12. Adjust FRC-2210X and FRC-2210Y to the desired flow rates.

> **NOTE**
> Monitor VCT level for increase.

> **NOTE**
> The addition of Boric Acid should be completed before the PMW, such that, the total blend volume remaining allows for at least 30 gallons of primary makeup water alone, to flow through the lines and flush out any remaining boric acid.

M. 13. When the desired amount of Boric Acid has been added, Then place the selector switch for FCV-2210Y to CLOSE.

N. 14. When the Boric Acid and water flow totalizers show that the proper amounts have been added to the VCT, then close V2512 or V2525, which ever was used.

O. 15. STOP the running BAM Pump and Place the running BAM pump switch to AUTO. ~~and ensure pump stops.~~

P. 16. Close FCV-2161.

Q. 17. Close FCV-2210X.

R. 18. Monitor for any abnormal change in temperature. Check Boronometer for undesirable change in Boron Concentration.

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Add new section 8.5.2 and 8.5.3 per attached sheets

8.0 INSTRUCTIONS: (continued)

8.5 (continued)

2. Manual Dilution

| NOTE |
|---|
| VCT level equates to 33.8 gallons per percent of scale on LIC-2226, VCT Level. |

A. Determine the desired volume of water to be added.

B. Ensure the Make-up Mode Selector switch is selected to MANUAL.

C. Ensure that FRC-2210X, Make-up Water Flow, is in MANUAL and reduce the controller output to zero (0).

D. Ensure that FRC-2210Y, Boric Acid Flow, is in MANUAL. and reduce the controller output to zero (0).

E. Ensure that FCV-2210Y, Boric Acid Valve, selector is in CLOSE.

F. Ensure that either the 1A or the 1B Primary Make-up Water Pump is running.

G. Place FCV-2210X, Reactor Make-up, selector switch in AUTO.

H. If diluting to the VCT, Then OPEN V2512, Reactor Make-up Water Stop Vlv.

I. If diluting directly to the suction of the charging pumps, Then OPEN V2525, Boron Load Control Valve.

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8.0 INSTRUCTIONS: (continued)

8.5 (continued)

2. (continued)

**CAUTION**
To preclude lifting the VCT relief valve while using V2525, do NOT allow the PMW flowrate to exceed the running charging pump flow rate.

J. Adjust FRC-2210X to the desired flowrate.

K. If necessary to maintain the desired VCT level, Then divert the letdown flow to the WMS by placing V2500, VCT Divert Valve, in the WMS position.

L. When the desired VCT level is reached, Then:

1. Return V2500, VCT Divert Valve, to the AUTO position.

2. Ensure that V2500 indicates CLOSED.

M. When the desired amount of PMW has been added, Then place the FCV-2210X selector switch in the CLOSE position.

N. CLOSE V2512 or V2525, whichever was used.

O. Ensure that FRC-2210X is in MANUAL and reduce the controller output to zero (0).

P. Monitor for unexpected results:

1. Abnormal change in the RCS temperature.

2. Undesired change in the RCS boron concentration by boronmeter indication.

TC #1-96-Ø17

8.0 INSTRUCTIONS: (continued)

8.5 (continued)

3. Manual Boration

**NOTE**
VCT level equates to 33.8 gallons per percent of scale on LIC-2226, VCT Level.

A. Determine the desired volume of boric acid to be added.

B. Ensure the Make-up Mode Selector switch is selected to MANUAL.

C. Ensure that FRC-2210X, Make-up Water Flow, is in MANUAL and reduce the controller output to zero (0).

D. Ensure that FRC-2210Y, Boric Acid Flow, is in MANUAL and reduce the controller output to zero (0).

E. Ensure that FCV-2210Y, Boric Acid Valve, selector is in CLOSE.

F. Ensure that either the 1A or the 1B Primary Make-up Water Pump is running.

**NOTE**
While it is acceptable to use either BAMT for RCS boration, it is preferable to operate the BAM Pump for the BAMT NOT designated as 'Tech Spec'.

G. START either the 1A or the 1B BAM Pump.

H. Place FCV-2210Y, Boric Acid Valve, selector switch in AUTO.

I. OPEN FCV-2161, Boric Acid Make-up Isolation.

J. If borating directly to the VCT, Then OPEN V2512, Reactor Make-up Water Stop Vlv.

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8.0 INSTRUCTIONS: (continued)

8.5 (continued)

3. (continued)

K. If borating directly to the suction of the charging pumps, Then OPEN V2525, Boron Load Control Valve.

L. Adjust FRC-2210Y to the desired flowrate.

M. If necessary to maintain the desired VCT level, Then divert the letdown flow to the WMS by placing V2500, VCT Divert Valve, in the WMS position.

N. When the desired VCT level is reached, Then:

1. Return V2500, VCT Divert Valve, to the AUTO position.

2. Ensure that V2500 indicates CLOSED.

O. When the desired amount of boric acid has been added, Then place the FCV-2210Y selector switch in the CLOSE position.

P. CLOSE FCV-2161, Boric Acid Make-up Isolation.

| CAUTION |
|---|
| To preclude lifting the VCT relief valve while using V2525, do NOT allow the PMW flowrate to exceed the running charging pump flow rate. |

Q. STOP the running BAM pump and place the selector switch in the AUTO position.

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8.0 INSTRUCTIONS: (continued)

8.5 (continued)

3. (continued)

R. If flushing the CVCS piping following boration is desired, Then:

1. Place FRC-2210X, Make-up Water Flow, controller in AUTO.

**CAUTION**
To preclude lifting the VCT relief valve while using V2525, do NOT allow the PMW flowrate to exceed the running charging pump flow rate.

2. Adjust FRC-2210X to the desired flowrate to flush the lines with a total of at least 30 gallons of PMW.

3. When the desired amount of PMW has been added, Then place the FCV-2210X selector switch in the CLOSE position.

4. Place FRC-2210X in MANUAL and reduce the controller output to zero (0).

S. CLOSE V2512 or V2525, whichever was used.

T. Ensure that FRC-2210Y, Boric Acid Flow, is in MANUAL and reduce the controller output to zero (0).

U. Monitor for unexpected results:

1. Abnormal change in the RCS temperature.

2. Undesired change in the RCS boron concentration by boronmeter indication.

TC # 1-96-017

PSL
CONTROL COPY
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PROCEDURE PRODUCTION

FLORIDA POWER & LIGHT COMPANY
NUCLEAR ENERGY DEPARTMENT
ST. LUCIE PLANT

# PREPARATION, REVISION, REVIEW/APPROVAL OF PROCEDURES

## 1.0 APPROVAL:

Reviewed by Facility Review Group ______ 1/30 1975

Approved by J.H. Barrow (for) Plant General Manager ______ 2/3 1975

Revision 67 Reviewed by FRG ______ 12/8 1995

Approved by J. Scarola Plant General Manager ______ 12/8 1995

## 2.0 PURPOSE:

2.1 This procedure provides administrative guidance for the preparation, review, approval and revision of all plant procedures and letters of instruction, for use at the St. Lucie Plant.

2.2 This procedure defines the instructions that shall be used by St. Lucie Plant personnel to assure conformance with NRC Regulatory Guides 1.33 and 1.68, NUREG-0737 and the Site Quality Manual (SQM 2.1 and 5.0).

S__OPS

DATE______
DOCT PROCEDURE
DOCN QI-5-1
SYS______
COMP COMPLETED
ITM 67

5.0 INSTRUCTIONS: (continued)

5.12 (continued)

2. Controlled vendor technical manuals may be utilized as references to safety or non-safety related NPWOs to provide technical guidance (e.g., DWGs, specifications, torque values, dimensional information, voltage/current values, etc.) to supplement an invoked plant approved procedure/guideline or the work scope/instructions without prior FRG Review/Plant General Manager approval. In this case, the vendor's step-by-step maintenance instructions are not being used.

3. Changes to technical manuals received from the vendor or changes initiated by FPL shall be forwarded to PEG/JB for review and approval.

4. New technical manuals received from vendors shall be numbered and controlled in accordance with QI 6-PR/PSL-1.

5. The maintenance and preventive maintenance requirements specified in technical manuals shall be considered when writing maintenance procedures. Vendor recommendations for preventive maintenance activities or frequencies contained in these Vendor Tech. Manuals may be deviated from, provided a technical review is performed by the respective maintenance engineering group.

6. Distribution of revisions to vendor technical manuals shall be maintained by the Information Services Supervisor or designee.

5.13 Adherence to Procedures:

1. A strict adherence to procedural requirements - Verbatim Compliance - is the policy expected and required of all St. Lucie Plant personnel.

2. A procedure shall be performed in a step by step manner, with each step being completed prior to the performance of the next step, unless exceptions allowed by the procedure or as specified by this procedure.

   A. Procedures and Instructions of an Administrative nature (Quality Instructions, ADMs, etc.) shall not be violated, but step by step implementation is not required. By nature, these types of procedures and instructions often do not lead themselves to sequential implementation.

   B. Procedures and instructions that are of a technical nature shall be followed sequentially except as specifically allowed by approved plant procedures.

5.0 INSTRUCTIONS: (continued)

5.13 (continued)

2. (continued)

B. (continued)

1. Required sign-offs and data entries shall be made as each step is performed.

2. If a procedure step cannot be completed as written, or if in the judgement of the individual performing a procedure, completion of a specific step(s) could result in an unsafe condition (e.g., personnel injury, damage to equipment, conditions outside the limits of the procedure etc.), conduct of the procedure shall be stopped, the system/components placed in a safe condition and the Nuclear Plant Supervisor shall be notified.

3. Deviation from Procedure Valve Checklists may be made provided the deviation is noted in ink on the applicable valve alignment and is approved (initialed and dated) by the Nuclear Plant Supervisor.

3. Personnel shall not give directions, guidance, recommendation, or clarifications which conflict with approved procedures.

4. Adherence to procedures shall be accomplished by use of one of the following methods:

A. Method 1 - Procedure Present During Performance of Activity: The types of procedures that shall be present and referred to directly are:

1. Those procedures developed for extensive or complex jobs where reliance on memory cannot be trusted.

2. Tasks which are infrequently performed.

3. Tasks which must be performed in a specified sequence and/or which verification is documented by initial or signature.

5.0 INSTRUCTIONS: (continued)

5.13 (continued)

B. Method 2 - Memorization: Method by which the procedural steps for the required actions are committed to memory. This method does not permit any deviation from the Procedural Adherence Policy.

1. Procedures for which actions should be committed to memory are Immediate Actions in Emergency Operating Procedures and Off Normal Operating Procedures.

2. Procedures for which actions may be committed to memory are routine procedural actions that are frequently repeated and may not require the procedure to be present during performance of the activity. However, copies of procedures shall be available to the user at his/her work location for reference during performance of the task, if necessary.

5. Procedural adherence may be accomplished by use of a Temporary Change, if necessary.

6. When used in a procedure the word "shall" is used to denote a requirement, the word "should" to denote a recommendation and the word "may" to denote permission, neither a requirement nor a recommendation.

7. Independent Verification:

A. Independent Verification has been defined in ADM-17.06, "Independent Verification." Definitions of Independent Verification should not be added to procedures as they may conflict with the guidance outlined in ADM-17.06.

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT
ADMINISTRATIVE PROCEDURE NO. 0010120
REVISION 79



1.0 TITLE:

CONDUCT OF OPERATIONS

2.0 REVIEW AND APPROVAL:

Reviewed by Plant Nuclear Safety Committee 1/17 1975

Approved by J. H. Barrow (for) Plant General Manager 1/22 1975

Revision 79 Reviewed by Facility Review Group 12/21 1995

Approved by J. Scarola Plant General Manager 12/21 1995

3.0 SCOPE:

3.1 Purpose:

This procedure defines the responsibilities and conduct of the Operations Department during the performance and documentation of all departmental activities. This procedure provides instruction to ensure that plant operations are conducted in an effective, consistent, professional and businesslike manner as per the operating license, plant procedures and applicable regulatory requirements.

This procedure applies to all persons in the Operations Department. It identifies operational requirements and management policies necessary to ensure the daily conduct of plant operations is consistent with good operational and engineering practices.

ST. LUCIE PLANT
ADMINISTRATIVE PROCEDURE NO. 0010120, REVISION 79
CONDUCT OF OPERATIONS

**APPENDIX D**
**CREW RELIEF/SHIFT TURNOVER**
(Page 5 of 5)

1. (continued)

D. Instruction for an Interim or Short Term Relief/Shift Turnover.

1. If a specific watchstander requires a short term relief for a period of less than 2 hours, then the following instructions provide the minimum requirements for shift relief:

    a. General watchstation status.

    b. Off-normal conditions.

    c. Tests in progress.

2. The applicable unit ANPS shall be notified immediately after the shift turnover has been completed.

3. If an individual is expected to be absent for period of greater than 2 hours, then an Individual Relief/Split-Shift Turnover shall be performed.

## APPENDIX E
## NOTIFICATION OF OPERATIONS SUPERVISOR/FPL MANAGEMENT
(Page 1 of 3)

1. The Nuclear Plant Supervisor is responsible for notifying higher station authorities and appropriate station personnel. Advance notification should be made when possible. The following situations require prompt, verbal notifications:

   Notify the Operations Supervisor for the following situations:

   A. Any event that would cause entry into an Emergency Operating Procedure (EOP).

   B. Any event requiring phone call notification to the NRC.

   C. Any event that will generate an LER.

   D. Inadvertent radioactive liquid or gaseous release.

   E. Major equipment failure or malfunctions.

   F. Unexplained or unplanned reactivity changes.

   G. Forced power reduction.

   H. Major personnel injury or radiation overexposure.

   I. Any LCO that would require unit shutdown within the next 24 hours.

   J. Any operational event that generates an In House Event (IHE) Report AND causes heightened awareness to FPL sources offsite.

   K. Any release that is or is potentially, damaging to the environment.

   L. Load restrictions or inability to meet load dispatcher requirements. This includes, but is NOT limited to the following:

      1. A planned power escalation is unexpectedly halted for any reason and can not be resumed within one hour.

      2. If at a power level less than 100 percent, any unexpected condition that would prevent a future power escalation and can not be resolved within two hours.

      3. If at a power level less than 100 percent and the plant is unable to support an unexpected request for more power from the load dispatcher.

ST. LUCIE PLANT
ADMINISTRATIVE PROCEDURE NO. 0010120, REVISION 79
CONDUCT OF OPERATIONS

**APPENDIX F**
**LOG KEEPING**
(Page 2 of 9)

2. Chronological Logs:

A. Log books and/or computerized logs shall be maintained at the RCO, NO/SNPO, NTO/NPO and ANPO normal stations. Entries are to be in concise and complete enough to reconstruct the events of the shift. Particular attention should be made to the entries pertaining to any abnormal condition that occurs. Times for each entry shall be as near correct as possible using military time. The entries are to be made in chronological order.

1. Evolutions, manipulations and operations that are performed, observed and monitored by operators NOT actively assuming the responsibilities of a particular watch station shall be recorded in the applicable watch station chronological log and initialed by that operator. The operator should notify the responsible watchstander of the log entry.

2. When it is necessary to insert additional information after the fact, Then the entry shall be recorded with the actual time of occurrence, the words Late Entry in parenthesis, and the information to be logged.

| | | |
|---|---|---|
| Example: | 1234 | Started the 1A EDG for surveillance run |
| | 0827 | (Late Entry) Filled the 1A2 SIT with the 1B HPSI Pump in accordance with OP 1-0410021 |
| | 1345 | Secured the 1A EDG. Surveillance run SAT. |

3. When it is necessary to correct information recorded in error, then the entry shall be recorded with the actual time of occurrence, the words "Corrected Entry" in parenthesis, and the information to be logged.

| | | |
|---|---|---|
| Example: | 1234 | Started the 1B EDG for surveillance run |
| | 1345 | Secured the 1A EDG. Surveillance run SAT. |
| | 1234 | (Corrected Entry) Started the 1A EDG for surveillance run |

4. Entries in the RCO log should include, but are NOT to be limited to, the following:

a. Conditions at the beginning of each watch.

**APPENDIX F**
**LOG KEEPING**
(Page 3 of 9)

2. Chronological Logs: (continued)

A. (continued)

4. (continued)

b. Significant changes in plant conditions.

Examples: 1. Mode changes.

2. Load changes.

3. Reactivity changes.

4. Startups and Shutdown.

5. Time of Reactor criticality.

c. Any new condition that would limit unit generation.

Examples: 1. Condenser back pressure at administrative limits.

2. Chemistry parameters limiting operation.

d. Special tests, including periodic and surveillance testing, for major equipment.

Examples: 1. Start and stop times for periodic or surveillance tests and outcome (SAT or UNSAT), for major equipment.

2. Post maintenance testing and outcome, for major equipment.

e. Control problems associated with major equipment or systems.

Examples: 1. Changes in plant work arounds.

**APPENDIX M**
**PROCEDURAL COMPLIANCE AND IMPLEMENTATION**
(Page 1 of 6)

1. Controlled procedures are available in both Control Rooms and shall be implemented and complied with in accordance with the instructions provided in QI 5-PR/PSL-1, "Preparation, Revision, Review/Approval of Procedures."

2. In the event of an emergency where procedural guidance does NOT exist or in which a specific emergency is NOT addressed by an approved procedure, then Operations personnel shall take action to protect the health and safety of the public, minimize personnel injury, and damage to the facility.

3. Numerous tasks performed by the operators are repetitive and routine in nature. These tasks come under the guidance of the memorization method of adherence to procedures in accordance with QI 5-PR/PSL-1, "Preparation, Revision, Review/Approval of Procedures," and may be performed from memory. These tasks, which are listed in the following sections, are considered to be skill of the trade for qualified operators. Each listed task shall have one or more of the below justification reasons:

   **(A)** Task is routine and not complex - satisfactory completion assured by routine training and observation.

   **(B)** Task is routine and has a low level of complexity - satisfactory completion assured by completion of verification checklist and independent verification.

   **(C)** Posted instructions in place as reference.

   **(D)** Satisfactory completion assured by multiple levels of review and/or feedback from system.

A. **General Control Tasks**

   1. Racking IN and OUT of 6.9 KV, 4.16 KV, and 480V breakers. **(B)**

   2. Turning ON and OFF 480V MCC breakers. **(D)**

   3. Writing clearances and NPWOs. **(A,D)**

   4. Changing chart paper. **(A,D)**

   5. Placing controllers in MANUAL or AUTO. **(A,D)**

## APPENDIX M
## PROCEDURAL COMPLIANCE AND IMPLEMENTATION
(Page 2 of 6)

1. (continued)

B. **Reactor Control Operator**

1. Divert Letdown to Control VCT level. **(A,D)**
2. Check Sheet 1 of AP 1-0010125. **(A,B,D)**
3. Refueling Operations - movement of machine, etc. **(A,D)**
4. Adjusting Main Generator loading, including Megavars and Megawatts (manipulation of D[illegible] controls). **(D)**
5. Swapping Auxiliary and Start-up Transformers. **(D)**
6. Adjusting CEA position (eg. ASI control). **(D)**
7. Manipulation of control valve[illegible] DVs, FCVs) to control Heatup and Cooldown rates. **(D)**
8. Pumping down Reactor Drain Tank. **(A,D)**
9. Placing CST on recirc. **(A,D)**

C. **Senior Nuclear Plant Operator**

1. Generic Rounds Sheets. **(A)**
2. Swapping HUTs. **(A,D)**
3. Blowing down BAMT level transmitters. **(C)**
4. Operator Readings and AP 0010125 checks. **(A,B,D)**
5. Recirculating of HUTs, WMTs, and AWSTs. **(A,D)**
6. Backwashing ICW/CCW strainers. **(C)**

**FIGURE 4**
**TEMPORARY CHANGE REQUEST**
(Page 1 of 3)

A

Reference Information: (**Originator** to complete)

St. Lucie Unit # COMMON TC # 0-96-014

Procedure Title: CONDUCT OF OPERATIONS

Procedure Number: AP 0010120 Rev. 79

Reason for change: INCORPORATE MANAGEMENT DIRECTIVES

THIS TC SUPERCEDES TC 0-96-011

Originator: CZACHOR Phone: 7091 Date: 1/29/96

B

Procedural Controls: (**Originator** to complete)

| Yes | No | |
|---|---|---|
| ☐ | ☒ | Is the intent of the procedure altered? (Tech. Spec. 6.8.3.A) If yes, a TC is NOT applicable. A PCR is required. |
| ☐ | ☒ | Is this Temporary Change for a one-time use? If yes, this TC can be executed one time only. If no, this TC may be used up to 90 days, and the originator of the TC shall submit a procedure change request incorporating this TC at the same time the TC is approved. |

Department Head or Designee C H [signature] 1/29/96

| Yes | No | |
|---|---|---|
| ☐ | ☒ | Is this T.C. for a Q.I.? If yes, the Quality Manager or designee and the Dept. Head or designee who is jurisdictionally responsible for the Q.I. shall sign. |

Quality Manager or Designee ____________ __/__/__

Department Head or Designee ____________ __/__/__

C

Temporary Change Contents: (**Originator** to complete)

Does this Change:

| Yes | No | |
|---|---|---|
| ☐ | ☒ | Incorporate complex or extensive changes? If Yes, Subcommittee required. ______ Subcommittee Initials |
| ☐ | ☒ | Modify instrument setpoints? |
| ☐ | ☒ | Delete an independent verification? |
| ☐ | ☒ | Alter a QC holdpoint? |
| ☐ | ☒ | Modify a procedural step which alters a regulatory requirement as identified in the procedure? |
| ☐ | ☒ | Alter the first execution of a procedure? (Preop, LOI) |
| ☐ | ☒ | Addition of any chemicals? |

**NOTE**
If any of the above criteria are marked yes, prior FRG review is required.

# FIGURE 4
## TEMPORARY CHANGE REQUEST
(Page 2 of 3)

TC # 0-96-014

**D**

**10 CFR 50.59 Screening**

| | Yes | No |
|---|---|---|
| 1. Does the change represent a change to the facility as described in the SAR? | ___ | ✓ |
| 2. Does the change represent a change to procedures as described in the SAR? | ___ | ✓ |
| 3. Is the change associated with a test or experiment not described in the SAR? | ___ | ✓ |
| 4. Could the change affect nuclear safety in a way not previously evaluated in the SAR? | ___ | ✓ |
| 5. Does the change require a change to the Technical Specifications? | ___ | ✓ |

NOTE
If the answer to ALL the above 10 CFR 50.59 screening questions are no, (Questions 1 - 5), then a safety evaluation is not required.

STA review (signature) ________ Date 1/29/96

**E**

| Does this change: (**NPS** to complete) | Yes | No |
|---|---|---|
| 1. Compromise the separation of redundant trains of equipment? | ___ | ✓ |
| 2. Potentially isolate pressure reliefs? | ___ | ✓ |
| 3. Defeat automatic signals? | ___ | ✓ |
| 4. Defeat mechanical or electrical interlocks? | ___ | ✓ |
| 5. Alter the completion of an evolution due to an operator work around. | ___ | ✓ |

If **yes** to No. 5, authorization from the Plant General Manager or Site Vice President shall be obtained.

________________________ Date ___/___/___

| Yes | No | |
|---|---|---|
| ☐ | ☑ | Prior FRG review required? |

NOTE
If any of the above criteria are marked yes, discuss possible alternatives with the originator.

NPS Signature ________ Date 1/29/96

**F**

FRG Review:

Plant General Manager Approval ________________________ Date ___/___/___

FRG Number ______-______ This change shall be reviewed (if prior FRG review is not required) by the Facility Review Group and approved by the Plant General Manager within 14 days of the authorization date. (Tech. Spec. 6.8.3.C)

REJECTED by FRG/Plant General Manager ________________________ Date ___/___/___

Reason: ________________________________________________

________________________________________________

Return to Originator ________________________

It is the responsibility of the originator of the rejected temporary change to cancel the change in the appropriate Control Room, destroy all field copies and halt all subsequent evolutions using this temporary change.

**FIGURE 4**
**TEMPORARY CHANGE REQUEST**
(Page 3 of 3)

G

TC # 0-96-014

Approval: (This change shall have prior approval by a **NPS** and one member of the **plant management staff**.) (Tech. Spec. 6.8.3.B)

Plant Management Staff Signature ____________ Date 1/29/96

NPS Signature ____________ Authorization Date 1/29/96

H

Cancellation Authorization ____________(NPS/ANPS) Date ___/___/___

Reason: ____________

ST. LUCIE PLANT
ADMINISTRATIVE PROCEDURE NO. 0010120, REVISION 79
CONDUCT OF OPERATIONS

## APPENDIX B
## SHIFT OPERATIONS POLICIES


4. (continued)

A. (continued)

4. P - Prove

a. Prove to yourself that the actions that were just performed produced the desired results.

b. Observe and verify the following:

1. The task was performed correctly.

2. The actual response was the expected response.

3. The component/system is in the proper configuration to support the intended operation.

4. The proper component was operated.

TC 0-96-014

INSERT NEW STEP 5 (attached)

5. ~~RTGB Manipulation~~

A. ~~Only licensed operators are permitted to manipulate the controls that directly affect the reactivity or power level of a reactor except for training purposes. A trainee may manipulate controls only under direct visual supervision of a licensed operator.~~

6. Unit Reliability

A. The NPS/ANPS should make every effort to prevent putting the plant in a situation where a single failure would jeopardize plant safety or availability.

Systems listed under AP 0010142, "Unit Reliability - Manipulation of Sensitive Systems" warrant particular attention.

Maintenance or testing should not be allowed on an in-service train or channel with the opposite train out-of-service or another channel in Trip, except for Tech. Spec. required surveillances or to prevent a plant shutdown.

TC 0-96-014

change after event

## APPENDIX B
## SHIFT OPERATIONS POLICIES

5. Reactivity Manipulations

   A. Reactivity manipulations in the course of normal plant operations is defined as the insertion of positive and negative reactivity due to manipulation of the following:

      1. CEA insertion and withdrawal.

      2. Addition of water and/or boric acid to the VCT or Charging Pumps' suction.

      3. Turbine/Generator load changes.

      4. Placing a purification Ion Exchanger in service, (any time V2520, "Ion Exchanger Bypass Valve," position is changed from bypassing the ion exchanger(s) to directing flow through the ion exchanger(s)).

   B. All reactivity manipulations in the course of normal plant operations, both positive and negative, shall have prior approval from the SRO fulfilling the role of the Control Room Command function, except as provided for in step 5.D.

   C. When reactivity manipulations are being performed, both positive and negative, the SRO fulfilling the role of the Control Room Command function shall directly supervise the manipulation and additionally assume the role of a reactivity manager, except as provided for in step 5.D.

   D. In the event of off-normal and emergency conditions, Reactor Control Operators are authorized to perform reactivity manipulations without the presence of and approval of an SRO, if in his/her judgement immediate intervention is required to protect the health and safety of the public and/or challenging of plant safety functions. The SRO fulfilling the role of the Control Room Command function shall be notified of the manipulation as soon as possible.

   E. Crew Relief/Shift Turnover shall NOT take place for Reactor Control Operators or the Assistant Nuclear Plant Supervisor while reactivity manipulations are in progress.

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**APPENDIX B**
**SHIFT OPERATIONS POLICIES**

5. (Continued)

   F. Reactivity manipulations shall be performed only by those individuals possessing an active license applicable to the unit on which the manipulation is being performed. The only exceptions are persons reactivating a license or in a bonafide training role in pursuit of obtaining a license; they may perform reactivity manipulations under direct visual supervision of a licensed operator with an active license.

ST. LUCIE PLANT
ADMINISTRATIVE PROCEDURE NO. 0010120, REVISION 79
CONDUCT OF OPERATIONS

## APPENDIX D
## CREW RELIEF/SHIFT TURNOVER
(Page 4 of 5)

1. (continued)

C. Instructions for an Individual Relief/Split-Shift Turnover

1. If a specific watchstation shift is being split by two individual watchstanders, then the following instructions provide the minimum requirements for shift relief:

a. The off-going watchstander shall review applicable plant log sheets to determine the existence of any off-normal condition or trends.

TC 2-96-014

b. The off-going watchstander shall complete the applicable Turnover Check Sheet (Data Sheet 1) for their watchstation.

INSERT NEW STEPS d+e (attached)

c. The off-going watchstander shall verbally transmit and explain the information as recorded on their applicable Turnover Check Sheet (Data Sheet 1) to the on-coming watchstander.

f ~~d~~. The on-coming watchstander shall review the following and acknowledge that review by initialing Check Sheet 1 of AP 1(2)-0010125, "Schedule of Periodic Test, Checks, and Calibrations."

1. Applicable Watchstation Chronological Log.

2. Applicable Watchstation Operator Log Readings.

3. Night Order Book.

4. NPWO, ANPS, and NWE shall review equipment out-of-service log.

g ~~e~~. The applicable unit ANPS shall be notified immediately after the shift turnover has been completed.

TC 0-96-014

## APPENDIX D
## CREW RELIEF/SHIFT TURNOVER

1.

C.

1.

d. On-coming and off-going control room watchstanders shall conduct a face-to-face complete walkdown of the RTGBs and control panels.

e. The on-coming watchstander shall make a chronological log entry indicating he/she has assumed the responsibilities of the watchstation.

ST. LUCIE PLANT
ADMINISTRATIVE PROCEDURE NO. 0010120, REVISION 79
CONDUCT OF OPERATIONS

## APPENDIX D
## CREW RELIEF/SHIFT TURNOVER
(Page 5 of 5)

1. (continued)

D. Instruction for an Interim or Short Term Relief/Shift Turnover.

1. If a specific watchstander requires a short term relief for a period of less than 2 hours, then the following instructions provide the minimum requirements for shift relief:

   a. General watchstation status.

   b. Off-normal conditions.

   c. Tests in progress.

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INSERT NEW step d (attached)

2. The applicable unit ANPS shall be notified immediately after the shift turnover has been completed.

3. If an individual is expected to be absent for period of greater than 2 hours, then an Individual Relief/Split-Shift Turnover shall be performed.

TC 0-96-014

# APPENDIX D
## CREW RELIEF/SHIFT TURNOVER

1.

D.

1.

d. Control room watchstanders with the responsibility of the Operator at the Controls or the Control Room Command function shall conduct a face-to-face complete walkdown of the RTGBs and control panels with the individual assuming their responsibility.

# FLORIDA POWER & LIGHT COMPANY

## DOCKET NO. 50-335

## ST. LUCIE PLANT UNIT NO. 1

## FACILITY OPERATING LICENSE



1. The Nuclear Regulatory Commission (the Commission) having found that:

   A. The application for license filed by Florida Power & Light Company (the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commissions's rules and regulations set forth in 10 CFR Chapter 1 and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the St. Lucie Plant, Unit No. 1 (facility) has been substantially completed in conformity with Construction Permit No. CPPR-74 and the application, as amended, the provisions of the Act and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   E. The licensee is technically and financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

   F. The licensee has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this operating license will not be inimical to the common defense and security or to the health and safety of the public;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use at any time byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act, and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Sections 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below;

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2700 megawatts (thermal), provided that the construction items, preoperational tests, startup tests, and other items identified in Enclosure 1 to this license have been completed as specified in Enclosure 1. Enclosure 1 is an integral part of, and is hereby incorporated in this license.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 134 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2700 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHIELD BUILDING INTEGRITY

1.28 SHIELD BUILDING INTEGRITY shall exist when:

a. Each door is closed except when the access opening is being used for normal transit entry and exit;

b. The shield building ventilation system is in compliance with Specification 3.6.6.1, and

c. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.30 Site Boundary means that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.32 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.33 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.34 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.35 Unrestricted area means an area, access to which is neither limited nor controlled by the licensee.

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - $F_r$

1.36 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.

Power Level

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

August 28, 1980

NOTE TO: R. Tedesco
T. Novak
G. Lainas

I agree with E. Jordan's memo in that further debate on this issue is probably not warranted at this time. Please ensure that your staff is aware of this interpretation and that this will be the NRC position on this matter at this time.

Darrell G. Eisenhut

Enclosure

cc: E. Jordan
J. Scinto

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 2 2 1980

SSINS #0200

MEMORANDUM FOR: E. J. Brunner, Chief, RO&NSB, RI
R. C. Lewis, Acting Chief, RO&NSB, RII
R. F. Heishman, Chief, RO&NSB, RIII
G. L. Madsen, Chief, RO&NSB, RIV
J. L. Crews, Chief, RO&NSB, RV

FROM: E. L. Jordan, Assistant Director for Technical Programs
Division of Reactor Operations Inspection, IE

SUBJECT: DISCUSSION OF "LICENSED POWER LEVEL" (AITS F14580H2)

Dating back at least to 1974, there have been many lengthy "discussions" regarding the exact meaning of "full, steady-state licensed power level" (and similarly worded power limits). We do not believe the real safety benefits that might be derived from an NRC-wide agreement would be worth the further expenditure of manpower in meetings, etc. that would be required to achieve a consensus.

We do realize that some common uniform basis for enforcing maximum licensed power is needed by I&E inspectors. Therefore, until and unless an NRC-wide position is put forward and agreed upon (and as stated, I&E does not propose to initiate proceedings to that end), I&E will use the following guidance.

The average power level over any eight hour shift should not exceed the "full steady-state licensed power level" (and similarly worded terms). The exact eight hour periods defined as "shifts" are up to the plant, but should not be varied from day to day (the easiest definition is a normal shift manned by a particular "crew"). It is permissible to briefly exceed the "full, steady-state licensed power level" by as much as 2% for as long as 15 minutes. In no case should 102% power be exceeded, but lesser power "excursions" for longer periods should be allowed, with the above as guidance (i.e., 1% excess for 30 minutes, 1/2% for one hour, etc., should be allowed). There are no limits on the number of times these "excursions" may occur, or the time interval that must seperate such "excursions," except note that the above requirement regarding the eight hour average power will prevent abuse of this allowance.

CONTACT: H. W. Woods, IE
49-28180

~~8010160407~~ XA 2/1

8

TOTAL P.03



AUG 22 1980

The above is considered to be within the licensing basis and, therefore, acceptable to us, and it is also fair to the utilities and their ratepayers.

E. L. Jordan, Assistant Director for
Technical Programs
Division of Reactor Operations Inspection
Office of Inspection and Enforcement

cc: R. C. DeYoung, IE
S. E. Bryan, IE
D. Eisenhut, NRR
D. Ross, NRR

## 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.

Auto

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

None of the alarms came in because just into charging pump (10 Vol

c) Makeup flow indication and alarms

d) Vol\ 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.