UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

April 29, 2003

Florida Power and Light Company
ATTN: Mr. J. A. Stall, Senior Vice President
Nuclear and Chief Nuclear Officer
P. O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: TURKEY POINT NUCLEAR PLANT - NRC INSPECTION REPORT 50-250/03-07 AND 50-251/03-07

Dear Mr. Stall:

On April 14, 2003, the Nuclear Regulatory Commission (NRC) completed a follow-up to a safety system design and performance capability inspection at your Turkey Point Nuclear Plant. The enclosed report documents the inspection findings which were discussed with Mr. B. Jefferson and other members of your staff on February 21, 2003, and with Mr. T. Jones and other member of your staff on April 14, 2003.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your license. The inspectors reviewed selected procedures and records and interviewed personnel.

This report documents two NRC identified findings, both of which were determined by the NRC to be violations. The first violation was evaluated under the risk significance determination process as having very low safety significance (Green). The second finding was evaluated using traditional enforcement and determined to be a Severity Level IV violation. This violation was also determined by NRC management to be of very low safety significance (Green). Because these findings are of very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations, consistent with Section VI.A of the NRC Enforcement Policy. If you contest either of these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for you denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at Turkey Point.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document

Room or from the Publically Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

**/RA/**

Charles R. Ogle, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure: NRC Inspection Report 50-250/03-07,
50-251/03-07 w/Attachment

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| NAME | ROGERS | MUNDAY | OGLE | | | | |
| DATE | **4/25/2003** | **4/28/2003** | **'4./29/2003** | 4/ /2003 | 4/ /2003 | 4/ /2003 | 4/ /2003 |
| E-MAIL COPY? | YES NO | YES NO | YES NO | YES NO | YES NO | YES NO | YES NO |
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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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|---|---|
| Docket Nos.: | 50-250, 50-251 |
| License Nos.: | DPR-31, DPR-41 |
| Report Nos.: | 50-250/03-07, 50-251/03-07 |
| Licensee: | Florida Power & Light Company (FPL) |
| Facility: | Turkey Point Nuclear Plant, Units 3 & 4 |
| Location: | 9760 S. W. 344$^{th}$ Street<br>Florida City, FL 33035 |
| Dates: | February 18 - April 14, 2003 |
| Inspectors: | Walter G. Rogers, Senior Reactor Analyst |
| Approved By: | Charles R. Ogle, Chief<br>Engineering Branch 1<br>Division of Reactor Safety |

Enclosure

# SUMMARY OF FINDINGS

IR 05000250/2003-007, IR 05000251/2003-007; Florida Power and Light; onsite 2/18/03 - 2/21/03 and 2/24/03 - 4/14/03 in-office; Turkey Point Units 3 and 4; Section 4, Other Activities.

This safety system design and performance capability follow-up inspection was conducted by regional inspectors. Two Green non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. Inspector Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. Several emergency operating procedures developed by the licensee for mitigating a station blackout contained procedural inadequacies. These inadequacies included lack of appropriate acceptance criteria as well as improper use of and inadequate caution statements. Six examples were identified by the inspectors.

  Six examples of a non-cited violation of Technical Specification 6.8.b, Procedures and Programs and licensee Administrative Procedure 0-ADM-101, Procedure Writer’s Guide, were identified. This violation was more than minor because if left uncorrected, it could become a more significant safety concern. The finding is of very low safety significance because only one initiating event (loss of offsite power) was involved, three of four available emergency diesel generators (EDGs) would have to fail to get to this condition; the low probability that, given three EDGs fail, the fourth would operate properly; the possibility that offsite power would be restored prior to core damage; and there was a possibility that operators would be able to recover from the performance deficiency without overloading the EDG, due to operator training on the limitations of the EDGs. (Section 4OA5)

- Green. The licensee failed to update the Final Safety Analysis Report (FSAR) regarding their method of coping with potential reactor coolant system losses during a station blackout by reestablishing reactor coolant pump seal injection.

  A non-cited violation of 10 CFR 50.71(e) was identified. This violation is subject to traditional enforcement since it had the potential for impacting the regulatory process. Specifically, the NRC relies on the licensees to update FSARs to reflect the latest information developed for the facility. This ensures that the NRC has an accurate description of the facility when conducting inspections and evaluating license amendments. The finding is of very low safety significance because not updating the FSAR did not have any actual safety consequences. (Section 4OA5)

B. Licensee Identified Findings

None.

<u>Report Details</u>

## 1. REACTOR SAFETY

**Cornerstones: Initiating Events, Mitigating Systems**

1R21 <u>Safety System Design and Performance Capability (71111.21)</u>

a. <u>Inspection Scope</u>

During the previous safety system design and performance capability (SSDPC) inspection, the NRC team identified four unresolved items (URIs) associated with the licensee's compliance with the requirements of 10 CFR 50.63, Loss of All Alternating Power. (NRC Inspection Report 50-250, 251/02-06, ADAMS Accession No. ML030080300) The inspector reviewed these four items through review of applicable licensee corrective action documents, interviews with cognizant operations and electrical engineering staff, and independent review of existing emergency operating procedures (EOPs).

b. <u>Findings</u>

The specific findings are discussed under each individual URI in Section 4OA5 of this inspection report.

## 4. OTHER ACTIVITIES

4OA5 <u>Other Activities</u>

1. <u>(Open) URI 50-250,251/02-06-01</u>: Adequacy of SBO Strategy/Analysis and Loss of AC Power EOPs

This unresolved item remains open pending NRC review of the licensee's revised station blackout (SBO) analysis.

During the previous SSDPC inspection, the NRC team was unable to verify that changes made to the EOPs did not adversely impact the licensee's ability to mitigate an SBO for the required duration. Specifically, the team observed that the licensee's EOPs were revised in 1998 to delete restoration of reactor coolant pump (RCP) seal cooling upon regaining power from the alternate alternating current (AAC) emergency diesel generator (EDG) and instead, directed a reactor coolant system (RCS) cooldown and depressurization. The team noted that previous licensee correspondence associated with the SBO Rule stated that upon restoring electrical power using the AAC EDG, RCP seal cooling would be restored and the plant would remain at hot standby. In addition, the original NRC issued safety evaluation report indicated that the RCS inventory was sufficient to cope with a station blackout provided no RCS shrinkage occurred. The inspectors were concerned that the revised mitigation strategy provided by the EOPs could impact both these statements. In response, the licensee initiated Condition Report (CR) 02-2224. Under this CR an analysis was performed to determine if the present EOP mitigation strategy would successfully mitigate an SBO consistent with the present EOP mitigation strategy.

Licensee calculation, PTN-ENG-SENS-02-065, Engineering Evaluation to Demonstrate Compliance with Station Blackout Requirements, Revision 0, dated 2/10/03 was reviewed by the inspectors. The evaluation involved two scenarios. The first scenario assumed a controlled plant cooldown using atmospheric dump valves, one charging pump, and additional RCS inventory addition through the cold leg accumulators. The RCS end state after 8 hours was approximately 380 degrees Fahrenheit (°F) and 390 pounds per square inch gauge (psig). During the 8 hours covered by this analysis, core cooling was maintained but, pressurizer level was lost and voiding in the reactor pressure vessel head occurred. The second scenario contained in the calculation did not involve an RCS cooldown but instead maintained the plant in hot standby. (This represented a more significant challenge to containment.) Again, this analysis demonstrated satisfactory RCS performance and containment parameters remained below design limits.

The inspectors noted that there was no traceable standard by which the inspectors could determine whether the engineering evaluation had included all requisite assumptions and parameters. However, the licensee provided the inspectors with a comparable analysis (same computer code used) that was accepted by the Office of Nuclear Reactor Regulation technical reviewers during the original acceptance of compliance with 10 CFR 50.63 for the St. Lucie plant, thereby implying this would be an acceptable methodology. However, further review by NRC technical personnel will be necessary to confirm that the licensee's coping analysis was technically adequate. This matter will remain open pending this review.

2. <u>(Closed) URI 50-250,251/02-06-02:</u> Adequacy of 10 CFR 50.59 Reviews Associated With EOP Changes Concerning SBO

<u>Introduction:</u> A Green NCV was identified for failure to comply with 10 CFR 50.71(e) requirements to periodically update the Final Safety Analysis Report.

<u>Description:</u> During the previous SSDPC inspection, the team observed that EOP changes in 1998, modifying the SBO mitigation strategy, did not acknowledge the revisions as changes to the facility or procedures as described in the Updated Final Safety Analysis Report (UFSAR). The team considered the matter unresolved, pending further NRC review.

In response, the licensee initiated CR 03-0007. Under this CR, the licensee performed a root cause evaluation and extent of condition review. The licensee's analysis attributed this condition to their failure to incorporate into the FSAR their method of coping with potential RCS inventory losses during an SBO by re-establishing RCP seal injection. Although, this approach was committed to by the licensee during communications associated with their SBO submittals in the 1991 timeframe and captured in the associated NRC SER it was not incorporated into the FSAR by the licensee during subsequent revisions. As a result, the appropriate information was not available when the FSAR review was performed in 1998. Also, based upon a review to determine if information was properly incorporated into the UFSAR from 75 other SERs, the licensee determined that this was an isolated case. Based upon a review of this CR and licensee interviews, the inspector came to the same conclusion as the licensee.

Analysis: This violation was subject to traditional enforcement since it had the potential to impact the regulatory process. Specifically, the NRC relies on licensees to periodically update FSARs to reflect the latest information developed for the facility. This ensures that the NRC has an accurate description of the facility when conducting inspections and evaluating license amendments. The NRC characterized the violation at a Severity Level IV since the failure to update the UFSAR did not impede or influence regulatory action related to changes made to the facility or the NRC's review of proposed license amendments. The violation was also determined to be of very low safety significance (Green) since there were no actual safety consequences.

Enforcement: 10 CFR 50.71(e) requires the licensee to periodically update the FSAR to reflect the latest information developed. This includes information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirements since the original FSAR or latest update. Contrary to this requirement, the licensee failed to update the FSAR regarding their method of coping with potential RCS inventory losses during an SBO by reestablishing RCP seal injection. The approach was committed to by the licensee during communications associated with their SBO submittals in 1991 and captured in the NRC SER dated August 5, 1991. This Severity Level IV violation is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR 03-0007. This violation was also evaluated by NRC management as having very low safety significance (Green) since there were no actual safety consequences. It is identified as NCV 50-250,251/03-07-01, Failure to Update UFSAR with SBO Mitigation Information.

3. (Closed) URI 50-250,251/02-06-03: Adequacy of Procedure Guidance and Training for SBO Mitigation

Introduction: A Green NCV was identified involving multiple examples of procedural inadequacies in the procedures used to mitigate an SBO.

Description: During the previous SSDPC inspection, the team was unable to confirm that the operators would be able to adequately manage the limited capacity of the AAC EDG in all SBO scenarios. Further, it was unclear to the inspectors whether the procedures provided the necessary guidance to operators as to which equipment to load onto the AAC EDG. This matter was considered unresolved, pending additional inspection.

During this inspection, the inspector postulated plant conditions consistent with those that the licensee was required to mitigate as required by the SBO rule. Specifically, Units 3 and 4 were assumed to have experienced a non-recoverable loss of offsite power (LOSP) with EDG 3A providing onsite power to both units and secondary side heat removal operating on both units. In addition, it was postulated that, Unit 4 would have a constant 100 gallon per minute (gpm) RCS leakrate. [For ease of nomenclature, Unit 4 will be referred to as the SBO unit and Unit 3 the non-affected unit.] From this postulated scenario, the applicable EOPs that would be utilized by the licensee were reviewed. The review was to determine whether generic and site-specific technical information had been incorporated into the procedures, whether that information had been included using appropriate human factors considerations as required by the writer's guide, and to determine if the procedures were written consistent with applicable quality assurance program requirements. This review was conducted considering the requirements established in the following references:

- Technical Specification 6.8.b, "Procedures and Programs." This technical specification requires that emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33 would be established, implemented, and maintained.

- An NRC Order modifying the Turkey Point Operating License dated February 23, 1984. This order required that the licensee implement the requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements," and Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability," criterion I.C.1., "Guidance for the Evaluation and Development of Procedures for Transients and Accidents."

- 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," as implemented through the licensee's Quality Assurance Program Topical Report.

- Technical Specification 6.8.a, "Procedures and Programs," which captures the activities of Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 and, Sections 5.1 and 5.3 of ANSI N18.7-1972; thereby requiring procedures be established, implemented, and maintained.

- 10 CFR 50.63, Loss of All Alternating Current Power, and Regulatory Guide 1.155, August 1988, Station Blackout.

The results of the review were:

a. Initially, the controlling EOP for the SBO Unit was 4-EOP-ECA-0.0, Loss of All AC Power. At the response not obtained (RNO) for Step 10 of 4-EOP-ECA-0.0, operators were directed to restore power to the SBO unit by branching into procedure 4-ONOP-004.2, Loss of 4A 4KV Bus. From the procedure actions in 4-ONOP-004.2 operators were to establish the electrical configuration to re-energize SBO unit buses from a non-affected unit's EDG. Prior to energizing the Unit 4 safety buses from EDG 3A, operators were directed to reduce electrical loads on Unit 3. This activity was accomplished in RNO sub-step (a) of Step 13 in procedure 4-ONOP-004.2. The procedure specifically stated:

> *Request that the Unit 3 Reactor Control Operator (RCO) perform the following:*
>
> *a. On the Unit 3 4KV bus which is supplying 3D 4KV bus, place the following in PULL-TO-LOCK or OFF:*

- *Non-running safeguards equipment*
- *Non-essential loads*

This language was not precise and did not contain appropriate quantitative or qualitative acceptance criteria for determining that an important activity had been satisfactorily accomplished. Specifically, there was no information in this step indicating what “non-essential loads” were to be turned off or placed in pull-to-lock. This same inadequately defined action existed in companion procedures 4-ONOP-04.3, Loss of 4B 4KV Bus, at the RNO of Step 12; 3-ONOP-04.3, Loss of 3B 4KV Bus, at the RNO of Step 12; and 3-ONOP-04.2, Loss of 3A 4KV Bus, at the RNO of Step 13.

Administrative Procedure, 0-ADM-101, Procedure Writer’s Guide, Section 5.5.2.1.a, required steps be written in short and precise language that present exactly the task which the individual is to perform. [The inspectors observed that other references also required more precise acceptance criteria for this step number. Specifically, ANSI N18.7-1972, Administrative Controls for Nuclear Power Plants, Section 5.3.1, Procedure Scope, required in part that, “Each procedure shall be sufficiently detailed for a qualified individual to perform the required function without direct supervision....” Likewise, 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures and Drawings,” required in part that “Activities affecting quality shall be prescribed by documented ...procedures. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.” Finally, FPL Topical Quality Assurance Report, section 5.2.4, required in part that “Quality instructions shall require that procedures affecting quality include adequate quantitative and qualitative acceptance criteria. These acceptance criteria requirements apply to ... operations.”]

The inspectors noted that the site specific technical information to ensure appropriate loading of the EDG 3A was contained in Drawing 5613-E-1712, Emergency Diesel Generator 3A Station Blackout Load List. However, as discussed above, this information was not translated precisely into the procedure steps specified above.

The failure to establish appropriate acceptance criteria for EDG loading in the SBO EOP steps described above, is a violation of Administrative Procedure, 0-ADM-101. This is the first example of a violation for inadequate procedures.

b. With the completion of Step 14 in ONOP-004.2, operators were directed to proceed to Step 32 of Procedure 4ECP-ECA-0.0, Loss of all AC Power, by the caution immediately prior to Step 11 in Procedure 4-EOP-ECA-0.0. This activity would occur prior to placing any loads on the newly energized Unit 4 safety buses. The caution specifically stated:

*C A U T I O N*

*When power is restored to the A or B 4KV bus, recovery actions should continue with CAUTIONS prior to Step 32 and then Step 32.*

No procedure step specifically directed operators to transition to Step 32 of Procedure 4-ECP-ECA-0.0. The use of this caution as a transition was confirmed by the inspectors during interviews of qualified operations personnel. This same direction would also be applicable to the caution prior to Step 11 in the companion procedure, 3-EOP-ECA-0.0.

Administrative Procedure, 0-ADM-101, Procedure Writer's Guide, Section 5.5.12.9.c, states that "Notes and cautions may advise in a passive voice regarding actions or transitions which may become necessary due to changes in plant conditions as long as the same information is contained within the steps of the procedure."

The inappropriate use of a caution statement to solely direct operator actions is a violation of Administrative Procedure, 0-ADM-101. This is an additional example of a violation for inadequate procedures.

c. The RNO at Step 35 for Procedure 4-EOP-ECA-0.0 instructed operators to power the Unit 4 load centers from the newly energized 4KV safety buses with EDG 3A as the power source. However, the caution prior to this step established site-specific starting and steady-state maximum electrical loadings for the Unit 4 EDGs without mentioning the Unit 3 EDGs. No specific caution was provided to protect the Unit 3 EDGs from damage.

The Westinghouse Emergency Response Guidelines, Revision 1C, constituted the generic technical information for this facility. The caution prior to Step 25 of ECA-0.0, Loss of All AC Power, in this generic technical information stated:

*"The loads placed on the energized ac emergency bus should not exceed the capacity of the power source"*

This generic technical information was not appropriately translated by the licensee into Procedure 4-EOP-ECA-0.0. Failure to establish appropriate cautions to prevent overloading the EDG used as the power source as described above is contrary to the requirements of Technical Specification 6.8.b, "Procedures and Programs." This is identified as an additional example of a violation for inadequate procedures.

d. On the non-affected unit, Unit 3, operators would initially respond to the loss of offsite power with Procedure 3-EOP-ES-0.1, Reactor Trip Response, as the controlling procedure. At RNO sub-step (b) for Step 5, (f) for Step 9, and (d) for Step 10, operators were directed by that procedure to shed non-essential loads using Attachment 2, "UNIT 3 COMPONENT KW LOAD RATING CHART," if EDG maximum loading would be exceeded. Included in the list of non-essential loads contained in Attachment 2 were the terms charging pump and pressurizer heaters.

The inspectors noted that Drawing 5613-E-1712, Emergency Diesel Generator 3A Station Blackout Load List, indicated that a charging pump and 150 KW of pressurizer heaters were used to maintain the unit in safe shutdown (the companion drawings would be 5613-E-1713 for EDG 3B, 5614-E-1712 for EDG 4A and 5614-E-1713 for EDG 4B). Pressurizer heaters and a charging pump help ensure that a steam bubble is maintained in the pressurizer, that the pressurizer is not voided, and that subcooling margin is maintained. The inspectors observed that these conditions were critical to not degrading the normal safe shutdown capability of the non-affected unit. Based on review of Regulatory Guide 1.155, Position C.3.3.5 and NUMARC 87-00 (including December 27, 1989, supplemental questions and answers), required that the normal safe shutdown capability of the non-affected unit not be degraded for compliance with 10 CFR 50.63.

The inspectors, concluded that not properly translating the EDG blackout load list contained in Drawing 5613-E-1712 into Attachment 2 of Procedure 3-EOP-ES-0.1 had the potential to degrade the normal safe shutdown ability of the non-affected unit and was contrary to the requirements of Technical Specification 6.8.b, “Procedures and Programs.” This is the fourth example of a violation for inadequate procedures.

e. Upon completing Procedure 4-EOP-ECA-0.0, operators would proceed into either Procedure 4-EOP-ECA-0.1, Loss of All AC Power Recovery Without SI Required, or 4-EOP-ECA-0.2, Loss of All AC Power Recovery With SI Required. Depending upon the length of time necessary to reach this point in the event mitigation strategy, either procedure could be selected by operators.

Assuming Procedure 4-EOP-ECA-0.1 was selected, operators would begin to re-energize large electrical loads connected to Unit 4's safety related Bus 4A (powered from Unit 3's EDG 3A) at Step 4. However, the caution prior to Step 4 established site-specific starting and steady-state maximum electrical loadings for a Unit 4 EDG only. This caution resulted from a caution contained in generic technical information associated with this potential event. Specifically, the caution prior to Step 3 in the generic ECA-0.1, Loss of All AC Power Recovery Without SI Required stated:

> *“The loads placed on the energized ac emergency bus should not exceed the capacity of the power source”*

The licensee inadequately translated this caution into the EOPs as a caution prior to Step 4 in Procedure 4-EOP-ECA-0.1 by not including any discussion of the Unit 3 EDGs. Without a caution discussing the Unit 3 EDGs prior to Step 4, the inspectors were concerned that no procedural controls existed to prohibit operators from overloading the operating EDG with the equipment being placed into service at Step 4. In addition, the RNO of Step 5 referenced an Attachment 1, Unit 4 COMPONENT KW LOAD RATING CHART, for component

KW load ratings when additional electrical loads were to be applied to the EDG later in Step 5. Again, the caution at the top of this attachment only referenced electrical controls for the Unit 4 EDGs. Also, these same inadequacies existed at the same place in the Unit 3 companion procedure, 3-EOP-ECA-0.1, Loss of All AC Power Recovery Without SI Required.

The inspectors concluded that not properly including cautions to prevent overloading the appropriate EDG, in the procedures described above, was contrary to the requirements of T.S. 6.8.b, “Procedures and Programs.” This was identified as the fifth example of an inadequate procedure violation.

f. Assuming operators transition to Procedure 4-EOP-ECA-0.2, Loss of All AC Power Recovery With SI Required, upon completion of 4-EOP-ECA-0.0, operators would begin to re-energize large electrical loads connected to Unit 4's safety related bus 4A (powered from Unit 3's EDG 3A) at Step 5. However, the caution prior to Step 5 established site-specific starting and steady-state maximum electrical loadings for a Unit 4 EDG only. This caution resulted from a caution contained in generic technical information associated with this event. Specifically, the caution prior to Step 5 in the generic ECA-0.2, Loss of All AC Power Recovery With SI Required stated:

*“The loads placed on the energized ac emergency bus should not exceed the capacity of the power source”*

The licensee inadequately translated this caution into the EOPs as a caution prior to Step 5 in Procedure 4-EOP-ECA-0.2 by not including any discussion of the Unit 3 EDGs. Without a caution discussing the Unit 3 EDG prior to Step 5, the inspectors were concerned that no procedural controls existed to prohibit operators from overloading the operating EDG with the equipment being placed into service at Step 5.

The inspectors concluded that not properly including cautions to prevent overloading the appropriate EDG, in the procedures described above, was contrary to the requirements of TS 6.8.b, “Procedures and Programs.” This is the sixth example of an inadequate procedure violation.

The inspectors also observed that the licensee’s EOP validation/verification program required as part of the February 23, 1983, NRC Order failed to identify the inadequate procedure steps/cautions identified above. This was considered a missed opportunity to identify these discrepancies.

Analysis: Collectively, the violation examples in sections a. - f. above constituted a performance deficiency. The performance deficiency was “The EOPs used to mitigate a station blackout contained numerous procedural inadequacies.”

This performance deficiency was more than minor because if left uncorrected, it could become a more significant safety concern. Under Significance Determination Process (SDP) Phase 1, the deficiency was associated with mitigating systems and represented a potential loss of safety function. Due to the nature of the deficiency, a Phase 3 analysis was performed by a senior reactor analyst. The performance deficiency was determined to be of very low safety significance (Green). The major factors leading to the significance determination were:

- Only one initiating event, loss of offsite power, was associated with the performance deficiency.
- Redundancy in the number of EDGs, with four onsite and, only one required for successful mitigation.
- The low probability that, given three EDGs fail, the fourth would operate properly.
- The probability that offsite power would be restored prior to core damage.
- Due to operator training and experience on the limitations of the EDGs, there was a reasonable probability that operators would not execute the procedure(s) as written and would be able to recover from the performance deficiency without overloading the EDG. Also, in a number of the possible accident sequences there would be ample time for operators to diagnosis the situation and establish adequate accident mitigation actions.

Enforcement: Both Administrative Procedure O-ADM-101 and T.S. 6.8.b, “Procedure and Programs,” require that EOPs used to mitigate the consequences of a SBO be adequate. Contrary to the above, on April 14, 2003, the inspectors observed inadequacies in the procedures used to mitigate a SBO as described in Sections a. although f. above. Because this violation is of very low safety significance and has been entered into the licensee’s corrective action program (CR 02-2224), it is being treated as an NCV, consisted with Section VI.A of the NRC Enforcement policy. This is identified as NCV 50-250,251/03-07-02, Inadequacies in SBO Mitigation Procedures (Six Examples). URI 50-250,251/02-06-03: Adequacy of Procedure Guidance and Training for SBO Mitigation, is closed.

4. (Closed) URI 50-250,251/02-06-04: Acceptability of Reactor Coolant Pump High Temperature O-rings Having a Different Material Hardness

During the previous SSDPC inspection, the team identified an unresolved item associated with the licensee’s evaluation of the adequacy and acceptability of replacement O-rings in reactor coolant pump seal packages. Specifically, the licensee had approved and installed RCP seal kits (including O-rings) from Framatome Technologies Incorporated (FTI) with O-rings of a harder material than in the Westinghouse specifications. The team questioned the impact of using the harder material O-rings on the licensee’s SBO analysis. In response, the licensee initiated CR 02-2151 and performed IEE 072667. The licensee concluded that the FTI seal kit testing bounded the Westinghouse testing and the seal kits were equivalent. The inspectors reviewed the licensee’s analysis and considered it adequate.

The inspectors did not identify any explicit regulatory requirements associated with RCP seals or O-rings and these components were not classified as safety related. Also, the inspector did not identify any technical discrepancies in the FTI test documentation that would void the licensee's conclusion of equivalency. Hence, no violation of regulatory requirements was identified. Therefore, this item is closed.

4OA6 Management Meetings

The inspector presented the results of the inspection to Mr. B. Jefferson and other members of the licensee's staff at an interim exit meeting on February 21, 2003, who acknowledged all but one of the findings. Following additional in-office review of the licensee's procedures, the inspector presented the final results of the inspection to T. Jones, and other members of the licensee's staff in a final exit held on April 14, 2003. Proprietary information is not included in this inspection report.

## SUPPLEMENTAL INFORMATION

### KEY POINTS OF CONTACT

Licensee Personnel

M. Guth, Electrical Engineer
B. Jefferson, Acting Site Vice President
T. Jones, Plant Manager/Site Vice President
M. Lacal, Operations Manager
W. Parker, Licensing Manager
B. Stamp, Operations Supervisor
A. Zielonka, Engineering Manager

NRC Personnel

K. Green-Bates, Resident Inspector, Turkey Point Nuclear Plant
C. Patterson, Senior Resident Inspector, Turkey Point Nuclear Plant

### LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

| | | |
|---|---|---|
| 50-250,251/03-07-01 | NCV | Failure to Update UFSAR with SBO Mitigation Information (Section 4OA5.2) |
| 50-250,251/03-07-02 | NCV | Inadequacies in SBO Mitigation Procedures (Six Examples) (Section 4OA5.3) |
| 50-250,251/02-06-02 | URI | Adequacy of 10 CFR 50.59 Reviews Associated With EOP Changes Concerning SBO (Section 4OA5.2) |
| 50-250,251/02-06-03 | URI | Adequacy of Procedure Guidance and Training for SBO Mitigation (Section 4OA5.3) |
| 50-250,251/02-06-04 | URI | Acceptability of Reactor Coolant Pump High Temperature O-rings Having a Different Material Hardness (Section 4OA5.4) |

Discussed

| | | |
|---|---|---|
| 50-250,251/02-06-01 | URI | Adequacy of SBO Strategy/Analysis and Loss of AC Power EOPs (Section 4OA5) |

## APPENDIX

## LIST OF DOCUMENTS REVIEWED

### Procedures

4-EOP-ECA-0.0, Unit 4 Loss of ALL AC Power, dated 12/18/02
3-EOP-ECA-0.0, Unit 3 Loss of ALL AC Power, dated 4/30/02
3(4)-EOP-ECA-0.1, Loss of All AC Power Recovery Without SI Required, dated 4/30/02
3(4)-EOP-ECA-0.2, Unit 3 Loss of All AC Power Recovery With SI Required, dated 2/22/02
ECA-0.0, 0.1 and 0.2, Westinghouse Owners Group Emergency Response Guidelines
3(4)-ONOP-004.2, Loss of 3A 4KV Bus, dated 10/16/01
3(4)-ON0P-004.3, Loss of 3B 4KV Bus, dated 10/16/01
3(4)-EOP-ES-0.1, Reactor Trip Response dated 4/30/02

### Drawings

5613-E-1712, Emergency Diesel Generator 3A Station Blackout, Rev. 7
5614-E-1712, Emergency Diesel Generator 4A Station Blackout, Rev. 6
5613-E-1713, Emergency Diesel Generator 3B Station Blackout, Rev. 3
5614-E-1713, Emergency Diesel Generator 4B Station Blackout, Rev. 3

### Condition Reports

CR 02-2151, Framatome high temperature RCP O-rings are not an identical replacement for the Westinghouse high temperature RCP O-rings
CR 02-2224, Discrepancy between current mitigation strategies and original regulatory responses for SBO

### Miscellaneous Documents

FP&L Letter L-89-144, dated April 17, 1989, Subject: Information to Resolve Station Blackout
FP&L Letter L-90-56, dated March 29, 1990, Subject: Information to Resolve Station Blackout
FP&L Letter L-90-338, dated September 21, 1990, Subject: Comments on NRC's Safety Evaluation for Station Blackout
FP&L Letter L-91-136, dated May 14, 1991, Subject: Additional Information for Station Blackout
NRC Safety Evaluation Report dated June 15, 1990, Safety Evaluation for Proposed Implementation of the Station Blackout Rule (10 CFR 50.63)
NRC Safety Evaluation Report dated July 31, 1991, Supplemental Safety Evaluation for the Proposed Implementation of the Station Blackout Rule (10 CFR 50.63)
10 CFR 50.59 Safety Review for Procedure 3-EOP-ECA-0.1, Unit 3 Loss of All AC Power Recovery Without SI Required, dated 6/3/98
10 CFR 50.59 Safety Review for Procedure 3-EOP-ECA-0.2, Unit 3 Loss of All AC Power Recovery With SI Required, dated 6/3/98
Jeumont Industry Pressurized Water Reactor Coolant Pump High Temperature O-Ring for Shaft Seals Qualification Report dated 4/3/94