UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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June 22, 2006

James J. Sheppard, President and
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SUBJECT: ERRATA FOR SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000498/2006002 AND 05000499/2006002

Dear Mr. Sheppard:

Please replace the Inspection Scope in Section 4OA5, "Implementation of Temporary Instruction (TI) 2525/165 - Operational Readiness of Offsite Power and Impact on Plant Risk" page 27 of the Report Details in NRC Inspection Report 05000498/2006002 and 05000499/2006002, dated May 18, 2006, with the enclosed revised change. This change is necessary to revise the numbering of the Temporary Instruction and the Inspection Scope template that was reviewed during the inspection period.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly 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).

Should you have any questions concerning this inspection update, we will be pleased to discuss them with you.

Sincerely,

/RA/

Claude E Johnson, Chief
Project Branch A
Division of Reactor Projects

Docket Nos.: 50-498
50-499
License Nos.: NPF-76
NPF-80

Enclosure:
Page 27 of NRC Inspection Report 05000498/2006002 and 05000499/2006002

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SUNSI Review Completed: __CEJ_ ADAMS: √ Yes G No Initials: _CEJ__
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4OA5 Other

.1 Temporary Instruction 2515/165, Operational Readiness of Offsite Power

a. Inspection Scope

The objective of TI 2515/165, "Operational Readiness of Offsite Power and Impact on Plant Risk," was to confirm, through inspections and interviews, the operational readiness of offsite power systems in accordance with NRC requirements. On March 6 -17, 2006, the inspectors reviewed licensee procedures and discussed the attributes identified in TI 2515/165 with licensee personnel. In accordance with the requirements of TI 2515/165, the inspectors evaluated the licensee's operating procedures used to assure the functionality/operability of the offsite power system, as well as, the risk assessment, emergent work, and/or grid reliability procedures used to assess the operability and readiness of the offsite power system.

The information gathered while completing this Temporary Instruction was forwarded to the Office of Nuclear Reactor Regulation for further review and evaluation.

- OPOP04-AE-0005, "Offsite Power System Degraded Voltage," Revision 0.
- Technical Specification Limiting Condition for Operation Action 3.8.1.1.e
- OPOP01-ZO-0006, "Extended Allowed Outage Time," Revision 13
- OPOP01-ZG-002, "STP Coordinator," Revision 1
- OPOP04-AE-0003, "Loss of Power to one or more 13.8kV Standby Bus," Revision 6
- OPOP04-AE-0004, "Loss of Power to one or more 4160 ESF Bus," Revision 10
- ERCOT Operating Guide Section 2.10, "System Voltage Profile," May 1, 2005
- ERCOT Operating Guide Section 4, "Emergency Operation," September 1, 2004
- NRR Safety Evaluation of Revised Blackout Position, dated July 24, 1995
- OPGP03-ZO-0045, "CenterPoint Energy Real Time Operations Emergency Operations Plan," Revision 1
- South Texas Project Interconnection Agreement (Reliant Energy, CP&L, San Antonio, and Austin /STP), dated August 15, 2002

b. Findings

No findings of significance were identified.

.2 <u>(Closed) URI 05000498;499/2005006-02</u>: Inadequate Motor-Operated Valve Operation Method

<u>Introduction</u>: The inspectors identified an NCV of 10 CFR Part 50, Appendix R, Section III.L.3, in that the method used to position motor-operated valves ("hot-sticking") following a fire in the control room was not independent of the fire area. Specifically, a portion of each valve control circuit was located in the control room. A fire affecting those circuits could result in maloperation or overthrusting of the valves.

<u>Description</u>: The licensee utilized a motor-operated valve repositioning method called "hot-sticking." Operators repositioned or checked the position of a given valve by pushing in either the open or close contactor at the motor control center breaker for the valve. Operators were trained that a contactor will "suck in" if the valve is out of its required position. Then the valve will travel to the requested position and the contactor will "pop out." For valves in the required position, the contactor will immediately "pop out." The inspectors identified a concern with this method because the method utilizes circuits that are not independent of the fire area. If the control room circuit fails or hot shorts, the method will not work as intended. The reliance on circuits that are within the fire area is not consistent with 10 CFR Part 50, Appendix R, Section III.L.3, which states, in part, ". . . the alternative shutdown capability shall be independent of the specific fire area(s)." The failure modes are described below:

1. **Open Circuit**: A control circuit failure (open circuit) could result in improper indication to operators when hot-sticking valves. In this instance, following a hot-sticking attempt, the MOV contractor would immediately "pop out." Operators were trained that this response indicates that: "the valve is already in the required position." In reality, the valve should be in the opposite position.

2. **Hot Short**: A hot short could cause a valve to start repositioning to an inappropriate position after operators had performed the hot-sticking method for the valve. Since an operator opens the breaker after repositioning the valve, the valve would still be able to reposition during the few seconds before the operator opened the breaker. Some of the valves had very short stroke times (about 10 seconds).

3. **Overthrust/Torque**: In the case where the necessary control room circuits are undamaged and a valve is in its required position, valves and actuators can be overthrust and overtorqued well in excess of manufacturers ratings. The hot-sticking method drives the valves into its seats with locked rotor torque. The licensee performed a detailed analysis to evaluate the impact on the MOVs. The licensee concluded that, while catastrophic valve/actuator failure is not expected, the stress to some valve components would exceed the yield point.

As an initial corrective action, the licensee trained the plant operators to ensure that they understood the vulnerabilities associated with the hot-sticking method. The licensee verified that adequate indication was available in all cases to ensure that maloperation of valves could be promptly identified and corrected.

Analysis: The failure to ensure that all circuits relied on for safe shutdown in response to a control room fire was free of the fire area was a performance deficiency. The issue was more than minor because it affected the reliability objective of the Equipment Performance attribute under the Mitigating Systems Cornerstone. Specifically, MOVs that are relied upon to achieve postfire safe shutdown were less reliable because parts of their control circuits could be damaged by the fire. A Senior Reactor Analyst evaluated the safety significance of this issue. The frequency of fires that require evacuation of the control room and use of the remote shutdown panels at South Texas Project are 7E-6/yr. In the base case, it is assumed that the conditional core damage probability of a control room evacuation is 0.1. That is, approximately 10 percent of control room evacuations are assumed to result in core damage. Therefore, to obtain a delta-core-damage-frequency increase of greater than 1E-6/yr, the conditional core damage probability of a control room evacuation, given the performance deficiency, would have to increase to at least 0.24.

The Senior Reactor Analyst noted that, in response to a control room fire, the licensee would attempt to recover all three trains of safe shutdown equipment rather than just the one credited train. Further, most valves were already in their required positions, which negated the need for repositioning. Finally, the licensee had valve position indication at the remote shutdown panel for all affected valves. Accordingly, the operators could quickly troubleshoot and address a mispositioned valve. Therefore, the Senior Reactor Analyst concluded that the increased risk from the hot-sticking method was insufficient to cause more than a very small change in the conditional core damage probability. Also, the implications of large early release would not be relevant to a risk increase in this instance. Considering the above, the Senior Reactor Analyst evaluated the safety significance of this finding using Manual Chapter 0609, “Significance Determination Process,” Appendix F, and determined that the finding constituted a low level of degradation for postfire safe shutdown equipment. As such, the finding was of very low safety significance.

Enforcement: 10 CFR Part 50, Appendix R, Section III.L.3, states, in part, ". . . the alternative shutdown capability shall be independent of the specific fire area(s).” Contrary to the above, approximately 25 MOVs utilized for mitigation of a fire in the control room, the licensee specified a valve repositioning method that relied upon circuits that were not independent of the fire area. Because this issue is of very low safety significance and has been entered into the corrective action program as CR 05-8004, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000498;499/2006002-04).

.3 (Closed) URI 05000498;499/2005006-03: Inadequate Fire Protection Alternate Shutdown Analysis

Introduction: The inspectors identified a noncited violation of 10 CFR Part 50, Appendix R, Section III.L.1, because the thermohydraulic analysis was inconsistent with actions allowed in the South Texas Project licensing basis for a control room evacuation. The inconsistencies affected the control room fire timeline that operators must meet in order to successfully accomplish safe shutdown. Specifically, the analysis inappropriately allowed four additional manual actions to be performed from the control room while the license basis allowed only one manual action to be performed prior to evacuating the control room. The other four manual actions are required to be performed in the field.

Description: The inspectors performed a walkdown with plant operators to verify that alternate shutdown actions could be performed within the time limits derived by the thermohydraulic analysis.

The inspectors identified that Calculation NC-7079, “Fire Hazards Analysis,” Revision 1 (thermohydraulic analysis), contained inappropriate assumptions. For alternate shutdown outside the control room (control room fire), Fire Hazards Analysis Report, Section 2.4.4, “Alternate Shutdown,” credits tripping the reactor from the control room and nothing more. Licensee Calculation NC-7079 assumed that the following additional actions would be accomplished prior to exiting the control room:

- • Isolating main steam isolation valves
- • Isolating feedwater
- • Securing charging
- • Isolating letdown

Since the licensee had not obtained NRC approval for the deviations to the licensing basis, crediting performance of the additional actions and using the timeline for completion of the actions from the control room was inappropriate. Further, if the licensee completed the actions outside the control room, there would be an impact on the thermohydraulic analysis results and the associated timeline. Specifically, performing these actions inside the control room ensured that the reactor coolant system process variables remained within those values predicted for a loss of normal ac power, as required by 10 CFR Part 50, Appendix R, Section III.L.1. For a loss of normal ac power at South Texas Project, the predicted pressurizer level (a reactor coolant system process variable) remains well within the indicating range.

The NRC Enforcement Manual, Section 8.1.7, states, in part: “Failure to have an adequate written evaluation available for an area in which Appendix R compliance is not apparent will be taken as an indication that the area does not comply with NRC requirements . . . .”

Analysis. The failure to have an adequate written evaluation available for a control room fire scenario was a performance deficiency. This issue was more than minor because it affected the Mitigating Systems Cornerstone attributes of protection from external factors (fire). The inadequate analysis overestimated the amount of time available when accomplishing shutdown actions and, during walkdowns, the inspectors could not verify compliance with the requirements. An NRC Senior Reactor Analyst evaluated this issue. The frequency of fires that require evacuation of the control room and use of the remote shutdown panels at South Texas Project is 7E-6/yr. In the base case, it is assumed that the conditional core damage probability of a control room evacuation is 0.1. That is, approximately 10 percent of control room evacuations are assumed to result in core damage. Therefore, to obtain a delta-core-damage-frequency increase of greater than 1E-6/yr., the conditional core damage probability of a control room evacuation, given the performance deficiency, would have to increase to at least approximately 0.24.

The Senior Reactor Analyst noted that the additional actions that operators would take prior to evacuation of the control room would cause a delay of approximately 90 seconds beyond the times assumed in the analyzed recovery action timeline. The Senior Reactor Analyst determined that a delay of this magnitude was insufficient to cause more than a very small change in the conditional core damage probability. Also, the implications of large early release would not be relevant to a risk increase in this instance. Based on the above, the Senior Reactor Analyst evaluated the safety significance of this finding using Manual Chapter 0609, "Significance Determination Process," Appendix F, and determined that the finding constituted a low level of degradation for postfire safe shutdown analysis. Accordingly, the finding was of very low safety significance.

Enforcement: 10 CFR Part 50, Appendix R, Section III.L.1, requires that reactor coolant system process variables be maintained within those predicted for a loss of normal ac power. License Condition 2.E specifies, "STPNOC shall implement and maintain in effect all provisions of the approved fire protection program as described in the . . . Fire Hazards Analysis Report." The Fire Hazards Analysis Report, Section 2.4.4, "Alternate Shutdown," credits tripping the reactor from the control room and nothing more. The licensee used Calculation NC-7079 to demonstrate compliance with 10 CFR Part 50, Appendix R, Section III.L.1. Contrary to the above, Calculation NC-7079 inappropriately credited several additional actions from the control room, including main steam isolation valve closure, feedwater isolation, securing charging, and isolating letdown. Because this issue is of very low safety significance and has been entered into the corrective action program as CR 05-8507, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000498;499/2006002-05).

4OA6 Meetings, Including Exit

The results of the ALARA inspection were presented to Mr. Gary Parkey, Vice President of Generation, and Mr. Thomas Jordan, Vice President of Engineering, and other members of the staff, on February 9, 2006.

On February 27, 2006, the inspector conducted a telephonic exit meeting to present the emergency preparedness inspection results to Mr. A. Morgan, Supervisor, Emergency Planning, who acknowledged the findings.

On April 10, 2006, the inspector conducted a telephonic exit meeting to present the electrical URI inspection results to Mr. K. Taplett, Licensing Staff Engineer, who acknowledged the findings.

The results of the resident inspection were presented to Mr. Gary Parkey, Vice President of Generation, and other members of licensee management on April 13, 2006.

During each exit meeting, the inspectors asked the licensee representatives whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

Technical Specification 3.5.2 requires, in part, that, with only two of three required essential cooling water loops operable, three loops be restored to operable within 7 days or be in at least hot standby within 6 hours. Contrary to this, Unit 2 continued to operate at 100 percent power while ECCS Train 2C was inoperable for an indeterminate time greater than 7 days due to missing T-drains in MOV Motor 2-SI-0019C. The licensee entered the performance deficiency into their corrective action program as CR 05-13606 for resolution. This finding is of very low safety significance because of the availability of two other trains.

ATTACHMENT: SUPPLEMENTAL INFORMATION