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

December 1, 2006

Virginia Electric and Power Company
ATTN.: Mr. David A. Christian
Sr. Vice President and
Chief Nuclear Officer
Innsbrook Technical Center - 2SW
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION - NRC SPECIAL INSPECTION REPORT 05000280/2006011 AND 05000281/2006011

Dear Mr. Christian:

On October 27, 2006, the United States Nuclear Regulatory Commission (NRC) completed a special inspection at your Surry Power Station, Units 1 and 2. The inspection reviewed the circumstances associated with the partial loss of offsite power to both units and the Unit 2 manual reactor trip event that occurred on October 7, 2006. In accordance with Management Directive 8.3, NRC Incident Investigation Program, a special inspection was warranted because the event involved significant unexpected system interactions and the estimated conditional core damage probability exceeded 1E-6. The enclosed special inspection report documents the inspection findings which were discussed on November 1, 2006, with Mr. D. Jernigan and other members of your staff.

The inspection was performed in accordance with Inspection Procedure 93812, "Special Inspection," and focused on the six areas discussed in the enclosed charter. The inspection examined activities conducted under your licenses as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your licenses. The team reviewed selected procedures and records, conducted field walkdowns, observed activities, and interviewed personnel.

Based upon the results of this inspection, the report documents three unresolved items of which the related evaluations are ongoing, and one self-revealing finding of very low safety significance (Green). This finding was determined to involve a violation of NRC requirements; however, because of its very low safety significance, and because it was entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. In addition, one licensee-identified violation, which was determined to be of very low safety significance (Green), is documented in this report. If you contest any non-cited violation in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Surry Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if any, 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 by Michael E. Ernstes acting for/

Charles A. Casto, Director
Division of Reactor Projects

Docket Nos.: 50-280, 50-281
License Nos.: DPR-32, DPR-37

Enclosures: NRC Special Inspection Report 05000280/2006011, 05000281/2006011
w/Attachments: 1. Supplemental Information
2. Special Inspection Team Charter

cc w/encls: (See page 3)

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Report to D. A. Christian from Charles A. Casto dated December 1, 2006

SUBJECT: SURRY POWER STATION - NRC SPECIAL INSPECTION REPORT 05000280/2006011 AND 05000281/2006011

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-280, 50-281
License Nos.:	DPR-32, DPR-37
Report Nos.:	05000280/2006011, 05000281/2006011
Licensee:	Virginia Electric and Power Company (VEPCO)
Facility:	Surry Power Station, Units 1 & 2
Location:	5850 Hog Island Road Surry, VA 23883
Dates:	October 16, 2006 through October 27, 2006
Inspectors:	J. Reece, Senior Resident Inspector L. Miller, Senior Reactor Inspector R. Rodriguez, Reactor Inspector
Approved by:	Charles A. Casto, Director Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000280/2006-011, IR 05000281/2006-011; 10/16/2006 - 10/27/2006; Surry Power Station, Units 1 & 2; Special Inspection Report.

The report documents special inspection activities conducted by a senior resident inspector, a senior reactor inspector and a reactor inspector to review a partial loss of offsite power to both units and a Unit 2 manual reactor trip. One NRC identified Green non-cited violation was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). 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. NRC-Identified and Self-Revealing Findings

Cornerstone: Emergency Preparedness

- Green. A self-revealing non-cited violation of 10 CFR 50.72(a)(4) was identified. During the October 7, 2006, partial loss of offsite power event, the licensee failed to activate the Emergency Response Data System (ERDS) within one hour of an Alert declaration. The ERDS was not made operable until approximately five and one-half hours after the Alert declaration due to an upgrade to the telephone exchange that had been done seven days prior to the event.

  The finding is more than minor due to its impact on the Emergency Preparedness cornerstone objective to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency and the related attribute of Emergency Response Organization response. The finding is of very low safety significance (Green) because it involved a "failure to implement" (in distinction to a "failure to meet") an NRC emergency planning standard. The cause of the finding is related to the cross-cutting area of human performance, in that, the licensee failed to reprogram the telephone exchange following a telephone system change which occurred prior to the event. Upon discovery, the licensee immediately reprogrammed the telephone exchange and entered the problem into their corrective action program as condition report CR 002183. (Section 4OA3.6)

B. Licensee-Identified Violation

A violation of very low safety significance, which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and the licensee's corrective action tracking numbers are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Event

On October 7, 2006, at approximately 1711 hours, Unit 2 was manually tripped based on indications associated with main steam flow, main steam pressure, and steam generator feedwater flow and level perturbations. Normal offsite power was lost to both Unit 1 and one of the Unit 2 emergency buses due to flying debris that impacted the 'A' and 'C' Reserve Service Station Transformers' (RSST) electrical conductors. Exhaust steam discharging from opened Unit 2 cross-under piping relief valves (CURV) impacted the adjacent turbine building siding creating flying debris. The dedicated Unit 1, emergency diesel generator (EDG) #1, started and loaded safety system emergency bus 1H. EDG #3, shared between Units 1 and 2, started and automatically loaded to the 2J emergency bus as designed. The alternate AC diesel generator (AAC DG) automatically started but was not manually loaded by the operators to an emergency bus, because a breaker lockout signal had occurred on the 1J emergency bus normal supply breaker. As expected EDG #2 did not start since the 'B' RSST was not affected, and continued to supply power to the Unit 2, 2H emergency bus. This left the Unit 1, 1J emergency bus de-energized.

Due to loss of emergency bus 1J power, a semi-vital bus also lost power on Unit 1. This semi-vital bus powers non-safety related loads associated with secondary side systems. To stabilize the unit from the steam/feedwater transient induced by the loss of normal power to secondary side equipment, Unit 1 operators lowered power to approximately 71%. Unit 2 was stabilized in Hot Shutdown. At 1801 hours the licensee declared an Alert due to the partial loss of offsite power resulting from flying debris damage to safety related equipment. At 1911 EDG #3, the shared EDG, was placed on the 1J emergency bus, leaving the 2J emergency bus de-energized. At 2137 EDG #3 was transferred back from the 1J emergency bus to the 2J emergency bus. Following troubleshooting, the affected breaker lockout contacts associated with protective relaying were reset and the 1J emergency bus was energized at 2154 hours by closing its normal supply breaker.

The 'A' RSST was not damaged, as determined by licensee inspections, and was returned to service supplying power to the Unit 1, 1J emergency bus at 0209 hours on October 8. The Alert was subsequently terminated at 0540. The 'C' RSST bus bar experienced minor damage and was repaired. The 'C' RSST was returned to service supplying the 'F' transfer bus at 1446 hours and normal offsite power was restored to all safety system emergency buses at 1656 hours on October 8. The licensee subsequently retracted their Alert declaration following a critique which determined that their definition of safety significant equipment was incorrect.

NRC Response

Upon notification of the event, the resident inspectors reported to the site and Region II partially manned the Region II incident response center to monitor the event. The NRC reviewed the safety significance and risk of the event and concluded that a special inspection was warranted in accordance with Management Directive 8.3," NRC Incident Investigation Program," due to significant unexpected system interactions and the estimated conditional core damage probability exceeded 1E-6. The special inspection team conducted inspection activities in accordance with the attached “Special Inspection Team Charter” and Inspection Procedure 93812, “Special Inspection.” The objectives of the inspection were to: (1) review the facts surrounding partial loss of site power on October 7, 2006; (2) assess the licensee’s response and investigation of the event; (3) identify any generic issues associated with the event; and (4) conduct an independent extent of condition review. The following six charter items were inspected to meet these objectives.

a. Develop a complete sequence of events related to the event.
b. Conduct an independent extent of condition review of the event. As appropriate, provide any new information that is identified that would affect the risk analysis, to the Region II, Senior Reactor Analyst.
c. Identify and evaluate the effectiveness of the actions taken by the licensee in response to this event.
d. Determine the cause, common cause potential, and corrective actions for the alternate AC diesel generator output breaker lockout.
e. 1. Determine and assess the licensee’s previous actions and lesson learned associated with the cross-under piping safety relief valves lifting during an evolution that previously occurred on Unit 1.
   2. Determine and assess what actions were taken or proposed by the licensee to preclude damage from occurring to the turbine building caused by cross-under piping safety relief valve lifting events.
   3. Determine if there are any generic implications associated with the cross-under safety relief valve steam exhaust piping configuration and potential damage effect. Promptly communicate any potential generic issues to regional management.
f. Evaluate the licensee’s decision making process associated with the licensee’s implementation of the Surry emergency plan and whether the appropriate emergency classifications were declared on October 7, 2006. Assess the appropriate corrective actions associated with this event related to the implementation of the emergency plan.

4. OTHER ACTIVITIES

4OA3 Event Followup - Special Inspection (93812)

.1 Sequence of Events (Charter Item a.)

The team developed the following sequence of events for the loss of the offsite power event using control room logs and trace recorder printouts, computer event recorder data, Technical Support Center (TSC) Logs, and licensee event notification reports. All times are eastern daylight time.

Date/Time	Condition/Event
October 7, 2006	
17:11:18	Unit 2 was at 100% power when the main turbine governor and intercept valves spuriously close due to a main turbine electro-hydraulic control (EHC) system problem. Subsequently, the governor valves reopened

followed by the intercept valves opening within a few seconds. This resulted in Unit 2 generator output decreasing from 852 MWe to 0 MWe and back to over 200 MWe. Additionally, this transient impacted the main steam and feedwater systems resulting in multiple alarms including steam flow - feedwater flow mismatch for all three steam generators (SGs), generator motoring - turbine low differential pressure, high steam flow and level error for all three SGs.

17:11:27 Unit 2 high pressure turbine exhaust pressure increased above normal operating pressure because the high pressure turbine control valves reopened prior to the low pressure turbine intercept valves. Consequently, all 12 moisture separator reheater (MSR) CURVs (6 on each side of the turbine building) opened to relieve the excessive pressure (maximum was 300 psig) created by the valve opening sequence. During this time, sections of the turbine building outside wall were removed and ejected into the air by the steam exhaust flow from the CURVs opening.

17:11:32 Some of the turbine building wall sections ejected by the steam flow contacted conductors associated with the ‘A’ and ‘C’ RSSTs which subsequently resulted in a fault on these transformers and a loss of power to their loads. The ‘A’ RSST is the normal power feed to the 1J 4160V emergency bus and the ‘C’ RSST is the normal power feed to the 1H and 2J 4160V emergency buses. The ‘B’ RSST which was unaffected continued to power the 2H 4160V emergency bus during the event.

The loss of the 1J emergency bus also resulted in the loss of the Unit 1 semi-vital bus. Unit 1 operators entered the applicable abnormal operating procedures, started a third condensate pump and initiated a power reduction to approximately 73% due to affected secondary side plant systems.

17:11:34-39 Alarm for Unit 2 pressurizer power operated relief valve (PORV) open was received (spurious due to loss of the semi-vital bus).

17:11:38 Due to the severe feedwater and steam system transients, related SG level error alarms, Tave - Tref deviation alarm, and the sound of steam flow from the turbine building, the Unit 2 control room operators initiated a manual reactor trip, which caused a main turbine trip, and entered their emergency procedure E-0, “Reactor Trip or Safety Injection.”

After a turbine trip the station service loads are normally transferred automatically to the RSSTs. However, since ‘A’ and ‘C’ RSSTs were faulted, power was loss to the ‘A’ and ‘C’ reactor coolant pumps. The forced flow from these reactor coolant pumps is the source of pressurizer spray flow.

17:11:43 Unit 2 auxiliary feedwater pumps started based on steam generator low-low levels.

17:11:56 Unit 2 feedwater isolation occurred due to reactor trip signal in conjunction with a low Tave signal.

17:14:18 Unit 2 operators closed main steam trip valves due to the main turbine #4 stop valve not indicating full closed.

17:16 Unit 2 Operators transitioned to emergency procedure (EP) 0.1, "Reactor Trip Response."

17:18 Unit 1 semi-vital bus power supply was swapped from 1J1 bus to 1H1 bus (powered from the 1H emergency bus).

17:51 Unit 1 was stable at 71% power.

17:59:20 Unit 2 pressurizer PORV opened and closed.

18:01 Licensee declared an Alert due to criteria in Surry Power Station Emergency Plan Tab K-11, Missile damage to safety related equipment.

18:15 Unit 1 operators attempted to supply power to the 1J emergency bus from the AAC DG; however, the 1J emergency bus normal supply breaker, 15J8, fails to close.

19:11 Operators transferred EDG #3 from the 2J to 1J emergency bus.

21:37 Operators transferred EDG #3 back to the 2J emergency bus.

21:54 Operators were successful in energizing the 1J emergency bus via the AAC DG.

October 8, 2006

01:00 Unit 1 control room log late entries discussed related TS action statements.

02:09 'A' RSST was re-energized from offsite power and aligned to 'D' transfer bus and the 1J emergency bus. The AAC DG was subsequently removed from service.

05:40 Licensee terminated the Alert emergency classification and entered into a recovery phase.

08:59 Unit 1 commenced power increase to 100% rated thermal power.

10:13 A one hour notification to the NRC was made due to the discovery at 0920 hours by the licensee that a Notification of Unusual Event (NOUE) had not been declared on October 7, 2006, as required due to loss of offsite power to the Unit 1 transfer buses D and F.

10:42 Unit 1 was at 100% rated thermal power.

13:50 ‘C’ RSST was re-energized following repairs to the transformer’s conductors.

14:46 ‘F’ transfer bus was re-energized; offsite power now restored to onsite emergency buses.

15:12 EDG #1 was removed from the 1H emergency bus.

16:56 EDG #3 was removed from the 2J emergency bus.

.2 Independent Extent of Condition Review (Charter Item b.)

a. Inspection Scope

The inspectors conducted an independent extent of condition review of the event to determine if other plant system or component configurations could lead to a similar event resulting in the ejection of turbine building siding and impact of important plant components. The inspectors performed plant walk-downs and reviewed applicable plant design basis documents, procedures, and Updated Final Safety Analysis Report (UFSAR) sections.

b. Findings and Observations

There were no findings of significance.

The inspectors reviewed plant system and component configurations on Units 1 and 2 and determined that the Unit 1 CURVs on the south side of the turbine building still retained the same configuration relative to the associated CURV discharge piping. Additionally, the Unit 1 turbine building siding in close proximity to the CURV discharge piping was of the original construction configuration and thus presented a vulnerability to the plant; i.e., an event leading to full opening of the six relief valves would result in removal and ejection of the siding. Since the RSSTs, station service and main transformers were located on the south side of the turbine building, a similar event resulting in a partial or full loss of offsite power was plausible. The inspectors verified that the licensee had captured this problem in their corrective action program. Additionally, the inspectors reviewed interim corrective action implemented by the licensee to secure the affected Unit 1 siding per Work Order (WO) 00757998-01 with guidance from Engineering Transmittal (ET-NPD-CME-06-0005), “Turbine Building Siding Evaluation and Structural Enhancement Guidance.” Refer to Section 4OA3.5 of this report for additional information relative to past and present corrective actions.

.3 Licensee Response to the Event (Charter Item c.)

a. Inspection Scope

The inspectors reviewed the licensee's response to the event relative to implementation of Technical Specifications (TS), emergency and abnormal procedures, and timely incorporation of this event into the overall plant risk analysis used in maintenance rule risk evaluations. The inspectors' review of the licensee response in the area of Emergency Planning is documented in Section 4OA3.6 of this report. The inspectors also reviewed the licensee's reactor trip report and a draft root cause evaluation and interviewed plant personnel to verify the licensee had appropriately reviewed the plant response to the trip and identified any problems or issues with respective corrective actions.

b. Findings and Observations

There were no findings of significance.

The inspectors identified four issues associated with the licensee's response to the event. The first issue involved the Unit 2 PORV, 2RC-PCV-2455C, which cycled open and then closed approximately 50 minutes following the reactor trip. The inspectors recognized that there was a significant reduction in pressurizer spray flow available due to the loss of 'A' and 'C' Reactor Coolant Pumps and that the operators initial response to the event was very good. However, the inspectors concluded that the operators should have been aware of pressurizer pressure trends and taken proactive steps to prevent pressure from reaching the PORV set point. Industry operating experience has shown that PORVs are susceptible to sticking in an open position creating complications to an event. This concern was discussed with the licensee who generated condition report 005013.

The second issue found by the inspectors during the evaluation of the licensee's response to the event was found through interviews with plant personnel. Interviews revealed that a recorder had been connected to a card within the Unit 2 EHC cabinet per Work Order 00734213-01 since February 3, 2006, to monitor several parameters. Although the potential impact from a malfunction of the recorder had been evaluated by the licensee, the inspectors noted that this evaluation was not documented in the root cause evaluation.

The third issue observed by the inspectors involved not maintaining evidence of equipment malfunction. The inspectors found that the licensee concluded that the EHC malfunction causing the main turbine transient was the result of a spurious overspeed protection control (OPC) signal. The licensee surmised in their root cause evaluation that a faulty connector on an associated card was possibly responsible for the spurious signal. However this could not be verified because the licensee had already shipped the card to a vendor for failure analysis and had no other evidence for review. The inspectors concluded that quarantine of the affected cabinets and subsequent analysis with attendant documentation to positively identify a root cause was not maintained.

Not maintaining the card was likely the reason the licensee could not determine a definitive answer to the EHC malfunction.

The fourth issue identified by the inspectors involved the licensee's risk assessments of plant conditions. Upon arrival at the site on October 23, 2006, approximately 2 weeks after the event, the inspectors requested a risk assessment of this event which included a CURV event from June 29, 2006, and the existing configuration of the Unit 1 cross-under relief valves on the south side of the turbine building. This configuration could result in removal and ejection of turbine building siding on discharge of the relief valves similar to that which occurred on Unit 2 on October 7, 2006, and on the Unit 1 north side turbine building CURVs on June 29, 2006. The licensee did not have the risk calculations available and initiated action to perform the risk evaluation, and subsequently incorporated the results into the risk matrix for Unit 1 maintenance rule evaluations. Pending further NRC review of the licensee actions to perform and complete the risk evaluation, this issue will be tracked as an unresolved item (URI) 5000280, 281/2006011-01, Evaluation of Risk Analysis for Unit 1 For Cross-under Relief Valve Events.

.4 Alternate AC Diesel Generator Output Breaker Lockout (Charter Item d.)

a. Inspection Scope

Following the main turbine transient, faults on the 'A' and 'C' RSST resulted in a partial loss of offsite power. The #1 and #3 EDGs restored power to the 1H and 2J buses as designed. However, the AAC DG could not be loaded on to the 1J bus. By design, this should have been achieved by placing the 1J emergency bus normal supply breaker, 15J8, control switch in the main control room to the closed position. The inspectors reviewed several elementary diagrams and corrective action documents to independently assess the causal factors for the switch's failure to close, common cause potential, and the appropriateness of the licensee's corrective actions.

b. Findings and Observations

Following the unexpected actuation of the Unit 2 CURVs, pieces of siding contacted two transformers causing a partial loss of offsite power. Faults and subsequent lockouts of the 'A' and 'C' RSSTs resulted in undervoltage (UV) conditions on the 1H, 1J, and 2J emergency buses. These UV conditions initiated automatic starting and loading of the #1 and #3 EDGs and restored power to the 1H and 2J buses. By design, the #3 EDG preferentially loaded the 2J emergency bus on a UV condition on that bus. The same UV conditions automatically started the AAC DG. However during the event, the AAC DG could not be loaded onto the 1J emergency bus as designed by closing the 1J normal supply breaker, 15J8, from the main control room. The inspectors independently verified that the cause of this event was a circuit design deficiency. When the AAC DG selector switch 43-35J8 was placed in the AAC position, the lockout relays (86/A and 86/PW) that input to the trip circuit for breaker 15J8 were not defeated as intended by the design.

The design objective of selector switch 43-35J8 when placed in the AAC position, was to defeat the following input signals to the trip coil of circuit breaker 15J8, and enable manual closure from the control room:

- RSST pilot wire lockout relay;
- RSST differential relay; and,
- RSST to transfer bus feeder breaker open.

The inspectors determined that the associated emergency bus under/degraded voltage relay was blocked by operation of the circuit breaker control switch in the control room when loading the AAC DG onto the emergency bus. However, the circuit still permitted DC voltage to travel through the closed contact of the RSST pilot wire lockout relay (86/PW), as well as the differential lockout relay (86/A), thereby maintaining a signal on the 1J normal supply breaker, 15J8, trip coil. With the trip coil energized, the breaker could not be closed. Since the AAC DG could only supply power to the 1J emergency bus using this breaker, the 1J emergency bus could not be loaded onto AAC DG.

A review of the breaker circuitry for the normal feed to the 2H emergency bus (breaker 25H8) revealed that a similar condition existed on Unit 2 if a fault were to occur at the 'B' RSST. The licensee initiated CR002805 for evaluation and resolution of this issue. Subsequently, design change 06-052 was implemented to insert available spare contacts of the AAC selector switch into the 1J and 2H feeder breaker's trip circuits to fully disable the RSST trip signals. This modification allowed the station to supply the 1J and 2H buses from the AAC DG by operating the selector switches from the control room as intended by the initial design.

One of the licensing basis functions of the AAC DG was to meet the station blackout requirements of 10 CFR 50.63, "Loss of all alternating current power." The inspectors questioned whether the AAC DG with the pre-modified circuitry met the licensing basis function. The inspectors verified that with the circuit modification corrected, the AAC DG met the licensing basis function of 10 CFR 50.63.

Pending further NRC review of the pre-modified configuration, this issue will be tracked as URI 05000280, 281/2006011-02, Breaker Design Deficiency Prevents the AAC DG from being Loaded as Designed.

.5 Corrective Actions Associated with CURVs (Charter Item e.)

.5.1 Previous Actions, Lessons Learned, and Actions Taken to Preclude Recurrence (Charter Item e.1.)

a. Inspection Scope

The inspectors reviewed the licensee's corrective action database for documented issues and problems similar to the those leading to the partial loss of offsite power event to determine the adequacy any corrective actions implemented. The inspectors also interviewed licensing personnel to obtain similar documentation from legacy corrective action databases.

b. Findings and Observations

The inspectors identified several previous events involving CURV actuations of which two documented events related damage to the turbine building. The first event, in 1996, was documented in Plant Issue S-1996-0231, "When 2RL and 2IL ("C" MSR reheat step & intercept) were closed IAW 1-OSP-TM-001, the crossunder safety valves lifted ... window in the turbine building shattered. (The south wall between columns B4 & B5)." Documentation from that time did not address corrective actions, if any, to address the cause of the damaged window.

The second event occurred in 2006 and was documented in Plant Issue S-2006-2851. The event's description read , "Unit 1 Turbine Crossunder SV (North side) lifted shortly after tagging out 1-EH-SOV-1RL. During the lift portions of the Unit 1 TB sheet metal were pulled loose and entrained into the steam path. Sheet metal and other debris are scattered from Unit 1 alleyway to the "D" building (sizes range from whole sheets to smaller pieces. No one was injured by the falling debris. The Crossunder SVs seated by themselves after ~4 minutes (prior to reducing power). Unit 1 was ramped down further (to ~91% power) using 0-AP-23.00 at 155 MWe/hr to increase the margin between Cross-under pressure and the actual SV setpoints. 1-MS-SV-108A continued to simmer significantly after lifting (but did display an improving trend)."

The inspectors' review of Plant Issue S-2006-2851 revealed documentation that stated, "Several whole sheets and several partial sheets of metal were caught in the path of the steam flow and hurled across the unit 1 alleyway towards the "D" building approximately 100 feet away." The inspectors noted that the root cause evaluation for the Plant Issue stated, "No additional short term corrective actions are required," and that the long term corrective actions did not address the removal of turbine building siding and potential effects on the station. The inspectors also noted that one Plant Issue S-2006-2851 resolution to engineering was, "to determine and initiate required actions to prevent Turbine Building wall degradation due to safety relief valve actuations (to include Unit 2)." However, the inspectors determined that the licensee implemented no interim corrective actions to secure siding as mentioned in Section 4OA3.2 above. Additionally, the inspectors determined that there were no resolutions to evaluate potential impact on the station such as the event that occurred on October 7, 2006 resulting in a partial loss of offsite power. Pending further NRC review of the licensee actions relative to corrective action, this issue will be tracked as URI 05000280, 281/2006011-03, Review of Licensee Corrective Actions to Preclude Recurrence of Damage From Cross-under Relief Valve Actuations.

.5.2 Generic Implications of CURV Configuration (Charter Item e.2.)

a. Inspection Scope

The inspectors performed a review of the plant configuration relating to CURVs as compared to other plant operating experience (OE) reports to identify any potential generic implications.

b. Findings and Observations

There were no findings of significance.

The inspectors' review of other related OE revealed that one other plant had a similar configuration of the CURVs; however, the plant did not have a turbine building shell to enclose related secondary equipment. Therefore, CURV actuations at this plant did not result in damage to structures. The inspectors therefore concluded that there were no significant generic implications of this event for other plants.

.6 Emergency Plan Implementation (Charter Item f.)

a. Inspection Scope

The inspectors completed one inspection sample by reviewing the October 7, 2006 event. The inspectors reviewed the applicable logs, associated licensee procedures and discussed the event with operators, and members of the licensee's staff and management.

The special inspection team was tasked to evaluate the licensee's decision making process associated with the licensee's implementation of the Surry Emergency Plan and whether the appropriate emergency classifications were declared on October 7, 2006.

The special inspection team was also tasked to assess the appropriate corrective actions associated with this event related to the implementation of the emergency plan.

b. Findings and Observations

The licensee identified two items associated with their emergency classifications of the event. On October 8, the licensee determined Unit 1 should have been in an Notice of Unusual Event (NOUE) while the normal plant transfer buses were out of service. This item met the criteria for a licensee identified violation and is discussed in Section 4OA7 of this report. Subsequent licensee review revealed that Unit 2 should not have been in an Alert classification and withdrew their notification of an Alert. The inspectors assessed the licensee's review and determined that Unit 2 had not met the criteria to be in an Alert classification. Since they self-identified the issue, no violation of NRC regulations occurred.

Failure to Activate the Emergency Response Data System (ERDS) as Required

Introduction: A self-revealing non-cited violation (NCV) of 10 CFR 50.72(a)(4) of very low safety significance was identified. The licensee failed to activate the Emergency Response Data System (ERDS) not later than one hour after declaring an emergency class Alert or higher.

Description: On October 7, 2006, during the partial loss of offsite power event, the ERDS could not be activated when required. The ERDS was not made operable until approximately five and one-half hours after the Alert declaration. The inspectors noted that the cause of the problem was due an equipment program rendered inadequate from an upgrade to the telephone exchange that was performed seven days prior to the event. The inspectors determined that this is contrary to 10 CFR 50.72(a)(4) which requires that ERDS be activated as soon as possible but not later than one hour after declaring an emergency class Alert or higher.

Analysis: The inspectors reviewed Inspection Manual Chapter (IMC) 0612 and determined that the finding is more than minor due to its impact on the Emergency Preparedness cornerstone objective to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency and the related attribute of Emergency Response Organization response. The inspectors performed a Significance Determination Process (SDP) review as required by IMC 0609, Appendix B and determined that the finding is of very low safety significance (Green) because it involved a “failure to implement” (in distinction to a “failure to meet”) an NRC emergency planning standard. The cause of the finding is related to the cross-cutting area of human performance, in that, the licensee failed to reprogram the telephone exchange to accept both 8 or 9 pre-fix following a telephone system change which occurred prior to the event. Upon discovery, the licensee immediately reprogrammed the equipment and entered the problem into their corrective action program as CR 002183.

Enforcement: 10 CFR 50.72(a)(4) requires that ERDS be activated as soon as possible but not later than one hour after declaring an emergency class Alert or higher. Contrary to this on October 7, 2006, ERDS could not be activated within the required time period due to equipment program problems as the result of previous telephone system upgrades. Because this finding is of very low safety significance and because it was entered into the licensee’s corrective action program as CR002183, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000280, 281/2006011-04, Failure to Activate ERDS Within One Hour of an Alert Declaration.

4OA6 Meetings, including Exit

On November 1, 2006, the special inspection team leader presented the inspection results to Mr. D. Jernigan and other members of the staff. The licensee acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violation

The following finding of very low significance was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for characterization as an NCV.

Title 10 CFR 50.54(q) states that a licensee authorized to possess and operate a nuclear power reactor shall follow and maintain in effect emergency plans which meet the standards in 10 CFR 50.47(b) and the requirements in Appendix E of this part.

10 CFR 50.47(b)(4) states that a standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures. The licensee's action level scheme was defined in the Surry Power Station Emergency Plan, revision 50. The licensee's Emergency Manager Controlling Procedure, EPIP-1.01, revision 47, specifically identified the indication of loss of offsite power (Emergency Action Level (EAL) Tab H-3) as loss of offsite power to unit specific transfer buses (Unit 1: D & F; Unit 2: E & F) and required a declaration as a Notification of Unusual Event (NOUE).

Contrary to the above, on October 7, 2006, Unit 2 was manually tripped when multiple alarms associated with steam flow and feed flow mismatch were received and crew noted the sound of steam flow in the turbine building. During the transient, the CURVs on both sides of the turbine building lifted causing portions of the turbine building siding to detach and contact two phases of the overhead bus for the 'A' and 'C' RSSTs. The Shift Manager, the Manager of Operations, a qualified Station Emergency Director, and the Shift Technical Advisor actively reviewed the EALs for applicability; however, the licensee failed to identify that entry conditions were met for EAL Tab H-3 for Unit 1 and failed to declare the required NOUE for Unit 1. After reviewing Plant Computer System trend data the licensee determined that the EAL criteria for a NOUE for Unit 1 was met, but was not declared. Event Notification 42890, Discovery of After-the-Fact-Emergency Condition - NOUE, was made on October 8, 2006 at 1013 hours. The finding was of very low safety significance because, although it involved an actual event, the event was a NOUE, the finding only involved a failure to comply with the emergency plan and there were no indications of Planning Standard problems. This issue was entered into the licensee's corrective action program as CR 002191.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

M. Adams, Director, Nuclear Station Safety and Licensing
J. Bischof, Vice President, Nuclear Engineering
J. Costello, Supervisor, Emergency Preparedness
M. Crist, Manager, Nuclear Operations
S. Davis, Supervisor, Nuclear Unit Operations
B. Garber, Supervisor, Licensing
J. Grau, Manager, Nuclear Oversight
C. Gum, Supervisor, Nuclear Security Operations
E. Hendrixson, Engineer
L. Hilbert-Semmes, Supervisor, Operation
T. Huber, Manager, Engineering
D. Jernigan, Site Vice President
L. Jones, Manager, Radiation Protection and Chemistry
J. Keithly, Supervisor Health Physics Operations
L. Lastauckas, Reactor Operator, Operations
C. Luffman, Manager, Protection Services
D. Miller, Supervisor Health Physics Technical Services
M. Oppenhimer, Manager, Nuclear Design Engineering
R. Simmons, Manager, Outage and Planning
K. Sloane, Director, Nuclear Station Operations and Maintenance
B. Stanley, Manager, Nuclear Maintenance
T. Steed, Manager, Organizational Effectiveness
M. Wilson, Manager, NuclearTraining

ITEMS OPENED AND CLOSED

Opened

05000280, 281/2006011-01	URI	Evaluation of Risk Analysis for Unit 1 For Cross-under Relief Valve Events
05000280, 281/2006011-02	URI	Breaker Design Deficiency Prevents the AAC Diesel from being Loaded as Designed
05000280, 281/2006011-03	URI	Review of Licensee Corrective Actions to Preclude Recurrence of Damage From Cross-under Relief Valve Actuations

Opened and Closed

05000280, 281/2006011-04	NCV	Failure to Activate ERDS Within One Hour of an Alert Declaration

LIST OF DOCUMENTS REVIEWED

Procedures and Documents

- EPIP-1.01, Emergency Manager Controlling Procedure, Rev. 47
- SPS EAL Basis, Emergency Action Level Technical Basis Document, Rev. 10
- Technical Specifications Surry Power Station Units 1 and 2, 7/26/06
- NCRODP (Nuclear Control Room Operator Development Program) -35-S, Vital and Emergency Electrical Distribution System, Rev. 39
- Updated Final Safety Analysis Report, Chapter 8 Electrical Systems, Rev. 38
- Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 4
- Event Notification 42885, Non-emergency, Normal road access to Surry Power Station became flooded and impassible due to sustained heavy rainstorms. This flooding compromised the Surry off-site response capability, 10/7/06 at 1340
- Event Notification 42888, Alert, Manual scram due to unusual noise in turbine building, missile damage to ‘A & C’ Reserve Station Service Transformers.10/7/06 at 1801
- Event Notification 42888, Alert terminated, ‘A’ Reserve Station Service Transformer in service with bus ‘1J’ energized, 10/08/06 at 0545
- Event Notification 42890, Discovery of after-the-fact-emergency condition - unusual event, 10/7/06 at 1711
- DCN 06-052, Modify Circuit Breaker Logic for loading AAC Diesel Generator onto the Emergency Buses
- 11448-FE-21D, D.C. Elementary Diagram Generator 1 & Transformers Protection, Rev. 37
- 11448-FE-21J, Elementary Diagram 4160V Circuits, Rev. 18
- 11448-FE-21K, Elementary Diagram 4160V Circuits, Rev. 20
- 11448-FE-21P, Elementary Diagram Pilot Wire Relaying, Rev. 7
- 11548-FE-21Q, 4160V Bus 2H BKR 25H3 & 25H8, Rev.10
- Unit 1 Control Room Logs for 10/07 - 09/06
- Unit 2 Control Room Logs for 10/07 - 09/06
- 1-AP-10.05, “Loss of Semi-Vital Bus)”
- Event Log, VEPCO Surry Unit 2, 10/06/06 - 19:26:44:526 through 10/07/06 - 17:11:37:962
- SDBD-SPS-ESS, “System Design Basis Document for Station Service Power System Surry Power Station”
- SDBD-SPS-EP, “System Design Basis Document for Emergency Power System Surry Power Station”
- SDBD-SPS-EG, “System Design Basis Document for Emergency Diesel Generator System Surry Power Station”
- Surry Master EP Timeline, 10/07/06
- Surry Unit 2 Reactor Trip Report S2-10-07-06\
- Interim Report Root Cause Evaluation RC-000005 Surry, Unit 2, “Manual Unit Trip in Response to Steam Flow Feed Flow mismatch with a loss of ‘A’ and ‘C’ Reserve Station Service Transformers”
- Surry Master EP Timeline, 10/07/06
- Surry Unit 2 Reactor Trip Report S2-10-07-06
- Various plant computer trend graphs of plant parameters

- Interim Report Root Cause Evaluation RC-000005 Surry, Unit 2, "Manual Unit Trip in Response to Steam Flow Feed Flow mismatch with a loss of 'A' and 'C' Reserve Station Service Transformers"
- Equipment Specification 677088, page 5 of 10
- Category 2 RCE 96-0231 Unit 1 Cross Under Safety Valves Lifted During Testing
- Operation Decision Making document for CR002219 dated 10-13-06 and approved 10-14-06

<u>Condition Reports (CR)</u>

- CR002173, Surry LEOF, water was observed leaking through two light fixtures above the Dose Assessment Area.
- CR002174, Surry LEOF, the front door and back door emergency exit lights were inoperative.
- CR002183, The NRC communicators attempted to connect the Emergency Response Data System link to the NRC without success. The procedure was attempted several times and both in the Main Control Room and the TSC.
- CR002185, Upon declaration of Alert, Surry Security did not initially notify Community Alert Network (CAN) in accordance with EPIP-3.05 Action/Expected Response. Instead, ITC Security was notified and directed to initiate back-up ERO augmentation notification in accordance with CPIP-3.4.
- CR002186, During the Emergency on 10/8/06 the ERFCS indications for Unit 2 became extremely questionable. Most digital indications were magenta. Most analog indications were bad or poor. SPDS showed red in two categories. Unit 1 indications were never affected.
- CR002187, During the Emergency on 10/8/06 the voltages for the 480V Emergency Bus load centers and MCCs on Unit 2 varied widely. The reading for each component was steady, but the voltages among the components ranged from approximately 400 volts to 550 volts. I did not check the Station Service voltages or the Unit 1 voltages.
- CR002188, Following Unit 2 trip and loss of power the MCR and Annex area were unable to access LAN applications such as webtop and eSOMS.
- CR002189, CERC recorder Deck B tape failed during CERC activation on 10/7/06.
- CR002190, Following Unit 2 trip and loss of power, the Station did not have adequate food for on-shift personnel. Food was ordered from TSC but none was delivered to Operations.
- CR002191, After reviewing PCS Trend data it appears NOUE Tab H-3 was missed. Also, time when information was available relative to the Alert classification time is being reviewed. [Accuracy of ALERT classification being reviewed – RSST are not safety-related.]
- CR002192, Adverse road conditions (flooding and blocked roads) delayed response and [ERO staffing] goals were not satisfied in some cases.
- CR002193, Some equipment was unavailable due to the loss of power in the Technical Support Center during the 10/7/06 Alert. This hampered emergency response. Examples: No air conditioning, Fax machines were inoperative, LAN connection was down [not really down, but unavailable on workstations because of power loss according to IT), thus access to Webtop was unavailable, Cordless telephones were inoperative, including the TSC-LEOF/CERC/CEOF network, Limited lighting (very difficult to see).
- CR002194, Neither CERC/CEOF nor LEOF were able to make eSOMS log entries during an emergency response on 10/7/06. (eSOMS was unavailable in other locations

due to LAN connection problems, See CR002188 and CR002193)

- CR002195, During the Alert response on 10/7/06, CERC/CEOF emergency response organization members released after initial augmentation were provided a telephone number to call for information. When the CERC/CEOF was deactivated a voice mail message was recorded providing this information, e.g., status, expectations for relief, etc. However, these numbers did not roll-over to voice mail.
- CR002219, Unit 2 reactor was manually tripped, loss of A & C RSST, Declaration of Alert
- CR002220, AAC diesel could not be initially loaded to 1J bus
- Station Deviation Report Number: 1-90-989
- Station Deviation Report Number: S1-90-1096
- Deviation Report: S-94-2352
- Deviation Report: S-94-1862
- Deviation Report: S-96-0231
- Deviation Report: S-96-0255
- Deviation Report: S-96-1536
- Commitment Tracking 3398
- Commitment Tracking 3399
- Commitment Tracking 3400
- Commitment Tracking 3664

October 13, 2006

MEMORANDUM TO: James Reece, Team Leader
Special Inspection Team

FROM: William D. Travers, Regional Administrator ***/RA/***

SUBJECT: SURRY SPECIAL INSPECTION TEAM CHARTER

A Special Inspection Team (SIT) has been established for Surry to inspect and assess the facts surrounding the partial loss of offsite power and Unit 2 manual reactor trip. The Team Leader will be James Reece, of Region II. Team members will be assigned as appropriate based on the issues identified. Your inspection should begin on October 23, 2006.
The objectives of the inspection are to: (1) review the facts surrounding partial LOSP on October 7, 2006; (2) assess the licensee's response and investigation of the event; (3) identify any generic issues associated with the event; and (4) conduct an independent extent of condition review.

For the period during which you are leading this inspection and documenting the results, you will report directly to me. The guidance in Inspection Procedure 93812, "Special Inspection" and Management Directive 8.3, "NRC Incident Investigation Procedures," applies to your inspection.

If you have any questions regarding the objectives of the enclosure charter, contact Charles A. Casto at (404) 562-4500.

Docket Nos.: 50-280, 50-281
License Nos.: DPR-32, DPR-37

Enclosure: Special Inspection Team Charter

cc w/encl: (See page 2)

CONTACT: Eugene F. Guthrie, DRP/RII
404-562-4662

cc w/encl: (See page 2)

L. Reyes, OEDO
W. Kane, DEDO
S. Campbell, OEDO
J. Dyer, NRR
S. Monarque, NRR
R. Zimmerman, NSIR
V. McCree, RII
C. Casto, RII
H. Christensen, RII
N. Garrett, RII

SPECIAL INSPECTION TEAM (SIT) CHARTER
PARTIAL LOSS OF OFFSITE POWER AT SURRY AND UNIT 2
MANUAL REACTOR TRIP

Basis for the Formation of the SIT - On 10/07/06 at approximately 5:11 p.m., Unit 2 was manually tripped based on indications associated with main steam flow, main steam pressure, and steam generator feedwater flow and level perturbations. The site was in ALERT from 6:01 p.m. until 5:40 a.m. on October 8 due to a partial loss of offsite power. Normal offsite power was lost to both Unit 1 and one of the Unit 2 emergency busses due to flying debris that impacted the 'A' and 'C' Reserve Service Station transformers electrical conductors. The flying debris results from exhaust steam from the Unit 2 moisture separator reheater cross-under piping safety relief valves impacting the outside of the turbine building.

The dedicated Unit 1, number 1 emergency diesel generator, started and loaded its safety system emergency buss. The number 3 emergency diesel generator, shared between Units 1 and 2, started and automatically transferred to the other Unit 1 safety system emergency buss since that buss became de-energized before the Unit 2 safety system emergency buss de-energized. The alternate AC emergency diesel generator automatically started but was not manually loaded due to a generator output breaker lockout. This left the Unit 2, 2J, safety system emergency buss de-energized until the alternate AC emergency diesel generator output breaker was reset and closed at 9:54 p.m. The number 2 diesel generator did not start as expected, since the 'B' Reserve Service Station transformer was not affected and continued to supply power to the Unit 2, 2H buss. The 'A' Reserve Service Station transformer was not damaged, as determined by licensee inspections, and was returned to service supplying power to the Unit 1, 1J, safety system emergency buss at 2:09 a.m. on October 8 The 'C' Reserve Service Station transformer buss bar experienced minor damage and was repaired. The 'C' Reserve Service Station transformer was returned to service at 2:47 p.m. and normal offsite power was restored to all safety system emergency busses at 5:27 p.m. on October 8. Unit 1 remained at power during the event and was stabilized at approximately 73 percent rated thermal power. Unit 2 was stabilized in Hot Shutdown.

Objectives of the SIT - The objectives of the inspection are to: (1) review the facts surrounding partial LOSP on October 7, 2006; (2) assess the licensee's response and investigation of the event; (3) identify any generic issues associated with the event; and (4) conduct an independent extent of condition review. To accomplish these objectives, the following will be performed:

a. Develop a complete sequence of events related to the event.
b. Conduct an independent extent of condition review of the event. As appropriate, provide any new information that is identified that would affect the risk analysis, to the Region II, Senior Reactor Analyst.
c. Identify and evaluate the effectiveness of the actions taken by the licensee in response to this event.

d. Determine the cause, common cause potential, and corrective actions for the alternate AC diesel generator output breaker lockout.

e. 1. Determine and assess the licensee's previous actions and lesson learned associated with the cross-under piping safety relief valves lifting during an evolution that previously occurred on Unit 1.
2. Determine and assess what actions were taken or proposed by the licensee to preclude damage from occurring to the turbine building caused by cross-under piping safety relief valve lifting events.
3. Determine if there are any generic implications associated with the cross-under safety relief valve steam exhaust piping configuration and potential damage effect. Promptly communicate any potential generic issues to regional management.

f. Evaluate the licensee's decision making process associated with the licensee's implementation of the Surry emergency plan and whether the appropriate emergency classifications were declared on October 7, 2006. Assess the appropriate corrective actions associated with this event related to the implementation of the emergency plan.

g. Document the inspection findings and conclusions in an inspection report within 30 days of the inspection.

Inspection Dates: October 23, 2006 until objectives are met.

References:

1. NRC Inspection Procedure 93812, Special Inspection
2. Region II, ROI 2296 Rev. 1, Management Directive 8.3 Decision Documentation Form
3. Management Directive 8.3, NRC Incident Investigation Program
4. Manual Chapter 0612, Power Reactor Inspection Reports
5. Manual Chapter 0609, Significance Determination Process