U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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License Nos:	DPR-32, DPR-37
Report No:	50-280/97-07, 50-281/97-07
Licensee:	Virginia Electric and Power Company (VEPCO)
Facility:	Surry Power Station, Units 1 & 2
Location:	5850 Hog Island Road Surry, VA 23883
Dates:	July 13 - August 23, 1997
Inspectors:	R. Musser, Senior Resident Inspector K. Poertner, Resident Inspector P. Byron, Resident Inspector R. Gibbs, Reactor Inspector (Section M8.1) P. Hopkins, Project Engineer (Sections 08.2, 08.3, 08.5, M8.2, M8.5 and E8.2) J. Blake, Senior Project Manager (Section M2.1)
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Enclosure

EXECUTIVE SUMMARY

Surry Power Station, Units 1 & 2
NRC Inspection Report Nos. 50-280/97-07, 50-281/97-07

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection; in addition, it includes the results of announced inspections by three regional inspectors.

Operations

- The licensee has good controls for the tagout process and is focussing on backlog reduction (Section 01.2).

- Based on observations and a document review, the Seismic Monitoring System was operable (Section 01.3).

- The main control board position indications for selected containment isolation valves satisfied procedural requirements (Section 01.4).

- The Unit 1 Containment Spray System was properly aligned and in generally good condition (Section 02.1).

- The recently implemented employee concerns telephone reporting system was a good idea; however, the licensee's efforts in informing employees of the new service had not been effective in some groups (Section 08.1).

- A Non-cited Violation was identified for failure to isolate and deactivate containment isolation trip valve 1-RC-TV-1519A within fours hours as required by TS 3.8.C (Section 08.2).

Maintenance

- Technicians performing inadequate core cooling monitor power supply maintenance were cautious and consistent in their repeat backs. The technicians utilized good work practices and did a good job (Section M1.1).

- Personnel performing maintenance activities on the Number 1 Emergency Diesel Generator louvers had drawings, procedures, and work orders at the job site. There was adequate management and technical support. The inspectors expressed concern to licensee management that the Vendor Technical Manual recommended setpoint had not previously been incorporated into the maintenance procedure (Section M1.2).

- Three observed surveillance activities were accomplished in accordance with approved procedures and were completed satisfactorily (Section M1.3).

- Alternate alternating current diesel generator testing was conducted in accordance with approved procedures. Problems encountered during performance of the test were properly dispositioned by the operating crew. During performance of the test, elevated coolant temperatures were observed and were attributed to diesel exhaust entering the radiator fan suction. The engineering transmittal addressing the issue lacked documentation to justify the conclusions reached. An inspection followup item was identified to review this item further (Section M1.4).

- A Station Nuclear Safety and Operating Committee (SNSOC) meeting that approved an Engineering Transmittal associated with elevated alternate alternating current diesel coolant temperatures did not sufficiently challenge the conclusions reached or require further review or testing to determine the extent of the problem. Based upon previous observations, matters reviewed by SNSOC were normally thorough. This instance is considered an isolated weakness (Section M1.4).

- Testing of the Number 1 Emergency Diesel Generator was conducted satisfactorily. The failure to ensure that repairs were scheduled for a problem recognized during the previous monthly test indicates a lack of ownership of the component and is considered a weakness in both system engineering and scheduling (Section M1.5).

- The Flow Accelerated Corrosion Inspection Program for Surry is a well-organized, mature program, with inspection-established growth rates, and program-predicted life expectancies for all major carbon steel components in the secondary piping systems (Section M2.1).

- The Flow Accelerated Corrosion inspection scope, inspection expectations, and in some cases inspection results, were not clearly communicated to plant management and other people outside of the program (Section M2.1).

- No problems were observed regarding corrective actions for equipment problems. Trending of station deviations and corrective action for adverse trends identified by that process were assessed as a strength (Section M8.1).

Engineering

- The licensee performed an adequate evaluation and implemented appropriate actions to alleviate concerns with hydrogen concentration within the Unit 1 containment (Section E1.1).
- A Non-cited Violation was identified for Emergency Service Water Pump 1-SW-P-1A being inoperable due to inadequate missile shielding for the pump's discharge piping for a period of time greater than allowed by TS 3.14.B (Section E8.1).

Plant Support

- Health physics practices were observed to be proper (Section R1).
- Security and material condition of the protected area perimeter barrier were acceptable (Section S1).

Report Details

Summary of Plant Status

Unit 1 operated at power the entire reporting period. On August 21 power was reduced to 49 percent to repair a turbine intercept valve. The valve was repaired and the unit returned to 100 percent power on August 22.

Unit 2 operated at power the entire reporting period. On July 24 power was reduced to 64 percent to allow removal of the B main feedwater pump for motor repairs. The pump motor was repaired and the unit returned to 100 percent power on July 26.

I. Operations

01 Conduct of Operations

01.1 General Comments (71707, 40500)

The inspectors conducted frequent control room tours to verify proper staffing, operator attentiveness, and adherence to approved procedures. The inspectors attended daily plant status meetings to maintain awareness of overall facility operations and reviewed operator logs to verify operational safety and compliance with Technical Specifications (TSs). Instrumentation and safety system lineups were periodically reviewed from control room indications to assess operability. Frequent plant tours were conducted to observe equipment status and housekeeping. Deviation Reports (DRs) were reviewed to assure that potential safety concerns were properly reported and resolved. The inspectors found that daily operations were generally conducted in accordance with regulatory requirements and plant procedures.

01.2 Review Of Tagouts

a. Inspection Scope (71707)

The inspectors reviewed the active tagout packages located in the shift office.

b. Observations and Findings

Fifty-seven tagout packages were reviewed and the inspectors noted that 11 were older than one year. Of these, 6 were for Unit 1 and 5 for Unit 2. The oldest tagout (1-91-SA-0004) was issued on January 29, 1991. The licensee is in the process of making a decision to either relocate the compressor associated with the tagout or scrap it. The inspectors noted that 4 of the 11 tagouts require an outage in order to close. Three of the oldest tagouts on Unit 1 were for abandoned equipment and 3 tagouts on Unit 2 were used to procedurally control equipment.

The inspectors did not identify any problems with the tagouts which have been active less than one year. The licensee is making an effort to minimize the tagout backlog. A list of the active tagouts is generated and reviewed daily to focus on tagouts which can be closed or to schedule work which would enable the tagout to be closed. The inspectors were informed that this effort has the attention of senior management and Operations issues a tagout backlog report to senior management monthly.

c. Conclusions

The inspectors consider that the licensee has good controls for the tagout process and is focussing on backlog reduction.

O1.3 Seismic Monitor Operability

a. Inspection Scope (71707)

The inspectors inspected accessible portions of the Seismic Monitoring System and reviewed completed Periodic Tests (PTs).

b. Observations and Findings

The Seismic Monitoring System has three accelerometers. Two are located in the Unit 1 containment. The third accelerometer and a triggering device are located outside the Protected Area. The inspectors reviewed the TS and the Updated Final Safety Analysis Report (UFSAR). The inspectors also reviewed completed procedures 1-PT-31.1, "Seismic Instrument Test," Revision 3, and 1-PT-31.3, "Seismic Instrument Status Check Recording," Revision 2. These were scheduled to be performed monthly for the first seven months of 1997. However, 1-PT-31.3 was not performed during March 1997, when procedure 1-PT-31.4, "Seismic

Instrumentation Calibration," Revision 3, was performed. Procedure 1-PT-31.4 incorporated the requirements of 1-PT-31.3. Completed semi-annual procedures 1-PT-31.2, "Seismic Instrument Functional Test," Revision 2, and IMP-C-MISC-45, "Seismic Instrument Preventative Maintenance," Revision 1, were also reviewed. The inspectors noted that all the PTs were performed within the specified time period and the results were satisfactory.

c. Conclusions

The inspectors concluded that based on their observations and document review that the Seismic Monitoring System was operable.

O1.4 Containment Isolation Valve Lineup

a. Inspection Scope (71707)

The inspectors reviewed the procedures controlling the position of the Unit 1 containment isolation valves and observed their position indication on the control room panel.

b. Observations and Findings

The position of containment isolation valves is controlled by the following procedures:

- 1-OPT-CT-211, "Containment Integrity Verifications For: Inside CTMT (Containment) Manual and LMC (Leakage Monitoring Connections) Vlvs. (Valves), and Various Flanges and Hatches," Revision 3
- 1-OPT-ZZ-05, "Verification of Local and MCB (Main Control Board) Valve Position Indication for Przr (pressurizer) PORVs (Power Operated Relief Valves)," Revision 1
- 1-OPT-ZZ-06, "Verification of Local and Remote Valve Position Indication of Safety Relief Valves Inside Containment," Revision 2
- 1-OPT-ZZ-07, "Position Indication Verification of Inside Containment Valves," Revision 0

Procedure 1-OPT-CT-211 is required to be performed prior to exceeding 200 degrees F and the other procedures are performed on a refueling outage frequency. The remote controls and position indication for containment isolation valves are clustered in a color coded area of the MCB. Valve positions are verified as part of the operators' turnover process. The inspectors reviewed the above procedures and verified that the position indication of several containment isolation valves accurately indicated the required position.

c. Conclusions

The main control board position indications for selected containment isolation valves satisfied procedural requirements.

O2 Operational Status of Facilities and Equipment

O2.1 Unit 1 Containment Spray System Walkdown

a. Inspection Scope (71707)

During the inspection period the inspectors performed a walkdown of accessible components associated with the Unit 1 Containment Spray (CS) System. The walkdown encompassed the CS pump suction and discharge piping and the chemical addition tank and associated piping. The inspectors referenced the system Piping and Instrument Diagrams (P&IDs) and valve alignment procedures for proper system alignment and component descriptions.

b. Observations and Findings

The inspectors checked system hangers and supports, general housekeeping, valve positions, labeling, and control room indications. The inspectors determined that the system was properly aligned and in generally good condition. The inspectors verified that procedure 1-OP-CS-001A, "Containment Spray System Alignment," Revision 1, adequately aligned the CS system for standby operation.

c. Conclusions

The Unit 1 Containment Spray System was properly aligned and in generally good condition.

08 Miscellaneous Operations Issues (92700, 92901)

08.1 Employee Concerns Program

a. Inspection Scope (40500)

The inspectors reviewed a new aspect of the licensee's employee concerns program involving reporting concerns via a toll free telephone number.

b. Observations and Findings

During the inspection period, the inspectors reviewed a new aspect of the licensee's employee concerns program. Effective May 1, the licensee added a toll free telephone number which allows employees to confidentially, or anonymously, report safety concerns to the company. A pamphlet describing this process was prepared and distributed to the Nuclear Business Unit employees. On August 5, the inspectors surveyed a few employees from the majority of the on-site departments to determine if employees had been made aware of the new reporting system. The results were mixed. The vast majority of the employees polled in the maintenance, operations, and security departments were not aware of the new phone number. This information was brought to the attention of station management, nuclear oversight, and the individual at Virginia Power Corporate Office who administers the licensee's employee concerns program. Station management stated that corrective actions would be taken to ensure that on-site personnel were aware of the telephone reporting system.

c. Conclusions

The inspectors concluded that the recently implemented employee concerns telephone reporting system was a good idea; however, the licensee's efforts in informing employees of the new service had not been effective in some groups.

08.2 (Closed) LER 50-280/96002-00: containment isolation valve inoperable greater than Technical Specification (TS) requirements due to personnel error. On March 3, 1996, with Unit 1 at 100 percent power, an Operations Periodic Test (OPT) 1-OPT-22.011, was performed on containment isolation trip valve 1-RC-TV-1519A. On March 5, 1996, while performing a review of the above OPT, a discrepancy was identified in that the recorded stroke time of 6.62 seconds was outside the TS limits. A retest resulted in declaring the valve inoperable and entry into a four hour TS 3.8.C action statement. The containment penetration was isolated as required by TS.

The licensee took immediate and long term action associated with this event. The inspectors concluded, based upon reviews of associated records and interviews, that the corrective actions were appropriate and adequately implemented.

Failure to isolate and deactivate containment isolation trip valve 1-RC-TV-1519A within fours hours is a violation of TS 3.8.C. This non-repetitive, licensee-identified and corrected violation is being treated as an Non-cited Violation (NCV) consistent with Section VII.B.1 of the NRC Enforcement Policy. This is identified as NCV 50-280/97007-01.

08.3 (Closed) EA 96-231/01013, 01023: unit operation with inoperable hydrogen analyzers and the Emergency Operating Procedure (EOP) did not provide instructions for function switch positions. The licensee's corrective actions were documented in a letter dated September 12, 1996. The inspectors reviewed the associated documentation including the associated LER, procedural changes, and training records and found that corrective actions had been adequately implemented.

08.4 (Closed) EA 95-223/01013, 01023, 01033: severity level III problem associated with failure to follow administrative procedures, failure to properly control maintenance activities, and failure to follow operating procedures during a refueling outage. Corrective actions included procedural changes to better integrate and control outage activities, revisions to the licensed operator and shift technical advisor continuing training programs, development of additional operations standards to communicate managements expectations, and outage preparatory training. The inspectors reviewed the licensee's corrective actions and found that they were adequate.

08.5 (Closed) LER 50-280, 281/96004-00, 01: hydrogen analyzers inoperable due to procedural deficiencies caused by personnel error. The inspectors verified by observation that the function switches on the local and remote panels were in the SAMPLE position. The main power switch on the main control room post accident monitoring panel was maintained in the STANDBY position. A review of the TS and TS basis, vendor manual, and the associated procedures show that implementation of corrective actions had been accomplished.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Unit 1 A Inadequate Core Cooling Monitor (ICCM) Power Supply Replacement

a. Inspection Scope (62707)

The inspectors observed maintenance performed on the Unit 1 A ICCM power supply.

b. Observations and Findings

On August 5, 1997, the inspectors observed Instrument & Control (I&C) technicians troubleshoot a problem with the plasma display for the A ICCM system. Work Order (WO) 370977-01 was issued to allow the technicians to troubleshoot the problem. The technicians had diagnosed a defective low voltage power supply to be the source of the problem. The inspectors observed the technicians pre-job briefing which was thorough. The technicians had procedure IMP-C-RC-124, "Checking, Repairing or Replacing a Component in the ICCM System," Revision 5, at the job site and utilized it.

The replacement low voltage power supply did not solve the problem. The technicians concluded that both the low and high voltage power supplies had to be replaced. Power supply replacements solved the problem with the plasma display.

c. Conclusions

The inspectors noted that the technicians were cautious and were consistent in their repeat backs. The inspectors consider that the technicians utilized good work practices and did a good job.

M1.2 Repair of Number 1 Emergency Diesel Generator (EDG) Radiator Louvers

a. Inspection Scope (62707)

The inspectors reviewed maintenance activities performed on the Number 1 EDG radiator louvers.

b. Observation and Findings

The licensee was experiencing high coolant temperature alarms on EDG Number 1 and the operators observed that one set of radiator louvers did not operate properly. WO 370314 was issued to troubleshoot the problem and on August 12, 1997, troubleshooting was initiated. The electricians determined that the louver controller was defective and replaced the controller. Electrical Corrective Maintenance (ECM) procedure 0-ECM-0701-01,"Emergency Diesel Generator Maintenance," Revision 3, contained temperature settings for the controller in Ohmic values which converted to 160 degrees F. The licensee recognized that this value was at the low end of the allowable band and an Engineering Transmittal (ET) was generated to change the listed setpoint in the licensee's setpoint program. ET S-97-0291, Revision 0, was issued to provide a revised setpoint of 175 degrees F which Engineering based on the Vendor Technical Manual (VTM).

The inspectors observed a portion of the restoration effort and noted that the craft had the WO and the procedure at the job site but did not have the applicable pages of the VTM. The inspectors reviewed the WO at the job site and the completed document, procedure 0-ECM-0701-01, ET S-97-0291, and VTM 38-EO-35-00001, pages 214 and 215. The VTM stated that the allowable temperature range for the controller was 160 degrees F to 190 degrees F and the controller should be set at 175 degrees F. The ET specified the same temperature range and setpoint as the VTM, but added a tolerance of -0/+5 degrees F.

The new louver controller was installed and EDG Number 1 was started at approximately 2:00 a.m., on August 13, for Post Maintenance Testing. The EDG was run for approximately 2 hours and jacket water temperature was 178-180 degrees F and there were no high temperature alarms. The louvers opened as required and maintained temperature within band with the controller temperature set at 175 degrees F. The EDG was declared operable at 3:30 a.m.

c. Conclusions

Personnel performing maintenance activities on the Number 1 EDG louvers had drawings, procedures, and WOs at the job site. There was adequate management and technical support. The inspectors expressed concern to licensee management that the VTM recommended setpoint had not previously been incorporated into the maintenance procedure.

M1.3 Surveillance Observations (61726)

On August 13, the inspectors observed portions of procedure 1-PT-8.1, "Reactor Protection System Logic (For Normal Operations)," Revision 14, being performed. The inspectors noted that the I&C technicians used the procedure. The technicians kept the operators informed and the Control Room Operator kept the Senior Reactor Operator informed. Repeat backs were evident and the inspectors considered that communications were very good. The inspectors observed that a clipboard was frequently passed over the control board. This observation was discussed with Operation's management who stated that they did not consider it to be a good practice and would address the issue with the operators. The surveillance was considered to be successful.

On August 13, the inspectors observed the performance of procedure 0-OPT-SW-002, "Emergency Service Water Pump, 1-SW-P-1B," Revision 10 P1. The pump flow was at the low end of the acceptable range. The surveillance was considered to be acceptable, but the pump was placed in an alert status.

On August 20, the inspectors observed portions of the Unit 1 Turbine Driven Auxiliary Feedwater Pump (1-FW-P-2) monthly surveillance performed in accordance with procedure 1-OPT-FW-003, "Turbine Driven Auxiliary Feedwater Pump 1-FW-P-2," Revision 9. The inspectors observed that the pump flow met the surveillance requirements. The surveillance was considered to be acceptable.

Three surveillance activities observed by the inspectors were accomplished in accordance with approved procedures and were completed satisfactorily.

M1.4 Alternate Alternating Current (AAC) Diesel Generator Testing

a. Inspection Scope (61726)

The inspectors observed portions of procedure 0-OSP-AAC-001, "Quarterly Test of 0-AAC-DG-0M, Alternate AC Diesel Generator," Revision 4.

b. Observations and Findings

On August 18, the inspectors observed the performance of procedure 0-OSP-AAC-001, "Quarterly Test of 0-AAC-DG-0M, Alternate AC Diesel Generator," Revision 4. The inspectors attended the pre-job briefing conducted in the control room, observed activities at the AAC diesel and reviewed the completed test data.

During the prestart checks required by the procedure, the operator determined that engine lube oil level did not meet the procedure requirement specified. Step 6.1.16 required that engine lube oil level be between the add mark and the full mark of the dipstick (engine stopped and oil cold). Actual oil level was slightly above the full mark. The procedure was halted until the system engineer was contacted and a one time procedure change was initiated to allow continuation of the test. The inspectors discussed the oil level requirement with the system engineer and reviewed the procedure change. The inspectors determined that operation with the slightly elevated oil level was acceptable.

During the four hour loaded run required by the procedure, jacket water outlet temperature and oil cooler water outlet temperature exceeded the normal temperature band specified in the procedure within one hour of operation. Oil cooler water outlet temperature stabilized at 149 degrees F, then later trended down to 145 degrees F 2.5 hours into the run and remained at this value for the remainder of the test. Maximum oil cooler water outlet temperature specified in the procedure was 145 degrees F. Jacket water outlet temperature stabilized at 205 degrees F then trended down to 200 degrees F 3 hours into the run and remained at this value for the remainder of the test. Maximum jacket water outlet temperature specified in the procedure was 200 degrees F. All other operating parameters were within specified normal limits.

The system engineer was contacted and responded to the AAC diesel building to determine why cooling water temperatures were elevated. Initial efforts focused on the radiator fans located on the roof of the AAC diesel building. Both fans were operating and turning at the required rpm. While operations was checking the operation of the fans, the operator determined that temperature under the radiators (fan intake) was elevated and that diesel exhaust fumes were being drawn into the radiators. The diesel exhaust piping and silencer are at approximately the same elevation and in close proximity to the intake of the radiator fans. The inspectors observed that temperatures were elevated at the intake to the radiators. The inspectors also noted that outside air temperature dropped noticeably during the four hour period that the diesel was in operation.

Subsequent to the diesel run, an Installation Problem Report was submitted to isolate or reconfigure the diesel engine exhaust to prevent exhaust gasses from being drawn across the radiator. Engineering also generated ET 97-0295 to justify operability of the AAC diesel. The ET stated that the elevated cooling water temperatures were due to exhaust gasses being drawn into the radiators due to the prevailing wind direction. During the diesel run, wind direction was initially from 285 degrees F and then shifted to 324 degrees F approximately one hour and 20 minutes into the run. The ET also stated that operation of the diesel at the temperatures observed was acceptable. Engine coolant temperatures were below the alarm setpoint of 208 degrees F and an automatic diesel shutdown would not occur until 219 degrees F. This ET was reviewed and approved by plant management at a Station Nuclear Safety and Operating Committee (SNSOC) meeting on August 19.

The inspectors attended the SNSOC meeting that approved the ET and reviewed the approved document. The inspectors questioned the conclusions reached in the transmittal. In particular, the inspectors questioned the statement that under the worst case ambient temperature and wind direction, the AAC diesel would still function as designed without reaching the shutdown setpoint for engine coolant temperature. The inspectors were unable to determine the basis for this conclusion. The inspectors also questioned the conclusion that the drop in ambient temperature had no effect on coolant temperatures. These items were discussed with the licensee. At the conclusion of the inspection period the inspectors had not been provided documentation supporting the conclusions reached in the ET, however; the licensee was in the process of reviewing the inspectors' concerns and planned to revise the ET.

The AAC diesel generator is not a TS required component. The licensee has a 14 day administrative limit on diesel operability. If the diesel is inoperable greater than 14 days the station administrative procedure requires that a special report be submitted to the NRC within 30 days. Until the inspectors further review AAC diesel coolant temperature concerns and long term actions to resolve this issue, this is identified as Inspection Followup Item (IFI) 50-280, 281/97007-02.

c. Conclusions

AAC diesel generator testing was conducted in accordance with approved procedures. Problems encountered during performance of the test were properly dispositioned by the operating crew. During performance of the test, elevated coolant temperatures were observed and were attributed to diesel exhaust entering the radiator fan suction. The engineering transmittal addressing the issue lacked documentation to justify the conclusions reached. An IFI was identified to review this item further.

The SNSOC meeting that approved the ET did not sufficiently challenge the conclusions reached or require further review or testing to determine the extent of the problem. Based upon previous observations, matters reviewed by SNSOC were normally thorough. This instance is considered an isolated weakness.

M1.5 Number 1 Emergency Diesel Generator Monthly Test

a. Inspection Scope (61726)

On August 3, the inspectors observed the performance of the monthly testing of the Number 1 EDG.

b. Observations and Findings

On August 3, the inspectors observed portions of the performance of surveillance procedure 1-OPT-EG-001, "Number 1 Emergency Diesel Generator Monthly Start Exercise Test," Revision 8-P1. The test was completed satisfactorily and in accordance with the surveillance procedure. However, during the testing with the engine at full load, an "EDG 1 TRBL" alarm was received in the control room. Investigation at the engine revealed that the hot engine alarm was lit. The condition rectified itself when the second set of louvers on the engine's radiator opened reducing coolant temperature to less than 180° F. The engine's radiator louver controller is designed to control engine coolant temperature below 180° F. However, if the controller fails to maintain temperature, the louvers are designed to fully open (as demonstrated during the test) and keep engine temperature below the alarm set point. Because the engine has the design feature which fully opens the radiator louvers on a high engine temperature, operability of the engine was not challenged.

This same problem occurred during the previous month's (July) engine test and was supposed to have been corrected during the next month's (August) surveillance test. Due to a scheduling error, the repair was scheduled in late 1998. This situation reflects a lack of ownership of the component and is considered a weakness in the both system engineering and scheduling. The louver controller was repaired following the test and is discussed further in section M1.2.

c. Conclusions

Testing of the Number 1 EDG was conducted satisfactorily. The failure to ensure that repairs were scheduled for a problem recognized during the previous monthly test indicates a lack of ownership of the component and is considered a weakness in both system engineering and scheduling.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Flow Accelerated Corrosion (FAC) Program

a. Inspection Scope (IP 49001)

The inspectors reviewed procedures, records, and documents related to the monitoring of FAC in secondary piping and components at the Surry Nuclear Power Station. The review included the most recent results of FAC inspections of both units, as well as the plans for the next FAC inspection of Surry Unit 2.

This inspection was scheduled to review the licensee's actions in response to the licensee's DR (DR S-97-0895) generated during an NRC inspection of Surry Unit 1, FAC inspection activities during the Spring 1997 Refueling Outage (RFO). (Ref. Section M2.3 of NRC Integrated Inspection Report No. 50-280, 281/97003).

b. Observations and Findings

The licensee's corrective action plan for DR S-97-0895 was designed to provide a detailed self-assessment of the inspection program, along with independent reviews of the program by Electrical Power Research Institute (EPRI) and Institute of Nuclear Power Operators. The DR was generated because of the perception that the "Secondary Piping & Component Inspection Program," (as the FAC inspection program is formally titled) had identified a larger number of inspected components that required replacement than normal. The lead responsibility for corrective action for the DR was assigned to the licencee's corporate engineering staff, with December 31, 1997, as the scheduled completion date for all corrective actions.

The inspectors reviewed the following procedures, reports, and documents relative to the FAC inspection program and the corrective action for DR S-97-0895:

- Virginia Power Administrative Procedure VPAP-0807, "Secondary Piping and Component Inspection Program," Revision 0, October 1, 1992.

- General Nuclear Standard, STD-GN-0033, "Secondary Piping and Component Inspection Program," Revision 6, January 22, 1997.

- Technical Report No. ME-0107, "Surry Unit 1, 1997 Refueling Outage, Results of Secondary Piping & Component Inspection Program," Revision 0, July 17, 1997.

- Technical Report No. ME-0103, "Surry Unit 2, 1996 Refueling Outage, Results of Secondary Piping & Component Inspection Program," Revision 0, December 16, 1996.

- EPRI Letter Report, "Supplemental Assistance to Virginia Power's Program to Control Flow-Accelerated Corrosion at the Surry and North Anna Nuclear Power Plants," June 20, 1997.

- ET CME-0016, "Updated Preliminary Piping Inspection List - Surry Power Station - Unit 2 - 1997 Refueling Outage," Revision 0, April 28, 1997.

- ET CME-0016, "Updated Preliminary Piping Inspection List - Surry Power Station - Unit 2 - 1997 Refueling Outage," Revision 1, August 12, 1997.

- ET CME 94-012, "Core Uprate Impact on Flow Accelerated Corrosion, Surry Power Station Units 1 & 2," May 17, 1994.

- Memorandum, "Significant Event Report, Surry Unit 1 Sulfate Intrusion," July 5, 1996.

- ET CME-97-0009, "Secondary Piping and Component Inspection Program Component Critical Life, Surry Power Station - Unit 1," Revision 0, February 3, 1997.

- ET CEM-97-0009, "18" Reinforced Fabricated Tee Replacement, Surry Unit 1," Revision 0, February 10, 1997.

Procedure VPAP-0807 and STD-GN-0033 constitute the licensee's program for control of FAC. The "Secondary Piping & Component Inspection

Program" has historically been the responsibility of the licensee's corporate engineering staff.

Technical Report Nos. ME-0107 and ME-0103 were the final engineering reports detailing the inspection and replacement of secondary piping and piping components during the most recent outages at each of the Surry units. The EPRI letter report described the review of the licensee's program that was conducted as a part of the corrective action to DR S-97-0895. The remaining documents were examples of correspondence documenting preparations for, and implementation of, parts of the program.

Along with the text of Technical Report No. ME-0107 for Surry Unit 1, the inspectors also reviewed the data sheets that were a part of the report. (There were ~140 data sheets for the component inspections, and ~60 data sheets for baseline inspections of replaced components.) The review of the inspection data sheets was conducted in an effort to understand the scope of the inspection, and how the results were categorized.

The inspectors noted that the advertised inspection sample size of 140 components was misleading in that it under represented the actual scope of the work conducted during the outage. Because FAC is most prevalent where there are changes in the piping configuration, the licensee's program requires that a segment of the upstream and downstream components, attached to the sample, or "main-component," be inspected along with the sample component. (In some cases, the upstream or downstream component was found to be the most severely eroded.)

The concern, which caused the generation of DR S-97-0895, was that prior to the outage, engineering scheduled 20 components to be replaced, and inspections found an additional 22 components to be replaced. This made it appear as if there were surprises in the inspection findings. Through the review of the documentation, and discussions with the engineering personnel responsible for the program, the inspectors were convinced that there were no surprise findings during the Spring 1997, Surry Unit 1 outage. Part of the problem was in the way pre-outage inspection information was transmitted to the site; in that 9 of the 22 additional components were predicted to need replacement, but had not been transferred from the inspection list to the replacement list. The additional 13 components appeared to be ones that were known to be approaching their margin and were inspected to check predicted corrosion rates.

c. Conclusions

The FAC inspection program for Surry is a well-organized, mature program, with inspection-established growth rates, and program-predicted life expectancies for all major carbon steel components in the secondary piping systems.

The FAC inspection scope, inspection expectations, and in some cases inspection results, were not clearly communicated to plant management and other people outside of the FAC inspection program.

M8 Miscellaneous Maintenance Issues (92902, 90712)

M8.1 Corrective Action for Equipment Problems

a. Inspection Scope (62700)

This portion of the inspection was conducted to review the licensee's corrective actions for equipment problems. The inspection was conducted primarily through the review of corrective actions for hardware problems identified on station DRs. In order to complete the inspection, the licensee was requested to provide a listing of all station deviations written in November 1996, January 1997, and June 1997. This listing was reviewed by the inspectors in an effort to select a sample of hardware deviations for a more detailed review of corrective actions, and also to identify any trends in equipment performance that the licensee should be addressing. The inspectors selected a sample of approximately 15 deviations and completed a detailed review of corrective actions. The inspectors also reviewed the Deviation Trending Reports for the fourth quarter 1996 and the first quarter 1997 in order to determine if the licensee was identifying and responding to negative trends in equipment performance which were observed by the inspectors. The inspectors also reviewed the licensee's procedures applicable to this area, which included VPAP-1601, "Corrective Action," Revision 7, and VPAP-1501, "Deviation Reports," Revision 8.

b. Observations and Findings

The inspection resulted in the following observations and findings: No problems were noted during the review of the corrective actions for the sample deviations. Review of the trending of deviations determined that adverse trends were being appropriately identified, and corrective actions for the identified trends were being adequately implemented.

Trending of deviations was assessed as a strength in the licensee's corrective action program.

c. Conclusions

No problems were observed regarding corrective actions for equipment problems. Trending of station deviations and corrective action for adverse trends identified by that process were assessed as a strength.

M8.2 (Closed) Licensee Event Report (LER) 50-281/95003-01: transmitters out of calibration due to use of a gauge that was not temperature compensated. This item will be closed by the closure of Enforcement Action (EA) 95-053 Violation 01014 in Section M8.3.

M8.3 (Closed) EA 95-053 Violation 01014: minimum number of pressurizer pressure instruments not operable. On February 10, 1995, during the Unit 2 RFO, I&C technicians performing calibration checks discovered the "as-found" data on the first of three pressurizer pressure protection transmitters was not within the allowable tolerance specified in the calibration procedure. The technicians repeated the test with another gauge and obtained the same results. The "as-found" calibration data for the three pressure transmitters were found to be reading between 24 and 30 psig high. DR 95-2148 was issued to track the issue. On February 23, 1995, a Root Cause Evaluation (RCE) team was assembled in parallel with a request for a safety assessment by the Nuclear Analysis and Fuels Department. The Nuclear Analysis and Fuels Department evaluation determined that the condition observed had been bounded by a previous evaluation for power operations. However, minimum pressurizer pressure requirements would have been exceeded for the pressurizer low-pressure reactor trip and the pressurizer low-low pressure safety injection actuation during postulated transients. Both events are prohibited by TS and LER 50-281/95003-00 was issued to document this condition.

The three pressurizer transmitters had been installed on June 18, 1994, and field calibrated with Unit 2 in Cold Shutdown. On June 24, 1994, with Unit 2 in Hot Shutdown (2235 psig and 547 degrees F) the technicians made calibration adjustments to the three pressurizer transmitters. The unit operated continually until it entered its RFO on February 3, 1995. The RCE team determined that the final calibration of the transmitters was performed in a sub-atmospheric condition (9.7 psia). However, these transmitters are different than other transmitters on site in that they have a sealed reference side instead of being open to the atmosphere. One side of the transmitters saw 14.7 psia and the other side saw 9.7 psia which resulted in an instrument span shift. It was also determined that the gauge that was used to measure pressure was not temperature compensated which resulted in an additional 10 psi shift. This event is described in more detail in Section 5.2 of NRC Inspection Report No. 50-280, 281/95-06 and LERs 50-281/95003-00 and 01.

The RCE team concluded that the pressurizer low-pressure reactor trip and the pressurizer low-low-pressure safety injection functions had been inoperable from June 24, 1994, to February 4, 1995. The RCE team concluded that there were multiple causal factors including: inadequate training of the Metrology Laboratory staff, inadequate procedures which failed to address calibration in sub-atmospheric conditions, improper testing of Measuring and Test Equipment (M&TE), and poor work practices. The team recommendations included: calibration procedures be reviewed and revised if necessary; additional training for maintenance and M&TE personnel be provided; elimination of non-temperature compensated gauges; and perform an assessment/audit of the M&TE program. Quality Assurance (QA) performed an assessment of the site M&TE program which was documented in QA Report No. S95-11. In addition, Station Nuclear Safety (SNS) performed a self assessment of the Site Deviation Corrective Action Program. The results of this assessment were documented in a SNS report dated February 13, 1995. The assessors addressed programmatic process issues and did not address program implementation issues.

The inspectors reviewed the licensee's response dated June 15, 1995, to the violations identified in NRC Inspection Report No. 50-280, 281/95-06. The licensee's response identified that their corrective actions for these violations were identified in RCE 95-04 and LER 50-281/95003-01. The inspectors also reviewed RCE 95-04 and the corrective action items which had been assigned to the Commitment Tracking System (CTS). The lesson plans for maintenance and M&TE continuing training were reviewed. Continuing training is mandatory and the training verification documents were reviewed. The inspectors also reviewed vendor test reports, ET 95-0887, Revision 0, revised purchasing specifications, revised procedures and other related documents. QA Report No. S95-11 was reviewed including all the QA follow up documents in which QA verified that the corrective actions for the 10 findings from the assessment had been completed. The inspectors considered that the QA M&TE assessment and RCE 95-04 were excellent licensee products, in that they were thorough and addressed issues in depth.

CTS item 2963 had been issued to Corporate Engineering to review digital Heise gauge usage and assist in the evaluation of environmental effects on M&TE and procedure revisions. Corporate Engineering reviewed all of the pressurizer pressure instruments inside containment. The existing calculations associated with the identified pressure transmitters were reviewed to determine if atmospheric pressure changes had been taken into account in the Channel Statistical Analysis (CSA). ET No. CEE 97-010, Revision 0, documented the review. Pressure transmitters whose signals are used in safety analysis and/or for Emergency Operations Procedure (EOP) usage were considered significant. Corporate Engineering determined that the CSAs for significant pressure transmitters should be revised. Engineering also recommended that the reactor coolant wide range loop pressure instruments (loops 1/2-RC-PT-1402 and 1/2-RC-PT-1402-1) should have their CSA assumptions revisited. Corporate Engineering determined that the five psi variation in ambient pressure experienced by the loop pressure transmitters as containment moves from sub-atmospheric to atmospheric conditions does not cause a safety concern when that change, reflected in the loop error calculation, is used in determining setpoints which appear in the Surry EOPs. ET No. CEE 97-043, Revision 0, documented this review. CTS item 2963 will be closed and the completion of the CSAs for the subject pressure transmitters will be tracked by Engineering Task Tracking (ETT) No. 97-0301. The inspectors reviewed ETs CEE 97-010 and 97-043 and found them adequate.

The inspectors verified that all of the corrective actions had been completed. The inspectors noted that these corrective actions also address the item covered by Section M8.4.

M8.4 <u>(Closed) EA 95-053 Violation 02014</u>: failure to adequately establish measures to identify and correct a pressurizer transmitter problem. On February 10, 1995, during the Unit 2 RFO, I&C technicians calibration checks discovered the "as-found" data, for the first of three pressurizer pressure protection transmitters to be calibrated, was not within the allowable tolerance specified in the calibration procedure. The RCE team determined that on June 24, 1994, I&C technicians issued DR 94-1352 which identified an indication discrepancy between the Unit 2 pressurizer pressure control and protection channels. I&C personnel concluded that the Heise gauge may have been misread when the pressurizer protection transmitters were calibrated. The calibration was checked and found too low by approximately 30 psig, and the pressurizer protection transmitters were adjusted.

On June 25, 1994, Operations submitted DR 94-1353 to document a pressurizer low pressure alarm during startup. The DR also noted that the pressurizer protection channels were indicating approximately 15 to 20 psi higher than the pressurizer control channels.

Both DRs were assigned to the I&C Department to determine the cause and provide corrective actions. I&C personnel mistakenly assumed that both DRs described the same condition. DR 94-1353 was closed out on July 14, 1994, as it was considered to be redundant to DR 94-1352 which was closed out on August 4, 1994, based on the fact that it was a personnel error in reading the Heise gauge. The RCE determined that the cause of the event was a binding linkage in the Heise gauge.

The licensee's corrective actions included counseling personnel on properly dispositioning DRs and increased training on awareness on M&TE issues. The inspectors reviewed management letters and the lesson plans for the training given during continued training. Section M8.3 contains other corrective actions for this event. The inspectors consider the licensee's corrective actions to be adequate.

M8.5 (Closed) LER 50-280/96003-00: control room air handling units inoperable due to mechanical failure. The inspectors verified by review of the RCE, engineering evaluations, procedural changes as recommended by engineering, and observation of unit operations that corrective actions have been adequately implemented.

III. Engineering

E1 Conduct of Engineering

E1.1 Unit 1 Containment Hydrogen Concentration (37551)

a. Inspection Scope

The inspectors reviewed the licensee's actions related to detectable hydrogen concentration within the Unit 1 containment.

b. Observations and Findings

Following the return of Unit 1 to service from a RFO in May 1997, the licensee determined that the upper pressurizer manway was not properly sealed and was leaking slightly. Reactor Coolant System leakage was well below TS requirements. As a precaution, the licensee began performing containment air samples to check hydrogen concentration. A sample performed on June 12 indicated hydrogen concentration levels in containment to be approximately 0.37 percent. A follow-up sample on June 26 indicated that hydrogen levels were remaining stable. However, a sample performed on July 10 indicated that hydrogen concentration levels had increased to approximately 0.54 percent. Based on the increasing trend of hydrogen concentration, the licensee began evaluating the impact of hydrogen in the containment on the accident analysis and its effect on continued operation. The sample performed on July 17 indicated a hydrogen concentration of 0.6 percent. The licensee established an upper administrative limit for containment hydrogen concentration at 0.8 percent. These evaluations were documented in ET No. NAF-970185, "Justification for Continued Operation (JCO) S1-97-002," and Safety Evaluation 97-091.

This matter was reviewed in detail by the inspectors. A number of questions were raised by the inspectors. These included; 1) the basis for the acceptability of the 0.8 percent administrative limit, and 2) if adequate mixing was occurring in containment preventing the formation of hydrogen pockets with increased concentrations. The Surry accident analysis for a large break Loss of Coolant Accident (LOCA) assumes that hydrogen concentration prior the accident is below the lower detectable limit of 0.2 percent. The inspectors questioned how having a higher starting hydrogen concentration than assumed by the accident analysis was acceptable for continued operation. The licensee demonstrated that the accident analysis assumed that the hydrogen recombiners had a capability to process 50 Standard Cubic Feet Per Minute (SCFM) of containment gases, while in actuality the installed recombiner had a capacity of at least 60 SCFM. This extra capacity would provide adequate margin to maintain post accident hydrogen levels below the 4 percent explosive limit with an initial hydrogen concentration of 0.8 percent or less. The licensee also demonstrated that with the air flow into the pressurizer cubicle, adequate mixing of the hydrogen and containment atmosphere would occur preventing the formation of hydrogen pockets. In addition to the review by the resident staff, the licensee's safety evaluation, JCO, and ET were reviewed by the NRC's Office of Nuclear Reactor Regulation. No additional concerns were raised.

Following the licensee's adoption of the 0.8 percent hydrogen concentration administrative limit, the licensee began a process of bleeding and feeding the containment atmosphere. This was performed by removing air from the containment with containment vacuum pumps, and returning air to the containment via the containment instrument air compressors. This process was initiated on approximately July 19, with containment hydrogen concentration at approximately 0.6 percent. Following initiation of the bleed and feed of containment, hydrogen concentration initially continued to increase. Hydrogen concentration peaked at 0.64 percent and then began a downward trend plateauing at approximately 0.52 percent.

Because the current Unit 1 operating cycle will not end until October 1998, the licensee sought out a way to continuously remove hydrogen from containment without shutting down and repairing the pressurizer manway. The licensee found a commercially produced product, a Portable Autocatalytic Recombiner (PAR), which could be installed inside containment and through an auto-initiated reaction, would combine the hydrogen in containment with oxygen to form water. A PAR was purchased and installed in the Unit 1 containment on August 22. Through data obtained from the PAR manufacturer, the licensee expects that within three months hydrogen levels within containment will be less than 0.2 percent.

c. Conclusions

The inspectors concluded that the licensee performed an adequate evaluation and implemented appropriate actions to alleviate concerns with hydrogen concentration within the Unit 1 containment.

E8 Miscellaneous Engineering Issues (92700)

E8.1 (Closed) LER 280/96001-00: emergency service water pump inoperable due to loss of missile protection for piping. This event was reported to the NRC following the licensee's determination that Emergency Service Water (ESW) Pump 1A was inoperable for an extended period of time (in excess of the TS Limiting Condition for Operation (LCO) permitted time) due to a loss of missile protection for the pump's discharge piping. Specifically, a trench was made to perform repairs to piping in the non-safety related fish screen system. In performing this action, the licensee failed to recognize that the minimum soil depth of 5 feet required for missile protection of the ESW pump discharge piping was not in place. The pump was declared inoperable, compensatory actions were promptly taken, and the pump was returned to operable status that same day.

The licensee's corrective actions for this event entailed the following: 1) revising procedure GMP-C-102, "Excavation, Backfill, and Subgrade Preparation," Revision 3, to include instructions to ensure compensatory measures are in place for loss of missile protection due to having less than the required 5 feet of earth coverage, 2) coaching engineering personnel involved with the design change process to reinforce the requirements of General Nuclear Standard STD-GN-0001, "Design Change Preparation," and 3) posting signs at the Low Level Intake structure in the vicinity of the ESW pump discharge lines indicating that the soil provides missile protection for these lines. The inspectors reviewed these actions and found them to be satisfactory.

The failure to provide missile protection for ESW piping is a violation of TS 3.14.B in that ESW Pump 1-SW-P-1A was inoperable for a period of approximately 36 days. This non-repetitive, licensee-identified and corrected violation is being treated as a Non-cited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy. This matter is identified as NCV 50-280/97007-03.

E8.2 <u>(Closed) LER 50-281/96003-00</u>: unit 2 Pressurizer Safety Valve (PSV) as-found lift setting out of tolerance. On May 12, 1996, with Unit 2 in a refueling shutdown, setpoint testing on PSV 2-RC-SV-2551B revealed that the as-found lift setting was out of tolerance at 2374 psig. TS 3.1.A.3.b requires the as-found setting be 2485 psig $\pm$ 3 percent (2411 to 2559 psig). The inspectors reviewed the test results performed by engineering and the results of the disassembly and inspection by the vendor. The as-left lift settings were within TS requirements. The inspectors determined that the tests performed by the licensee and the vendor were adequate to resolve the particular problem. Setpoint drift is a phenomenon that is recognized by the licensee and the NRC.

<u>IV. Plant Support</u>

R1 Radiological Protection and Chemistry Controls (71750)

On numerous occasions during the inspection period, the inspectors reviewed Radiation Protection (RP) practices including radiation control area entry and exit, survey results, and radiological area material conditions. No discrepancies were noted, and the inspectors determined that RP practices were proper.

S1 Conduct of Security and Safeguards Activities (71750)

On numerous occasions during the inspection period, the inspectors performed walkdowns of the protected area perimeter to assess security and general barrier conditions. No deficiencies were noted and the inspectors concluded that security posts were properly manned and that the perimeter barrier's material condition was properly maintained.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on September 3 and 11, 1997. On September 11, the Station Manager disagreed with the appropriateness of including the FAC program communication misunderstanding as a finding in the report's Executive Summary. Specifically, the Station Manager stated that the finding had no regulatory significance.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. Adams, Superintendent, Engineering
R. Allen, Superintendent, Maintenance
G. Bischof, Director, Design Engineering
R. Blount, Assistant Station Manager, Nuclear Safety & Licensing
I. Breedlove, Senior Staff Engineer, NES/ME
D. Christian, Station Manager
M. Crist, Superintendent, Operations
C. Hooper, Technical Analyst, NES/ME
M. Kansler, Vice President, Nuclear Operations
G. Miller, Corporate Licensing
L. Miller, Senior Staff Chemist, Nuclear Licensing & Operations Support
B. Shriver, Assistant Station Manager, Operations & Maintenance
D. Sommers, Supervisor Corporate Nuclear Licensing
T. Sowers, Superintendent, Training
B. Stanley, Director, Nuclear Oversight
W. Thorton, Superintendent, Radiological Protection

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 49001:	Inspection of Erosion/Corrosion Monitoring Programs
IP 61726:	Surveillance Observation
IP 62700:	Maintenance Implementation
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 90712:	Inoffice Review of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901:	Followup - Plant Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering

ITEMS OPENED AND CLOSED

Opened

50-280/97007-01	NCV	inoperable containment isolation valve (Section 08.2).
50-280, 281/97007-02	IFI	AAC diesel coolant temperature concerns and long term actions to resolve the issue (Section M1.4).
50-280/97007-03	NCV	inoperable ESW pump due to loss of missile protection (Section E8.1).

Closed

50-280/96002-00	LER	containment isolation valve inoperable greater than Technical Specification (TS) requirements due to personnel error (Section 08.2).

96-231/01013, 01023	EA	unit operation with inoperable hydrogen analyzers and the EOP did not provide instructions for function switch positions (Section 08.3).
95-223/01013, 01023, 01033	EA	severity level III problem associated with failure to follow administrative procedures, failure to properly control maintenance activities, and failure to follow operating procedures during a RFO (Section 08.4).
50-280, 281/96004-00, 01	LER	hydrogen analyzers inoperable due to procedural deficiencies caused by personnel error (Section 08.5).
50-281/95003-01	LER	transmitters out of calibration due to use of a gauge that was not temperature compensated (Section M8.2).
95-053/01014	EA	minimum number of pressurizer pressure instruments not operable (Section M8.3).
95-053/02014	EA	failure to adequately establish measures to identify and correct pressurizer transmitter problem (Section M8.4).
50-280/96003-00	LER	control room air handling units inoperable due to mechanical failure (Section M8.5).
50-280/96001-00	LER	emergency service water pump inoperable due to loss of missile protection for piping (Section E8.1).

50-281/96003-00	LER	unit 2 pressurizer safety valve as-found lift setting out of tolerance (Section E8.2).
50-280/97007-01	NCV	inoperable containment isolation valve (Section 08.2).
50-280/97007-03	NCV	inoperable ESW pump due to loss of missile protection (Section E8.1).