UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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January 12, 2001

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 INSPECTION REPORT NOS. 50-280/00-09, 50-281/00-09

Dear Mr. Christian:

On December 15, 2000, the NRC completed an inspection at your Surry Power Station, Units 1 and 2. The enclosed report presents the results of that inspection which were discussed on December 15, 2000, with Mr. Blount and other members of your staff.

The inspection was an examination of activities conducted under your licenses as they relate to the identification and resolution of problems, and compliance with the Commission's rules and regulations and with the conditions of your licenses. Within these areas, the inspection involved selective examination of procedures and representative records, observations of activities, and interviews with personnel.

On the basis of the sample selected for review, there were no findings of significance identified during this inspection. The team concluded that problems were properly identified, evaluated and resolved within the problem identification and resolution programs. However, during the inspection, several examples of minor problems were identified that included time to resolve issues and the adequacy of evaluations and resolutions.

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Sincerely,

/RA/

Kerry D. Landis, Chief
Reactor Projects Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report

Attachments: 1. Documents Reviewed
2. NRC's Revised Reactor Oversight Process

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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.:	50-280/00-09, 50-281/00-09
Licensee:	Virginia Electric and Power Company (VEPCO)
Facilities:	Surry Power Station, Units 1 & 2
Location:	5850 Hog Island Road Surry, VA 23883
Dates:	December 4 - 15, 2000
Inspectors:	L. Garner, Senior Project Engineer, Region II R. Gibbs, Senior Reactor Inspector, Region II K. Poertner, Resident Inspector, Surry Power Station
Approved by:	K. Landis, Chief, Reactor Projects Branch 5 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000280-00-09, IR 05000281-00-09, on 12/04-12/15/2000; Virginia Electric and Power Co., Surry Power Station, Units 1 & 2. Annual baseline inspection of the identification and resolution of problems.

The inspection was conducted by a senior project engineer and senior reactor inspector from Region II and a resident inspector from the Surry Power Station. No findings of significance were identified.

Identification and Resolution of Problems:

The licensee was effective at identifying problems and entering them into the corrective action program. The threshold for entering problems into the corrective action program was low. Operating experience was appropriately incorporated into plant procedures and activities. Generally, problems entered into the corrective action program were adequately evaluated and appropriate corrective actions were identified. Category 1 and 2 root cause evaluations, performed for the most and next most significant issues, respectively, were thorough and specified corrective actions were appropriate. Although risk was not formally used in prioritizing issues, corrective actions were usually implemented in a timely manner commensurate with their safety significance. Licensee audit and self-assessment results were consistent with NRC observations and identified deficiencies were entered into the corrective action program. Based on interviews, a safety conscious work environment was present where employees felt free to raise nuclear safety concerns. However, some negative observations were identified involving time to resolve some issues and the adequacy of some evaluations and resolutions. These negative observations involved issues that were of very low safety significance.

Report Details

4 OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

.1 Effectiveness of Problem Identification

a. Inspection Scope

The inspectors reviewed Unit 1 and Unit 2 operating logs from December 1, 2000, through December 15, 2000, to determine if deficiencies were being entered into the corrective action program. The inspectors also toured portions of the service water (SW), high head safety injection (HHSI), containment spray, component cooling, and auxiliary feedwater systems to determine if deficiencies existed that had not been entered into the corrective action program.

The inspectors reviewed items from the licensee's corrective action database associated with high risk significant components and systems identified in the plant specific probabilistic risk assessment. The components and systems selected for review included the high head and low head safety injection (HHSI and LHSI) systems, the emergency diesel generators (EDGs) and electrical distribution system and the service water (SW) system. In addition, the inspectors reviewed items involving the operations, maintenance, engineering, chemistry, emergency preparedness, and security organizations. Specifically, the inspectors selected 56 plant issues from approximately 3300 generated since October 1999 for review. The inspectors evaluated the plant issues to determine the licensee's threshold for identifying problems and entering them into the corrective action program. The plant issues reviewed are listed in Attachment 1.

The inspectors reviewed the 28 operating experience (OE) items identified in Attachment 1 to determine if they were appropriately evaluated for applicability and if problems identified through these reviews were entered into the corrective action program.

The inspectors reviewed a Nuclear Oversight audit of the corrective action program and three station level self-assessments to determine if audit and self-assessment findings were entered into the licensee's corrective action program.

b. Findings

During the selection of items to review, from plant issue reports, audits, self-assessments and other licensee documents, the inspectors observed that the licensee's threshold for identifying problems and entering them into the corrective program was sufficiently low. In addition to capturing self-disclosing events, equipment failures and human errors, formal processes such as audits, self-assessments performed by the licensee's staff and outside organizations, and OE reviews were effective in identifying issues. Visual inspection of various safety-related and risk significant components revealed that material condition was satisfactory and minor equipment deficiencies were identified as evidenced by attached deficiency tags.

.2 Prioritization and Evaluation of Issues

a. Inspection Scope

The inspectors reviewed the plant issues, audit and self-assessments discussed in the inspection scope of section 4OA2.1 to determine if issues were being appropriately prioritized in accordance with their safety significance. The inspectors also determined whether the cause or root cause analysis and the specified corrective actions were appropriate and in accordance with licensee procedures. Virginia Power Administrative Procedure (VPAP) 1604, "Root Cause Evaluations," Revision 2, provided instructions on performing Category 1 (the most rigorous) and Category 2 root cause evaluations (RCEs). Category 3 RCEs (the lowest level cause evaluation) were defined and discussed in VPAP-1601, "Corrective Actions." The corrective action, goals and monitoring for all Maintenance Rule (a)(1) items were reviewed to determine if they were acceptable.

The inspectors attended a Station Nuclear Safety and Operating Committee (SNSOC) meeting and reviewed selected SNSOC meeting minutes to determine if plant issues were being properly reviewed and if appropriate management attention was applied to plant issues.

The inspectors reviewed the seven potential problem reports (PPRs) listed in Attachment 1 to determine the adequacy of the resolutions and associated evaluations.

The inspectors verified that the corrective actions for five non-cited violations (NCVs) identified in NRC inspection reports were appropriately developed and prioritized and were either implemented or scheduled to be implemented.

b. Findings

In general, issues entered into the corrective action program were properly prioritized and evaluated. Although risk significance was not formally used in prioritizing issues, the methodology used, based on procedural guidance, experience and judgement, was effective in identifying and prioritizing issues such that the issues were evaluated and corrective actions were developed commensurate with their safety significance. Category 1 and 2 root cause evaluations were thorough and identified corrective actions were appropriate. Other evaluations were normally satisfactory and corrective actions were generally implemented in a timely manner. Some exceptions, of very low safety significance, were noted concerning the adequacy of evaluations and timeliness of implementing corrective actions. Some issues involved licensing basis applicability or questions, and these issues took substantial time (greater than 6 months) to research and evaluate.

The inspectors identified an OE response that did not fully address the item. Plant Issue S-2000-1968 addressed emergency diesel generator load swings due to governor oil viscosity. The response stated that diesel load swings had not been a problem at Surry but the evaluation incorrectly assumed that a 10W30 oil was utilized in the governor when an ISO Grade 68 oil is utilized. The licensee reopened the OE item until the item could be fully reviewed.

The inspectors also identified an example in which probabilistic risk analysis (PRA) insights could have been used more appropriately. During the development of the evaluation for PPR 00-017 (still in draft at the time of this inspection), PRA insights were referenced as confirming engineering's conclusions. Engineering had concluded that the likelihood of the design basis large break loss of coolant accident (LOCA) with the loss of offsite power and a failure of an entire DC bus was very small. As a result, engineering determined that the present technical specification allowed outage time for certain engineered safety feature actuation instrumentation was acceptable. The risk information used to support the acceptable conclusion involved the use of only that one sequence of events. Considering risk from all sequences, including all size LOCAs, which rely on the automatic function associated with the instrumentation would have been more appropriate. The licensee's PRA personnel agreed with this observation but indicated that the final conclusion would not change.

The inspectors noted several items that took a long time to evaluate. For example:

- OE item (Plant Issue S-2000-0571), concerning non-conservative engineered safety feature actuation instrumentation technical specification allowed outage times, was placed in the corrective action program on March 14, 2000, but was not escalated to a PPR (PPR 00-017) until July 13, 2000. The evaluation was not complete at the time of this inspection (approximately nine months). The evaluation was in draft and the responsible individual stated that the review should be completed shortly.

- PPR 99-047, concerning no thermal relief valves installed on component cooling water piping per the applicable piping code, was issued on November 29, 1999. The licensee has confirmed that either operating conditions or procedural controls would most likely preclude a significant piping failure. However, the evaluation to justify not installing or determining which pipes should have thermal relief valves installed is not scheduled to be completed before the later part of 2001.

- PPR 99-028, concerning not upgrading the classification of the main feedwater regulating valves and the associated bypass valves to safety-related as stated in PPR 93-024, was issued on June 28, 1999. PPR 93-024 was closed to an engineering department tracking item which, for a long time, did not rise to a priority level to be completed. The item was completed this year and PPR 99-028 was closed out. The inspectors verified that a similar situation did not exist with other engineering department tracking items. The new corrective action program, implemented within the last two years, should preclude similar situations from occurring.

NRC review of PPR 98-045 was incomplete at the end of the inspection period. This PPR involves whether or not check valves need to be installed in the inside recirculation spray pump discharge piping. The potential concern is if a loss of offsite power would occur after the pumps have started for a LOCA, the water in the discharge piping and spray headers would flow back through the pumps causing them to rotate backwards. When the emergency diesel generators start and repower the pump motors, the pump motors could stall or fail resulting in a failure of both trains of inside recirculation spray.

This item is identified as an unresolved item (URI) 50-280, 281/00009-01 pending review of the risk significance associated with this condition.

.3 Effectiveness of Corrective Actions

a. Inspection Scope

The inspectors reviewed findings concerning the effectiveness of corrective actions from the Nuclear Oversight corrective action audit, three station level self-assessments and two quarterly trend reports to determine if they were generally consistent with NRC findings from this inspection and those documented in the nine NRC inspection reports issued between December 6, 1999, and October 31, 2000. The inspectors also assessed the effectiveness of corrective actions by observing if issues repeated, e.g. issues reoccurred when they should have been prevented by previous corrective actions.

b. Findings

Corrective actions were typically effective in resolving issues as demonstrated by few repeat events. The results of audits and self-assessments concerning corrective action effectiveness were generally consistent with NRC observations and findings documented in NRC inspection reports.

A notable exception was the corrective actions developed and implemented to address the tripping of the auxiliary ventilation exhaust filter fans. The corrective actions, including modifications and procedure changes, failed to correct this condition. Corrective actions implemented in December 1999 and those implemented in April 2000 (fans tripped during testing) did not prevent the fans from tripping again during testing in October 2000. This issue is discussed in NRC Integrated Inspection Reports Nos. 50-280, 281/00-03 and 00-05 and Licensee Event Reports (LERs) 50-280, 281/00001-00 and 01, "Filtered Exhaust Fan Failure Results in Technical Specifications Violation," and LER 50-280, 281/00003-00, "Both Filtered Exhaust Fans Inoperable Due To Operation Close To Trip Setpoint."

.4 Assessment of Safety-Conscious Work Environment

a. Inspection Scope

Personnel, both licensee and contractor, were interviewed concerning their willingness to report nuclear safety issues, generate plant issue documents on equipment and procedure deficiencies, and knowledge of and willingness to use the employee concerns program. The inspectors reviewed "Employee Concerns Program Assessment," dated June 8, 2000.

b. Findings

The inspectors concluded that a safety conscious work environment was present where employees felt free to raise nuclear safety concerns. Plant personnel were willing to identity issues as evidenced by the number of plant issue reports submitted, the low threshold for identifying issues and the content of the issues. This conclusion was also confirmed through interviews with personnel from operations, maintenance, engineering, security and health physics.

The employee concerns program receives approximately 3 or less concerns per year. Non-operations individuals interviewed were typically not familiar with the program; however, none thought it was necessary because of the ease of submitting a plant issue report or raising concerns to their supervisor. The inspectors noted that information describing the program was not posted on bulletin boards in the main entrance or major hallways of the plant as was the practice for other plant programs.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors discussed the inspection results with Mr. Richard Blount, the Site Vice President, and other members of licensee management at the conclusion of the inspection on December 15, 2000. On January 12, 2001, Mr. Foster was informed via telephone call of URI 50-280, 281/00009-01. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any of the material 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
R. Blount, Site Vice President
D. Bucheit, Supervisor, Nuclear Engineering
M. Crist, Director Nuclear Oversight
B. Foster, Director, Nuclear Station Safety and Licensing
T. Sowers, Director Nuclear Operations and Maintenance
B. Stanley, Supervisor, Licensing

NRC

K. Landis, Chief, Branch 5, Division of Reactor Projects, Region II

ITEM OPENED

50-280, 281/00009-01	URI	Determine the risk significance of the inside recirculation spray pumps not having a check valve installed in the discharge piping (Section 4AO2.2)

DOCUMENTS REVIEWED

Audits

Nuclear Oversight Audit Report 00-03, "Emergency Planning"
Nuclear Oversight Audit Report 00-07, "Radiological Protection / Process Control Program"
Nuclear Oversight Audit Report 00-09, "Corrective Action"
Nuclear Oversight Audit Report 00-11, "Maintenance"

Licensee Event Reports

50-280, 281/00001-00 and 01	"Filtered Exhaust Fan Failure Results in Technical Specifications Violation"
50-280, 281/00003-00	"Both Filtered Exhaust Fans Inoperable Due To Operation Close To Trip Setpoint"

Non-cited Violations

50-280/99009-01	"Failure to provide adequate PMT instructions for the replacement of relay 1-RP-RLY-271XB"
50-281/99007-04	"Failure to properly complete post maintenance testing prior to declaring the A LHSI pump operable"
50-280/00003-01	"Failure to follow AMSAC logic test procedure"
50-280,281/00003-04	"Failure to comply with the requirements of the Physical Security Plan"
50-281/99007-03	"Failure to meet the requirements of TS 3.16.B"

Operating Experience Items (by plant issue number)

S-1999-2378	"Failure of Closing Spring Charging Motor of ABB K-3000 Air circuit Breaker"
S-1999-2560	"10 CFR Part 21 Report - ABB K-Line Breaker Defect"
S-1999-2581	"ITT Barton Nuclear Advisory Letter on Model 752 Differential Pressure and Model 753 Gage Pressure Transmitters"
S-1999-2733	"Emergency Diesel Generator Stub Shaft Bracket Bolting Failure"
S-2000-0091	"ITT Industries - Vendor Notification #36512"
S-2000-0098	"WOE 00053-PHWR Event-Limitorque SMB-2 Actuator Drive Sleeve Part Change"
S-2000-0099	"Refurbished Pumps Were Coated With Rust Inhibitors That Contained Sulfur"
S-2000-0525	"Leak Sealant Compound was Not Within Manufacturers Tolerance"
S-2000-0571	"PS36743: Summer - Plant Event #36743"
S-2000-0615	"Reactor Building Containment Hatch Performance Test Failures"
S-2000-1374	"ABB 5HK350-3000A Breaker, Misalignment of Control Device Contact Due to Lack of Updated Vendor Information"
S-2000-1967	"ASCO Hydramotor Relief Valve Staking"
S-2000-1870	"Discrepancy in Offsite Dose Calculation Software"
S-2000-2006	"ASCO Hydramotor Relief Valve Staking"
S-2000-1727	"SEN 213: Steam Generator Tube Failure"
S-2000-0230	"OE 10567: Reduced Reactor Coolant Pump Seal Leakoff Flows"

S-2000-2686 “OE 10397: Westinghouse Inverter Capacitor Failure”
S-2000-2827 “Part 21 99-08: ABB HK Circuit Breaker Trip Roller Hardness Below Specifications”
S-2000-2469 “OE 10311: Failure of ABB 5HK350-3000 Breakers due to an Inadequate Design Upgrade”
S-2000-1999 “OE 10463: Air Discovered in Safety Injection Pump Casings”
S-2000-2665 “OE 10392: Main Feedwater Regulating Valve Maintenance Deficiencies”
S-1999-2467 “Possible Defect in a Swagelok Tube Fitting”
S-2000-1093 “OE 10912: Division 3 EDG Output Voltage Flucuations due to Degraded Potentiometer”
S-2000-1254 “WEST 00-03: Pressurizer Upper Level Instrument Line Safety Classification”
S-2000-0057 “OE 10419: Electro-Hydraulic Governor Valve Rando Movement/Transients”
S-2000-1464 “OE 11102: Diesel Generator Trip from Low Jacket Coolant Pressure”
S-2000-0672 “OE 10790: Surge Arrestor Failure Causes Transformer Isolation”
S-2000-1968 “OE 11321: Emergency Diesel Generator Load Swings”

<u>Plant Issue Reports</u>

S-1998-1575 “Degrading Condenser Vacuum Due to Maintenance Activities”
S-1999-2711 “Thermal Relief Valve Protection Not Provided for all CC Equipment”
S-2000-2473 “Unit 1 Reactor Trip Due to Work on Wrong Unit EHC System”
S-2000-1656 “EDG Level Switch Modified By Vendor - It Will Not Fit”
S-2000-2621 “2-SS-TV-204A Failed to Close During Manual Operation Attempt”
S-2000-2614 “1-SW-PCV-100D Failed to Open During Performance Test”
S-2000-2575 “Sixty Three False PTs Were Downloaded from PTSS into WICATS”
S-2000-2455 “2-DG-TV-208A Failed Type C Test @11.5 SCFH”
S-2000-2451 “Program for Updating Reference Values and Acceptance Criteria for MOVs and TVs Appears to be Ineffectively Implemented”
S-2000-2363 “Bolts/Fasteners in Flange Connections for Piping to RWST Chillers are Corroded”
S-2000-2352 “As Found Relief Valve Set Points Out of Spec.”
S-2000-2343 “Valve 02-SI-56-CKVALVE Failed its Open and Inspect IST Seat Leak Test”
S-2000-2342 “Maintenance Signed Off Work Complete Which was not Complete”
S-2000-2286 “Valve 2-CC-TV-240B Would not Stroke”
S-2000-2268 “Inspection of Snubber 2-RC-HSS-144 Found no Fluid”
S-2000-2177 “2-RC-6 and 2-RC-45 Would not Close Manually”
S-2000-2102 “PI S-2000-0838 has an Overdue Maintenance Rule Evaluation”
S-2000-2089 “Containment Vacuum Pressure Transmitters 02-CV-PT-201A&B Were Found Out of Calibration”
S-2000-2080 “Unit 1 RCP Shaft Alert Annunciator has been Coming in and Immediately Clearing for about a Week”
S-2000-2005 “WO 432575-01 was Issued with PMT Required but no PMT was Included in the WO”
S-2000-2001 “Scheduled Maintenance Outage for the Unit 2 “C” Charging Pump Exceeded its Expected Out of Service Time”
S-2000-1986 “Freeze Seal Box for 2-CH-27 Placed in Wrong Location”
S-2000-1700 “Pump 1-EE-P-1A Rotor Clearances Out of Spec.”

S-2000-1648 "Maintenance Plugged Five Tubes in 01-CW-E-1D on 07/22/00"
S-2000-1576 "EDG Output Breaker 15J3 Control Switch Would not Red Flag When Breaker was Closed"
S-2000-1568 "Evaluation of External Leak on Check Valve 1-SI-61"
S-2000-1367 "Jacking Device for 1-MS-FCV-104A Would not Engage"
S-2000-1314 "Total Torque Values for Four MOVs Exceeded the Limits in Engineering Transmittal CME 99-0066, Rev. 0 and Rev. 1"
S-2000-1312 "PMs for 1-CC-RV-134 and 135 Exceeded Their Late Date"
S-2000-0988 "Surveillance of Transmitter 1-CH-FT-1113-IXMITR Determined it Was Out of Spec."
S-2000-1097 "1-SW-64 Piping Near the "C" RSHX has a Through Wall Leak"
S-2000-1203 "Three MOV Torque Settings Were Set Above Their Available Motor Torque Capability"
S-2000-1013 "B Hi CLS Signal Generated when a Lead was Lifted"
S-1999-2412 "C RSST Isolation due to Electrical Fault"
S-2000-0795 "1-SW-MOV-103B Motor Ground"
S-2000-0812 "Pin Hole leak on Rupture Disc 1-RC-TK-2"
S-2000-0806 "CRDM Fan has Increasing Vibration"
S-2000-0805 "CRDM Fan has Excessive Vibration"
S-2000-0804 "Potential Through Wall Leak on 1-RH-4"
S-2000-0786 "NI-32 Failure"
S-2000-0755 "MCR Leakage Testing Issues"
S-2000-0315 "2-CW-MOV-206A Leaks by 140 GPM"
S-2000-0705 "2-CW-MOV-206A Seat Leakage"
S-2000-0690 "Milk Analysis not performed as Required"
S-2000-0618 "3-EE-RLY-SFD2-Relay time delay was to long"
S-2000-0516 "1-CC-E-1A in Alert after Performing 1-OSP-SW-002"
S-2000-0579 "CC HX 1A Endbell Requires Twelve Weld Repairs"
S-2000-0510 "CC HX 1C tubes Coated with Shells and Sand"
S-2000-0457 "Safety Evaluation 92-93A no Longer Valid for Current Plant Analysis"
S-2000-0449 "1-CC-E-1C Inoperable"
S-2000-0401 "Lack of Documentation Supporting the Configuration of the Charging Pump Exhaust Hoods"
S-2000-0409 "1-SW-840 Internals Degraded"
S-2000-0234 "Terry Turbine Governor Exhibits Excessive Divergent Oscillations"
S-2000-0741 "1-GW-TK-1B Pressure Increased with the Tank Isolated"
S-2000-0540 "TSC Filter Testing does not Support the Iodine Removal Efficiency Assumed in the Dose Analysis"
S-2000-0581 "3 Empty Sample Canisters not Blanked as required"

<u>Potential Problem Reports</u>

93-024 "Main Feedwater Regulating and Bypass Valve Classification"
98-045 "Potential for IRS Pumps to Fail to Restart With Subsequent LOOP"
98-052 "AFW Flow Indication Improperly Classified as NSQ"
99-028 "Main Feedwater Regulating and Bypass Valve Classification"
99-047 "CC Thermal RVs Not Installed"

00-006 “MCR Chiller / TS 3.23"
00-017 “Failure of DC Power Causes Loss of Protective Function”

Procedures

VPAP-1501, “Deviations,” Revision 12
VPAP-1601, “Corrective Actions,” Revision 13
VPAP-1604, “Root Cause Evaluation Program,” Revision 2
VPAP-1801, “Program and Management Oversight of Quality,” Revision 6

Root Cause Evaluations - Category 1

S-2000-2473 “Unit 1 Reactor Trip Due to Personnel Error”
S-2000-1666, “#2 EDG Failure to Start during Return to Service Testing”

Root Cause Evaluations - Category 2

S-2000-1656 “EDG Fuel Oil Level Switch Issues”
S-2000-1255 “Inadvertent Weapon Discharge at Firing Range”
S-1999-2412, “C RSST Electrical Fault”
S-1999-2205 “Unit 2 Condensate Polisher Resin Performance Problems”

Self-assessments

Station Level Self-assessment “Corrective Action Program” approved March 31, 2000
Station Level Self-assessment “Root Cause Effectiveness” approved August 26, 2000
Station Level Self-assessment “Corrective Action Effectiveness” approved October 28, 2000
“Employee Concerns Program Assessment,” approved June 8, 2000

Surry Power Station - NRC Inspection Reports

Nos. 50-280/99-07, 50-281/99-07
Nos. 50-280/99-08, 50-281/99-08
Nos. 50-280/99-09, 50-281/99-09
Nos. 50-280/00-01, 50-281/00-01
Nos. 50-280/00-02, 50-281/00-02
Nos. 50-280/00-03, 50-281/00-03
Nos. 50-280/00-04, 50-281/00-04
Nos. 50-280/00-07, 50-281/00-07 (Safety System Design Inspection)
Nos. 50-280/00-08, 50-281/00-08 and Office of Investigations Report No. 2-2000-013

Trend Reports

Station Nuclear Safety: “Second Quarter 2000 DR Trend Report”
Station Nuclear Safety: “Third Quarter 2000 DR Trend Report”

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.