UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report Nos.: 50-280/95-07 and 50-281/95-07

Licensee: Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060

Docket Nos.: 50-280 and 50-281

License Nos.: DPR-32 and DPR-37

Facility Name: Surry 1 and 2

Inspection Conducted: April 2 through 29, 1995

Lead Inspector: M. W. Branch, Senior Resident Inspector — Date Signed: 5-24-95

Inspectors: D. M. Kern, Resident Inspector
G. A. Harris, Resident Inspector

Approved by: G. A. Belisle, Section Chief — Date Signed: 5-24-95
Reactor Projects Section 2A
Division of Reactor Projects

SUMMARY

Scope:

This routine resident inspection was conducted on site in the areas of plant status, operational safety verification, maintenance inspections, engineering inspections, plant support, and Licensee Event Report follow-up. Inspections of backshift activities were conducted on April 19, 21, and 26.

Results:

Operations

The Unit 2 Safety Injection system configuration was properly controlled and material condition was good (paragraph 3.2).

Overtime controls, including monitoring for indicators of fatigue, were effective during the Unit 2 refueling outage (paragraph 3.3).

Conservative decision making was noted when Unit 1 was manually tripped to ensure plant conditions were bound when problems were encountered after a control rod dropped into the core (paragraph 3.1).

Maintenance

Material condition of the Spent Fuel Pool Cooling System did not meet the plant's normal high standards (paragraph 3.4).

Engineering

A weakness was noted in procedure 1-OPT-RX-004, Reactor Power Calorimetric Using Feed Flow With P-250 Out of Service (Manual), (paragraph 5.1).

Plant Support

Confined space entry (CSE) training was comprehensive and appropriate to support personnel safety at the station. Personnel demonstrated sound knowledge of CSE requirements and atmosphere test equipment (paragraph 6.2).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*W. Benthall, Supervisor, Licensing
*H. Blake, Jr., Superintendent of Nuclear Site Services
*R. Blount, Superintendent of Maintenance
*D. Christian, Station Manager
J. Costello, Station Coordinator, Emergency Preparedness
D. Erickson, Superintendent of Radiation Protection
*B. Garber, Licensing
B. Hayes, Supervisor, Quality Assurance
D. Hayes, Supervisor of Administrative Services
*D. Llewellyn, Superintendent, Nuclear Training
C. Luffman, Superintendent, Security
*R. MacManus, Supervisor, System Engineering
*J. McCarthy, Assistant Station Manager
*A. Price, Assistant Station Manager
*S. Sarver, Superintendent of Operations
R. Saunders, Vice President, Nuclear Operations
K. Sloane, Superintendent of Outage and Planning
*E. Smith, Site Quality Assurance Manager
T. Sowers, Superintendent of Engineering
*B. Stanley, Supervisor, Station Procedures
J. Swientoniewski, Supervisor, Station Nuclear Safety

Other licensee employees contacted included plant managers and supervisors, operators, engineers, technicians, mechanics, security force members, and office personnel.

NRC Personnel

*M. Branch, Senior Resident Inspector
D. Kern, Resident Inspector
G. Harris, Resident Inspector

*Attended Exit Interview

Acronyms used throughout this report are listed in the last paragraph.

2. Plant Status

Unit 1 began the inspection period at 100% power. On April 12, control rod J-7 dropped to the fully inserted position, causing a turbine runback. Operators stabilized the unit at 74% power. A CRDM cable failure precluded operators from recovering control rod J-7 to the desired group position. The fully inserted control rod resulted in a higher axial flux difference than was permitted for power operations above 50%. Operators reduced reactor power to 45% and manually tripped the reactor (paragraph 3.1).

Unit 1 was restarted and placed on-line on April 16 following CRDM System repairs. Operators observed an unexpected steam flow imbalance between SGs as the reactor approached full power. The B SG was producing about 200,000 pounds mass/hour less steam flow than either the A or C SGs. Initial licensee evaluation concluded that the reactor trip had removed an oxide layer from the B SG U-Tube region which resulted in a temporary reduction of heat transfer capability. Operators stabilized reactor power at 98.5% to reduce the magnitude of the steam flow imbalance. Unit 1 remained at 98.5 % power through the end of the inspection period.

Unit 2 operated at 100% power throughout the entire inspection period.

3. Operational Safety Verification (71707)

The inspectors conducted frequent tours of the control room to verify proper staffing, operator attentiveness and adherence to approved procedures. The inspectors attended plant status meetings and reviewed operator logs on a daily basis to verify operational safety and compliance with TSs and to maintain overall facility operational awareness. Instrumentation and ECCS lineups were periodically reviewed from control room indication to assess operability. Frequent plant tours were conducted to observe equipment status, fire protection programs, radiological work practices, plant security programs and housekeeping. Deviation reports were reviewed to assure that potential safety concerns were properly addressed and reported.

3.1 50.72 Report

At 5:56 pm, on April 12, Unit 1 was manually tripped from 45 percent reactor power. All control rods inserted into the core as designed. Auxiliary feedwater automatically initiated as designed on low-low steam generator level following the trip. No primary safety or power operated relief valves were actuated during the event. No secondary safety relief valves actuated; however, the A main steam power operated relief valve was actuated briefly at the start of the transient. It subsequently closed and reseated. All electrical buses transferred properly following the trip and all emergency diesel generators were operable.

The manual reactor trip was well coordinated and demonstrated conservative decision making. There were no Technical Specification or procedural requirements to shutdown the unit. Plant maintenance and engineering personnel had determined that a high probability existed for dropping additional control rods if an orderly shutdown was attempted. Troubleshooting was being performed as a result of a dropped control rod, which occurred earlier that day at approximately 8:20 am.

At 9:20 pm the licensee informed the NRC via the Emergency Notification System of the event. The inspectors determined that the event was properly reported as required.

3.2 SI System Lineup Verification

The inspectors reviewed the material conditions and positions of approximately 70 SI system valves within the Unit 2 Safeguards Building to verify that the SI system was correctly configured to perform the LHSI safety function. The general material condition of the system was good. Minor discrepancies including packing leaks, a missing label, and a degraded position indicator were discussed with the Supervisor of Shift Operations and appropriate corrective actions were initiated. All SI system valves inspected were found properly positioned in accordance with procedure 2-OP-SI-001A, SI System Alignment, revision 2. The current position of SI system valves was properly documented in the system lineup library maintained by the Operations Department. The inspectors concluded that the Unit 2 SI system configuration was properly controlled and maintained.

3.3 Personnel Overtime Controls During Outage

VPAP-0103, Working Hours and Limitations, revision 2, establishes policy to control overtime work. The inspectors reviewed VPAP-0103, timekeeping records, interviewed personnel, and observed outage work activities to determine whether overtime controls were effectively implemented.

VPAP-0103 clearly defined administrative overtime limits for personnel who perform safety-related functions. These limits and approval authority were consistent with NRC GL 82-12, Nuclear Power Plant Staff Working Hours. The inspectors reviewed overtime records for the February-March, 1995 period. With limited exceptions, workers requested and received Station Manager's approval prior to exceeding station administrative overtime limits. The Administrative Services department reported authorized and unauthorized overtime usage (above the administrative limits) to the station manager on a weekly basis. When appropriate, station DRs were initiated to resolve discrepancies regarding pre-authorization to exceed administrative overtime limits. The inspectors determined that overtime controls were implemented consistent with VPAP-0103.

The inspectors observed that, on several occasions, a single request to exceed overtime limits for selected activities was written for an entire department (i.e., electrical maintenance - 60 people, mechanical maintenance - 90 people). However, only approximately 10 to 15% of those persons authorized under the blanket overtime authorization actually exceeded the guidance. The inspectors questioned whether the program continued to preclude excessive worker fatigue when blanket overtime authorization of this scale was granted. The inspectors discussed this concern with the Station Manager who noted that several actions to prevent worker fatigue and monitor for fatigue were effective during the outage. These included rescheduling of work

to maintain crew continuity, frequent in-plant observation tours by management, human factors assessment of DRs and job rework, and QA oversight. In addition, workers received behavioral observation training annually. The inspectors independently interviewed workers during the outage and attended daily work progress meetings. Workers did not appear to be adversely affected, physically or mentally, due to working overtime. The quality of work performed was not negatively impacted by use of overtime. The inspectors concluded that licensee overtime controls, including continued monitoring for indicators of fatigue, satisfied the intent of NRC GL 82-12.

3.4 Spent Fuel Pool Cooling and Purification System Walkdown

On April 11 and 12, the inspectors performed a valve alignment verification and an evaluation of the material condition of the spent fuel cooling pumps, skimmer, and purification system components. The inspectors noted that the material condition of these components, while generally adequate, was not in keeping with that in the rest of the station. Discrepancies noted included the following:

- Several pressure gage isolation valves were found to be out of their specified full open position. For example, the spent fuel cooling pump A suction pressure gage isolation valve, 1-FC-6 was found nearly closed causing the gage to read abnormally low. In another example, spent fuel cooling pump B discharge pressure gage isolation valve, 1-FC-10, was found throttled. These isolation valves were for gages that provided local indication only.

- Both inboard and outboard mechanical seals for the 1A and 1B spent fuel cooling pumps appeared to have been leaking as evidenced by the accumulation of a considerable amount of boron.

- The oil bubbler for the 1B spent fuel cooling pump was observed to be abnormally low for a period of two days while the pump was in service. Oil was added to the pump on the third day.

- The coatings for both the 1A and 1B spent fuel cooling water pumps and their base plates were significantly degraded.

- Spent fuel pool cooling heat exchanger component cooling water pressure gage, 1-CC-PI-101B, read offscale high.

- The local power control hand switches for the 1A spent fuel cooling water pump was not labelled in accordance with station standards. This was the only example of a labeling problem identified.

- Both the 3A and 3B purification pumps had extensive rust on the base plates and their coatings were degraded.

- Both the 2A and 2B skimmer pumps had oil and water accumulation on their base plates resulting in an undesirable radioactive mixed waste.

The inspectors concluded that despite the noted discrepancies the spent fuel cooling water pump, skimmer and purification pumps were operable. These discrepancies were discussed with station management for investigation and correction as necessary.

Within the areas inspected, no violations or deviations were identified.

4. Maintenance Inspections (62703)

During the reporting period, the inspectors reviewed the following maintenance activities to assure compliance with the appropriate procedures.

4.1 Battery Status Update

In October 1994, the 2A station battery became inoperable due to a degraded cell as documented in NRC Inspection Report Nos. 50-280/94-32 and 50-281/94-32. NRC and licensee follow-up identified several programmatic weaknesses which resulted in failure to recognize battery degradation in a timely manner. Station management subsequently issued a management expectation policy statement which directed that a station DR be written each time a battery cell is found in the Alert performance range. This action was taken to ensure that indications of battery degradation were raised to management's attention before a battery became inoperable. The licensee also initiated a RCE to assess battery performance throughout the station. The inspectors reviewed battery maintenance activities to determine whether the policy statement had been implemented in an effective manner.

The number of DRs pertaining to the 1A and 2A station batteries dramatically increased since October 1994. In addition, the number of DRs reported pertaining to the EDG, TSC, Black, and security inverter batteries also increased as the policy statement was implemented. Prior to October, very few battery problems were documented. The inspectors observed that improved documentation of marginal battery performance has been beneficial in highlighting continuing problems with the station batteries for management, driving corrective actions to be taken, and in increased maintenance support.

In December 1994, six 1A station battery cells were replaced. The replacement cells were spares obtained from another facility. The replacement cells had been in service for approximately two years at the other facility, and then stored for approximately three

years. Several of the replacement cells exhibited marginal performance following installation, warranting increased attention. Two cells were replaced a second time prior to reactor startup in December 1994. In March 1995, eighteen 2A station battery cells were replaced. This included replacement of the cell which had caused the battery to be inoperable in October 1994. The jumper which had temporarily bypassed the inoperable cell was removed and the 2A station battery was restored to its design condition. Several additional maintenance activities have been completed in the past six months to improve the condition of the EDG, TSC, Black, and security inverter batteries.

Marginal cell voltage and non-uniform specific gravities were observed following cell replacements on 1A and 2A station batteries. These conditions persisted despite repeated battery equalization charges. System engineers determined that low equalization charge voltage was a major contributor to these problems. It was not possible to increase the equalization charge voltage further for these batteries due to DC bus voltage limitations. This limited the amount of battery improvement that can be gained from a charge. The RCE team discussed this situation with the vendor and continued to evaluate alternate solutions to improve the condition of the station batteries.

The licensee RCE team determined that following the 1A and 2A cell replacements, station battery performance was acceptable. However, there were RCE recommendations that addressed 1A cell replacement during the Fall 1995 Unit 1 RFO and possible station battery replacement at a later date. The RCE team also examined the appropriateness of post maintenance test selection for cell replacements following problems encountered on the 1A replacement.

The inspectors reviewed the licensee RCE for station battery problems and determined that it was thorough. The increase in DRs associated with battery problems indicated increased sensitivity to early problem identification and resolution. The licensee's lower threshold for problem identification as outlined in the October 1994 management policy statement has been effective in early identification of equipment degradation. Additionally, as outlined in the licensee RCE, even though the batteries remain operable, further maintenance actions and possible battery replacement are necessary to restore the material condition of some of the more degraded batteries.

Within the areas inspected, no violations or deviations were identified.

5. Engineering Inspections (37551)

During the reporting period, the inspectors reviewed engineering activities to assure compliance with the appropriate procedure and TS requirements.

5.1 Review of Reactor Power Calorimetric Manual Method

During a walkdown of the Unit 1 mechanical equipment room the inspectors observed that one of the local feedwater temperature indicators was missing. This instrument had been removed to allow installation of the downstream RTD into the piping thermowell. The thermowell for the RTD had developed a leak and had been plugged. The feedwater RTD provides input to the P-250 computer for temperature information used in the computerized calorimetric calibration of NIs.

The P-250 computer used for calorimetric calibration of the NIs was out of service and the licensee was planning to perform a manual calorimetric if the computer could not be returned to service in time. The computer was returned to service and the computerized calorimetric was completed on time.

The inspectors reviewed procedure 1-OPT-RX-004, Reactor Power Calorimetric Using Feed Flow With P-250 Out of Service (Manual), revision 1. This was the procedure that the licensee would have used to perform the calorimetric if the computer was not available. The inspectors noted that the procedure required the use of the local temperature instruments to measure feedwater temperature. As noted previously, one of the referenced temperature instruments was removed and would not have been available if needed. The three local feedwater temperature instruments were designated 1-FW-TI-154-A, B, and C. These local temperature instruments were made by Ashcroft with a span of 200 - 700 °F with 5 degree graduations on the gage face. Given the needed precision of the calorimetric, the inspectors reviewed the assumed accuracy in the licensee's calculational basis for NI setting and adjustment.

Calorimetric assumptions and calculations were contained in "Phase 2 Results of Nuclear Unit Efficiency Study for Surry Unit 1 (VEPCO Contract No. FHN-074-0219) April 25, 1984." This study defined the assumed feedwater temperature error as the sum of instrument specified tolerances + instrument drift + readout error. The total feedwater temperature error assumed was 1.75 °F. The inspectors reviewed licensee's vendor information on the Ashcroft local temperature instruments to determine specified accuracy. Additionally, I&C calibration data was reviewed to determine calibration accuracy.

The vendor manual specifies and the licensee's lab calibrates these type instruments to an accuracy of +/- 1% of full scale (1% of 500 degrees or 5 °F). The licensee's field calibration specifies a +/- 2% full scale accuracy (2% of 500 degrees or 10 °F). Neither of these values were within the values assumed in the calorimetric calculation. It should be noted that, the calorimetric formula applies instrument uncertainty in a statistical manner. Therefore, the increased feedwater

temperature error impact on the assumed calorimetric uncertainty is unclear but believed to be slight.

The inspectors also reviewed previous calibration records for one of the three local feedwater temperature instruments. The inspectors selected 1-FW-TI-154A for review. The licensee's records indicated that this instrument was last calibrated on May 4, 1990, by WO 094487. Additionally, the inspectors noted that the local feedwater temperature instruments were not in the periodic calibration program. The inspectors held discussions with reactor engineering personnel and questioned the acceptability of using non-calibrated gages with accuracies outside those assumed in the calculation for performing calorimetric calibrations. The licensee is currently performing a review to determine the potential impact on calorimetric uncertainties and management has placed a hold on the use of procedures 1 & 2-OPT-RX-004. The inspectors did not identify cases where this procedure had been recently used to provide values for adjustment of NIs. The inspectors considered this issue to be a weakness in the licensee's program and will continue to follow the licensee's actions.

Within the areas inspected, no violations or deviations were identified.

6. Plant Support (71707, 71750)

The inspectors conducted facility tours, work activity observations, personnel interviews, and documentation reviews to determine whether programs were effectively implemented to comply with regulatory requirements in the areas of radiological protection and security.

6.1 Plant Tour Observations

The inspectors observed radiological control practices and radiological conditions throughout the plant. Portal and hand-held monitors were observed to be in good condition and within proper calibration periodicities. Radiological posting and control of contaminated areas were good. The licensee continued decontamination efforts to manage the amount of contaminated floor area to a minimum. The inspectors observed good worker practices regarding radwaste minimization and tool issue from the contaminated tool locker.

Selected aspects of plant physical security were reviewed during regular and backshift hours to verify controls were in accordance with the security plan and implementing procedures. This review included security measures, vital and protected area barrier integrity, maintenance of isolation zones, personnel access control, searches of personnel, packages and vehicles, and escorting of visitors. No discrepancies were noted.

6.2 Confined Space Entry Training

Certain portions of the station have been identified as confined spaces, for which personnel require specialized training prior to access. The training and requirements of VPAP-1901, Industrial Safety and Health, revision 4, were established for personnel safety. The primary hazards of concern are flammable gas, toxic fumes, and an oxygen deficient or enriched atmosphere. The inspectors attended confined space entry training prior to inspecting LHSI components in the valve pit level of the Unit 1 and Unit 2 Safeguards Buildings. The instructor was knowledgeable and the training material was comprehensive. RP personnel demonstrated sound knowledge of confined space entry requirements and atmosphere test equipment usage when accompanying the inspectors to the valve pits. The inspectors concluded that the level of training was appropriate to support personnel safety during confined space entries.

Within the areas inspected, no violations or deviations were identified.

7. Licensee Event Report Follow-up (92700)

The inspectors reviewed LERs to verify accuracy, description of cause, previous similar occurrences, and effectiveness of corrective actions. The inspectors considered the need for further information, possible generic implications, and whether the events warranted further on-site follow-up. The LERs were also reviewed with respect to the requirements of 10 CFR 50.73 and the guidance provided in NUREG 1022, Licensee Event Report System, and its associated supplements.

(Closed) LER 50-281/93-03, Unit 2 Automatic Reactor Trip Due to Low Steam Generator Water Level Coincident with Steam/Feedwater Flow Mismatch Resulting from Spurious Closure of "A" MFRV. The LER describes the August 3, 1993 Unit 2 reactor trip. Safety systems responded as designed and the event did not pose a threat to public health and safety. The trip resulted from a sudden and unanticipated loss of FW flow to the A SG. The licensee determined that a 15 vdc MFRV controller power supply failed, causing the A MFRV to fail closed. The event, causal analysis, and corrective actions were previously documented in NRC Inspection Report Nos. 50-280/93-20 and 50-281/93-20. The inspectors reviewed the resultant RCE and confirmed that corrective recommendations had been implemented. The LER was accurate and met the requirements of 10 CFR 50.73.

Within the areas inspected, no violations or deviations were identified.

8. Exit Interview

The inspection scope and findings were summarized on May 5, 1995, with those persons indicated in paragraph 1. The inspectors described the

areas inspected and discussed in detail the inspection results addressed in the Summary section and those listed below.

Item Number	Status	Description/(Paragraph No.)
LER 50-281/93-03	Closed	Unit 2 Automatic Reactor Trip Due to Low Steam Generator Water Level Coincident with Steam/Feedwater Flow Mismatch Resulting from Spurious Closure of "A" MFRV (paragraph 7).

Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

9. Index of Acronyms and Initialisms

CFR	CODE OF FEDERAL REGULATIONS
CRDM	CONTROL ROD DRIVE MECHANISM
CSE	CONFINED SPACE ENTRY
DR	DEVIATION REPORT
ECCS	EMERGENCY CORE COOLING SYSTEM
EDG	EMERGENCY DIESEL GENERATOR
FW	FEEDWATER
GL	GENERIC LETTER
LER	LICENSEE EVENT REPORT
LHSI	LOW HEAD SAFETY INJECTION
MFRV	MAIN FEEDWATER REGULATING VALVE
NI	NUCLEAR INSTRUMENT
NRC	NUCLEAR REGULATORY COMMISSION
QA	QUALITY ASSURANCE
RCE	ROOT CAUSE EVALUATION
RP	RADIOLOGICAL PROTECTION
RTD	RESISTANT TEMPERATURE DETECTOR
SG	STEAM GENERATOR
SI	SAFETY INJECTION
TS	TECHNICAL SPECIFICATION
TSC	TECHNICAL SUPPORT CENTER
VDC	VOLT DIRECT CURRENT
VPAP	VIRGINIA POWER ADMINISTRATIVE PROCEDURE
WO	WORK ORDER
°F	DEGREES FAHRENHEIT