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

Report Nos.: 50-280/95-19 and 50-281/95-19

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: October 15 through November 4, 1995

Lead Inspector: L W Garner For 12-1-95
M. W. Branch, Senior Resident Inspector Date Signed

Other Inspectors:

D. M. Kern, Resident Inspector
W. K. Poertner, Resident Inspector
L. W. Garner, Project Engineer

Approved by: G A Belisle 12/1/95
G. A. Belisle, Chief Date Signed
Reactor Projects Branch 5
Division of Reactor Projects

SUMMARY

Scope:

This routine resident inspection was conducted on site in the areas of plant status, operational safety verification, maintenance and surveillance inspections, refueling activities, engineering review, plant support, self-assessment, and Licensee Event Report followup. Inspections of backshift and weekend activities were conducted.

Results:

Plant Operations

Command and control during Unit 1 startup for physics testing was good (paragraph 5.2).

A non-cited violation was identified for failure to properly implement a tagout procedure. Manipulation of the wrong component resulted in offsite power being lost to the 1H and 2J emergency busses (paragraph 9.1).

Maintenance

A personnel error during Reactor Protection System (RPS) testing resulted in both Unit 1 source range nuclear instruments becoming inoperable. The personnel error was classified as a non-cited violation (paragraph 4.2).

A strength in procedure usage was identified when Instrument and Control technicians identified an omission in a RPS logic test procedure (paragraph 4.2).

Root Cause Evaluation 95-11 developed several sound recommendations which are intended to enhance control room annunciator reliability (paragraph 8.2).

Engineering

Use of an additional reactivity chart recorder greatly improved control room communication between the Reactor Engineer and the reactor operators manipulating control rods during physics testing (paragraph 5.2).

Units 1 and 2 were uprated to 2546 MWT in October and August 1995, respectively. Engineers demonstrated a thorough understanding of the P-250 secondary calorimetric program and changes related to the core uprate. The decision to limit reactor power based on the feedwater flowrate calorimetric during core uprate activities demonstrated a sound safety perspective (paragraph 6.3).

Plant Support

Central alarm station operators were knowledgeable and performed physical security plan duties in an alert manner (paragraph 7).

REPORT DETAILS

1. Persons Contacted

<u>Licensee Employees</u>

*W. Benthall, Supervisor, Procedures
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
R. Garner, Outage and Planning
*D. Hayes, Supervisor of Administrative Services
*C. Lovett, Supervisor, Licensing
*C. Luffman, Superintendent, Security
*J. McCarthy, Assistant Station Manager
*F. McConell, Superintendent of Materials
*S. Sarver, Superintendent of Operations
R. Saunders, Vice President, Nuclear Operations
*B. Shriver, Assistant Station Manager
K. Sloane, Superintendent of Outage and Planning
E. Smith, Site Quality Assurance Manager
T. Sowers, Superintendent of Engineering
*B. Stanley, Station Director of Nuclear Oversight
*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.

<u>NRC Personnel</u>

#*M. Branch, Senior Resident Inspector
D. Kern, Resident Inspector
*K. Poertner, Resident Inspector
L. Garner, Project Engineer

*Attended Exit Interview on November 8, 1995.
#Attended Exit Interview on November 29, 1995.

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

2. Plant Status

On October 19, criticality was achieved for Unit 1 low power physics testing following the RFO. The unit was placed on the electrical grid on October 21 and was operating at 100% power at the end of the inspection period. Unit 1 core uprate was implemented during the RFO and no unexpected operational problems resulted during the startup. The unit is currently licensed to operate at 2546 MWT which represented an increase of approximately 30 megawatts electrical.

Unit 2 operated at or near 100% power throughout the 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 indications 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.

Spent Fuel Pool Cooling

During the inspection period, the inspectors reviewed the design of the Spent Fuel Pool Cooling System with respect to core offload capability. The spent fuel pool cooling system consists of two pumps and two heat exchangers. The heat exchangers and pumps are arranged for cross-connected operation if required. One pump and heat exchanger provide 100% heat removal capability. The original design requirements of the system stated that the system has to maintain fuel pool temperature below 140° F, when one-third of a core is placed in the fuel pool 150 hours after shutdown, and below 170° F when one and two thirds cores are placed in the fuel pool 150 hours after shutdown. These temperatures are based on a maximum component cooling water temperature of 105° F. With two heat exchangers in service, spent fuel pool temperatures can be maintained less than 140° F assuming worst case conditions, i. e., maximum component cooling water temperature and full core offload.

The inspectors determined that the original spent fuel pool design addressed a full core offload. The inspectors reviewed operating logs and determined that spent fuel pool temperature was maintained less than 106° F during the Unit 1 refueling outage with the entire core offloaded to the spent fuel pool. The inspectors determined that the UFSAR and spent fuel pool design calculations assumed that the core offload was not placed in the spent fuel pool until 150 hours after shutdown. The inspectors could find no administrative controls in place that would ensure that the 150 hour criteria was met prior to core offload. The licensee is reviewing this item and has initiated a commitment tracking item to follow its resolution. The recent Unit 1 core offload was not started until approximately 240 hours after shutdown. The inspectors will review this item further when the licensee has completed their evaluation.

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

4. Maintenance and Surveillance Inspections (62703, 61726)

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

4.1 Charging Pump Component Cooling Head Tank Sight Glass Replacement

On November 2, the inspectors witnessed work activities associated with replacing the Unit 1 charging pump component cooling head tank sight glass. The work activity was accomplished in accordance with WO 00287868. The scope of the work order had been expanded to include replacing the inlet and outlet sightglass isolation valves due to leakage past the seat. This activity was accomplished in accordance with skill of the craft criteria. Since replacement of the isolation valves required that automatic level control for the head tank be defeated, an operator was stationed at the head tank to unisolate makeup if required. The inspectors reviewed the work package and monitored maintenance activities in progress. Activities observed were conducted appropriately and the sightglass was returned to service without incident.

4.2 RPS Logic Testing

On October 18, a personnel error during RPS logic testing per 1-PT-8.2, Reactor Protection Logic, revision 6P-1, resulted in both Unit 1 source range nuclear instruments becoming inoperable for approximately one minute. While the source range instrumentation was inoperable, the RCS boron concentration was 2402 PPM, all control rod assemblies remained fully inserted in the core, and no positive reactivity changes occurred.

A shutdown margin calculation was performed. The results of the calculation determined the shutdown margin to be 8.9% (K-effective equal to 0.911). This item is the subject of LER 280/95-011. Failure to correctly perform a step in 1-PT-8.2 constitues a violation, i. e., failure to follow procedures. The inspectors reviewed the violation, the immediate corrective actions, and the issues surrounding its root cause. This licensee identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII of the NRC Enforcement Policy. This NCV is identified as NCV 50-280/95-19-01, Failure To Follow RPS Logic Testing Procedure.

After the inadvertent source range de-energization, RPS testing was suspended until the event was investigated and the cause determined. Subsequent to the test resumption, the I&C crew that was to complete the test was counseled on management's expectations concerning procedure utilization and was briefed on the previous crew's error. The inspectors observed personnel at the protection racks satisfactorily complete 1-PT-8.2. The inspectors verified that test switches were correctly aligned, test pushbuttons were properly actuated, and test lamp indications or relay actuations were received as specified by the procedure.

During the test, a technician observed, as he was closing a protection cabinet door, that test selector switch #1 had not been returned to its normal position. After verifying that no additional testing was to be performed in this protection cabinet and that the remaining test steps did not require this test switch be left selected to position 5, the technician returned the switch to its normal position. This was discussed with and concurred in by the other technician performing the test. A procedure review revealed that step 5.75 had failed to require the switch be returned to its normal position. A similar problem was encountered when step 5.80 was performed. The technician submitted a procedure change request to correct this deficiency. This procedure, as well as, the corresponding Unit 2 procedure were subsequently revised.

The inspectors reviewed RPS drawing number 113E244 sheet 19 and verified that leaving the test switch in a selected position would have no adverse effect on the RPS. Identification of this procedural deficiency by I&C personnel was considered as a strength, in that, it demonstrated proper procedure usage with attentiveness to expected equipment conditions.

Within the areas inspected, one NCV was identified.

5. Refueling Activities (71711)

5.1 Hot Rod Drop Testing

On October 18, the inspectors witnessed Shutdown Bank A and B hot rod drops being performed in accordance with 1-NPT-RX-014, Hot Rod Drops By Bank, revision 1. Initial conditions were properly established and precautions were observed. The inspectors independently verified that control rod designations marked on the chart recorder paper corresponded to the actual control rods being tested and that disconnected control rod position indicating signal cables were reconnected to their proper locations. The test data demonstrated that the control rod drop times were within TS criteria and that the control rods were latched to their drive mechanisms. On October 19, the test data for Control Banks A, B, C and D were reviewed. The control rods performed as expected and the test procedure and TS criteria were met.

5.2 Low Power Physics Testing

The inspectors witnessed portions of the low power physics testing conducted on Unit 1 for fuel cycle 14. The testing was conducted in accordance with procedure NPT-RX-008, Startup Physics Testing, revision 6. The inspectors attended the evolution briefing conducted prior to pulling the control rods to criticality, observed the reactor startup to criticality, monitored low power physics testing activities in progress and reviewed the results obtained. Command and control during the unit startup was good. All activities observed were satisfactory and the test results were well within the predicted values. The inspectors noted that an additional reactivity chart recorder was installed for use by the RO. This installation provided for better communication between the RO and reactor engineer. In the past, the RO was provided verbal direction as to rod motion direction and duration from the test director. The additional chart recorder allowed the RO to operate within the pre-established operational band controlled by the test director.

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

6. Engineering Review (37551)

6.1 Reactor Power Secondary Calorimetric Following Core Uprate

Actual reactor power is calculated once per eight-hour shift using a P-250 process computer based secondary calorimetric program (CALCALC). Operators compare all four PRNI channel indications with the CALCALC generated power level to verify PRNI accuracy. The PRNIs are then adjusted, as necessary, to maintain their input to RPS overpower protection circuitry within prescribed tolerances.

The CALCALC program can be run based upon either SFR or FFR. Units 1 & 2 were uprated to 2546 MWT in October and August 1995, respectively. Prior to core uprate, the licensee used CALCALC based on SFR to verify that reactor power was maintained within TS limits. The FFR based CALCALC was considered less accurate, but conservative, due to gradual feedwater nozzle fouling. Engineers anticipated that steam moisture content would increase following the core uprate, which would in turn affect SFR based CALCALC accuracy. Feedwater based CALCALC accuracy would not be affected. FFR based CALCALC power was higher than SFR based CALCALC computed power prior to core uprate on both units. The inspectors questioned how the licensee intended to verify that reactor power did not exceed the license limit during and after the core uprate.

ET No. CEE-95-065, Steam Moisture Content Increase, Core Uprate Project Effect on Steam Flow Derived Power by P-250 Computer SPS, Unit 2, revision 1, documented the anticipated core uprate effects on CALCALC computed power. Steam moisture content was postulated to increase from .25% to 1.0%. Increased moisture content would directly affect two CALCALC parameters; steam enthalpy and steam density. Actual steam enthalpy decreases below that used in the CALCALC program, making CALCALC computed power conservative relative to actual power. Steam density increases which causes indicated SFR to be less than the actual SFR. This second factor influences CALCALC in the nonconservative direction. Engineers determined that the net result would cause the SFR based CALCALC computed power to be slightly more conservative than before the core uprate. The inspectors reviewed ET No. CEE-95-065, verified the current P-250 CALCALC computer program used the enthalpy conservatism assumed by engineers, and independently calculated the affects of increased moisture content on the SFR based CALCALC program. The inspectors concluded that ET No. CEE-95-065 was technically sound. Engineers demonstrated a thorough understanding of the P-250 CALCALC program.

During power ascension for core uprate, operators monitored both FFR based and SFR based CALCALC reactor powers. The FFR based CALCALC remained higher than SFR based CALCALC computed power by approximately 0.5% to 1.0% on both units. Although the anticipated affects of higher moisture carryover on SFR based CALCALC were well understood, they could not be quantified until moisture carryover was quantified. The licensee decided to use the FFR based CALCALC value as the controlling reactor power indication to ensure the license limit was not exceeded following core uprate. The inspectors determined that this action demonstrated a sound safety perspective.

Engineers informed the inspectors that several post core uprate tests were scheduled to support return to the SFR CALCALC standard. Special tests 1-ST-0317 and 2-ST-0315, on Units 1 and 2, respectively, were in progress at the close of this report period to recalibrate feedwater flow instruments. Steam moisture carryover measurement tests are scheduled for December 1995. Engineers plan to use data from the above tests to rescale the flow instruments and update the P-250 CALCALC program to restore desired program accuracy. Criteria for rescaling feedwater and steam flow instruments were under development. The inspectors reviewed the P-250 CALCALC computer program with system engineers and discussed specific program changes which require evaluation in conjunction with flow instrument rescaling. The inspectors concluded that planned actions to restore SFR based CALCALC accuracy were reasonable.

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

7. 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, security, emergency preparedness, and fire protection.

The inspectors observed radiological control practices and radiological conditions throughout the plant. Portal and handheld monitors were observed to be in good condition and within proper calibration periodicities. Workers complied with radiation work permits and properly used required personnel monitoring devices. Radiological postings and control of contaminated areas were generally good. The licensee determined that on one occasion, contamination surveys had failed to detect fixed contamination on tools released from the RCA. On November 3, the licensee established additional controls including an independent verification survey by a second HP technician prior to releasing tools and equipment from the RCA. The inspectors determined that this interim action was adequate.

Selected aspects of plant physical security were reviewed during regular and backshift hours to verify that controls were in accordance with the physical 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 visitor escort. No discrepancies were noted. The inspectors observed security force practices from the CAS. The CAS operator was attentive to assigned duties and properly monitored perimeter surveillance devices required by the PSP. Vehicle access authorization was properly verified and controlled. Security force personnel were knowledgeable and performed their duties in an alert manner.

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

8. Self Assessment (40500)

8.1 Institute of Nuclear Power Operations Evaluation Review

On October 16, the inspectors reviewed the facility's INPO Evaluation Report, dated September 18, 1995, and discussed the report's findings with NRC management. No NRC follow up was planned for items identified in the report.

8.2 Control Room Annunciator Power Supplies

Problems were experienced with the power supplies to the Unit 1 A through E annunciator panels resulting in degraded operation of the annunciators in July 1995 and failure of these annunciators in August 1995. The events were discussed in NRC Inspection Report Nos. 50-280, 281/95-14 and 95-16. The licensee initiated RCE 95-11, RIS Power Supplies, to determine the cause of the power supply failures. The RCE was completed and approved on October 25, 1995.

The RCE determined that the power supply failures resulted from a ground loop created in 1994 by replacing an annunciator light bulb. The ground loop increased current flow and degraded the A through E panel power supplies. The RCE determined that the July event was initiated by a short circuit across a failed power supply light socket. This placed the failed power supply in parallel with the other power supplies and resulted in two other power supplies failing and the degradation of the remaining power supplies. The RCE determined that the August event was the result of two years of accumulative power supply degradation.

The RCE recommendations included evaluating replacement of the present system with a microprocessor based system, developing a procedure to monitor for power supply degradation, implementing a design change to install an annunciator ground detector alarm, and checking the condition of the Unit 2 annunciator power supplies during the next scheduled refueling outage. All these recommendations were accepted by licensee management. During the Unit 1 refueling outage, the licensee checked all the Unit 1 annunciator power supplies (panels A through K) and repaired/replaced any defective power supplies.

The inspectors consider that implementation of the above recommendations should greatly enhance the future reliability of the control room annunciators. The inspectors also verified that tracking items had been assigned where appropriate.

8.3 Four Inoperable Component Cooling Heat Exchangers

On October 7, 1995, the licensee determined that all 4 CCHxs had been inoperable from 10:50 p.m. on October 6, 1995, to 12:55 a.m. on October 7, 1995, due to inadequate service water flow. This event is discussed in NRC Inspection Report Nos. 280, 281/95-17. The licensee initiated RCE 95-13, Four Inoperable CCHxs, to determine the cause of the inadequate service water flow. The RCE determined that the inadequate service water flow resulted from tubesheet blockage by hydroids and other marine material. A contributing cause may have been air binding since the vacuum priming system was removed from service for maintenance during the event. The RCE determined that the debris resulted from returning the B high level intake structure to service after the bay had been drained for nine days allowing hydroids to slough off due to the changing environment and water velocity. Engineering analysis determined that the total heat removal capacity of the Component Cooling System was adequate for the required heat loads.

The RCE recommendations included revising procedures to require that the associated condenser waterboxes be flowed for a minimum of ten minutes prior to opening the associated service water supply valve if maintenance at the high level intake structure has been performed. Also, procedures were revised to require that one CCHx be isolated prior to returning associated condenser waterboxes or service water supply lines to service. RCE enhancements included establishing a PM program on the vacuum priming check valves to ensure that they function properly.

The licensee has issued LER 50-280/95-10 describing this event. Completion of the corrective actions to prevent recurrence will be reviewed during closure of the licensee's LER.

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

9. Licensee Event Report Followup (90712)

The inspectors reviewed LERs submitted to the NRC 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 followup. 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.

9.1 (Closed) LER 50-280, 281/95-09, Personnel Error Results in Loss of 4160 V Transfer Bus and Start of Emergency Diesel Generators.

During tagout activities, an operator erroneously removed the F transfer bus potential fuses resulting in a loss of power to the F transfer bus and the 1H and 2J emergency busses. This resulted in the #1 and #3 EDGs starting and energizing the 1H and 2J emergency busses.

This event was discussed in NRC Inspection Report Nos. 50-280, 281/95-17. The inspectors reviewed the LER and the root cause evaluation performed by the licensee after the event. The root cause determined that the error was due to the operator failing to perform an adequate self check. The operator was counseled and operations personnel were briefed on the details of the event and the lessons learned. The root cause evaluation recommended that a self check simulator be constructed and used during monthly and quarterly training to practice self checking and that simultaneous verification be performed when removing safety related equipment from service. Licensee management accepted the recommendation to construct a self check simulator but did not accept the simultaneous verification recommendation.

TS 6.4 requires written procedures and instructions for preventive or corrective maintenance operations. Instructions for tagging out equipment for maintenance activities were provided in VPAP 2002, Work Request and Work Order Task, revision 5. Failure to properly perform a tagout procedure constitutes a failure to comply with VPAP 2002 and TS 6.4. This licensee identified and corrected violation is being treated as a Non-Cited Violation, consistent with Section VII of the NRC Enforcement Policy. This NCV is identified as NCV 50-280, 281/95-19-02: Failure To Properly Perform Tagout Results In Loss Of Offsite Power To The 1H And 2J Emergency Busses.

9.2 (Closed) LER 50-280/95-011, Both Source Range Nuclear Instruments De-Energized due to Personnel Error. This item is discussed in paragraph 4.2 of this report.

Within the areas inspected, one NCV was identified.

10. Exit Interview

The inspection scope and findings were summarized on November 8 and 29, 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.)
NCV 50-280/95-19-01	Closed	Failure To Follow RPS Logic Testing Procedure (paragraph 4.2).
NCV 50-280, 281/95-19-02	Closed	Failure To Properly Perform Tagout Results In Loss Of Offsite Power To The 1H And 2J Emergency Busses (paragraph 9.1)
LER 50-280, 281/95-09	Closed	Personnel Error Results in Loss of 4160 V Transfer Bus and Start of Emergency Diesel Generators (paragraph 9.1).
LER 50-280/95-011	Closed	Both Source Range Nuclear Instruments De-Energized due to Personnel Error (paragraph 9.2).

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

11. Index of Acronyms

CAS	CENTRAL ALARM STATION
CCHx	COMPONENT COOLING HEAT EXCHANGER
CFR	CODE OF FEDERAL REGULATIONS
ECCS	EMERGENCY CORE COOLING SYSTEM
EDG	EMERGENCY DIESEL GENERATOR
ET	ENGINEERING TRANSMITTAL
F	FAHRENHEIT

FFR	FEEDWATER FLOWRATE
INPO	INSTITUTE OF NUCLEAR POWER OPERATION
LER	LICENSEE EVENT REPORT
MWT	MEGAWATTS THERMAL
NCV	NON-CITED VIOLATION
NRC	NUCLEAR REGULATORY COMMISSION
PM	PREVENTIVE MAINTENANCE
PPM	PARTS PER MILLION
PRNI	POWER RANGE NUCLEAR INSTRUMENT
PSP	PHYSICAL SECURITY PLAN
RCA	RADIOLOGICAL CONTROL AREA
RCE	ROOT CAUSE EVALUATION
RCS	REACTOR COOLANT SYSTEM
RFO	REFUELING OUTAGE
RIS	ROCHESTER INSTRUMENTATION SYSTEM
RO	REACTOR OPERATOR
RPS	REACTOR PROTECTION SYSTEM
SFR	STEAM FLOWRATE
SPS	SURRY POWER STATION
TS	TECHNICAL SPECIFICATION
UFSAR	UPDATED FINAL SAFETY ANALYSIS REPORT
VPAP	VIRGINIA POWER ADMINISTRATIVE PROCEDURE
V	VOLTS
WO	WORK ORDER