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

Report Nos.: 50-338/84-38 and 50-339/84-38

Licensee: Virginia Electric and Power Company
Richmond, VA 23261

Docket Nos.: 50-338 and 50-339 License Nos.: NPF-4 and NPF-7

Facility Name: North Anna 1 and 2

Inspection Conducted: October 6 - November 5, 1984

Inspectors: SAElrod 11/28/84
M. W. Branch (SRI) Date Signed

SAElrod 11/28/84
J. G. Luehman (RI) Date Signed

Approved by: SAElrod 11/28/84
S. Elrod, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine inspection by the resident inspectors involved 203 inspector hours on site in the areas of previous inspection findings, licensee event reports, IE Bulletins, previously identified items, design changes and modifications, plant startup from refueling, plant operations, surveillance and maintenance.

Results: Of the nine areas inspected, one violation was identified in the area of plant operations and is discussed in paragraph 12.

REPORT DETAILS

1. Licensee Employees Contacted

*E. W. Harrell, Station Manager
G. E. Kane, Assistant Station Manager
*M. L. Bowling, Assistant Station Manager
*L. Johnson, Superintendent, Technical Services
J. R. Harper, Superintendent, Maintenance
R. O. Enfinger, Superintendent, Operations
G. Paxton, Superintendent, Administrative Services
*A. L. Hogg, Jr., QC Manager
S. B. Eisenhart, Licensing Coordinator
J. R. Hayes, Operations Coordinator
J. P. Smith, Engineering Supervisor
R. C. Sturgill, Engineering Supervisor
D. E. Thomas, MechaNICAL Maintenance Supervisor
A. H. Stafford, Health Physics Supervisor
E. C. Tuttle, Electrical Supervisor
R. A. Bergquist, Instrument Supervisor
F. P. Miller, QA Supervisor
*F. T. Terminella, QA Supervisor

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

2. Exit Interview

The inspection scope and findings were summarized on November 6, 1984, with those persons indicated in Paragraph 1 above and the licensee acknowledged the findings.

3. Licensee Action on Previous Inspection Findings

(Closed) Infraction 338/79-39-06 Failure to Perform a Safety Evaluation in Accordance with 10 CFR 50.59(b) and NPS QAM. The licensee's response to the violation dated November 29, 1979, has been reviewed and the inspectors have no further questions.

(Closed) Violation 339/83-13-05 Improperly Set Intermediate Range (IR) Nuclear Instrument Trip Setpoints. The licensee responded to this violation in a letter dated November 18, 1983. The inspectors verified that, as stated in the response, the operating procedure covering ascension from mode 2 to mode 1 had been modified to include a step to visually verify illumination of the IR annunciators. Further, the inspectors verified that the licensee is predicting and then trending the IR trip-setpoint settings for each core load design. Finally, the licensee is trying to determine a fixed IR trip setpoint that will be acceptable for all future core load designs.

4. Unresolved Items

An unresolved item is a matter about which more information is required to determine if it is acceptable or may involve a violation or deviation.

Two unresolved items were identified during this inspection and they are discussed in paragraph 10.

5. Plant Status

Unit 1

The unit began the inspection period in a ramp-up to 100% power. The 100% power level was reached on October 8, 1984. At 1546 on October 12, 1984, a turbine/reactor trip occurred due to a loss of a power supply in the turbine Electro-Hydraulic Control (EHC) system. The loss of the power supply was the result of troubleshooting being done by an instrument technician. The reactor was taken critical late on October 12, 1984, reached 100% power on October 15, 1984 and ended the inspection period at or about that level.

Unit 2

During this inspection period, the unit completed a refueling outage. Work done in the outage included Appendix R, GDC 17 and environmental qualifications. Additionally, containment integrated leak rate testing was completed and the automatic shunt trip modification to the reactor trip breakers was installed.

On September 25, 1984, an inadvertent draining of the Casing Cooling System occurred. This is discussed in paragraph 12. Additionally, on November 1, 1984, an inadvertent injection of the Emergency Core Cooling System (ECCS) A & B Accumulator occurred. During a primary plant heatup with plant pressure at approximately 550 psig, the A & B Accumulators discharged approximately 350 gallons into the primary system. The discharge occurred when the outlet motor operated isolation valves (MOV) inadvertently opened while being energized. The MOVs were being energized as part of a preplanned evolution to establish conditions required by Technical Specification 3.5.1, however, the valves should have remained in the shut position. The licensee has determined that a spurious operation of the K628 relay (system pressure >2000 psi auto open signal) was the problem. The licensee has tested the circuitry and all equipment is operating properly. The licensee is planning on submitting Licensee Event Reports (LER) on both of the above events.

The reactor was started up on November 2, 1984, and following the completion of low power physics testing, the turbine was brought on line. The unit ended the inspection period in a secondary chemistry hold at 30% power.

6. Licensee Event Reports (LER)

(Closed) LER 338/83-67 Both level indicators for B ECCS accumulator tank improperly calibrated. This event was tracked as inspector followups 338 and 339/83-27-02 which were closed in inspection reports 338 and 339/84-06.

(Closed) LER 338/84-16 Contract employee received greater than 1.25 rem during the third quarter of 1984. This event was reviewed by the Region II Facilities Radiation Protection Section and will be discussed in upcoming inspection report 338 and 339/84-40.

(Closed) LER 339/83-41 Two containment isolation trip valves failed. As part of the scheduled corrective action, the licensee committed to evaluating the plunger spring and pin failures that caused one valve to fail and may have caused the other failure. The evaluation has been completed and the cause of the spring failure was determined to be hydrogen embrittlement. The licensee contacted the manufacturer (Valcor Engineering Corp.) and was told no other such failures had been reported to the company and further that such failure should not be a problem in the future because the springs are now manufactured of different material. Region II Materials and Processes Section has been informed to evaluate if there is a need for further followup.

At North Anna nitrogen is used a a redundant source of motive power for the Power Operated Relief Valves (PORV). The PORV nitrogen supply system has been a continuing problem as the following Licensee Event Reports (LER) document:

Unit 1 (50/338)

81-01, 81-02, 82-32, 82-40, 82-41, 82-91, 83-71

Unit 2 (50/339)

80-43, 80-50, 80-90, 81-63, 82-09, 82-23, 82-33, 82-42, 82-51, 83-30, 83-33, 83-39

In an effort to minimize such events, the licensee initiated a design change for each unit (82-S10 A & B, System Upgrade of the Nitrogen Supply to PORVs). The work done under these design changes included installing soft seat conversion kits in the nitrogen system relief valves, which were a large source of leakage, and installation of remote system pressure indicators to allow the control room operator to increase system pressure prior to receiving the low pressure alarm and falling below minimum acceptable system pressure.

The inspectors have reviewed the above listed LERs as well as the design changes and have no further questions. Additionally, discussions with various plant operators indicate that, although the system problems have not been completely remedied, actions taken to reduce system leakage have made the system "tighter" and the remote indicators in the control room are seen as useful in allowing the operator to correct falling system pressure before it becomes a significant problem. Problems noted with the annunciator response for the nitrogen system for the PORVs and drawing updates required by the design changes are discussed in paragraph 10.

The following LERs were reviewed and closed. The inspector verified that reporting requirements had been met, causes had been identified, corrective actions appeared appropriate, generic applicability had been considered, and the LER forms were complete. Additionally, for those reports identified by asterisk (*), a more detailed review was performed to verify that the licensee had reviewed the event, corrective action had been taken, no unreviewed safety questions were involved, and violations of regulations or Technical Specification conditions had been identified.

*339/80-43	PORV nitrogen system low pressure
*339/80-50	PORV nitrogen system low pressure
*339/80-90	PORV nitrogen system low pressure
*339/81-63	PORV nitrogen system low pressure
*339/82-09	PORV nitrogen system low pressure
*339/82-23	PORV nitrogen system low pressure
*339/82-42	PORV nitrogen system low pressure
*339/82-51	PORV nitrogen system low pressure
*339/83-30	PORV nitrogen system low pressure
*339/83-33	PORV nitrogen system low pressure
*339/83-39	PORV nitrogen system low pressure
339/83-41	Containment isolation valves failed to open
*339/84-04	Thermal overload devices not tested as required by T.S. 4.8.2.b.b
339/84-07	Control room emergency air supply smoke detector failed
*338/81-01	PORV nitrogen system low pressure
*338/81-02	PORV nitrogen system low pressure
*338/82-32	PORV nitrogen system low pressure
*338/82-40	PORV nitrogen system low pressure
*338/82-41	PORV nitrogen system low pressure
*338/82-91	PORV nitrogen system low pressure
*338/83-71	PORV nitrogen system low pressure
*338/83-67	Improper calibration on accumulator level channels
338/84-14	Startup reactor trip
*338/84-16	Occupational radiation exposure limit exceeded

7. IE Bulletins

(Closed) 338 and 339/83-BU-07 "Apparently Fraudulent Products Sold by Ray Miller Inc." The licensee has completed an extensive review of station purchasing documentation and a quality assurance audit has been completed in this area. The results of the investigators into the impact of products supplied to North Anna by Ray Miller Inc. are documented in a letter to the NRC dated April 5, 1984.

8. Followup of Previously Identified Items

(Closed) IFI 338/83-31-04 Possible Seismic considerations of temporary Primary Grade Water piping. As documented in inspection reports 338 and 339/84-04 almost all of this temporary piping has been removed. The failure to adequately address the seismic considerations was addressed under violations 338 and 339/84-04-02.

(Closed) IFI 338/79-39-07 Potential leak path from drain tanks. The licensee performed a design change (DCP 79-S73) to reroute the Volume Control Tank (VCT) relief line into the high level waste drain tank liquid space instead of the vapor space. The missing orifice in the piping was addressed in the licensee's response to a notice of violation (response dated November 29, 1979).

(Closed) IFI 338/83-11-02 Update of Unit 2 load list. This action was completed for the Unit 2 Load List which is what the item required.

(Closed) IFI 339/83-18-02 Minimum water volumes required by Technical Specification 4.7.14.1.1 required in all modes, however, logs list only modes 1-4. The inspection verified that the applicable logs have been changed to reflect the fact that these readings are required in modes 1-6.

(Closed) IFI 338/83-31-03 Identification and storage of fire retardant wood. Paragraph 3.5.2.b(3) of the Fire Protection Program outlines the proper identification of wood acceptable for use at the station. The station fire marshal has verified that the fire retardant wood in use at North Anna is treated so that long term outside storage should not significantly degrade fire retardant properties.

(Closed) IFI 339/83-24-03 Failure of pump performance tests to meet ASME XI requirements. Test 1-PT-57.1A has been changed to add acceptance limits. Test 1-PT-15.1 now requires the recording of ΔP, however flow data is not taken because the code allows the taking of either ΔP or flow for a fixed resistance system. The boric acid tank level in 1-PT-15.1 does not require an acceptable value for tank level specific to this test. The tank level is recorded in order to calculate inlet pressure for measuring ΔP since inlet pressure instrumentation is not installed. Finally, the reason for the difference in the "alert" high and low limits in 1-PT-14.3 and 1-PT-14.1 is that the limits have been determined from reference testing values obtained following repair of the individual pumps.

9. Recirculation Spray Heat Exchangers

During the 1984 Refueling Outage for both Units 1 and 2, several problems were discovered with the recirculation spray heat exchangers (Hx) which are located inside the Containment Building. The problems identified involved a crack being discovered in the welded seal area on the lower Hx heads and the possibility of the seal diaphragm being overstressed by external (containment) pressure during a containment pressurization accident. These problems were discussed in I/E Inspection Report 338 and 339/84-34 and LER 338/84-008 with major emphasis being the metallurgical problem involving cracks in the welded area of the diaphragm.

The issue of overstressing the Hx diaphragm by external (containment) pressure was evaluated and resolved by Engineering Work Request 84-613 dated September 8, 1984. The calculations performed by Stone & Webster indicated that during a Design Basis Accident (DBA) an unsealed diaphragm would be externally loaded. This external pressure loading would stress the diaphragm during the first 25 seconds of the (DBA) until service water could pressurize the internal surface of the diaphragm and negate the external effects. This external stress for the first 25 seconds would be slightly above yield but well below ultimate and the diaphragm would not fail. To eliminate any concern with external stress VEPCO installed two rings of silicone rubber sealant between the diaphragm and the head backing cover plate.

The review of VEPCO calculations and followup of any redesign of the Hx heads will be followed and reinspected as a result of inspector followup item 338, 339/84-34-03.

No violations or deviations were identified in this area.

10. Design, Design Changes, and Modifications (37700)

The design change and modifications program was reviewed by the inspectors. Specifically, the inspectors selected several design change packages and verified the following:

- design changes were reviewed and approved in accordance with Technical Specifications and established QA/QC requirements.
- design changes were controlled by established plant procedures.
- test records and results were reviewed and evaluated by appropriate licensee organizations.
- design changes were reviewed and approved in accordance with 10 CFR 50.59.
- operating procedures were modified and approved in a timely manner to reflect the modification.
- as-built drawings were updated in a timely manner to reflect the modification.

To evaluate the overall design change program, the inspectors reviewed Station Administrative Procedure 3.1 dated May 23, 1984, to ensure it provide necessary controls to effectively accomplish regulatory and industry requirements.

Section 5.2.2.1 of Administrative Procedure 3.1 discussed the controls of design change procedures. Specifically, the administrative procedure states, "Upon completion of each shift, the white working controlled copies are to be returned to the Design Change Log Room and those steps completed shall be signed in the Master Colored Controlled Copy by the supervisor immediately responsible and knowledgeable of activities completed." The inspectors questioned the Quality Assurance Manager and his staff on the actual method of controlling design change procedures and expressed the following regulatory concerns with the stated method:

- Technical Specification 6.10.2.a requires records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report be retained for the duration of the facility operating license.
- Amendment 4 of the VEPCO Topical Report VEP-1, "Quality Assurance Program, Operations Phase" commits VEPCO to collect, store, and maintain quality assurance records in accordance with NRC Regulatory Guide 1.88, Revision 2, 1976, which endorses ANSI N45.2.9-1974.
- Section 3.2 of ANSI N45.2.9-1974, provides instruction on what constitutes a valid quality assurance record.

The prevalent practice of allowing a person that ha not performed or witnessed an action step in a design procedure to sign for that action does not appear to be the present accepted industry practice nor does it appear to meet the intent of the above regulatory requirements. Additionally, the practice of discarding the "white-working copies" of the design change procedures detracts from the validity of the quality records.

Additional problems discovered during the review of specific design changes are listed below:

- DCP 84-05 (Automatic Actuation of Reactor Trip Breaker Shunt Trip Attachment, Unit 2). The Solid State Protection system manual has not been updated to reflect the recent modification.
- DCP 83-33 (Class IE SOV Replacement, Unit 2). A change to allow the installation of jumpers to keep TV-DA-200B was accomplished using the jumper logbook in lieu of a field change request as required by Station Administrative Procedure 3.1.
- Design Changes 82-S/O A & B: (System Upgrade of Nitrogen Supply to PORVs). The addition of remote system pressure indicators in the control room made it unnecessary to dispatch an operator to verify system pressure as required in the annunciator response procedure for low PORV N^2 pressure. Further, station drawings that include the nitrogen system (11715-FM-49A and 11715-FM-105A, including control room copy) have not been updated or annotated to reflect the addition of the pressure transmitters on Unit 1.

- An additional problem with the control of the Reactor Coolant Tave Upgrade Program for Units 1 and 2 was identified. Specifically, the coolant Tave upgrade, as approved by NRR, was not controlled by the design change program as required by Section 2.2.2 of the VEPCO Nuclear Power Station Quality Assurance Manual, Revision 3, dated June 18, 1984. There appears to be some question as to whether the QA Manual requirement come after the Tave upgrade was initiated. The licensee is reviewing the time elements of the above concern and this item is identified as unresolved item 338, 339/84-38-01 pending licensee review.

The problem associated with the control of design change procedures and the validity of the quality record is identified as unresolved item 338, 339/84-38-02, pending Region II review.

11. Plant Startup from Refueling (71711)

In preparation for Unit 2 startup following refueling, the inspectors reviewed 2-PT-94.0, "Refueling Nuclear Design Check" (the controlling document for the performance of physics testing) and its associated tests 2-PT-94.1-9. Additionally, the inspectors randomly verified as mode changes were made leading to startup, that the various required surveillances were completed. The inspectors witnessed portions of 2-PT-61.4, "RCS Pressure Isolation Valves - Leakage Test" which were required to be accomplished in accordance with Technical Specification surveillance requirement 4.4.6.2.2.d. Additionally, the inspectors witnessed the securing of the Residual Heat Removal (RHR) system using 2-OP-14.1, "Residual Heat Removal."

Prior to the Unit 2 startup, the inspectors independently verified the valve positions of critical valves in the Auxiliary Feedwater System (using 2-OP-31.2A), the SI accumulators (using 2-OP-7.3A), Refueling Water Storage Tank (RWST) (using 2-OP-7.7A), and NaOH Chemical Addition (using 2-OP-7.8A). The inspectors noted the following problems: 2-SI-173 had no tag, an operator had signed that 2-QS-36 was locked closed when in fact, it was only closed and 2-QS-68 and 69 were not in the positions indicated by the valve lineups. The position of the latter two valves did not degrade system operability. The problem with 2-QS-36 was immediately corrected by operations personnel and the two other problems were discussed at the monthly exit with the licensee.

On November 2, 1984, Unit 2 was started up and subsequently, the inspectors observed portions of the following physics tests:

2-PT-94.1 "Zero Power Testing Range"

2-PT-94.3 "Boron Endpoint Determination"

2-PT-94.4 "Isothermal Temperature Coefficient Measurement at Hot Zero Power"

12. Inadvertent Draining of Unit 2 Casing Cooling Tank

On October 25, 1984, while reperforming portions of Performance Test 2-PT-66.3, dated October 4, 1984, (Containment Depressurization Actuation Functional Test) approximately 82,000 gallons of water containing 2000 ppm boron inadvertently drained into the Unit 2 containment building basement. The specific events and details leading to the above flooding are listed below:

a. During the initial performance of 2-PT-66.3 on October 19, 1984, some equipment was either isolated or failed to meet established acceptance criteria. Attachment 6.3 of 2-PT-66.3 titled (Individual Equipment Test) specified terminal board and pin # to jumper or lift a lead from to cause individual equipment actuation. To individually test valves MOV-SW-204A and MOV-SW-204D, Attachment 6.3 stated to install a jumper at T.B. 905/5-6, for train "A". This action also resulted in the opening of MOV-RS-200A (Casing Cooling Pump Discharge Valve) which allowed the Casing Cooling tank to gravity drain to the containment sump.

b. Procedure 2-PT-66.3 was modified in 1981 to add Attachment 6.3. A facility modification occurring in late 1980 moved the contacts from 905/3-4 to 905/5-6 for valve MOV-RS-200A. This modification was not reflected in the 1981 performance test change and it appears that an outdated Elementary Schematic (ESK) 5BE was used to develop the 1981 change to performance test 2-PT-66.3.

c. Annunciator Response procedure 2-AR-10 for panel 2K Annunciator E-7 (Recirculation Spray Pump Casing Cooling Tank Level High/Low) specifies action necessary to respond to an Abnormal Condition. The response procedure requires an immediate determination of cause and termination of the event. However, the control room operator allowed approximately 82,000 gallons of casing cooling tank water to drain to the reactor containment basement floor unnoticed.

The defective performance test procedure 2-PT-66.3 and the failure to follow annunciator response procedure 2-AR-10 are identified as violations of Technical Specification 6.8.1 (339/84-38-03).

13. Routine Inspection

By observations during the inspection period, the inspectors verified that the control room manning requirements were being met. In addition, the inspectors observed shift turnover to verify that continuity of system status was maintained. The inspectors periodically questioned shift personnel relative to their awareness of plant conditions.

Through log review and plant tours, the inspector verified compliance with selected Technical Specifications and Limiting Conditions for Operations.

During the course of the inspection, observations relative to protected and vital area security were made, including access controls, boundary integrity, search, escort, and badging.

On a regular basis, radiation work procedures (RWPs) were reviewed and the specific work activity was monitored to assure the activities were being conducted per the RWPs. Radiation protection instruments were verified operable and calibration/check frequencies were reviewed for completeness.

The inspector conducted various plant tours and made frequent visits to the control room. Observations included: witnessing work activities in progress, verifying the status of operating and standby safety systems and equipment, confirming valve positions, instrument and recording readings, annunciator alarms, housekeeping and vital area controls.

No violations or deviations were identified in these areas.