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

Report Nos.: 50-338/90-05 and 50-339/90-05

Licensee: Virginia Electric and Power Company
Glen Allen, VA 23060

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

Facility Name: North Anna 1 and 2

Inspection at: VEPCO, Innsbrook Corporate Offices
Richmond, Virginia

Inspection Conducted: February 26-March 2, 1990

Inspectors: L R Moore for 3/19/90
M. Thomas, Reactor Engineer Date Signed

L R Moore for 3/19/90
B. Breslau, Reactor Engineer Date Signed

Approved by: Frank Jape 3/19/90
F. Jape, Section Chief Date Signed
Quality Performance Section
Division of Reactor Safety

SUMMARY

Scope:

This routine, unannounced inspection was conducted in the areas of followup on items identified during the design phase of the NRC SSOMI inspection. 50-339/89-200.

Results:

This inspection determined that the issues noted in the SSOMI report, 50-339/89-200, have been adequately addressed, except for two items identified in paragraphs 2.h. and 2.k. Paragraph 2.h involves Safeguards Information and will be followed by NRR. Paragraph 2.k., breaker coordination which the licensee is still analyzing, will be reviewed by Region II during a future inspection.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

V. Bhargava, Staff Engineer, Civil
R. Baldwin, Coordinator, Nuclear Training Support
*B. Bristow, Systems Engineer, Civil/Mechanical
L. Berry, Licensing Engineer, UFSAR Coordinator, (SPS)
P. Boulden, Senior System Engineer, (NAPS)
*R. Calder, Manager, Nuclear Engineering
B. Delamorton, Supervisor, Simulator Training Support
*G. Flowers, Manager, Electrical Engineering
*L. Hartz, Manager, ISI/NDE & Engineering Programs
S. Harvey, Supervisor, Testing (NAPS)
D. Heacock, Superintendent, Engineering, (NAPS)
*J. Hegner, Supervisor, Corporate Licensing (NAPS)
P. Knutsen, Supervisor, Nuclear Programs
*W. McBride, Supervisor, Nuclear Programs
T. Miller, Senior Engineer, Electrical Engineering
*F. Moore, VP Nuclear Engineering Services
R. Rasnic, Supervisor, Mechanical Engineering
R. Riley, Supervisor, Nuclear Project Engineering, (NAPS)
*C. Robinson, Jr., Manager, Civil/Mech. Engineering
*W. Rodill, Senior Staff Engineer
G. Rossetti, Design Control Engineer (NAPS)
M. Sartain, Senior Engineer, Nuclear Project Engineering, (NAPS)
D. Sommers, Supervisor, Corporate Licensing, (SPS)
J. Stall, Superintendent, Operation, (NAPS)
J. Wolak, Senior Engineer, Mechanical Engineering
C. Zalesiak, Staff Engineer, Civil Engineer/Mechanical Engineering (NAPS)

Other licensee employees contacted included engineers, technicians, and office personnel.

*Attended Exit Interview

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

2. Actions on Previous Inspection Findings (92701)

a. (Closed) IFI 339/90-05-01, Incorrect Differential Pressure Used in Sizing Service Water Reservoir Spray and Bypass System Isolation Valves (IC-1, Unresolved Item 50-339/89-200-01)

In response to questions raised by the NRC, VEPCO performed a review of safety related MOVs replaced or modified by the DCP or EWR processes for Units 1 and 2 and determined that the valves were adequately sized to meet design pressure differential and torque requirements. The licensee also reviewed EWRs which were generated to support maintenance or minor modifications which could impact valve performance.

The licensee documented the results of the expanded review in Technical Report No. ME-0021, Revision 1. Twelve EWRs were determined to require additional engineering evaluations. The additional evaluations determined that the modifications had no impact on the operability of the MOVs. The inspectors reviewed some of the additional evaluations and determined that the evaluations were adequate.

To ensure proper operation of the SW system spray array MOVs, the licensee stated that the torque switches were reset to a value of 5.0 to open and to close. Heavier spring packs were installed in the SW bypass valves. The inspectors reviewed EWR 89-191 which provided instructions for replacement of the spring packs. The spring packs were installed under WRs 627558,627559,627560, and 627561 for SW bypass valves 1-SW-MOV-123A,1-SW-MOV-123B,2-SW-MOV-223A, and 2-SW-MOV-223B, respectively.

The licensee also performed special test 1-ST-87, Differential Pressure Test on Service Water Bypass Valves 1-SW-MOV-123A and 2-SW-MOV-223B. The test was performed to determine the torque required to operate the SW bypass MOVs from an initially throttled position by simulating design basis conditions and measuring actual torque requirements using MOVATS equipment. The licensee determined from the test results that the maximum flow and differential pressure under design basis conditions were significantly less than the values assumed in preliminary engineering calculations.

Based on the inspectors' review, this item is considered closed.

b. (Closed) IFI 50-339/90-05-02, Setpoint Calculation Omission and Lack of an Approved Program for Performing Setpoint Calculations (IC-2, Unresolved Item 50-339/89-200-02).

The licensee was requested to review 10 setpoint calculations specified by the SSOMI team in order to ensure that the setpoints had been properly performed and safety margins had not been reduced. In their letter dated March 31, 1989, VEPCO also committed to develop a procedure for performing setpoint calculations and to review all modifications scheduled for implementation during the 1989 refueling outages for Units 1 and 2 to determine if any involved setpoint changes. The 10 setpoint calculations and the procedure for performing setpoint calculations were reviewed and discussed in NRC Inspection Report 50-338/89-10 and 50-339/89-10. Licensee personnel stated that it was determined from reviewing DCPs and EWRs scheduled for implementation during the 1989 refueling outages that the only design change involving instrument loop changes was DCP 87-29-2. This DCP was reviewed by the SSOMI inspection team.

Based on the information reviewed and the actions taken by the licensee, the inspectors considered that this item has been adequately addressed.

c. (Closed) IFI 50-339/90-05-03, Non Class 1E Loads Connected to Class 1E Buses Without Proper Isolation (IC-3, Unresolved Item 50-339/89-200-03)

This item was discussed and resolved during a meeting between NRR and VEPCO personnel on September 7, 1989. The meeting clarified a misunderstanding between the NRC SSOMI team and the VEPCO engineering staff. It was determined that, although the components in question were not classified as being safety-related, they were purchased to conform to Class 1E quality requirements.

In their June 30, 1989, letter to the NRC the licensee stated that procedure NAS-3012, Criteria Specification for Design and Identification of Electrical Cable Systems for North Anna Power Station Units 1 and 2, would be revised to distinguish between the isolation requirements of the original plant design and the isolation requirements of new designs. The inspectors verified that Revision 2 to procedure NAS-3012 distinguished between the isolation requirements of original plant design and plant design since April 1987, which is the time that NAS-3012 was first implemented.

Based on review of the actions taken by the licensee, the inspectors concluded that this item has been adequately addressed.

d. (Closed) IFI 50-339/90-05-04, Failure to Report Undersized Service Water Reservoir Bypass Isolation Valve Motor Actuators (IC-5, Unresolved Item 50-339/89-200-04)

The inspectors reviewed deviation reports 87-1405 and 87-1452 which identified problems with operation of the SW system bypass valves. Deviation report 87-1405 was written when bypass valve 1-SW-MOV-123B did not fully stroke closed on an initiation signal from the control room. The deviation report states that the valve in series with MOV-123B (2-SW-MOV-223B) was operable and closed. Licensee personnel stated that the deviation report was determined to be nonreportable because it met the requirements of 50.73(a)(2)(vi) which states in part that individual component failures need not be reported if redundant equipment in the same system was operable and available to perform the required safety function. This requirement was met because valve 2-SW-MOV-223B was operable and closed.

Licensee personnel also stated that it was determined from further investigation of the deviation report that the most probable cause of the valve failure to close was that the reactor operator did not realize that the bypass valves were throttleable and that the circuitry for the valve did not have a seal-in contact. Without a seal-in contact, the operator has to continually hold the handswitch in the open or closed position to move the valve to the fully open or closed position. Thus, there was no valve failure because the valve had actually performed as required.

Deviation report 87-1452 was written when it was discovered that the actuators for the SW bypass valves were undersized. Justifications discussed with the inspectors for the non-reportable determination were based on previously performed engineering calculations which demonstrated that the SW heat sink requirements could be met even with full bypass flow in the SW reservoir (i.e. full flow through the SW spray headers was not required during winter operation which was the condition at the time that the problem was identified); the SW bypass valves had functioned properly during the post modification operability test; and in order for the condition to exist where the SW bypass valves were in a positon that required a greater torque output to operate the valves than the actuators were capable of delivering, required a combination of operator error and multiple violations of operations procedures.

Based on the inspectors' review of the actions taken by the licensee at the time this problem was identified, the inspectors concluded that although there was a weakness in the documentation of this issue, the licensee's justification for not reporting this item was adequate.

e. (Closed) IFI 50-339/90-05-05, Inadequate Operator Training and Simulator Modeling (IC-6, Unresolved Item 50-339/89-200-05).

The inspectors reviewed Simulator Modification Report 890228090 and the presentations provided to the operators on the subject in order to verify that the simulator was corrected and the operators received training on the correct operations of the SW bypass valves. The inspectors also questioned licensee personnel concerning what actions were taken to determine why this modification did not get incorporated into the reactor operator training program. Licensee training personnel stated that at the time the SW bypass valves were changed, DCPs and EWRs were not reviewed with the same attention to detail as they are now. Licensee personnel stated that they have a high degree of assurance that there is complete information available for appropriate operator update training. This was based on the simulator upgrade program that was completed in December 1989; closer scrutiny of all DCPs and EWRs for potential impact on simulator hardware and/or software; and the performance feedback program which allows the training staff and licensed operations personnel utilizing the simulator to identify differences between plant system behavior and simulator response.

Based on review of the information provided, the inspectors concluded that the licensee has adequately addressed this item.

f. (Closed) IFI 50-339/90-05-06, Inadequate Loop Accuracy Calculation for Charging Flow Instrument (IC-7, Unresolved Item 50-339/89-200-06).

The licensee's corrective actions taken for this item and the concern noted in paragraph 2.2.6 of the SSOMI report are discussed in paragraph 2.o. of this report.

g. (Closed) IFI 50-339/90-05-07, Inadeqate Post-Modification Test Program IC-8, (Unresolved Item 50-339/89-200-07).

The inspector reviewed licensee Technical Report PE-0012. This report delineates the actions taken by the licensee in response to the SSOMI. The inspector reviewed four completed DCPs and six completed EWRs implemented during the outage to verify that adequate post-modification testing was included and documented.

Additionally, the inspector confirmed that additional testing, testing documentation, and/or testing requirements, where applicable, were included in the DCPs and EWRs. The DCPs and EWRs reviewed are listed in Appendix A.

The inspector also reviewed updated procedures covering the EWR and DCP process for preparation and testing of modifications. This enhancement ensures the modification did not inadvertently compromise ancillary system functions. These procedures required functional testing to at least one point upstream and one point downstream of the unaffected area of the modification. The changed procedures also require specific test instructions be developed with acceptance criteria to adequately test the functional operation of the modifications. Procedures reviewed are listed in Appendix B.

The licensee acknowledged that post-modification testing typically does not satisfy the requirements of the TS to declare the modified system as being operable, unless the post-modification testing meets the intent of the TS operability requirements. The inspector determined from his inspection that adequate administrative guidance is utilized in the modification test program.

h. (Closed) IFI 50-339/90-05-08, This Item Contains Safeguards Information, Transmitted Under Separate Cover Letter.

This item was not reviewed during this inspection. This item number is administratively closed. The Safeguards Branch in NRR has this concern under consideration and will subsequently provide a determination.

i. (Closed) IFI 50-339/90-05-09, Confirmation of Leak Detection Capability for Leak-Before-Break Analysis (MS-2, Unresolved Item 50-339/89-200-09). Design engineering was unable to provide information concerning the design and operation of the leak detection equipment utilized by the licensee in their commitment to detect a leakage rate of one gallon per minute leak in four hours.

The inspector reviewed the June 1989, vendor supplied documentation describing the licensee's reactor coolant system leakage detection system. This documentation provided an adequate description of the systems installed to identify, quantify, and locate leakage sources. Each system used to detect identified, unidentified, and intersystem leakage was described in sufficient detail. The details included such items as the description of the instrumentation, equipment calibrations, loop calibrations, surveillance, and periodic test.

Information was also provided as to which procedures were applicable to monitoring various conditions when leaks would be detected, such as through alarm response procedures, periodic test procedures, abnormal operating procedures, and surveillance procedures. Based on the inspector's review, this item is considered adequately addressed.

j. (Closed) IFI 50-339/90-05-10, Inadequate 10 CFR 50.59 Safety Evaluation (MS-3, Unresolved Item 50-339/89-200-10)MS-3, Unresolved Item 50-339/89-200-10, Several instances were noted where it did not appear that the requirements for performance of 10 CFR 50.59 safety evaluations were well understood. The procedures used at Surry and North Anna are not identical and differ from those used by corporate engineering in the design change and engineering work request processes.

The licensee committed to develop a common procedure. The common procedure will enhance consistency in the safety evaluations and reduce the probability of incorrect evaluations being preformed. The inspector determined from his review of the revised Surry and North Anna administrative procedures that each site's procedures accomplish safety evaluations in a consistent manner. These procedures adequately address the concern noted in the SSOMI.

k. (Open) IFI 50-339/90-05-11, Protective Devices on Safety Class Buses Not Coordinated (EP-1, Unresolved Item 50-339/89-200-11). A single fault at one of the 480-V buses could cause the 4160-V feeder breaker to trip, causing the loss of both of the 480-V load centers.

The licensee has completed Technical Report EE-0021, Coordination of Load Center Circuit Breakers, Revision 0. This report concluded that coordination does not exist between the 4160V load center feeder breakers and the load center main breakers for all potential faults. The report notes that better coordination could be maintained if the existing relays were replaced with GE IAC 53 relays. Further engineering analysis is required to study the settings and bases associated with new relays, protective devices and fault currents. This study is scheduled to be completed by April 15, 1990.

This SSOMI item number will be administratively closed. The results of the licensee's final analysis and corrective actions will be reviewed in a future inspection. This item will remain open and is identified as IFI 50-339/90-05-11.

l. (Closed) IFI 50-339/90-05-12, Inadequate Design Evaluation and Safety Evaluation of Quality Control Inspection Report (EP-2, Unresolved Item 50-339/89-200-12).

The inspector reviewed Technical Report CE-0020, Revision 0. This report provided the results of the licensee's review of 15 QCAR/IRs performed in conjunction with design changes. The licensee found on their initial review that all but three of the dispositions had been adequately substantiated; the three remaining required additional evaluations. The final conclusion was that no evidence existed to confirm a design review was compromised.

The inspector independently reviewed the 15 QCAR/IRs and supporting documentation and determined that the quality control verifications were adequately dispositioned.

m. (Closed) IFI 50-339/90-05-13, Oversized Holes in Base Plate Accepted Without Proper Justification (MC-1, Unresolved Item 50-339/89-200-13).

Additionally, the inspector noted the licensee has changed the Nuclear Design Control Manual, NDCM 3.7, Calculations, Revision 3. Revision 3 requires a reference section be included in each calculation; design inputs taken from existing sources are required to be reviewed to ensure they are correct and appropriate for the condition being analyzed. Where applicable, field walkdowns are required prior to finalizing a calculation. For example, fluid flow calculations for future piping modifications cannot be finalized until the system has received a walkdown. Change two to Revision 3 stipulates that it is not adequate to state "ok by engineering judgement" or any similar words.

The basis employed in any engineering analysis must be documented. The inspector determined from the above that guidance exists to ensure that an adequate design verification is performed on future modifications.

n. (Closed) IFI 50-339/90-05-14, Lateness in Updating UFSAR (CS-1, Unresolved Item 50-339/89-200-14). None of the UFSAR updates listed in DC-84-43-3 were implemented a the time of the SSOMI inspection.

The inspector's review indicated that the items noted in the SSOMI inspection report have been incorporated in the UFSAR. The licensee has developed and approved Nuclear Standard LINS-2802, Preparation and Control of UFSAR Updates, Revision 0. This standard requires Corporate Nuclear Licensing to establish a change program that ensures change packages are complete, controlled, tracked, and consistent with NRC requirements and commitments. It also requires that management be apprised of the status of the proposed changes and the changes are subsequently reviewed, approved, and that a revised UFSAR amendment is submitted to the NRC.

The licensee has implemented administrative procedure NL&P-ADM-3.2, Review, Coordination, Development and Tracking of Updated Final Safety Analysis Report Updates, Revision 0. The inspector verified that an adequate system had been implemented by conducting interviews with the Supervisors of Corporate Licensing and the UFSAR Coordinator coupled with a demonstration of the computerized program that tracks the UFSAR change packages. The above administrative procedure adequately implements LINS-2802 regarding the review, coordination, development and tracking of the UFSAR updates.

o. The NRC SSOMI inspection report addressed other concerns in addition to those identified as Unresolved Items. These concerns are addressed by their respective paragraph numbers.

(1) Paragraph 2.2.4, Too Many Revisions to Design Change Packages. The licensee has developed an action plan for reducing the number of avoidable field changes relative to DCPs for both North Anna and Surry Power Stations. The action plan identifies the responsible individuals and completion dates which are consistent with the licensee's Engineering Quality Plan. The Engineering Quality Plan is an effort implemented by the licensee's Corporate Engineering Department in order to improve the quality of its services and products. Attributes of the Action Plan aimed at reducing field changes include proper planning for the modification; a clear definition of the problem or need; proper level of participation in reviews and design development; good communications; control of the design organization; identification of lessons learned; and qualified individuals performing design, construction, and reviews.

In addition to the above actions, the licensee is considering a pilot program to determine if development of controlling procedures on site would result in lowering the number of DCP comments and resultant field changes.

The inspectors noted that the Action Plan is scheduled for implementation throughout 1990. While the Action Plan appears to be adequate for addressing the concern over excessive field changes, the overall effectiveness of the Action Plan cannot be assessed until after the Action Plan has been fully implemented.

Based on review of the licensee's actions, the inspectors concluded that this item has been adequately addressed.

(2) Paragraph 2.2.5, Engineering Work Requests - ALARA Considerations. Engineering work request procedure required the originator of a modification to consider the ALARA aspects of how operation of the plant with the modification completed might increase the radiation exposure of plant personnel.

The inspector reviewed the revised Administrative Procedure ADM-3.7, Engineering Work Request, Revision 07-27-89. Attachment 7.7, Modification Check List, Part 2, item 13, requires an ALARA review for post-modification installation. This procedure change provides adequate guidance concerning this issue.

(3) Paragraph 2.2.6, the inspection team's review of DC 87-29-2, determined the instrument loop accuracy (which was performed five months after the engineering standard was revised) did not consider current leakage in a postulated harsh environment.

In reponse to the SSOMI team's request, the licensee has conducted reviews of previous changes to the facility which may have affected instrumentation loops located in a harsh environment. The inspectors reviewed 10 of the approximately 71 calculation reviews performed. The licensee's calculation review process has been completed, except that the QA verification of the database information is ongoing and is scheduled to be completed by the end of April, 1990. Based on the results of the inspector's review and the implemented action of the licensee, this item is considered adequately addressed. Calculation packages reviewed are listed in Appendix C.

No violations or deviations were noted within the areas inspected.

3. Exit Interview

The inspection scope and results were summarized on March 2, 1990, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results listed below. Although reviewed during this inspection, proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

Item:	Description/Paragraph
50-339/90-05-11	Protection Devices on Safety Class Buses Were Not Coordinated, Paragraph 2.k.

4. Acronyms and Initialisms

ALARA	As Low As Reasonably Allowable
CFR	Code of Federal Regulations
DC	Design Change
DCP	Design Change Package
EWR	Engineering Work Request
FSAR	Final Safety Analysis Report
IFI	Inspector Follow-up Item
ISI	Inservice Inspection
MOV	Motor Operated Valve
MOVATS	Motor Operated Valve Analysis and Test System
NAPS	North Anna Power Station
NDCM	Nuclear Design Control Manual
NDE	Nondestructive Examination
NL&P	Nuclear Licensing and Programs
NRC	(United States) Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
QA	Quality Assurance
QCAR	Quality Control Activity Reports/Inspection Reports
SD	Station Directive
SPS	Surry Power Station
SSOMI	Safety System Outage Modification Inspection
SW	Service Water
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
UNR	Unresolved Items
V	Volt
VEPCO	Virginia Electric Power Company
VP	Vice President

APPENDIX A

UNIT 2 Outage

DCP	DESCRIPTION
85-08	ICCM System - T/Couple Repair & ICDM Cable Separation (FC 50)
87-12	ATWS FC Added Functional Testing To Blowdown TVs, FD Pumps and PTS To Be Performed Prior To Startup
88-12	RCS Level Indication

EWR	DESCRIPTION
87-673	Battery Charger Load Shed
88-100	CRDR - Vital Bus Status Lights
88-112	Replace SI Accumulator SOVs
88-330	Charging Pump Air Binding
88-357	FW Pipe Replacement

UNIT 1 Outage

DCP	DESCRIPTION
84-43	Service Water Relay Modifications

EWR	DESCRIPTION
89-179	FW Pipe Replacement

APPENDIX B

PROCEDURES REVIEWED

Document Number	Title
ADM-3.1 Revision	Control Of Design Change Implementation
ADM-3.7 Revision 07-27-89	Engineering Work Requests
ADM-11.1 Revision 12-15-88	Design Change Test Procedures
ADM-14.0 Revision 12-01-88	Tagging Of Systems And/Or Components
ADM-16.7 Revision 11-01-88	Work Order Processing
STD-GN-0001, Revision 8	Instructions For DCP Preparation (Review included changes 9 and 12)

APPENDIX C

CALCULATIONS REVIEWED

Document Number	Title
EE-0119	RCS WR Temperature Indicator
EE-0120	High Head SI Flow Indication
EE-0121	Main Steam Pressure Indication
EE-0126	Reactor Coolant System NR Pressure Indication
EE-0142	Pressurizer Level Indication
EE-0155	Letdown Flow Indication
EE-0161	Charging Water Flow Indication
EE-0175	RVLIS RCS WR Temperature Input Uncertainty
EE-0179	PZR Pressure Trip Uncertainty