U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Division of Reactor Inspection and Safeguards

Report No.: 50-339/89-200

Docket No.: 50-339

Licensee: Virginia Electric and Power Company (VEPCO)

Facility: North Anna Power Station Unit 2

Inspection at: VEPCO, Innsbrook Corporate Offices
Richmond, Virginia

Inspection Conducted: February 13 through 17 and February 27 through March 3, 1989

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1. INTRODUCTION AND SUMMARY

1.1. Background and Purpose

1.1.1 Background

The Nuclear Regulatory Commission (NRC) initiated the Safety System Outage Modification Inspection (SSOMI) Program in 1985. This program generally consists of two team inspection activities (1) an outage design inspection to evaluate planned design changes and modifications against regulatory requirements and licensee commitments; and (2) a preoperational readiness inspection to ensure plant readiness for startup through review of licensee controls, inspection of turnover package closeouts, verification walkdowns of installed systems, and observation of selected inprogress testing.

This report describes the activities and findings associated with the first phase of the SSOMI Program - the outage design inspection of North Anna Power Station, Unit 2.

Some of the items identified by the team may be potential enforcement findings. Region II will identify and execute any required enforcement actions.

1.1.2 Purpose

The purpose of this phase of the SSOMI Program was to examine, on a sampling basis, the detailed design and engineering required to support plant modifications planned during the current refueling outage. This assessment addressed the technical adequacy of modifications to ensure that the licensee had not violated any licensing commitments or regulatory requirements by designing and installing the modifications. Appendix B contains a complete list of all modification packages reviewed by the team.

1.2 Inspection Effort and Report Organization

1.2.1 Inspection Effort

NRC personnel conducted the inspection, with contractor assistance, at the licensee's engineering offices in Richmond, Virginia, during February 13 through 17 and February 27 through March 3, 1989. Inspection team members visited the North Anna site on March 1, 1989. Selected team members provided technical expertise and experience in each of the engineering disciplines evaluated during the inspection. Inspection activities concluded on March 3, 1989, with an exit interview held at the licensee's engineering offices, that was attended by those persons noted in Appendix A.

The SSOMI design inspection primarily emphasized the adequacy of design details or products as a means of measuring how well the design process had functioned. The team inspected five engineering disciplines within the scope of the project: instrumentation and controls, mechanical systems, electrical power, mechanical components, and civil/structural.

1.2.2 Report Organization

This inspection report is organized to present the team's findings in a format that will facilitate its use by different groups of readers with varied interests and responsibilities. Section 1.3 provides an overview of the team's activities and a summary of major findings organized by discipline. Sections 2 and 3 analyze the effectiveness of the licensee's design effort in terms of weaknesses and strengths, respectively.

Three appendices are attached to the body of the report. Appendix A lists personnel contacted during the inspection, and Appendix B lists all modification packages reviewed. Appendix C lists specific deficiencies organized by discipline and documented in detail to aid in their resolution.

1.3 Summary of Inspection Activities and Findings by Discipline

1.3.1 Instrumentation and Controls Discipline

The team reviewed the following design changes (DCs), engineering work requests (EWRs) and jumpers (i.e., temporary modifications) in the instrumentation and controls discipline:

(1)	DC 84-036-3	Addition of eight pressure transmitter loops, four temperature loops and four flow loops in the service water system.
(2)	DC 84-043-3	Complete instrumentation tie-in of the service water spray arrays.
(3)	DC 85-050-2	Addition of two, three phase relays for 4-kV undervoltage protection.
(4)	DC 87-012-2	Addition of ATWS system which would react after a common mode failure of the reactor protection system.
(5)	DC 87-025-2	Addition of position indicator lights on the main control board for pressurizer spray valves.
(6)	DC 87-029-2	Addition of flow transmitter and square root extractor in the charging flow loop and change in instrumentation ranges for control room indication.
(7)	DC 88-004-2	Elimination of reactor trip on turbine trip below 30 percent power.
(8)	DC 88-012-2	Addition of permanent reactor vessel level monitoring in the control room to be used when reactor vessel head is removed.
(9)	EWR 87-649	Replacement of motor-operated valve actuator due to inadvertent overthrusting.
(10)	EWR 87-658	Change in safety injection valve actuator gear ratio to improve stroke time.

(11) Jumper 842	Removal of temporary modification which eliminated a nuisance alarm associated with an inaccessible stuck limit switch. (Note that this was the only temporary modification scheduled to be worked this outage.)

In general, the team determined that VEPCO's program for performing design changes and modifications to the facility was adequate. Proper reviews were being performed and the design work appeared to be performed with adequate control. System interfaces were taken into consideration where required and appropriate attention was paid to regulatory requirements. The team found that design-related evaluations demonstrating compliance to Appendix R to 10 CFR 50, 10 CFR 50.59, ALARA, GDC-17 of Appendix A to 10 CFR 50 and electrical load growth were done in a detailed manner. The station jumper program was effective in controlling and reducing temporary modifications, and the setpoint program effectively prevented unauthorized setpoint changes.

Where the team identified weaknesses, those weaknesses usually appeared to result from poor engineering judgment or a less than full understanding of the problems identified due in part to a lack of staffing, rather than from overall weaknesses in VEPCO's design change programs. However, two programmatic weaknesses were identified. First, the team identified the need for an instrumentation setpoint procedure for establishing a consistent setpoint calculation methodology. Second, post-modification testing needed to be specified in the design change packages and engineering work requests to ensure engineering agreement with the testing required to demonstrate functionality of all affected equipment.

Also, the team's review of specific modification packages identified weaknesses in VEPCO's design and procurement of motor-operated valves (MOVs) in that certain service water system MOVs had been specified for a design differential pressure that was too small. Associated with this MOV issue, the team identified that the plant operators were not adequately trained with regard to operation of the service water bypass MOVs and that the simulator had been improperly modified to reflect operation of the service water bypass MOVs. Finally, the team found that a non-Class 1E transmitter on the service water system was improperly isolated from the Class 1E power supply.

1.3.2 Mechanical Systems Disciplines

The team reviewed the following design changes and engineering work requests in the mechanical systems discipline.

(1) DC 84-043-3 DC 84-031-3 DC 84-036-3	These DCs involved the addition of the new service water system spray arrays.
(2) DC 84-070	Fitting insulation around the pressurizer safety/relief valves to minimize waterhammer effects in downstream piping if valves open.
(3) DC 84-072-2	Upgrading of pipe supports for pressurizer safety/relief valve discharge piping.

(4)	DC 86-010-2	Removal of large-bore snubbers from primary coolant piping as a result of leak-before-break analysis.
(5)	DC 87-026-3	Reinstallation of steam generator downcomer flow resistance plate.
(6)	DC 88-004-2	Elimination of reactor trip on turbine trip below 30 percent power.
(7)	EWR 86-695	Removal of a block wall around the iodine filter unit in the containment ventilation system to permit changing of the filter.
(8)	EWR 87-022	Installation of isolation valve and calibration tee for charging pump auxiliary oil pump pressure switches.
(9)	EWR 88-112	Replacement of safety injection accumulator vent valves with environmentally qualified valves.
(10)	EWR 88-329	Installation of thermocouples on safety injection lines and pressurizer surge line to detect cold piping to prevent excessive thermal stresses.
(11)	EWR 88-330	Addition of vents to ensure that the charging pumps have a flooded suction to prevent air binding.
(12)	EWR 88-357	Inspection/replacement of nonsafety-related feedwater piping to ensure an acceptable wall thickness considering erosion/corrosion effects.
(13)	EWR 89-036 EWR 89-036B	Addition of diesel-driven air compressor and instrument air dryer in the instrument air system.

Overall, the team found that the licensee's modification programs appeared to be well controlled and effective in implementing the principles of 10 CFR 50, Appendix B.

The programs consisted of two primary elements: the design change (DC) process and the engineering work request (EWR) process. The DC was generally used for the more extensive modifications and the EWR for the smaller projects. The team reviewed the procedures controlling both processes (Procedures STDGN-001, Revision 8 and ADM-3.7, dated November 1, 1988, respectively) as well as a sample of modification packages of both types.

The team found that the procedures appeared to have the proper level of detail to ensure that modifications were correctly performed in a controlled, uniform manner. The team observed that the procedures appeared to provide the proper controls in areas that are often poorly controlled, such as setpoint changes and replacement equipment equivalency.

One weakness in the EWR procedure was observed by the team. Although the procedure required the originator of a modification to consider the ALARA aspects of installation, it did not require consideration of the ALARA aspects of the modification itself; that is, how operation of the plant with the completed modification might increase the radiation exposure of plant personnel and the general public.

Another aspect of the modification process, 10 CFR 50.59 safety evaluations, appeared to be well covered by the recently revised procedure, ADM-3.9, dated October 11, 1988. However, as evidenced in Finding MS-3 there were indications that what was required to perform a complete, comprehensive safety evaluation was not yet universally understood by persons who must perform them.

The team also observed that there appeared to be an inordinate number of revisions to many of the design changes reviewed. Although most of the changes were minor, such as dimensional changes, corrections for physical interferences, and changes to installation or testing procedures to correct for some unforeseen obstruction or difficulty, the number of changes seemed to indicate that there was insufficient attention to detail in the initial planning and generation of the modification packages.

1.3.3 Electrical Power Discipline

The team reviewed the following design changes and engineering work requests in the electrical power discipline.

(1)	DC 83-024-2	Addition of redundant fuses to prevent a fire in the main control room from adversely affecting emergency diesel generator operation.
(2)	DC 85-030-2	Replacement of station batteries.
(3)	DC 88-005-3	Addition of third reserve station service transformer to provide a back-up source of power for the 34.5 Kv transformer.
(4)	EWR 89-036	Addition of diesel driven air compressor to the instrumentation system.

As stated previously the design change was used for major plant modifications, and the engineering work request was used for minor modifications. Both types of modification packages contained the required design change information, installation instructions, and test procedures in sufficient detail to permit implementation of the modification as described in the DC or EWR. The DCs and EWRs reviewed by the team were comprehensive and well documented. Each package provided a high level of assurance that the design changes would meet the applicable licensing requirements and achieve the objective of the change without changing the plant design basis.

The team reviewed the planned design changes and modifications, the electrical design bases, and plant design margins. The scope of the review also included a verification that (1) adequate safety evaluations were performed to ensure that no unreviewed safety question existed, (2) independent design verifications were performed as required, and (3) post-modification tests were adequate and included acceptable acceptance criteria.

In general, the calculations that supported the modifications were documented in accordance with American National Standards Institute (ANSI) Standard N45.2.11. However, the team did note that the calculations supporting the switchyard modifications did not meet one or more of the following minimum requirements: (1) adequate references for design inputs, (2) evidence of review, and (3) summary of calculation results.

The team identified two findings as a result of the electrical power system review. First, the team identified that the 4.16-kV vital bus feeder breaker was not coordinated with its two downstream 480-Volt load center feeder breakers. This finding would have resulted in the loss of one complete safety division; it was the result of poor design practice and failure to meet VEPCO's commitment to Institute of Electrical and Electronics Engineers (IEEE) Standard 308. The second finding identified that VEPCO improperly dispositioned a quality control inspection report involving the routing of a nonseismic conduit over one of the four sets of station batteries. This finding raised concerns regarding adequacy of the review process both within the engineering organization and the station's 10 CFR 50.59 review process.

1.3.4 Mechanical Components Discipline

The team reviewed the following design changes and engineering work requests in the mechanical components discipline:

(1)	DC 84-072-2	Upgrade of pipe supports for pressurizer safety/ relief valve piping.
(2)	DC 84-085-3	Branch connections added to help monitor service water corrosion.
(3)	DC 86-010-2	Removal of large bore snubbers from primary coolant piping as a result of leak-before-break analysis.
(4)	EWR 87-022	Installation of an isolation valve and calibration tee for charging pump auxiliary oil pump pressure switches.
(5)	EWR 87-671	Modification of a snubber support baseplate to enlarge holes.

The modifications evaluated by the team involved a review of the associated piping stress analysis and pipe support designs. The majority of the calculations reviewed were performed by VEPCO's contractors, either Stone and Webster Engineering Corporation or Westinghouse. These calculations, in general, were technically adequate and supported the objective of the modification. However, the team did identify one VEPCO performed calculation where significant reanalysis was required to substantiate the modification (i.e., EWR 87-671) and was viewed to be an example of inadequate design review within the engineering organization. Overall, the team concluded that VEPCO's modification process in the mechanical components discipline was technically adequate.

1.3.5 Civil/Structural Discipline

The team reviewed the following design changes in the civil structural discipline:

(1) DC 84-031-3 Addition of service water reservoir spray and bypass piping.

(2) DC 84-035-3 Service water valve house structural analysis and design.

(3) DC 84-043-3 Service water reservoir improvements, final systems tie-ins (electrical/mechanical).

(4) DC 84-037-3 Service water reservoir improvements, buried piping.

In general the team found that the design changes were properly prepared and procedurally controlled. The calculations reviewed were performed by Stone and Webster Engineering Corporation as well as by VEPCO. The team was satisfied with the quality and content of the calculations.

The only weakness identified concerned timeliness in updating the final safety analysis report (FSAR). Specifically, the FSAR changes noted in design change 84-43-3 had not yet been made.

Additionally, the team interviewed VEPCO site personnel to evaluate how effectively engineering work requests and field change requests were processed. The team concluded that the site personnel initiated and executed those requests in accordance with the existing procedures and guidelines.

2. INSPECTION FINDINGS INDICATIVE OF LICENSEE WEAKNESSES

2.1 Design Verification

Design verification is the process of reviewing, confirming, or substantiating the design by one or more methods. When design reviews are used as the method of verification, the objective is to evaluate whether: (1) the inputs are correctly selected and incorporated in the design; (2) applicable codes, standards, and regulatory requirements are satisfied; (3) an appropriate design method is used; and (4) the design is suitable for the application.

During the inspection, the team assessed the quality of design verifications through the review of modification package details. This assessment revealed a weakness in the implementation of the design verification process, which suggested a need for greater attention to detail. Errors included: (1) incorrect sizing of service water motor-operated valve actuators; (2) improper isolation of nonClass 1E instruments from their Class 1E power supply, (3) inadequate design review of baseplate bolt hole enlargements, and (4) inadequate breaker coordination between safety class buses. The following sections describe examples that demonstrate areas of weakness in the licensee's design verification process.

2.1.1 Incorrect Sizing of Service Water Motor-Operated Valve Actuators

The team reviewed Design Change 84-043-3 and questioned VEPCO's justification of the 50 psi differential pressure used for sizing the service water valve actuators. VEPCO confirmed that the 50 psi differential pressure was too low and with Limitorque reanalyzed the actuators using 100 psi differential

pressure. The new calculation indicated that the installed actuators would not be able to deliver the required torque under the previously assumed 70 percent voltage condition. Additionally, the torque output of the motors would be limited by the actuator torque switch which had been set at a value coinciding with a valve differential pressure of 50 psi. As a result, several of the service water spray and bypass valves may not have operated as required under all design-basis conditions. The team considered this to be a safety-significant issue requiring resolution of the specific concern, and a review of all valves procured and designed by VEPCO since the facility was licensed. VEPCO's response documented in a letter dated April 28, 1989, confirmed that they had reviewed all safety-related MOVs that they had replaced or modified. The results of that review revised some torque switch settings and replaced actuator spring packs but, in general, concluded that all Unit 2 MOVs replaced or modified have been verified to meet differential design pressure and torque requirements.

2.1.2 Improper Isolation of NonClass 1E Instruments From Class 1E Power Supply

In the review of Design Change 84-036-3, the team identified that eight non Class 1E pressure transmitters were connected to vital instrument power buses without proper isolation. A nonClass 1E fuse block was used for isolation instead of a qualified Class 1E isolation device. A fault in the nonClass 1E portion of the system could potentially degrade the class 1E power supply. VEPCO's responses of March 31 and April 28, 1989, indicate that qualified isolation devices would be installed and that all modifications installed during the current outage would be reviewed to avoid similar problems. However, VEPCO has only committed to review this issue back until April 1987. It is the team's position that all similar isolation practices at North Anna need to be identified and corrected prior to the end of the next refueling outage. This matter will be the subject of a meeting between VEPCO and the NRC staff.

2.1.3 Inadequate Design Review of Engineering Work Request

EWR 87-671 enlarged the pipe support baseplate holes to facilitate installation during maintenance activities. The disposition of the associated field request indicated that the pipe support calculation had been reviewed and that the holes could be enlarged. The inspection team reviewed the subject pipe support calculation and identified that no design margin existed to justify enlargement of the baseplate holes. A reanalysis performed by VEPCO after the inspection team identified the concern, demonstrated that the existing design was adequate. However, the original disposition did not have adequate justification for permitting the change to be made. Therefore, VEPCO is requested to sample 10 pipe support field change requests, randomly selected, to ensure similar problems with design verification do not exist elsewhere in the facility. Also, VEPCO needs to address what programmatic controls are in place to ensure proper review of field change requests.

2.1.4 Lack of Breaker Coordination Between Class 1E Buses

VEPCO's Electrical Distribution Coordination Study identified that the 4-kV vital bus feeder breaker was not coordinated with its downstream 480-V load center feeder breakers, but concluded that the unit could be safety shutdown.

However, the inspection team was concerned that a fault in one of the two 480-Volt load centers would result in loss of both due to improper breaker coordination, resulting in the loss of one total electrical division. This lack of breaker coordination was indicative of a poor design practice and improper design verification. VEPCO was requested to review the subject breaker coordination to determine if the relays could be reset to provide adequate coordination.

2.2 Design Modification Interface Control/Design Process Control

Modifications were controlled at North Anna through the procedures for design changes, engineering work requests and jumpers (i.e., temporary modifications). Generally, the design change was utilized for the more significant modifications and the engineering work requests for smaller projects. As a result of this inspection, the team identified weaknesses in VEPCO's modification interface and design process control program: a plant change was not incorporated into operator training program and also was modeled incorrectly on the simulator, no methodology existed for evaluation of the effect of plant changes on instrumentation setpoint values, and post-modification testing did not perform all necessary functional testing subsequent to installation of the modification. These and other examples are also discussed below.

2.2.1 Inadequate Operator Training and Simulator Modeling

The addition of the new service water system also resulted in the addition of bypass valves which did not have a "seal-in" circuit. Therefore, the switch had to be held in the open position for the valve to open. VEPCO documented in Deviation Report 87-1405 an instance in which the bypass valves were thought to be fully open by the operator but were only partially open. Deviation Report 87-1405 identified that the most probable cause of the bypass valve not opening was failure of the operator to hold the valve's switch in the open position. The inspection team identified that the training received by the operators on the service water system did not specifically address operation of the bypass valves. Additionally, when the simulator was changed to reflect the new service water system configuration, the bypass valves were incorrectly modeled with switches that required only momentary contact to fully open the bypass valves. Therefore, neither VEPCO's specific training guidelines nor the simulator aided the operators understanding of how the bypass valves were controlled.

2.2.2 Lack of Programmatic Controls for Performing Setpoint Calculations

In its review of DC 87-029-2 which was issued to change instrument ranges, the team identified various errors in the loop accuracy calculations. These errors included the following: omission of measuring and test equipment accuracy, assumptions were not verified, and the instrument range change was not evaluated for its effect upon the associated setpoint. Also the team was concerned that VEPCO had been making hardward changes to instrument loops without the proper evaluation of the effect on the setpoint and associated safety margin. VEPCO was requested to review 10 specific setpoint calculations to ensure that the safety margin had not been adversely affected. The cause of these omissions was viewed by the inspection team to be the lack of an approved procedure for performing setpoint calculations and an unawareness of when setpoint calculations need to be revisited.

VEPCO's response of April 28, 1989 confirmed that the review of the 10 setpoint calculations was complete and where safety limits were applicable, the associated setpoints had a demonstrable margin of safety. Also in the response of March 31, 1989, VEPCO committed to review plant modifications installed during this refueling outage to assure that setpoint changes were properly performed and committed to develop a procedure for performing setpoints calculations.

2.2.3 Post-Modification Testing Requirements not Included in Change Packages

In its review of DC 87-012-2 for ATWS system installation, the inspection team noted that design change packages did not explicitly prescribe the necessary testing required to demonstrate functionality of the system and affected components following the change. Adequate modification control was not in place since the team identified two examples where the specific post-modification testing was being performed prior to the installation of the design change. Therefore, to ensure that modifications installed during the current outage were functionally tested subsequent to installation and to ensure that the capability of the affected systems to mitigate the design basis accidents had not been comprised, VEPCO was requested to incorporate the specific post-modification testing requirements into the design change packages and EWRs scheduled to be installed for the 1989 outage. Additionally, VEPCO committed to update the associated design change procedures to ensure that the required testing would be accomplished subsequent to the installation of future modifications. The purpose of including the post-modification testing requirements in the modification packages is to ensure that the engineering organization establishes the required scope of post-modification testing and that the site test group then selects and schedules the types of specific testing procedures to meet the testing objectives. At North Anna it appeared to the team that the site testing group was establishing the required scope of testing in lieu of the engineering organization.

2.2.4 Too Many Revisions to Design Change Packages

The team observed that there appeared to be an inordinate number of revisions to many of the design changes reviewed. Although most of the changes were minor, such as dimensional changes, corrections to account for physical interferences, and changes to installation or testing procedures to correct for some unforeseen obstruction or difficulty, the number of changes seemed to indicate that there was insufficient attention to detail in the initial planning and generation of the modification packages.

2.2.5 Engineering Work Requests - ALARA Considerations

The engineering work request procedure required the originator of a modification to consider the ALARA aspects of installation. The procedure did not require consideration of the ALARA aspects of the modification itself; that is, how operation of the plant with the modification completed might increase the radiation exposure of plant personnel.

2.2.6 Failure to Comply With Commitment Made in Response to a Previous NRC Violation

Violation 87-32-03 in NRC Inspection Report 50-338/87-32 and 50-339/87-32 cited VEPCO for failing to address effects such as leakage currents in total loop instrument accuracy calculations. In its response dated May 19, 1989, VEPCO committed to revise the associated engineering standard by August 31, 1989 to preclude further problems. During this inspection the team reviewed the Design Change 87-29-2 associated with the installation of a charging flow differential pressure detector. The team reviewed the associated calculation for determining the instrument loop accuracy, which was performed 5 months after the engineering standard was revised (i.e., February 10, 1989) and identified that it did not consider current leakage in a postulated harsh environment due to degradation of the cable insulation system. This is an example of improper commitment implementation in that the programmatic controls were in place but the design process result was unsatisfactory. After VEPCO became aware of this concern, the calculation was corrected with no detrimental effect. However, VEPCO is requested to review other previous changes to the facility which may have affected instrumentation loops located in a harsh environment.

2.3 Safety Evaluations

Licensees may make changes to the facility without prior Commission approval providing the proposed change does not involve a change to the technical specifications or an unreviewed safety question. Safety evaluations are performed to ensure that the aforementioned requirements are met. The team was satisfied with the level of detail in the safety evaluation checklist (i.e., Attachment 1 to Administrative Procedure ADM-3.9, October 11, 1988). However, the team was concerned that VEPCO did not correctly implement the safety evaluation procedure for the two following examples and consequently VEPCO may not completely understand the requirements for performance of safety evaluations.

2.3.1 Safety Evaluations for Removal of Block Wall and Effects of NonSafety-Related Equipment

Engineering Work Request 86-695 removed a portion of a block wall around the iodine filter unit. The associated safety evaluation only considered the effects of the wall removal of the function of the iodine filters. It failed to address the effect on radiation shielding on personnel/equipment and the effect on the original intended function of the wall. Engineering Work Request 89-036 and 89-036B added a diesel-driven air compressor and an air dryer, respectively, to the instrument air system. In the safety evaluation for both EWRs VEPCO reasoned that since no safety-related equipment was modified, no unreviewed safety question was involved. The team noted that whether or not safety-related equipment was being modified is not the only criteria by which to judge whether or not an unreviewed safety question was involved. A simple example of nonsafety-related equipment located over and potentially falling on safety-related equipment demonstrates the flaw in that logic.

2.3.2 Safety Evaluation Failed to Identify Obvious Error in Disposition of Quality Control Inspection Report (QCIR)

The inspection team reviewed the design change (DC 85-030-2) associated with the replacement of station batteries. Included in the design change package was a QCIR which identified that a nonseismic conduit was routed directly above one of the station batteries. The disposition of the QCIR by engineering was use-as-is with the justification that only one of the four channels would be lost during a seismic event. The inspection team identified that an improper engineering design verification was performed and that a properly performed safety evaluation should have identified the disposition as unacceptable. Subsequent to the team's finding VEPCO performed a seismic qualification of the electrical conduit to demonstrate that one battery channel was not lost during a seismic event. However, this subsequent justification did not resolve the concern that the safety evaluation should have identified the QCIR disposition error, which was due to either a procedural shortcoming or inadequate review.

2.4 Slow FSAR Updating

The team noted in its review of Design Change 84-43-3 that FSAR changes were identified for various revisions from July 1986 to June 1987. However, none of these changes were incorporated into the FSAR at the time of the inspection. VEPCO is required to submit FSAR changes no less frequently than annually for all changes made up to a maximum of six months prior to the date of filing.

3. INSPECTION FINDINGS INDICATIVE OF LICENSEE STRENGTHS

During the inspection, the team evaluated many design change packages and EWRs as identified in Appendix A including the removal of one jumper. Some problems were identified, however, the majority of positive findings led the team to conclude that, in general, controls were in place to result in an adequate design product. The team also found the licensee engineering staff to be technically knowledgeable and in general, familiar with the North Anna facility. The following sections discuss examples of licensee strengths.

3.1 Control of Design Input

The team found that the licensee's modification packages were supported by detailed, comprehensive design requirements. These documents provided adequate design-basis data and adequate procedural control existed to ensure the preparation, review and approval requirements were met. Where calculations existed, the team observed that the stated purpose was effectively supported by the analysis and documented in such a manner that an independent review could be readily performed. Contractor performed calculations were generally thorough, especially those reviewed in the mechanical components discipline, and demonstrated a good design communication process between VEPCO and the contractor.

3.2 Design Modification Control

Contrary to the previously concerns described in the design modification interface/design process control section, overall the team felt that the procedures governing the control of design modifications were quite good. The team was impressed with the coordination reviews performed for design changes at the 30 percent and 70 percent

conceptual completion levels which demonstrated good internal coordination between all affected organizations (engineering, installation, test, operations, etc). Additionally, VEPCO was performing the majority of modifications itself (approximately 80 percent) with the balance contracted, primarily to the original architect-engineer. The changes were supported by a competent engineering organization, centrally located in Richmond, but well represented at the site and available to the facility. The temporary modifications (i.e., jumpers) were well controlled and virtually nonexistant since only one was being worked during this outage and that was for removal.

3.3 Procurement of Safety-Related Electrical Components

The team reviewed Design Change 87-12-2, "ATWS Mitigation System Actuation Circuitry (ASMAC)" and identified it as an example of an acceptable procurement process. The review concentrated on the procurement of the ASMAC panel to be used in the new ATWS System. The panel was purchased by VEPCO as safety related with 10 CFR Part 21 and Appendix B to 10 CFR 50 invoked on the purchase order. The panel was purchased from United Controls, who in turn procured the safety related output relays from Electroswitch. The output relays were seismically qualified to Electroswitch Qualification Test Report 2983-3, "Qualification Inspection of Series 24 LOR, LOR/ER, and LSR Auxiliary Relays and Lockout Relays." The procurement of this panel was found to be well planned and effectively implemented by the cognizant VEPCO personnel.

APPENDIX A

PERSONNEL CONTACTED

NAME	POSITION
T. Abercrombie	Electrical Engineer
R. Berryman	Manager - Nuclear Analysis & Fuel
G. Bischof	Staff Engineer, Civil/Structural
*D. Blakenship	Senior Staff Engineer, Electrical - I&C
R. Boehling	Project Engineer - Nuclear
*P. Boulden	North Anna Power Station, System Engineer
*M. Bowlin	Assistant Station Manager
P. Bradley	Electrical Engineer
P. Buhl	Engineer, Electrical Engineering
*R. Calder	Manager, Nuclear Engineering
*R. Carroll	Nuclear Engineer
*W. Cartwright	Vice President - Nuclear
*D. Compton	Senior Staff Engineer - Civil/Eng. Mech.
G. Darden	Nuclear Engineer
*J. Davis	Manager Nuclear Site Services
B. Douglas	Senior Construction Specialist
B. Dunlap	Project Engineer - Nuclear
K. Dwivedy	System Engineer - Civil/Eng. Mech.
*M. Gettler	Superintendent - NSS
*D. Glasska	Senior Engineer (SEO) - Mechanical
J. Graf	Electrical Engineer
S. Harvey	Electrical Engineer
D. Heacock	Superintendent - Engineering
*J. Hegner	Supervisor - Licensing
B. Hill	Electrical Engineer
R. Hurd	Staff Engineer, Site Purchasing
B. Jones	QC Supervisor
*J. Leberstein	Licensing Engineer
J. Lewis	North Anna Power Station System Engineer
*J. MacCrimmon	Supervisor - Civil Engineering
*J. Maciejewski	Manager, Quality Assurance
D. Madden	Senior Engineer - Civil/Eng. Mech.
F. McLaden	North Anna Power Station - Maintenance Engineer
G. Midas	Project Engineer - Nuclear
*T. Miller	Electrical Engineer
*F. Moore	Vice President - Power Engineering Services
W. Murray	Nuclear Engineer
G. Pannell	Director, Licensing Group
R. Pavlik	Senior Staff Engineer - Civil/Eng. Mech.
M. Phillips	Electrical Engineer (Power)
M. Pinion	Project Engineer - Nuclear

APPENDIX A

PERSONNEL CONTACTED - CONT.

NAME	POSITION
C. Ranganath	System Engineer - Civil/Eng. Mech.
*R. Rasnic	Supervisor - Mech./Nuclear Eng.
J. Regic	Senior Staff Engineer
*R. Riley	Supervisor - Project Engineering
C. Robinson, Jr.	Manager, Civil Engineering
M. Sartain	Project Engineer - Nuclear
N. Smith	System Engineer - Nuclear Fuels
*W. Stewart	Senior Vice President - Power
K. Stacy	Electrical Engineer
R. Sturgill	North Anna Supervisor of System Eng.
E. Taylor	Staff Engineer - Mech./Nuclear
*W. Thomas, Jr.	Senior Staff Engineer - Mech./Nuclear
*W. Thompson	Manager, Electrical Engineering
J. Thornton	Staff Engineer - Civil/Structural
M. Vick	Electrical Engineer
A. Vig	Senior Staff Engineer - Civil/Eng. Mech.
M. Weeks	Contractor to VEPCO
L. Wroniewiz	Supervisor - North Anna Site Eng.
C. Zalesiak	Staff Engineer - Civil Eng./Eng. Mech.

*Attended Exit Meeting

APPENDIX B

DESIGN CHANGES (DC), ENGINEERING WORK REQUESTS (EWR)
AND JUMPERS REVIEWED BY THE INSPECTION TEAM

*DC 83-024-2	Appendix R, Emergency Diesel Generator
*DC 84-031-3	Reservoir Spray and Bypass System
*DC 84-035-3	Valve House Structural Analysis and Design
*DC 84-036-3	Valve House Electrical, Mechanical and Final Structural
*DC 84-37-3	Buried Piping
DC 84-043-3	Service Water Reservoir Improvements Final System Tie-In and Startup,
DC 84-070	Pressurizer Oven Installation, Rework
DC 84-072-2	Pressurizer Safety and Relief Valve Discharge Pipe Support Modifications
DC 84-085-3	Pipe Preservation - 14 inch Component Cooling Heat Exchanger Branch
*DC 85-030-2	Replacement of Station Batteries
*DC 85-050-2	Emergency Bus Undervoltage Relay Replacement
DC 86-010-2	Large Bore Snubber Leak-Before-Break Modifications
DC 87-012-2	ATWS Mitigation System Modification
DC 87-025-2	Control Room Design Review - Pressurizer Spray Valve Indicator Lights
*DC 87-026-3	Steam Generator Downcomer Flow Resistance Plate Installation
DC 87-029-2	Control Room Design Review - Installation Range Changes
DC 88-004-2	Eliminate Reactor Trip on Turbine Trip Below 30 Percent Power
DC 88-005-3	Installation of Third Reserve Station Service Transformer
DC 88-012-2	Reactor Coolant System Level Indication
EWR 86-695	Iodine Filter Upgrade
EWR 87-022	Install Isolation Valve and Calibration Tee for Charging Pumps
*EWR 87-649	Evaluate Overthrust of a Charging MOV Actuator
*EWR 87-658	Safety Injection MOV Failed Its Stroke Time
*EWR 87-671	Modify Snubber Support Base Plate
EWR 88-112	Replace Safety Injection Accumulator Solenoid Operated Valves
EWR 88-329	Installation of Instruments in Response to NRC Bulletin 88-08
EWR 88-330	Charging Pump Air Binding
EWR 88-357	Feedwater Pipe Replacement
EWR 89-036	Install Diesel-Driven Air Compressor
EWR 89-036B	Install Instrument Air Dryer (DRAFT)
Jumper 842	Alarm In Control Room for an Inaccessible Stuck Limit Switch

*Indicates that DC or EWR was installed during previous outage

APPENDIX C

FINDINGS

Nomenclature/Contents:

FINDING IC-1: Incorrect Differential Pressure Used in Sizing Service Water Reservoir Spray and Bypass System Isolation Valves. (Unresolved Item 89-200-01).

Discussion:

The inspection team reviewed the portions of Design Change 84-43-3 pertaining to the installation of the new service water reservoir spray and bypass system isolation valves. Specifically, the review concentrated on the methodology and assumptions made in sizing the motor actuators for those specific valves. During the review, it was noted that VEPCO Specification No. NAS-2018 referred to a maximum differential pressure rating of 50 psi which was used by the valve vendor in sizing the subject actuators. In response to the inspection team's concerns, VEPCO was unable to provide a justification for the 50-psi maximum differential pressure. VEPCO calculations performed during the inspection indicated that a differential pressure of approximately 100 psi could exist across the affected valves at a pump shutoff head condition. The inspection team then expressed concern that the valve actuators, sized for a differential pressure of 50 psi, might not be able to open or close the affected valves should the differential pressure be above the assumed 50 psi. It should be noted that valve thrust requirements are proportional to the differential pressure across the valve. VEPCO in conjunction with Limitorque, then performed calculations in order to determine if the actuators could still stroke the subject valves with an assumed differential pressure of 100 psi (pump shutoff head). The new calculations indicated that the installed actuators would not be able to deliver the required torque to the subject valves under the previously assumed 70 percent voltage. Additionally, the torque output of the actuators would be limited by the actuator torque switch which had been previously set for a valve differential pressure of 50 psi. As a result, several of the service water reservoir spray and bypass system isolation valves might not have operated as required under all design basis conditions.

VEPCO's March 31, 1989 response indicated that the service water spray valves needed the torque switches reset and the service water bypass valves needed new spring packs for their actuators. Since the spring packs cannot be procured prior to restart, VEPCO is instituting administrative controls to preclude adverse positioning the bypass valves in Modes 1 through 4. VEPCO's

April 28, 1989 response indicated that they had confirmed the adequacy of all safety-related MOVs that had been replaced or modified for Unit 2. As a result of this generic review, VEPCO indicated that additional torque switch setting changes were required. This item remains open pending completion of a similar review of Unit 1 MOVs that were replaced or modified.

Regulatory Basis:

Criterion III of 10 CFR 50 Appendix B requires that measures shall be established to assure that applicable regulatory requirements and design bases are correctly transferred into specifications, drawings and procedures.

References:

(1) VEPCO Design Change 84-43-3, "Service Water Reservoir Improvements, Final System Tie-In and Startup/North Anna Units 1&2."

(2) VEPCO Specification No. NAS-2018, "Specification for Motor Operated Butterfly Valves for Service Water Reservoir Spray and Bypass System Isolation Valves/North Anna Power Station Units 1 & 2," Revision 1.

FINDING IC-2: Setpoint Calculation Omissions and Lack of an Approved Program for Performing Setpoint Calculations. (Unresolved Item 89-200-02).

Discussion:

The team reviewed DC 87-29 issued for control room design review (CRDR) instrumentation range changes. This DC was issued to change the ranges of a few Class 1E transmitters, replace the charging flow transmitters and add a square-root extractor in charging flow loop. The team noted the following:

a. The loop accuracy calculation performed for the charging flow loop did not include measurement and test equipment (M&TE) accuracy.

b. The accuracy calculation had many unverified assumptions, including allowances for uncertainties in process variables following a design basis-event, calibration period, letdown pressure and charging pressure. Also, the team noted that there was no existing program for tracking and resolving these assumptions before making the modification operable.

c. The effect of instrumentation range changes on the associated setpoints was not evaluated.

Subsequent to inspection team's finding, VEPCO engineers performed setpoint calculations for transmitters which had undergone range changes for DC 87-29, and these calculations were acceptable.

The team requested to review the plant's setpoint calculations, but only the emergency operation procedure (EOP) indicating instruments uncertainty calculations were provided. These calculations used assumed values of drift instead of vendor-published actual values as related to the surveillance frequency of each loop. Also, these calculations used assumed values for calibration accuracy.

The team believed that VEPCO should have a corporate procedure for performing engineering evaluations of setpoints resulting from design changes. The team was informed by VEPCO engineers that the need of a uniform corporate level procedure had been previously identified by VEPCO and was currently being developed. In addition, VEPCO indicated that setpoint calculations for all safety-related systems would be reviewed as part of the design bases reconstitution (DBR) program scheduled to be completed within the next 3 to 5 years. To ensure that safety-related setpoints had not been adversely affected since the plant was licensed and since the DBR program was not scheduled for the immediate future, the inspection team requested VEPCO to review 10 safety-related I&C loops.

VEPCO's response of March 31, 1989, committed to develop a controlled procedure for performing setpoint calculations and to review all modifications for this outage to ensure setpoint calculations were properly performed. Also, in the April 28, 1989 response, VEPCO confirmed that the 10 setpoint calculations reviewed had no adverse impact on safety limits on the associated margin of safety.

The inspection team reviewed two administrative procedures, EEN-0211 for setpoint documentation and ADM-6.8 for administrative control of setpoint changes. The team noted that VEPCO had effective control which prohibited unauthorized changes to setpoints and an efficient way for documentation of setpoint changes.

This item remains open pending VEPCO's verification of the issuance of approved procedure for performing setpoint calculations.

Regulatory Basis:

Criterion III of Appendix B to 10 CFR 50 states, in part, "Design changes, including field changes, shall be subject to design control procedures commensurate with those applied to the original design...".

ANSI N45.2.11, section 8 Design change control states, in part, "Documented procedures shall be provided for design changes to approved design documents, including field changes, which assure that the impact of the change is carefully considered...".

References:

(1) DC 87-29

(2) VEPCO, Setpoint Document Standard EEN-0211, Revision 1, November 23, 1987

(3) VEPCO, Administrative Procedure for Setpoint Changes ADM-6.8, June 26, 1988

FINDING IC-3: NonClass 1E Loads Connected to Class 1E Buses Without Proper Isolation. (Unresolved Item 89-200-03).

Discussion:

DC 86-34-3 renovated the common service water system for North Anna Units 1&2 including installation of eight pressure transmitter loops, four temperature

loops and 4 flow loops. All of these instruments were classified as nonClass 1E. However, the eight pressure transmitter loops were powered from 120-V ac Class 1E vital instrument power buses without proper isolation. The only isolation between nonClass 1E circuits and the Class 1E power source was a nonClass 1E fuse block. In such a situation, a potential existed to degradate the Class 1E power source due to a fault on the non1E portion. Since this power source fed the instruments of many safety systems, degradation of the power source might affect the operation of many safety systems.

VEPCO's responses of March 31 and April 28, 1989, committed to providing the appropriate isolation device for the subject pressure transmitters, as well as, reviewing all modifications to be installed during the current outage and making any necessary changes prior to restart. Additionally, VEPCO committed to review all modifications made since April 1987, update the associated procedure and provide training to the affected personnel. However, the team believes that a complete review of this isolation design practice needs to be done for the entire North Anna facility independent of modification installation date. This review and any associated changes should be completed prior to the end of the next refueling outage. VEPCO has verbally requested the Project Manager to arrange a meeting with the NRC staff to discuss this matter. This item remains open.

Regulatory Basis:

Criterion III of Appendix B to 10 CFR Part 50 states in part; "Design changes, including field changes, shall be subject to design control procedures commensurate with those applied to the original design...."

References:

(1) DC 86-34-3

FINDING IC-4: MOV Thrust Calculation not Performed when Changing Actuator Gear Ratios. (Closed).

Discussion:

The team reviewed Engineering Work Request 87-658 which identified that safety injection valve SI-MOV-2867B failed to complete its required stroke in the allotted time. Corrective action was to change the valve actuator gear ratio from 63:1 to 55.8:1. The lower gear ratio would decrease the stroke time of the valve; however, it would also decrease the maximum thrust output of the actuator. No calculation was performed to demonstrate that the actuator as configured could still deliver the required thrust to stroke the valve under all assumed design-basis conditions. After the modification, normal stroke time testing performed at nominal plant conditions would not be adequate to ensure operation of this valve for all design basis conditions. After the team identified this finding, VEPCO generated a new calculation for this actuator which showed that the reduced thrust output would still be sufficient to operate this valve under all design basis conditions. This item is closed.

Regulatory Basis:

Criterion III to 10 CFR 50 Appendix B requires that design changes be subject to measures commensurate with those applied to the original design.

References:

(1) VEPCO Engineering Work Request 87-658 dated October 22, 1987

FINDING IC-5: Failure To Report Undersized Service Water Reservoir Bypass Isolation Valve Motor Actuators. (Unresolved Item 89-200-04).

Discussion:

As part of its review of DC 84-43-3, the team reviewed Deviation Reports 87-1405 and 87-1452 which identified problems with valve motor actuators installed by the design change. Deviation Report 87-1405 was written when one of the service water reservoir bypass valves failed to close on initiation of a signal from the control room. As a result, the valve was manually closed and all bypass motor-operated valves were isolated in the closed position.

On reviewing the specifications for the affected valves, a VEPCO engineer noted that the bypass valve motor actuators were apparently undersized. As a result, Deviation Report 87-1452 was written on December 22, 1987, documenting the apparent undersizing and recommended replacement of the actuators. On December 23, 1987, VEPCO determined that this deviation was "not reportable" because the bypass valves were de-energized and locked closed and that flow would have been maintained through all safety-related components if the bypass valves had failed in the open position.

This item should have been reportable under 10 CFR 50.72(b)(2)(i) or (iii)(B), or 10 CFR 50.73(a)(2)(ii)(B) or (v)(B) because when the deviation was written, VEPCO thought that the service water reservoir bypass isolation valves were inoperable because the actuators were undersized. Although flow would have been maintained to all safety-related components if the bypass valves had failed to close, this situation would be outside the plant design basis which assumes full flow through the service water spray headers. This item remains open pending justification from VEPCO for not reporting the undersized service water bypass valve motor actuators.

Regulatory Basis:

10 CFR 50.72(b)(2)(i) requires that a licensee notify the NRC within 4 hours of "any event, found while the reactor is shutdown, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principle safety barriers, being seriously degraded or being in an unanalyzed condition that significantly compromises plant safety."

10 CFR 50.72(b)(2)(iii)(B) requires that a licensee notify the NRC within 4 hours of "any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to remove residual heat."

Similar requirements for 30 day reports are identified in 10 CFR 50.73(a)(2)(ii)(B) or (v)(B).

References:

(1) VEPCO Deviation Report 87-1452, December 22, 1987

(2) VEPCO Design Change 84-43-3 "Service Water Reservoir Improvements, Final System Tie-In and Startup/North Anna Units 1&2"

(3) VEPCO Deviation Report 87-1405, December 8, 1987

FINDING IC-6: Inadequate Operator Training and Simulator Modeling. (Unresolved Item 89-200-05).

Discussion:

As a result of the team's review of Deviation Report 87-1405 associated with DC 84-43-3, a question arose concerning reactor operator training. Deviation Report 87-1405 identified the failure of a service water bypass motor-operated valve to close an initiation of a signal from the control room. VEPCO investigated the deviation and determined that the most probable cause of the valve failure to close was that the operator did not realize that the valves were throttleable and that the circuitry for the valve does 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).

During the inspection, the team determined that reactor operators had received training on the new service water reservoir spray and bypass system but that the training did not specifically cover the operation of the particular valves in question. In addition, when the simulator was changed several months later to reflect the service water system changes, the bypass valves were incorrectly modeled with seal-in versus throttleable circuitry. This item remains open pending VEPCO correcting the simulator bypass valve control switch modeling and verification that the reactor operators have been properly trained to recognize which switches have seal-in versus throttleable circuitry.

Regulatory Basis:

10 CFR Part 55.11 requires reactor operators be trained in the operation of all plant systems.

References:

(1) VEPCO Deviation Report 87-1405, December 8, 1987

FINDING IC-7: Inadequate Loop Accuracy Calculation for Charging Flow Instrument (Unresolved Item 89-200-06)

Discussion:

During the inspection, the team reviewed portions of Design Change 87-29-2 pertaining to the installation of a new charging flow differential pressure detector. The review concentrated on the environmental qualification of the

new detector and its effect on the overall instrument loop accuracy. During the review the team found that Calculation EE-0048 performed for determining the instrument loop accuracy, did not consider the effects of current leakage due the degradation of the cable insulation system in a postulated harsh environment. In a postulated harsh environment, cable insulation resistance decrease which increases leakage currents and cause a corresponding decrease in instrument accuracy. This was a repeat of violation 87-32-03 cited in NRC Inspection Report 50-338/87-32 and 50-339/87-32. During that inspection VEPCO was cited for failing to address the effects of characteristics such as leakage currents on total loop instrument accuracy calculations. In a response to this violation, submitted to the NRC on May 19, 1988, VEPCO committed to revising Engineering Standard STD-N-0025 by August 31, 1989, as necessary to preclude further violations. This standard had been changed; however, the changes made appeared to be inadequate for ensuring that loop accuracy calculations would be correctly performed when changes were made to environmentally qualified systems. Calculation EE-0048 was performed after the standard was changed.

As a result of this finding, VEPCO performed a new calculation which included the effects of the cable leakage currents. This calculation showed that the cable added an additional 0.3 percent to the previously assumed error of 5 percent. This item remains open pending VEPCO's review of other previous changes to the facility which may have affected instrumentation loops located in a harsh environment. Also, VEPCO is requested to issue an approved procedure for performing instrument setpoint calculations which would reduce the possibility of similar errors in future modifications.

Regulatory Basis:

10 CFR 50.49(d)(1) requires that the qualification file for electrical equipment important to safety specify the performance requirements under conditions existing during and following design basis accidents. 10 CFR 50.49(j)(2) requires that the qualification file for electrical equipment important to safety demonstrate that the equipment meets its specified performance requirements.

References:

(1) VEPCO Design Change 87-29-2, "CRDR Instrumentation Range Changes, North Anna Unit 2," November 3, 1988

(2) VEPCO Calculation EE-0048, "Instrument on Channel Accuracy for Charging and Letdown Flow," Revision 0, February 10, 1989

(3) NRC Inspection Report 50-338/87-32 and 50-339/87-32, November 25, 1987

(4) VEPCO Letter to NRC, Serial No. 88-230, May 19, 1988

(5) VEPCO Response I&C-002, "IR Loss in Cable," February 16, 1989

(6) VEPCO Standard STD-GN-0025, "Equipment Qualification Standard", Revision 6, Change 2, October 13, 1988.

FINDING IC-8: Inadequate Post Modification Test Program. (Unresolved Item 89-200-07).

Discussion:

Design Change 87-12 for installation of the ATWS mitigation system interfaced with various Class 1E systems such as the auxiliary feedwater system, reactor protection system, containment isolation portion of the steam generator sampling system, steam generator level monitoring system and nonClass 1E systems such as steam generator blow-down system and turbine trip system. During installation of this design change, wiring changes included providing interlocks and permissives between ATWS output relay contacts and initiating circuits for the above Class 1E and nonClass 1E systems. The team reviewed engineering requirements for the post-modification testing and noted that the scope of this testing was limited only to newly installed hardware. Since the modification involved disconnecting and/or reconnecting various relay contacts, limit switch contacts of Class 1E MOVs, and output contacts of various Class 1E instruments, the team was concerned that during installation, a potential existed for erroneous alteration of wiring or terminations which could be in the vicinity but not related to this modification. This could lead to a situation where the modified circuit(s) might operate properly for objectives of the modification but could have been disabled for other safety functions. Therefore, circuits of the affected systems should be verified by post-modification testing to ensure that the pre-modification capacity of the Class 1E system to mitigate an accident has not been comprised due to inadvertent error during installation of the modification.

Additionally, the team identified two examples where the post-modification testing was performed prior to the installation of the ATWS modification. Test Procedure 2-PT-57-4 "Safety Injection Functional Test" included auxiliary feedwater pump actuation on a safety injection signal and steam generator blowdown valves closing on a safety injection signal. Both of these tests were done prior to the ATWS modification installation. Therefore, this item remains open pending confirmation from VEPCO that:

(1) The post-modification testing requirements had been included in all the design change packages/engineering work requests installed during this outage and the required testing was implemented subsequent to modification installation.

(2) The design change process and engineering work request procedure have been updated to require inclusion of post-modification testing requirements from project engineering and implementation by the site test group.

(3) The two aforementioned safety injection tests were completed subsequent to the ATWS modification.

Regulatory Basis:

Appendix B, Criterion XI of Appendix B requires that a test program be established to assure that all testing required to demonstrate that structures, systems and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

In this regard ANSI N18.7-1976, Section 5.2.19, item (4), states; "The test program shall cover all required testing including tests during design,

fabrication and construction activities associated with maintenance and modifications during operational phase and the demonstration of satisfactory performance following plant maintenance and modifications or procedural changes."

References:

(1) DCP 87-12 ATWS modification package

(2) Periodic Test Procedure 2-PT-57-4 "Safety Injection Functional Test

FINDING MS-1: Contains Safeguards Information, Transmitted Under Separate Cover Letter (Unresolved Item 89-200-08).

FINDING MS-2: Confirmation of Leak Detection Capability for Leak-Before-Break Analysis. (Unresolved Item 89-200-09).

Discussion:

In DC 86-10-2, VEPCO had taken advantage of the relaxation of GDC-4 to eliminate several of the pipe snubbers on the primary coolant piping. As required, a leak-before-break analysis was performed on the affected piping. In the submittal to the NRC in support of this modification, a commitment was made to detect a leakage rate of 1-gallon per minute leak in 4 hours. The team requested details concerning the design and operation of the equipment upon which this claim was based. Although some information was provided concerning the operation of the equipment, no definitive information was provided concerning the design of the equipment. The information provided about the design seemed to be in conflict with the operational information. At the conclusion of the inspection, no definitive, nonconflicting information had been provided. The team was surprised that the design engineering organization was unable to provide this information over a period of approximately one and one-half weeks. This, along with observations in other sections of this report, would appear to indicate a weakness in the instrumentation and controls area of the design organization.

This item remains open pending a detailed description of how the aforementioned leakage rate is detected.

Regulatory Basis:

10 CFR 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Bases," requires that structures, systems, and components important to safety be protected from the dynamic effects of a loss-of-coolant accident. In 1987, this requirement was relaxed to allow elimination of certain postulated high energy line breaks and, also eliminate the snubbers originally designed to restrain the piping if the leak-before-break criteria of NUREG-1061, Volume 3 were satisfied. One requirement of such an analysis is the ability to detect a 1-gallon per minute leak in 4 hours.

ANSI N45.2.11-1974, Section 8.2, states that the designated organization shall have demonstrated competence in the specific design area of interest and have an adequate understanding of the requirements and intent of the original design.

References:

(1) DC-86-10-2 "Large Bore Snubber Leak-Before-Break Modifications, North Anna Unit 2," December 22, 1988

(2) NUREG-1061, Volume 3, "Evaluation of Potential for Pipe Breaks"

FINDING MS-3: Inadequate 10 CFR 50.59 Safety Evaluation (Unresolved Item 89-200-10).

Discussion:

The team found several instances where it did not appear that the requirements for performance of 10 CFR 50.59 safety evaluations were well understood. EWR 86-695 removed a portion of a block wall around the iodine filter unit in the containment ventilation system to facilitate changing of the filters. It also installed bolt-in-place dams at the resulting opening and at the original access to the room to prevent potential flooding of the filters. The team found that the 10 CFR 50.59 safety evaluation for this modification did not address significant technical considerations. Block walls are generally incorporated in nuclear plant design to provide radiation shielding for personnel and/or equipment. They may also perform other functions, such as support for other structures or equipment, or credit may be taken for them in the plant's high energy line break analyses. Therefore, removal of these walls has the potential to have other safety implications. In addition, it may affect other nonsafety-related, yet important, design considerations such as ALARA. The safety evaluation for EWR 86-695 was deficient in that it did not address any of the effects of removal of this wall with respect to the possible originally intended function(s) of the wall itself. It only addressed how removal of the wall would not affect the function of the iodine filters.

EWRs 89-036 and 89-036B added a diesel-driven air compressor and an air dryer, respectively, to the instrumentation system. At several locations in both EWRs, statements were made that "since this EWR does not modify any safety-related equipment,...no unreviewed safety question is created." Whether or not safety-related equipment was modified is not the only criterion in the unreviewed safety question determination since, it does not consider interactions of safety and nonsafety-related systems. It therefore appeared that an incorrect criterion was used.

This item remains open pending VEPCO issuing augmented procedural guidance with regard to safety evaluations.

Regulatory Basis:

10 CFR 50.59, "Changes, Tests, and Experiments," allows the licensee to make changes in the facility and procedures as described in the FSAR and to conduct tests and experiments not described in the FSAR without NRC approval if the changes, tests, or experiments not described in the FSAR do not involve a change in the technical specifications or an unreviewed safety question.

ANSI N45.2.11-1974: Section 8 states that documented procedures shall be provided for design changes to approved designs, including field changes, which assure that the impact of the change is carefully considered.

References:

(1) North Anna Administrative Procedure ADM-3.9, Safety Evaluation 10 CFR 50.59 Review, October 11, 1988

(2) EWR 86-695, Modification of Emergency Iodine Units, Revision NA, March 20, 1987.

(3) EWR 89-036, Installation of Backup Air Compressor, Revision NA, January 24, 1989.

(4) EWR 89-036B, Installation of Instrument Air Dryer, Draft, February 14, 1989.

FINDING MS-4: Inadequate Identification of Calculation Inputs (Closed).

Discussion:

The team reviewed two calculations associated with DC 87-026-3, Steam Generator Downcomer Flow Resistance Plates Installation, in which sources were not given for several of the inputs and formulae used. In every other respect, the calculations were excellent. However, since recourse to the originator was required to fully understand the analyses and verify the adequacy of the result, they did not meet ANSI N45.2.11 Section 4.2 requirements. This item is provided for information purposes only, and is considered closed.

Regulatory Basis:

10 CFR 50, Appendix B, Criterion III, Design Control, requires that measures shall be established to assure that design bases are correctly translated into specifications, drawings, procedures, and instructions.

ANSI N45.2.11-1974, Section 4.2, Design Analyses, requires that "analyses shall be sufficiently detailed as to purpose, method, assumptions, design input, references and units such that a person technically qualified in the subject can review and understand the analyses and verify the adequacy of the results without recourse to the originator."

Peferences:

(1) VEPCO Calculation SM-553, Tube Uncovery Time for North Anna Steam Generator Tubes Covered During a Steam Generator Tube Rupture, Revision 0, dated September 21, 1987.

(2) VEPCO Calculation SM-554, Determination of Power Level for Keeping Steam Generator Tubes Covered During a Steam Generator Tube Rupture, Revision 0, dated September 27, 1987.

(3) DC 87-026-3, Steam Generator Downcomer Flow Resistance Plates Installation.

FINDING EP-1: Protective Devices on Safety Class Buses Not Coordinated. (Unresolved Item 89-200-11).

Discussion:

The team reviewed DC 83-24, "Appendix R Emergency Diesel Generator." This design change was initiated to comply with NRC's IE Information Notice 85-09, which stated that a fire in the main control room could adversely affect the cables routed to the emergency diesel generator and 4160-V generator circuit breaker control circuits. This could result in loss of the control circuit fuses.

The design change added redundant fusing and a means of transferring between the normal and emergency fuses. The team agreed that the addition of these fuses would eliminate the problem noted in the information notice. As part of this review, the team observed that the Electrical Distribution System Coordination Study (Appendix R, Reanalysis Chapter 9, prepared by VEPCO in 1986) showed that the 4160-V vital bus feeder breaker was not coordinated with its downstream 480-V load center feeder breakers; the 4160-volt breaker supplies power to two load centers. The report also stated that coordination cannot be obtained in some cases between the 4160-V switchgear supply breakers and the downstream 480-V load center supply breakers. Since separate studies had shown that either unit at North Anna could safely shut down utilizing the opposite unit's 4160-V and 480-V power sources through the use of mechanical cross-connects on the charging and component cooling water systems, the requirements of Appendix R are not violated in this case. However, the team was concerned that a single fault at one of the 480-V buses could cause the 4160-V feeder breaker to trip, causing the loss of both 480-V load centers, which constitutes the loss of one division. The team recognized that this lack of breaker coordination was not a violation of the single failure criterion since the other division would still be available, but was indicative of poor design practices. This concern was applicable for all the safety class buses. This item remains open pending VEPCO's review of the protective device coordination to determine if the relays can be reset to provide adequate coordination.

Regulatory Basis:

This condition was not in accordance with the design philosophy of IEEE 308, Standard Criteria for Class 1E Power Systems, Section 6.2, Alternating Current Power Systems to which VEPCO was committed according to Chapter 8 of the update FSAR. Item Number 6 of IEEE 308, Section 6.2 states that protective devices should be provided to limit the degradation of Class 1E power systems.

References:

(1) VEPCO's "Appendix R, Reanalysis, Chapter 9," 1986

(2) DC 83-24, Appendix R, Emergency Diesel Generator

(3) NRC IE Information Notice 85-09

FINDING EP-2: Inadequate Design Evaluation and Safety Evaluation of Quality Control Inspection Report. (Unresolved Item 89-200-12).

Discussion:

The inspection team reviewed DC 85-30-2 which was associated with replacement of the station batteries during the previous outage. Also included in the design change package was Quality Control Inspection Report (QCIR) IR-N-86-281A dated March 21, 1986, which identified that a nonseismic 1¼-inch conduit was routed directly above the Class 1E battery located in Battery Room 2-III. The disposition of the QCIR was use-as-is with the justification being that only one of the four channels would be affected.

The inspection team noted that this QCIR disposition as written was a violation of Regulatory Guide 1.29 which require protection of safety-related equipment from unacceptable interaction with nonseismic items. In response to the inspection team's concern, VEPCO performed Seismic Analysis SEO-1064 Revision 0, which demonstrated that the conduit supports were structurally adequate to withstand the design-basis seismic event. VEPCO further explained that the engineer who dispositioned the QCIR was capable of performing the required seismic analysis but had elected to offer the aforementioned inappropriate system-based disposition.

As a result of this review, the inspection team had three concerns. First, as stated previously, a quantitative analysis was not performed which demonstrated that the conduit was designed to withstand the design-basis seismic event. Second, the 10 CFR 50.59 review did not identify the invalid disposition on the QCIR due to either a procedural breakdown or inadequate review. Third, an ineffective and/or nonexistent design verification was performed for the QCIR disposition within the design organization.

This item remains open pending VEPCO's review of a sample (minimum 10) of QCIR's performed in conjunction with design changes to ensure an adequate and substantiated disposition exists.

Regulatory Basis:

The arrangement of an unanalyzed component over seismic Category I equipment violates the seismic design requirement outlined in Regulatory Guide 1.29, Section C, paragraph 2. This paragraph states that those portions of structures, systems, or components whose continued function is not required but whose failure could reduce the functioning of a seismic Category I feature should be designed and constructed so that the safe shutdown earthquake would not cause such failure.

References:

(1) Analysis of North Anna 1, Stationary Battery Replacement, E-1, Rev. 3, February 5, 1986

(2) Calculation EE-009, 125-Vdc System Analysis, Revision 0, February 28, 1989

(3) Quality Control Inspection Report IR-N-86-281A, March 21, 1986

(4) DC 85-30-2, Replacement of Station Batteries

(5) Seismic Analysis SEO-1064, Revision 0

FINDING MC-1: Oversized Holes in Baseplate Accepted Without Proper Justification. (Unresolved Item 89-200-13).

Discussion:

The inspection team reviewed EWR 87-671 which involved removing pipe support 2-H55-WGCB-3B for accessibility to perform maintenance work on a valve. The support could not be reinstalled over the existing 1-inch diameter anchor bolts without damaging the threads, since the bolts were installed at 4°-6° angularity. Therefore, a field change request (FCR) was initiated to enlarge three out of the eight holes from 1-1/8-inch diameter to 1-1/4-inch diameter. The disposition of the FCR accepted this enlargement claiming that the associated pipe support calculation (SWEC calculation 12050-Z-1020, Rev. 0) had been reviewed.

The inspection team also reviewed the pipe support calculation and identified two concerns. First, the existing bolt interaction was at a maximum value of 1.0 (actual 1.04), considering all eight bolts were in shear. Second, the impact of baseplate flexibility on the anchor bolts was not included in the original calculation. Therefore, the disposition of the FCR was clearly inappropriate because it was solely based on the existing calculation and a new analysis needed to be performed prior to dispositioning.

After the inspection team's identification of this finding, a calculation was performed by VEPCO which utilized the GT STRUDL computer program to distribute the support loads properly through the frame. The baseplate was analyzed using the computer program BASEPLATE II to consider the appropriate baseplate flexibility. The calculation also considered higher anchor bolt allowable loads, due to higher concrete strength and longer embedment lengths which were as-built verified. The three anchor bolts with the oversized holes were excluded from the shear resistance of the anchor bolt qualification. Based on this analysis, the new anchor bolt interaction ratio was calculated to an acceptable value of 0.76. The team accepted this quantification method and recommended that it be added to the EWR.

This finding remains open pending VEPCO's review of 10 randomly sampled pipe support field change requests to ensure similar problems with design verification is not pervasive. Also, VEPCO needs to explain what changes have or need to be made to ensure that an adequate design verification is performed for future modifications.

Regulatory Basis:

ANSI N45.2.11-1974, Section 8, states that design changes, including field changes shall be justified and subjected to design control measures commensurate with those applied to the original design.

ANSI N45.2.11-1974, Section 6.1 states that measures shall be applied to verify the adequacy of design.

References:

(1) SWEC Calculation 12050-Z-1020, Revision 0, April 28, 1977

(2) VEPCO Calculation CE-0589, Revision 0, March 1, 1989

(3) Nuclear Standard STD-CEN-0024, Revision 1, December 9, 1987

FINDING CS-1: Lateness in Updating UFSAR. (Unresolved Item 89-200-14).

Discussion:

DC 84-43-3 summarized the UFSAR changes required by the service water system improvement. The changes were not only from DC 84-43-3, but from DC 84-3-3, DC 84-37, DC 84-35-3, etc. The UFSAR changes just appeared in DC 84-43-3, Rev. 9, dated July 10, 1986, revised by DC 84-43-3, Rev. 51, dated June 4, 1987. The following were examples of some of the UFSAR changes:

(1) Codes and Standards changes: (UFSAR Section 3.8.1.2.1)

i. ACI-318-83
ii. AISC 8th Edition
iii. ACI-301-84

(2) Mechanical Splices - Use of Dywidag Threaded Rebar Splices (USFAR Section 3.8.1.7.3)

(3) Added grade 60 rebar which was used for the service water system improvement (USFAR Section 3.8.1.7.2)

(4) Portable water from Orange, VA was used for concrete mixing (USFAR Section 3.8.1.7.1.3)

None of the FSAR updates listed in DC 84-43-3 were implemented at the time of the inspection. Since the system was put into service in 1987, it appeared that the licensee was late in updating the UFSAR.

Regulatory Basis:

10 CFR 50.71 Maintenance of records, making of reports, Section (e)(4) requires that "...revisions (to UFSAR) shall be filed no less frequently than annually and shall reflect all changes up to a maximum of 6 months prior to the date of filing."

References:

(1) DC 84-43-3, "Service Water Reservoirs Improvements, Final System Tie-In and Startup," Revision 9, July 10, 1986 and June 4, 1987.

(2) DC 84-35-3, "Service Water Reservoir Improvements, Valve House Structural Analysis and Design," Rev. 34, January 13, 1987.