VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

W. L. STEWART
SENIOR VICE PRESIDENT
POWER

August 25, 1989

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Serial No. 89-485
NOS:jmj
Docket Nos. 50-339
License Nos. NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION UNIT 2
SAFETY SYSTEM OUTAGE MODIFICATION INSPECTION
(DESIGN REPORT) 50-339/89-200

Virginia Electric and Power Company has received the report of the Safety System Outage Modification Inspection (SSOMI) which conveyed the results and conclusions of the design portion of the inspection for North Anna Unit 2. The report documented 18 issues which required addressing. Our responses to these issues (except for IC-4 and MS-4 which are closed and MS-1 which is Safeguards Material and is being submitted under separate cover) are enclosed in Attachment 1. The supporting engineering studies are provided in Attachment 2.

If you have further questions, please contact us.

Very truly yours,

W. L. STEWART

Attachments:

IE01
1/1

cc: U. S. Nuclear Regulatory Commission
101 Marietta Street, N.W.
Suite 2900
Atlanta, GA 30323

Mr. J. L. Caldwell
NRC Senior Resident Inspector
North Anna Power Station

ATTACHMENT 1

RESPONSE TO SSOMI REPORT

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION UNIT 2
SAFETY SYSTEM OUTAGE MODIFICATION INSPECTION
(DESIGN REPORT) 50-339/89-200

FINDING IC-1

INCORRECT DIFFERENTIAL PRESSURE USED IN SIZING SERVICE WATER RESERVOIR SPRAY AND BYPASS SYSTEM ISOLATION VALVES. (UNRESOLVED ITEM 89-200-01)

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.

RESPONSE TO IC-1

In response to NRC questions about design changes that required a change in safety related motor operated valves (MOVs) , VEPCO submitted reports on April 28, 1989 and June 30, 1989. These reports documented the adequacy of the Unit 2 and 1 (respectively) MOVs which have replaced under Engineering Work Requests (EWRs) and Design Change Packages (DCPs). Significant modifications (i.e., those that were performed in lieu of a changeout) were also included in this review.

Based on our experiences in the MOV upgrade program at Surry and the MOV Program underway in response to Generic Letter 89-10, "Safety Related Motor Operated Valve Testing and Surveillance", we have now expanded the North Anna review to include EWRs which were generated to support maintenance or minor modifications which could impact valve performance. This expanded review will be completed November 30, 1989.

FINDING IC-2

SETPOINT CALCULATION OMISSION AND LACK OF AN APPROVED PROGRAM FOR PERFORMING SETPOINT CALCULATIONS. (UNRESOLVED ITEM 89-200-02)

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.

RESPONSE TO IC-2

The investigation of the Westinghouse methodology for performing instrument uncertainties and the Instrument Society of America (ISA) standards and recommended practices for developing methodology and control of documents for setpoints, resulted in the development and issuance of two standards. STD-EEN-0304, "Calculating Instrumentation Uncertainties By the Square Root of the Sum of the Squares", was developed to provide a consistent, conservative method to address the error associated with setpoints. STD-GN-0030. "Revising Nuclear Plant Setpoints", was developed to identify the requirements for calculating, documenting and reviewing changes to setpoints.

This is discussed in more detail by Technical Report EE-0019 (attached).

With this submittal, this item is considered closed.

FINDING IC-3

NON CLASS 1E LOADS CONNECTED TO CLASS 1E BUSES WITHOUT PROPER ISOLATION.. (UNRESOLVED ITEM 89-200-03)

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 non-Class 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 response 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 practice needs to be done for the entire North Anna facility independent of modification 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.

RESPONSE TO IC-3

The NRC staff and VEPCO will meet on September 7, 1989, to resolve this item.

FINDING IC-5

FAILURE TO REPORT UNDERSIZED SERVICE WATER RESERVOIR BYPASS ISOLATION VALVE MOTOR ACTUATORS. (UNRESOLVED ITEM 89-200-04)

As part of its review of DC 84-43-3, the team reviewed Deviation Reports 87-1405 and 87-1452 which identified problems with the 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 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 valve, 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.

RESPONSE TO IC-5

The basis for not reporting either Deviation Report 87-1405 or 87-1452 was discussed in a conference call with the SSOMI design team on March 20, 1989. The NRC was told that the Service Water (SW) system was not outside of its design basis. Specifically, when the new SW Reservoir system was made operable in the spring of 1987, the SW Reservoir bypass MOVs were placed in the closed position and de-energized. The MOVs were de-energized in the closed position because the post modification operability test for the new valves could not be performed with the units operating. The post modification operability test required a Safety Injection (SI) signal to be initiated since the bypass MOVs are required to close on a SI signal.

The SW Reservoir is designed to ensure that sufficient cooling capability is available to either provide normal cooldown of the facility or to mitigate the consequences of an accident. The SW bypass MOVs are used to isolate the SW spray arrays during the cooler months in order to maintain the temperature of the SW Reservoir above freezing. During the warmer months, with both units operating at full power, the SW spray arrays are used to maintain the temperature of the SW Reservoir below the Technical Specification limit of 95 degrees F.

During the summer of 1987, simultaneous full power operation of both units did not occur due to forced and scheduled refueling outages. Following satisfactory completion of the post modification operability test during each units refueling the the SW bypass MOVs were opened.

Simultaneous full power operation of both units resumed in late November 1987 with the SW bypass MOVs opened or throttled. Based on previously performed engineering calculations, the SW heat sink requirements could be met even at this time even with full bypass flow in the SW Reservoir. The previously performed engineering calculations confirmed that the SW system could accommodate the design basis heat load for operation between the summer of 1987 and the submittal of Deviation Report 87-1452 on December 22, 1987 which resulted in closing and . the SW bypass MOVs.

A copy of this response will be attached to Deviation Reports 87-1405 and 87-1452.

With this submittal, this item is considered closed.

FINDING IC-6

INADEQUATE OPERATOR -TRAINING AND SIMULATOR MODELING.
(UNRESOLVED ITEM 89-200-05)

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 a failure of a service water bypass motor operated valve to close (on) an initiation 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 have 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.

RESPONSE TO IC-6

The simulator valve control models were corrected by Simulator Modification Report (SMR) 8902280900 on Feb. 28, 1989, to reflect the throttleable circuitry installed in the plant.

The operator training on this subject was completed on April 21, 1989. Special Licensed Operator Requalification Program presentations were conducted on April 13, 17, 18, and 20, 1989.

With this submittal, this item is considered closed

FINDING IC-7

INADEQUATE LOOP ACCURACY CALCULATION FOR CHARGING FLOW INSTRUMENT (UNRESOLVED ITEM 89-200-06)

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 to the degradation of the cable insulation system in a postulated harsh environment. ..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.

RESPONSE TO IC-7

Engineering standards have been developed that provide a methodology for calculating, documenting and reviewing changes to setpoints.

Setpoint calculations, which include cable leakage current, were performed on loops that function in a harsh environment and found to be acceptable.

This is discussed in more detail by Technical Report EE-0020 (attached).

With this submittal, this item is considered closed.

FINDING IC-8

INADEQUATE POST MODIFICATION TEST PROGRAM. (UNRESOLVED ITEM 89-200-07)

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 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 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.

RESPONSE TO IC-8

(1) Testing has been performed, reviewed and documented for DCPs and EWRs implemented during the 1989 refueling outages, as detailed in the enclosed Technical Report PE-0012.

(2) The Design Change process and the Engineering Work Request procedure have been modified to conform to VEPCO's engineering standards and will specify the post-modification testing to be performed.

(3) The aforementioned safety injection tests were not performed subsequent to the ATWS modification. However, post-modification testing was performed and documented to assure that the unmodified safety functions, which interface with the ATWS, were not adversely affected by the ATWS modification and will continue to perform their intended functions.

This is discussed in more detail by Technical Report PE-0012 (attached).

With this submittal, this item is considered closed.

FINDING MS-2

CONFIRMATION OF LEAK DETECTION CAPABILITY FOR LEAK-BEFORE-BREAK ANALYSIS. (UNRESOLVED ITEM 89-200-09)

In DC 86-10-2, VEPCO has 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.

RESPONSE TO MS-2

Indications of leakage are sensed by various systems, which provide detection of: identified, unidentified and intersystem leakages. The radiation monitoring system and the frequency and duration of the operation of the containment sump pumps provide qualitative indications of increasing RCS leakage. The determination of this increase in leakage will direct the performance of the appropriate procedures to determine the quantity and sources of the leakage. The accuracy of the RCS leakrate test, which can be performed in one hour, is better than 1 gpm.

This is discussed in more detail by the attached Stone and Webster report REACTOR COOLANT SYSTEM LEAKAGE DETECTION SYSTEMS NORTH ANNA UNITS 1 & 2

With this submittal, this item is considered closed.

FINDING MS-3

INADEQUATE 10 CFR 50.59 SAFETY EVALUATION. (UNRESOLVED ITEM 89-200-10)

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.

EWR 89-036 and EWR 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 regards to safety evaluations.

RESPONSE TO MS-3

Currently, there are several procedures in use for performing 10 CFR 50.59 safety evaluations. Although these procedures accomplish the same ends, the ones in use at North Anna and Surry are not identical and they differ from those used by engineering in the Design Change and Engineering Work Request processes.

VEPCO is now developing a common procedure to be used for the various functions at the various locations. The uniform procedure will enhance consistency in the safety evaluations and reduce the probability of incorrect evaluations being performed.

This procedure will be completed and in place by October 31, 1989.

FINDING EP-1

PROTECTIVE DEVICES ON SAFETY CLASS BUSES NOT COORDINATED. (UNRESOLVED ITEM 89-200-11)

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.

RESPONSE TO EP-1

The Appendix R review of the North Anna electrical systems required a thorough coordination study of all Appendix R equipment and power supplies to ensure that a fire-induced fault of one circuit in another fire zone would not result in the complete loss of an associated bus. It was noted, at that time, that coordination did not exist in all cases between the 480 volt buses and the 4160 volt breakers feeding them. However, this is acceptable since only fire-induced faults could cause this problem and the 480 volt buses and their 4160 volt feeder breakers are located in the same fire zone.

The NRC identified this issue again during the SSOMI and confirmed that it is neither an Appendix R violation nor a safety concern, but may be indicative of poor design.

This lack of coordination could result in the loss of both associated 480 volt buses only in a limited band of bus faults and, bus faults are rare. Although we consider the condition to be safe, though perhaps undesirable, and the NRC inspectors agreed with this assessment, VEPCO has initiated a further study of the issue.

This is discussed in more detail by Technical Report EE-0021 (attached).

FINDING EP-2

INADEQUATE DESIGN EVALUATION AND SAFETY EVALUATION OF QUALITY CONTROL INSPECTION REPORT. (UNRESOLVED ITEM 89-200-12)

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 non-seismic 1 1/2 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.

RESPONSE TO EP-2

VEPCO sampled 15 quality control reports. Twelve of these were found to be adequately substantiated during the initial examination. The remaining three required additional research to assess their dispositions.

These 3 were reviewed more closely because they lacked adequate reference to a technical basis at the time of the disposition. Subsequently, they were confirmed to be adequately substantiated and there was no evidence that the design review process was compromised.

This is discussed in more detail by Technical Report CE-0020(attached).

With this submittal, this item is considered closed.

FINDING MC-1

OVERSIZED HOLES IN BASEPLATE ACCEPTED WITHOUT PROPER JUSTIFICATION
(UNRESOLVED ITEM 89-200-13)

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 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.

RESPONSE TO MC-1

A review of 10 randomly selected field changes was conducted and the adequacy of the design justification was determined to be acceptable. In some cases, the design justification was described in the body of the field change while in the others, the evaluations were either recorded in the calculations or could be reasoned out with a little effort. In the ten cases examined, the plant remained consistent with the licensing basis following implementation of the pipe support design changes and subsequent field changes.

In order to enhance the field change system, NDCM procedure No. 3.7, Calculations, will be revised to require documentation of the rationale behind any 'engineering judgement' evaluations of design justifications. This revision will be completed by September 30, 1989.

This is discussed in more detail by Technical Report CE-0017 (attached).

FINDING CS-1

LATENESS IN UPDATING UFSAR. (UNRESOLVED ITEM 89-200-14)

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 (UFSAR Section 3.8.1.7.3)

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

(4) Portable water from Orange, VA was used for concrete mixing (UFSAR 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.

RESPONSE TO CS-1

The lateness in the updating the UFSAR is being resolved as follows:

1.0 A new standard has been written and is in review which more clearly defines the requirements for UFSAR change request package preparation and processing. In particular, the requirements for UFSAR change request package content and the definition of when the review of a change package is to be initiated have been strengthened. This new standard will be in place by 12/31/89.

2.0 Additional resources have been dedicated, on an interim basis, to the review and processing of the backlogged UFSAR change packages. Once the backlog of change request packages has been incorporated into the UFSARs, an evaluation will be performed to determine the permanent resource requirements.

3.0 As a result of the Surry SSFI, the sensitivity by Virginia Power has been increased in regard to control and maintenance of design basis information. The UFSARs are a summary of the design bases for each station and therefore the priority given to the preparation and review of UFSAR change request packages has increased. This will shorten the review time of UFSAR changes by the various Virginia Power organizations.

ATTACHMENT 2

ENGINEERING REPORTS FOR
SAFETY SYSTEMS OUTAGE MODIFICATION INSPECTION

VIRGINIA ELECTRIC AND POWER COMPANY