U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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Report Nos: 50-338/96-12, 50-339/96-12

Licensee: Virginia Electric and Power Company (VEPCO)

Facility: North Anna Power Station, Units 1 & 2

Location: 1022 Haley Drive
Mineral, Virginia 23117

Dates: November 3 through December 7, 1996

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ENCLOSURE 2

EXECUTIVE SUMMARY

North Anna Power Station, Units 1 & 2
NRC Inspection Report Nos. 50-338/96-12, 50-339/96-12

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a five week period of resident inspection. In addition, it includes the results of announced inspections by regional specialists and regional projects inspectors.

Operations

- Daily operations were generally conducted in accordance with regulatory requirements and plant procedures (Section O1.1).
- Safety system and operator response to a Unit 2 reactor trip was good. Minor equipment problems were promptly corrected. The trip's cause, a main generator failure due to damage caused by foreign material, required an extended unit shutdown for repairs (Section O1.2).
- Initial reviews of cold weather protection procedures found that activities were properly completed (Section O2.1).
- One NRC notification required by 10 CFR 50.72 was properly made by the licensee (Section O2.2).
- An Operator Work Around (OWA) meeting was an effective mechanism for keeping plant management informed of the status of each open OWA (Section O6.1).
- The Oversight organization continued to assess station performance effectively. A Management Safety Review Committee meeting complied with TS requirements, and substantive assessment issues were addressed in committee discussions (Section O7).

Maintenance

- Service Water Restoration Project work activities were well controlled. Supporting system manipulations were performed in accordance with regulatory requirements and commitments (Section M1.1).
- Debris was observed on the ledges above the radiator fans for three of the four emergency diesel generators (Section M2.1).
- A Non-Cited Violation (NCV) was identified for a failure to meet TS surveillance requirements for testing reactor trip bypass breakers. Two previous violations and three Licensee Event Reports (LERs) were closed (Section M8).

Engineering

- A violation was identified for a failure to ensure proper facility design control. Unit 2 safeguards area walls were found to not meet their design basis for controlling safeguards pump leakage since early in plant life. One Unresolved Item was closed (Section O8.1).

- The licensee's operability evaluation for non-conforming bolts in a recently installed pump was reasonable, and completed in a timely manner. The scheduled date for correcting the non-conformance was acceptable. Even though the problem was primarily caused by the manufacturer, problems in the licensee's design control program contributed to the wrong bolts being installed. For this reason a Notice of Violation was issued (Section E1.1).

- The licensee appropriately analyzed the impact of service water system restoration activities on a shutdown unit and made conservative decisions concerning shutdown unit status (Section E1.2).

- An NCV was identified for a failure to meet American Society of Mechanical Engineers Code Section XI requirements for testing the check valve functions of six steam generator decay heat removal isolation valves (Section E1.3).

- The Minor Modification Review Team meeting was professionally conducted. The inspectors had no concerns involving the disposition of the items (Section E1.4).

- The licensee's approach to resolving a potential problem with microbiological influenced corrosion in stainless steel service water piping was reviewed. One section of four-inch diameter pipe was scheduled for replacement based on findings to date. The licensee was at the investigation stage, and the total corrective action plan was under development. No concerns were identified by the inspectors (Section E2.1).

- The engineering design change package was well prepared for implementing a carbon steel piping replacement in the service water system and the quality of work being performed by the craftsmen was good (Section E2.2).

- A detailed cell inspection was performed on each cell of all the safety-related batteries. The inspectors concluded that battery maintenance was good. However, the inspectors observed significant sedimentation in two cells. The inspectors concluded that the sedimentation had been present on June 4, the time of the last surveillance inspection. Therefore, the fact that the surveillance did not record any problems with sedimentation was considered a weakness in the implementation of the inspection procedure (Section E2.3).

- A review of six Deviation Reports and one Potential Problem Report related to mechanical and structural engineering indicated good support for facilities and equipment by engineers (Section E2.4).

- The alarm response procedure for bus overvoltage was a workable procedure, and was consistent with statements made in letters to the NRC on the subject of overvoltages. The safety evaluation supporting the change in maximum allowable response time for performing tap changer manipulations was accurate and complete.

  Some corrective actions were proposed by the licensee to address NRC comments with regard to reading voltage at the 480 volt bus, but this did not represent any programmatic weakness. The overvoltage relays with alarms were a conservative design in relation to standard industry practice (Section E2.5).

- The inspectors reviewed a Deviation Report involving the failure of safety-related vital battery 1-I to meet one acceptance criterion during performance of the last service test. The inspectors agreed that the subsequent Operability Evaluation was reasonable, and that the requirement of the Technical Specification was met. An Inspection Follow-up Item was established to ensure review of certain procedure enhancements and future test results (Section E2.6).

- Review of a 1996 modification package to reinforce the supports for the component cooling water surge tank to resolve concerns about the seismic integrity of the tank indicated that the design was correctly accomplished, but that the actual installation did not meet specified dimensional tolerances. Also, a pipe support was installed at a point where no support was shown on the drawings. A Notice of Violation was issued (Section E3.1).

- The licensee's test procedure for the molded-case circuit breakers was good, in that, it checked that the magnetic element tripped within a specified band around the set point. The controls for establishing and verifying the set points for the magnetic elements were minimal. There was no set point document, drawing, or specific instructions to help ensure proper set points were maintained. Based on a sample of two newer circuit breakers, the licensee's informal method resulted in correct settings. Nevertheless, the lack of formal controls on the set point for the magnetic element of molded-case circuit breakers was a weakness in the licensee's design control program in that continued use of the informal method has increased probability to result in incorrect set points. Whether the lack of formal controls represents a violation of NRC requirements in the area of design control or procedures is under further review by the NRC. The issue is identified as an Unresolved Item concerning the control of set points for molded-case circuit breakers (Section E3.2).

- The safety and oversight committees were conducting their duties in an efficient and productive manner (Section E7.1).

- The engineering self-assessment program was only recently defined, and very few self assessments were completed (Section E7.2).
- An Inspection Followup Item was opened to review functional tests that will be conducted on either new or old snubbers in the upcoming Unit 1 refueling outage. One Unresolved Item was closed (Section E8.1).

Plant Support

- The licensee was closely monitoring collective and individual radiation dose exposure and meeting established As Low As Reasonably Achievable goals and occupational dose limits (Section R1.1).
- The licensee's water chemistry control program for monitoring primary and secondary water quality had been implemented in accordance with the TS requirements and the Electric Power Research Institute guidelines for pressurized water reactor water chemistry (Section R1.2).
- One non-cited violation was identified by the licensee for failure to comply with the conditions of the Certificate of Compliance for an NRC-approved shipping package (Section R1.3).
- Two previous violations were closed (Section R8).
- Security systems were in good working order and security manning was appropriate (Section S1).

Report Details

Summary of Plant Status

Unit 1 began the inspection period at full power and operated the entire inspection period at or near full power.

Unit 2 began the inspection period at full power and operated at or near full power until November 12. On that date, the unit tripped from full power due to a main generator fault. The unit was cooled down and entered hot shutdown on November 13, and cold shutdown on November 29. At the inspection period's end, the unit remained shutdown for main generator repairs.

I. Operations

O1 Conduct of Operations

O1.1 Daily Plant Status Reviews (71707)

The inspectors conducted frequent control room tours to verify proper staffing, operator attentiveness, and adherence to approved procedures. The inspectors attended daily plant status meetings to maintain awareness of overall facility operations and reviewed operator logs to verify operational safety and compliance with Technical Specifications (TSs). Instrumentation and safety system lineups were periodically reviewed from control room indications to assess operability. Frequent plant tours were conducted to observe equipment status and housekeeping. Deviations Reports (DRs) were reviewed to ensure that potential safety concerns were properly reported and resolved. The inspectors found that daily operations were generally conducted in accordance with regulatory requirements and plant procedures.

O1.2 Unit 2 Reactor Trip

a. Inspection Scope (71707, 93702)

On November 12, the inspectors in the field observed plant equipment responding to a Unit 2 reactor trip from full power. The inspectors proceeded to the control room and observed operations immediately following the trip. Additionally, the inspectors observed immediate post-trip equipment conditions in the turbine building, switchgear rooms, Auxiliary Feedwater (AFW) pump house, and Main Steam Valve House (MSVH). The inspectors attended the licensee's post-trip review and reviewed trip data to independently verify that safety systems and operator performance were as expected throughout the event.

b. Observations and Findings

The inspectors found that the reactor tripped from full power following a main generator trip. The main generator trip was caused by an actuation of the generator protective system. A review of trip data found that the trip signal was valid, and the inspectors verified that safety systems performed as designed for plant conditions during the trip. Operator response to the trip was appropriate and emergency procedures were properly followed. One intermediate range nuclear instrument was observed to be overcompensated, and one individual rod position instrument indicated slightly greater than 10 steps after the trip. Operators initiated emergency boration for the rod position problem in accordance with abnormal procedures. The rod position indication problem was corrected, and rod drop time testing was later performed as required by NRC Bulletin 96-01, Control Rod Insertion Problems.

Plant equipment conditions immediately following the trip were good except for the turbine building where numerous secondary relief valves lifted following the reactor trip. Most of the relief valves lifted due to the sudden increase in feedwater pressure, and all except one reseated when pressure was reduced. The relief valve which did not reseat was manually gagged by maintenance personnel. Additionally, the inspectors observed that a large amount of water was discharged from the turbine-driven AFW pump exhaust line during pump startup. No adverse effects upon pump response were observed.

The main generator trip was found to be initiated by actuation of the neutral ground fault relay. Testing confirmed that an actual fault condition occurred on the A phase of the generator. On November 14, generator inspections found that foreign material, a section of clear plastic, had lodged on the cooling tubes for several generator coils. The plastic cut off hydrogen cooling flow and allowed the coils to overheat and fail. The generator required major disassembly for coil replacement, and an extended forced outage was in progress at the inspection period's end.

On November 13, the unit was cooled to hot shutdown, and secondary systems were secured. The unit remained in hot shutdown until previously ongoing repair activities on the Service Water (SW) system were completed (Section E1.2). On November 29, the unit was cooled to cold shutdown.

c. Conclusions

The inspectors concluded that safety system and operator response to the Unit 2 reactor trip was good. Minor equipment problems were promptly corrected. The trip's cause, a main generator failure due to damage caused by foreign material, required an extended unit shutdown for repairs.

O2 Operational Status of Facilities and Equipment

O2.1 Cold Weather Preparations

a. Inspection Scope (71714)

On several occasions during the inspection period, the inspectors reviewed the initial implementation of the licensee's cold weather protection procedures.

b. Observations and Findings

The inspectors performed initial reviews of the licensee's procedures for cold weather protection and their implementation. Procedure 0-GOP-4, Cold Weather Operations, Revision 9, was performed monthly during cold weather or as directed by shift supervision. The inspectors reviewed the procedure completed on November 15 and found that operators had documented completing the actions necessary to protect safety-related systems from freezing. For components where operators identified material discrepancies, the inspectors verified that corrective actions were initiated.

The inspectors also walked down the status of freeze protection equipment near the Refueling Water Storage Tanks (RWSTs) and in various buildings containing safety-related equipment. On November 24, the inspectors identified that an approximately three-inch section of piping to RWST level transmitter 2-QS-LT-200D was wrapped in heat tracing, but was not fully insulated. The inspectors reported the discrepancy to shift supervision, and verified on December 6 that the discrepancy had been corrected. Additionally, the inspectors discussed with station managers the fact that with Unit 2 shutdown for a forced outage, the Unit 2 MSVH temperature was much lower than normal. The managers informed the inspectors that the Unit 2 MSVH temperature was being monitored and that action would be taken (e.g., installing temporary heaters) if MSVH temperature approached freezing.

At the inspection period's end, the inspectors were continuing their reviews of the licensee's cold weather protection activities.

c. Conclusions

Initial reviews of cold weather protection procedures found that activities were properly completed.

O2.2 NRC Notifications

a. Inspection Scope (71707)

The inspectors reviewed the following licensee notifications to the NRC to ascertain if the required reports were adequate, timely and proper for the events.

b. Observations and Findings

On November 12, the NRC was notified as required by 10 CFR 50.72 concerning reactor protection system and engineered safety feature actuations generated when Unit 2 tripped from full power. The inspectors found that the licensee's reporting actions were appropriate. Additional inspection activities and findings are discussed in Section O1.2.

c. Conclusions

One NRC notification required by 10 CFR 50.72 was properly made by the licensee.

O6 Operations Organization and Administration

O6.1 Operator Work-Arounds (OWAs) (71707)

On December 5, the inspectors attended an OWA status meeting conducted at the conclusion of the morning management meeting. There were 18 active OWAs, 2 items assigned as high priority (may affect nuclear safety), 7 assigned as medium priority (important but do not directly affect nuclear safety) and 9 assigned as low priority (minimal impact on plant operations). Each item's status was reviewed in sufficient detail to determine the progress in resolving the item. The meeting was an effective mechanism for keeping plant management informed of the status of each open OWA.

O7 Quality Assurance in Operations

O7.1 Oversight Meeting (40500)

On November 6, the inspectors met with Oversight personnel. Issues discussed included Oversight activities and findings since previous meetings. Copies of recent audits were provided for review. The inspectors also observed Oversight personnel observing plant activities on numerous other occasions during the inspection period and met briefly with them to discuss their observations. The inspectors concluded that the Oversight organization was continuing to assess station performance effectively.

O7.2 Management Safety Review Committee (MSRC) Meeting (40500)

On November 20, the inspectors attended a regularly scheduled MSRC meeting at the North Anna site, and observed Station Manager's plant status reports. The inspectors found that the MSRC meeting met TS 6.5.2 requirements for member composition and quorum and that the agenda appropriately included review items required by TS 6.5.2.7. The inspectors observed that the Station Manager's reports generated

significant self-critical discussions of station performance. The inspectors concluded that the MSRC meeting was in compliance with TS requirements and that substantive assessment issues were being addressed in the discussions.

O8 Miscellaneous Operations Issues (92901, 92700)

O8.1 (Closed) Unresolved Item (URI) 50-339/96004-01, review significance of safeguard area ventilation not meeting design basis

(Closed) Licensee Event Report (LER) 50-339/96001, potential unfiltered release path from the quench spray pump house to the environment

a. Scope

As a result of questions asked by the inspectors, on May 15, 1996, the licensee identified and reported to the NRC a potential unfiltered release path to the environment following a design basis accident. As a result of the problem, both trains of safeguards area ventilation systems were declared inoperable. During this inspection period, the inspectors reviewed the problem's significance.

b. Observations and Findings

The problem identified by the inspectors was that a flowpath existed between the safeguards area and the Quench Spray (QS) area sumps. The safeguards area contained the Low Head Safety Injection (LHSI) pumps, Outside Recirculation Spray pumps, and associated piping and valves. Leakage from these components was directed to a sump described in Updated Final Safety Analysis Report (UFSAR), Section 15.4.1.8, as sized to accommodate the design basis leakage from area components, including a 50 gallons per minute (gpm) LHSI pump seal leak lasting 10 minutes. The size of the safeguards sump and sump drainage were important because credit was taken that operators would notice any leakage of radioactive water to the sump and take action to isolate the leakage. The safeguards sump was found to be directly tied to the QS sump via a six-inch pipe connected between the two areas. The connection potentially allowed a portion of the design basis radioactive leakage to flow to the QS sump. This leakage could have inhibited prompt operator identification of leakage into the sump. Additionally, the QS area ventilation system was not designed to handle radioactive effluents, and an unfiltered and unmonitored release could result.

The inspectors reviewed the licensee's corrective action for this problem. Along with making a 10 CFR 50.72 report, the licensee reported this event in LER 50-339/96001. A temporary cap was placed over the pipe until a permanent repair was prepared. Later, a Design Change Package (DCP) was implemented to permanently plug the pipe and to install a pump in the QS sump.

The inspectors reviewed the event's significance. The LER stated that, "The event posed no significant safety implications because any increase in dose through the unfiltered release path would be well within the limits of 10 CFR 100." The inspectors reviewed the above statement with the licensee's nuclear fuels group personnel on September 5. The inspectors found that the pump seal leak contribution to off-site dose was small compared to total off-site dose referred to in UFSAR Section 15. With no filtration, the calculated dose associated with a LHSI pump seal leak was increased by a factor of 10, but was still only a small contributor to total dose. The inspectors concluded that the safety consequence of the cross-tied sumps, with regards to off-site dose, was minimal.

The inspectors also questioned the potential increase in control room dose due to the cross-tied sumps. Licensee engineers postulated that the expected control room doses would be similar to those estimated following a separate problem identified with inadequate operator procedures for responding to a fuel handling accident (NRC Inspection Report Nos. 50-338, 339/96-09). Similar to that problem, the control room doses could possibly exceed 10 CFR 50 General Design Criteria (GDC) 19 limits for operators. However, such estimates contained numerous conservative assumptions which meant that if an accident had actually occurred with the leakage path through the QS area, the actual dose to control room operators would not likely exceed the GDC 19 limits.

While reviewing the above problem on May 23, 1996, the inspectors also identified that the floor drain hole that permitted the B LHSI pump leakage to drain from its pump cubicle to the safeguards area sump was grouted closed. The grouted ends had been covered with paint indicating the drain hole had been in this condition for a long time period. The inspectors noted that the UFSAR Section 15, Accident Analysis, Condition IV - Limiting Faults for a Loss of Coolant Accident (LOCA) assumed a pump seal failure. Specifically, Section 15.4.1.8, Doses From Leakage From Emergency Core Cooling System Components, stated, "The 50 gpm leakage due to the pump seal failure is assumed to last 10 minutes subsequent to initiation of the leak. The leakage is limited to 10 minutes (500 gallons of total leakage) because the leak will be quickly detected by the safeguards area sump-level monitor and alarm." The grouted hole prevented the sump-level monitor and alarm from performing its function for a pump seal leak on the B LHSI pump.

This issue was discussed with system engineers. After review, DR N-96-1059 was issued to document the problem and track corrective actions. The licensee found that the drain holes were plugged in both the B LHSI and the B Outside Recirculation Spray pump cubicles. The licensee also found that with the drain holes plugged, seal leakage could accumulate to approximately 1500 gallons before draining over to the safeguards sump, which was three times that analyzed in the UFSAR. The holes had apparently been plugged since initial construction.

The licensee reviewed this issue for reportability to the NRC and concluded the following as documented in response to the DR:

- The condition was not outside of the design basis since maintaining off-site doses within 10 CFR 100 limits met design basis criteria.

- The associated safety equipment was not affected by the condition, and the ability to process the ventilation discharge through the charcoal filters was not altered. Therefore, the ability to mitigate the consequences of an accident was not significantly altered.

The inspectors found that the licensee's assessments of the significance of the plugged cubicle drain holes were appropriate. Shortly after discovery, the licensee unplugged the drain holes in both cubicles and verified that drains for all other cubicles were open.

The inspectors concluded that since early in plant life, the safeguards area configuration did not match that described in the plant design basis. The licensee's failure to ensure proper facility design was a violation of 10 CFR 50, Appendix B, Criterion III, which required the licensee to establish measures to ensure that applicable regulatory requirements for the design basis described in license documents be implemented. Contrary to this requirement, since early in plant life until May 1996, the licensee failed to ensure that safeguards area walls met the design basis for containing pump seal leakage as described in UFSAR Section 15.4.1.8. Specifically, a hole existed between the safeguards area and the QS area where the design required that the safeguards area be fully separated. Additionally, holes designed to exist between pump cubicles and the safeguards area sump in order to allow drainage of pump seal leakage were found to be plugged. This is identified as a violation of 10 CFR 50, Appendix B, Criterion III, requirements (50-339/96012-01). This violation is considered to have occurred in the Engineering functional area.

c. Conclusions

A violation was identified for a failure to ensure proper facility design control. Unit 2 safeguards area walls were found to not meet their design basis for controlling safeguards pump leakage since early in plant life. One URI was closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Observations (62707)

Throughout most of the inspection period, the inspectors reviewed Service Water Restoration Project (SWRP) efforts to replace sections of

the Train B SW headers to the Component Cooling Water (CCW) heat exchangers. During the SWRP, the licensee entered and exited various TS Action Statements specifically approved for the SWRP. The inspectors verified that TS compliance was maintained and that compensatory actions required by the NRC Safety Evaluation Report for the TS Action Statements were properly implemented by the licensee. The inspectors also observed that the overall control of SWRP work activities was appropriate. The only deficiency noted by inspectors during the work was that a flexible cable jacket was pulled out of a connector on valve 1-SW-MOV-113A. Due to the large amount of work in the area of the valve, it appeared that the jacket had probably been pulled out due to unknown individuals stepping on the cable. DR N-96-2687 was initiated, and the configuration was reviewed and found not to be an operability concern. A work request was initiated for corrective action. The inspectors concluded that overall the SWRP work activities were well controlled, and supporting system manipulations were performed in accordance with regulatory requirements and commitments.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Emergency Diesel Generator (EDG) Radiator Fan Exhaust Compartment Housekeeping (71707)

On December 4, the inspectors, accompanied by a system engineer, performed a walkdown of the EDG radiator fan exhaust compartments located on the Service Building roof. Debris was observed on the ledges above the radiator fans for three of the four EDGs. The debris included two four-by-six-inch metal plates. With the radiator fan blades exposed, a seismic event, which results in EDG starts or occurs while an EDG is operating, could cause the debris to fall onto the radiator fan blades with unknown consequences. The system engineer notified the shift supervisor of the observation so that the debris could be removed. Deviation Report N-96-2749 was issued concerning this condition.

M8 Miscellaneous Maintenance Issues (92902, 92700)

M8.1 (Closed) Violation 50-338/95015-01, failure to follow procedures for properly controlling safeguards area ventilation system (SAVS) maintenance. This event involved the requirement of following procedures to ensure that maintenance on safety-related equipment be properly preplanned and performed to minimize the impact on operability of other associated safety equipment. Senior reactor operators failed to verify whether maintenance work on one SAVS train would affect the other redundant equipment train.

The inspectors verified that the licensee performed adequate investigation and established the root causes for the event. Multiple personnel errors in this case caused the process that would have ensured the prevention of such an event to breakdown. The licensee took remedial actions through the disciplinary process, and the event was discussed at training sessions and shift turnovers and was instituted as a part of the Licensed Operator Requalification Program.

The inspectors verified that special training sessions took place, that remedial action was taken, and the training materials and procedures were reviewed in detail to ensure adequacy. The inspectors concluded that the licensee's response dated October 12, 1995, and the corrective actions were appropriate and had been adequately implemented.

M8.2 (Closed) Violation 50-339/95020-02, failure to comply with 3.6.1.3 for air lock outer door rendered inoperable by open test connection. This violation concerned an inoperable Unit 2 containment air lock outer door due to valve 2-CE-4 being left uncapped. When the containment air lock outer door became inoperable, unknowingly the plant was then under a TS Limiting Condition for Operations. TS 3.6.1.3, Action A, required that the operable door be locked within 24 hours or that the plant be placed in hot standby within the next six hours and placed in cold shutdown within the following 30 hours.

The inspectors reviewed the licensee's response and commitments to this event to determine the adequacy and appropriateness of corrective actions taken and the implementation of those actions. The inspectors verified that the licensee's response to the violation clearly documented the efforts expended to preclude recurrence of the problem.

The inspectors reviewed procedure 2-PT-62.1, Containment Air Locks - Leakage Rate, Revision 17, and verified that the procedure had been revised and updated. The licensee's efforts to correct the causes of the event included coaching the individuals who performed the procedure on the importance of procedure implementation and self-checking. The event and lessons learned became part of the licensee's requalification program. The inspectors concluded that the licensee's response dated January 9, 1996, and corrective actions were appropriate and had been adequately implemented.

M8.3 (Closed) LER 50-338, 339/96009, reactor trip bypass breaker missed surveillance due to inadequate surveillance test procedure. This LER reported the identification on October 10, 1996, that a surveillance test was not being performed as required by TS 3.3.1.1. The surveillance tests were not testing the reactor trip bypass breaker manual shunt trip prior to placing the breaker in service. The shunt trip was instead being tested immediately after closing the normally racked-in breaker. Licensee personnel identified the deficiency after visiting another facility and noting differences in the testing procedures. The licensee postulated that the procedure inadequacy was caused by personnel mis-interpreting TS wording and failing to identify that a procedure change was needed following a 1986 TS change. A review of NRC Generic Letter 85-09, Technical Specification for Generic Letter 83-28, Item 4.3, revealed that the prior interpretation was incorrect. As corrective action, the licensee modified surveillance test procedures to require that the bypass breaker be racked out and the shunt trip tested prior to closing the breaker. The inspectors verified that these procedure changes were completed. Additionally, a UFSAR change was planned to update test sequence descriptions.

TS Surveillance Requirement 4.3.1.1.1, Table 4.3-1, Notation 8, requires that the reactor trip bypass breaker local manual shunt trip be tested prior to placing the breaker into service. Contrary to this requirement, from approximately June 9, 1986, until approximately October 10, 1996, reactor trip bypass breakers were placed in service without first testing the local manual shunt trip. This is identified as a violation for an inadequate surveillance test, in that, the test did not meet TS surveillance requirements. This licensee-identified and corrected violation is being treated as an NCV, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-338, 339/96012-02).

M8.4 (Closed) LER 50-338/96010, automatic reactor trip due to failure of a generator negative phase sequence relay. This LER discussed an October 24, 1996, Unit 1 trip from full power due to a failure in the main generator negative phase sequence relay. The licensee's response to the event and corrective actions for the associated equipment failures were reviewed and discussed in NRC Inspection Report Nos. 50-338, 339/96-10.

M8.5 (Closed) LER 50-339/96003, automatic reactor trip resulting from main generator stator coil failure due to personnel error. This LER discussed a November 12, 1996, Unit 2 trip from full power due to a main generator failure. The licensee's response to the event and initial corrective actions for the event are reviewed and discussed in Section O1.2. The inspectors also verified that appropriate corrective actions were initiated for all associated minor equipment failures following the trip.

III. Engineering

E1 Conduct of Engineering

E1.1 Problem with Bolts - SW and Auxiliary Service Water Pump

a. Inspection Scope (37550)

The inspectors reviewed a problem, documented in DR N-96-2359, involving the seismic qualification for a replacement SW pump and stress on the pump's flange bolts. Requirements related to this review included American National Standards Institute (ANSI) B31.7, Power Piping.

b. Observations and Findings

The licensee was in the process of replacing four safety-related SW pumps due to aging concerns. Replacement SW pump 2-SW-P-1A was installed in December 1995. The remainder were in manufacturing.

In October 1996, the manufacturer's Quality Control inspector, while inspecting the second pump at the factory, identified a discrepancy in the bolts at various column flanges and the bowls. The drawings indicated one-inch diameter bolts were required, but the actual hardware installed was three-quarter inch diameter bolts. The manufacturer

contacted the licensee to ask for approval to ship the pump with smaller bolts about October 14, 1996. This prompted the licensee to review the seismic qualification report (AR120N, Revision 1), the outline drawings, and the bill-of-material for the pump received in 1995. The licensee identified a discrepancy between the seismic report and the bill-of-material regarding the bolts in question. The seismic qualification report was based on one-inch diameter, A325, high strength, dynamic loading bolts, but the bill-of-material showed one-inch diameter, A307, low strength, non-dynamic loading, structural bolts.

Concerned about operability of the installed pump in light of the new information, the license requested the manufacturer to perform a seismic re-analysis based on the actual bolts supplied with the new pump (one-inch diameter, A307). When the licensee reviewed this re-analysis, they identified errors. The re-analysis used the wrong bolt tensile strength area and compared calculated stresses to the wrong criteria (wrong code), and therefore reached a wrong conclusion. Revision 1 of the seismic report provided by the manufacturer did not have these errors. The licensee then performed their own re-analysis on October 16.

The licensee's analysis showed a calculated stress for the Design Basis Earthquake condition of 28.6 ksi on the bolts compared to ANSI B31.7 allowable stress of 16.8 ksi. ASME Section III, Appendix F, states that the pump would be operable if the stress did not exceed 0.7 times the Ultimate Tensile Strength or 42 ksi. Therefore, the licensee concluded the pump was operable. Replacement of the non-conforming bolts with proper bolts was scheduled for January 13, 1997. In addition, on October 21, 1996, the licensee issued a revised purchase specification calling for A193 Group B7 high strength bolts to be supplied with the new pumps. Also, the licensee planned to inspect two Auxiliary Service Water pumps, which were classified as important to safety, to determine the bolt type used in these pumps. If necessary, those bolts would be replaced as well.

The inspectors inquired whether a notification of defect would be issued pursuant to 10 CFR 21. The licensee stated that evaluation of the need for a report was in progress.

The licensee had documents, namely the seismic report and the bill-of-material, which contained a discrepancy and deficiency with regard to the material of the bolts in question. However, the licensee's design reviews failed to identify these before installation of the first replacement pump. This problem in the design control program contributed to installation of non-conforming bolts, and constitutes a violation of 10 CFR 50, Appendix B, Criterion III, Design Control, which requires that measures shall be established to ensure the design basis are correctly translated into drawings. The matter is identified as a violation (50-339/96012-03).

c. Conclusions

The inspectors concluded that the licensee's operability evaluation for non-conforming bolts in a recently installed pump reached a proper conclusion in accordance with NRC guidance, and was completed in a timely manner. The scheduled date for correcting the non-conformance was acceptable. Even though the problem was primarily caused by the manufacturer, reviews in the licensee's design control program did not identify that the wrong bolts were installed. For this reason a Notice of Violation was issued.

E1.2 Review Supporting Plant Cooldown During SW Maintenance (37551)

During the licensee's SWRP to replace sections of the Train B SW piping going to the CCW heat exchangers (Section M1.1), a Unit 2 forced outage was required for main generator repairs. The inspectors reviewed the licensee's decisions with regard to the shut down status of Unit 2 to ensure that regulatory requirements were met and that Unit 2 was maintained in a safe shutdown condition. The inspectors reviewed the licensee's submittal for TS changes required to support the SWRP. In the submittal documentation, the licensee stated that if an outage was required for either unit during the SWRP, an assessment would be performed. The inspectors found that shortly after shutdown, the licensee completed the required assessment which was contained in an Engineering Transmittal (ET) No. NAF-96201. The inspectors reviewed the ET and found that it accurately assessed the situation. Specifically, it analyzed the reliability of available SW equipment to support shutdown operations and made appropriate recommendations for compensatory actions. Additionally, the inspectors noted that managers considered the ET and other available information and made a conservative decision to keep Unit 2 in hot shutdown and not proceed to cold shutdown until the SWRP was completed. The inspectors concluded that the licensee appropriately analyzed the impact of the SWRP on the unit shutdown and made conservative decisions concerning unit status.

E1.3 Inservice Testing of Decay Heat Release (DHR) Stop-Check Valves

a. Inspection Scope (37551)

During the week of November 12, the inspectors reviewed the engineering basis for deleting Inservice Test (IST) requirements for DHR stop-check valves 1/2-MS-19, -58, and -96.

b. Observations and Findings

During the previous inspection period, the inspectors observed that operators used the Unit 2 Decay Heat Release (DHR) valve to control temperature during a forced outage. The DHR valve was an air-operated Steam Generator (SG) atmospheric relief valve located on a common header supplied from all three SGs. The header was normally isolated by manually-operated stop-check valves 1/2-MS-19, -58, and -96. On October

22, the inspectors observed that all three Unit 2 stop-check valves were open to allow DHR valve use. The inspectors questioned engineers concerning the IST history for the stop-check valves and were informed that ISTs for the valves had been deleted in early 1996. Licensee engineers provided the inspectors with a memorandum from C. Snow to B. Foster dated October 24, 1996, describing the basis for deleting ISTs.

The inspectors reviewed the memorandum and found that it stated that the ISTs were deleted because UFSAR Section 15.2.13.3 provided an analysis of simultaneous blowdown of all three SGs due to a failure of the three inch DHR line. The memorandum stated that American Society of Mechanical Engineers (ASME) Code Section XI, paragraph IWV-1100, required testing only for valves which were, "required to perform a specific function in ... mitigating the consequences of an accident." Since the UFSAR did not take credit for the valves' isolation in analyzing for a DHR header break, the licensee concluded that ISTs were not required. The inspectors accepted this conclusion for the accident analyzed in UFSAR Section 15.2.13.3.

However, the inspectors identified that a different accident function for the valve had not been properly considered by the licensee. Specifically, UFSAR Section 15.4.2.1 described analyses for a Main Steam Line Break (MSLB) occurring within the isolation boundaries for the SG (e.g., upstream of the stop-check valves). The analyses in that section assumed that only one SG would be depressurized for such a failure. On November 19, the inspectors informed licensee engineers that for this assumption to be valid when the DHR valve is in use, the check valve feature of the stop-check valves must function correctly. The stop-check valves were needed to prevent backflow from intact SGs into a faulted SG in order to prevent simultaneous blowdown of all SGs. The stop-check valves served a function similar to check valves in the steam lines supplying the turbine-driven AFW pump, for which ISTs were performed. After reviewing the issue, licensee engineers submitted DR N-96-2674 on November 22. The DR stated that IST testing for the stop-check valves should have been continued in order for operators to be allowed to use the DHR valve for temperature control as was done on October 22 through 24.

On November 26, licensee engineers provided the inspectors with a copy of ET NAF-96207, Review of Decay Heat Removal Using the Decay Heat Release Valve and Impact on the Plant Safety Analyses North Anna Power Station Units 1 and 2, dated November 26, 1996. The ET evaluated the effect of a failure of the check valve function on the accident analyses for a MSLB. The ET concluded that for the plant situation existing on October 22 through 24, (reactor in MODE 3, all rods in, main feed isolation valves shut, and main steam non-return valves shut), operation of the DHR valve was acceptable and the licensing basis MSLB analysis remained valid with an assumed failure of a DHR stop-check valve. The inspectors found that the ET prepared on November 26 demonstrated that for the plant configuration actually occurring when the DHR valve was used, the problem did not have major safety significance.

Overall, the inspectors found that the licensee had deleted the ISTs for the DHR stop-check valves without proper supporting analysis or controls to prevent their use. In order to delete testing the check valve functions of 1/2-MS-19, -58, and -96, use of the DHR valve should have been prevented by modifications or procedural controls, or the licensing basis MSLB analyses should have been re-preformed assuming a stop-check valve failure. Neither of these actions were taken until questions were raised by the inspectors. The licensee's failure to ensure proper IST for the valves was a violation of TS 4.0.5 surveillance requirements which required that ASME Code Class 1, 2 and 3 components be inspected and tested in accordance with ASME code Section XI. Contrary to this requirement, the licensee deleted testing requirements for the stop-check function for the six DHR isolation valves without properly evaluating or limiting their function during accident conditions. Three of the six valves were opened for use from October 22 through 24, 1996. This failure constitutes a violation of minor significance and is being treated as an NCV, consistent with Section IV of the NRC Enforcement Policy (50-338, 339/96012-04).

c. Conclusions

An NCV was identified for a failure to meet ASME code Section XI requirements for testing the check valve functions of six steam generator DHR isolation valves.

E1.4 Minor Modification Review Team (MMRT) Meeting (37551)

On December 4, the inspectors attended a MMRT meeting. Since these meetings had been temporarily suspended after the October 16 meeting because of outages, there were several new Requests for Engineering Assistance to be reviewed. The primary purpose of the meeting was for management to approve, modify, or cancel proposed minor modifications. The reviews considered nuclear plant safety, personnel safety, and economic benefits, generally in that order. The discussions also considered if a minor modification was the proper vehicle for addressing an issue. The meeting was professionally conducted. The inspectors had no concerns involving the disposition of the items.

E2 Engineering Support of Facilities and Equipment

E2.1 Resolution of Microbiological Influenced Corrosion (MIC) Corrosion of Stainless Steel SW Piping

a. Inspection Scope (37550)

The inspectors reviewed the licensee's ongoing activities in the area of leaks in the SW piping. The regulatory requirement relevant to the scope of inspection was 10 CFR 50, Appendix B, Criterion XVI, Corrective Action.

b. Observations and Findings

The licensee was resolving the problem described in DR N-96-2492 which identified the presence of two pin hole leaks on a four-inch diameter stainless steel alloy 316 SW piping weld. The licensee inspected the piping for the possibility of MIC since this form of corrosion had been prominently identified in the carbon steel SW piping, and had been identified in a stainless steel weld failure in the SW System in 1993.

A review was made by the inspectors of the Nondestructive Examination (NDE) Procedure used for evaluating the MIC found in the stainless steel SW line, No. NDE-RT-102, Radiographic Examination to Detect ...MIC, Revision 0, dated November 15, 1996. The attack of MIC occurs in the weld or in close proximity to the weld in stainless steel piping. The procedure was discussed with a corporate NDE, Level III, radiographic specialist and the X-ray film for welds WS-18E Line 4"-WS-56-163-Q3 Weld Nos. 60, 61, 64, and 65 were reviewed with the radiographer for the presence of MIC attack. Further discussions were held with the radiographer concerning the evaluation of the length of the MIC defect size and the inspectors considered those evaluations to be conservative. Also, the licensee noted that any weld defects, lack of fusion, etc., were added to the defect size in this evaluation. The licensee radiographed 15 welds and only one of these welds did not have any areas of MIC attack. The radiographer shot six film for each weld. The defect sizes noted in the 14 welds varied from ⅛ inch to 1½ inches in length with the longest cumulative length being 3½ inches in weld No. 89 in Line 4"-WS-56-163-Q3.

The inspectors accompanied one of the licensee's inspectors performing the weekly visual examination of the welds in this service piping and discussed the results and inspection requirements for this examination.

While the licensee was still developing their SW action plan (some of the actions would depend on inspections that had not yet been completed), the inspectors reviewed the part that had been formulated as of November 22, 1996. The licensee started preparation for replacement of a section of four-inch diameter pipe, and was preparing to visually inspect all accessible welds in stainless steel SW piping. The licensee stated that the visual inspections would include 1½-, 3-, 4-, and 8-inch diameter piping. The licensee was also considering selecting a sample of welds for radiography.

c. Conclusions

The licensee's approach to resolving a potential problem with MIC in stainless steel SW piping was reviewed. One section of 4-inch diameter pipe was scheduled for replacement based on findings to date. The licensee was at the investigation stage, and the total corrective action plan was under development. Evaluations performed by the licensee were reviewed and found to be conservative. No concerns were identified by the inspectors.

E2.2 Repair-Replacement of Carbon Steel SW Piping

a. Inspection Scope (37550)

The inspectors reviewed Design Change Notice (DCN) No. 94-010, Repair/Replacement of Exposed SW Piping to/from CCHXs, North Anna Units 1&2, and reviewed work being performed to implement the DCN. Requirements for the scope of this inspection were 10 CFR 50, Appendix B, Criterion III, Design Control; and Criterion V, Instructions, Procedures and Drawings.

b. Observations and Findings

The inspectors reviewed DCN package No. 94-010, including written descriptions of the change, isometric drawings for pipe support locations, and the erection control isometric. The package was adequate for the field implementation underway. The inspectors discussed the SW repair/replacement project with the cognizant engineer, and accompanied one of the assigned engineers into the auxiliary building basement where some of the exposed SW piping was being replaced. Using the reviewed DCN package, the inspectors reviewed some of the ongoing work.

The inspectors reviewed the welding technique sheet being used to weld the carbon steel piping (Welding Technique No. 103). The root passes were welded using Gas Tungsten Arc Welding (GTAW) and the remaining part of the weld was made using Shielded Metal Arc Welding (SMAW). The inspectors reviewed the material certifications of the bare wire used in GTAW welding, the covered electrodes used in SMAW welding, and the certifications for one section of piping (piping is ASME SA106 Group B). The welder performance qualification records were reviewed for three of the welders performing work on this modification. The welds appeared to be of good quality and no problems were identified with the documentation.

c. Conclusions

The engineering DCN package was well prepared for implementing a carbon steel piping replacement in the SW System, and the quality of work being performed by the craftsmen was good.

E2.3 Condition of Safety-Related Batteries

a. Inspection Scope (37550)

The inspectors performed a detailed visual inspection of each cell in the safety-related vital and diesel generator batteries for both units (total of twelve battery banks). The inspectors also performed a general inspection of the racks and environment for the batteries. The inspection was performed in accordance with Institute of Electrical and Electronics Engineers (IEEE) 450, IEEE Recommended Practice for Maintenance Testing of Large Lead Storage Batteries for Generating Stations, Section 4.3, Inspections. This standard was not mentioned in

the UFSAR, but was used as a guide. This particular inspection item was chosen, in part, because there have been several problems reported with safety-related batteries throughout the industry.

b. Observations and Findings

The batteries and racks were well maintained. The rooms for the vital batteries were well maintained and ambient conditions were acceptable. The ambient conditions for the diesel generator batteries, which were located with the generators, were acceptable. All charger output currents and voltages were normal. The electrolyte level was within specification. No cracks or corrosion were observed. The last pilot cell data was posted at each battery.

Cell No. 30 in bank 1-III and cell No. 60 in bank 1-IV had sedimentation of significant depth below one or two plates. In response to this observation, the licensee initiated DR N-96-2635. As part of the DR resolution, the manufacturer's technical representative came to the site and inspected the batteries. His report stated that the two cells mentioned above were showing signs of reconversion to sponge lead from the positive plate settlement.

The concern with sedimentation is that it could cause the voltage to decay. The inspectors reviewed copies of the data sheets from the last surveillance tests which checked the voltage, specific gravity and level of all cells. The surveillance had been performed on June 4, 1996, according to PT-96-86B. The inspectors reviewed copies of the data sheets for the most recent capacity test which had been conducted on September 24, 1994, according to PT-88D. The inspectors observed that all the cells showed good voltage and capacity in the surveillances, although the cell inspections had not identified any problem with sedimentation.

The licensee indicated to the inspectors that they would define an enhanced monitoring program for the cells with excess sedimentation. The 1-III battery had a date code February 1987, and the 1-IV battery had a date code of January 1992.

DR N-96-0421 indicated that a short-circuit had occurred at the main terminals of battery 1-IV during preparations for a service test. The inspectors examined the condition of the battery terminals. The inspectors observed that some metal was melted away, but the damage was of no consequence to future performance of the battery.

c. Conclusions

A detailed cell inspection was performed on each cell of all the safety-related batteries. The inspectors concluded that battery maintenance met requirements. However, the inspectors observed significant sedimentation in two cells. The inspectors concluded that

the sedimentation had been present on June 4, the time of the last surveillance inspection. Therefore, the fact that the surveillance did not record any problems with sedimentation was considered a weakness in the implementation of that inspection.

E2.4 Review of Mechanical and Structural Related Potential Problem Reports and Deviation Reports

a. Inspection Scope (37550)

The inspectors reviewed the disposition and corrective actions for the below listed Potential Problem Report (PPR) and DRs to determine if they were adequately reviewed, evaluated, resolved and corrected. The items reviewed were:

Item No.	Condition
PPR 96-018	Internal Pressure Concerns in Closed Piping in Containment During a Design Basis Accident
DR N-96-0384	A Pinhole Leak at Engine Sump of EDG
DR N-96-0450	Leakage on One-Inch Diameter Chemical Addition Piping Expansion Joint
DR N-96-0534	Response Spectra for Auxiliary Building
DR N-96-0843	Effects of Post-accident Temperature on the Recirculation Spray (RS) and Quench Spray (QS) Systems
DR N-96-1169	Quench Spray System Pump Flow Uncertainty
DR N-96-2498	Steam Generator Blowdown Tank Supports

As appropriate, the inspectors verified corrective actions by reviewing revised procedures, and drawings. The inspectors reviewed calculations and modifications as appropriate.

b. Observations and Findings

While some of the DRs were still in the process of evaluation or analysis, the inspectors concluded that the performance of the engineers was good. However, the inspectors had a comment on a tank isolation sequence carried out by Operations personnel. The corrective action for DR N-96-2498 included draining and isolating the steam generator blowdown tank for each unit. When the inspectors verified this action, he noted that one drain valve (1-BD-33) was in the correct position, but it was not tagged. Review of the system revealed that the position of valve 1-BD-33 was of no consequence to accomplishing the intent of the evolution as long as the other valves were in the correct position. This represented an inattention to detail in preparing the evolution instruction, since the tagging record did not include this valve.

c. Conclusions

A review of six Deviation Reports and one Potential Problem Report related to mechanical and structural engineering indicated good support for facilities and equipment by engineers.

E2.5 Overvoltage on 480 V Bus

a. Inspection Scope (37550)

The inspectors reviewed a situation where relatively frequent overvoltages occurred on one or more 480 V buses. Most of the overvoltage conditions occurred during plant shutdowns, but they occurred during power operation as well. This situation involved changing a previously submitted response to an NRC request for information and performance of a safety evaluation (10 CFR 50.59). Documents reviewed at the outset included the following:

- Letter, VEPCO to NRC, dated June 4, 1982, on the subject of "General Design Criterion 17 Analysis North Anna Units 1 and 2."
- Letter, VEPCO to NRC, dated September 10, 1996, on the subject of "General Design Criterion 17; Revised Commitment Related to Response to Overvoltage Conditions."
- Safety Evaluation 96-SE-OT-4B.
- Type 1 Report NP-3085 (partial).
- Alarm Response Procedure, 1F-H5, 480 V or 4 kV Emergency Bus Volts Hi/Lo.

Inspection activity of this item included a walkthrough of an alarm response procedure involving manual operation of an on-load transformer tap changer and reading of voltage indicators and computer outputs. Also, drawings, calibration data and voltage relays were inspected.

Requirements relevant to this inspection scope were: 10 CFR 50, Appendix A, Criteria 13, 17, and 19; Regulatory Guide 1.33, Appendix A, Item 5; Regulatory Guide 1.97 and 10 CFR 50.59.

b. Observations and Findings

The source transformer had an on-load automatic tap changer to maintain the voltage on the 4 kV bus. Design basis loading scenarios dictated that fixed taps on the 4160-480 V transformer must be set to boost voltage. With voltage at the 4 kV bus at the low end of the tap changer control band, voltage at the 480 V bus would be in specification. With voltage at the 4 kV bus at the high end of the tap changer control band, overvoltage would occur on a lightly loaded 480 V bus. Overvoltage

relays were installed on each 4 kV and main 480 V bus to warn operators of an overvoltage condition. The relays were wired to a common control room annunciator point and individual computer points.

When an overvoltage alarm was received, the operator was prompted by the alarm response procedure to determine which buses were normal and which were abnormal. This was done primarily through checking of voltmeters and the relevant computer alarm points. In the case of 4 kV normal and 480 V overvoltage, the procedure was to put the source transformer tap changer in manual, and depress voltage on the 4 kV bus to the low end of the acceptable range. The tap changer would then be returned to automatic mode. The tap changer control band was about 160 volts and the tap changer made an adjustment of 26 volts per step.

The maximum calculated voltage on a 480 V bus with the 4 kV voltage in specification was 521.9 V. Between 1982 and a recent procedure change, the operators were allowed 15 minutes to take the actions described above. The procedure was changed recently to allow a response time of two hours. The two hours applied only to performing the tap changer manipulations. The change was supported by a safety evaluation, and the NRC was notified of the procedure change by letter. The procedure also covered the case of diesel generator as the power source.

The 480 V bus voltage could be read at a computer terminal in the main control room and other locations. This was accomplished through potential transformers, ac/dc transducers, a multiplexer and the Emergency Response Facility computer. The transducers were wired to read line-to-neutral voltage, and the multiplexer applied a factor of 1.73. The purpose of this factor was to present a line-to-line magnitude voltage which was more recognizable to operators.

The inspectors read the following voltages at the Emergency Response Facility terminal for the 1H1 bus: A=493, B=530 and C=519. These voltages were significantly unbalanced and two of the three were above the alarm set point of 515 V. The licensee initiated DR N-96-2634 to resolve this apparent discrepancy. Using an accurate portable voltmeter, the licensee read voltages directly on the 480 V bus and at various points in the potential circuit. These readings indicated that voltages were normal and balanced -- 507, 504 and 506. The readings also indicated that the multiplexer was introducing an error in the computer readout of voltage. The inspectors requested a copy of the last calibration of the voltage loop at the multiplexer. The last calibration had been performed on April 26, 1996. One of the phases had a significant error, and was adjusted at that time. Since that time, the multiplexer had drifted out of calibration. As a result of the inspection findings related to reading voltage at the 480 bus, the licensee was considering the following three actions:

- Re-wire the transducers to read line-to-line voltage.

- In the interim, revise the ERF nomenclature to indicate the reading is line-to-neutral multiplied by 1.73.

- Reduce the calibration interval of the multiplexer.

c. Conclusions

The alarm response procedure for bus overvoltage was a workable procedure, and was consistent with statements made in letters to the NRC on the subject of overvoltages. The safety evaluation supporting the change in maximum allowable response time for performing tap changer manipulations was accurate and complete.

Some corrective actions were proposed by the licensee to address NRC comments with regard to reading voltage at the 480 V bus, but this did not represent any programmatic weakness. The overvoltage relays with alarms were a conservative design in relation to standard industry practice.

E2.6 Adequacy of Voltage for Diesel Generator Breaker Close Coil

a. Inspection Scope (37550)

From a summary of DRs written on electrical systems since November 1995, the inspectors selected DR N-96-0334 for further review. This DR, initiated February 19, 1996, involved the failure of safety-related vital battery 1-I to meet one acceptance criterion during performance of the last service test.

The relevant requirements for this inspection item were TS 4.8.2.3.2.d (battery service test) and 10 CFR 50, Appendix A, Criterion 17, Electric Power Systems.

b. Observations and Findings

The acceptance criterion from 1-PT-87H, DC Distribution System Service Test (Train A), that was not met in the last service test was to have at least 115.6 VDC at the battery terminals at t=10 seconds. The measured voltage was 115.3 VDC.

The reason for this criterion was to demonstrate that the breaker close coil for diesel generator 1H was operable. The criterion was determined by calculation and directly related to the close coil rated minimum operating voltage of 70 VDC. As confirmed by the inspectors, the source of this value was a letter from Brown Boveri Co., to VEPCO, dated June 24, 1986, on the subject of "Reliable Minimum Close Coil Voltage." A similar criterion applied to battery 2-I, but the other vital batteries did not have a corresponding criterion because calculations indicated that the voltage at t=10 seconds was not critical.

The licensee completed an Operability Determination on February 20, 1996. The report, which was reviewed by the inspectors, concluded that the battery was operable based primarily on the logic that the service

test result was only 0.3 V less than the conservatively calculated minimum voltage. Also, the report referenced a test conducted on an identical breaker in December 1985 which demonstrated breaker operation with a control voltage of about 40 VDC.

The Operability Determination also recommended that the output breaker for diesel generator 1H be tested to demonstrate operability at control voltages less than 70 V. This was expeditiously done, and the breaker was demonstrated to operate with a minimum control voltage of about 40 V. The inspectors asked about the test methodology to determine a minimum operating voltage. The test engineer explained that he slowly increased voltage to the close coil being tested with a rheostat. The inspectors commented that a similar test conducted at another site had resulted in damaging a close coil which had a rated minimum operating voltage of 100 VDC and a rated energize time of one minute. The test engineer recollected that he increased from zero to operate voltage in about 15 seconds. The rated energize time was not readily available. The breaker in question operated several times during diesel generator surveillance tests since the special test was conducted. While the inspectors agreed that the diesel generator output breaker was OPERABLE, there still remained the question of whether the test procedure affected the useful life of the close coil.

The inspectors noted that the criterion for battery voltage at t=10 seconds did not provide margin above the calculated value. Should the service test result be only slightly above the criterion, the procedure would not require any further evaluation. However, in such a case, given an 11-year old battery, normal aging could very possibly result in below criterion results at the next scheduled test. To address this concern, the inspectors reviewed results of the last service test on battery 2-I, which supplied diesel generator 2H output breaker control power. The test result was 118.5 VDC as compared to a required 111.4 VDC. Therefore, there was sufficient margin in the last test results to offset normally expected aging. The licensee stated they would re-consider their t=10 second criterion in light of battery aging.

Given the importance of the components involved, the inspectors decided that this issue should be re-visited at a future date. The specific items to review are:

- Enhancement to the voltage criterion in the service test procedure to account for aging.

- Enhancement to the test procedure which demonstrated close coil operating voltage.

- Evaluation of the close coil ratings versus the test procedure used in February 1996.

- Evaluation of future service test results versus the close coil rating.

The issue will be tracked under an Inspection Follow-up Item (50-338, 339/96012-05).

c. Conclusions

The inspectors reviewed a DR involving the failure of safety-related vital battery 1-I to meet one acceptance criterion during performance of the last service test. The inspectors agreed that the subsequent Operability Evaluation was reasonable, and that the requirement of the TS was met. An Inspection Follow-up Item was established to ensure review of certain procedure enhancements and future test results.

E3 Engineering Procedures and Documentation

E3.1 Modification of the CCW Surge Tank Support

a. Inspection Scope (37550)

From a summary of DRs assigned to structural engineers, the inspectors selected randomly DR N-96-2095 for review. Resolution of this DR involved a modification, and the inspectors reviewed that modification as well. The upper level requirement applicable to review of the DR was 10 CFR 50, Appendix B, Criterion XVI, Corrective Action. The requirement applicable to review of the modification was 10 CFR 50, Appendix B, Criterion III, Design Control. The DR and modification involved 10 CFR 50, Appendix A, Criterion 2, Design Bases for Protection Against Natural Phenomena.

b. Observations and Findings

DCN 96-014 reinforced the supports for the CCW surge tank to resolve concerns about the seismic integrity of the tank. From a design perspective, this DCN met the requirements, and resolved the DR. However, inspection of the completed work by the inspectors identified that the dimensional tolerances indicated on Drawing No. N-96014-3-S-001, Sheets 1 through 8 were exceeded. For example, the dimensions for certain anchor bolts carried a tolerance of minus 0 inches, plus 2 inches. Actual dimensions varied from the specified minus tolerance by 1½ inches and the plus tolerance by 1¼ inches. Also, the inspectors observed that one pipe support for a 3-inch diameter pipe was attached to the original un-reinforced portion of the tank support and that pipe support did not show on the modification drawings. DCN 96-014 was signed as completed on October 11, 1996, by the Operational Readiness Review. The licensee did not produce any documentation indicating that the discrepancies were noted or evaluated. The inspection finding described above constitutes a violation of 10 CFR 50, Appendix B, Criterion V, which requires that activities affecting quality shall be accomplished in accordance with documented drawings. The matter was identified as a Violation (50-338, 339/96012-06).

c. Conclusions

Review of a 1996 modification package (DCN) to reinforce the supports for the CCW surge tank to resolve concerns about the seismic integrity of the tank indicated that the design was correctly accomplished, but that the actual installation did not meet specified dimensional tolerances. Also, a pipe support was installed at a point where no support was shown on the drawings. A Notice of Violation was issued.

E3.2 Set Points for Molded-Case Circuit Breakers

a. Inspection Scope (37550)

The inspectors reviewed the control of set points for safety-related molded-case circuit breakers. This inspection topic was chosen, in part, because lack of controls on molded-case circuit breaker set points at other sites resulted in breakers inadvertently tripping during starting.

Relevant requirements for this inspection topic included 10 CFR 50, Appendix B, Criterion III, Design Control and Criterion V, Instructions, Procedures and Drawings. Guidance was provided by IEEE std 741-1986, IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations, Sections 5.1.3.1 and 6.1.4. Also, the inspectors referenced IEEE std 242-1986, IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems, Section 9.3.3.5, Instantaneous Settings.

b. Observations and Findings

The great majority of motor control centers at the site were furnished by Klockner-Moeller Co. The original circuit breakers had an adjustable inverse-time (thermal) element and a fixed instantaneous (magnetic) element. Many of the original circuit breakers had been replaced with a newer style due to a problem described in a 10 CFR 21 report (NRC Information Notice 93-22) and other age related failures. The newer style circuit breakers had an adjustable thermal element and an adjustable magnetic element.

The frame size and thermal element set point were shown on the one-line diagrams which depicted the 480 V motor control centers and loads. The magnetic set point was not shown on the drawings. The inspectors inquired as to how the set points for the magnetic element were established and controlled. The engineers responded that the craftsperson or technician installing a new breaker would know (i.e., skill of the craft) to make the settings the same as the one being replaced. The validity of this method was based on the philosophy that operating experience has shown the set points to be appropriate (i.e., no inadvertent tripping). The licensee pointed out that the magnetic set point, although fixed, was indicated on the nameplate of the original style circuit breakers. This was confirmed by the inspectors.

The licensee stated that the method by which a craftsperson or technician set up a new breaker was not explicitly covered in any procedure nor was it covered in any required training.

The inspectors reviewed Electrical Maintenance Procedure 0-EPM-0304-01, Testing Non-Containment-Isolation 480 V Breaker Assemblies, Revision 18. The inspectors noted that both the thermal and magnetic elements were tested. The magnetic element was checked to carry 70 percent of the set point value and trip at 125 to 140 percent of the set point value.

The inspectors examined six safety-related in-service circuit breakers, four original style and two newer style. The examination confirmed that the thermal element set point matched the one-line diagram. The magnetic element set points were recorded from the nameplate for the original style and from the setting dial for the newer style. The inspectors concluded that the set points for the magnetic element of the newer style breakers matched what would have been the fixed setting on the original style breaker having that thermal set point. The inspectors also compared the set points to the motor full load current and locked rotor current of the respective loads, and concluded that the set points were correct. The circuit breakers inspected were:

Compartment No.	Load	Breaker Style
2H1-2N-J3	MOV-2890C	original
2H1-2N-B3	2-CV-P-03A	original
2H1-2N-A4	2-HC-F-01	original
2H1-2S-C1	MOV-2720A	original
2H1-2S-C3	2-RS-P-3A	newer
2H1-2S-K4	2-HV-F-40A	newer

c. Conclusions

The licensee's test procedure for the molded-case circuit breakers was good, in that, it checked that the magnetic element tripped within a specified band around the set point. The controls for establishing and verifying the set point for the magnetic element were minimal. There was no set point document, drawing or specific instructions to help ensure proper set points were maintained. Based on a sample of two newer circuit breakers, the licensee's informal method resulted in correct settings. Nevertheless, the lack of formal controls on the set point for the magnetic element of molded-case circuit breakers was a weakness in the licensee's design control program, in that, the continued use of the informal method has increased the probability to result in incorrect set points. Whether the lack of formal controls represents a violation of NRC requirements in the area of design control or procedures is under further review by the NRC. The issue is identified as Unresolved Item (50-338, 339/96012-07).

E7 Quality Assurance in Engineering Activities

E7.1 Safety Committees

a. Inspection Scope (37550)

The inspectors reviewed the functioning of the two safety and oversight committees.

b. Observations and Findings

The inspectors reviewed some of the minutes of past meetings for the two safety and oversight committees and attended meetings of both of these committees. During the meeting of the Station Nuclear Safety and Operating Committee, the inspectors observed the review and approval of a temporary modification (required 10 CFR 50.59 screening), the approval of a special report on SW/ground water levels, the approval of eight procedure changes, and a change to a curve for nuclear fuel burnup rate.

A review was made of the last three MSRC meeting minutes, and it was noted by the inspectors that the committee was performing the functions delineated in the committee's procedures. This committee is the off-site oversight review committee. The inspectors attended a part of the meeting held at the North Anna Station on November 20, 1996. Three root cause evaluations (two involving unit trips) and various improvement initiatives for North Anna were discussed. The items were well presented and resulted in lively and productive discussions.

c. Conclusions

The safety and oversight committees were conducting their duties in an efficient and productive manner.

E7.2 Engineering Self-Assessment

a. Inspection Scope (40500)

A review of the licensee's engineering self assessment program was conducted by the inspectors to see if strengths and weaknesses were identified and if corrective actions were initiated for the weaknesses.

b. Observations and Findings

The inspectors reviewed the procedure for engineering self-assessments, NASES (North Anna Station Engineering Services)-1.12, Controlling Procedure for Engineering Self Assessment, Revision 0. Site Engineering Services performs a wide variety of services and the procedure is intended as a guideline only. The Engineering Review Board is responsible for controlling the scope of the proposed self assessments. Since this program was recently implemented, very few self-assessments were completed. The inspectors reviewed two partially completed

assessments, one on System Engineering and one on Post Maintenance Testing. The inspectors noted during the MSRC meeting that the Nuclear Oversight Group had identified that an upper tier document which addresses self-assessment (PAP-0104) was under modification because it did not spell out management's expectations for self-assessments.

c. Conclusions

The engineering self-assessment program was only recently defined, and very few self-assessments were completed.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) URI 50-339/96009-03, review anomalies in large bore snubber test data. The inspectors completed their review of large bore snubber functional test data. No other anomalies were identified. Snubbers with anomalies in test data were either repaired, replaced or subsequently satisfactorily tested.

The licensee plans to procure replacement large bore snubbers from a different vendor prior to the next refueling outage, i.e., the Unit 1 refueling outage scheduled to begin in May 1997. Testing of the new snubbers will use different test equipment and test techniques. An Inspection Followup Item is opened to review functional tests that will be conducted on either new or old snubbers in the upcoming Unit 1 refueling outage (50-338, 339/96012-08).

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 As Low As Reasonably Achievable (ALARA)

a. Inspection Scope (83750)

The inspectors reviewed licensee records of personnel radiation exposure and discussed ALARA program details, implementation and goals with the licensee. The site collective dose and individual exposures were compared to licensee established ALARA goals and occupational dose limits, respectively.

b. Observations and Findings

The licensee provided the inspectors with reports of personnel radiation exposure for calendar year (CY) 1995 and the first three quarters of 1996. Those reports indicated that the licensee had initially established an ALARA goal of 459 person-roentgen equivalent man (rem) for the 1995 site collective dose. That goal included a projected 425 person-rem for the Unit 2 SG Replacement Project and related outage activities. After the actual exposure for that outage was determined to have been 340 person-rem, the annual goal was reduced from the initial

goal of 459 person-rem to 367 person-rem. The final collective dose for CY 1995 was 359 person-rem. The annual goal for the 1996 site collective dose was established at 364 person-rem. That goal included a projected 188 person-rem for the Unit 1 Refueling Outage (RFO) and 149 person-rem for the Unit 2 RFO. The actual exposures for those outages were 184 and 117 person-rem, respectively. The actual exposure during the first three quarters of 1996 was 305 person-rem. The licensee indicated that, given the current work schedule for the remainder of the year, the 1996 annual goal was not expected to be exceeded. The inspectors determined that the licensee was meeting their established ALARA goals. The licensee's personnel radiation exposure records also indicated that the maximum individual exposures for 1995 and the first three quarters of 1996 were less than 2.6 rem and within the limits specified in 10 CFR 20.1201(a).

c. Conclusions

Based on the above reviews and observations, the inspectors concluded that the licensee was closely monitoring collective and individual radiation dose exposure, and that the licensee was meeting established ALARA goals and occupational dose limits.

R1.2 Water Chemistry Controls

a. Inspection Scope (84750)

The inspectors reviewed implementation of selected elements of the licensee's water chemistry control program for monitoring primary and secondary water quality. The review included examination of program guidance and implementing procedures, and analytical results for selected chemistry parameters. Those procedures and data were compared to specific ard programmatic requirements, i.e., the TSs required the licensee to monitor primary coolant for specific chemistry parameters and to implement a program for monitoring secondary water chemistry to inhibit steam generator tube degradation.

b. Observations and Findings

The inspectors reviewed VPAP-2201, Nuclear Plant Chemistry Program, Revision 1, and determined that it included provisions for sampling and analyzing reactor coolant at the prescribed frequency for the parameters required to be monitored by the TSs. That procedure also included provisions for monitoring primary and secondary water quality based on established industry guidelines and standards. Although the licensee's procedure did not specifically indicate that their program included implementation of the Electric Power Research Institute (EPRI) guidelines for Pressurized Water Reactor (PWR) primary and secondary water chemistry, the inspectors used those guides as references for evaluating the effectiveness of the licensee's program. The inspectors noted that VPAP-2201 listed the sampling frequency and typical values for each parameter to be monitored. Action levels applicable to various operational modes were given where appropriate. Guidance was also

provided for actions to be taken if analytical results exceeded prescribed limits. The inspectors determined that the above guidance and procedures were consistent with the applicable TS requirements and EPRI guidelines.

The inspectors also reviewed trend plots and records of analytical results for selected parameters generated during the period April through October 1996. The parameters selected included dissolved oxygen, chloride, fluoride, sulfate, and dose equivalent iodine-131 in reactor coolant and dissolved oxygen, iron, copper, sodium, hydrazine, silica, and sulfate in secondary systems. Those parameters were maintained well within the relevant TS limits and within the EPRI guidelines for power operations.

c. Conclusions

Based on the above reviews, the inspectors concluded that the licensee's water chemistry control program for monitoring primary and secondary water quality had been implemented in accordance with the TS requirements and the EPRI guidelines for PWR water chemistry.

R1.3 Transportation of Radioactive Materials

a. Inspection Scope (TI 2515/133)

The inspectors evaluated the licensee's transportation of radioactive materials programs for implementing the revised Department of Transportation and NRC transportation regulations for shipment of radioactive materials as required by 10 CFR 71.5 and 49 CFR Parts 170 through 179.

b. Observations and Findings

The inspectors reviewed the licensee's actions regarding a shipment of licensed material which exceeded the limit for specific activity delineated in the Certificate of Compliance (CoC) for the cask used to transport the material. On October 31, 1996, the licensee's radioactive waste shipping personnel were reviewing shipping records and found that the most recent revision of the CoC for the shipping cask had not been properly filed in the cask's reference notebook. The licensee maintains a cask reference notebook for each NRC licensed cask used by the facility and licensee personnel use those notebooks when preparing radioactive material shipments. On April 16, 1996, CoC No. 6601, Revision 25, for the Model No. CNS 8-120A package was issued by the NRC Office of Nuclear Material Safety and Safeguards, and was received at the site on or about May 6, 1996. Further review of shipping records by the licensee on November 5, 1996, revealed that on July 9, 1996, the cask was used for a shipment (No. 96-05) of radioactive waste to the Chem-Nuclear waste disposal site at Barnwell, SC. 10 CFR 71.12(c)(2) authorizes any licensee to transport licensed material in a package for which a CoC has been issued by the NRC, provided that the licensee complies with the terms and conditions of the CoC. Condition 5

(b)(1)(i) of CoC No. 6601 specifies that the average concentration of the package contents shall not exceed 0.3 millicurie per gram of radionuclides for which the A_2 quantity in Appendix A of 10 CFR Part 71 is more than 1 curie. Contrary to that condition, the average concentration of radionuclides for which the A_2 quantity in Appendix A of 10 CFR Part 71 is more than 1 curie for the package contents of Shipment No 96-05 was 0.495 millicurie per gram. On November 6, 1996, the licensee called the NRC Region II Office to report the violation of 10 CFR 71.12(c)(2) and indicated that shipments of radioactive waste for disposal had been immediately suspended and that an investigation had been initiated. During this inspection the licensee's report documenting that investigation was reviewed by the inspectors. That report indicated that the root cause of the violation was poor document control of the CoC. Other contributing factors identified by the investigation were incomplete procedural checklist guidance for preparing and inspecting shipments, and incomplete guidance for application of the software used as an aid in shipping preparations. The report also included recommendations for corrective actions to prevent recurrence of this event. Those recommendations were: delineating the CoC requirements in the cask shipping checklists used by the personnel who prepare the casks for shipment and in the checklists used by Quality Assurance/Quality Control personnel who inspect the shipments; filing the CoCs in their proper location in the cask reference notebooks immediately upon receipt; and establishing a required reading program and signoff system for revised shipping procedures. The licensee indicated that those corrective actions would be implemented prior to the resumption of radioactive waste shipments planned for December 6, 1996. This licensee-identified violation for failure to comply with the conditions of the Certificate of Compliance for an NRC approved shipping package is being treated as an NCV, consistent with Section VII.B.1 of the Enforcement Policy (50-338, 339/96012-09).

c. Conclusions

Based on the above reviews, the inspectors determined that the licensee's efforts in identifying and initiating corrective actions for the violation were adequate. One NCV was identified by the licensee for failure to comply with the conditions of the CoC for an NRC-approved shipping package.

R8 Miscellaneous RC&P Issues (92904)

R8.1 (Closed) Violation 50-338, 339/96007-04, failure to label a container of licensed material. During a previous inspection, the inspectors observed a trailer in the outside protected area which contained contaminated scaffolding but did not bear a completed radioactive material label. The licensee's response to the violation indicated that the container was relabeled properly, that other containers were inspected to ensure proper labeling, and that this event was discussed with Health Physics (HP) personnel during departmental meetings and subsequent training sessions. During this inspection no further

examples of improper container labeling were observed by the inspectors. Licensee records of attendance and discussion topics for the referenced departmental meetings and training sessions were also reviewed by the inspectors. The inspectors concluded that the licensee's response dated October 8, 1996, and the corrective actions were appropriate and had been adequately implemented.

R8.2 (Closed) Violation 50-338, 339/96007-05, failure to follow procedures for minimizing the potential spread of radioactivity to unrestricted areas. During a previous inspection it was determined that a individual was released from the protected area with clothing which exceeded the licensee's procedural release limit for radioactive contamination. The licensee's response to the violation indicated that the contaminated clothing had been retrieved, that HP procedures had been revised to provide additional guidance to HP personnel for dealing with personnel that alarm portal or personal contamination monitors, and that HP personnel were trained in those procedural changes. During this inspection the licensee's revised HP procedures were reviewed and found to include specific guidance for responding to alarms by portal and personal contamination monitors. The inspectors also reviewed the licensee's records for training on those procedure changes. The inspectors concluded that the licensee's response dated October 8, 1996, and the corrective actions were appropriate and had been adequately implemented.

S1 Conduct of Security and Safeguards Activities (71750)

On November 11, the inspectors toured various plant security systems with a security team leader. Specifically, selected protection area perimeter fencing and isolation zones, Central and Secondary Alarm Stations, various vital area accesses, and the secondary power supply were inspected. The inspectors observed that the protected area perimeter fencing was in good condition with no openings. Isolation zones were free of objects and clearly marked. The inspectors observed that the Central and Secondary Alarm Stations were properly manned and that intrusion systems were functioning properly. Various vital area accesses and boundaries were inspected and found to be in good working order. The inspectors concluded that security systems were in good working order and security manning was appropriate.

VI. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on December 20, 1996. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

W. Anthes, Superintendent, Outage and Planning
B. Foster, Superintendent Station Engineering
E. Grecheck, Assistant Station Manager, Operations and Maintenance
J. Hayes, Superintendent, Operations
D. Heacock, Assistant Station Manager, Nuclear Safety and Licensing
M. Kansler, Vice President, Nuclear Operations
P. Kemp, Supervisor, Licensing
T. Maddy, Superintendent, Security
W. Matthews, Station Manager
M. McCarthy, Director, Nuclear Oversight
D. Roberts, Supervisor, Station Nuclear Safety
H. Royal, Superintendent, Nuclear Training
D. Schappell, Superintendent, Site Services
R. Shears, Superintendent, Maintenance
A. Stafford, Superintendent, Radiological Protection

NRC

B. Buckley, Project Manager
F. Reinhart, Acting Project Director

INSPECTION PROCEDURES USED

IP 37550: Engineering

IP 37551: Onsite Engineering

IP 40500: Effectiveness of Licensing Controls in Identifying, Resolving, and Preventing Problems

IP 62707: Maintenance Observations

IP 71707: Plant Operations

IP 71714: Cold Weather Preparations

IP 71750: Plant Support Activities

IP 83750: Occupational Radiation Exposure

IP 84750: Radioactive Waste Treatment, and Effluent and Environmental Monitoring

IP 92700: Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities

IP 92901: Followup - Plant Operations

IP 92902: Followup - Maintenance

IP 92903: Followup - Engineering

IP 92904: Followup - Plant Support

IP 93702: Prompt Onsite Response to Events at Operating Power Reactors

TI 2515/133: Implementation of Revised 49 CFR Parts 100-179 and 10 CFR Part 71

ITEMS OPENED AND CLOSED

Opened

50-339/96012-01	VIO	Failure to Meet 10 CFR 50 Appendix B Criterion III Requirements for Design Control of Safeguards Area Walls (Section O8.1)
50-338, 339/96012-02	NCV	Inadequate Surveillance Test Procedure for Placing Reactor Trip Breakers in Service (Section M8.3)
50-339/96012-03	VIO	Inadequate Design Controls Contribute to Non-Conforming Bolts in Column Flange for Service Water Pump 2-SW-P-1A (Section E1.1)
50-338, 339/96012-04	NCV	Improper Deletion of DHR Isolation Valve Inservice Testing (Section E1.3)
50-338, 339/96012-05	IFI	Battery Service Test and Rating of Diesel Generator Breaker Close Coil (Section E2.6)
50-338, 339/96012-06	VIO	Component Cooling Surge Tank 1-CC-TK-1 Support Structure not Installed Per Drawings (Section E3.1)
50-338, 339/96012-07	URI	Control of Set Points for Molded-Case Circuit Breakers (Section E3.2)
50-338, 339/96012-08	IFI	Review Functional Tests That Will Be Conducted On New or Old Snubbers During Unit 1 Refueling Outage (Section E8.1)
50-338, 339/96012-09	NCV	Failure to Comply With Conditions of CoC for NRC Approved Shipping Package (Section R1.3)

Closed

50-338/95015-01	VIO	Failure To Follow Procedures For Properly Controlling Safeguards Area Ventilation System (SAVS) Maintenance (Section M8.1)
50-339/95020-02	VIO	Failure to Comply With 3.6.1.3 for Air Lock Outer Door Rendered Inoperable by Open Test Connection (Section M8.2)
50-339/96001	LER	Potential Unfiltered Release Path From the Quench Spray Pump House to the Environment (Section O8.1)
50-339/96003	LER	Automatic Reactor Trip Resulting From Main Generator Stator Coil Failure Due to Personnel Error (Section M8.5)
50-339/96004-01	URI	Review Significance of Safeguard Area Ventilation Not Meeting Design Basis (Section O8.1)
50-338, 339/96007-04	VIO	Failure to Label a Container of Licensed Material (Section R8.1)
50-338, 339/96007-05	VIO	Failure to Follow Procedures for Minimizing the Potential Spread of Radioactivity to Unrestricted Areas (Section R8.2)
50-338, 339/96009	LER	Reactor Trip Bypass Breaker Missed Surveillance Due to Inadequate Surveillance Test Procedure (Section M8.3)
50-339/96009-03	URI	Review Anomalies in Large Bore Snubber Test Data (Section E8.1)
50-338/96010	LER	Automatic Reactor Trip Due to Failure of a Generator Negative Phase Sequence Relay (Section M8.4)
50-338, 339/96012-02	NCV	Inadequate Surveillance Test Procedure for Placing Reactor Trip Breakers in Service (Section M8.3)
50-338, 339/96012-04	NCV	Improper Deletion of DHR Isolation Valve Inservice Testing (Section E1.3)
50-338, 339/96012-09	NCV	Failure to Comply With Conditions of CoC for NRC Approved Shipping Package (Section R1.3)