U.S. NUCLEAR REGULATORY COMMISSION

REGION II

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Licensee:	Virginia Electric and Power Company (VEPCO)
Facility:	North Anna Power Station, Units 1 & 2
Location:	1022 Haley Drive Mineral, Virginia 23117
Dates:	January 12 through February 22, 1997
Inspectors:	R. McWhorter, Senior Resident Inspector R. Gibbs, Resident Inspector R. Musser, Senior Resident Inspector, Surry (Section M1.3) L. Garner, Project Engineer (Sections O1.2, M1.1, M1.4, M1.5, and E8.2) E. Girard, Reactor Inspector (Sections E1.1 and E8.1)
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ENCLOSURE 2

EXECUTIVE SUMMARY

North Anna Power Station, Units 1 & 2
NRC Inspection Report Nos. 50-338/97-01, 50-339/97-01

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection; in addition, it includes the results of announced inspections by regional specialists and a regional project engineer.

Operations

- Daily operations were generally conducted in accordance with regulatory requirements and plant procedures. Good equipment material conditions were evident by problem-free plant operation during the inspection period (Section O1.1).
- An operator did not fully understand abnormal control board pressurizer spray line temperature indications (Section O1.2).
- A Unit 1 Low Head Safety Injection (LHSI) System walkdown revealed that the system was properly aligned and maintained (Section O2.1).
- A Violation (VIO) was identified for failures to meet 10 CFR 70.24 requirements for criticality monitoring for new fuel storage since the issuance of the facility operating license. The licensee recently submitted a request to the NRC to be exempted from these requirements in the future (Section O2.2).

Maintenance

- The observed outside Recirculation Spray (RS) pump work activities were performed in a quality manner. Two examples were observed in which questioning attitude could have been enhanced, and one example was noted in which a good questioning attitude was displayed (Section M1.1).
- The licensee collected equivalent data to meet Technical Specification (TS) surveillance requirements when a weekly surveillance test was not performed as planned on various Unit 2 batteries (Section M1.2).
- Four surveillance activities were successfully performed in accordance with test procedures (Sections M1.3, M1.5, M1.6 and M1.7).
- Post maintenance testing of a steam flow protection channel demonstrated that the channel functioned properly (Section M1.4).

Engineering

- A Non-cited Violation (NCV) was identified, involving a failure to follow a procedure step relating to the uncertainty in Motor Operated Valve (MOV) opening thrust measurements (Section E1.1.b.2).

- The inspectors raised issues regarding the susceptibility of the licensee's quench spray valves to pressure locking, possible thermal binding of the pressurizer power operated relief valve block valves, and whether residual heat removal system isolation valves should be evaluated for pressure locking. The NRC will address these issues in a safety evaluation of the licensee's response to Generic Letter (GL) 95-07. An Inspection Followup Item (IFI) was identified to track these issues (Section E1.1.b.6).

- Strengths in the MOV program were identified which included technically capable and dedicated personnel, effective efforts to organize and develop data to resolve the final concerns developed during the NRC review, and good diagnostic test assessments (Section E1.1.b.9).

- The licensee generally met the intent of GL 89-10 by verifying MOV design-basis capabilities. However, the licensee did not have quantitative dynamic test data to support the reliability of the methods used to determine the torque requirements for butterfly valves and this was considered a weakness. The licensee committed to resolve this weakness through differential pressure testing and/or application of the Electric Power Research Institute (EPRI) Performance Prediction Model to a sample of their butterfly valves. An IFI was identified to track completion of the commitment actions (Section E1.1.c).

- Following a surveillance test failure, the licensee adequately evaluated out-of-specification data prior to returning the Unit 2 B outside recirculation spray pump to operable status. However, the Engineering Transmittal (ET) did not clearly document all factors considered in this evaluation (Section E2.1).

- As a result of increased licensee awareness and initiatives, numerous Deviation Reports (DRs) were submitted concerning Updated Final Safety Analysis Report (UFSAR) discrepancies. The inspectors reviewed the DRs and verified that there were no significant safety concerns or unreviewed safety questions (Section E7.1).

- An Unresolved Item (URI) was identified to track NRC reviews of the DC power system and charging pump start logic for compliance with regulatory requirements (Section E8.2).

Plant Support

- Radiation areas and high radiation areas were properly posted and were locked when required by 10 CFR 20. Independent radiation measurements found that the licensee's postings and survey results were conservative (Section R1.1).

Report Details

Summary of Plant Status

Unit 1 and 2 operated the entire inspection period at or near full power.

I. Operations

O1 Conduct of Operations

O1.1 Daily Plant Status Reviews (71707, 40500)

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 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. DRs were reviewed to assure 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. Good equipment material conditions were evident by problem-free plant operation during the inspection period.

O1.2 Operator Knowledge of Control Board Indications (71707)

On January 22, during a control board walkdown, the inspectors observed that the Unit 1 pressurizer B spray line temperature was approximately 75°F lower than the A spray line temperature. The spray valve indicating lights associated with the spray valve position controllers revealed that the B spray valve was cracked open and the A spray valve was closed. Thus, the temperature indications were reversed from the expected values. This was discussed with the Unit 1 control operator. The operator was aware that there was a work request outstanding on the spray valve controllers and indicating lights but was unable to explain how the indicated temperatures related to the problem described on the work request. The operator was not able to address if the temperature mismatch was indicative of a new problem or was one that was already addressed. Further review indicated that the existing work request addressed the problem. This was discussed with the Operations Superintendent. The inspectors concluded that, in this instance, the operator did not fully understand the abnormal indications on the control board.

O2 Operational Status of Facilities and Equipment

O2.1 LHSI System Walkdown (71707)

During the weeks of January 13 and February 18, the inspectors performed a walkdown of the Unit 1 LHSI system in the Safeguards Building and

Quench Spray Pump House. The UFSAR, TS, system drawings, and procedure 1-OP-7.1A, Valve Checkoff - Low Head Safety Injection, Revision 15, were reviewed and used as references for the walkdown. The inspectors found that all system components were aligned in accordance with applicable requirements and were in good material condition. The inspectors concluded that the Unit 1 LHSI system was properly aligned and maintained.

O2.2 Criticality Accident Monitoring

a. Inspection Scope (71707)

On February 12 and 13, the inspectors reviewed plant systems and licensee procedures for criticality accident monitoring for new fuel stored on site prior to placement in the spent fuel pool. The inspectors reviewed the licensee's compliance with UFSAR, TS, and 10 CFR 70.24 requirements.

UFSAR Sections 3.1.53, 4.3.2.7, 9.1.1 and 12.1.4 described plant features supporting the storage of nuclear fuel prior to placement in the spent fuel pool. These features included new fuel storage rack design to prevent inadvertent criticality, new fuel handling equipment design, radiation monitoring system design, and new fuel storage rack seismic qualifications. UFSAR descriptions also stated that analyses were completed to demonstrate that if the new fuel storage rack area were filled with aqueous foam, a k_{eff} less than or equal to 0.9 would be maintained. TS 3.3.3.1 and TS Table 3.3-6 delineated the requirement that one area radiation monitor in the new fuel storage area, also described as a criticality monitor, be operable when fuel was present in the Fuel Building.

10 CFR 70.24 requirements for criticality accidents applied to the possession of special nuclear material (new reactor fuel) from removal from transportation containers until handled or stored beneath water shielding (placed in the spent fuel pool). For plants licensed after December 6, 1974, these requirements included:

- A gamma- or neutron-sensitive monitoring system which would energize alarm signals if accidental criticality occurs for each area where material was handled, used or stored [70.24(a)].

- A monitoring system of specific design sensitivity (capable of detecting a criticality that produced an absorbed dose of 20 rads combined neutron and gamma radiation at an unshielded distance of two meters from the reacting material) [70.24(a)(1)].

- A monitoring system consisting of two detectors [70.24(a)(1)].

- Emergency procedures for personnel evacuation upon system alarm [70.24(a)(3)].

- Procedures for the conduct of drills to familiarize personnel with the evacuation plan [70.24(a)(3)],

- Designation of responsible individuals for determining the cause of the alarm [70.24(a)(3)], and

- Placement of radiation survey instruments in accessible locations for use in an emergency [70.24(a)(3)].

b. Observations and Findings

The inspectors found that the UFSAR sections discussed above accurately described existing plant features. The new fuel storage rack area was consistent with the UFSAR description and was in good material condition. One radiation monitor (1-RMS-RM-152) was permanently installed in the new fuel storage area, and the inspectors verified that the monitor was operable in accordance with TS requirements. A second radiation monitor (1-RMS-RM-153) was attached to the fuel building bridge crane which was parked over the new fuel storage area when not in use, and the inspectors verified that the monitor was operable. Both radiation monitors used gamma-sensitive geiger-mueller detectors, had radiation level meters locally and in the control room, and had audible and visible alarm indications locally and in the control room. The inspectors concluded that the UFSAR and TS requirements for criticality accident-related features were met by the licensee.

The inspectors reviewed the facility operating licenses and determined that the licensee did not have an exemption from 10 CFR 70.24 requirements at the time of the inspection. The inspectors noted that on January 28, 1997, the licensee submitted a request for exemption from the 10 CFR 70.24 requirements to the NRC. The letter stated that the exemptions existed during facility construction, but were not incorporated into the 10 CFR 50 operating licenses when they were issued. Concerning the licensee's compliance with 10 CFR 70.24 requirements listed above, the inspectors found:

- One gamma-sensitive radiation monitoring system (1-RMS-RM-152) capable of detecting high radiation levels and actuating an alarm was installed in the new fuel storage area. No radiation monitoring system was installed in the new fuel receiving area where new fuel was unloaded from transportation containers for movement to the storage area. The new fuel storage area radiation monitor was located in the same end of the Fuel Building, but was on a level above the receiving area and shielded from the receiving area by concrete structures.

- The ability of the new fuel storage area radiation monitor to meet the requirements for monitor sensitivity could not be demonstrated. The licensee had no readily available information demonstrating that the monitor had been analyzed for compliance with the 10 CFR 70.24 sensitivity requirements.

- The new fuel storage area radiation monitor consisted of only one detector. Although a second radiation monitor and detector on the fuel building bridge crane was frequently present in the area, it was periodically moved away from the new fuel rack area during fuel handling activities and did not continuously meet the requirement for a second detector.

- Emergency procedures for evacuating the area upon a radiation monitor alarm were available for use, but did not specifically require evacuation of the area upon receipt of an alarm. The inspectors reviewed abnormal operating procedure 0-AP-5.1, Common Unit Radiation Monitoring System, Revision 9, and found that it contained direction for control room operators to respond to an alarm on the new el storage area radiation monitor. The procedure directed operators to investigate the validity of alarms, notify Health Physics (HP) personnel to survey the area, stop fuel movements in the area, verify any automatic ventilation system lineup changes, and notify personnel in the area. Additionally, if the alarm occurred during fuel handling, operators were directed to enter other abnormal procedures for fuel handling accidents including 0-AP-30, Fuel Failure During Handling, Revision 6. Procedure 0-AP-30 contained direction to evacuate the area after placing the fuel in a safe location. However, 0-AP-5.1 did not contain a step directing area evacuation upon receipt of any valid alarm during situations other than fuel handling.

- The licensee did not conduct drills specifically to familiarize personnel with evacuation plans. Licensee managers informed the inspectors that drills had not been conducted to meet this requirement.

- Licensee personnel were designated to determine the cause of system alarms. As discussed above, procedure 0-AP-5.1 required control room operators to respond to alarms on the new fuel storage area radiation monitor and take actions which were appropriate to determine the cause of the alarm.

- Numerous radiation survey instruments of various types were available for use by emergency response personnel and were located in an accessible storage area near the HP shift office.

The inspectors reviewed the significance of the above non-compliance items and concluded that the safety significance was low.

The inspectors concluded that the licensee failed to meet 10 CFR 70.24 requirements for criticality accidents, and no exemption from the requirements had been granted by the NRC. Specifically, contrary to 10 CFR 70.24 requirements, radiation monitors installed in fuel handling and storage locations did not provide full coverage for the area used for new fuel receipt and consisted of only one permanently installed detector. Additionally, emergency procedures did not clearly direct area evacuation upon receipt of any alarm, and drills were not conducted to familiarize personnel with evacuation plans. This condition had existed since issuance of the facility operating license until the inspection on February 13, 1997. During that time, the new fuel receiving and storage areas were used to handle, use and store new fuel assemblies on a regular basis prior to each unit refueling outage. This is identified as a violation of 10 CFR 70.24 requirements (50-338, 339/97001-01). This violation will be considered as an engineering violation.

c. Conclusion

A violation was identified for failures to meet 10 CFR 70.24 requirements for criticality monitoring for new fuel storage since the issuance of the facility operating license. The licensee recently submitted a request to the NRC to be exempted from these requirements in the future.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Unit 2 Outside RS Pump Maintenance

a. Scope (62707)

On January 21 through 23, the inspectors observed corrective maintenance associated with the Unit 2 A RS pump, 2-RS-P-2A. Observed work included: correcting a leak on the seal head tank accumulator per 0-MCM-0650-01, Disassembly, Inspection, and Repair of the Outside Recirculation Spray Pump Seal Accumulator, Revision 0-P1; replacement of a seal system water fill valve per WO 00358048-01; and, repair fittings on mechanical seal system water fill line per WO 00357986-02.

b. Observations and Findings

Much of the work observed was performed as skill of the craft. The work performed was of high quality. Procedures were followed when applicable. Two areas in which questioning attitude could be enhanced were discussed with the Maintenance Superintendent. The first involved the proper orientation of the replacement fill valve. Maintenance personnel noted that the installed valve was oriented with the indicated flow direction opposite the direction of flow when the valve is opened to fill the system. The individuals discussed this among themselves and decided to install the replacement valve in the as-found orientation. When questioned by the inspectors, the craftsmen contacted Operations who referred to a flow diagram and informed maintenance that the valve should be turned around from the as-found configuration. Engineering subsequently reviewed the issue and determined that based upon operating conditions it was more important for the valve to be installed in an orientation backwards from its indicated flow direction. This would allow the valve to better seal to prevent air leakage into the system which is under a vacuum during normal operation. The second instance involved an observation by the inspectors that the two mounting bracket bolts were not lubricated prior to torquing and had no washers installed. These bolts were installed through a bracket and then screwed into the accumulator head flange, thus compressing the accumulator head gasket. The other bolts that attached the accumulator head were lubricated prior to torquing. The licensee indicated that they would evaluate these observations. In both instances, maintenance personnel had the opportunity to identify that deviations from normal practices existed and to pursue these deviations with the appropriate organization. The inspectors observed that maintenance personnel questioned if a small bore pipe was correctly supported and initiated a DR to resolve the observation.

c. Conclusions

The observed RS pump work activities were performed in a quality manner. Two examples were observed in which questioning attitude could have been enhanced, and one example was noted in which a good questioning attitude was displayed.

M1.2 Potential Missed Battery Surveillances Review

a. Inspection Scope (61726)

On January 30, licensee engineers and maintenance personnel submitted DR N-97-274 documenting failures to perform weekly surveillance procedures for several Unit 2 batteries during a recent refueling outage. The DR stated that reviews of other tests and maintenance procedures performed during the outage demonstrated that the surveillance requirements of TS 4.8.1.2 and 4.8.2.2.2 were met during the period the weekly procedures were not performed. During the weeks of February 3 and 10, the inspectors reviewed the problem to verify the validity of the licensee's conclusions that TS surveillance requirements were met.

b. Observations and Findings

The inspectors found that test 2-PT-85, DC Distribution System, Revision 30, was missed for various Unit 2 batteries during a three-week period of September 1996. The DR stated that during this period procedures 2-PT-86A/B, DC Distribution Systems H/J Bus, Revision 18, and maintenance procedures for battery equalization charges and test discharges recorded equivalent data. The inspectors reviewed the data recorded by these procedures and compared them to 2-PT-85. The inspectors found that the other procedures did record the same data as 2-PT-85 and contained data required to be checked weekly by TS surveillance requirements.

The inspectors reviewed record copies of the procedures performed during the period in question. With the assistance of system engineers, the inspectors found that for all cases where 2-PT-85 was not performed, the other procedures were performed and included records of the information which would have been recorded by 2-PT-85.

The inspectors questioned why 2-PT-85 was not performed as planned during the three-week outage period. Licensee engineers informed the inspectors that the test was the responsibility of the electrical maintenance department. During the unit outage, when equalization charges and test discharges on the batteries were being performed, maintenance personnel failed to perform the weekly surveillance test as scheduled. At the inspection period's end, further investigations and planning of corrective actions were continuing.

c. Conclusions

The inspectors concluded that the licensee collected equivalent data to meet TS surveillance requirements when a weekly surveillance test was not performed as planned on various Unit 2 batteries.

M1.3 Quadrant Power Tilt Ratio (QPTR) Surveillance Test (61726)

On January 29, the inspectors observed the performance of 1-PT-23, Quadrant Power Tilt Ratio (QPTR) Determination, Revision 19, for Unit 1. This test was performed to satisfy TS Surveillance requirement 4.2.4.1.a. to demonstrate that the core QPTR did not exceed 1.02. The QPTR was calculated utilizing the P-250 plant computer in accordance with paragraph 6.3 of 1-PT-23. The inspectors observed the operator performing the work to be closely following the procedure and using good self-check practices. The inspectors reviewed the results of the surveillance and concluded that the QPTR calculated (1.007) fully satisfied the requirements of TS 3.2.4.

M1.4 Unit 2 Steam Flow Channel Post Maintenance Testing (62707)

On February 4, Unit 2 main steam flow channel III failed downscale. The licensee determined that a circuit card associated with instrument F-MS-2474 had failed. The circuit card was replaced, the circuit was successfully tested, and the flow channel was returned to service. The inspectors verified that related protection channels were placed in trip as required by TS and observed post maintenance testing of the channel. The testing was performed in accordance with 2-ICP-MS-F-2474, Steam Generator A Steam Flow and Feed Flow Protection Channel III (F-MS-2474 and F-FW-2477), Revision 1. The inspectors independently verified that portions of the data were correctly recorded, the data met acceptance criteria, and the channel functioned properly.

M1.5 Unit 2 B Motor Driven Auxiliary Feedwater (AFW) Pump Quarterly Test (61726)

On February 4, the inspectors witnessed performance of the quarterly surveillance test of the Unit 2 B motor driven AFW pump. The inspectors verified that the test was performed in accordance with 2-PT-71.3Q, 2-FW-P-3B, B Motor Driven AFW Pump, and Valve Test, Revision 15, and that the data recorded was correct and met the acceptance criteria contained in the procedure.

M1.6 LHSI Pump Surveillance Test

a. Inspection Scope (61726)

The inspectors observed the quarterly pump operability test for Low Head Safety Injection Pump 2-SI-P-1B to ensure TS 4.5.2.f.2 and TS 4.0.5 requirements were satisfied. Additionally, the inspectors reviewed selected sections of the UFSAR and historical test records.

b. Observations and Findings

On February 11, the inspectors observed operators performing 2-PT-57.1B, Emergency Core Cooling Subsystem-Low Head Safety Injection Pump (2-SI-P-1B), Revision 27-P2. The purpose of the test was to demonstrate pump operability requirements for discharge pressure and pump vibration. Prior to the test, the inspectors ensured the acceptance criteria in the controlling procedure agreed with the TS requirements. Additionally, the inspectors reviewed the associated pump performance curve (UFSAR Figure 6.3-7) to verify the test accurately reflected what was assumed in the UFSAR. The inspectors also verified that performance of the test had been approved by management, was properly planned, and the associated Limiting Condition for Operation was entered. During the test, the inspectors independently measured and calculated pump data associated with pump discharge pressure requirements and found them to be within TS requirements. The inspectors also verified that test instruments had been properly calibrated. Upon completion of the test, the inspectors reviewed the completed test procedure for accuracy and completeness. Additionally, the inspectors ensured proper review was performed by the Senior Reactor Operator and no problems were noted.

The inspectors later reviewed historical test records from 1996 to ensure that the testing frequencies had been completed in accordance with TS requirements. The inspectors also reviewed the operating data and no adverse trends were evident.

c. Conclusions

The inspectors concluded that the quarterly pump operability test for Low Head Safety Injection Pump, 2-SI-P-1B, was properly performed and that TS requirements were satisfied.

M1.7 Turbine Valve Freedom Test (61726)

On February 14, the inspectors observed operators performing 2-PT-034.3, Turbine Valve Freedom Test, Revision 18-P1. The test was required by TS 4.7.1.7.2.a to demonstrate the operability of the turbine governor and throttle valves. The inspectors found that operators performed the evolution carefully and in accordance with approved operating procedures. The inspectors noted that communication between the reactor cperator and the unit senior reactor operator was good and the level of supervisory oversight during the test was appropriate. The inspectors verified that the valves performed acceptably during the test and that TS requirements were met. The inspectors concluded that the Unit 2 turbine valve freedom test met TS requirements and was properly performed.

III. Engineering

E1 Conduct of Engineering

E1.1 Generic Letter 89-10 Program Implementation

a. Inspection Scope (Temporary Instruction 2515/109)

This inspection provided an assessment of the licensee's implementation of GL 89-10, Safety-Related-Motor-Operated Valve Testing and Surveillance. The inspection was conducted through a review of the licensee's GL 89-10 implementing documentation and through interviews with licensee personnel. Documents reviewed included the licensee's technical overview and closeout document (PE-0016, Revision 6), MOV program (VPAP-0805, Revision 6), diagnostic testing procedure (0-ECM-1505-01, Revision 17), gate and globe valve thrust and torque calculations, guidelines for addressing MOV design issues (NASES-3.10, Revision 4), butterfly valve torque calculations and assessments of test results, and summary matrices of MOV available valve factors and margins.

To assess details of the licensee's implementation of GL 89-10, the inspectors selected the sample of valves tabulated following this paragraph for particular attention. Other valves were also addressed, where appropriate to the areas being reviewed by the inspectors.

2-CH-2275B	High Head Safety Injection Pump Recirculation Valve
2-RC-2536	Power Operated Relief Valve (PORV) Block Valve
2-RH-2700	A Hot Leg to Residual Heat Removal (RHR) Pump Suction Valve
2-SI-2864B	Low Head Safety Injection Pump Discharge Valve
2-SI-2867A	Boric Acid Injection Tank Inlet Valve
1-SW-105A	Recirculation Spray Heat Exchanger Return Header Valve
1-SW-117	Auxiliary Service Water Pump Discharge Valve
1-SW-123A	Service Water Discharge Winter Bypass Valve
1-SW-104C	Recirculation Spray Heat Exchanger Outlet Valve
2-SW-220B	Auxiliary Service Water Return Header Valve
2-CC-200B	Component Cooling Supply to RHR Heat Exchanger Valve

b. Observations and Findings

1. Scope of MOVs Included in the Pr am

The scope of valves originally in the licensee's GL 89-10 program was reviewed and determined acceptable by the NRC during Inspections 50-338, 339/91-09 and 93-16. In the current inspection the inspectors verified that the scope had not been changed. The MOV program contained a total of 249 MOVs (consisting of 142 gate, 22 globe, and 85 butterfly valves).

2. Determinations of Settings and Verifications of Capabilities for Gate and Globe Valves

MOV Sizing and Switch Settings

North Anna's gate valve thrust and torque calculations utilized standard industry equations. An assumed stem friction coefficient of 0.20 was used to convert thrust to torque. Generally, the valve factors used in the gate valve design thrust calculations were based on original vendor information and were lower than are now typically used in the nuclear industry. Except for Westinghouse valves, a valve factor of 0.20 was used for double disc gate valves and 0.30 for flex and solid wedge gate valves. Westinghouse gate valve thrust requirements were calculated using the Westinghouse equation and a valve factor of 0.55. The licensee added 15 percent margin to the minimum total thrust requirements calculated for their gate valves to account for variations in valve factor, load sensitive behavior, and degradation of stem lubricant. Further, the thrust requirements calculated for the Westinghouse and other gate valves were increased to account for diagnostic equipment uncertainties and torque switch repeatability. Maximum thrust limits were based on the component with the lowest capability and were lowered to account for diagnostic equipment uncertainties and torque switch repeatability.

Globe valve thrust requirements were calculated using standard industry equations with a 1.1 valve factor and a 0.20 stem friction coefficient. Similar to the gate valves, the globe valves' minimum and maximum calculated thrust requirements were adjusted to account for diagnostic equipment uncertainties and torque switch repeatability.

Grouping and Design-Basis Capability

North Anna divided their MOVs into 42 groups. A group consisted of identical valves each having the same drawing number. The only differences found within a valve group were slight variations in the design-basis differential pressure requirements for particular valves. The gate and globe valve groups were separated into two categories depending on whether or not the licensee had used dynamic test results to justify the design-basis capabilities of the valves in the group.

The first category of gate and globe valves contained 19 groups. The inspectors found that the capabilities of these valves had been justified through dynamic testing. The number of valves dynamically tested from each group met GL 89-10, Supplement 6 recommendations. From their review of the licensee's test results, the inspectors found that the settings determined using the licensee's methodology bounded the actual requirements for all of the dynamically tested MOVs, except 2-SI-

2864B. This valve had a calculated minimum required thrust of 8155 lbs. From testing, the thrust at flow cut-off at 100 percent of design-basis differential pressure was 8792 lbs. Based on this test result, North Anna used 115 percent of 8792 lbs as the minimum required thrust applied to all MOVs in the valve group. The inspectors found that the licensee had satisfactorily demonstrated design-basis capability for MOVs in the 19 groups in the first category.

The second category of gate and globe valve groups consisted of 13 groups that were impractical to dynamically test. No dynamic test data were available for these valve groups. The inspectors requested that North Anna personnel calculate an available valve factor (AVF) for each MOV using the formulas given below.

AVF (Close) = (Th * [1 - (LSB + U)]) - PL - SR/ (Disc Area * DBDP)

AVF (Open) = (Th * [1 - (LSB + U)]) - PL + SR/ (Disc Area * DBDP)

where,

Th	=	thrust available for limit switch control, thrust at torque switch trip for torque switch control
LSB	=	load sensitive behavior
U	=	uncertainty (instrument and other uncertainties combined by square root sum of squares method)
PL	=	packing load
SR	=	stem rejection load
DBDP	=	design-basis differential pressure

Based on the above calculations, the lowest available valve factors were for MOVs FW-154C (0.64) and RH-2700 (0.67). All other MOVs in these groups had available valve factors of 0.70 or higher. Based on industry experience and the results of the licensee's tests, the inspectors considered the MOVs with available valve factors above 0.70 to have adequate capabilities. FW-154C and RH-2700 are discussed below:

FW-154C was a Crane, 16-inch, 900# class, flex-wedge gate valve that was powered from a non-vital electrical bus. This valve performed a back-up close function to isolate main feedwater. Based on industry experience with this valve design, the inspectors considered the 0.64 available valve factor of this valve sufficient to ensure its design-basis capability. Further, the inspectors noted that the valve provides a back-up rather than a primary isolation function.

RH-2700 was a Copes-Vulcan, 14-inch, 2500# class, parallel-disc gate valve with open and close safety functions. The inspectors reviewed the static diagnostic trace for this MOV and verified that it showed no abnormalities in valve performance. Based on general industry experience with parallel-disc gate valve testing and the absence of any performance abnormalities, the inspectors considered the 0.67 available valve factor of this valve sufficient to ensure its design-basis capability.

The inspectors accepted the licensee's verification of the design-basis capability of the non-dynamically tested gate and globe valves based on currently available valve factors. However, there was no assurance that these values of available valve factor would be maintained long-term and the inspectors were concerned that the reliability of the licensee's analytical method of determining thrust requirements for these valves had not been demonstrated through dynamic testing. Licensee personnel indicated that they would review industry and North Anna test results as part of their long-term test program and would look for further support for the application of their thrust determination methodology to their non-dynamically tested MOVs. The inspectors considered this appropriate.

Opening Diagnostic Measurement Uncertainty

In 1993 and 1994 the licensee's diagnostic equipment vendor, Liberty Technologies, reported previously unrecognized opening measurement uncertainty in their Customer Service Bulletins 031 and 037. The NRC inspectors verified that the licensee's current diagnostic procedure provided appropriate requirements to address this uncertainty. Additionally, the inspectors asked licensee personnel to review tests performed prior to the issuance of the Customer Service Bulletins to verify that the uncertainty had been adequately evaluated. The licensee completed this review and found no problem in the tests performed prior to Customer Service Bulletins. However, the licensee also reviewed more recent tests and found that a diagnostic procedure step addressing the uncertainty had not been performed for one test. Procedure 0-ECM-1505-01, revision 17-P2, Attachment 5, step 4 was not completed in evaluating opening diagnostic test results for valve 1-RS-100A in February 1996 and, as a result, the licensee failed to recognize that the test results indicated that the operability of this valve was in question. When this was discovered by the licensee during the current inspection, it was identified for resolution in DR N-97-68, dated January 9, 1997. Licensee corrective actions included issuance of a work request to inspect the valve and reduce the seating thrust (No. 078981) and immediate completion of an operability evaluation. The inspectors reviewed the completed operability evaluation and found that it demonstrated that the valve was operable based on comparisons of opening and closing motor current and spring pack deflection data. The inspectors considered the missed procedure step to be a violation of 10 CFR 50, Appendix B, Criterion V (Instructions, Procedures, and Drawings). This licensee identified and corrected violation is being treated as an Non-cited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy (50-338, 339/97001-02).

Load Sensitive Behavior

The licensee determined a load sensitive behavior mean and variability from their tests. These values were used in calculating available valve factors (as described above) for valves in groups that were not dynamically tested. The mean was applied as a bias value of 5.27 percent (LSB in the equation) and the variability value of 12.54 percent (95 percent confidence level) was combined with other uncertainties. The inspectors found the licensee's determinations satisfactory. However, they noted that the mean and variance in load sensitive behavior had been determined using a limited amount of data (only 10 data points) and questioned whether the results of future tests would be analyzed to strengthen the determination. The licensee stated that the results of any future dynamic testing for GL 96-05, Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves, would be evaluated to provide further confidence in the values used.

Stem Friction Coefficient

The licensee typically used a 0.20 stem friction coefficient in determining static settings. To assess this assumption, North Anna personnel first converted their static data to equivalent dynamic test data by using the fold line method described in NRC Report NUREG/CR 6100. The equivalent dynamic test data was then used in a statistical analysis which yielded a 95 percent confidence level stem friction coefficient value of 0.23. This study contained 34 data points. North Anna personnel screened their GL 89-10 MOVs using the 0.23 stem friction coefficient. No operability concerns were identified. The inspectors noted that North Anna should use the 0.23 stem friction coefficient value to statically set-up their MOVs instead of just for screening. North Anna personnel stated that in the future they would use the value obtained from their testing in establishing the static set-ups.

Torque Switch Repeatability

The licensee used the guidance from Limitorque Maintenance Update 92-02 to obtain values for torque switch repeatability. These values were combined with other random uncertainties in determining available valve factors. The inspectors found that the licensee's methodology satisfactorily accounted for torque switch repeatability.

Linear Extrapolation

Procedure NASES-3.10 required dynamic testing to be performed at a differential pressure of least 80 percent of the design-basis value for satisfactory extrapolation of the test data to design-basis conditions. However, the licensee did not have a stated absolute minimum pressure or contact seating force below which extrapolation would not be performed. The inspectors found no instances where this resulted in inappropriate testing or evaluations. North Anna personnel stated that they would revise their procedure to specify that the MOV coordinator or Engineer should be contacted for an evaluation if absolute minimum pressure or contact seating forces were low.

3. Determinations of Operating Requirements and Verifications of Capabilities for Butterfly Valves

The North Anna GL 89-10 program included 85 butterfly valves that were separated into 10 groups. The valves operated at moderately low design-basis differential pressures (125 psi maximum) and ranged in size from 8 to 24-inches. The valves were from two manufacturers, Allis Chalmers and Contramatics.

The torques required to operate the valves under design-basis conditions were either determined by the licensee using the vendor's equations (Allis Chalmers valves) or were provided by the vendor (Contramatics valves). As the valves were limit seated, their torque capabilities were based on the limiting component (motor, operator, or valve). The licensee's spreadsheet calculation indicated that all of the valves had capability margins at least 20 percent above the vendor torque requirements and, in most cases, the margins exceeded 50 percent. This provided margin for any discrepancy between actual torque requirements and vendor requirements.

The licensee had dynamically tested the number of valves from each group recommended by GL 89-10, Supplement 6; but had not measured the operating torques. Lacking quantitative torque measurements, it was not possible to demonstrate whether the vendor torque requirements were reliable. In most cases the licensee's dynamic tests had been performed at or above design basis differential pressure, which provided increased confidence in the capabilities of the valves to perform their design-basis functions. However, uncertainties remained because the potential adverse effects of reduced voltage and increased temperature during design-basis operation were not addressed. Subsequent to the inspection, in a January 20, 1997 letter to the NRC, the licensee submitted commitments to resolve these uncertainties. The licensee committed to perform instrumented dynamic testing and/or to apply the Electric Power Research Institute (EPRI) Performance Prediction Model to validate their design-basis calculation methodology for determining butterfly valve torque requirements. The commitments proposed were considered adequate to resolve this issue (see Section E1.1.c).

4. Periodic Verification

The licensee incorporated MOV periodic verification requirements into the preventive maintenance (PM) tasks specified in their database. The inspectors found that the database specified lubrication, diagnostic testing, and actuator inspection intervals and identified when the tasks were last performed. This implementation of periodic verification was considered adequate for closure of GL 89-10. The NRC may re-assess the licensee's long-term periodic verification program as part of its review of GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves", dated September 18, 1996.

5. Post Modification Testing

Licensee personnel stated that post modification test requirements were determined by engineers using post maintenance testing requirements as guidance. The licensee's implementation of post maintenance testing for GL 89-10 had been previously reviewed by the NRC and determined acceptable during Inspection Report 50-338, 339/94-04.

To assess the adequacy of the post modification testing implemented by the licensee, the inspectors selected and reviewed the testing recorded for five modifications. These modifications were either completed during the past two years or were still in process. The modifications were identified as follows: Nos. 273758-01 (valve replacement), 287719-01 (change to limit seating), 317913-01 (motor pinion gear replacement), 317856-04 (actuator modification), and 287720-01 (change to limit seating). The inspectors found that the testing specified was appropriate and concluded that the licensee had implemented acceptable post modification testing.

6. Pressure Locking and Thermal Binding

The inspectors reviewed the evaluation of gate valves susceptible to pressure locking and/or thermal binding which the licensee had completed in response to GL 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves. In letters to the NRC dated February 7 and July 3, 1996, the licensee identified valves that were susceptible to pressure locking and/or thermal binding and corrective actions.

The licensee's GL 95-07 submittals stated that an analytical method was utilized to demonstrate that the actuators on valves RC-MOV-X535, RC-MOV-X536, SI-MOV-X836, and SI-MOV-X869A and B could develop adequate thrust to overcome pressure locking. Pressure locking thrust requirements for these valves were determined using either a method developed by Virginia Power or one from Westinghouse Owners Group(WOG)/Commonwealth Edison. The inspectors independently calculated the thrust required to overcome pressure locking and the actuator capability for these valves and concluded that the actuators were able to develop the thrust required to overcome pressure locking. The inspectors used the WOG/Commonwealth Edison methodology with the appropriate GL 89-10 parameters for calculating the thrust required to overcome pressure locking.

The licensee's GL 95-07 submittals stated that valves SI-MOV-X867A, B, C, and D; 1-RS-MOV-100B; and 2-RS-MOV-200A and B were not susceptible to pressure locking. The licensee indicated that the valves might initially pressure lock in attempting to open, such that the motors would be incapable of unseating the valves and would undergo locked rotor conditions. However, they considered that the respective safety injection or recirculation spray pump would start and the discharge pressure applied to the upstream side of each valve would equalize bonnet pressure allowing the valves to open prior to reaching their thermal overload settings. The inspectors considered that even temporary actuator operation at a locked rotor condition was not an acceptable solution. The inspectors independently calculated the thrust required to overcome pressure locking and the actuator capability for these valves and concluded that the actuators were able to develop the thrust required to overcome pressure locking without reaching locked rotor conditions. The inspectors used the WOG/Commonwealth Edison methodology with the appropriate GL 89-10 parameters for calculating the thrust required to overcome pressure locking.

The licensee's GL 95-07 submittals stated that valves SI-MOV-X860A and B were not susceptible to pressure locking, as the valve bonnets were vented. The inspectors verified the installation of vent lines on the Unit 2 valves through a review of Design Change Package (DCP) 96-106 which implemented the bonnet vent modification on September 21, 1996.

The licensee's GL 95-07 submittals stated that the PORV block valves, RC-MOV-X535 and RC-MOV-X536, were susceptible to thermal binding. The licensee evaluated thermal binding caused by stem growth and determined that their valves had sufficient capability to overcome this binding. However, the licensee did not evaluate thermal binding caused by mechanical interference due to different expansion and contraction characteristics of the valve body and disk materials. The inspectors informed the licensee that thermal binding caused by mechanical interference between valve body and disk should be evaluated and corrective actions implemented if required in order for the NRC staff to determine if the licensee met the intent of GL 95-07.

The licensee's GL 95-07 submittals stated that quench spray pump discharge valves, QS-MOV-X01A and B, were not susceptible to pressure locking. The licensee indicated that the valves might initially pressure lock in attempting to open, such that the motors would be incapable of unseating the valves and would undergo locked rotor conditions. However, they considered that the quench spray pumps would start and that the discharge pressure applied to the upstream side of each valve would equalize bonnet pressure allowing the valves to open prior to reaching the thermal overload setting. The inspectors considered that actuator operation at a locked rotor condition was not an acceptable solution. The inspectors informed the licensee that these valves should be reevaluated for pressure locking and corrective actions implemented if required in order for the NRC staff to determine if the licensee met the intent of GL 95-07.

The licensee's GL 95-07 submittals stated that RHR system isolation valves, RH-MOV-X700/X701 and RH-MOV-X720A and B, were determined to be potentially susceptible to pressure locking. Since North Anna Power Station is licensed to achieve hot shutdown, the licensee concluded that these valves were outside the scope of GL 95-07. Therefore, the licensee did not evaluate these valves for pressure locking. The NRC staff is continuing its evaluation of this issue.

The adequacy of the licensee's actions to address pressure locking and thermal binding remain under NRC evaluation. During the current inspection, the inspectors raised issues regarding the susceptibility of the licensee's quench spray valves to pressure locking, possible thermal binding of the PORV block valves, and whether RHR system isolation valves should be evaluated for pressure locking. In the future, the NRC staff will address these issues in their safety evaluation of the licensee's response to GL 95-07. Followup of adequacy of GL 95-07 actions was identified as an IFI (50-338, 339/97001-03).

7. Trending

The licensee's trending requirements were specified by VPAP-0805, Motor-operated Valve Program, Revision 6, and consisted of a periodic review and issuance of MOV program reports indicating the status of programmatic improvement measures, types of problems found, trend information, failure rates, etc.

The inspectors obtained and reviewed the last two MOV program reports, which were dated May 5 and December 31, 1996, and covered nine month periods. Additionally, the inspectors performed a review of approximately 60 licensee DRs documenting MOV problems and reviewed trendable MOV data maintained in a licensee VOTES database. The inspectors found that the licensee was tracking failure trends and maintaining data for use in identifying trends in MOV degradation. The inspectors considered this adequate for GL 89-10 closure.

The inspectors had one negative comment. The amount of attention which the program reports devoted to causes of MOV failures and to discussing some significant MOV problems was considered too limited. For example, regarding MOV problems, the inspectors noted that the reports provided little discussion of improper gear combinations that had been documented in a number of DRs. The cause was not mentioned. This problem had resulted in many evaluations and corrective maintenance actions.

8. NRC Information Notice 92-18, Potential For Loss Of Remote Shutdown Capability During A Control Room Fire

Information Notice (IN) 92-18 alerted licensees of the potential for loss of safe shutdown capability during a fire in the control room. The IN reported that hot shorts occurring during the fire could potentially cause the MOVs needed for safe shutdown to go to a stall condition. This stall could result in valve and/or actuator damage that would preclude use of the MOVs for shutdown.

The inspectors reviewed the licensee's evaluation of IN 92-18, which concluded that thermal overload devices provided adequate MOV protection if a hot short occurred. The licensee provided the inspectors with an NRC Safety Evaluation issued to North Anna which stated that the licensee had proposed satisfactorily electrical isolation to assure that hot shorts would not preclude safe shutdown (letter to the licensee dated November 18, 1982). The inspectors determined that further internal NRC review of this issue was required to determine if further inspection or review were warranted.

9. Strengths

The inspectors observed a number of strengths in the licensee's implementation of GL 89-10. Particular examples included:

- Technically capable and dedicated personnel.
- Effective efforts to organize and develop data to resolve the final concerns developed during the NRC review.
- Good diagnostic test assessments.

c. Conclusions

The inspectors determined that the licensee had generally met the intent of GL 89-10 in verifying the design-basis capabilities of their MOVs. However, the licensee had not adequately established the reliability of their vendor-based methods for determining butterfly valve torque requirements and this was considered a weakness. In a letter to the NRC dated January 20, 1997, the licensee committed to resolve this issue through further validation of their methods for determining butterfly valve torque requirements. They proposed to accomplish this by performing instrumented differential pressure testing on one 18-inch Contramatics and one 24-inch Contramatics valve and by either performing instrumented differential pressure testing or applying the EPRI Performance Prediction Model to four representative Allis Chalmers valves. The letter stated that the Contramatics valve testing would be completed during the 1997 Unit 1 refueling outage and that any application of the EPRI model would be completed by the end of 1997. Testing of the Allis Chalmers valves, if performed in place of applying the EPRI model, would be completed during the Spring 1998 Unit 2 refueling outage. The letter also indicated that the NRC staff would be notified of the results and status by the end of 1997. Based on the results of this inspection and the above commitments, the NRC concluded that their review of the licensee's implementation of GL 89-10 could be closed. NRC verification of the licensee's completion of the above commitments was identified as an IFI (50-338, 339/97001-04).

The inspectors noted several licensee strengths, which are described in Section E1.1.b.9.

One related NCV was identified, involving failure to follow a procedure step relating to the uncertainty in valve opening thrust measurements. This NCV is described in Section E1.1.b.2.

E2 Engineering Support of Facilities and Equipment

E2.1 Surveillance Test Engineering Evaluation Review

a. Inspection Scope (37551)

During the weeks of February 3 and 10, the inspectors reviewed Engineering Transmittal (ET) TI-97-001, Evaluation of High Differential Pressure on 2-RS-P-2B, dated February 1, 1997, and discussed its conclusions with test engineers. The ET was written to evaluate the results of an outside recirculation spray pump, 2-RS-P-2B, surveillance test in which the pump differential pressure was found to be above the required action range. The ET concluded that the pump could be considered operable, and the inspectors reviewed the ET to verify that the pump's return to operable status was supported by engineering analysis and that applicable code requirements for Inservice Testing (IST) were met.

b. Observations and Findings

The ET stated that the pump's history had been reviewed, and no adverse behavior was found. Additionally, the evaluation noted that all other test parameters were satisfied within acceptable ranges. The ET stated that the pump was operating satisfactory based on the wide oscillations observed in the pump discharge pressure gage, difficulties obtaining consistent indication, and the narrow amount by which the value exceeded the limit (2.5 psid greater than the 192.3 psid limit). The ET recommended that instrument snubbers be installed after which new baseline values would be established, as well as, decreasing the test frequency from 18 months to 9 months.

The inspectors obtained and reviewed pump performance data from past tests for 2-RS-P-2B and other outside recirculation spray pumps. The inspectors found that the values for past tests were consistently high for 2-RS-P-2B and that this was the first time that the pump's differential pressure had entered the required action range. These findings supported the ET's conclusion that comparing current test data and past history did not indicate an adverse trend. While reviewing past data, the inspectors also verified that past tests had been performed within the required surveillance test intervals.

The inspectors then met with test engineers and reviewed how the licensee's actions complied with IST code requirements. The inspectors reviewed the applicable code sections which required that the pump remain inoperable until the cause was determined and corrected. The correction was normally required to be replacement or repair. An allowed alternative action was to complete an analysis to demonstrate that the condition did not impair operability followed by establishing a new set of reference values. The inspectors questioned how the ET met these requirements and were provided additional information from the engineers. This information was not clearly stated in the ET but had been discussed in the Station Nuclear Safety and Operating Committee (SNSOC) meeting held prior to returning the pump to operable status.

The engineers informed the inspectors that in addition to the facts discussed in the ET, the SNSOC had also discussed difficulties experienced by operators performing the test. Tests in the past had been performed under the Engineering Department's cognizance, and this was the first time the test was performed entirely by Operations personnel. As a result, the engineers believed that operators may not have been sensitive to the effects that pump heatup could have on the test. The pump's recirculation test flow path was extremely small in volume and quickly heated up during the test. This was believed to cause a wide variance in data if it was not collected at consistent times following pump start. This fact had been known to engineers during past tests, but the test procedure did not provide operators with guidance to ensure consistency in times to collect data. The engineers stated that the intent was to install the instrument snubbers and establish a new set of baseline data during the next test. Additionally, the test procedure would be revised to ensure that pump performance data was taken promptly at the end of the five minute waiting period required by the code. Based on the additional information, the inspectors found that the licensee's evaluations supported returning the pump to operable status.

c. Conclusions

The inspectors concluded that following a surveillance test failure, the licensee adequately evaluated out-of-specification data prior to returning the pump to operable status. However, the ET did not clearly document all factors considered in this evaluation.

E7 Quality Assurance in Engineering Activities

E7.1 Review of UFSAR Commitments

A recent discovery of licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for a special focused review that compared plant practices, procedures and/or parameters to the UFSAR description. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters.

The inspectors noted that as a result of increased licensee awareness and initiatives, numerous DRs were submitted concerning UFSAR discrepancies during this inspection period (DRs N-97-334 through -338, -348 through -353, -452, and -482). The inspectors reviewed the DRs and verified that there were no significant safety concerns or unreviewed safety questions.

E8 Miscellaneous Engineering Issues (37551, 92700, 92903)

E8.1 (Closed) IFI 50-338, 339/95008-03, MOV stem rejection thrust and MOV program changes. This item addressed followup of MOV-related issues and was divided into two parts. Part 1 was for followup on a finding that licensee GL 89-10 calculations determined valve stem rejection thrusts from differential pressure rather than upstream pressure. It appeared that this could be non-conservative in some instances. Part 2 was for followup on licensee actions to ensure satisfactory lubrication of valve stems. Lubrication problems had been identified during torque testing and the licensee planned to develop "lessons learned" and implement appropriate changes to their MOV lubrication program practices. The inspectors followup for each part was as follows:

Part 1: The licensee had evaluated their use of differential pressure in place of upstream pressure. The inspectors reviewed the results of the evaluation and the actions taken, which were documented in Engineering Transmittal CME 96-0079. The inspectors found that the evaluation demonstrated that the impact of the error was generally small. The error did not result in the operability of any valve being questioned.

Part 2: Licensee actions to address the lubrication program changes were specified in DR N 95-759 and resulted in Commitment Tracking System (CTS) items 02-95-2172-07 through -010. These CTS items specified development and implementation of a lubrication program that incorporated the "lessons learned" from investigation of valve stem lubrication problems. The items also provided for monitoring the effectiveness of the program. The inspectors verified the procedure (0-MPM-0400-05, Revision 2) and planning database entries that the licensee developed to implement the new lubrication program. Additionally, the inspectors confirmed that the licensee had evaluated the effectiveness of the program one refueling outage after revising stem lubrication practices. The study containing this evaluation was documented in Engineering Transmittal SE-96-059 and concluded there was no significant change in lubricant performance over the 18 month period between stem re-lubrications.

E8.2 (Closed) Licensee Event Report (LER) 50-338, 339/96006, charging pump interlock logic renders pumps inoperable due to a design error. The licensee reported that if the C charging pump was operating on a train with that train's charging pump not available for service, then a loss of DC power (opposite the train powering the C charging pump) would result in no charging pumps operating. The interlock logic would trip the running C charging pump and the loss of DC control power on the other train would prevent that train's charging pump from automatically starting. Thus, a single failure could cause the charging pumps not to be able to automatically perform their safety function during certain accidents. The ability to manually close the charging pump (the one that loses DC control power) breaker remained available. This was reported as a condition that alone could have prevented the fulfillment of the safety function of a system needed to mitigate the consequences of an accident.

In the LER, the licensee indicated that a modification was installed to correct this condition. The inspectors reviewed the pre-modification elementary electrical drawings and confirmed that the interlock logic would respond as reported in the LER. The inspectors noted that this condition existed only on one train since original licensing. In 1996, wiring modifications were made in accordance with DCP 95-226, Charging Pump Interlock Modification/NAS/Unit 1, and DCP 95-227, Charging Pump Interlock Modification/NAS/Unit 2, to replicate the design on the other train. The inspectors also confirmed that DCP 96-231, Charging Pump Interlock Modification/NAS/Units 1&2, corrected this condition on both units.

After reviewing UFSAR Section 3.1, Conformance with AEC General Design Criteria; 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors; Appendix K to Part 50, Emergency Core Cooling System Evaluation Models; and associated design criteria in Appendix A to Part 50, General Design Criteria For Nuclear Power Plants, the inspectors asked the licensee to clarify their design bases for the charging pumps and supporting DC power system. On February 12, the licensee informed the inspectors that failure of a DC power system train was not an active failure, i.e., the DC system can experience only passive failure modes. Therefore per UFSAR Chapter 3, a loss of DC power does not have to be considered during the first 24 hours of an accident. In addition, the licensee indicated that the interlock logic design reported in the LER was most likely acceptable and that they were evaluating withdrawing the LER. At the end of the report period, the inspectors were reviewing the licensee's position that the DC power system had no active failure modes. In addition, the inspectors were reviewing the licensee's licensing bases for compliance with regulatory requirements contained in 10 CFR 50.46 and associated criteria. Pending completion of these reviews, this item is identified as an URI (50-338, 339/97001-05).

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radiation Area Locks and Postings

a. Inspection Scope (71750)

During the period from January 23 - 27, the inspectors reviewed the posting of radiation areas and the labeling of radioactive material containers to verify that 10 CFR 20 requirements were met by the licensee. The inspectors performed direct radiation measurements to verify the accuracy of the licensee's radiation surveys and postings. Additionally, the inspectors verified the proper locking of doors to high radiation areas required to be locked by 10 CFR 20 and TS 6.12.1.

b. Observations and Findings

The inspectors checked approximately 30 doors leading into the Radiologically Controlled Area (RCA) and found that all were properly posted as required by 10 CFR 20.1902. Areas designated as high radiation areas and locked high radiation areas in the Auxiliary Building were checked and found to be posted and locked where required. Numerous containers of radioactive material in the Auxiliary Building and outside areas were checked and found to be marked in accordance with 10 CFR 20.1904.

The inspectors obtained a radiation survey instrument and measured radiation levels at various locations in the Auxiliary Building. The inspectors checked that radiation areas did not contain radiation levels which would require high radiation area postings. Additionally, the inspectors verified the accuracy of selected radiation survey results listed on copies of survey maps located at high radiation area entry points. The inspectors also checked that informational signs denoting general radiation levels at various points in the Auxiliary Building accurately reflected area radiation levels. The inspectors found that in all cases, actual radiation levels measured were at or lower than the levels posted for the areas.

c. Conclusions

Radiation areas and high radiation areas were properly posted and were locked when required by 10 CFR 20. Independent radiation measurements found that the licensee's postings and survey results were conservative.

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 March 10, 1997. The licensee disagreed that they were in violation of 10 CFR 70.24 (See Section O2.2). This was based on: 1) correspondence, dated May 11, 1988, that the NRC had sent to the Tennessee Valley Authority indicating that an exemption request was not necessary, and 2) that an implicit exemption to the requirements existed when TS Table 3.3-6, Radiation Monitoring Instrumentation, was approved. Item 1 of this table required only one criticality monitor for the fuel storage pool area. The licensee acknowledged the other 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 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

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92903:	Followup - Engineering
TI 2515/109:	Inspection Requirements For Generic Letter 89-10, Safety Related Motor-Operated Valve Testing and Surveillance

ITEMS OPENED AND CLOSED

Opened

50-338, 339/97001-01	VIO	Failure to Install a Radiation Monitoring System, Establish Procedures, and Conduct Training as Required by 10 CFR 70.24 (Section O2.2).

50-338, 339/97001-02	NCV	Failure to Follow Procedure Step in Evaluating MOV Diagnostic Test (Section E1.1.b.2).
50-338, 339/97001-03	IFI	Resolution of GL 95-07 Issues (Section E1.1.b.6).
50-338, 339/97001-04	IFI	Validation of Methods for Determining Butterfly Valve Torque Requirements (Section E1.1.c).
50-338, 339/97001-05	URI	Review DC Power System Failure Modes and Compliance With 50.46 (Section E8.2).

Closed

50-338, 339/95008-03	IFI	MOV Stem Rejection Thrust and MOV Program Changes (Section E8.1).
50-338, 339/96006	LER	Charging Pump Interlock Logic Renders Pumps Inoperable Due to Design Error (Section E8.2).
50-338, 338/97001-02	NCV	Failure to Follow Procedure Step in Evaluating MOV Diagnostic Test (Section E1.1.b.2).