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

Report Nos.: 50-369/88-09 and 50-370/88-09

Licensee: Duke Power Company
422 South Church Street
Charlotte, NC 28242

Docket Nos.: 50-369 and 50-370 License Nos.: NPF-9 and NPF-17

Facility Name: McGuire 1 and 2

Inspection Conducted: March 19, 1988 - April 22, 1988

Inspector: [signature] for 4/28/88
W. Orders, Senior Resident Inspector Date Signed

Accompanying Personnel: D. Nelson
R. Croteau

Approved by: T. A. Peebles 4/29/88
T. A. Peebles, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine unannounced inspection involved the areas of operations safety verification, surveillance testing, maintenance activities, and follow-up on previous inspection findings.

Results: In the areas inspected, three violations were identified. One violation was identified which included two examples for failure to follow procedure during safety valve testing and an inadequate procedure for slave relay testing. A second violation was identified which involved an inoperable component cooling train and the third violation involved a failure to perform post maintenance testing which rendered a nuclear service water train inoperable.

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REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *T. McConnell, Plant Manager
   B. Travis, Superintendent of Operations
   *M. Sample, Superintendent of Maintenance
   B. Hamilton, Superintendent of Technical Services
   R. Sharpe, Compliance Engineer
   J. Boyle, Superintendent of Integrated Scheduling
   L. Firebaugh, OPS/NPE/MNS
   *S. LeRoy, Licensing, General Office
   *D. Baxter, OPS/MNS/NPD
   *S. Copp, Planning Engineer
   R. Panner, Compliance
   J. Snyder, Performance Engineer
   *N. Atherton, Compliance
   W. Reeside, Operations Engineer
   R. White, IAE Engineer
   *G. Gilbert, MNS/NPD

   Other licensee employees contacted included construction craftsmen, technicians, operators, mechanics, security force members, and office personnel.

   *Attended exit interview

2. Exit Interview (30703)

   The inspection findings identified below were summarized on April 22, 1988, with those persons indicated in paragraph 1 above. The following items were discussed in detail:

   (OPEN) Violation 370/88-09-01, Failure to follow procedure for Pressurizer Code Safety Valve Testing and Inadequate Procedure for Slave Relay Testing. (See paragraphs 5 and 9).

   (OPEN) Violation 369/88-09-02, Inoperable Component Cooling (KC) Train Due to Inoperable Nuclear Service Water (RN) Valve. (See paragraph 10).

   (OPEN) Violation 369/88-09-03, Inoperable RN Train Due to a Failure to Test RN-21. (See paragraph 10).

   The licensee representatives present offered no dissenting comments, nor did they identify as proprietary any of the information reviewed by the inspectors during the course of their inspection.

3. Unresolved Items

An unresolved item (UNR) is a matter about which more information is required to determine whether it is acceptable or may involve a violation or deviation. There were no unresolved items identified in this report.

4. Plant Operations (71707, 71710)

The inspection staff reviewed plant operations during the report period to verify conformance with applicable regulatory requirements. Control room logs, shift supervisors' logs, shift turnover records and equipment removal and restoration records were routinely perused. Interviews were conducted with plant operations, maintenance, chemistry, health physics, and performance personnel.

Activities within the control room were monitored during shifts and at shift changes. Actions and/or activities observed were conducted as prescribed in applicable station administrative directives. The complement of licensed personnel on each shift met or exceeded the minimum required by Technical Specifications.

Plant tours taken during the reporting period included, but were not limited to, the turbine buildings, the auxiliary building, Units 1 and 2 electrical equipment rooms, Units 1 and 2 cable spreading rooms, and the station yard zone inside the protected area.

During the plant tours, ongoing activities, housekeeping, security, equipment status and radiation control practices were observed.

Use of overtime by operations was reviewed to verify compliance with TS requirements. Documentation showed that the maximum overtime limits were exceeded approximately nine times in 1987-1988, primarily for outage support. It is allowable to exceed the maximum limits for very unusual circumstances. Operations Management Procedure 1-7, Shift Manning and Overtime Requirements, requires that overtime worked in excess of guidelines be authorized in advance by the station manager or his designee (another high level of management). The inspector noted that in September of 1987 two instances of exceeding the maximum overtime limits were not authorized in advance. The licensee received a violation for this issue (see Inspection Report 369, 370/87-26) in the September 1987 period and corrective actions have been taken.

a. Unit 1 Operations

Unit 1 began the reporting period at full power. On March 23, 1988, the unit experienced a spurious safety injection (SI), main steam isolation and reactor trip. The spurious SI signal was generated in a Solid State Protection System (SSPS) cabinet containing the circuitry for A train low steam line pressure SI and main steam isolation. Licensee technicians had just completed testing in this cabinet and were closing the door. The spurious signal occurred when the door was shut. All A train SI components actuated and the high head pump injected into the primary coolant system. Main steam isolation and feed water isolation occurred, generating a turbine trip and resultant reactor trip. Operators determined that the SI was inadvertant and secured the injection in approximately nine minutes. No major problems occurred during the transient. All A train SI components functioned as designed during the transient, except that the reactor trip was caused by the turbine trip above P-8 instead of directly by the SI actuation as would be expected. The cause for this was determined later and is discussed below. During the event a low steam generator level in the C steam generator caused auxiliary feedwater (CA) to initiate a second time.

After the unit was stabilized, Instrumentation and Electrical (IAE) technicians attempted to determine the cause of the SI signal. The spurious signal was duplicated several times by agitating the SSPS cabinet. However, the signal could not be repeated following the shutting of a nearby heavy door. Further investigation could not determine the exact cause of the problem. The licensee considers that a small piece of loose wire or other conductor had shorted or grounded the circuitry upon agitation of the cabinet. According to the licensee, the additional agitation conducted during trouble-shooting, and finally the shutting of the nearby door served to vibrate the conductor away from the electrical contacts. A thorough cleaning of the cabinet was performed. This produced some small wire fragments which could have caused what the licensee postulates. The licensee also considers that the points of contact occurred in the low steamline pressure SI circuit downstream from the point where the reactor trip function branches off, thereby causing the SI to initiate without causing a direct reactor trip. Following the cleaning a complete functional surveillance was performed with no problems. The unit was restarted and achieved full power the following day. No similar problems in SSPS have developed.

On March 27, load was decreased to 86 percent power due to decreased load demand on the grid and was back at full power at 1247 a.m. on March 28. Later on March 28 power was reduced to 46 percent to support removal of a voltage regulator control drawer. Unit 1 returned to 100 percent power on March 29.

On April 16, Unit 1 was manually tripped from 100 percent power due to decreasing level in the C steam generator (SG) caused by the C feed regulating valve (FRV) failing shut. The C FRV shut due to a blown fuse on a card controlling the valve. The card was later tested and found to be the cause of the blown fuse. Auxiliary feed water (CA) auto started on low steam generator levels but 1SA-49, steam supply from the B SG to the turbine driven CA pump, did not indicate open due to problems with the position indicating limit switches. The valve was actually open. The limit switches were adjusted, the FRV card was replaced, and the unit returned to 100 percent power on April 18.

b. Unit 2 Operations

Unit 2 operated at full power for the entire period. The SI on Unit 1, discussed above, did however affect Unit 2. Due to the alignment of the common portion of the Nuclear Service Water (RN) System, the operating Unit 2 RN train was isolated by the single train SI on Unit 1. Had both SI trains on Unit 1 actuated, the common portion of RN would have realigned to ensure continued RN operation on Unit 2. Various Unit 2 component temperatures elevated, but Unit 2 operators diagnosed and corrected the problem in time to prevent any additional required actions.

No violations or deviations were identified.

5. Surveillance Testing (61726)

Selected surveillance tests were analyzed and/or witnessed by the inspector to ascertain procedural and performance adequacy and conformance with applicable Technical Specifications.

Selected tests were witnessed to ascertain that current written approved procedures were available and in use, that test equipment in use was calibrated, that test prerequisites were met, that system restoration was completed and test results were adequate.

Detailed below are selected tests which were either reviewed or witnessed:

PROCEDURE	EQUIPMENT/TEST
PT/0/A/4150/05	Pressurizer Safety Valve Setpoint Test
PT/1/A/4403/007	RN 1A Flow Balance Test
PT/2/A/4200/28A	SSPS Slave Relay Tests
PT/1/A/4208/03A	Train 1A NS Heat Exchanger Performance Test
PT/1/A/4252/01B	CA Pump 1B Performance Test
PT/1/A/4601/08B	SSPS Train B Periodic Test
PT/1/A/4403/01B	RN Train 1B Performance Test
PT/1/A/4403/01A	RN Train 1A Performance Test
PT/1/A/4206/01A	NI Pump 1A Performance Test
PT/1/A/4252/01A	CA Pump 1A Performance Test
PT/1/A/4204/01B	ND Pump 1B Performance Test
PT/2/A/4209/01A	NV Pump 2A Performance Test
PT/2/A/4206/01A	NI Pump 2A Performance Test
PT/0/A/4350/38	125 VDC Battery Service Test

See paragraph 9 for further information concerning PT/0/A/4150/05.

On March 22 at 10:02 AM, a procedure error in a performance periodic test procedure caused re-alignment of power sources to several Unit 2 non-safety containment ventilation systems. A step in procedure PT/2/A/4200/28A, Slave Relay Test, directs the opening of a "sliding link" to prevent ventilation units from tripping during the test. This occurs in the section of the procedure that tests slave relays in the Train A Safety Injection SSPS circuitry. The step specifies opening sliding link H-3 in cabinet 2ATC8. When the system was actuated in a subsequent step, several non-safety containment ventilation systems experienced a shunt trip to re-align their power sources to non-safety buses. The affected systems were lower containment ventilation (VL), Upper Containment Ventilation (VU), and Control Rod Drive Mechanism Cooling Ventilation (VR). These non-safety systems are designed to "load shed" in the event of an ESF actuation to lessen the electrical load on safety system power sources. The load shed took place because the wrong sliding link was specified. A step at the completion of the test specifies closing sliding link I-2 in the same cabinet. I-2 is the correct sliding link that should have been opened originally. Having the wrong sliding link open during the test did not otherwise adversely effect the test or plant operation. The procedure error constitutes an example of an inadequate procedure and is therefore an apparent violation. (370/88-09-01).

The licensee states that the procedure error occurred in a recent re-write of the procedure, but has no explanation for why it occurred. Another case of an unexplained procedure change occurred recently in an operations procedure which was discussed in NRC inspection report 50-369, 370/88-04. In that case two procedure steps were interchanged during re-write. A violation was issued for that occurrence, but the licensee's corrective action appeared to be limited to the operations organization where the problem occurred. When the corrective actions and lessons learned are shared with other departments, similar problems may be prevented.

6. Maintenance Observations (62703)

Routine maintenance activities were reviewed and/or witnessed by the resident inspection staff to ascertain procedural and performance adequacy and conformance with applicable Technical Specifications.

The selected activities witnessed were examined to ascertain that, where applicable, current written approved procedures were available and in use, that prerequisites were met, that equipment restoration was completed and maintenance results were adequate.

No violations or deviations were identified.

7. Follow-up on Previous Inspection Findings (92702)

The following previously identified items were reviewed to ascertain that the licensee's responses, were applicable, and licensee actions were in compliance with regulatory requirements and corrective actions have been completed. Selective verification included record review, observations, and discussions with licensee personnel.

(CLOSED) Violation 369, 370/85-06-04. Failure to Take Prompt Corrective Action to Notify Operations Personnel of Potential Degradation of Auxiliary Feedwater System and Correct Improper Valve Installation. Nuclear Station Modifications (NSM's) were completed by July 21, 1985 which installed temperature monitors to detect check valve leakage and replaced the stop check valves with a different design valve. The completed NSM's were reviewed and selected check valves were physically verified to be in place by the inspector. The violation was initially denied but the NRC determined that the violation occurred as stated in the Notice of Violation. The corrective actions stated in the response have been completed and this item is closed.

(CLOSED) Inspector Followup Item 369, 370/86-28-04, Testing of Safety Valves. Procedure PT/0/A/4150/05, now contains instructions on observing the trend of safety valve lift setpoints. See section 9 for more information on this PT.

(CLOSED) Unresolved Item 369/86-28-07, Blocking of Safety Functions. This issue dealt with blocking of the low pressure safety injection signal when a safety valve opened and caused excessive blowdown while in hot standby, Mode 3. The licensee has reinforced the policy of not blocking automatic safety actuations except when directed by approved procedures or 10 CFR 50.54(x). This item is closed.

(CLOSED) Violation 369, 370/86-28-01, Failure to Report. Corrective actions have been taken and this item is closed.

8. Licensee Event Report (LER) Followup (90712, 92700)

The following LER's were reviewed to determine whether reporting requirements have been met, the cause appears accurate, the corrective actions appear appropriate, generic applicability has been considered, and whether the event is related to previous events. Selected LER's were chosen for more detailed followup in verifying the nature, impact, and cause of the event as well as corrective actions taken.

(CLOSED) Licensee Event Report 369/86-17, Both Trains of Hydrogen Mitigation System Inoperable. Multiple failures of hydrogen ignitors during quarterly surveillance testing resulted in both trains of the hydrogen mitigation system being declared inoperable. The licensee has evaluated the life expectancy of the ignitors and estimates six years as a conservative life expectancy. Corrective action includes replacing the ignitors every four years.

(CLOSED) Licensee Event Report 370/86-03, Unidentified Reactor Coolant Leakage Due to Leaking Valves Resulting in Shutdown. Four valves were found to be leaking and subsequently repaired.

(CLOSED) Licensee Event Report 370/86-06, Failure to Maintain Required Boration Flow Path Due to Personnel Error. The event was discussed in an operations staff meeting and shift supervisor meeting.

(CLOSED) Licensee Event Report, 370/86-19, Missed Surveillance on Essential Auxiliary Power Systems. Corrective actions have been completed and this item is closed.

(CLOSED) Licensee Event Report 369/87-02, Both Trains of Containment Spray System Inoperable. This event resulted in violation 369/87-04-01 and corrective actions are being tracked in followup to the violation.

9. Pressurizer Safety Valve Setpoint Testing

During a review of the completed data sheets for PT/0/A/4150/05, Pressurizer Safety Valve Setpoint Test, it was noted that the maintenance personnel signing the data sheets for satisfactory lift checks were not the personnel who were trained to perform the tests. In a letter to the NRC dated July 22, 1987, the licensee committed to allowing only specially trained personnel to work and test these valves. This commitment was made in response to violation 369/86-28-06 which involved a primary system safety valve opening at a pressure outside the T.S. limit. The tests in question were performed on September 14, 1987; June 11, 1987 and June 13, 1987. The licensee stated that the qualified individuals were present but non-qualified individuals signed the data sheet for the qualified individuals. The licensee stated that in the future the data sheets will be clearly annotated if a non-qualified person signs for a qualified person and the data sheet will contain the name of qualified person performing the test.

Another problem with the test performed on pressurizer code safety valve 2NC1 on June 11, 1987, was discovered. The data sheet for this test listed a lift pressure of 2513 psi for the second lift of the valve. The procedure specified that each lift must be within the TS required range of 2485 psig plus or minus 1 percent (2461 to 2509). The other lift pressures were within the required band and the average of the three lifts was also within the required range. This is a second example of an apparent violation (370/88-09-01) of T.S. 6.8.1 for failure to properly implement the written procedure for pressurizer safety valve setpoint testing. Corrective actions for violation 369/86-28-06 should have prevented this violation from occurring in that personnel were trained on the specific requirements. Contributing to the violation was the fact that the data sheet did not clearly specify that each lift must be within the required band.

The licensee initiated a Problem Investigation Report when this item was brought to their attention by the inspector. The licensee has stated that the maintenance and quality control personnel involved believe the actual lift setpoint was in the required band but the data was incorrectly recorded. All parties state they were aware each lift was required to be within plus or minus 1 percent of 2485 psig. The proposed corrective actions include changing the procedure, counciling the persons involved, and expediting the development of a training program in this area.

During the review of the pressurizer safety valve setpoint test, Electric Power Research Institute (EPRI) Report NP-4235, Set-point Testing of Safety Valves Using Alternative Test Methods, was reviewed. This EPRI report was prepared to present the results of tests performed and to correlate alternate test methods for safety valve tests to testing using full pressure steam as the test medium. The full pressure steam test method most closely simulates the actual conditions which the valve experiences in the system. It is noted, however, that McGuire has loop seals in the lines from the pressurizer to the code safety valves so steam is not actually on the valves. The licensee uses an alternate test method using nitrogen as the pressure medium rather than steam. The EPRI test results appear to indicate that:

a. Tolerance bands using the nitrogen test method need to be much tighter than plus or minus 1.0% in order to assure the valve will lift at TS required 2485 plus or minus 1.0% psi while installed in the system. The licensee currently uses a plus or minus 1.0% tolerance band using nitrogen.

b. The actual valve lift setpoint using steam will be lower than the setpoint using nitrogen. It was discovered that generic correlations could not be made to relate nitrogen tests to actual in place lift setpoints; however, correlations on a valve-by-valve basis can be made with a higher degree of confidence. The method for determining the valve correlations is given in Appendix E of the EPRI report. The licensee currently does not use any correlations to correct the lift point using nitrogen to the lift point using a steam medium.

T.S. 3.4.2.2 requires that pressurizer code safety valves have a lift setting of 2485 psig plus or minus 1 percent and the lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. Based on the information available in EPRI Report NP-4235, it is not clear that the pressurizer code safeties would lift within the TS required range at normal system operating temperature and pressure. The licensee has indicated that the EPRI report does not take into consideration the fact that McGuire has loop seals in the lines to the pressurizer code safeties. According to the licensee, the temperature of the water at the code safety is 140 degrees F and testing at Wyle Laboratories has confirmed direct correlation between nitrogen lifts under ambient conditions and nitrogen with 140 degree F water at the valve inlet. The licensee stated that Catawba and Oconee send their pressurizer code safeties to Wyle to have hot lifts performed since neither have loop seals. This item was still being reviewed at the end of the inspection period.

10. Nuclear Service Water Valves Inoperable

On March 9, 1988, during a system walkdown a licensee engineer discovered Unit 1 Nuclear Service Water (RN) valve 1RN190B, (RN to component cooling (KC) heat exchanger 1B throttle valve) to be inoperable. A travel stop that limits the maximum open position of the valve had become repositioned toward the closed direction and was found to be loose. If the valve had been called upon during an Engineered Safety Features (ESF) actuation, the repositioned stop would have prevented required RN flow to the KC heat exchanger from occurring, and would have possibly prevented the KC system from adequately cooling critical components in a design basis accident. The travel stops are precisely set during RN system flow balance tests designed to balance RN system flow among all the RN heat loads, including KC. The licensee estimates that RN flow to the KC heat exchanger would have been reduced approximately 1500 gpm below the required 6000 gpm in the event of an ESF actuation. The licensee considers that the heat removal capability of the B train of KC, although impaired, was adequate due to low RN temperature and the clean condition of the KC heat exchanger. This hypothesis is based on design engineering analysis. Upon discovery, the licensee declared the KC train inoperable and took prompt action to restore the travel stops to the most recent tested position.

The last RN flow balance test was conducted on January 29, 1988, at which time the valve stops were adjusted. The licensee could not produce any documentation or evidence of authorized work conducted on 1RN190B that may have affected the travel stop positions since this last test. It is known that the travel stops have been routinely used to secure the valve in the shut position to facilitate isolation for other work. A work request usually documents these occasions. The licensee concluded that the hex nuts securing the travel stop had vibrated loose allowing the stop to drift in the closed direction. In normal operation the valve is throttled significantly in the closed direction with the travel stop performing no function, thus it was free to drift upon loosening of the nuts.

Technical Specification 3.7.3 requires that both trains of KC be operable during operation in Modes 1, 2, 3, and 4. One train may be inoperable for up to 72 hours in these modes. The mis-positioned travel stops on 1RN190B resulted in train B of KC being technically inoperable from the last documented position of the valve on January 29, until the discovery of the problem on March 9. This is an apparent violation (369/88-09-02) of the action statement requirements of Technical Specification 3.7.3.

In an unrelated event, on March 28, 1988, the licensee determined that valve 1RN-21, RN Strainer 1A automatic backwash valve, underwent maintenance on February 4, 1988, without subsequent retest. This valve is designed to automatically open upon high differential pressure across the 1A RN strainer thereby allowing backwash flow to clean the strainer. This flow is diverted from the total A train RN flow. Upon an ESF signal, RN-21 shuts, if open, to ensure that all RN flow is supplied to ESF heat loads cooling critical components in accident situations. A packing adjustment was performed on RN-21 on February 4, 1988, under a work request to investigate and correct a packing leak. The work request incorrectly identified that a retest was not required. Maintenance personnel tightened the packing but determined that the packing leak could not be stopped without over tightening the packing thereby impairing valve stroke. The work request had a contingency to be re-scheduled until an outage if tightening the packing was unsuccessful. The work request was returned to planning for this purpose. Contributing to this problem was the fact that maintenance clearance was also deemed to be not required which resulted in operations being not fully informed of the extent of work being conducted on the valve.

Technical Specification 4.0.5 states that testing to ASME Code requirements is required to properly retest ESF components following maintenance. Station Directive 3.2.2, Identifying and Performing Plant Retesting, implements these requirements by identifying the components and types of maintenance that require retests as well as identifying the retest required. RN-21 is identified as a component requiring retest. Adjustment of stem packing is an example of maintenance requiring retest. In this case a valve stroke timing test should have been performed since the valve is required to shut within 60 seconds upon receipt of an ESF signal.

On March 28, 1988, the licensee detected the error on the work request and immediately added RN-21 to the TS Action Item Log for RN train A which was currently declared inoperable for unrelated reasons. The performance of a stroke timing test at that point could have determined valve operability, however, additional packing adjustment took place first. During the subsequent stroke timing test, the valve failed to shut. Licensee review of operator aids computer (OAC) data revealed that the valve was actually required to automatically open and shut numerous times between February 4 and March 28 to correct high strainer differential pressure. Typical stroke times on these occasions, as recorded by the OAC, show that the valve shut well within the maximum time permitted, (approximately 10 seconds vs maximum 60 seconds) The licensee has stated that it is likely that satisfactory results would have been obtained had a formal stroke timing test been conducted after the initial packing adjustment. It was likewise hypothesized that the valve would have performed its safety function during the time of unknown inoperability. The licensee considers that the final packing adjustment caused the valve to fail to shut.

Technical Specification 3.7.4 requires that two trains of RN be operable in modes 1, 2, 3, and 4. One train may be inoperable for up to 72 hours in these modes. The failure of the work request to properly identify that a retest was required caused the requirements contained in TS 4.0.5 to be omitted. This resulted in RN-21 and thus train A of RN to be inoperable from February 4 to March 28, 1988. Mitigating factors, however, lessen the significance of the RN-21 inoperability. The total flow diverted with RN-21 open is approximately 700 gpm. This amount is a small portion of the total RN flow of greater than 12,000 gpm available from one train during accident conditions. Also, RN temperature during the time of inoperability was in the range of 40 to 50F, well below the design temperature of 78F. Design engineering evaluation indicates that ample heat sink existed for A train RN to perform its safety function. The prime concern and root cause of this particular event is the failure to retest as described above. This item is identified as an apparent violation (369/88-09-03) of Technical Specification 4.0.5.

The inoperability of the two RN valves discussed above occurred over an extended period of time resulting in numerous occasions when both trains of RN or the systems they support were rendered inoperable. Most notable is the overlapping period (February 4 to March 9) when both RN valves were inoperable rendering both trains of KC inoperable. The NRC recognizes the mitigating factors discussed above and considers the safety significance of these specific events to be minimal. However, had conditions been less favorable or other components been involved, the safety significance could have been much greater. The NRC is particularly concerned with the events associated with the RN-21 problem from the standpoint of maintenance work control and retesting. The inspectors have initiated a thorough study of the work control process to determine if sufficient controls are in place to prevent missed retests and unknown inoperabilities of safety system components.

11. Ground Water Detection

On April 12, 1988, NRR technical staff reviewed the Groundwater Monitoring System and conducted a walkdown of selected monitoring wells (Auxiliary Building East and West wall exterior monitors, Auxiliary Building north wall interior monitors PP-51, QQ-56 and PP-61), and Auxiliary Building drain sump "C". System operators and surveillances were found to be consistant with T.S. 3.4.7.13 requirements. On April 13, 1988 design calculations of hydrostatic and buoyancy influences and overturning potential for the Auxiliary Building, Reactor Buildings, and Diesel Generator Buildings were audited at licensee's corporate engineering offices. This review is part of the NRC's review of Duke's request for Technical Specifications 3.4.7.13 changes dated January 27, 1988.