UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report Nos.: 50-269/95-01, 50-270/95-01 and 50-287/95-01

Licensee: Duke Power Company
422 South Church Street
Charlotte, NC 28242-0001

Docket Nos.: 50-269, 50-270 and 50-287

License Nos.: DPR-38, DPR-47 and DPR-55

Facility Name: Oconee Units 1, 2 and 3

Inspection Conducted: January 1 - 28, 1995

Inspector: P. E. Harmon, Senior Resident Inspector — 2/16/95 Date Signed

W. K. Poertner, Resident Inspector
L. A. Keller, Resident Inspector
P. G. Humphrey, Resident Inspector

Approved by: R. V. Crlenjak, Chief — 2/17/95 Date Signed
Reactor Projects Branch 3

SUMMARY

Scope: This routine, resident inspection was conducted in the areas of plant operations, surveillance testing, maintenance activities, plant support, onsite engineering and technical assistance.

Results: No violations or deviations of NRC requirements were identified during this inspection period.

The Unit 2 feedwater control valve repair effort was well planned, coordinated and implemented, paragraph 2.c.

Another example of a Unit 3 motor operated valve failure occurred due to dirty torque switches, paragraph 3.a.(1).

Although the licensee's Quality Standards Manual was revised to identify the Condenser Circulating Water (CCW) pumps as safety-related, work procedures associated with the CCW pumps were not upgraded, paragraph 3.a.(6).

An inspector followup item was identified regarding low Unit 2 control battery capacities. Although these batteries have been in service less than 2 years, tests revealed that the 2CA battery was at 80.1 percent capacity and the 2CB battery was at 77 percent capacity. IEEE Standard 450 recommends battery replacement when

capacity has degraded to below 80 percent. The licensee performed an operability analysis that determined these batteries were still operable. The licensee is still evaluating root cause and long-term actions, paragraph 3.b.(5).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*B. Peele, Station Manager
E. Burchfield, Regulatory Compliance Manager
*D. Coyle, Systems Engineering Manager
J. Davis, Engineering Manager
T. Coutu, Operations Support Manager
*W. Foster, Safety Assurance Manager
*J. Hampton, Vice President, Oconee Site
D. Hubbard, Superintendent, Instrument and Electrical (I&E)
C. Little, Electrical Systems/Equipment Manager
J. Smith, Regulatory Compliance
*G. Rothenberger, Operations Superintendent
R. Sweigart, Work Control Superintendent

Other licensee employees contacted included technicians, operators, mechanics, security force members, and staff engineers.

*Attended exit interview.

2. Plant Operations (71707)

a. General

The inspectors reviewed plant operations throughout the reporting period to verify conformance with regulatory requirements, Technical Specifications (TS), and administrative controls. Control room logs, shift turnover records, temporary modification log and equipment removal, and restoration records were reviewed routinely. Discussions were conducted with plant operations, maintenance, chemistry, health physics, instrument & electrical (I&E), and engineering personnel.

Activities within the control rooms were monitored on an almost daily basis. Inspections were conducted on day and night shifts, during weekdays and on weekends. Inspectors attended some shift changes to evaluate shift turnover performance. Actions observed were conducted as required by the licensee's Administrative Procedures. The complement of licensed personnel on each shift inspected met or exceeded the requirements of TS. Operators were responsive to plant annunciator alarms and were cognizant of plant conditions.

Plant tours were taken throughout the reporting period on a routine basis. During the plant tours, ongoing activities, housekeeping, security, equipment status, and radiation control practices were observed.

b. Plant Status

All three units operated at or near full power throughout the inspection period.

c. Feedwater Control Valve Repair Effort

Unit 2 feedwater flow oscillations were investigated by the licensee and determined to be a function of the spool type positioner pilot valves on the feed control valve operators. Operations with assistance from Engineering and Maintenance developed a repair plan to place the control valves in manual and clean the pilot valve. Since manual control of the feed valve required close coordination and communication, a 10 CFR 50.59 evaluation and formal pre-job briefing package were developed. Operations personnel briefed the inspectors prior to the job and described their efforts to ensure a safe, orderly evolution.

The inspectors reviewed the licensee's plans and determined that all reasonable precautions were taken. A dedicated control room operator coordinated the manual control of the feedwater control valves while Maintenance personnel cleaned the pilot valves. The valves were repaired one at a time, exercised and placed back in automatic without incident. The repairs eliminated the oscillations on both control valves.

The inspector considered this evolution to be well planned, coordinated, and implemented. The decision to work the valves on-line rather than accept the continuing oscillations or challenge the system during a unit shutdown was conservative and appropriate.

Within the areas reviewed, no violations or deviations were identified.

3. Maintenance and Surveillance Testing (62703 and 61726)

a. Maintenance activities were observed and/or reviewed during the reporting period to verify that work was performed by qualified personnel and that approved procedures adequately described work that was not within the skill of the craft. Activities, procedures and work orders (WO) were examined to verify that proper authorization and clearance to begin work was given, cleanliness was maintained, exposure was controlled, equipment was properly returned to service, and limiting conditions for operation were met. The following maintenance activities were observed or reviewed in whole or in part:

(1) Inspection and Preventive Maintenance of Torque Switch Contacts (WO 94095115, PIP 94-1758)

On December 12, 1994, condensate supply valve 3C-391 to the Unit 3 Turbine Driven Emergency Feedwater (TDEFW) pump opened properly during a performance stroke test, but failed to close. Licensee Motor Operated Valve (MOV) technicians determined the problem to be dirty torque switch contacts. The MOV worked properly after cleaning the torque switch. This was the third occurrence of Unit 3 MOV failures due to dirty torque switch contacts since the last refueling outage. This issue was originally documented in NRC Inspection Report 94-11 as an Inspector Followup Item 94-11-02. The dirty torque switch contacts were attributed to the inadvertent omission of a step in the maintenance procedure which required their cleaning. Due to the time of the procedure change, this inadvertent preventive maintenance omission only affected Unit 3. As a result, there were 56 MOVs that did not receive the recommended torque switch cleaning.

When the issue was first identified, the licensee inspected the torque switch contacts on a representative sample (i.e., 5 MOVs in Unit 3). The inspections did not show a problem with dirty torque switch contacts so the licensee concluded that the remaining population of susceptible MOVs could wait until the next refueling outage (June 1995) for cleaning.

Due to this latest failure (valve 3C-391), the licensee inspected 25 additional MOVs. The inspections did not reveal any additional problems with dirty torque switches. On January 3, 1995, the inspector observed the inspection of Low Pressure Injection System valve 3LP-3. The inspection included as-found resistance readings across the torque switch for both the open and close direction. Following the resistance measurements the torque switch contacts were cleaned and the valve was fully stroked. The inspector concluded that the inspection and cleaning activities were adequate to identify torque switch problems and/or prevent torque switch failures for the valves included in this effort.

The inspector noted that there were 26 MOVs that were still not inspected. The licensee indicated that they could not perform the inspection on these valves at power due to potential plant transients or excessive dose. The inspector agreed with the licensee's rationale for not performing on-line inspections for these 26 MOVs, but expressed concern that the vulnerability could exist on these remaining valves for a prolonged length of time since the next scheduled outage for Unit 3 is June in 1995. The licensee stated that they would evaluate the possibility of placing some or all of these valves on the Unit 3 hot list, for torque switch inspection/cleaning following any Unit 3 reactor trip or forced outage. The inspectors will continue to track this issue under IFI 94-11-02.

(2) Inspection and Maintenance of Standby Shutdown Facility 4.16 KV Breaker OTS1-1 (WO 94065185)

On January 9, 1995, the inspector observed preventive maintenance activities associated with circuit breaker OTS1-1. Activities observed included main and auxiliary contact cleaning, lubrication of moving parts, and measurement of critical tolerances. All activities observed were satisfactory.

(3) Reactor Building Pressure Instrument Power Supply Replacement (TI/0/A/150/03)

There are two channels of Post Accident Monitoring (PAM) reactor building pressure indication for each unit. Testing in December 1994 revealed that the power supplies for these transmitters on Unit 3 had low output voltages (22.86 Vdc versus the required 28 Vdc). An engineering evaluation determined that the channels were conditionally operable, in that they were able to produce the minimum required voltage for the full span of indication. The licensee determined that the cause of the low voltage was aging capacitors internal to the power supply, and therefore, the power supplies had to be replaced at the earliest opportunity. On January 19, 1995, the inspector observed the replacement of the Unit 3 Channel "A" power supply and the subsequent test of the instrument string for proper calibration. All activities observed were satisfactory. The inspector noted that the replacement power supply was a different model than the original. The inspector reviewed the licensee's "acceptable substitute" evaluation and found it to be satisfactory.

(4) Clean Out 2A Component Cooler Tube Side (WO 94064439)

The inspector reviewed the work package and activities in progress associated with the cleaning of the 2A component cooler. The cleaning activity was associated with the low pressure service water (LPSW) side of the cooler. During the cleaning activity, the spare component cooler shared between Units 1 and 2 was placed in service on Unit 2 to provide component cooling for the Unit 2 loads. Work activities observed were accomplished in accordance with approved procedures. No discrepancies were noted.

(5) Conduct Performance Test on Battery SY1 (WO 94068053)

The inspector reviewed the work package and activities in progress associated with the performance test of switchyard battery SY1. The maintenance activity was accomplished in accordance with procedure IP/0/A/3000/023A, 125 VDC 230 KV Switchyard Battery Performance Test. The procedure disconnects the battery, then connects an external load to the battery and performs a discharge test of the battery for approximately eight hours to determine battery capacity. Work activities observed were accomplished in accordance with the procedure. No discrepancies were noted.

(6) Inspect and Repair 2A CCW Pump Strut (WO 94053189)

Beginning January 18, 1995, the inspector observed portions of the inspection and repair of the support struts on the 2A Condenser Circulating Water (CCW) pump. The work effort required removal of the pump motor, shaft and core barrel. Included in the strut inspection and repair was the replacement of the pump top seal. The inspector observed that this seal had a significant leak prior to the pump shutdown.

The inspector reviewed the strut repair in progress on January 23, 1995. The procedure utilized for the welding was MP/0/A/1800/072, Structural Steel - Miscellaneous Steel - Non-QA - Welding Repair and Modification, Task MM-OT-6069. The use of a non-QA welding procedure for the welding repairs of the struts which support the pump shaft to the core barrel was questioned by the inspector since the pump was considered safety-related. The licensee stopped the welding activity and issued a QA procedure for the welding work effort. However, the licensee stated that other work not associated with the welding (i.e., the removal and replacement of the pump and the installation of new packing) would be done with non-QA procedures because QA procedures did not exist.

The licensee had made a commitment in their March 14, 1994 response to the Service Water Inspection (IR 50-269,270, 287/93-25) to have a plan developed to properly classify equipment and upgrade procedures by April 15, 1994. Although a plan was initiated, the inspector noted that a clear definition of the work scope with completion dates had not been included. Although the licensee's Nuclear System Directive 307, "Quality Standards Manual" had been updated to identify certain systems/components (including the CCW pumps) as performing safety-related functions, the procedures associated with these components, including the CCW pumps, had not been reclassified to a QA status. This issue will be discussed further during a DPC/NRC Management meeting scheduled for February 6, 1994.

(7) Meteorological Equipment Checks (IP/O/B/1601/003)

On January 3, 1995, the inspector reviewed activities in progress during the weekly performance of IP/O/B/1601/003 for documenting and maintaining the meteorological equipment. The inspector noted that the work was authorized per an acceptable WO. The inspector concluded that the work was performed in accordance with the procedure and was of good quality.

(8) NI-3 Neutron Flux Instrumentation Calibration (IP/0/A/0301/003C-1)

The inspector reviewed efforts in progress on January 12, 1995, associated with the calibration of the Unit 2 NI-3 Gamma-Metrics wide range neutron flux monitoring channel. The work was completed in accordance with the procedure and the data forms were properly documented. The inspector verified that all calibration equipment was within the calibration due date as required, and that the activity had been authorized per an acceptable WO. The activity was determined by the inspector to have been performed to acceptable standards.

(9) Engineered Safeguards System Analog Channel C On Line Calibration (IP/0/A/0310/014C)

On January 4, 1995, the inspector reviewed the monthly testing and calibration of the Unit 2 Engineered Safeguards Channel "C" Reactor Building and reactor coolant pressure instrument components. The inspector verified that the WO associated with this activity was written in sufficient detail and had the appropriate authorizations. The inspector concluded that the work effort was performed per the procedure and was properly documented.

b. The inspectors observed surveillance activities to ensure they were conducted with approved procedures and in accordance with site directives. The inspectors reviewed surveillance performance, as well as system alignments and restorations. The inspectors assessed the licensee's disposition of any discrepancies which were identified during the surveillance. The following surveillance activities were observed or reviewed:

(1) Low Pressure Injection Pump Test (PT/3/A/0203/06A)

On January 18, 1995, the inspector witnessed the quarterly test of the 3A Low Pressure Injection Pump. All activities observed were satisfactory and all pump performance parameters were within the acceptance criteria.

(2) Unit 3 Penetration Room Ventilation System (PRVS) Monthly Test (PT/3/A/0170/05)

On January 10, 1995, the inspector witnessed the performance of the monthly test for the Unit 3 PRVS. This test demonstrated that the PRVS would operate at design flow and verified the stroke time of various valves in the system. All acceptance criteria were met and all activities observed were satisfactory.

(3) Control Rod Movement (PT/1/A/600/15)

The inspector witnessed the monthly performance test conducted on the Unit 1 control rods. The performance test implemented the requirements of Technical Specification (TS) 4.1.2, Minimum Equipment Test Frequency. This TS requires that each control rod be exercised monthly. To verify proper operation, the test procedure exercised groups 1 through 6 approximately 10 percent and groups 7 and 8 approximately 2.5 percent. Previous performances of the procedure required that all rods be moved 2.5 percent. The licensee increased the rod movement requirement on groups 1 through 6 to flush the control rod drive mechanism check valves in order to prevent corrosion buildup, which previously resulted in slow rod drop times. The licensee reduced power to less than 96 percent to perform the rod movement test. The inspector concluded that the test procedure was conducted in a controlled and professional manner by the operations staff. No difficulties were encountered during the performance of the test procedure.

(4) Control Rod Drive System (PT/2/A/600/15)

The inspector reviewed activities in progress during the movement of the Unit 2 control rods on January 19, 1995. This was essentially the same test as the one for Unit 1 described in paragraph 3.b.(3) above. The licensee had intended to change the Control Rod Movement procedure to move the rods 10 percent versus 2.5 percent, as they did for the Unit 1 test. Prior to the rod movement exercise, the licensee decided against moving the rods 10 percent and stayed with the original 2.5 percent. The licensee's decision was based on the fact that the feedwater control system was operating at a wider control band than desired. When combined with this problem, the larger movement of the control rod could have upset the system. The inspector concluded that the activity was performed to acceptable standards.

(5) Performance Test On Battery 2CA (IP/0/A/3000/023)

The inspector witnessed performance testing of the Unit 2, 2CA 125 Vdc Instrument and Control Battery on January 12, 1995. The results of this test showed the battery to be at 80.1 percent capacity. The Design Basis Document for the 125 Vdc Vital Instrumentation and Control System (OS-0254.00-00-2006) and Test Acceptance Criteria ONTC-0230-001-002, required that the battery capacity be above 80 percent.

Testing of the Unit 2, 2CB 125VDC Vital Instrumentation and Control Battery was performed on January 4, 1995. The inspector reviewed the associated test results, which revealed this battery to be at 77 percent capacity. Because the capacity factor was below 80 percent, the licensee performed an operability evaluation (PIP 2-095-0013). The 2CB battery was determined to be operable based on its test voltage profile exceeding the minimum voltage required to meet the load voltage demands during a design basis event. However, the licensee placed certain restrictions on Unit 2 battery operation/lineup. Specifically, the electrical alignment was prohibited from having the 2CB battery from being the sole supply for a unit (DC systems separated), and no cells could be removed from any of the control batteries.

The licensee reported that there were no spares for the existing batteries, Exide FTC 23 (lead-calcium type), and that the batteries were no longer manufactured by Exide. A special retooling by Exide was necessary to manufacture these when installed in January 1993. (Note: these batteries are approximately 2 years old). It was further reported by the licensee that spare batteries were not purchased, because of problems encountered in maintaining vendor requirements for the stored batteries.

Since both batteries were replaced in January 1993, the tests referenced above were the initial performance tests as required by IEEE Standard 450. The 1993 replacement was a result of testing performed in July 1992, which revealed the 2CA battery to be at 72 percent and the 2CB battery at 76 percent capacity. (Note: As discussed in Inspection Report 92-18, these tests in 1992 were reported to have been the first performed on the batteries.) At that time, an operability evaluation was performed by the licensee which determined the batteries to be operable. However, the licensee committed to bringing the batteries up to 80 percent capacity within 6 months and to replace them within one year.

As addressed in Inspection Report 92-18, the IEEE 450 Standard includes requirements for performing battery capacity tests every five years and annual capacity tests on batteries that show signs of degradation and/or battery capacity less than 85 percent. The standard further requires that the batteries be replaced if battery capacity is less than 80 percent.

The licensee reported that they were attempting to purchase replacements for those battery cells that tested below the acceptable limits. However, they indicated that this procurement probably could not be accomplished in the immediate future since the manufacturer requires special setup for these obsolete batteries. Although the licensee reported that the vendor representative suggested that additional testing was an option that should be considered, additional testing of these batteries is not planned in the immediate future. As of the end of the inspection period, the licensee was still evaluating their long-term actions.

This issue will be tracked as an Inspector Followup Item: 50-270/95-01-01, Unit 2 Control Battery Capacities.

Within the areas reviewed, licensee activities were satisfactory.

4. Onsite Engineering (37551)

During the inspection period, the inspectors assessed the effectiveness of the onsite design and engineering processes by reviewing engineering evaluations, operability determinations, modification packages and other areas involving the Engineering Department.

a. 10 CFR 50.72 Notifications

(1) Unreviewed Safety Question on Main Steam System

NRC Inspection Report 93-31 documented a concern regarding the potential to blow down a unit's steam generators from a break in the auxiliary steam header (Deviation 50-269,270,287/93-31-01). A conference call between NRC (Region II and NRR) and the licensee was conducted on March 9, 1994, to discuss whether the potential for a single failure to blow down both steam generators constituted an unreviewed safety question (USQ). During the call, the licensee maintained that the vulnerability in question did not constitute an USQ because it was bounded by the Final Safety Analysis Report (FSAR) Chapter 15 steam line break. The licensee agreed to provide the NRC their engineering analysis which provided the basis for concluding that there was no USQ, and that no corrective action was necessary. NRR subsequently reviewed the licensee's analysis and determined that the postulated event involved an USQ per 10 CFR 50.59, in that it presented the possibility for an accident of a different type than any evaluated in the safety analysis report.

On January 6, 1995, the licensee was provided with the results of NRR's review. On January 9, 1995, a conference call was held between NRC (Region II and NRR) and the licensee. During the call, the licensee stated that they would conservatively consider the vulnerability to be an USQ, pending further review. The licensee's immediate corrective actions were to close one of the two steam supply valves to the common steam headers (auxiliary steam and Turbine Driven Emergency Feedwater supply) on all 3 units. This effectively eliminated the vulnerability in question. Additionally, the licensee made a one-hour report in accordance with 10 CFR 50.72(b)(ii)(B) and agreed to provide a revised response to Deviation 50-269,270,287/93-31-01 by February 9, 1995. As of the end of the inspection period the licensee was still evaluating what long-term actions they might take regarding this issue. The inspectors concluded that the licensee's actions were adequate to eliminate the vulnerability in question. The inspectors will continue to track this issue under Deviation 50-269,270,287/93-31-01.

(2) Babcock & Wilcox Identifies Error in the Emergency Core Cooling System (ECCS) Evaluation Model

On January 26, 1995, the licensee reported to the NRC via a red phone call that Babcock and Wilcox (B&W) had identified a potential safety concern regarding the large break Loss of Coolant Accident (LOCA) analysis in the generic analysis for B&W designed plants. B&W had discovered an error in the non-conservative direction which resulted in an error of greater than 50 degrees F in the final peak clad temperature, a condition requiring notification of the NRC under 10 CFR 50.72(b)(1)(ii)(B). The analysis was preliminary and generic, but indicated that the actual peak clad temperatures could exceed 2200 degrees F assuming the worst case initial conditions at the beginning of the assumed LOCA. B&W suggested ameliorating this by reducing the operating band for axial imbalance, a measure of relative power levels in the top and bottom halves of the core. The axial imbalance limits are specified in the Core Operating Limits Report (COLR).

The licensee addressed this issue by restricting the allowable axial imbalance which could be present at the beginning of the event. The new, restrictive limits for axial imbalance were imposed after review by the Plant Operations Review Committee (PORC), and implemented by a Conditional Operability Evaluation. The new limits are considered temporary until B&W completes their analysis for Oconee and provides a new initial condition limit for axial imbalance.

The inspector attended the PORC meeting, and reviewed the Conditional Operability Evaluation and the instructions provided to the control room operators. The conclusions reached and the actions taken were conservative and thorough.

(3) Retraction of Previous Notification Regarding Single Failure Vulnerability of Reactor Building Spray

During the previous reporting period, the license made a notification to the NRC in accordance with 10 CFR 50.72.(b).(1).(ii).B, identifying that the plant abnormal procedures required that both trains of reactor building spray be secured assuming a single failure disables one of the two reactor building sump lines, and that this condition was beyond the design basis assumptions outlined in the Maximum Hypothetical Accident Safety Evaluation Report (see NRC Inspection Report 94-38).

Subsequent to the notification, the licensee performed an analysis and determined that the 10 CFR Part 100 Dose Limits would not have been exceeded assuming both trains of reactor building spray were secured at the time of switchover to the reactor building sump. Consequently, the notification to NRC was determined not to be required, and the licensee retracted it on January 17, 1995. The licensee plans to maintain the procedure changes originally implemented to correct the potential inoperability problem in effect.

Within the areas reviewed, licensee activities were satisfactory and no violations or deviations were identified.

5. Plant Support (71750)

The inspectors assessed selected activities of licensee programs to ensure conformance with facility policies and regulatory requirements. During the inspection period, the following areas were reviewed: Radiological Controls, Physical Security and Fire Protection.

During the week of January 13, 1995, the inspector witnessed portions of the activities associated with a Unit 3 spent fuel cask load. Activities observed included spent fuel assembly re-shuffle, cask movement, and cask loading. All activities observed were satisfactory. The inspector noted that spent fuel pool water clarity was significantly improved compared to previous evolutions.

No violations or deviations were identified.

6. Inspection of Open Items (92902 and 92903)

The following open items were reviewed using licensee reports, inspection record review, and discussions with licensee personnel, as appropriate:

a. (Closed) Unresolved Item 50-269/93-26-01: Load Shed System Not Single Failure Proof

This is the same issue as that described in LER 50-269,270,287/93-09 (see paragraph 7.a below). The corrective actions described in the LER were found to be satisfactory.

b. (Closed) Unresolved Item 50-287/94-01-01: Improperly sized High Pressure Injection (HPI) Orifice Plates.

The licensee discovered on January 15, 1994, that the normal and emergency injection pressure breakdown orifice in the Unit 3, 3B2 injection line was a 7/8 inch (.875) diameter orifice versus the required 0.78 inch diameter. The discovery resulted from an inspection by the system engineer to determine why the flow rates were higher in the Unit 3 system than those in Units 1 and 2. Three other Unit 3 orifices were inspected and found to be the wrong size (.875 versus 0.78 inch).

The licensee installed the correct orifice plates and performed a past system operability evaluation which indicated the equipment had been operable with the larger orifice plates. Based on the operability evaluation and good effort by the licensee to detect and correct the problem, this issue is closed.

7. Review of Licensee Event Reports (92700)

The below listed Licensee Event Report (LER) was reviewed to determine if the information provided met NRC requirements. The determination included: adequacy of description, compliance with Technical Specification and regulatory requirements, corrective actions taken, existence of potential generic problems, reporting requirements satisfied, and the relative safety significance of each event. The following LER was closed:

a. (Closed) LER 269,270,287/93-09: Design and Installation Deficiencies in Load Shed Circuitry Result in Technical Specification Violations

This LER involves two issues. The first issue involves the incorrect wiring of the load shed channel 1 slave relay in switchgear 3TD going undetected and the channel being inoperable from March 1987 to August 1993. This issue will be closed out in the future under Violation 50-287/93-24-01.

The second issue associated with this LER was discovered as part of the licensee's review for the above issue. The licensee discovered that the TD switchgear load shed channels for all three units were not single failure proof. Channel 1 of load shed was powered from 125 Vdc panel board DIA, but one set of relay contacts required to actuate channel 1 load shed was from a relay (RSL2X) powered from DIB. Additionally, channel 2 of load shed was powered from DIB, but one set of contacts required to actuate channel 2 load shed was from a relay (RSL1X) powered from DIA. This problem only existed for bus TD. Therefore, any single failure that would deenergize DIA or DIB, would defeat both load shed channels for the TD switchgear. This condition had existed since plant construction.

The licensee was able to eliminate the single failure vulnerability by modifying the wiring on relays RSL1X and RSL2X such that all the contacts for these relays were in the appropriate string of load shedding circuitry. The licensee performed an engineering evaluation to determine the effects of a design basis accident coincident with a failure of load shedding on the TD switchgear on one unit. The evaluation concluded that the additional loads resulting from failure of TD to load shed would not have exceeded the capability of the Keowee underground feeder path, including the CT-4 transformer. The inspectors reviewed the licensee's evaluation and concluded that it was adequate. Additionally, the inspectors witnessed the wiring modifications as they were performed as well as the post modification tests. All activities observed were satisfactory.

8. Exit Interview

The inspection scope and findings were summarized on February 1, 1995, with those persons indicated in paragraph 1 above. The inspectors described the areas inspected and discussed in detail the inspection findings in the summary and listed below. The licensee did not identify as proprietary any of the material provided to or reviewed by the inspectors during this inspection.

Item Number	Description/Reference Paragraph
50-270/95-01-01	INSPECTOR FOLLOWUP ITEM: Unit 2 Control Battery Capacities, paragraph 3.b.(5).