UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

April 28, 2000

NOED 99-2-003
EA 2000-056
EA 2000-068

Duke Energy Corporation
ATTN: Mr. G. R. Peterson
Site Vice President
Catawba Nuclear Station
4800 Concord Road
York, SC 29745

SUBJECT: NRC INTEGRATED INSPECTION REPORT NOS: 50-413/00-02 AND 50-414/00-02, AND NRC OFFICE OF INVESTIGATION REPORT NO. 2-1999-012

Dear Mr. Peterson:

This refers to the inspection conducted February 13 through April 1, 2000, at the Catawba facility. The enclosed report presents the results of this inspection.

During the inspection period, your conduct of activities at the Catawba facility was generally characterized by safety-conscious operations, sound engineering and maintenance practices, and careful radiological work controls.

Based on the results of this inspection, the NRC has determined that five violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII.B.1 of the Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the violations or severity level of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II, the Resident Inspector at the Catawba Nuclear Station, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In addition, on January 31, 2000, the NRC Office of Investigations (OI) completed an investigation regarding falsification of firewatches in March 1999, at your facility. The synopsis of OI Investigation Report No. 2-1999-012 is enclosed for your information. Section F8.1 of the enclosed inspection report provides further details on this matter and the resultant NCV.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and any response will be placed in the NRC Public Document Room.

Sincerely,

/RA/

Charles R. Ogle, Chief
Reactor Projects Branch 1
Division of Reactor Projects

Docket Nos. 50-413, 50-414
License Nos. NPF-35, NPF-52

Enclosures: 1. NRC Inspection Report Nos: 50-413/00-02 and 50-414/00-02
2. Synopsis of NRC Office of Investigation Report 2-1999-012

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-413, 50-414
License Nos.:	NPF-35, NPF-52
Report Nos.:	50-413/00-02, 50-414/00-02
Licensee:	Duke Energy Corporation
Facility:	Catawba Nuclear Station, Units 1 and 2
Location:	422 South Church Street Charlotte, NC 28242
Dates:	February 13 - April 1, 2000
Inspectors:	D. Roberts, Senior Resident Inspector R. Franovich, Resident Inspector M. Giles, Resident Inspector E. Lea, Project Engineer (Section F8.1) J. Coley, Reactor Inspector (Section M1.2, M7.1) S. Shaeffer, Senior Resident Inspector - McGuire (Section E8.1)
Approved by:	C. Ogle, Chief Reactor Projects Branch 1 Division of Reactor Projects

EXECUTIVE SUMMARY

Catawba Nuclear Station, Units 1 and 2
NRC Inspection Report 50-413/00-02, 50-414/00-02

This integrated inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a seven-week period of resident inspection, as well as the results of an announced inspection by one regional inspector and the Senior Resident Inspector from the McGuire Nuclear Station. In addition, an in-office review was conducted by a Region II project engineer [Applicable template codes and the assessment for items inspected are provided below.]

<u>Operations</u>

- An automatic Unit 1 reactor trip occurred on February 13, 2000, due to the failure of a pin connector on the turbine electrical trip solenoid valve. Observations of control room activity and review of the plant's response identified no operator or system deficiencies. The pin connector was the same type connector that had failed and caused an automatic Unit 2 trip on December 30, 1999. The actions specified in the licensee's corrective action program in response to the Unit 2 trip were reasonable despite the similar failure and subsequent reactor trip on Unit 1. (Section O2.1; [POS -1A; NEG - 2A])

- A non-cited violation was identified regarding a non-compliance with Technical Specification 3.8.4 and 3.8.1 when two emergency diesel generator battery bank cells were found to be below the minimum Technical Specification required voltage, but were dispositioned as a routine out-of-tolerance condition. Consequently, the 1A emergency diesel generator remained inoperable while the unit was operating in Mode 1 from the time of the initial discovery (October 2, 1999) until the problem was resolved (October 9, 1999). (Section O8.1; [NCV - 1A, 2B, 3A])

<u>Maintenance</u>

- Observed inservice examination activities were performed using approved procedures by certified examiners. The inspection results were properly recorded and evaluated in accordance with the appropriate test procedures. The Code repair and replacement packages reviewed by the inspectors, were complete and met American Society of Mechanical Engineers Sections XI and V Code requirements. (Section M1.2; [POS - 2B])

- A non-cited violation of Technical Specification 5.4.1 was identified regarding an inadequate procedure for removing inoperable nuclear service water pumps from service. Specifically, prior to February 23, 2000, the licensee's procedure for removing one pump from service did not take into account statements contained in the Technical Specification 3.7.8 Bases for maintaining a train of nuclear service water operable with both units operating in Mode 1. (Section M8.2; [NCV - 1A, 4B])

- A non-cited violation was identified for failure to implement required actions of Technical Specifications 3.7.12 and 3.0.3 regarding the Unit 1 auxiliary building filtered exhaust ventilation system (ABFVES). Specifically, after maintenance was performed on inlet vortex damper 1ABFD-13 on June 16, 1999, the 1A train of the ABFVES was made inoperable and remained inoperable until August 5, 1999, without the appropriate plant actions performed as required by Technical Specifications 3.7.12. In addition, while the 1A train of the ABFVES was inoperable from June 16, 1999, to August 5, 1999, the 1B train of the ABFVES was also inoperable for approximately 14 hours on June 29, 1999, and for approximately 8 hours on July 27, 1999, without the appropriate plant actions

performed as required by Technical Specifications 3.0.3. (Section M8.3; [NCV - 1A, 2B])

- The licensee did not perform an in-depth review of the status of the 1B auxiliary building filtered exhaust ventilation system train during the period of time that the 1A train was inoperable. NRC inspectors identified four occasions in which Unit 1 did not have an operable train, which constituted violations of Technical Specification 3.0.3. The licensee subsequently issued Revision 1 to Licensee Event Report 50-413/99-15 to address the safety significance of this discovery. (Section M8.3; [NEG - 4C, 5B])

Engineering

- A non-cited violation of 10 CFR 50, Appendix B, Criterion V, was identified concerning Unit 1 ice condenser basket coupling screws that were found to be missing during the End-of Cycle 11 refueling outage (April 21, 1999, - May 23, 1999) and, therefore, not installed in accordance with station drawings. (Section E8.1; [NCV - 4A])

Plant Support

- A non-cited violation of Technical Specification 5.4.1 was identified for missed fire watch tours between March 9 - 28, 1999, that had been established for degraded fire barrier penetrations. The tours were not completely performed by four individuals who had indicated on hourly fire watch logs that they had completed them. In three of the four cases, the violations were determined to be willful. (Section F8.1; [NCV-1C, 3A])

- The licensee took appropriate actions to identify and correct the instances of willfully missed fire watch tours in March 1999. (Section F8.1; [POS - 5A, 5C])

Report Details

Summary of Plant Status

Unit 1 began the inspection period at 100 percent reactor power. A reactor trip from 100 percent occurred on February 13, 2000, from a failed pin connector associated with the secondary electrical trip solenoid valve. Following repair activities, a reactor startup was commenced on February 14, 2000, with the unit entering Mode 1 the same day. Reactor power was increased to approximately 58 percent when a hydraulic oil leak was discovered on 1CF-60, the D steam generator feedwater system containment isolation valve. Reactor power was reduced to 16 percent to allow for repair and subsequent post-maintenance testing of the valve. Power ascension was commenced on February 16, 2000, to 100 percent reactor power. The unit reached 100 percent reactor power the same day, and remained at full power for the duration of the inspection period.

Unit 2 began the inspection period at 100 percent reactor power. Reactor power was reduced to approximately 94.5 percent on March 8, 2000, to support main steam safety relief valve testing. The unit was shutdown from 94.5 percent power on March 11, 2000, and cooled down to Mode 5 (cold shutdown) the same day to begin the End-of-Cycle 10 refueling outage. The unit entered Mode 6 (reactor vessel head studs de-tensioned) on March 15, 2000, and the core was fully off-loaded on March 19, 2000. Following refueling activities, the unit was returned to Mode 5 conditions on April 1, 2000, and remained there for the duration of the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors conducted frequent control room tours to verify proper staffing, operator attentiveness and effective communications, and adherence to approved procedures. The inspectors: (1) attended operations shift turnovers and site direction meetings to maintain awareness of overall plant status and operations; (2) reviewed operator logs to verify operational safety and compliance with Technical Specifications (TS); (3) periodically reviewed instrumentation, computer indications, and safety system lineups, along with equipment removal and restoration tagouts, to assess system availability; (4) reviewed the TS Action Item Log (TSAIL) for both units daily for potential entries into limiting conditions for operation (LCO) action requirements; (5) conducted plant tours to observe material condition and housekeeping; (6) routinely reviewed Problem Investigation Process reports (PIPs) to ensure that potential safety concerns and equipment problems were being addressed; (7) observed the Unit 1 and Unit 2 startup and shutdown activities, respectively, to assess operator and plant performance; and (8) observed reduced inventory and mid-loop operations for Unit 2 to verify that the licensee established adequate defense-in-depth and contingencies for maintaining core cooling and minimizing shutdown risk. The inspectors identified no significant problems or concerns from the above reviews or observations.

O2 Operational Status of Facilities and Equipment

O2.1 Automatic Reactor Trip Due to a Short in Main Turbine Electrical Trip Solenoid Connector

a. Inspection Scope (93702, 71707, 37551, 40500)

On February 13, 2000, at 6:31 p.m., Unit 1 experienced an automatic trip from 100 percent power. Prior to the trip, the unit was operating at normal steady-state conditions. The inspectors responded to the site to ensure plant conditions were stable with the unit in hot standby (Mode 3). Control room annunciators indicated that this automatic trip appeared very similar to the Unit 2 automatic reactor trip that occurred on December 30, 1999, [reported in Licensee Event Report (LER) 50-414/99-006-00, and documented in NRC Inspection Reports 50-413,414/00-01 and 99-08]. The inspectors reviewed PIP C-00-00615 and C-99-05255 (for the Unit 2 trip), the licensee's post-trip report, and the Updated Final Safety Analysis Report (UFSAR), Chapter 15, Accident Analysis, to assess operator actions and verify that plant equipment responded appropriately to the Unit 1 trip.

b. Observations and Findings

Following the reactor trip, control room operators entered E-0, Reactor Trip or Safety Injection, and subsequently transitioned to ES 0.1, Reactor Trip Response, for post-trip recovery actions. The inspectors determined that the unit was stable in Mode 3. No significant deficiencies in plant conditions were identified.

Based on the earlier Unit 2 trip, which was caused by a short within the electrical pin connector attached to the turbine electrical trip solenoid valve, and control room indications that suggested that this trip was of a similar nature, licensee troubleshooting activities were initiated to inspect the same pin connector on Unit 1. Engineering and maintenance personnel determined on February 14, 2000, that the trip was caused by a faulty pin connector on the turbine electrical trip solenoid valve, which failed open and reduced emergency trip system (ETS) pressure. This resulted in pressure switches on the ETS header sensing the low pressure and sending a signal to the solid state protection system that the turbine had tripped. This, in turn, caused the reactor trip. The failed pin connector was replaced along with a fuse and associated relays. Restart of Unit 1 was subsequently approved by the Plant Operations Review Committee (PORC) and the unit was restarted and placed on-line on February 15, 2000.

Due to the repetitive failure of this type pin connector, the inspectors reviewed PIP C-99-05255 to assess the adequacy of the corrective actions proposed following the Unit 2 trip. The proposed corrective actions indicated that the Unit 1 pin connector would be inspected during the next refueling outage and the feasibility of periodic inspections and replacement would be evaluated. Discussions with engineering personnel indicated that thermographic inspection was performed on the Unit 1 pin connector following the Unit 2 trip. The results did not indicate that a degraded condition existed. The inspectors concluded that these corrective actions were reasonable. This conclusion was further supported by the fact that satisfactory circuit integrity was being demonstrated through the turbine electrical trip solenoid valve being constantly energized to remain in the closed position.

The inspectors questioned the licensee about other installations of these pin connectors in order to understand how many are used in applications such that a failure could result in another reactor trip or failure of some safety-related component. The licensee indicated that an evaluation of risk-susceptible applications would be performed. Following this discussion, the licensee discovered that the root cause of the pin connector failure was due to the degradation of the insulating material used in between the pins. This degradation, which occurred because the insulating material was not rated for continuous use in applications where the environmental temperature was approximately 120-125 degrees Fahrenheit, reduced the resistance across the pins and allowed a short to occur in the connector. The failure mechanism was found through a microscopic examination of the pin connector in which cracks caused by the shorting condition were identified. This root cause differed from the root cause identified for the previous Unit 2 trip, which was attributed to loose solder in the pin connector. Following the discovery of microscopic cracks in the failed pin connector used on Unit 1, the failed pin connector used on Unit 2 was reevaluated. Microscopic cracks were also identified in the insulating material used in the pin connector. Based on this discovery, the licensee revised LER 50-414/99-006, changing the root cause to reflect the degraded insulating material.

The inspectors reviewed PT/0/A/4150/02, Revision 3, Transient Investigation, to assess the response of the plant as compared with that specified in the UFSAR, Chapter 15, Accident Analysis. No plant or system deficiencies were identified.

c. Conclusions

An automatic Unit 1 reactor trip occurred on February 13, 2000, due to the failure of a pin connector on the turbine electrical trip solenoid valve. This pin connector was the same type connector that had failed and caused an automatic Unit 2 trip on December 30, 1999. The actions specified in the licensee’s corrective action program in response to the Unit 2 trip were reasonable despite the similar failure and subsequent reactor trip on Unit 1. Observations of control room activities and review of the plant’s response identified no operator or system deficiencies.

O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) LER 50-413/99-017-00: Operation Prohibited by TS 3.8.4 and 3.8.1 concerning DC Power Supply to Diesel Generator 1A due to an Inadequate Procedure

The 125 volts direct current (VDC) auxiliary battery 1DGBA supplies 125 VDC essential auxiliary power to the 1A Emergency Diesel Generator (EDG). Normally on a float charge from its associated battery charger, 1DGBA supplies Class 1E power to various EDG loads and control loads. The battery consists of 94 nickel cadmium cells and is sized to carry its assigned loads for two hours.

On October 2, 1999, maintenance technicians performed IP/0/A/3710/017, Revision 30, Periodic Inspection And Maintenance For SAFT Model SBM277-2 Storage Battery, to satisfy TS surveillance requirement 3.8.4.2 for the 1A EDG. As part of this surveillance test, voltage readings were taken across the cells in 1DGBA, and were compared to the TS-required minimum voltage of 1.36 VDC while on a float charge. Maintenance technicians identified that the voltage readings across cells 9 and 74 were below the TS minimum limit. Voltages across all 94 cells were recorded on Enclosure 11.1, Calibration Checklist, with the unsatisfactory voltages circled for cells 9 and 74. This enclosure stated that the TS minimum limit was >1.36 VDC, with an administrative limit of > 1.40 VDC. Upon completion of this procedure, the maintenance technicians informed their supervision of the unsatisfactory voltage readings for the two affected cells and were directed to inform the shift work manager (SWM).

The technicians notified the SWM that two cells did not meet test acceptance criteria, but failed to clearly communicate to the SWM that these cells served a TS component and that the readings were below a TS minimum limit. The maintenance technicians placed copies of the recorded data in a drop box for a later engineering review of “out-of-tolerance” conditions. On October 6, 1999, engineering personnel responsible for documenting out-of-tolerance conditions generated PIP C-99-04053 for this discrepancy. The issue was reported to the cognizant system engineer, who, on

October 7, 1999, determined that the low cell voltages rendered the 1A EDG inoperable. Based on the evaluation, the engineer concluded that the 1A EDG had been inoperable since October 2, 1999, when the surveillance data was first collected. The 1A EDG was declared inoperable by operations on October 7, 1999, and placed on an equalizing charge for 24 hours. PIP C-99-04079 was generated to document the failure to properly declare the 1A EDG inoperable five days earlier. After the equalization charge, cell 9 voltage indication was 1.441 VDC, and cell 74 was 1.400 VDC. Because its voltage was right at the licensee's administrative minimum limit, cell 74 was jumpered out of the battery bank in accordance with a temporary modification. An accompanying engineering evaluation determined that battery 1DGBA could still perform its intended design function in this configuration. Operations declared battery 1DGBA operable on October 9, 1999. Additionally, PIP C-99-04087 was generated to determine why the two cells failed to retain the TS-required voltage.

The inspectors reviewed the completed surveillance test procedure, the work package, different training modules, and the licensee's root cause determination for this event. The licensee concluded that inadequate procedures resulted in the technicians failing to clearly communicate to the shift work manager that the battery cell voltages did not meet the TS minimum limit. The technicians processed the failure as an out-of-tolerance condition instead of a TS operability issue. Out-of-tolerance conditions were traditionally associated with instrumentation test criteria for which acceptable tolerance limits (that do not impact operability) were specified. The inspectors identified that, although the procedure contained misleading information concerning the handling of out-of-tolerance conditions, guidance was provided in the Limits and Precautions section, which was signed by the technician, to report immediately to the work supervisor any problem that renders the equipment inoperable. When interviewed by the inspectors, the technician stated that it was understood that there were TS implications associated with the test data and that it potentially affected EDG operability. The inspectors also identified training material for maintenance supervisors and acting supervisors that included a supervisory responsibility to notify the SWM of any TS operability issue immediately. The maintenance supervisor, who was in an acting status and had completed the training, delegated the duty of notifying the SWM to the technician. The inspectors have discussed the poor communications aspect of this event with licensee management. These aspects were being addressed in the licensee's corrective action program.

The inspectors reviewed the safety analysis performed by the licensee for the degraded battery bank. The analysis indicated that, even with two of the cells in battery 1DGBA slightly below the TS limit, it would still have been able to supply design basis accident loads. In the degraded condition, battery 1DGBA was capable of supplying a minimum of 112.2 VDC with a required battery bank voltage of 105 VDC. The inspectors reviewed the TSAIL and verified that the opposite train 1B EDG was operable during the period of time that the 1A EDG was inoperable. Consequently, the inspectors concluded that this event had minimal actual or potential safety consequences.

Technical Specification 3.8.4, DC Sources - Operating, requires the Train A and Train B EDG DC electrical power subsystems to be operable in Modes 1, 2, 3 and 4. Required Action C.1, specifies, with one EDG DC electrical power subsystem inoperable, that the licensee immediately enter applicable conditions and required actions of LCO 3.8.1, AC Sources - Operating, for the associated EDG made inoperable. LCO 3.8.1, Condition B, requires the inoperable EDG be restored to operable status within 72 hours, or perform the actions of Condition G, which requires the unit to be placed in Mode 3 within six hours, and in Mode 5 within 36 hours. On October 7, 1999, it was determined that battery 1DGBA and the associated 1A EDG had been inoperable for approximately five days. It was restored to an operable status on October 9, 1999. This period of inoperability exceeded that allowed by TS 3.8.1.

The licensee's failure to restore the 1A EDG within 72 hours or be in hot standby within the following six hours or cold shutdown in the next 36 hours constituted a violation of TS 3.8.1. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. This violation is included in the licensee's corrective action program in PIPs C-99-04053, 04079, and 04087 . This violation is identified as NCV 50-413/00-02-01: Failure to Comply with TS 3.8.4 and 3.8.1 with the 1A Emergency Diesel Generator Inoperable. This LER is closed.

O8.2 <u>(Closed) LER 50-413/00-001-00</u>: Reactor Trip Caused by a Pin to Pin Short Circuit within an Electrical Connector on the Turbine Electrical Solenoid Valve

This LER documented the Unit 1 automatic reactor trip discussed in Section O2.1 of this inspection report. No regulatory concerns were identified. This LER is closed.

O8.3 <u>(Closed) LER 50-414/99-006-01</u>: Reactor Trip Caused by an Electrical Ground in an Electrical Connector on the Turbine Electrical Trip Solenoid Valve

This LER was revised to include the results of a more detailed analysis of a failed pin connector associated with an ETS solenoid dump valve that caused a Unit 2 trip in December 1999. The more detailed failure analysis followed a similar failure on a Unit 1 valve that caused a unit trip in February 2000 (discussed in Section O2.1 of this report). No regulatory concerns were identified. This LER is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 <u>General Comments on the Conduct of Maintenance and Surveillance Activities (62707, 61726)</u>

The inspectors observed all or portions of the following maintenance and surveillance activities:

- PT/0/A/4150/19, Revision 19, 1/M Approach to Criticality
- OP/0/A/6550/011, Revision 24, Internal Transfer of Fuel Assemblies and Components
- MP/0/A/7450/048, Revision 6, Temporary Alteration of Station Dampers
- PT/2/A/4200/011, Revision 7, Emergency Boration Flow Rate Verification
- OP/2/A/6150/009, Revision 24, Boron Concentration Control
- MP /0/A/7150, Revision 21, Ice Basket Weight Determination
- SM/0/A/8510/007, Revision 10, Ice Basket Corrective Maintenance and Tracking

Maintenance and surveillance activities were performed using good workmanship, proper procedural adherence, and appropriate controls for using calibrated measuring and test equipment. Appropriate radiological practices were also observed where necessary.

M1.2 <u>Inservice Inspection (ISI) - Observation of Work Activities</u>

a. <u>Inspection Scope (73753)</u>

The inspectors observed liquid penetrant and ultrasonic examinations on four, 6-inch safety injection system pipe welds; ultrasonic examinations of the upper and lower circumferential welds for the volume control tank; and eddy current bobbin and rotating coil calibrations and subsequent examinations for the A and C steam generator tubing for Unit 2. In addition, the inspectors reviewed radiographic film and repair and replacement documentation for one completed Unit 1 corrective modification package, one Unit 2 minor modification package, and one Unit 2 corrective maintenance work order. These observations were performed to determine whether the ISI, and repair and replacement of Class 1, 2, and 3 pressure retaining components at the Catawba facility were performed in accordance with TSs, the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel (B&PV) Code (1989 Edition, no Addenda, Sections XI and V), and correspondence between NRC staff and the licensee.

b. Observations and Findings

For each method of examination observed above, the inspectors verified that approved procedures were being followed, examination personnel were knowledgeable of the examination method and operation of the test equipment, examination personnel with the proper level of qualification and certification were performing the examination activities, and examination results and evaluation of the results were recorded as specified in the applicable nondestructive examination procedure. No findings were identified during the examinations observed or as a result of the repair and replacement reviews.

c. Conclusions

Inservice examination activities observed were performed using approved procedures by certified examiners. The inspection results were properly recorded and evaluated in accordance with the appropriate test procedures. The Code repair and replacement packages reviewed by the inspectors, were complete and met ASME Sections XI and V Code requirements.

M7 Quality Assurance in Maintenance Activities

M7.1 Licensee Assessments of ISI Activities (73753)

The inspectors evaluated the effectiveness of licensee's controls for identifying, resolving and preventing problems in ISI by reviewing the corrective actions taken for items identified in Self Assessment No. SA-98-07. This assessment was conducted on the Catawba Unit 2, Outage 2 End of Cycle 9, ISI Plan. After thorough examination of the problems identified, the inspectors concluded that the licensee's controls were effectively identifying and resolving issues within the corrective action program.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (Closed) LER 413/99-016-00: Operation Prohibited by Technical Specification 3.8.1 and 3.7.8 Due to Inoperable Diesel Generator 1B for Greater than 72 Hours

(Closed) Notice of Enforcement Discretion (NOED) 99-2-003: Catawba Unit 1 Inoperable 1B Emergency Diesel Generator and Nuclear Service Water System

The above items were associated with three test failures (output breaker tripped open on overcurrent) of the 1B EDG after planned maintenance during the week of November 15, 1999. The first test failure was initially attributed to the maintenance effort, specifically, the replacement of eight heim joints on November 16, 1999 (see NRC Inspection Report 50-413,414/99-07). Following the second and third test failures after the new heim joints were verified to be installed correctly, the licensee determined that all three failures were caused by the improper operation of the EDG electronic governor assembly (EGA). The EGA was replaced and successfully tested on November 20, 1999, and the 1B EDG was returned to operable status at 11:09 p.m. that night.

Technical Specification LCO 3.8.1 requires the licensee to restore an inoperable EDG to operable status within 72 hours or place the reactor in Hot Standby (Mode 3) within the following 6 hours. Because the 1B EDG had been inoperable since 4:15 a.m. on November 16, 1999, the licensee requested a NOED from the NRC to allow an additional 48 hours to troubleshoot and repair the EDG. The NOED was granted based on the low risk implications of the extension and several compensatory measures implemented by the licensee (and verified by the inspectors). The EDG was declared operable prior to the expiration of the NOED. However, because the EDG was inoperable for greater than the TS-allowed 72 hours, the licensee reported the TS non-compliance in LER 50-413/99-016-00. The only additional corrective action planned by the licensee was to have the failed EGA sent to the manufacturer for a more detailed failure analysis. The inspectors identified no enforcement issues related to the failed component or the licensee's efforts to restore the EDG within the completion times specified in TS LCO 3.8.1. The EDG has successfully passed four consecutive surveillance tests since November 1999. Therefore, LER 50-413/99-016 and NOED 99-2-003 are both closed.

M8.2 (Closed) Unresolved Item (URI) 50-413/99-07-02: 1B EDG Inoperability Due to Successive Test Failures following Maintenance - NOED 99-2-003

The URI was opened as a tracking mechanism for the NOED described in Section M8.1 above and to track three separate followup issues related to the licensee's activities surrounding the 1B EDG failure.

Heim Joint Maintenance Procedure Issue:

The first issue was associated with certain sections of the EDG maintenance procedure MP/0/A/7400/001, Diesel Fuel Oil Injection Pump Removal, Replacement and Adjustment, Revision 24, which were not documented as having been performed. These sections included steps associated with the proper installation of the heim joints that had been replaced on November 16, 1999. The procedure sections in question specified the proper angle of alignment between the heim joints and the fuel rack linkages to which they were connected. It also documented as-left millimeter settings, a measurement of heim joint length with respect to the fuel racks. The licensee investigated the omitted steps and determined that these actions actually had been performed, but were simply not signed in the procedure because the steps appeared in sections related to fuel pump replacement, which itself was not being performed. The licensee agreed that these steps should have been initialed by the technicians. The procedure was revised to include sections specifically dedicated to heim joint replacement. The inspectors verified that the revised procedure properly included the steps in question. The inspectors, who had also observed some of the maintenance and reviewed the heim joint alignment following the first EDG failure, did not identify any misalignment between the joints and the fuel racks. The inspectors considered the licensee's followup activities and corrective actions adequate to address this documentation issue.

Common Mode Failure Determination Issue:

The second issue was a question of whether or not the licensee's common mode failure determination for the 1A EDG was adequate following the test failures of the 1B EDG. The inspectors concluded that the licensee's documentation of the common-mode failure determination was weak in that the evaluation was not documented until four or five days following the initial failure of the 1B EDG. Technical Specification 3.8.1 required that the licensee determine the absence of a common mode failure potential for the opposite train (operable) EDG within 24 hours or test the operable EDG within 24 hours. The licensee contended that engineers and operators had verbally discussed and ruled out the potential for common-mode failures throughout the troubleshooting efforts following each of the three test failures of EDG 1B, but acknowledged that the documentation of those determinations could have been enhanced.

Also at issue was whether or not the final determination itself was adequate. The licensee documented in its corrective action program (PIP C-99-04675) that the 1B EDG output breaker trips did not present a common mode failure potential due to the fact that the 1A EDG had been successfully passing its surveillance tests without any of the problems the 1B EDG had been experiencing. This reliance on previous successful surveillance tests did not address the specific failure mechanism for the 1B EDG, which itself had passed tests up to November 16, 1999. The licensee was planning to develop better guidance on how to adequately perform and document common mode failure determinations. The 1A EDG has been successfully tested several times since November 1999, and has not exhibited any of the EGA problems identified for the 1B EDG. No further inspection is planned for this item.

<u>TS 3.7.8 Compliance Issue</u>:

The third issue concerned the licensee's compliance with TS 3.7.8, Nuclear Service Water System. Upon removing the 1B EDG from service on November 16, 1999, the licensee also entered TS LCO 3.7.8 for the 1B nuclear service water (RN) pump. According to the TS Bases, an RN pump is considered inoperable when either its normal or emergency power source (the 1B EDG) is inoperable. The TS Bases further states that an RN train is considered operable with one unit's RN pump if one unit's RN system flowpath to the containment spray (NS) heat exchangers, the auxiliary feedwater (CA) system, and the non-essential header of the service water system is isolated (or equivalent flow restrictions). The inspectors noted that the operators had not verified that these RN flow paths were isolated for either unit. Because the Unit 1 and Unit 2 RN systems were cross-tied (normal alignment) with both units operating in Mode 1, the inspectors questioned the operability of the Unit 2B RN sub-train and its associated diesel generator given the degraded status of the 1B EDG and RN sub-train.

The operators had entered TS LCO 3.7.8 for Unit 2 in accordance with the governing operating procedure for removing the 1B RN pump from service, OP/0/A/6400/006C, Nuclear Service Water System, Revision 224, Enclosure 4.11. However, they had not made a similar TS 3.8.1 LCO entry for the 2B EDG, which itself required the 2B RN train to be operable and had a 1-hour required action to perform offsite power alignment verification. When questioned, operators explained that they were simply following the guidance in the operating procedure and did not feel as if the 2B RN pump's operability was actually compromised, even with the loads mentioned above not verified to be isolated as described in the TS Bases. After these questions were raised by the inspectors, and before the NOED was requested, the operators switched from the Enclosure 4.11 alignment to Enclosure 4.12 on November 19, 1999, which required them to verify the isolation of the above-mentioned RN loads. This facilitated the removal of Unit 2 from the TS 3.7.8 LCO condition. Unit 1 remained in the LCO condition until the 1B EDG and 1B RN pump were declared operable on November 20, 1999.

As indicated in Inspection Report 50-413,414/00-01, Section M7.1, the licensee subsequently established a multi-disciplined team to look at the interrelated aspects of RN/EDG/offsite power. This team recommended the performance of a system flow balance test to evaluate the 2B RN train's capability of supplying the 2B EDG during the non-isolated configuration that existed between November 16 - 19, 1999. The test results from test procedure PT/0/A/4400/008B, Revision 30, RN Flow Balance Train B, reiterated the importance of having to isolate the NS heat exchanger flow paths in order to maintain a train of RN operable when one train-specific RN pump was inoperable. The licensee acknowledged that, even though the NS heat exchangers are normally isolated during Mode 1 operations, there are occasions that they could become unisolated due to routine flushing or flow testing activities. The inspectors subsequently identified one such case on November 18, 1999, in which the 1B NS heat exchanger was unisolated for several hours during the period that it was assumed to be isolated to comply with the TS Bases. To address that condition, the licensee contended, without performing an additional detailed flow analysis, that even with RN flow (on the order of 4500 gallons per minute) through this heat exchanger while the 1B RN pump was technically inoperable, there still would have been enough flow from the operable RN pumps to the 2B EDG for it to perform its intended function during an accident. This was a result of other RN system loads [like the inoperable 1B EDG and the CA system backup supply piping] being isolated during the time of consideration.

The inspectors ultimately concluded that operating procedure OP/0/A/6400/006C, Revision 224, Enclosure 4.11 did not contain adequate guidance for maintaining compliance with the TS Bases when removing an inoperable RN pump from service. Specifically, adequate controls were not in place to ensure that at least one NS heat exchanger remained isolated when relying on one RN pump in a flow loop (Unit 1 and 2 same-train components comprised a loop) to meet the flow demands of operable EDGs in that loop. The licensee revised the enclosure on February 23, 2000, to ensure that flushing or flow activities that could un-isolate the NS heat exchangers are not ongoing when an RN pump is inoperable. This restriction was also placed on any RN to CA pipe flushing/flow testing activities. The licensee also plans to revise the TS Bases to eliminate the statement implying that both an emergency and a normal power source are required for an RN pump to be operable. This revision would be consistent with the definition of "operable" in the front of the TS, which indicates that either the normal or emergency electrical power source is sufficient to maintain the supported system operable.

Technical Specification 5.4.1 requires that written procedures shall be established, implemented, and maintained covering the activities in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978; which includes procedures for operation and shutdown of the safety-related service water system (Item 3.m). Contrary to the above, prior to February 23, 2000, the operating procedure for removing an inoperable RN pump from service, OP/0/A/6400/006C, Revision 2, Enclosure 4.11 was inadequate in that it did not ensure that system configuration would be maintained in accordance with statements contained in the TS 3.7.8 Bases for considering a train of RN operable with one unit's train-related pump inoperable. This could have resulted in degraded conditions for those components supported by RN and assumed to be operable. This Severity Level IV violation is being treated as an NCV, consistent with Section VII.B.1 of the NRC Enforcement Policy. It is identified as NCV 50-413,414/00-02-02: Inadequate Operating Procedure Used to Maintain RN System Configuration Control When

Removing One Pump from Service. This issue was included in the licensee's corrective action program in PIP C-00-00355. The URI is closed.

M8.3 (Closed) LER 50-413/99-015-(00,01): Inoperability of Auxiliary Building Ventilation System in Excess of Technical Specification Limits Due to Improperly Positioned Vortex Damper

On June 16, 1999, maintenance was performed on auxiliary building filtered ventilation exhaust system (ABFVES) vortex damper 1ABFD-13 in accordance with MP/0/A/7450/019, Revision 010, Clarage Type A.F.P. 1550A Series Fan Corrective Maintenance. The maintenance consisted of cleaning and lubricating the vortex damper gear drives. These gear drives position their associated damper blade, which in conjunction with the other damper blades, control the amount of air that is introduced into the ventilation system fan unit. In order to access the damper gear drives, the damper gear retaining disc and the gear drive mechanism had to be removed by technicians. Following completion of the maintenance, technicians had reinstalled the gear drive mechanism and the damper gear retaining disc. Operations subsequently restored the vortex damper to service.

On August 2, 1999, during TS Surveillance Requirement (SR) 3.7.12.4 testing of the auxiliary building filtered ventilation exhaust system, performed in accordance with PT/0/A/4450/04A, Revision 43, Auxiliary Building Filtered Exhaust System Performance Test, test personnel identified that train 1A failed to meet its acceptance criteria. Technical Specification SR 3.7.12.4 requires verification that one ABFVES train can maintain the emergency core cooling system (ECCS) pump rooms (Train A and B) at negative pressure relative to adjacent areas. Test results indicated that the 1A and 1B residual heat removal pump rooms, the 1B safety injection pump room, and the 1A chemical and volume control pump room were not being maintained at a negative pressure. Operators declared the ABFVES train 1A inoperable and entered it into the TSAIL. The licensee generated PIP C-99-03127 to document this test failure. Engineering personnel later identified that the ABFVES train 1A inlet vortex damper, 1ABFD-13, was nearly closed instead of being in the required throttled position. The damper was repositioned, and subsequent successful testing restored the train to an operable condition on August 5, 1999.

The auxiliary building filtered exhaust system exhausts potentially contaminated air from the auxiliary building, the ECCS pump rooms, and non-safety portions of the auxiliary building. The system consists of two independent and redundant trains for each unit. Upon receipt of a safety injection signal, non-safety-related portions of the system are isolated, and only air taken from the ECCS pump rooms is exhausted through a filter package, a heater/demister, and the train's associated fan and ductwork.

The inspectors reviewed the completed work package, the PIP associated with this event, performed system walkdowns with the cognizant system engineer, and reviewed procedure revisions of the governing maintenance procedures. The inspectors concluded that procedures used to perform the original maintenance did not contain sufficient guidance to ensure that when the damper was reassembled, the as-left position of all mating gears integral to the damper would be able to provide normal (full-open) and post-accident (throttled) damper blading positions as required. This conclusion was consistent with the root cause determination performed by the licensee. The inspectors noted upon reviewing the revised maintenance procedure that as-left blading position was required to be recorded in the normal position, and if blading position was changed in the post-accident alignment, readjustment would be required to ensure that normal blading position would be achieved. The inspectors concluded that these and other procedural corrections should prevent recurrence of this issue.

Technical Specification 3.7.12, Auxiliary Building Filtered Ventilation Exhaust System, requires two ABFVES trains to be operable in Modes 1, 2, 3, and 4. Required Action A.1, specifies that, with one ABFVES train inoperable, it shall be restored to an operable status within seven days. On August 26, 1999, engineering personnel determined that the 1A ABFVES train was inoperable between June 16, 1999, and August 5, 1999, due to ABFVES damper positioning problems. This period of inoperability exceeded that allowed by TS and Unit 1 had not been placed in Mode 3 (Hot Standby) in six hours and in Mode 5 (Cold Shutdown) in the following 36 hours.

After reviewing Unit 1 TSAIL entries from June 16, 1999, through August 5, 1999, the inspectors discovered that, during the period of time that train 1A was inoperable, there were four occasions in which the 1B ABFVES train was also inoperable. Having both trains of ABFVES inoperable placed Unit 1 in a TS 3.0.3 condition. Had TS 3.0.3 been invoked, the resulting plant conditions for two of the four occasions in which the 1B train was inoperable for greater than the specified TS 3.0.3 time limit (i.e., approximately 14 hours and 8 hours on June 29, 1999, and on July 27, 1999, respectively) would have been the same as if the required actions of TS 3.7.12 were performed as required. However, having both Unit 1 ABFVES trains inoperable was not initially developed in the LER. The inspectors discussed this omission with the licensee who indicated that the status of the 1B train was not investigated because, for this event, the Unit 2 system was available to filter any radionuclides that potentially leaked from the affected Unit 1 ECCS pump rooms into the general auxiliary building area. The inspectors confirmed that actual ECCS pump and component leakage was less than that assumed in the licensee's safety analysis. Based on that information, and the fact that the ABFVES was not called upon to perform any post-accident functions during the time period in question, the inspectors concluded that this event had no actual safety consequences. Revision 1 to the LER was issued on February 21, 2000, to document the inspectors' findings related to the 1B train, and to better describe the minimal safety consequences of this event, as well as corrective actions that were planned.

The LER also contained results of a query that was performed to determine whether or not the failure to perform adequate retests of equipment following maintenance activities was a recurring issue. The licensee identified three previous occurrences that involved inadequate retesting. Because of the apparent recurring nature of this issue, the licensee has included this trend in its corrective action under PIP C-99-03127. In addition, due to the number of auxiliary building ventilation issues in recent years, an organization in engineering was developed in Fall 1999, as part of a management focus initiative, dedicated to improving the licensee's overall performance in the area of maintaining and operating safety-related ventilation systems.

In summary, after maintenance was performed on inlet vortex damper 1ABFD-13 on June 16, 1999, the 1A train of the ABFVES was made inoperable and remained inoperable until August 5, 1999, without the appropriate plant actions performed as required by TS 3.7.12. Additionally, while the 1A train of the ABFVES was inoperable from June 16, 1999, to August 5, 1999, the 1B train of the ABFVES was also inoperable for approximately 14 hours on June 29, 1999, and for approximately 8 hours on July 27, 1999, without the appropriate plant actions performed as required by TS 3.0.3. This Severity Level IV violation of TS 3.7.12 and 3.0.3 is being treated as a NCV, consistent with Section VII.B.1 of the NRC Enforcement Policy. It is identified as 50-413/00-02-03: Non-Compliance with TS 3.7.12 and 3.0.3 Due To Inoperable ABFVES Trains. This LER and its supplement are closed.

III. Engineering

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) URI 50-413/99-03-03: Review of Missing Ice Basket Coupling Screws in the Unit 1 Ice Condenser

This URI was identified to review the licensee's final past operability determination concerning Unit 1 ice condenser baskets where the number of missing coupling ring screws exceeded the specified acceptance criteria as identified in Westinghouse Nuclear Safety Advisory Letter 98-012. The NRC staff reviewed the licensee's operability determinations for conditions identified in the subject URI and independently evaluated the conclusions and assumptions identified in PIP 1-C99-1734. This PIP detailed the locations and number of ice condenser basket coupling screws that were identified during the End-of-Cycle 11 refueling outage (April 21, 1999, - May 23,1999) as missing or never installed, which exceeded the acceptance criteria for missing coupling screws based on the vendor information. Based on its reviews, the NRC staff concluded that the identified missing screw locations did not adversely impact the past operability of the Catawba Unit 1 ice condenser. In addition, the staff reviewed the licensee's methodology for sampling the ice condenser for extent of condition reviews and concluded that the licensee's sample size was adequate to assume current operability. However, based on the identified material condition problem, the inspector concluded that a violation of 10 CFR Part 50, Appendix B occurred. Specifically, Criterion V requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, and drawings, and shall be accomplished in accordance with the established instructions, procedures, and drawings. Contrary to this, the ice condenser coupling screws on station drawing CNM 1201.17-0030, Sheet 13 were not installed for ice condenser baskets identified in PIP 1-C99-1734. This Severity Level IV Violation is being treated as a NCV consistent with Section VII.B.1 of the NRC Enforcement Policy. It is identified as NCV 50-413/00-02-04: Ice Condenser Coupling Screws Not Installed in Accordance with Station Drawings. This violation is in the licensee's correction action program as PIP 1-C99-1734. This URI is closed.

IV. Plant Support

F8 Miscellaneous Fire Protection Issues (92904)

F8.1 (Closed) URI 50-413,414/99-02-05: Missed Hourly Fire Watch Patrols

The inspectors conducted an in-office review of documentation necessary to address the concerns associated with URI 50-413,414/99-02-05, Missed Hourly Fire Watch Patrols. As described In NRC Inspection Report 50-413,414/99-02, dated May 5, 1999, the licensee determined that required hourly fire watch duties had not been performed in some areas of the plant. The fire watch requirements were established due to inoperable fire barrier penetrations, which had been identified previously. The licensee concluded that fire watch activities had not been performed after a review of security records for doors for the affected areas. As a result of their investigation, the licensee identified a two and one-half week period, between March 9 - 28, 1999, in which some of the areas could not have been accessed by individuals who signed the logs indicating that the fire watch duties had been completed. In all, approximately 45 instances occurred where doors monitored by the security system were not accessed by the hourly fire watches as required. The licensee took immediate corrective actions after concluding, for three of four individuals involved, that there was deliberate intent to falsify records indicating completion of fire watch tours.

During subsequent NRC inspections and investigations surrounding this issue, the NRC confirmed that the four individuals who signed the logs could not have performed the fire watch duties as indicated for certain areas. A review of information regarding this issue and plant access records to areas monitored by the security computer-aided door system led the inspectors to conclude that three of the individuals willfully signed the fire watch log for tours that had not been completed. The inspectors concluded that the one remaining individual likely missed the areas in question as a result of human error.

The inspectors found that the licensee had established adequate training for those individuals responsible for performing the fire watch duties. Records indicated that the four individuals attended the fire watch training. The inspectors' determination that adequate training had been provided was based on a description of the training in the information reviewed and was supported by security records indicating that in the vast majority of cases, the fire watches were performed properly.

The Catawba Facility Operating License, Item 2.C.(8) states, in part, that Duke Energy Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report, as amended, for the facility and as approved in the SER (Safety Evaluation Report) through Supplement 5. Technical Specification 5.4.1 requires that written procedures shall be established, implemented, and maintained covering the activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, which included administrative procedures for the plant fire protection program (Item 1.l).

The UFSAR, Section 16.0, Selected Licensee Commitments (SLC), and the licensee's directive NSD 316, Revision 1, Fire Protection Impairment and Surveillance, delineated fire watch requirements. NSD 316 applied to all fire protection features within the owner controlled areas as designated by the UFSAR/SLC. Section 16.9-5, Fire Barrier Penetrations, of the UFSAR stated in part:

> All fire barriers (walls, floor/ceiling, cable tray enclosures and other fire barriers) and all sealing devices in the fire barrier penetrations (fire doors, fire dampers, cable, pipe and ventilation duct penetration seals) separating safety from non-safety related areas, redundant analyzed Post Fire Safe Shutdown Equipment, or the Control Complex from the remainder of the plant shall be operable at all times.
>
> With one or more of the above required fire barriers penetrations and/or sealing devises inoperable, within one hour either establish a continuous fire watch on at least one side of the affected penetration, or verify the OPERABILITY of the fire detectors on at least one side of the inoperable penetration and establish an hourly fire watch patrol.

In accordance with Nuclear System Directive 316, the licensee initiated Impairment and Compensatory Measure (ICM) Form 99-60 in response to degraded fire barriers.

The licensee's failure to perform some of the required hourly fire watch activities between March 9 - 28, 1999, was a violation of the above requirements. Although the violation was determined in some cases to be willful, the inspectors determined that because of the non-supervisory status of the individuals involved, and the fact that the licensee took immediate corrective actions, this Severity Level IV violation is being treated as an NCV, consistent with Section VII.B.1 of the NRC Enforcement Policy. The violation is in the licensee's corrective action program as PIP O-C99-1118. It is identified as NCV 50-413,414/00-05: Failure to Properly Perform Hourly Fire Watches.

In addition, the inspectors also determined that there were other tools/aids and guidance established by station administrative procedures, which, if implemented properly, could have provided additional assurance that fire watch duties would have been performed properly. Nuclear System Directive (NSD) 316.6, General Practices, stated that when practical the ICM Form should be posted in the affected area at a visible location as close to the impaired feature as possible. As an alternative, the ICM form can be retained and carried by the person performing the fire watch. NSD 316.5.5, Personnel and Supervision Responsible for Performing a Fire Watch, states, in part, that "If an impairment requires a fire watch surveillance that covers more than one room, area, or unit, then separate ICM Forms (Appendix A) are required for each room, area, or unit. Getting (sic) supervisor approval prior to performing surveillance, if deviations are needed."

The inspectors determined that only one ICM form was initiated for the performance of the fire watch tours in question. The single ICM form was located on the fourth floor of the service building in the single point of contact maintenance office areas. When interviewed by the inspectors, contract supervisory personnel responsible for implementing the program were unfamiliar with the ICM form requirements contained in the NSDs. No immediate supervisory approval had been granted to deviate from the posting requirements. A licensee management representative indicated during this inspection period that wallet-sized cards listing the affected areas were provided to fire watch personnel. However, certain of the individuals performing fire watch duties indicated they were unfamiliar with these cards. The inspectors concluded that the licensee's failure to comply with their administrative requirements for implementing the fire watch program may have defeated a barrier to human error.

The licensee's failure to properly implement NSD 316 by either posting the ICM forms at each of the various locations or having the personnel performing the fire watch duties hand carry the forms during their rounds, was also a violation of the licensee's procedures and NRC requirements. However, the inspectors determined that the failure to post or carry the forms as required, or request supervisory approval to deviate from the NSD requirement, constituted a violation of minor significance and is not subject to formal enforcement action. This item is closed.

V. Management Meetings

X1 Exit Meeting Summary

The inspector presented the inspection results to members of licensee management at the conclusion of the inspection on April 5, 2000. The licensee acknowledged the findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Beadle, Emergency Preparedness Manager
R. Beagles, Safety Review Group Manager
M. Boyle, Radiation Protection Manager
G. Gilbert, Regulatory Compliance Manager
R. Glover, Operations Superintendent
P. Grobusky, Human Resources Manager
P. Herran, Engineering Manager
R. Jones, Station Manager

R. Parker, Maintenance Superintendent
G. Peterson, Catawba Site Vice-President
F. Smith, Chemistry Manager
D. Sweigart, Safety Assurance Manager

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 73753:	Inservice Inspection
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support
IP 93702:	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED AND CLOSED

Opened

50-413/00-02-01	NCV	Failure to Comply with TS 3.8.4 and 3.8.1 with the 1A EDG Inoperable (Section O8.1)
50-413,414/00-02-02	NCV	Inadequate Operating Procedure Used to Maintain RN System Configuration Control When Removing One Pump from Service (Section M8.2)
50-413/00-02-03	NCV	Non-Compliance with TS 3.7.12 and 3.0.3 Due To Inoperable ABFVES Trains (Section M8.3)
50-413/00-02-04	NCV	Ice Condenser Coupling Screws Not Installed in Accordance with Station Drawings (Section E8.1)
50-413,414/00-02-05	NCV	Failure to Properly Perform Hourly Fire Watches (Section F8.1)

Closed

50-413/99-017-00	LER	Operation Prohibited by TS 3.8.4 and 3.8.1 concerning DC Power Supply to Diesel Generator 1A due to an Inadequate Procedure (Section O8.1)
50-413/00-001-00	LER	Reactor Trip Caused by a Pin to Pin Short Circuit within an Electrical Connector on the Turbine Electrical Solenoid Valve (Section O8.2)
50-414/99-006-01	LER	Reactor Trip Caused by an Electrical Ground in an Electrical Connector on the Turbine Electrical Trip Solenoid Valve (Section O8.3)

50-413/99-016-00	LER	Operation Prohibited by Technical Specification 3.8.1 and 3.7.8 Due to Inoperable Diesel Generator 1B for Greater than 72 Hours (Section M8.1)
99-2-003	NOED	Catawba Unit 1 Inoperable 1B Emergency Diesel Generator and Nuclear Service Water System (Section M8.1)
50-413/99-07-02	URI	1B EDG Inoperability Due to Successive Test Failures following Maintenance - NOED 99-2-003 (Section M8.2)
50-413/99-015-(00,01)	LER	Inoperability of Auxiliary Building Ventilation System in Excess of Technical Specification Limits Due to Improperly Positioned Vortex Damper (Section M8.3)
50-413/99-03-03	URI	Review of Missing Ice Basket Coupling Screws in the Unit 1 Ice Condenser (Section E8.1)
50-413,414/99-02-05	URI	Missed Hourly Fire Watches (Section F8.1)

LIST OF ACRONYMS USED

ABFVES - Auxiliary Building Filtered Ventilation Exhaust System
ASME - American Society of Mechanical Engineers
B&PV - Boiler and Pressure Vessel
CA - Auxiliary Feedwater
CFR - Code of Federal Regulations
DC - Direct Current
EA - Enforcement Action
ECCS - Emergency Core Cooling System
EDG - Emergency Diesel Generator
EGA - Electronic Governor Assembly
ETS - Emergency Trip System
ICM - Impairment/Compensatory Measures Form
IFI - Inspector Followup Item
ISI - Inservice Inspection
LCO - Limiting Conditions for Operation
LER - Licensee Event Report
NCV - Non-Cited Violation
NOED - Notice of Enforcement Discretion
NRC - Nuclear Regulatory Commission
NS - Containment Spray
NSD - Nuclear System Directive
PIP - Problem Investigation Process
PM - Preventative Maintenance
PORC - Plant Operations Review Committee
RN - Nuclear Service Water
SER - Safety Evaluation Report
SLC - Selected Licensee Commitments
SR - Surveillance Requirement
SWM - Shift Work Manager
TS - Technical Specification

TSAIL	-	Technical Specification Action Item Log
UFSAR	-	Updated Final Safety Analysis Report
URI	-	Unresolved Item
VDC	-	Volts Direct Current

SYNOPSIS

The U.S. Nuclear Regulatory Commission, Region II, Office of Investigations, initiated this investigation on April 15, 1999, to determine if three firewatch personnel who were formerly employed as contractors at the Duke Energy Corporation Catawba Nuclear Station, had failed to perform firewatches and falsified firewatch records as though they had completed their watches.

The evidence developed during this investigation substantiated that the three contract personnel deliberately and intentionally failed to perform the firewatches as required by procedures, and that they deliberately and intentionally falsified firewatch records by indicating that the firewatches had been performed.

APPROVED FOR RELEASE ON APRIL 17, 2000