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

Report Nos.: 50-413/95-24 and 50-414/95-24

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

Docket Nos.: 50-413 and 50-414 License Nos.: NPF-35 and NPF-52

Facility Name: Catawba Nuclear Station Units 1 and 2

Inspection Conducted: November 19, 1995 - December 30, 1995

Inspectors: R E Carroll for telcon 1/19/96
R. J. Freudenberger, Senior Resident Inspector Date Signed

P. A. Balmain, Resident Inspector
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Approved by: R. V. Crlenjak 1/19/96
R. V. Crlenjak, Chief Date Signed
Projects Branch 1
Division of Reactor Projects

SUMMARY

Scope: Inspections were conducted by resident and regional inspectors in the areas of plant operations, maintenance, engineering, and plant support. As part of this effort, backshift inspections were conducted.

Results: Plant Operations Significant management involvement in zero power physics testing was noted. Specifically, the Plant Operation Review Committee (PORC) approved the shift briefing packages for zero power physics testing and management personnel were present in the control room throughout testing. The licensee's actions to ensure rigid control of reactivity and correction of problems encountered in the previous Unit 1 zero power physics testing were effectively implemented (paragraph 2.2). The PORC review of the repair plan for valve 2NI-169 was considered an effective forum for discussing isolation options and potential safety issues associated with the valve repair (paragraph 2.3).

Enclosure

Maintenance — Resolution of and/or compensatory actions for several degraded components were appropriately dispositioned by the licensee (paragraphs 3.1, 3.2, 3.3, 3.4, and 3.5).

Engineering — The modification of the station's carbon dioxide fire suppression systems to provide a pneumatic predischarge alarm was an overall appropriate resolution and proper safety enhancement to a previous failure of an electric predischarge alarm. The licensee's safety contingencies taken during post modification testing were good. However, the actual operation and response of the suppression system was not understood during post modification test development and resulted in the unexpected discharge of several carbon dioxide storage cylinders into an auxiliary feedwater pump area during testing of the modification (paragraph 4.1).

Plant Support — An emergency organization drill and critique process was effective in identifying areas for improvement (paragraph 5.1).

REPORT DETAILS

Acronyms used in this report are listed in the last paragraph.

1.0 PERSONS CONTACTED

Licensee Employees

B. Addis, Training Manager
S. Coy, Radiation Protection Manager
* J. Forbes, Engineering Manager
W. Funderburk, Work Control Superintendent
T. Harrall, IAE Superintendent
* D. Kimball, Safety Review Group Manager
W. McCollum, Catawba Site Vice-President
W. Miller, Operations Superintendent
* K. Nicholson, Compliance Specialist
* M. Patrick, Safety Assurance Manager
* G. Peterson, Station Manager
R. Propst, Chemistry Manager
D. Rogers, Mechanical Superintendent
* Z. Taylor, Regulatory Compliance Manager
* D. Tower, Regulatory Compliance Engineer

* Attended exit interview.

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

2.0 PLANT OPERATIONS (NRC Inspection Procedures 40500, 71707 and 92901)

Throughout the inspection period, control room observations and facility tours were conducted to observe operations activities in progress. During these inspections, discussions were held with operators, supervisors, and plant management. Some operations activity observations were conducted during backshifts. Licensee meetings were attended by the inspector to observe planning and management activities. The inspections evaluated whether the facility was being operated safely and in conformance with license and regulatory requirements. In addition, the inspections assessed the effectiveness of licensee controls and self-assessment programs in achieving continued safe operation of the facility.

2.1 PLANT STATUS

Unit 1 Summary

Unit 1 began the report period at 100% power and remained at or near full power until December 15 when power was decreased to 88% for a main turbine control valve performance test. At that time, the 1C2 heater drain tank pump was removed from service so that a leak in the pump discharge vent line could be repaired. The unit returned to 100% power later on December 15, although at a lower gross output since the heater drain tank pump remained out of service. The pipe repair was completed on December 16 and power was decreased to 97% to support returning the heater drain tank pump to service. The unit was subsequently returned to 100 percent power. On December 17, the 1C2 heater drain pump tripped on overcurrent; however, reactor power level was not affected. The unit remained at or near 100 percent power until December 20, when power was decreased to 97% so that the 1C2 heater drain tank pump could be returned to service following repairs to a short in the motor leads. Reactor power was returned to 100% on December 21 and remained at or near 100% power for the remainder of the inspection period.

Unit 2 Summary

Unit 2 began the period in Mode 4 at 0% power, preparing for entry into Mode 3. The reactor was taken to Mode 5 on November 20 for replacement of valve 2NI-169 (2D Cold Leg Injection Check Valve), which had failed reactor coolant system pressure boundary valve testing. Following replacement of all five reactor vessel head conoseals, which had developed leaks as a result of the mode change for valve 2NI-169 replacement, the unit returned to Mode 4 on November 25 and was synchronized to the grid on November 30. The unit reached 100% power on December 7 and remained at or near full power until December 13, when power was reduced to 94% so that low-pressure feedwater heater 2G3 could be isolated for repair of a vent line leak. Attempts to isolate the heater were not successful, so the unit was returned to 100% power on December 14. Power was decreased to 87% early on December 15 for a main turbine control valve movement performance test. A second attempt to isolate the feedwater heater was successful at that time. The feedwater repair was made, and the unit returned to 100% power later that day. The unit remained at or near 100% power for the remainder of the inspection report period.

2.2 Control Room Operator Performance During Zero Power Physics Testing

During the last Unit 1 startup from refueling on March 23, 1995, a reactivity excursion occurred during zero power physics testing as documented in NRC Inspection Report 50-413,414/95-10, Section 3.a. An excessive positive reactivity addition resulted when a shutdown bank was withdrawn 14 steps in the central area of the core (area of high differential worth). This positive reactivity addition caused reactivity indication to over-range and wrap around to the negative range. The test coordinator did not realize that the reactivity indication was over-ranged and, assuming that reactivity was much lower than actual, directed the reactor operator to continue to withdraw control rods, resulting in a power excursion to roughly 3.5 percent power range.

A Significant Event Investigation Team organized to assess the event identified two root causes: (1) inadequate planning, preparation and job briefing for the testing and (2) inadequate work practices by the control room operator. Corrective actions to prevent recurrence were implemented during the Unit 2 zero power physics testing observed during this inspection period.

The inspector attended a PORC meeting which reviewed and approved the zero power physics testing briefing packages to assess the effectiveness of the committee's review. The problems from the previous restart were discussed and changes in how the evolution would be conducted for the current restart were thoroughly reviewed.

The inspector also was present in the control room for portions of initial approach to criticality and zero power physics testing. A reactivity computer was installed in the back of the control room, which included a strip chart recorder for trending reactivity, flux, temperature and pressurizer level data. A strip chart recorder was also installed in the control room horseshoe area where the designated reactor engineer could monitor core changes resulting from rod manipulations during testing. Alarms were provided for reactivity deviations greater than ±40 pcm. Additionally, a Senior Reactor Operator was involved to provide oversight of the execution of the procedures. During the test evolutions, a management representative (station manager, engineering manager, or operations superintendent, etc.) was present in the horse shoe area to provide additional oversight and assessment. Inspector observations of various shifts implementing zero power physics testing revealed that communication between the Test Engineer, the Reactor Operator, and the Senior Reactor Operator were not consistently implemented. In some cases, the Senior Reactor Operator was directly involved in communications between the Test Engineer and the Reactor Operator, while in other cases, the Senior Reactor Operator provided an oversight function. The inspector concluded that consistent implementation of communication protocol could reduce the potential of errors in these communications. Licensee management also recognized the inconsistency in the implementation of communication protocol between the shifts and planned to determine if actions were appropriate.

As indicated above, significant management involvement in zero power physics testing was noted by the inspector. Specifically, the PORC approved the shift briefing packages for zero power physics testing and management personnel were present in the control room throughout testing. The inspector concluded that the licensee's actions to ensure rigid control of reactivity and correction of problems encountered in the previous Unit 1 zero power physics testing were effectively implemented.

2.3 Pressure Boundary Check Valve Leakage

On November 19, with Unit 2 in Mode 4, valve 2NI-169 (2D Cold Leg Injection Check Valve) failed a reactor coolant system pressure boundary valve test. Leakage past the valve seat was 2.8 gpm, thereby placing the unit in TS 3.4.6.2.f. The licensee unsuccessfully attempted to seat the valve via mechanical agitation. Consequently, Unit 2 was cooled down and depressurized to Mode 5 on November 20 so that valve repairs could be made. A PORC meeting was held to evaluate the maintenance repair plan, which involved establishing a freeze seal to isolate the valve so that it could be inspected and repaired. Because safety injection system piping was too hot for a freeze seal to be established, the valve could not be isolated for repair until November 21 when temperatures had sufficiently dissipated. The valve and its associated piping were cut out, and a replacement valve, prefabricated with associated piping, was welded into place on November 22.

The depressurization to mode 5 early on November 20 was followed by the identification of leakage on four of five of the reactor vessel head conoseals. The reactor coolant system was drained to 38% wide range level (below the top of the head, above the vessel flange) on November 23 so that conoseal work could be performed. The licensee disassembled and inspected all five of the conoseals and found no problems associated with workmanship or foreign material and did not identify any nicks or scratches in the seating surfaces. Nonetheless, all five conoseals were replaced and the unit was returned to Mode 4 on November 25. The licensee suspects that the conoseal leaks were induced by the depressurization associated with the mode change.

The inspector considered the PORC review of the repair plan an effective forum for discussing isolation options and potential safety issues associated with the valve repair. Post-maintenance pressurized air decay testing (using an inflatable bladder to segment the valve test area) is conducted on system pressure boundary check valves at the McGuire plant, whereby leaks are discovered and corrected before the reactor coolant system is pressurized. This testing improves the reliability of the valves' performance during pressure boundary testing, thereby reducing the risk of having to cooldown and depressurize to repair a leaking check valve. The practice at McGuire was discussed with Catawba plant management. No safety concerns associated with valve repair, restoration, and testing practices at Catawba were identified.

2.4 (Closed) Violation 50-413,414/94-27-01: Inaccurate Tagout Results in Both Control Room Ventilation Trains Being Inoperable

On October 18, 1995, TS 3.0.3 was unknowingly entered when the access panel on the Train B Control Room Area Ventilation air handling unit was opened in support of routine maintenance. Outward air flow through the open access panel (as a result of unsecured backdraft damper 2CR-D-10) created a bypass leakage flowpath. This caused the running Control Room

Area Ventilation Train A to be inoperable in that it could not maintain the control room pressurized to the TS required ⅛ inch water column. Damper 2CR-D-10, like other motor operated Control Room Area Ventilation dampers, had been modified the year before per NSM CN-50433 to be a manual/backdraft damper with a securing device. The pre-planned tagout utilized to generate the Train B Control Room Area Ventilation maintenance tagout had not been updated to reflect NSM CN-50433. Consequently, despite instructions on the air handling unit breaker cabinet, backdraft damper 2CR-D-10 was verified "closed" instead of "secured closed."

As indicated in the licensee's response, key aspects of this event (with particular emphasis on operator responsibilities) were addressed during subsequent licensed operator requalification training. The inspector confirmed that this occurred, as well as verified that the Control Room Area Ventilation pre-planned tagouts have since been revised to reflect NSM CN-50433 and that "secured closed" is now provided as a selection on the tagging program pick list. As TS 3.0.3 was not violated (i.e., Control Room Area Ventilation Train A was restored well within the "Shut Down" Action Statement Limits) and interviews with the Operations staff indicated that all pre-planned tagouts were now being verified correct before each use, the inspector considered this item closed.

2.5 (Closed) Violation 50-414/94-27-02: Inadequate Procedures for Realignment of Auxiliary Feedwater Following an Automatic Start

Two days after Unit 2 restarted from a reactor trip on October 18, 1994, its motor driven auxiliary feedwater pumps' flow control valves (2CA-40,44,56 and 60) were found to be closed instead of open as required by TS 4.7.1.2.1.a.4. This failure to return the Auxiliary Feedwater system to its standby alignment was set up by conditional procedure inadequacies in EP/2/A/5000/ES-0.1, Reactor Trip Response, and OP/2/A/6100/05, Unit Fast Recovery. Accordingly, the inspector verified that both of these procedures were revised to assure that the Auxiliary Feedwater system will be returned to its standby alignment after being secured. Other corrective actions included stressing operator responsibilities (particularly control board monitoring) during subsequent licensed operator requalification training and revising the Control Room Indication Checklist in OMP 2-22, Shift Turnover, to recognize that the controllers for the Auxiliary Feedwater flow control valves are required to be set at 100% open when reactor power is $\geq$ 10%. Having confirmed that these other actions were taken and that the subject valves receive a signal to open when the Auxiliary Feedwater system is automatically started, the inspector had no further concerns. This item is closed.

2.6 (Closed) LER 413/94-06, Technical Specification 3.0.3 Entered Due to Inadequate Work Practices

The event and corrective actions associated with this LER are addressed in paragraph 2.4 of this report under the closure of Violation 50-413,414/94-27-01. In view of the human performance initiatives that have been initiated site-wide, in conjunction with the specific corrective actions taken for the event, this LER is closed.

3.0 **MAINTENANCE** (NRC Inspection Procedures 62703, 61726 and 92902)

Throughout the inspection period, maintenance and surveillance testing activities were observed and reviewed. During these inspections, discussions were held with operators, maintenance technicians, supervisors, engineers and plant management. Some maintenance and surveillance observations were conducted during backshifts. The inspections evaluated whether maintenance and surveillance testing activities were conducted in a manner which resulted in reliable, safe operation of the facility and in conformance with license and regulatory requirements.

3.1 Auxiliary Feedwater Discharge Check Valve Backleakage

During this inspection period, the licensee received indications of elevated temperatures for check valve 2CA-53 which is the Unit 2 turbine driven AFW pump discharge check valve to steam generator 2B. The valve functions to prevent backleakage from steam generator 2B into the AFW system when the turbine driven pump is in standby and also functions to allow forward flow into S/G 2B when the pump operates. Elevated temperatures for 2CA-53 indicated that some amount of backleakage from the main feedwater system was occurring.

The licensee periodically operated the turbine driven AFW pump to cool the valve to acceptable temperatures as needed when the temperature exceeded established AFW system operating procedure limits. As the trend of increasing 2CA-53 temperature continued the licensee initiated increased temperature monitoring and engineering evaluation. Review of data generated from the licensee's evaluation confirmed that the backleakage from the valve was minimal and did not compromise the valve's ability to fulfill its safety functions. The inspector reviewed the recent maintenance history for 2CA-53. Maintenance work order documentation revealed that work was performed on this valve during the most recent refueling outage in October 1995. The inspector observed that the work order documentation showed that the edges of the valve disk were rounded to prevent dragging of the disk and no activities were performed that would affect the seating surfaces of the valve.

The licensee plans to perform a turbine driven AFW pump outage in January 1996 to replace the leaking check valve. Based on this review the inspector concluded that operation of 2CA-53 with a small amount of backleakage did not impact operability of the check valve and the licensee took appropriate actions to initiate repair of the valve.

3.2 Heater Drain Pump Motor Lead Failure

On December 17 motor leads for the Unit 1 1C2 heater drain pump motor shorted to ground and caused the motor to trip. The loss of the heater drain pump resulted in a minor secondary plant transient and a loss of some generating efficiency.

The licensee's subsequent investigation determined that the cause of the event was due to rubbing of the leads on the inside of the motor lead terminal box cover. This in turn caused the insulation of the leads to fail and result in the short. The licensee replaced the failed 1C2 Heater Drain pump motor and developed an inspection plan of pump motors which have similar motor lead configurations. With the exception of the Service Water pumps, safety-related pumps were not considered susceptible to this problem due to the use of smaller cable sizes and larger terminal boxes.

The licensee performed inspections of motor leads on several nonsafety-related standby pump motors. Inspections of the condensate pump motors revealed no rubbing or vibration problems. The Unit 2 2B Hotwell pump motor leads were found to be rubbing and had insulation wear. They were subsequently repaired. The Unit 1 Hotwell pump motor had no evidence of damage. The licensee will continue inspections of potentially susceptible nonsafety-related pump motor terminations as pumps are rotated to standby during routine operation or as they become available during power reductions and unit outages. Similarly, the Service Water pumps will be inspected during upcoming system outages. Based on this review the inspector considered the licensee's actions appropriate.

3.3 Steam Generator Feedwater Isolation Valve Actuator Nitrogen Leak

On December 10, an Operator Aid Computer alarm alerted operators to decreasing pressure in the nitrogen accumulator associated with the valve actuator for valve 1CF-51, the Steam Generator 1C Main Feedwater Isolation Valve. The safety function of the valve is to terminate flow in either direction. The nitrogen accumulator provides the passive force to close the valve. Hydraulic pressure is used to maintain the valve open.

Initially, the nitrogen accumulator was recharged as necessary to maintain pressure above the alarm setpoint of 2150 psig. The alarm response indicated that stroke time of the valve in the closed direction was not affected until pressure dropped below 1100 psig. Maintenance technicians identified the leak as originating at both the mount and the locknut on each of the two solenoid valves. The solenoid valves are mounted directly on the actuator. The leakage rate was determined to be sufficiently slow to allow for a planned frequency of verification and repressurization without routinely causing low pressure alarms. Based on the slow leakrate, the licensee planned to repair the leaks at the next shutdown opportunity.

The inspector walked down equipment which was placed in the interior doghouse (Main Steam and Feedwater valve and pipe vault) to support repressurization of the accumulator and reviewed information regarding the status of the valve which had been provided to operators. The inspector concluded that appropriate actions were taken to ensure continued operability of the valve and equipment placed in the area was appropriately controlled.

3.4 Steam Generator PORV Block Valve Packing Leak

During the inspection period, an existing packing leak on the 1A S/G PORV Block Valve (1SV-27A) worsened to the point that two concerns were identified by the licensee, prompting the initiation of PIP 1-C95-2328. The PIP documented evaluation of the impact of the leakage on off-site dose assuming a tube rupture in the 1A steam generator and the effect of increased temperature and humidity on the environmental qualification of the motor operator installed on the valve. The S/G PORV Block Valves are not addressed by Catawba TS; nonetheless, these valves need to be open to support operability of the S/G PORVs themselves. Although TS allow unlimited operation with one S/G PORV inoperable, the licensee chose to pursue isolation of the leak by placing the block valve on its backseat. This option was chosen to maintain operational flexibility in responding to potential plant events.

A function of the S/G PORV Block Valve is to isolate the PORV should it fail open. In order to preserve this function the licensee manually placed the S/G PORV Block Valve on its backseat (which successfully terminated the leak) with limited force applied to prevent valve damage. The motor operator was then engaged by initiating a close stroke of the valve. The close stroke was interrupted after the operator engaged but before the valve came off its backseat. The valve was then stroked electrically to demonstrate it still met its stroke time criteria. This operation was performed several times to demonstrate repeatability of the process.

The inspector concluded that this configuration provided for an optimum resolution of the current S/G PORV Block Valve packing leak. The leak was isolated while the S/G PORV remained operable and the S/G PORV Block Valve remained available to isolate a postulated failed open S/G PORV.

The root cause of the packing leak on valve 1SV-27A was not able to be determined at this time. However, external leaks on the S/G PORV Block Valves have been a recurring problem at Catawba. All four of the Unit 1 S/G PORV Block Valves are planned for replacement in the next refueling outage (1EOC9 - summer 1996).

3.5 Diesel Generator Jacket Cooling Water System Check Valve Leakage

On December 13 the 2B Diesel Generator was declared inoperable because the jacket cooling water keep warm system temperature dropped to 138°F (140°F is the minimum temperature required for operability); as a result, the unit entered a 72-hour action statement to return the inoperable diesel generator to service. An action associated with the inoperable diesel generator is the implementation of surveillance requirement 4.8.1.1.1.a. On December 15, contrary to this surveillance requirement, the control room operators exceeded the 8-hour interval for verifying the availability of off-site power. There have not been any other occurrences of this missed surveillance within the last two years and the licensee was evaluating the root cause and drafting an LER to document the error. Therefore, corrective actions will be reviewed after the LER is submitted.

The licensee determined that the cause of the low temperature was a leaking check valve that allowed diversion of flow that should have traveled from the system heater to the engine. The licensee opened and inspected the check valve (2KD-21) and found no visible signs of damage. The valve was a swing check valve manufactured by Borg Warner, whereas the keep warm system check valves associated with the other three diesel generators were Walworth swing check valves. The licensee speculated that the implementation of a modification during the recent Unit 2 refueling outage may have altered system dynamics slightly and caused the check valve to leak. The modification involved a change in location of the temperature sensor for controlling the standpipe heater. The sensor had been located at the standpipe itself; it was relocated to the diesel generator engine to ensure that temperature was controlled at the critical system component. The licensee suspects that the check valve always leaked, but that leakage was so marginal that sufficient flow was returning to the diesel generator engine. Because the temperature sensor was relocated to the diesel generator engine, temperature of the keep warm system increased slightly and caused a thermostatic valve to open and divert an increased amount of water from the engine. The increased leakage was sufficient to cause temperature to drop to the alarm setpoint and reveal the leakage by 2KD-21.

The licensee decided to replace the valve but did not have sufficient time remaining in the action statement to perform the maintenance and restore it to service. As a result, valve replacement was scheduled for December 20. A performance test was conducted December 14 in an attempt to restore operability of the diesel generator. During this test the diesel generator was required to be shutdown due to the failure of a fuel injector and a fuel supply line fitting. The licensee appropriately characterized this test as a valid failure and plans to submit a special report to the NRC regarding this event.

The inspector reviewed the results of the licensee's investigation of this failure with engineering personnel and observed the failed injector after it had been removed from the engine. The nozzle end of the failed injector was blackened with carbon soot. Normally this area is clean and free of soot. Based on the condition of the injector, the licensee determined that debris in the injector seating area prevented the injector from seating adequately in the engine cylinder, which resulted in exhaust gases from engine operation to leak by the outside diameter of the injector. The exhaust leak-by caused excessive heat to build up that led to abnormal expansion of internal and external fuel injector parts which caused the injector to seize. Seizure of the injector allowed a rapid pressure increase in the fuel line between the fuel pump and fuel injector that blew off the fuel line fitting and created and substantial fuel leak. The inspector reviewed corrective action documentation (PI 2-C95-2394) and concluded the licensee's actions in response to this failure were appropriate. The performance test was successfully rep[illegible] on December 16, and the diesel generator was declared operable.

During the week of December 18, valve 2KD-21 was replaced with an Anderson Greenwood wafer check valve, and the diesel generator was returned to service on December 21. The Anderson Greenwood valve is of a different design and features a spring loaded device to facilitate valve seating. The inspector observed portions of the maintenance and concluded that the valve replacement was an appropriate corrective action. No concerns with the maintenance were identified.

3.6 (Closed) Violation 50-414/94-27-03, Failure to Follow Procedures Results in Reactor Trip

On October 18, 1994, Unit 2 tripped during the simultaneous performance of the Train B Solid State Protection System/Reactor Trip Breaker bi-monthly testing (which caused an expected General Warning Alarm Condition for Train B) and the quarterly analog channel operational test (ACOT) of 7300 process instrumentation associated with channel #4 to the Solid State Protection System (SSPS). Confronted with an apparent conflict, the IAE technicians performing the tests decided to return the SSPS Train A multiplexer switch to "NORMAL." However, while repositioning this multiplexer switch from "A+B" back to "NORMAL," the switch had to pass through the "INHIBIT" position. This caused a momentary General Warning Alarm Condition for Train A, which when combined with the existing Train B alarm condition made up the necessary logic to cause the reactor trip.

The ACOT associated with SSPS Channel #4 had begun as scheduled on October 17, 1994, but completion was delayed in order to effect repairs on the pressurizer pressure channel #4 trip bistable. When the ACOT was resumed on October 18, 1994, Operations questioned its compatibility with scheduled Train B SSPS/reactor trip breaker testing. As the conflict described above was not readily apparent, testing personnel assured Operations that the tests could be done simultaneously. To preclude similar occurrences in the future, the licensee implemented a formal policy to evaluate the impact of carryover work, as well as requiring prior approval by the Station Manager for performing any cross train work. The inspector verified that this policy was appropriately addressed in Site Directive 3.0.8, Scheduling Philosophy For Priority Work, and Work Process Manuals WPM 600, Scheduling Work On The Master Schedule, and WPM 700, Work Execution. Confirming that this policy is being controlled by the work scheduling process and that station personnel have been made aware of the policy, as well as the expectation to stop work and seek supervisory help when an unexpected work situation exists, the inspector considered this item to be closed.

3.7 (Closed) Violation 413/94-27-04: Failure to Follow Procedure Results in Reinstatement of Reactor Trip Bistables

On October 18, 1994, Unit 1 power range nuclear instrument channel N-44 was declared inoperable and the affected reactor trip channel bistables were placed in the tripped condition per AP/1/A/5500/16, Malfunction of Nuclear Instrument Systems. During the following two days, N-44 troubleshooting activities resulted in the installation of associated control power fuses on three separate occasions. This unknowingly reinstated the associated high neutron flux and positive rate trip bistables, which is not in compliance with TS 3.3.1 for a power range channel considered to be inoperable. It was determined that the conflict with replacing control power fuses could have been identified had the review and discussion between IAE technicians and operations personnel (required by IP/0/A/3890/01, Controlling Procedure for Troubleshooting and Corrective Maintenance) been at the expected level of detail.

As indicated in the licensee's response, corrective actions included: (1) re-emphasizing the expectations for IAE technicians and operations personnel supporting troubleshooting activities; (2) revising AP/1(2)/A/5500/16 to assure operations maintains positive control of the control power fuses; and (3) developing a method to maintain high neutron flux and positive rate bistables in a tripped condition during troubleshooting, maintenance, and testing activities. Aside from verifying that these actions were appropriately completed, the inspector reviewed subsequent changes made to IP/0/A/38390/01 that require the preparation of, and adherence to, a detailed troubleshooting/corrective action plan (including the performance of a consequence assessment that is approved by the Operational Control Group). The inspector determined that these enhancements to IP/0/A/38390/01, in conjunction with the revision to AP/1(2)/A/5500/16 requiring the operations shift manager to maintain control over the control power fuses until authorized for reinstallation by the operations superintendent (or designee), should preclude similar occurrences. Having confirmed that other Nuclear Instrumentation System related procedures have been identified for review to ensure the desired communications and controls are reflected, the inspector had no further questions. This item is closed.

3.8 (Closed) LER 414/94-07, Reactor Trip and Technical Specification Violation

The events and corrective actions associated with this LER are addressed in paragraphs 2.5 and 3.6 of this report under the closures of Violations 50-414/94-27-02 and 03, respectively. In view of the human performance initiatives that have been initiated site-wide, in conjunction with the specific corrective actions taken for each of the events, this LER is closed.

4.0 **ENGINEERING** (NRC Inspection Procedure 37551)

Throughout the inspection period, the inspectors reviewed engineering evaluations, root cause determinations, and modifications. During these inspections, discussions were held with operators, engineers, and plant management. The inspection evaluated the effectiveness of licensee controls in identifying and appropriately documenting problems, as well as implementing corrective actions.

4.1 Carbon Dioxide Fire Suppression System Modifications

During this inspection period the inspector reviewed the licensee's installation and testing of the AFW Cardox system modification. Cardox systems are carbon dioxide fire suppression systems and are installed in the AFW and DG areas in both units. The modifications consisted of the addition of a pneumatic predischarge siren and timer to ensure that an audible alarm is sounded prior to discharge of carbon dioxide if these systems are actuated. These pneumatic features supplement existing electrical alarms and were added to ensure timely evacuation from areas which could become a severe personnel safety hazard due to the displacement of oxygen in the area. The existing electrical alarm system malfunctioned previously in the Unit 1A DG room such that carbon dioxide was discharged into the DG room without any audible predischarge warning (see NRC Inspection Report 50-413,414/95-22).

The inspector reviewed the licensee's safety evaluation screening of the modification and found it to be appropriate. The inspector also witnessed post modification testing of the Unit 1 AFW Cardox system. The inspector observed that oxygen monitor instrumentation was being used in the area to verify a breathable atmosphere and rescue personnel with a SCBA were stationed in the room during testing. The licensee took these precautions because the pneumatic siren operates using carbon dioxide gas from the cardox system and a small amount of carbon dioxide was expected to discharge from the sirens. When the testing was initiated the sirens performed properly but operated for longer than expected. When the AFW cardox systems were being restored to operable, following the modifications, the licensee recognized that several carbon dioxide cylinders were empty. A subsequent investigation revealed the contents of several of the cylinders unexpectedly discharged into the AFW areas instead of the small amount expected.

The inspector discussed the development of the post modification testing and potential causes for the volume of carbon dioxide discharged during the AFW cardox system test with engineering personnel involved in the test development. From these discussions, the inspector observed that the post modification test for the AFW cardox systems was based on the understanding of the DG cardox system operation which is similar, but not identical to the AFW cardox systems.

Based on this review, the inspector concluded that the modification of the station's carbon dioxide fire suppression systems to provide a pneumatic predischarge alarm was an overall good resolution and safety enhancement in response to a previous failure of an electric predischarge alarm. The licensee's safety contingencies taken during post modification testing were good; however, the actual operation and response of suppression system was not understood during post modification test development and resulted in the unexpected discharge of several carbon dioxide storage cylinders into an auxiliary feedwater pump area during testing of the modification.

5.0 **PLANT SUPPORT** (NRC Inspection Procedure 71750)

Throughout the inspection period, facility tours were conducted to observe activities in progress. Some tours were conducted during backshifts. The tours included entries into the protected areas and the radiologically controlled areas of the plant, including emergency response facilities. Observations included assessments of radiological postings and work practices. During these inspections, discussions were held with radiation protection and security personnel. The inspections evaluated the effectiveness of the programs to assess whether activities were performed safely and in conformance with license and regulatory requirements.

5.1 Emergency Organization Drill

On December 6, the licensee conducted an emergency preparedness organization drill. The inspector participated in the drill, attended the post drill critique, and reviewed PIP C96-0125 which documented observations from the critique for action item tracking and trending. The inspector observed that the drill and critique process were effective in identifying areas for improvement.

6.0 **NRC PERSONNEL ON SITE**

On December 4, members of NRC management met with members of Duke Power Company management at the Catawba Nuclear station to present the findings of a recent Systematic Assessment of Licensee Performance (SALP) performed by the NRC. The results of the assessment are documented in Inspection Report 50-413,414/95-99.

7.0 EXIT

The inspection scope and findings were summarized on January 3, 1996, with those persons indicated by an asterisk in paragraph 1. An interim exit was conducted on November 30, 1995. The inspector described the areas inspected and discussed in detail the inspection results. A listing of inspection findings is provided. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

Type	Item Number	Status	Description and Reference
VIO	50-413,414/ 94-27-01	Closed	Inaccurate Tagout Results in Both Control Room Ventilation Trains Being Inoperable (paragraph 2.4).
VIO	50-414/94-27-02	Closed	Inadequate Procedures for Realignment of Auxiliary Feedwater Following an Automatic Start (paragraph 2.5).
VIO	50-414/94-27-03	Closed	Failure to Follow Procedures Results in Reactor Trip (paragraph 3.6).
VIO	50-413/94-27-04	Closed	Failure to Follow Procedure Results in Reinstatement of reactor Trip Bistables (paragraph 3.7).
LER	413/94-06	Closed	Technical Specification 3.0.3 Entered Due to Inadequate Work Practices (paragraph 2.6).
LER	414/94-07	Closed	Reactor Trip and Technical Specification Violation (paragraph 3.8).

8.0 ACRONYMS

ACOT - Analog Channel Operational Test
AFW - Auxiliary Feedwater System
CFR - Code of Federal Regulations
DG - Diesel Generator
EOC - End of Cycle
gpm - gallons per minute
IAE - Instrument and Electrical
LER - Licensee Event Report
NRC - Nuclear Regulatory Commission
NSM - Nuclear Station Modification
OMP - Operations Management Procedure
pcm - percent milli rho

PIP	-	Problem Investigation Process
PORC	-	Plant Operation Review Committee
PORV	-	Power Operated Relief Valve
R&R	-	Removal and Restoration (Tagging Order)
SCBA	-	Self Contained Breathing Apparatus
S/G	-	Steam Generator
SSPS	-	Solid State Protection System
TS	-	Technical Specifications
VIO	-	Violation