UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
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ATLANTA, GEORGIA 30303-8931

March 16, 2001

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

SUBJECT: CATAWBA NUCLEAR STATION - NRC INSPECTION REPORT 50-413/01-03 AND 50-414/01-03

Dear Mr. Peterson:

On February 16, 2001, the NRC completed an inspection at your Catawba Nuclear Station. The enclosed report documents the inspection findings which were discussed on February 15, 2001, with Mr. P. Herran and other members of your staff.

The inspection examined activities conducted under your licenses as they relate to the identification and resolution of problems, and compliance with the Commission's rules and regulations and with the conditions of your licenses. The inspectors reviewed selected procedures and representative records, observed activities, and interviewed personnel.

On the basis of the sample selected for review, it was concluded that in general, problems were properly identified, evaluated, and corrected. During the inspection, however, an apparent violation was identified related to the failure to identify the root cause(s) and establish effective corrective actions to prevent repetitive water hammer events on the Unit 1 residual heat removal system which have resulted in the repeated failure of system support snubbers. This issue has not yet been characterized by the Significance Determination Process and has therefore not yet been dispositioned. Accordingly, please be advised that the number and characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review. No response regarding the apparent violation is required at this time.

There were also two issues of very low safety significance (Green) identified during the inspection concerning the full scope of problems not being identified with respect to an inoperable reactor vessel level instrumentation system channel and inoperable post accident monitoring recorders. These findings were determined to be examples of a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a Non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest the Non-cited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the United States Nuclear Regulatory

Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Catawba facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Robert C. Haag, Chief
Reactor Projects Branch 1
Division of Reactor Projects

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

Enclosure: NRC Inspection Report 50-413/01-03, 50-414/01-03
w/Attached NRC's Revised Reactor
Oversight Process

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-413, 50-414
License Nos:	NPF-35, NPF-52
Report No:	50-413/01-03, 50-414/01-03
Licensee:	Duke Energy Corporation
Facility:	Catawba Nuclear Station, Units 1 and 2
Location:	420 Concord Road York, SC 29745
Dates:	February 5 - 16, 2001
Inspectors:	M. Shannon, Senior Resident Inspector - Oconee (lead inspector) M. Giles, Resident Inspector E. Girard, Reactor Inspector, Region II M. Maymi, Reactor Inspector (in training), Region II
Approved by:	R. Haag, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000413-01-03, IR05000414-01-03, on 02/5-16/2001, Duke Energy Corporation, Catawba Nuclear Station, Units 1 & 2, annual baseline inspection of the identification and resolution of problems.

The inspection was conducted by two resident inspectors and a regional reactor inspector. The inspection identified two Green findings, both of which were considered examples of a non-cited violation (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using the Significance Determination Process (SDP) found in Inspection Manual Chapter 0609. Findings to which the SDP does not apply are indicated by “no color” or by the severity level of the applicable violation.

In addition, the inspection identified a potentially safety significant apparent violation that will require additional review and analysis.

Identification and Resolution of Problems

Overall, the licensee’s corrective action program was effective at identifying, evaluating, and correcting problems. The threshold for entering problems into the corrective action program was sufficiently low. Reviews of operating experience information were comprehensive. In general, the licensee properly prioritized items (by Action Category) in its corrective action program database, which ensured that timely resolution and appropriate causal factor analyses were employed commensurate with safety significance. Some exceptions were noted in the area of problem identification, where all relevant issues of problems were not identified and equipment performance was adversely affected. The inspection identified three exceptions in the area of prioritization and evaluation of issues, where more comprehensive root cause determinations would have provided more effective evaluations and corrective actions. In the area of effectiveness of corrective actions, it was noted that the corrective action program was not timely in resolving various documentation deficiencies with Technical Specification (TS) surveillances, Updated Final Safety Analysis Report changes and TS bases changes.

Previous non-compliance issues documented as non-cited violations were properly tracked and resolved via the corrective action program. The results of the last comprehensive corrective action program audit conducted by the licensee (September 1999) were properly entered and dispositioned in the corrective action program. Based on discussions with plant personnel and the apparently low threshold for items entered in the corrective action program database, the inspectors concluded that workers at the site generally felt free to raise safety concerns to their management.

A. <u>Inspector Identified Findings</u>

Cornerstone: Initiating Events

- To Be Determined. An apparent violation of 10 CFR 50, Appendix B, Criterion XVI was identified for the failure to identify a root cause and establish effective corrective actions to prevent repetitive water hammer events in the Unit 1 residual heat removal (ND) system which have caused the repeated failure of snubbers on supports 1-R-ND-0226 and 1-R-ND-0596. (Section 40A2.b.(2).2)

Cornerstone: Mitigating Systems

- Green. The first example of a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI was identified for a failure to identify a condition adverse to quality which contributed to a Unit 1 reactor vessel level instrument system (RVLIS) channel being inoperable. A quality control inspector did not initiate a Problem Investigation Process report after identifying that a RVLIS system terminal board was not reconnected (wired) in accordance with electrical drawings. Because of an electrical drawing error, the terminal board was then wired incorrectly and resulted in a failure to meet Technical Specification 3.3.3. Function 4 requirements for an inoperable RVLIS channel from June 1999 to November 4, 2000.

  Because other indications would have been available to the operators to mitigate the consequences of an accident, and based on the probability that the operators would have used the conservative indication of decreasing reactor vessel level from the operable RVLIS channel, the inspectors determined that this issue was of very low safety significance. (Section 40A2.a.(2).2)

- Green. The second example of a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI was identified for a failure to identify a condition adverse to quality which contributed to not recognizing that four post accident monitoring control room recorders in Unit 1 were inoperable from September 24 through September 29, 2000, and degraded from September 29 through October 19, 2000. Specifically, operators did not review applicable electrical drawings in order to identify which components were supplied from a failed electrical breaker. Consequently, they did not recognize that post accident monitoring control room recorders, which are used in the emergency operating procedures to determine mitigation strategies, were no longer operable.

  Because other indications would have been available to the operators to use in lieu of these accident monitoring recorders and because the Technical Specification Limiting Condition for Operation requirements were not exceeded, the inspectors determined that this issue was of very low safety significance. (Section 40A2.a.(2).3)

A. Licensee Identified Violations

A violation of very low significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. The violation is listed in section 4OA7 of this report.

Report Details

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

a. Effectiveness of Problem Identification

(1) Inspection Scope

This annual inspection reviewed licensee corrective action program (CAP) activities and included a review of CAP documents for issues documented in NRC inspection reports and the plant issues matrix. The inspectors focused on open corrective actions, on non-cited violations (i.e., correction of previous examples of non-compliance with NRC regulations), and on issues and corrective actions from operating experience reviews. For further insight into potential problems, CAP entries were discussed with the resident inspectors who routinely evaluated these activities as part of the NRC baseline inspection program.

The inspectors reviewed Problem Investigation Process reports (PIPs), which served as the licensee's formal means of documenting equipment and human performance problems, concerns, issues, and events. The inspectors also reviewed other CAP documents, including completed corrective actions documented in PIPs, and operating experience program (OEP) documents to verify that industry-identified problems potentially or actually affecting Catawba were appropriately entered into and resolved by the formal CAP process. Items included in the OEP effectiveness review were NRC Information Notices, industry or vendor-generated reports of defects and noncompliance under 10 CFR Part 21, and vendor information letters. A detailed listing of PIPs and OEP documents that were reviewed during this inspection is included at the end of this report.

The inspectors toured areas of the plant containing equipment important to safety. This included walkdowns of systems and components with issues documented in the PIPs that were being reviewed. The inspectors discussed issues identified during the PIP reviews with various system engineers, maintenance personnel, procedure writers, and other plant personnel to determine if the corrective action system was effective for identifying and tracking conditions adverse to quality (CAQ).

(2) Findings

.1 General

In general, the licensee's threshold for entering problems into the CAP was satisfactory. The inspectors did not identify any plant equipment problems or industry-related issues that had not been entered in the CAP. Based on the total number of PIPs generated at the Catawba site each year, the observed low threshold for documenting issues, and discussions with plant personnel, the inspectors concluded that the licensee's CAP was being effectively implemented for the identification and resolution of problems. This conclusion was based on a review of over 100 licensee initiated PIPs; which, for but a few discussed in the following sections, were considered by the inspectors to appropriately identify and resolve applicable problem areas.

.2 Quality Control (QC) Failure to Identify a Condition Adverse to Quality for Improper Installation of Important to Safety Plant Equipment

A Green finding was identified and dispositioned as an example of a non-cited violation (NCV) for a failure to identify a condition adverse to quality which contributed to a Unit 1 reactor vessel level instrument system (RVLIS) channel being inoperable from June 1999 to November 4, 2000. Specifically, the QC inspector did not initiate a PIP after identifying that a RVLIS system terminal board was not installed (wired) in accordance with electrical drawings during replacement activities. This contributed to a failure to meet Technical Specification (TS) 3.3.3. Function 4 requirements for an inoperable RVLIS channel.

During June 1999, terminal boards related to RVLIS were replaced. The technicians performed independent verifications of the wiring prior to removal of the terminal board leads and performed independent verification of the relanded terminal board leads. Subsequently, a QC inspector performed a review of the work performed using engineering drawings and identified that two leads were landed differently (reversed) than the drawings indicated. The QC inspector brought this to the attention of the technicians, and the leads were relanded in accordance with the electrical drawings. However, neither the QC inspector nor the technicians documented that the terminal board had not been installed in accordance with the electrical drawings in either a PIP or in the work order. Subsequently, on November 4, 2000, the licensee discovered that the “A” train RVLIS upper level detector, used for reactor vessel level indication, had a blown circuit fuse and had been inoperable (with a failed high reactor vessel level indication) since June 1999. The licensee subsequently determined that the electrical drawings were in error, and issued the TS required fourteen day report to the NRC for the RVLIS channel being inoperable for greater than 30 days.

In addition to other proposed corrective actions, licensee PIP 00-05558 discussed the lack of questioning attitude by the technicians when the wiring was found not to be in conformance with the drawings and the inadequate post maintenance testing plan specified for the terminal board replacement. The corrective actions did not identify that the QC inspector had failed to initiate a PIP when he initially identified that the wiring was not in conformance with the engineering drawings, even though the technicians had performed an independent verification of the wiring prior to and after the terminal board replacement.

The failure to initiate a PIP in June 1999 for the as found wiring discrepancy resulted in a missed opportunity to identify the electrical drawing error and prevent the extended inoperability of the RVLIS channel. This deficiency is more than minor because it resulted in an actual impact on safety in that the “A” train RVLIS channel remained inoperable for approximately 18 months. In addition, the failure of QC inspector to adequately document personnel work errors, identified during a QC review, could mask potential inadequate personnel performance issues and appropriate corrective actions would not be implemented. The inaccurate information from this RVLIS channel, which would have always indicated a full reactor vessel, could have resulted in some confusion following certain loss of coolant accidents; thereby affecting the mitigating system cornerstone. However, because other indications would have been available to the operators to mitigate the consequences of an accident, and based on the high probability that the operators would have used the conservative indication of decreasing reactor vessel level from the operating RVLIS channel, this issue has very low safety significance (Green).

10 CFR 50, Appendix B, Criterion XVI, requires that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. These requirements are implemented through the licensee's Quality Assurance Program by Nuclear System Directive (NSD) 208, Problem Investigation Process. NSD 208 requires that the improper installation of equipment important to safety be documented in the PIP program. The failure of the QC inspector to document the wiring error for the RVLIS terminal board when identified in June 1999 was considered to be a failure to identify a condition adverse to quality and determined to be the first example of a violation of 10 CFR 50, Appendix B, Criterion XVI. This violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as the first example of NCV 50-413/01-03-01: Failure to Identify Conditions Adverse to Quality. This violation has been captured in the licensee's corrective action program as a revision to PIP C-00-05558.

.3 Failure to Identify Safety Related Components Supplied from a Failed Power Supply Breaker

A Green finding, dispositioned as another example of a NCV, was identified for the failure to identify a condition adverse to quality which contributed to not recognizing that four post accident monitoring control room recorders were inoperable from September 24 through September 29, 2000, and degraded from September 29 through October 19, 2000. Specifically, operators did not review applicable electrical drawings in order to identify which components were supplied from a failed electrical breaker and therefore did not recognize that post accident monitoring control room recorders were no longer operable.

On September 24, 2000, operations personnel identified supply breaker 1EKPY-03 in the trip free (tripped) position and initiated a work request to address the deficiency. At the time, the operators inappropriately assumed the 1A hydrogen analyzer still had power available and was operable (based on the lit indication lights on the hydrogen analyzer.) However, based on discussions with licensee personnel and the documentation in PIP C-00-04692, the inspectors concluded that the operators had not verified the loads supplied by this breaker by either reviewing the equipment load list or electrical diagrams following discovery of the breaker in the trip free condition. Based on this failure, the licensee did not understand the consequences of this supply breaker being in the tripped condition. On September 29, 2000, during the TS required 92 - day hydrogen analyzer channel calibration, technicians found the supply breaker in the tripped position. The technicians reset the breaker and the channel calibration was successfully completed. At that time, the technicians initiated PIP C-00-04692 to document that the supply breaker had been found in the tripped condition. Subsequently, on October 19, 2000, when the breaker was removed for testing, it was noted that the breaker "failed to mechanically operate properly" in that slight vibration would cause the breaker to trip. The breaker was subsequently replaced.

During a review of an electrical drawing to address one of the recommended corrective actions, the licensee determined that the breaker supplied control power to the Train “A” control room chart recorders for containment sump level, containment hydrogen concentration, containment pressure, and control room hydrogen concentration. These recorders are part of the post accident monitoring instrumentation and are included in a 30 day limiting condition for operation (LCO) as specified by TS 3.3.3. Since the breaker was left in the tripped condition between September 24 and September 29, 2000, the associated recorders were rendered inoperable. Based on the licensee’s determination that a slight vibration could cause the breaker to open, the inspectors concluded that a degraded condition had existed from September 29, 2000, until the breaker was finally replaced on October 19, 2000.

The corrective actions for PIP C-00-04692 had recommendations to improve the channel check for the hydrogen analyzer and to provide awareness training to the licensee’s staff for understanding the operation of the hydrogen analyzer. The PIP did not identify that the operators failed to review the electrical drawings when the breaker was found in the trip free position on September 24, 2000, and therefore did not determine the operability of the affected equipment. This failure to determine what components were affected resulted in a failure to identify that four post accident monitoring instruments were inoperable. This resulted in a failure to take prompt corrective actions to replace the defective breaker.

The failure to review the electrical diagrams to determine operability of equipment was considered to be more that minor because it had an actual impact on safety in that the operators did not realize that these recorders, which are used in the emergency operating procedures to determine mitigation strategies, were not functioning. For example, the operators use the containment pressure recorder to determine if a safety injection signal had existed for high containment pressure in EP/1/A/5000/E-0, Reactor Trip or Safety Injection, and for this period the failed indication would have provided misleading information to the operators if it had been needed. Additionally, the failure to identify this human performance error in the PIP precluded the CAP from addressing this item. Because other indication would be available and because the TS Limiting Conditions for Operation requirements were not exceeded, this issue was determined to have very low safety significance (Green).

10 CFR 50, Appendix B, Criterion XVI, requires that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. This requirement is implemented through the licensee’s Quality Assurance Program by NSD 208, which requires “that a PIP be initiated when equipment or personnel do not perform as expected.” The failure of the operators to perform an adequate review to determine the operability of the post accident monitoring instrumentation was considered to be a failure to identify a condition adverse to quality and determined to be the second example of a violation of 10 CFR 50, Appendix B, Criterion XVI. This violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as the second example of NCV 50-413/01-03-01: Failure to Identify Conditions Adverse to Quality. This violation has been captured in the licensee’s corrective action program as a revision to PIP C-00-04692.

b. Prioritization and Evaluation of Issues

(1) Inspection Scope

The inspectors reviewed PIPs that were assigned various Action Categories to determine whether issues were properly prioritized and evaluated in accordance with NSD 208. The Action Categories (1 through 4) were defined in NSD 208 and were numbered based on decreasing significance. Action Category 1 PIPs were "significant" CAQs that required formal root cause evaluations, while Action Category 4 PIPs were low level CAQs or conditions not adverse to quality, neither of which required any type of causal evaluation. The majority of the reviewed PIPs were screened as Action Category 3, with the remainder falling into Action Categories 1, 2, and 4. Action Category 2 PIPs were defined as CAQs for which management could use its discretion in deciding whether to perform a formal root cause evaluation. Action Category 3 PIPs were problems for which an "apparent cause" analysis was sufficient in fixing the immediate problem.

(2) Findings

.1 General

In general, the licensee's threshold for prioritization and evaluation of problems in the CAP was considered to be satisfactory. The inspectors noted that the technical adequacy and depth of evaluations, as documented in the corrective action program, were generally acceptable. However, as discussed below, the inspectors did identify PIPs where the root cause evaluations were considered to be incomplete. This finding is similar to a previous licensee corrective action assessment finding that categorized some of the root cause evaluations as "narrow" in scope and detail (see Section 4OA2.d). Based on the total number of PIPs with root cause evaluations that were reviewed during this inspection, the inspectors concluded that the licensee's corrective action program was being effectively implemented for the prioritization and evaluation of problems. This conclusion was based on a review of over 100 licensee initiated PIPs; which, for but a few discussed in the following sections, were considered by the inspectors to appropriately prioritize and evaluate the identified problem areas.

.2 Failure To Identify A Root Cause and Establish Effective Corrective Actions To Prevent Repeated Water Hammer Events In The Residual Heat Removal System

Brief Overview

An apparent violation (EEI) of 10 CFR 50, Appendix B, Criterion XVI was identified for the failure to identify the root cause(s) and establish effective corrective actions to prevent repetitive failures of snubbers on supports 1-R-ND-0226 and 1-R-ND-0596 (hereafter referred to as snubbers 226 and 596). The licensee concluded that the failed snubbers were caused by water hammer events in the Unit 1 residual heat removal (ND) system and documented this in PIP C-99-02978. A review of the safety significance of the water hammer events on the Unit 1 ND system is pending.

Background

The review of historical data noted that snubber testing performed since September 1986 documented five failures on snubber 226, and six failures on snubber 596. The most recent failures, in which both snubbers failed, occurred in July 1999 and were documented in PIP C-99-02926. Snubbers 226 and 596 are located on the 1A ND pump discharge piping. Snubber operability and testing requirements are documented in Selected Licensee Commitments (SLC) Manual requirement 16.9-13, Snubbers. The licensee, in accordance with these SLC requirements, performs visual inspections, freedom of motion testing, and functional testing. The failures mentioned above were identified during the specified freedom of motion testing. The inspectors noted that snubbers 226 and 596 were last stroked successfully in October 1997, but were found in a failed (locked-up) condition on July 1999.

Assessment

The inspectors, upon review of PIP C-99-02978, noted that numerous historical failures had been occurring, and had been previously documented in Problem Investiagtion Reports (PIRs) 1-C90-00219, 1-C91-00103, and 1-C91-00364, and PIPs 1-C93-00847, and 1-C99-02926. However, these corrective action documents did not identify the actual cause of the water hammer, the actual location of the water hammer or a specific operational evolution during which the water hammer condition was occurring.
PIP C-99-02978 indicated that corrective actions implemented by PIP 1-C93-00847, due to failures that occurred in October 1993, had adequately resolved the water hammer problem. However, since the water hammers were still occurring, the corrective actions were inadequate due to not clearly establishing the root cause(s).

Corrective actions specified in PIP 1-C93-00847, which were implemented in an attempt to prevent future failures, consisted of modifying ND system procedures to prevent rapid depressurization of ND suction and discharge piping, and to enhance operator training to make operators aware of the potential for water hammer damage created by isolating hot reactor coolant fluid in the ND system and allowing it to depressurize and cause voids to form in the system. As a result of the July 1999 failures, the licensee installed an accelerometer on hanger 1-R-ND-0226 to obtain data for determining the operational conditions during which the water hammer events were taking place. Based on accelerometer data obtained by the licensee during the October and November 2000 refueling outage, several potential water hammer events occurred. However, at the conclusion of the inspection, the licensee had not inspected the two snubbers to determine if they had been damaged during those events.

In addition, the inspectors observed that PIP C-99-02978, generated to address the need for further evaluation, was improperly screened as a Category 4 PIP, not requiring a root cause analysis to be performed. The inspectors, after review of NSD 208, concluded that this screening was inappropriate based on the screening guidelines established for plant equipment that is important for maintaining nuclear safety, including risk significant components.

Based on the number of repetitive failures which have occurred on snubbers 226 and 596 since 1986, the inspectors concluded that the corrective actions implemented thus far have been inadequate in preventing recurring failures and that the root cause(s) has yet to be determined by the licensee. The inspectors also determined that based on the potential significance of water hammer events in an emergency core cooling system, the licensee should have generated actions to identify the root cause and evaluate the effects of the water hammer condition. The inspectors noted that the licensee has not performed an evaluation of whether design pipe loading limits have been exceeded; has not performed non-destructive testing or detailed piping inspections on the ND system discharge piping to identify whether any fatigue failure damage has occurred; has not addressed potential water hammer conditions on the opposite train or Unit 2; has not performed an operability evaluation to determine if the ND system discharge piping is currently operable with recurring water hammers; and has not determined if the snubber failures, due to excessive design loading, constitute an unanalyzed condition.

Significance

Because water hammer events can greatly exceed piping design specifications and have led to piping failures at other facilities, the inspectors concluded that having water hammer events at unspecified times and with unknown magnitudes constituted a potential safety significant condition. For this specific case, a failure of the ND system piping could lead to an inter-system loss of coolant accident (LOCA) outside of containment, and would require the operators to take manual actions (under high stress conditions) to isolate the ruptured piping and restore reactor coolant inventory. Because of the unknown impact that these water hammer events have had on the ND system, the safety significance of this issue cannot be assessed at this time. Therefore, further reviews by the NRC staff of information currently available and/or developed by the licensee in the future will be necessary.

Enforcement

10 CFR, Appendix B, Criterion XVI, requires that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. This requirement is implemented through the licensee's Quality Assurance Program by NSD 208. Contrary to the requirements of Criterion XVI, the actions taken by the licensee in determining a root cause and in implementing corrective actions to prevent water hammer events since 1986 have been inadequate in correcting this CAQ. The inspectors considered this failure to promptly identify and correct a CAQ as a violation of 10 CFR 50, Appendix B, Criterion XVI. This violation is being treated as an apparent violation, and pending review of the safety significance, is identified as EEI 50-413/01-03-02: Failure to Promptly Identify and Correct the Unit 1 Residual Heat Removal System Water Hammer Condition. This apparent violation is in the licensee's corrective action program as a revision to PIP C-99-02978.

.3 Incomplete Evaluations for Equipment Deficiencies

A negative observation was identified with two examples in which the documented evaluations for equipment deficiencies were not thorough and that the potential for future degradations was not addressed.

The first example dealt with the evaluation for surveillance stroke testing failures of valve 2CA-40, Motor Driven Auxiliary Feedwater Pump B Discharge to Steam Generator

D Control Valve, and of valve 2CA-44, Motor Driven Auxiliary Feedwater Pump B Discharge to Steam Generator C Control Valve. Both valves utilize a Fisher Model 3582i positioner. A similar valve, 2CA-60, Motor Driven Auxiliary Feedwater Pump A Discharge to Steam Generator A Control Valve, which also utilizes the Fisher Model 3582i positioner, had previously failed stroke time testing performed on May 12, 2000. The root cause of this failure, documented in PIP C-00-02514, was determined to be debris found in the positioner pneumatic signal ports. A corrective action from this PIP was to inspect other valves utilizing the Fisher Model 3582i positioner due to the potential for common-mode failure from foreign material located in the positioner.

On July 26, 2000, the signal ports for valves 2CA-40 and 2CA-44 were inspected and no foreign material was found. During post-maintenance testing (stroke-time testing), both valves failed their initial test and a second subsequent stroke test. It was noted that 2CA-40 had a stroke-time acceptance range of 1.70 to 5.10 seconds with actual stroke times of 0.5 and 0.4 seconds, and 2CA-44 had an acceptance range of 1.55 to 4.65 seconds with actual stroke times of 0.9 and 0.9 seconds. Prior to these failures, 2CA-40 and 2CA-44 had previously demonstrated acceptable valve operation with respective stroke times of 2.40 and 2.90 seconds.

Following the valve retest failures, the licensee performed an evaluation which concluded that operation of both valves was acceptable. The extent of the licensee's written evaluation supporting this conclusion was that 10 seconds was the plant design limiting stroke time for these valves. The inspectors questioned the adequacy of this evaluation because it did not provide the root cause of why both valves failed to meet the established stroke time acceptance criteria. The inspectors concluded that by not determining the root cause of these failures, the licensee would not be able to adequately evaluate the condition of both valves or to determine whether internal degradation existed. During subsequent quarterly stroke testing, valves 2CA-40 and 2CA-44 have exhibited stroke times that were similar to test results obtained prior to the failures on July 26, 2000.

The second example dealt with a potentially degraded condition on the Unit 1 turbine driven auxiliary feedwater (TDAFW) pump turbine which had a high temperature bearing condition. On November 20, 2000, following the shutdown of the TDAFW pump after auxiliary feedwater system flow balance testing, a high temperature alarm was received for the turbine outboard bearing. Following the shutdown, the actual bearing temperature increased to approximately 195 degrees F which exceeded the alarm setpoint of 190 degrees F. As appropriate, operations personnel generated PIP C-00-05895 to document this condition and to request that engineering perform an evaluation to address this unexpected condition. The subsequent evaluation concluded that the cause of the high bearing temperature was due to bearing cooling being secured when the pump was stopped. This conclusion was based on the fact that the bearing lube oil is only supplied to the bearings when the TDAFW pump is operating. Based on this, the licensee concluded that no adverse condition existed and no further corrective actions would be necessary.

The inspectors reviewed PIP C-00-05895 and concluded that the documented engineering evaluation was lacking in that it had not obtained or considered the pump operating parameters prior to the high temperature condition nor had any comparison of bearing temperatures been reviewed. To further assess this condition, the inspectors reviewed the data from Unit 1 TDAFW pump test completed on February 15, 2001, and the Unit 2 TDAFW pump test completed on February 1, 2001. The recorded temperatures for the Unit 1 TDAFW turbine's outboard bearing while running was 150 degrees F, as compared to the Unit 2 TDAFW turbine's outboard bearing temperatures of 110 degrees F.

During the inspection, the licensee identified to the inspectors that prior trending of TDAFW pump turbine bearing temperatures had been performed and some temperature trending data was given to the inspectors. An engineer for the licensee contended that he was aware of previous bearing temperatures and considered them when evaluating PIP C-00-05895, although it was not documented in the PIP. Following the inspection on February 22, 2001, the licensee generated PIP C-01-00868 to document the inconsistencies between the Unit 1 and Unit 2 TDAFW pump turbine outboard bearing temperatures and also identified that increasing trends could result in additional control room alarms and operability issues. The inspectors were still concerned that the higher temperature (approximately 40 degrees F) for the Unit 2 TDAFW pump turbine outboard bearing had not been adequately addressed from a root cause perspective. Based on a maximum operating temperature limit of 220 degrees F for the lube oil which had not been exceeded, the licensee concluded that the TDAFW pump turbine bearing high temperature would not render the TDAFW pump inoperable.

c. Effectiveness of Corrective Actions

(1) Inspection Scope

The inspectors reviewed PIPs to assess the licensee's actions in determining causal factors, to develop and implement appropriate actions to correct the adverse condition, and, if significant, prevent recurrence. These PIPs were primarily related to cornerstones in the Reactor Safety strategic performance area of the NRC inspection program; however, PIPs were also reviewed in the areas of Radiation Safety and Safeguards to maintain some distribution across all NRC inspection program cornerstones. PIPs associated with past NCVs were reviewed to verify that the associated problems were corrected.

The inspectors reviewed industry operating experience issues that were evaluated in the past two years to determine if this information had been appropriately assessed for applicability to the station and whether applicable issues were incorporated into the station's corrective action program. Items reviewed for the OEP included vendor information letters (VILs), NRC Information Notices (INs), and NRC Generic Letters (GLs) .

In addition, the inspectors interviewed plant personnel directly involved with the corrective action program, as well as those cognizant of specific technical issues, to verify and understand corrective actions associated with the items listed above.

(2) Findings

.1 Untimely Corrective Actions for Documentation Deficiencies

A negative observation was identified regarding the licensee's corrective action process not being timely in resolving various documentation deficiencies related to TS surveillance, Updated Final Safety Analysis Report (UFSAR) and TS bases changes. The basis for this observation is provided in the following examples:

- Surveillance Testing Deficiencies Associated With the 7300 Channel Calibration Procedures

  PIP C-98-04725, dated December 9, 1998, documented that "some 7300 channel calibration procedures do not provide explicit instructions to ensure satisfactory completion of the requirements of a channel calibration as defined by Technical Specifications...the procedure layout is ambiguous such that it does not provide specific guidance for verification of the trip function status light which would provide satisfactory operation of the channel alarms and trip functions... Continuing to rely on informal work practices is not prudent and is not in compliance with established station directives and processes." The 7300 procedures are related to the calibration and testing of the reactor protection system and safety injection/emergency core cooling system protection circuits.

  The inspectors did not identify any specific procedures that may have been deficient and the licensee did not document any evaluation that may have been done to determine the preferred priority of revisions to the 7300 procedures. The licensee indicated that the recommended revisions specified with this PIP were added to the procedure enhancement program for the 7300 procedures, which included approximately 23 separate enhancements to 262 procedures. Although the original plan was to have all 262 related procedures updated by the end of 1999, as of the last update, on August 22, 2000, to the corrective actions for PIP C 98-04725, only about 60 procedures had been revised and approved for use and the next due date for completion of the procedure revisions was March 3, 2001. The inspectors concluded that the licensee's efforts to resolve 7300 calibration procedure deficiencies lacked prioritization or evaluation to support the continued delays in revising the procedures.

- Surveillance Testing Deficiencies Associated with Improper Setting of the Rate Thumb Wheel Resulting in Improper Rate Setting in the 7300 Cabinets

  PIP C-99-00079, dated January 7, 1999, documented that a thumb wheel was found out of adjustment in one of the 7300 cabinets. This resulted in an inappropriate setting for the lead/lag circuit for the reactor protection system delta T circuit. This also adversely affect the over power delta temperature (OPDT) and over temperature delta temperature (OTDT) circuits. Thus neither the OPDT not the OTDT circuits met the requirements for TS allowable values. This issue was identified as Licensee Event Report (LER) 414/98-07 and NCV 99-01-03. The licensee's immediate corrective actions included a directive to the technicians to not adjust the rate thumb wheel unless specified in the procedure.

Proposed corrective actions for this violation included revising the related 7300 procedures so that the technicians would still be able to adjust the rate thumb wheel to zero to speed up completion of the calibration procedure. In addition, the procedure would also have a step to ensure that the rate thumb wheel was then returned at the proper setting. At the time of the inspection, it was noted that as of the last update, on December 20, 2000, only four of the 44 applicable procedures had been revised and approved for use. The due date for completion of all of the procedures was April 2, 2001. The inspectors concluded that the licensee's corrective action program was not timely in resolving this issue.

- <u>Deficiencies with the Documentation in the UFSAR Related to the Need and Effects of the TDAFW Pump Steam Supply Heat Trace System</u>

  PIP C-97-00616, dated March 6, 1997, documented that "an engineering review of the heat trace system installed on the steam supply line to the TDAFW pump has identified concerns that should have been resolved when this problem occurred several years ago...instances of configuration management deficiencies, original design deficiencies, omissions of relevant information from the UFSAR, and potential failures to accurately determine reportability were discovered."

  PIP C-97-00616, corrective action number 4, proposed an evaluation be conducted for inclusion of the TDAFW pump steam line heat trace system into the UFSAR. The licensee noted that "as the TDAFW pump steam line system is relied upon to maintain operational readiness of the safety related TDAFW pump, a discussion should be included in the UFSAR." Subsequent discussions with the licensee noted that the heat trace system, relating to the TDAFW pump steam supply, should have been included in the UFSAR to address the potential for causing a steam line rupture in the auxiliary building. The inspectors noted that neither the potential unanalyzed condition nor the potential outside design basis issues were addressed as potential issues to be included in the UFSAR update.

  The inspectors noted that at the time of the inspection, the UFSAR had not been updated to include the reliance of the heat trace system for operability of the TDAFW pump or the potential adverse effects. The inspectors concluded that the licensee has not updated the UFSAR for this specific issue within a reasonable timeframe.

- <u>Deficiency with the TS Bases for Surveillance Requirement 3.8.4.8</u>

  PIP C-00-00944, dated March 2, 2000, documented that the vital battery amperage value acceptance criteria, specified in the TS bases, is "totally wrong." In addition, this PIP noted that "anyone looking at the test results and comparing them to the bases could be mistaken that the batteries were operable based on the values given when the batteries could very much be inoperable." The inspectors noted that the values used in the TS surveillance procedure were correct for these batteries.

The inspectors noted that at the time of this inspection, the TS bases for TS 3.8.4.8 had not been revised. A TS bases change can be performed by the licensee as necessary, after appropriate evaluations. The inspectors concluded that the licensee's corrective action program did not appear to be timely in resolving this TS bases deficiency based on the importance of the TS and the need to have accurate TS bases.

d. Effectiveness of Self-Assessments and Audits

(1) Inspection Scope

The inspectors reviewed the licensee's most recent self-assessment of the corrective action program to verify if findings and recommended areas for improvement were being entered into the licensee's CAP, and that appropriate corrective actions were taken to resolve identified CAQs or program deficiencies. As applicable, self-assessment findings were compared to recent NRC findings. The self-assessment was conducted by the Nuclear Assessment and Issues Division, Nuclear Performance Assessment Section from the Duke Energy General Office from September 13-30, 1999, and was identified as SA-99-35 (ALL)(RA), Level 3 Assessment of the Corrective Action Program, requested by Safety Assurance Business Excellence Steering Team. The findings from this assessment were documented in PIP G-99-00352.

(2) Findings

The inspectors determined that the findings noted in the previous sections of this inspection report were similar to those identified in the 1999 licensee self-assessment of the corrective action program. The review indicated that the self-assessment was thorough and effective in identifying deficiencies in the corrective action program and other programmatic areas. These deficiencies were routinely entered into the CAP, with areas for improvement being identified for all three Duke facilities.

e. Assessment of Safety-Conscious Work Environment

(1) Inspection Scope

The inspectors discussed the issue of maintaining a safety conscious work environment while performing follow-up activities related to the PIP review. Specifically, personnel were asked questions regarding any reluctance to initiate PIPs and adequacy of corrective actions for identified issues.

(2) Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. P. Herran, as well as other members of licensee management and staff, at the conclusion of the inspection on February 15, 2001. A subsequent meeting was held with Mr. R. Sweigart and other licensee staff members to discuss inspection results. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations. The following finding of very low significance was identified by the licensee and constitutes a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as a NCV.

NCV Tracking Number	Requirement Licensee Failed to Meet
50-413/01-03-03	10 CFR 50, Appendix B, Criterion III, requires in part that the design bases is correctly translated into drawings. 10 CFR 50, Appendix B, Criterion XI, requires in part that all testing required to demonstrate that components will perform satisfactorily in service is identified and performed. An error in the electrical drawings for the Unit 1 reactor vessel level indication system (RVLIS) circuitry was introduced during a previous drawing revision on July 1, 1985, which led to the improper wiring of the RVLIS instrumentation in a June 1999 modification. Following the modification activities, the licensee did not develop an adequate post modification testing plan for the RVLIS electrical circuitry, resulting in one channel of RVLIS being inoperable for 18 months. This is captured in the licensee's corrective action program under PIP C-00-05558.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Peterson, Vice President, Catawba Nuclear Station
R. Glover, Manager, Plant Operations
M. Boyle, Manager, Radiation Protection
G. Gilbert, Manager, Regulatory Compliance
R. Sweigart, Manager, Safety Assurance
R. Jones, Station Manager, Catawba Nuclear Station
R. Parker, Manager, Maintenance
P. Herran, Manager, Engineering

ITEMS OPENED AND CLOSED

Opened

50-413/01-03-02	EEI	Failure to Promptly Identify and Correct the Unit 1 Residual Heat Removal System Water Hammer Condition (Section 40A2.b(2).2)

Opened and Closed During this Inspection

50-413/01-03-01	NCV	Failure to Identify Conditions Adverse to Quality - two examples (Sections 40A2.a(2).2 and .3)

LIST OF DOCUMENTS REVIEWED

PIP Number	Action Category	PIP Description
C-01-00001	3	Reactor makeup water storage tank level indication is increasing erratically due to suspected freezing of the transmitter.
C-01-00024	3	Unable to obtain samples from reactor makeup water storage tanks. Sample lines may be frozen.
C-01-00083	3	Component cooling miniflow valve 2KC-C40B failed to open.
C-01-00201	4	The partial stroke testing performed on the main steam isolation valves during operations conflicts with the basis statement for TS 3.7.2.1.
C-01-00418	3	The 2A RN pump went into the “Required Action” range during Section XI pump testing.
C-00-00335	3	Operations use of technical specification 3.7.8 may not be conservative.

PIP Number	Action Category	PIP Description
C-00-00592	1	(NCV 00-03-05) OAC point for ND/NS sump pumps was deleted from processing and it was required to comply with compensatory action 0-C98-4098.
C-00-01692	3	Auxiliary feedwater valve 2CA-15A failed to open during loss of normal suction test.
C00-01795	3	A procedure for safety injection and chemical and volume control check valve testing did not specify the correct test instrumentation.
C-00-02101	4	Inconsistency between technical specification 3.3.2 and 3.7.5 for swapping the auxiliary feedwater suction source to nuclear service water.
C-00-02176	3	The automatic isolation pathways for the containment purge system and the containment air release and addition system are not checked in accordance with selected licensee commitments.
C-00-02813	3	Possible inadequate engineering assessment of operability and failure of work request review process to identify operability concerns.
C-00-02941	1	The 2B feedwater pump trip logic was not maintained in a trip condition during pump calibrations.
C-00-03097	4	Quality control inspections and procedure signoffs were not completed for motor-operated valve maintenance.
C-00-03194	3	Repeated repairs to the closed limit switch on manual valve 1CA-67.
C-00-03250	4	The Unit 2 power range detector was discovered inoperable. There was no clear procedure flowpath to restore the detector to service.
C-00-03483	2	The 1B containment spray pump breaker reopened following the auxiliary safeguards start signal.
C-00-03492	4	Evaluate nuclear service water valve verification surveillance procedure to determine if it meet the intent of TS 3.7.8.1.
C-00-04089	3	Unable to correct out of tolerance condition on out of core instrumentation. Engineering evaluation is needed.
C-00-04242	4	Evaluate charging system check valve back leakage test for adequacy.

PIP Number	Action Category	PIP Description
C-00-04330	3	Knife switches in area terminal cabinet 2AC23 were found open and identified with an extremely old work order.
C-00-04410	3	Unplanned entry into technical specification 3.3.4 due to remote shutdown instrumentation surveillance test failure.
C-00-04547	3	The pre-defined work order for inspection of nuclear service water return to the standby nuclear service water pond does not require orifice inspection.
C-00-04565	3	The on-duty operations shift was not aware that a work order was being performed and that potential past operability issues were involved.
C-00-04646	3	Surveillance testing requirements for diesel generator and 4160 bus protective relaying are unclear.
C-00-04752	3	Industry operating experience issue for evaluating transients and updating fatigue analyses for the pressurizer main and auxiliary spray lines
C-00-04810	3	Engineering evaluation required for out of tolerance condition found on isolation amplifier NM306.
C-00-05201	3	Inadequate wall thickness at two charging system welds.
C-00-05228	3	Questioned whether capillaries of non-safety containment pressure switches had ever been filled. Set points were found to be low.
C-00-05272	3	No surveillance work orders referenced selected licensee commitments 16.7-9.5 and -9.6 for testing auxiliary feedwater valves 1CA-48 and -187.
C-00-05424	3	Potential RN strainer fouling in that the resistance factor is decreasing over time.
C-00-05709	1	While performing a pressurizer channel calibration, an invalid pressurizer low pressure signal actuated the Unit 1 train B safety injection logic.
C-00-05735	3	Upon venting the vacuum refil rig, the pressurizer level dropped rapidly.
C-00-05770	3	Possibly missed Residual heat removal breaker alignment verification required by technical specification surveillances 3.4.7.3 and 3.4.8.2 prior to entering mode 5.
C-00-05828	4	Increase in freeze protection corrective maintenance.

PIP Number	Action Category	PIP Description
C-00-05844	3	The set points for cold leg accumulator release valves 1NI-74 and -86 were outside of the test acceptance criteria.
C-00-05888	4	TS 3.7.1 does not provide clear guidance with respect to main steam safety valve testing.
C-00-05907	3	The calibration procedure for pressurizer level does not correctly compensate for the static pressure shift of 2235 psi.
C-00-06016	3	Inservice testing found Residual heat removal pump 1A differential pressure below requirement and flow marginally acceptable.
C-00-06140	4	Evaluate current condition of diesel generator cooling water heat exchangers which are corroding.
C-00-06316	3	The RN “B” train pump went into “Alert” range during Section XI pump testing.
C-00-06493	4	Technical specification 3.7.1 does not provide clear guidance for main steam safety valve testing.
C-00-06495	3	Low pressure service water header pressure detectors are freezing.
C-99-00747	1	(NCV 99-04-06) Valves failed normal response time testing, the acceptance criteria used for IWV testing may also be incorrect.
C-99-01057	3	Procedures may not satisfy the channel Operational Test requirements of Selected Licensee Commitments Tables 16.11-2 and -3.
C-99-02373	1	(NCV 99-07-03) Charging pump 1B low flow and pressure is causing the pressurizer level to decrease.
C-99-02467	3	Primary thermal power error of 3%.
C-99-02518	3	Charging pump flow control valve 2NV-294 not controlling pressure level.
C-99-02519	3	Unit 2 “D” bank of pressurizer heaters would not energize.
C-99-03482	2	The 2A containment spray pump tripped during an auxiliary safeguards test. The breaker closed then immediately opened and did not reclose.

PIP Number	Action Category	PIP Description
C-99-03097	1	(NCV 99-07-04) Testing procedure enhancement for the diesel generator undervoltage and degraded voltage relays should have been reported in 1996 as a missed surveillance.
C-99-03139	1	(NCV 00-01-02) Prohibited use of control room pressure boundary compensatory actions.
C-99-03940	3	Unplanned entry into Technical Specification 3.3.1 due to detector current meter fluctuations.
C-99-04053	4	Out of tolerance identified on the emergency diesel generator battery cell 9.
C-99-04079	2	Failure to declare auxiliary diesel power battery 1DGBA inoperable on finding that the voltage was below technical specification limits on two cells.
C-99-04087	3	Documents evaluation for PIPs C-99-04913 and -4053.
C-99-04433	3	Control room ventilation can't be restored within the 45 minutes stated in the station blackout design basis documents and the response to the NRC.
C-99-04897	2	Feedwater valve 2CF-51 failed to fully close during test.
C-99-04913	3	Uncertainty in technical specification requirements for safety related pressurizer heaters when energizing power is unavailable.
C-99-04953 and -04053	4	Voltage below technical specification limits on two cells of auxiliary diesel power battery 1DGBA.
C-98-00056	3	The 7300 process and control channel calibration procedures do not verify bistable test contacts return to service during restoration.
C-98-00612	1	(NCV 99-02-02) TS tables 3.3-3 and 4.3-2 item 11.a conflicts with item 11.c, literal compliance has not been met.
C-98-02094	3	A review of operation surveillance procedures identified instruments and indications not covered by maintenance calibration procedures or the work management system preventive maintenance program.
C-98-02207	2	The 2A containment spray pump tripped during an auxiliary safeguards test. The breaker closed then immediately opened and did not reclose.

PIP Number	Action Category	PIP Description
C-98-02463	1	The analog channel operational test procedure for over temperature delta temperature was incorrect. As a result, the channel was not calibrated within allowable limits.
C-98-03252	2	The penetration and bypass leakage rate measured on containment purge 2A carbon absorber bank exceeded the acceptance requirements.
C-98-03331	4	(Operating Experience) Respond to prevent event questions described in Significant Event Report 3-98.
C-98-03866	3	Emergency operating procedure FR-H.1 may not successfully mitigate a loss of all feedwater.
C-98-04064	3	(Operating Experience) Response to SOER 98-2 for circuit breaker failures.
C-98-04064	3	Significant Operating Experience Report 98-2 regarding industry circuit breaker failures.
C-98-04098	1	(NCV 99-07-06) FSAR and SER describe an interlock between the liquid waste system and the solid state protection system that apparently was not installed.
C-98-04627	3	Technical specification 3.7.10, Control Room Area Ventilation System, specifies an alignment which is not defined and the technical specification basis are unclear and inaccurate.
C-97-01579	1	(NOV 98-01-05) Vortex formation at outlet of the AFW condensate storage tank could lead to air intrusion into suction of pumps.
C-97-01620	2	The 2A containment spray pump tripped during in-service testing.
C-97-01639	1	Current plant procedures are not adequate to ensure compliance with technical specification 4.5.3.2.
C-97-03621	1	Scenario exists where turbine driven auxiliary feedwater pump flow to two steam generators cannot be stopped.
C-96-02830	3	(Operating Experience) Potentially incorrect material in keys connecting residual heat removal flow control valve stems to actuators.
C-95-02073	3	The containment purge ventilation system carbon filter failed the bypass leakage performance test.
C-94-01555	3	Corrective actions to consider periodic testing to ensure orifice degradation in the RN system is identified.

Operating Experience Program Documents

OEP #	Description
00-024508	Manufacturer deficiency with Barton 752/753 transmitters. PIP C-99-04330
00-024349	Notification from vendor of possible defective circuit board. PIP C-00-00367
00-024219	High temperature paint supplied as QA-1 has not been qualified. PIP C-00-00231
00-024125	Trip rollers supplied by ABB for use in breakers may not be properly hardened 10 CFR part 21. PIP C-99-05128
00-024170	Crane Liberty Technology Service Bulletin regarding potential for inaccuracies in data taken by Valve Vision. PIP C-00-00421
00-024370	Westinghouse Nuclear Service Advisory Letter concerning the structural analysis for ice baskets. PIP C-00-00275
00-025089	SOER 99-1 Loss of Grid, to analyze recent operating experiences involving loss of grid. PIP C-00-00366
00-024287	Ingersoll Dresser Company vendor information letter regarding charging pump shaft deficiency. PIP C-99-02373

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

Attachment