U.S. NUCLEAR REGULATORY COMMISSION

REGION V

Report Nos. 50-206/93-05, 50-361/93-05, 50-362/93-05

Docket Nos. 50-206, 50-361, 50-362

License Nos. DPR-13, NPF-10, NPF-15

Licensee: Southern California Edison Company
Irvine Operations Center
23 Parker Street
Irvine, California 92718

Facility Name: San Onofre Units 1, 2 and 3

Inspection At: San Onofre, San Clemente, California

Inspection Conducted: February 17 through March 31, 1993

Inspectors: C. W. Caldwell, Senior Resident Inspector
D. L. Solorio, Resident Inspector
T. A. Greene, NRR

Approved By: H. J. Wong [signature] 4/30/93
H. J. Wong, Chief
Reactor Projects Section II
Date Signed

Inspection Summary

Inspection on February 17 through March 31, 1993 (Report Nos. 50-206/93-05, 50-361/93-05, 50-362/93-05)

Areas Inspected: Routine resident inspection of Units 1, 2 and 3 Operations Program including the following areas: operational safety verification, radiological protection, security, evaluation of plant trips and events, monthly surveillance activities, monthly maintenance activities, defueling activities, independent inspection, licensee event report review, and followup of previously identified items. Inspection procedures 30702, 37700, 40500, 60705, 60710, 61726, 62703, 64704, 71707, 83201, 90712, 92700, 92702, 92720, and 93702 were covered.

Safety Issues Management System (SIMS) Items: None

Results:

General Conclusions and Specific Findings:

Strengths

During an audit of the implementation of Licensee Event Report (LER) corrective actions, the Quality Assurance department noted that several commitments made in an LER may not have been implemented and that the LER contained an inaccurate statement (Paragraph 8.d).

The inspector noted that the performance of the Station Emergency Director during an Emergency Preparedness evacuation exercise had improved, based on previous drill observations.

Weaknesses

NRC Inspection Report 50-362/92-16 noted that repeated failures of the reactor coolant baffle bolts were attributed to a failure to formally control the implementation of interim corrective actions. The inspector reviewed several Nuclear Organization division procedures for performing division investigations and noted that several did not provide guidance to implement interim corrective actions (Paragraph 8.b).

A temporary waiver of compliance was needed due to the apparent failure to properly evaluate a Part 21 notification for safety-related battery chargers (Paragraph 8.a).

Significant Safety Matters:

None

Summary of Violations:

None

Open Items Summary:

During this inspection period, six new followup items were opened and two were closed.

DETAILS

1. Persons Contacted

Southern California Edison Company

H. Ray, Senior Vice President, Nuclear
*R. Krieger, Station Manager
J. Reilly, Manager, Nuclear Engineering & Construction
*B. Katz, Manager, Nuclear Oversight
R. Rosenblum, Manager, Nuclear Regulatory Affairs
K. Slagle, Deputy Station Manager
*R. Waldo, Operations Manager
*L. Cash, Maintenance Manager
*D. Breig, Manager, Station Technical
M. Short, Manager, Site Technical Services
M. Wharton, Manager, Nuclear Design Engineering
P. Knapp, Manager, Health Physics
*W. Zintl, Manager, Emergency Preparedness
D. Herbst, Manager, Quality Assurance
*C. Chiu, Manager, Quality Engineering
J. Schramm, Plant Superintendent, Unit 1
V. Fisher, Plant Superintendent, Units 2/3
*R. Joyce, Maintenance Manager, Units 2/3
*M. Herschthal, Manager, Nuclear Systems Engineering
*A. Thiel, Manager, Electric Systems Engineering
*J. Rainsberry, Plant Licensing Manager
*C. Anderson, Supervisor, Emergency Preparedness
*J. Fee, Health Physics Assistant Manager
*G. Hammond, Supervisor, Onsite Nuclear Licensing
*J. Jamerson, Lead Engineer, Onsite Nuclear Licensing
*T. Llorens, Onsite Nuclear Licensing
*D. Axline, Onsite Nuclear Licensing
*L. Mayweather, Onsite Nuclear Licensing
J. Reeder, Manager, Nuclear Training
H. Newton, Manager, Site Support Services
*J. Hirsch, Supervisor, Power Generation
*R. Neal, Supervising Engineer
*M. Tolson, Engineer, Fire Protection
*J. Peattie, Refueling Engineer

City of Riverside

*C. Harris, Site Representative

NRC

*H. Wong, Chief, Reactor Projects Section II, Region V

*Denotes those attending the exit meeting on March 31, 1993.

The inspectors also contacted other licensee employees during the course of the inspection, including operations shift superintendents, control room supervisors, control room operators, QA and QC engineers, compliance engineers, maintenance craftsmen, and health physics engineers and technicians.

2. Plant Status (94702)

Unit 1

The Unit was permanently shutdown on November 30, 1992. The core was off-loaded to the spent fuel pool on March 6, 1993. On March 23, 1993, the inspector participated in a meeting with the licensee and the Office of Nuclear Reactor Regulation (NRR) to discuss Unit 1 decommissioning issues.

Unit 2

The Unit operated at power during the inspection period.

Unit 3

The Unit operated at power during the inspection period.

3. Operational Safety Verification (37700, 64707, 71707)

The inspectors performed several plant tours, verified the operability of selected emergency systems, reviewed the tag out log, and verified proper return to service of affected components. Particular attention was given to inspection of housekeeping, examination for potential fire hazards, fluid leaks, excessive vibration, and verification that maintenance requests had been initiated for equipment in need of maintenance. The inspectors also observed selected activities by licensee radiological protection and security personnel to confirm proper implementation of, and conformance with, facility policies and procedures in these areas.

a. Site Problem Report Program

During a plant walkdown of the Unit 3 cable spreading room, the inspector noted a deficiency tag on a nonsafety-related cable tray. The cable tray was situated near a safety-related cable conduit. The deficiency tag read, "Does not meet length requirement of 39412, cera-blanket should extend to UYXYE6." The inspector noted that a nonconformance report (NCR) had been issued for the condition, but had been invalidated.

Emergency Preparedness personnel provided the inspector with a Site Problem Report (SPR) which had been issued in 1991 to correct field document 39412, "Regulatory Guide 1.75 Separation Barriers." Apparently, field document 39412 initially required the cera-blanket to extend farther along the tray than necessary. Because the cera-blanket installed in the plant was long enough to comply with Regulatory Guide (RG) 1.75, "Physical Independence Of Electrical

Systems," an SPR was then issued to correct field document 39412, listing the new locations the cera-blanket covered.

The inspector walked down the affected tray with the Fire Protection Cognitive Engineer and verified that the application of the cera-blanket met RG 1.75. The inspector noted, however, that the SPR did not contain a justification for the change to field document 39412. Additionally, during the exit meeting the licensee identified that the tag found by the inspector should have been removed once the corrective actions had been taken. The inspector will review the SPR program to determine how evaluations are documented and follow this review as an inspector followup item (50-362/93-05-01).

b. Spent Fuel Pool Emergency Makeup Surveillance - Unit 1

On March 5, 1993, the Operations Department, in conjunction with Nuclear Projects, conducted a test of the Unit 1 spent fuel pool (SFP) emergency makeup system. While reviewing a Quality Assurance (QA) audit of the test, the inspector noted that there were no requirements to periodically inspect the hose used to provide a flow-path from the emergency water source (previously the auxiliary feedwater storage tank) to the SFP. The inspector discussed this issue with the Unit 1 Superintendent.

To address this concern, the licensee added a requirement to inspect the hose biannually to S01-12.9-11, "Miscellaneous Surveillances." Additionally, a step was included in monthly surveillance S01-12.3-42, "Spent Fuel Pit System Safety Related Alignment," to perform S01-12.9-11 if water was found in the hose storage locker. The inspector considered the licensee's corrective actions adequate.

c. Units 2 and 3 Battery Jar Lid Leaks

On March 5, 1993, while replacing Unit 3 safety-related station battery bank 3D1, the licensee observed droplets of dried electrolyte on the side walls of two battery jars. In order to address the question of battery integrity, the licensee performed vendor-recommended pressure-testing on the batteries in the bank. The test consisted of pressurizing the battery jar to one pound per square inch, and monitoring the pressure for 15 seconds. Results of less than 95% of the original pressure were considered a leaking battery. Small leaks were detected at the battery jar lid interfaces on five cells of battery bank 3D1. It was noted that the battery jar lids had been secured to the battery jars with plastic cement. The leaking jars were replaced and 3D1 was subsequently put into service.

On March 24, 1993, the licensee pressure-tested both the Unit 3 safety-related battery bank 3D2, and the temporary safety-related battery bank B00X. Several jars were found to be leaking in both battery banks. Battery bank B00X was the temporary supply for the

2D1 battery loads while battery bank 2D1 was being replaced with a new battery bank. The replacement was necessary in order to remove batteries with cracked jar lids (see NRC Inspection Report 50-362/92-23). All of the new battery cells for 2D1 were also tested and leaking batteries are to be replaced before placing 2D1 back in service.

The 3D2 battery bank was in service at the time of the pressure test. The licensee performed an evaluation which addressed postulated level losses for the batteries during a seismic event. The evaluation results indicated that the batteries with leaking jar lids would remain operable during a seismic event. Additionally, Exide, the battery vendor, provided the licensee with a letter confirming that the batteries would continue to function electrically during and after a seismic event. However, the inspector noted that the letter did not contain a justification for the assessment. The licensee had sent one of the leaking cells to the vendor to determine the cause of the seal problem between the jar lid and the jar.

The licensee plans to conduct a seismic test of the two worst leaking battery jars, in accordance with IEEE 535-1986, "Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations." The results of the seismic test and the resultant corrective actions will be reviewed by the inspector as inspector followup item (50-361/93-05-02).

No violations or deviations were identified.

4. Evaluation of Plant Trips and Events (93702)

a. Core Protection Calculator Abnormalities - Units 2 and 3

On February 25, 1993, the licensee observed differences between the core axial power distribution calculated by their design code (CECOR), the core operating limits supervisory system (COLSS), and Core Protection Calculators (CPCs) for both Units. CECOR and COLSS use the in-core neutron flux detectors, while the CPCs (input to the reactor protection system) use the ex-core detectors to calculate core axial power. CECOR and COLSS were observed to produce a saddle-shaped axial power distribution, while the CPCs produced a cosine-shaped curve. The licensee expected all three systems to calculate a saddle-shaped axial power curve. The axial power distribution differences were observed during a data review performed by the licensee's Nuclear Fuels Analysis (NFA) group, which was being done as part of the licensee's expanded ability to review core performance. The review identified that the calculated power at the top of the core as predicted by the CPCs was less than that predicted by COLSS or CECOR. As a result, the licensee concluded that the departure from nucleate boiling ratios (DNBRs) calculated by the CPCs might have been non-conservative since the core power used in the DNBR calculation might be less than actual core power. Upon reaching this conclusion, the licensee notified the NRC Region V office.

The licensee preliminarily considered the cause of the difference in the axial power distribution for the CPCs to be attributed to the methods used to develop the Shape Annealing Matrix (SAM). The CPCs used the SAM, a 3 x 3 matrix of addressable constants, which was created during startup testing to develop an axial power distribution. The SAM values were determined by tests performed during reactor startup which recorded the ex-core detector's response to a power increase. An analysis was performed on this matrix to determine uncertainty factors which were applied to the DNBR calculation (performed by the CPCs) to ensure conservatism. In any given cycle, once the SAM was constructed in conjunction with DNBR penalty factors, the CPC should have generated a DNBR that conservatively modeled the actual axial power distribution under all conditions.

As immediate compensatory actions, the licensee determined that a penalty factor (corresponding to approximately 15% power reduction) should be imposed into the CPC to account for any possible non-conservatism in calculated DNBRs. On February 25, 1993, a 15% power was reduction was initiated for both Units.

Additional corrective actions included inducing a power swing to develop a new SAM, and a re-determination of penalty factors for the CPCs of each Unit. Using the new SAM, the CPC calculated axial power distribution was compared to the axial power distribution calculated by COLSS and CECOR. The new CPC calculated axial power distribution showed the expected shape and appropriate conservatisms. The new SAM was also analyzed to ensure it would continue to correctly model the axial power distribution through the end of each Unit's fuel cycle. Additionally, the licensee initiated weekly monitoring of the axial power distribution generated by the CPCs. On March 4 and March 6, 1993, power was increased to 100% after incorporating the new data into the CPCs for Units 3 and 2 respectively.

The safety significance of the effects of the CPC cosine-shaped axial power distribution on the DNBR will be assessed as part of the licensee's long-term corrective actions. These actions will include determination of whether or not the plant was bounded by the uncertainty analysis, and the cause of the unexpected performance. These evaluations will be performed by Combustion Engineering and NFA, and will be documented in a root cause report, scheduled to be completed in May 1993. The licensee also committed to issue an LER, once the root cause report is issued. The inspector will review startup core physics data used to develop the SAMs, penalty factors, and the licensee's evaluations as inspector followup item (50-361/93-05-03).

b. Latching Relay Failures - Units 2 and 3

On March 9, 1993, operators noted that the Unit 2 spent fuel pool (SFP) pump P009 failed to continue running when started. Prior to that, on February 28, 1993, the auxiliary feedwater discharge to steam generator 2-E088's electro-hydraulic isolation valve, 2HV4714, failed to open during performance of a quarterly valve test. Additionally, the inspector noted that 2HV4714 had failed three previous surveillance tests, dating back to March 1992. The licensee was not able to reproduce the failure mechanism for the first two valve surveillance failures. After the second failure, the relay was replaced. The third and fourth failures of 2HV4714, and the failure of SFP pump P009, were attributed to the failure of relays to latch. The latching relays were manufactured by Square D. The following NCRs were written to document the relay failures, and to provide corrective actions: 93020069, 92090003, 92050025 and 92030118. The relays for the latest two failures (2HV4714 and P009) were replaced with improved models and the equipment was subsequently returned to service.

Following the SFP pump relay failure, the electrical group within Station Technical (STEC) identified all locations of the suspect relays. A total of 46 relays were identified. Many of the relays were in identical components in both Units. An evaluation assessing the consequences of the failure of the relays to latch was performed. The inspector preliminarily reviewed the evaluation for several components and that review continues. It was noted that STEC had not included documentation of its evaluation in the NCR prior to being questioned by the inspector. Long-term corrective actions include developing a schedule for the replacement of all suspect relays.

The site Independent Safety Engineering Group (ISEG) will issue a root cause report for the relay failures. The inspector will review the report and the licensee's root cause evaluations for the previous failures of 2HV4714 as inspector followup item (50-361/93-05-04).

No violations or deviations were identified.

5. Bi-Monthly Surveillance Activities (61726)

During this report period, the inspectors observed or conducted inspection of the following surveillance activity:

a. Observation of Routine Surveillance Activities (Unit 2)

S023-V-3.5.9 "High Pressure Safety Injection and Low Pressure Safety Injection Discharge Stop-Check Valve."

No violations or deviations were identified.

6. Monthly Maintenance Activities (62703)

During this report period, the inspectors observed or conducted inspection of the following maintenance activities:

a. Observation of Routine Maintenance Activities (Unit 2)

MO93031405000	"Disconnect Battery Bank 2B007 and Connect Battery Bank SA1806EB00X to Battery Charger S21806EB001. SA1806EB00X Becomes a 1E Battery. Support for Temporary Facility Modification 2-92-PKA-002."
MO93030768000	"Station Technical Requires MOVATS Testing (Current and Switches) After Packing Adjustment on Boiler and Condenser Maintenance Order 92111463."
MO92111463001	"Emergency Boration Valve Packing Leak. This MO for Adjustment Only."

b. Observation of Routine Maintenance Activities (Unit 3)

MO93021022000	"Disconnect Battery Bank 3B007 and Connect Battery Bank SA1806EB00X to Battery Charger S31806EB001. SA1806EB00X Becomes A 1E Battery. Support MO for Temporary Facility Modification 3-92-PKA-004."

No violations or deviations were identified.

7. Plant Modification and Refueling Activities (60705, 60710)

On March 2, 1993, the licensee initiated Unit 1 core offload as part of permanently shutting down the Unit. On March 6, 1993, the core offload was completed. A total of 157 fuel assemblies were removed from the reactor vessel and placed in the Unit 1 spent fuel pool. The inspector observed the defueling operations from both the control room and the reactor cavity and noted that the activities were conducted in a professional and safe manner.

No violations or deviations were identified.

8. Independent Inspection (71707, 82301, 90712, 92720)

a. Temporary Waiver of Compliance For Unit 2 Battery Charger Current Output Below Technical Specification Limits (71707)

On February 25, 1993, the licensee identified that three of eight battery chargers (2D1, 2D2 and 3D4) in Units 2 and 3 may not meet Technical Specification (TS) 3.8.2, "Electrical Power Systems - D.C. Sources," requirements for current output of 300 amps. The licensee identified this condition shortly after replacement of several battery charger circuit cards for the Unit 2 battery 2D1 charger, which had developed indications of a circuit card failure.

On February 25, 1993, at 7:40 p.m., the licensee tested the output of the 2D1 battery charger, and found the output to be 270 amps. As a result of 2D1 output current failing to meet the TS requirement of 300 amps, and because the other two chargers were in a similar configuration without the chargers' output having been verified, the licensee concluded that the three chargers were inoperable. With two chargers inoperable on Unit 2, TS 3.0.3 was entered.

The licensee had discussions with the NRC offices of Region V, and NRR, orally requesting a waiver of compliance to allow restoration of the 2D1 charger to an operable status. At 8:30 p.m., Region V, with concurrence from NRR, granted a waiver to extend the time to shutdown the reactor for four hours. At 9:43 p.m. the 2D1 charger was declared operable. Subsequent testing of the 2D2 charger found its output to be 287 amps, and the 3D2 charger output was found to be 306 amps. All three charger outputs were adjusted to above 300 amps to provide an adequate safety margin.

STEC and the Nuclear Engineering Design Organization performed calculations to determine the minimum required current to be supplied by the chargers to perform their intended safety function. The minimum current was found to be 200 amps, which was documented in NCRs 93020062, 93020063, and 93020064. Since all three charger output currents were found to be greater than 200 amps, the inspector considered the safety significance to be low.

The inspector noted that in 1989 the licensee had received a Part 21 notification for the chargers. The notification was consequently dispositioned by STEC. On March 29, 1993, an LER was submitted stating that the initial Part 21 evaluation failed to identify the potential of the charger output current to be less than the TS requirement. The LER will be reviewed by the inspector during routine inspection activities to determine whether licensee actions related to the 1989 Part 21 report were adequate.

b. Review of Division Investigation Report Requirements (40500, 92720)

The inspector reviewed the Division Investigation Report (DIR) procedural requirements for the following divisions: Nuclear Oversight (NOD), STEC, Operations (OPS), Maintenance, Health Physics (HP), and Chemistry. The inspector noted that the various divisions' procedures did not include provisions for handling the implementation of interim corrective actions. The inspector also noted that time requirements for issuing reports, and extensions for reports which could not be completed within the recommended time frame, varied among the divisions.

In NRC Inspection Report 50-362/92-16, the inspector concluded that failures of the reactor coolant pump (RCP) baffle bolts were attributed to the failure of Safety Engineering (SE) to formally implement interim corrective actions. On March 3, 1993, SE issued a root cause human performance evaluation, RCE-93-003, which arrived at the same conclusion. The inspector noted that it appeared that lessons learned from the failure to implement interim corrective actions (where warranted) for the RCP baffle bolts were not used to assess the adequacy of DIR procedures for groups other than SE. The licensee indicated that they had initiated efforts to provide more formality in their DIR procedures throughout the nuclear organization. The inspector will review the implementation of enhancements to the DIR procedures as inspector followup item (50-361/93-05-05).

c. <u>Emergency Preparedness Drill Observations - Unit 2</u> (82301)

On March 17, 1993, the inspector observed the licensee's emergency preparedness (EP) exercises for Unit 2 from the Technical Support Center. The purpose of the drill was to practice a site evacuation.

The inspector noted that performance of the Station Emergency Director had improved, based on previous drill observations. However, the inspector identified that a few observers, not acting as coaches, were helping players with emergency activities. EP managers informed the inspector that observers had been instructed not to participate in activities during the drill, and that they would reemphasize that directive for future drills. The EP manager indicated that the drill was the first one of the year, with many new players and observers. He expected that over the next few drills, prior to the annual exercise, there would be no participation from observers. The inspector considered the licensee's actions adequate.

d. <u>Licensee Event Report (LER) Revision Due to Inaccurate Information</u> (90712)

On February 19, 1993, the licensee contacted the inspector to acknowledge that several commitments made in LER 2-91-007, Revision 0, may not have been implemented. These deficiencies were identified by QA during an audit of the corrective actions as stated in the LER. This LER discussed the shutdown of Unit 2 due to the loss of controlled bleedoff flow (seal flow) to a reactor coolant pump. The loss of controlled bleedoff (CBO) flow was due to the failure of baffle bolts within the reactor coolant pump. The LER discussed the licensee's corrective actions.

Two deficiencies were found by the licensee's QA organization. One deficiency was that the use of an anti-embedment technique, as stated in the LER, was not documented in the maintenance orders for the Unit 2 reactor coolant pumps P002 and P003 for the baffle repairs. An assessment addressing the consequences of not using the anti-embedment technique to reassemble the baffles was performed by

SE. As of the end of the inspection period, the evaluation was being reviewed by the inspector and NRR. Additionally, the licensee identified that one statement was inaccurate with regard to the evaluation of the loss of controlled bleedoff (CBO) flow. In the LER, the licensee stated that the seals would maintain their integrity without CBO flow, per vendor tests. QA identified that the test actually conducted was a loss of component cooling water flow which cools the CBO flow, and not a loss of CBO flow. During the exit meeting the licensee stated that they planned to revise the LER.

The inspector also questioned the licensee as to what actions they planned take to assure themselves of the accuracy of future submittals. The licensee indicated that a DIR had been initiated to address these concerns, and that corrective actions would be implemented where warranted, based on the results of that investigation. The inspector will review the the revised LER and the DIR as an unresolved item (50-361/93-05-06).

No violations or deviations were identified.

9. Review of Licensee Event Reports (90712, 92700)

a. Through direct observations, discussion with licensee personnel, or review of the records, the following Licensee Event Reports (LERs) were closed:

Unit 2

93-01 Rev 0, "Main Steam Safety Valve Inoperable"

92-09 Rev 1, "Mis-alignment of Unit 2 Salt Water Cooling Pump P112 Emergency Seal Water Supply Isolation Valve"

The inspector verified that the licensee's corrective actions were as stated in the LER. The inspector also found that almost all of the planned corrective actions stated in the LER had been completed. In particular, the licensee had reviewed all station technical procedures to identify those needing revision to comply with the requirements of licensee procedure SO123-0-20, "Use of Procedures," regarding equipment status controls for the manipulation of plant equipment. The review led to the upgrade of 56 procedures. A similar review of chemistry procedures (Regulatory Commitment Tracking System No. 9207034) was scheduled for completion in July 1993.

b. LER Review To Determine Compliance With Requirements of 50.73

The inspector reviewed the following LERs to determine if they were written in compliance with the requirements of 10 CFR 50.73:

LER #	DATE
50-206/92-003	01/18/93
50-206/92-002	08/11/92
50-206/92-001	07/23/92
50-206/91-021	12/06/91
50-206/91-020	11/25/91
50-206/90-018, REV 0	10/22/90
50-206/90-018, REV 1	12/31/92
50-206/91-019	12/02/91
50-206/91-018	11/20/91
50-206/91-017	11/18/91
50-206/91/016	09/30/91
50-206/91-015	09/16/91
50-206/91-014, REV 0	09/09/91
50-206/91-014, REV 1	03/30/92
50-206/91-013, REV 0	07/31/91
50-206/91-013, REV 1	12/06/91
50-206/91-013, REV 2	03/05/92
50-361/92-012	08/31/92
50-361/92-011	10/29/92
50-361/92-010	06/03/92
50-361/92-009, REV 1	11/19/92
50-361/92-008	05/22/92
50-362/92-009, REV 0	07/14/92

All, but 4 of the Licensee Event Reports (LERs), were submitted within a month after the discovery of the event. For the 4 exceptions, 2 were submitted in a month plus one day and 2 in a month plus two days.

All the LER narratives describing the event contained enough details to understand the event and to assess its safety significance. A couple of LERs did not contain a separate abstract. However, these LERs contained relatively brief narratives and so the narrative also acted as the abstract. In general, the LERs were clear and specific so that a reader could understand the complete event as required by 10 CFR 50.73.

No violations or deviations were identified.

10. Exit Meeting

On March 31, 1993, an exit meeting was conducted with the licensee representatives identified in Paragraph 1. The inspectors summarized the inspection scope and findings as described in the Results section of this report.

The licensee acknowledged the inspection findings and noted that appropriate corrective actions would be implemented where warranted. The licensee did not identify as proprietary any of the information provided to or reviewed by the inspectors during this inspection.