UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

October 7, 2003

Joseph E. Venable
Vice President Operations
Waterford 3
Entergy Operations, Inc.
17265 River Road
Killona, Louisiana 70066-0751

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000382/2003006

Dear Mr. Venable:

On September 20, 2003, the NRC completed an inspection at your Waterford Steam Electric Station, Unit 3. The enclosed report documents the inspection findings which were discussed on September 22, 2003, with Mr. J. R. Douet General Manager, Plant Operations, and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has identified four issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that violations are associated with these issues. These violations are being treated as noncited violations (NCVs), consistent with Section VI.A of the Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the violations or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Waterford Steam Electric Station, Unit 3 facility.

Since the terrorist attacks on September 11, 2001, NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during Calendar Year 2002 and the remaining inspection activities for Waterford 3 were completed in

June 2003. The NRC will continue to monitor overall safeguards and security controls at Waterford 3.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure(s), and your response will be made 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/reading-rm/adams.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

William B. Jones, Chief
Project Branch E
Division of Reactor Projects

Docket: 50-382
License: NPF-38

Enclosure(s):
NRC Inspection Report
050000382/2003006

w/attachment: Supplemental Information

cc w/enclosure(s):
Senior Vice President and
Chief Operating Officer
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286-1995

Vice President, Operations Support
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286-1995

Wise, Carter, Child & Caraway
P.O. Box 651
Jackson, Mississippi 39205

General Manager, Plant Operations
Waterford 3 SES
Entergy Operations, Inc.
17265 River Road
Killona, Louisiana 70066-0751

Manager - Licensing Manager
Waterford 3 SES
Entergy Operations, Inc.
17265 River Road
Killona, Louisiana 70066-0751

Chairman
Louisiana Public Service Commission
P.O. Box 91154
Baton Rouge, Louisiana 70821-9154

Director, Nuclear Safety &
Regulatory Affairs
Waterford 3 SES
Entergy Operations, Inc.
17265 River Road
Killona, Louisiana 70066-0751

Michael E. Henry, State Liaison Officer
Department of Environmental Quality
Permits Division
P.O. Box 4313
Baton Rouge, Louisiana 70821-4313

Parish President
St. Charles Parish
P.O. Box 302
Hahnville, Louisiana 70057

Winston & Strawn
1400 L Street, N.W.
Washington, D.C. 20005-3502

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-382
License No.:	NPF-38
Report No.:	05000382/2003006
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, Louisiana
Dates:	June 22 through September 20, 2003
Inspectors:	M. C. Hay, Senior Resident Inspector G. F. Larkin, Resident Inspector D. L. Stearns, Project Engineer
Approved By:	W. B. Jones, Chief, Project Branch E
ATTACHMENTS:	Supplemental Information

SUMMARY OF FINDINGS

IR05000382/2003006; 06/22/2003-09/20/2003; Waterford Steam Electric Station, Unit 3; Adverse Weather Protection, Operability Evaluations, Surveillance Testing, Other Activities.

The report covered a 3 month period of inspection by resident inspectors. The inspection identified four Green findings. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the Significance Determination Process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified a noncited violation of 10 CFR 50.63 for the failure to maintain a station blackout coping analysis that adequately encompassed plant conditions prescribed by the station blackout recovery emergency operating procedure. This resulted in the failure to evaluate for a reactor coolant system cooldown to a 400°F cold leg temperature, as prescribed by procedure, since the coping analysis assumed the reactor coolant system cold leg would be maintained at 545°F during station blackout conditions.

This finding is greater than minor because it affected the reactor safety mitigating system cornerstone objective to ensure the capability of systems that respond to initiating events to prevent undesirable consequences. The significance of the finding was determined to be of very low safety significance because the deficiency was confirmed not to result in loss of the capability to cope with a station blackout per Generic Letter 91-18 guidance (Section 1RO1).

Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to translate into specifications, procedures, and instructions design criteria for the diesel generator air start system. This resulted in the failure to maintain each diesel generator air receiver capable of starting the diesel engine five times.

This finding is greater than minor because it affected the reactor safety mitigating system cornerstone objective due to the degradation of the design basis capability of the starting air system. The significance of the finding was determined to be of very low safety significance because the deficiency did not represent an actual loss of the starting air system safety function per Generic Letter 91-18 guidance. Additionally, surveillance testing has demonstrated the capability of each diesel generator to start within the required 10 seconds (Section 1R15).

<u>Green.</u> A self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" was identified for the failure to maintain design control of an overcurrent relay. This resulted in the failure to maintain normally open contact gap distances in accordance with vendor specifications. This design control deficiency was determined to be the most probable cause for loss of power to a safety related bus on July 24 and July 27, 2003.

The finding is greater than minor because it affected the reactor safety mitigating system corner stone and if left uncorrected the finding could become a more significant safety concern. The significance of the finding was determined to be of very low safety significance because the deficiency did not result in the loss of safety-related equipment for greater than its Technical Specification allowed outage time (Section 1R22).

Cornerstone: Barrier Integrity

- <u>Green</u>. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to maintain design control of the switchgear ventilation system. This resulted in a potential common mode failure of safety related Dampers SVS-101 and SVS-102, due to loss of the nonsafety-related instrument air system.

  The finding is greater than minor because if left uncorrected the finding could become a more significant safety concern. The significance of the finding, which is under the Barrier Integrity cornerstone, was determined to be of very low safety significance because the finding only represented a degradation of the radiological barrier function provided for the control room (Section 4OA5).

B. <u>Licensee-Identified Violations</u>

None.

REPORT DETAILS

Summary of Plant Status: The plant began the period at full rated thermal power and operated at full power for the entire period except for a planned power reduction and coast down. On September 5, 2003, power was reduced to approximately 88 percent power for planned surveillance testing of the high-pressure turbine. On September 18, 2003, the plant began to coast down from 100 percent power. The plant was at 99 percent power on September 20.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

With a potential for hurricanes present in the vicinity of the facility during the inspection period, the inspectors reviewed the licencee's ability to cope with a station blackout. On September 2, 2003, the inspectors completed area walkdowns to verify that appropriate seasonal preparations were made to protect the station's switchyard and emergency diesel generators. These areas were selected based on their function to supply normal and emergency electrical power during adverse weather conditions. The inspectors also reviewed the licensee's station blackout recovery procedure, coping analysis, and the Final Safety Analysis Report.

b. Findings

Introduction. A Green noncited violation of 10 CFR 50.63 was identified for the failure to maintain a station blackout coping analysis that adequately encompassed plant conditions prescribed by the station blackout recovery emergency operating procedure.

Description. A review of the station blackout coping analysis contained in Engineering Document EC-E89-016, "Station Blackout Response for Waterford," Revision 2, was performed. The inspectors noted that the coping analysis assumed that the reactor coolant system cold leg temperature would be maintained at approximately 545°F during the duration of the station blackout. The inspectors reviewed Emergency Operating Procedure OP-902-005, "Station Blackout Recovery," Revision 11, and noted that actions to cool the reactor coolant system to a cold leg temperature of 400°F, to maintain reactor system subcooling greater than 28°F, was prescribed. The inspectors determined that the cooldown to 400°F adversely affected the analysis in Engineering Document EC-E89-016 the following ways: 1) the quantity of condensate storage pool water required for decay heat removal was underestimated; and 2) the time to uncover the core due to the effects of the cooldown was underestimated.

The inspectors raised these concerns to the responsible engineering and operations personnel. The licensee stated that changes were made to the station blackout recovery emergency operating procedure that resulted in the potential for plant conditions being outside those evaluated in the station blackout coping analysis. The licensee performed an evaluation and determined that reactor coolant system inventory and heat removal inventory remained within the acceptance limits. Therefore, the station blackout recovery methodology remained acceptable.

Analysis. The deficiency associated with this finding was the failure to maintain an appropriate station blackout coping analysis resulting in the potential for plant conditions to be unanalyzed. This finding is greater than minor because it affected the reactor safety mitigating system cornerstone objective to ensure the capability of systems that respond to initiating events to prevent undesirable consequences. The significance of the finding was determined to be of very low safety significance because the deficiency was confirmed not to result in loss of the capability to cope with a station blackout per Generic Letter 91-18 guidance.

Enforcement. 10 CFR 50.63 requires that the capability for coping with a station blackout be determined by an appropriate coping analysis. The licensee's station blackout analysis failed to evaluate for a reactor coolant system cooldown to 400°F, assuming temperature would remain at 545°F. The failure to maintain an appropriate station blackout coping analysis is being considered a violation of 10 CFR 50.63. Because the failure to maintain an appropriate station blackout coping analysis was of very low safety significance and has been entered into the licensee's corrective action program as Condition Report CR-WF3-2003-2452, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000382/2003006-01, Inadequate Station Blackout Coping Analysis.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial System Walkdowns

The inspectors performed the following three partial system equipment alignment inspections during this inspection period:

- On July 15, 2003, the inspectors walked down the accessible portions of the mechanical and electrical components of the switchgear ventilation system Train A. This walkdown was completed while the hydramotor for switchgear ventilation system Valve SVS-201B and air handling unit Damper AH-30 was removed for maintenance.

- On August 4, 2003, the inspectors completed a partial equipment alignment inspection of the 125V DC electrical distribution system Train AB. System configuration was assessed using Operating Procedure OP-006-003, "125 Volt DC Electrical Distribution," Revision 9, as well as applicable chapters of the Final Safety Analysis Report. A walkdown of accessible portions of the system was performed to assess material condition and housekeeping issues that could adversely affect system operability.

- On September 8, 2003, the inspectors completed a partial equipment alignment inspection of the control room emergency filtration system Train A while Train B was inoperable. System configuration was assessed using Operating Procedure OP-003-014, "Control Room Heating and Ventilation (HVC)," Revision 7, as well as applicable chapters of the Final Safety Analysis Report. A walkdown of accessible portions of the system was performed to assess material condition, such as system leaks and housekeeping issues, that could adversely affect system operability.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors conducted six inspections to assess whether the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capabilities, and maintained passive fire protection features in good material condition.

The following areas were inspected:

- Fire Zone RAB 2 on July 15, 2003
- Fire Zone RAB 8A, 8B, and 8C on July 16, 2003
- Fire Zone RAB 15, 16, 17, 18, 19, 20, and 21 on August 27, 2003
- Fire Zone RAB 1, 8B, and CTB on September 8, 2003
- Fire Zone RAB 5, 6 and 7 on September 14, 2003
- Fire Zone RAB 33, 35, 36 and 37 on September 16, 2003

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors completed an inspection of internal and external flood protection features associated with the nuclear plant island structure on September 17, 2003. The inspection included a review of the Updated Final Safety Analysis Report, selected design calculations, condition reports, and a walkdown of select flood protection features. The inspectors toured the fuel handling building lower level, cooling tower areas Trains A and B, and the nuclear plant island internal and external flood wall penetrations.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed the ability of the spent fuel pool cooling system to remove the decay heat of the spent fuel during refueling outages involving a full core offload. The inspection consisted of reviewing the following documentation:

- HI-961586, “Thermal Hydraulic Analysis of the Waterford-3 Spent Fuel Pool”
- Final Safety Analysis Report, Section 9.1.3, “Fuel Pool System”
- Administrative Procedure PE-001-015, “Generic Letter 89-13 Heat Exchanger Test Basis,” Revision 3
- ASME Standard OM-S/G-1997, Part 21, “Inservice Performance Testing of Heat Exchangers in Light-Water Reactor Power Plants”

The inspectors noted that performance testing was not performed nor required to be performed for these heat exchangers and determined to complete this inspection during the next refueling outage after verifying adequate system performance.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On September 9, 2003, the inspectors observed two licensed operator simulator training exercises. During the first exercise, the inspectors evaluated the operators’ ability to recognize, diagnose, and respond to a dropped control element assembly followed by a loss of turbine building cooling water and a main feedline break inside containment. The second simulator exercise involved a steam generator tube leak developing into a tube rupture coupled with a failure of two control element assemblies to insert and the failure of High-Pressure Safety Injection Pump B to start on a safety injection actuation signal. The inspectors observed and evaluated the following areas:

- Understanding and interpreting annunciator and alarm signals
- Diagnosing events and conditions based on signals or readings
- Understanding plant systems
- Use and adherence of Technical Specifications
- Crew communications including command and control
- The crew's and evaluator's critiques

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

During the inspection period, the inspectors reviewed licensee implementation of the Maintenance Rule. The inspectors considered the characterization, safety significance, performance criteria, and the appropriateness of goals and corrective actions. The inspectors assessed the licensee's implementation of the Maintenance Rule to the requirements outlined in 10 CFR 50.65, and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2. The inspectors reviewed the following two systems that displayed performance problems:

- Main turbine generator seal oil system
- Instrument air system

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors reviewed risk assessments for planned or emergent maintenance activities to determine if the licensee met the requirements of 10 CFR 50.65(a)(4) for assessing and managing any increase in risk from these activities. The following four risk evaluations were reviewed:

- On June 16, 2003, the diverse reactor trip system and the diverse emergency feedwater actuation system were declared inoperable

- On June 27, 2003, during the loss of safety-related Electrical Bus 311B while performing emergent replacement of faulted Overcurrent Relay SSDEREL31B-10B

- On July 4, 2003, during emergent repairs on Main Feedwater System Isolation Valve Number 2

- From August 2-6, 2003, during emergent repairs on Battery Charger AB1

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the technical adequacy of five operability evaluations to verify that they were sufficient to justify continued operation of a system or component. The inspectors considered that, although equipment was potentially degraded, the operability evaluation provided adequate justification that the equipment could still meet its Technical Specification, Updated Final Safety Analysis Report, and design-bases requirements and that the potential risk increase contributed by the degraded equipment was thoroughly evaluated. The following evaluations were reviewed:

- Operability evaluation addressing temperature of component cooling water from the outlet of the shutdown heat exchanger exceeding the design temperature of the system (Condition Report 2003-02557)

- Operability evaluation addressing the failure to account for process-dependent effects for determining steam generator moisture carryover potentially impacting reactor thermal power calculations (Condition Report 2003-02423)

- Operability evaluation addressing a lube oil leak from Train B emergency diesel generator turbocharger lube oil strainer (Condition Report 2003-02308)

- Operability evaluation addressing pipe voids affecting Emergency Core Cooling System Train A (Condition Report 2003-02427)

- Operability evaluation addressing air leakage affecting Emergency Diesel Generator Starting Air System Train B (Condition Report 2003-01942)

b. Findings

Introduction. A Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified that applied to both Trains A and B emergency diesel generating air start systems. Specifically, the inspectors identified that the design basis for each diesel generator starting air system was not properly translated into specifications, procedures, and instructions. This resulted in the failure to maintain each diesel generator air receiver capable of starting the diesel engine five times.

Discussion. The design basis for the air start system, as stated in the Final Safety Analysis Report, Section 9.5.6.3, “Safety Evaluation,” Revision 11-A, is that each diesel generator is equipped with two air receivers, and that each receiver is sized to store enough air to crank and start the engine five times, based on an initial nominal air receiver pressure of 245 to 255 psig, without the use of the air compressors when starting manually. The inspectors noted that there was no sizing calculations performed for the receivers and that the licensee utilized preoperational test results which demonstrated that the design basis was satisfied at the time the plant was licensed. The preoperational testing results are as follows:

Diesel Generator	Air Receiver	Air Receiver Pressure	Number of Starts
Train A	3A-1	250 psig	6
Train A	3A-2	250 psig	4
Train A	3A-2	255 psig	5
Train B	3B-1	245 psig	5
Train B	3B-2	250 psig	5

The inspectors noted that the nominal air receiver pressure for all the receivers was being maintained in a pressure band between approximately 239 psig and 251 psig. The inspectors noted that the nominal operating pressure range for the receivers went below the test pressures that were used to demonstrate each receiver could support five diesel starts. Upon questioning, the licensee stated it was their understanding that the five-start capability for each receiver was a sizing criteria used to purchase the receivers and that maintaining the capability for five starts per receiver was not required.

The inspectors reviewed applicable documentation that described the initial licensing basis of the diesel generator air start system. Standard Review Plan (NUREG-0800) Section 9.5.6, states that General Design Criterion 17 as related to the capability of the diesel engine air starting system to meet independence and redundancy criteria is satisfied when, as a minimum, the air starting system is capable of cranking a cold diesel engine five times without recharging the receivers. NRC Safety Evaluation Report (NUREG-0787) related to the operation of Waterford Steam Electric Station, Unit 3, Section 9.5.6, states that each emergency diesel generator has an independent and redundant air starting system consisting of two separate full capacity air starting subsystems each with sufficient air capacity to provide a minimum of five consecutive cold engine starts. This meets the requirements of General Design Criterion 17.

The inspectors determined that the five-start capability for each receiver was an initial design requirement and was required to be maintained in order to satisfy 10 CFR Part 50, Appendix A, General Design Criterion 17. The inspectors determined that the failure to translate into specifications, procedures, and instructions the design basis for each diesel generator air start system was a violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.”

Analysis. The deficiency associated with this finding was the failure to maintain design control of the diesel generator starting air system resulting in the failure to maintain the capability of each air receiver to provide for five diesel starts. This finding was greater than minor because it affected the reactor safety mitigating system cornerstone objective due to the degradation of the design-basis capability of the starting air system.

The significance of the finding was determined to be of very low safety significance because the deficiency did not represent an actual loss of the starting air system safety function per Generic Letter 91-18 guidance. Additionally, surveillance testing has demonstrated the capability of each diesel generator to start within the required 10 seconds.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that measures be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The failure to translate into specifications, procedures, and instructions the design basis for each diesel generator air start system to maintain the capability of each air receiver to provide for five diesel starts is a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Because the failure to maintain design control of the air start system was of very low safety significance and has been entered into the licensee's corrective action program as Condition Report CR-WF3-2003-01942, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000382/2003006-02, Inadequate Design Control of the Diesel Generator Starting Air System.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors performed a review of operator workarounds. This review evaluated the cumulative affects of current operator workarounds to assess the overall impact affecting the operators' ability to respond in a correct and timely manner to plant transients and accidents.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed a permanent plant modification that reduced the number of bolts used to secure the containment side fuel transfer tube flange. The inspectors verified that the modification did not adversely affect the fuel transfer tube design requirements specified in the Final Safety Analysis Report. The modification was implemented to decrease the amount of time to secure the fuel transfer tube flange resulting in lower personnel radiation exposure. The inspectors reviewed the following documentation during this inspection activity:

- Engineering Calculation EC-M94-003, "Fuel Transfer Tube Flange Bolt Reduction," Revision 1

- Waterford 3 Final Safety Analysis Report

- Drawing G-175, Sheet 3, "Reactor Containment Building Piping Penetrations," Revision 15

- Underwater Construction Corporation Dive Logs for July 3, 4 8, 9 and 10, 2003

- Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," Revision 0

- Condition Report 2003-01846

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed postmaintenance tests to verify system operability and functional capabilities. The inspectors considered whether testing met design and licensing bases, Technical Specifications, and licensee procedural requirements. The inspectors reviewed the testing results for the following six activities:

- Main Feedwater Isolation Valve MVAAA184B following emergent repairs on July 4, 2003, involving replacement of a pneumatic regulator. The inspectors reviewed Work Order 26372.

- Switchgear Ventilation System Valve SVS-201B Hydramotor following planned maintenance activities to correct a slight oil leak on July 15, 2003. The inspectors reviewed Work Order 25382.

- Control room emergency filtration unit Train A Recirculation Damper HCV-213A after the damper drifted open without operator action on June 26, 2003. The inspectors reviewed Work Order 26143.

- Emergency feedwater Pump B following a planned maintenance outage on July 23, 2003. The inspectors reviewed Work Orders 20341, 50232671, and 50010416.

- Emergency diesel generator Train B turbine lube oil filter following leak repairs on August 18, 2003. The inspectors reviewed Work Order 7182.

- Emergency diesel generator Train A following a planned maintenance outage on September 2, 2003. The inspectors reviewed Work Orders 25272 and 25273.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed or reviewed the following six surveillance tests to ensure the systems were capable of performing their safety function and to assess their operational readiness. Specifically, the inspectors considered whether the following surveillance tests met Technical Specifications, the Updated Final Safety Analysis Report, and licensee procedural requirements:

- Surveillance Procedure OP-903-046, “Emergency Feedwater Pump Operability Check,” Revision 15, performed on July 7, 2003. This surveillance tested the functional capability of Emergency Feedwater Pump B.

- Surveillance Procedure OP-903-068, “Emergency Diesel Generator Operability and Subgroup Relay Verification,” Revision 12, performed on July 8, 2003. This surveillance tested the functional capability of emergency diesel generator Train A to start within 10 seconds upon demand and portions of the engineered safety features actuation system subgroup relay that are testable during power operations.

- Surveillance Procedure OP-903-030, “Safety Injection Pump Operability Verification,” Revision 13, performed on July 23, 2003. This surveillance tested the functional capability of the high pressure safety injection system Pump B.

- Electrical Maintenance Procedure ME-005-072, “General Electric Overcurrent Relay Model 12IAC66T Calibration,” Revision 10, performed on July 24, 2003. This surveillance verified the functional capability of overcurrent relays to provide overcurrent protection for safety-related switchgear.

- Surveillance Procedure OP-903-063, “Chilled Water Pump Operability Verification,” Revision 11, performed on August 12, 2003. This surveillance tested the functional capability of essential chillwater Pump B.

- Surveillance Procedure PE-005-005, “Controlled Ventilation Area System Surveillance,” Revision 5, performed on September 11, 2003. This surveillance tested the functional capability of the controlled ventilation area system Train B.

b. Findings

Introduction. A Green self-revealing, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, was identified for the failure to maintain design control of an overcurrent relay resulting in the failure to maintain normally open contact gap distances in accordance with vendor specifications.

Description. On July 24, 2003, during reinstallation of Overcurrent Relay SSDEREL31B-10B into Motor Control Center 311B, the relay inadvertently actuated resulting in a loss of power to safety-related 480V Electrical Bus 311B. Plant personnel removed the relay, verified the relay was calibrated correctly, interviewed personnel involved, reinstalled the relay, and returned power to Electrical Bus 311B believing that human error resulted in the relay actuating during its installation. On July 27, 2003, Overcurrent Relay SSDEREL31B-10B again inadvertently actuated resulting in loss of power to Electrical Bus 311B. Operators noted no abnormal indications or annunciators prior to the bus deenergizing, and verified that no maintenance activities were being performed. The faulty relay was replaced.

The licensee performed an investigation into the possible causes for the overcurrent relay to inadvertently actuate. During this investigation, the licensee identified that a set of normally open contacts were not being maintained as required by the vendor specifications. Specifically, the vendor specifications describe that the contact gaps should be maintained from 0.010 to 0.020 inches. The licensee noted that one set of contacts was physically bent and exhibited a gap of 0.008 inches. When closed, this contact would actuate the overcurrent relay resulting in the loss of power to Electrical Bus 311B.

Analysis. The deficiency associated with this event was the failure to maintain design control of Overcurrent Relay SSDEREL31B-10B resulting in the most likely cause for loss of safety-related Electrical Bus 311B on July 24 and July 27, 2003. The finding was greater than minor because if left uncorrected the finding could become a more significant safety concern. The significance of the finding, which is under the Mitigating Systems cornerstone, was determined to be of very low safety significance because the finding did not result in the loss of safety-related equipment for greater than its Technical Specification allowed outage time.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” states, in part, that “Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.” The failure to maintain design control of Overcurrent Relay SSDEREL31B-10B resulted in not maintaining the appropriate contact gap is a violation of 10 CFR Part 50, Appendix B, Criterion III. Because the failure to maintain design control of the overcurrent relay was of very low safety significance and has been entered into the licensee corrective action program as Condition Reports CR 03-02090 and CR 03-02064, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000382/2003006-03, Design Control of Overcurrent Relay.

Cornerstone: Emergency Preparedness

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed a temporary plant modification of the automatic control element drive mechanism timing module which supports an indirect method for obtaining control element assembly position indication. The modification involved decoupling the dropped rod contacts for control element assembly Number 26 from the plant monitoring computer. The modification prevented the faulty contacts from sporadically toggling which was inadvertently resetting the plant monitoring computer pulse counter for the control element assembly when no rod movement had taken place. Operators were required to perform an operator workaround to manually update control element assembly position during rod movement of element Number 26. The inspectors reviewed the following documentation to verify that the modification did not adversely affect applicable design basis and licensing requirements:

- Engineering Request W3-2003-0408-000
- Technical Specification 3.2.3.2, “Position Indicator Channels-Operating”
- Design Basis Document W3-DBD-048, “Plant Monitoring Computer”

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors reviewed two licensed operator simulator training sessions conducted on September 9, 2003, to determine if major elements of the emergency plan were acceptably tested. The inspectors evaluated performance by focusing on the risk-significant activities of emergencey classification, notification, and protective action recommendations. In addition, the inspectors reviewed the drill critiques and interviewed personnel responsible for collecting and evaluating the Drill/Exercise performance indicator data.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP2 Access Control (71130.02)

a. Inspection Scope

On August 22, 2003, the inspectors walked down the licensee protected area to verify that vehicle controls were being adequately implemented in accordance with the licensee's physical security plan and implementing procedural requirements.

b. Findings

The details of the findings, which were discussed with Entergy Operations, Inc. will be documented separate from this report.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors reviewed data for the Mitigating Systems and Barrier Integrity cornerstone performance indicators from the second quarter of 2002 through the second quarter of 2003. This data was reviewed to verify accuracy of the licensee's reported data, using requirements of NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2. The following performance indicators were reviewed:

- Safety System Functional Failures
- Reactor Coolant System Activity
- Reactor Coolant System Leakage

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

a. Inspection Scope

The inspectors reviewed the licensee's corrective actions associated with two degraded conditions. The first condition involved the failure of Battery Charger AB on August 2, 2003. The inspectors reviewed Condition Reports CR-WF3-2003-02168 and -02165 to verify the licensee identified the full extent of the issues, performed appropriate evaluations, and specified suitable corrective actions. The inspectors also reviewed the corrective and preventive maintenance history on Battery Charger AB1 to ensure that maintenance activities were completed in accordance with vendor recommendations and design specifications. The inspectors also verified that previous corrective actions addressing Battery Charger AB1 performance deficiencies did not contribute to the recent battery charger failure.

The second condition involved the failure to maintain design control of each diesel generator air start system as discussed in Section 1R15, “Operability Evaluations,” of this report.

b. Findings and Observations

There were no findings identified associated with the two reviewed samples; however, the inspectors identified that the licensee missed several opportunities to correct the design control issue affecting the diesel generating air start system that occurred since initial licensing of the plant.

In 1980 during initial review and approval of the Final Safety Analysis Report the NRC specifically questioned the adequacy of the air start system. Question 40.63 stated, “You (Waterford 3) state that each air receiver is sized to store enough air to crank and start a cold engine three times without the use of the air compressor. This is not acceptable. We (NRC) require, as a minimum, each of the redundant starting systems for each standby diesel generator should be capable of cranking a cold diesel engine five times without the use of the air compressor. Revise your design accordingly.”

In September through October of 1990 a safety system functional inspection and design basis documentation evaluation was performed on the emergency diesel generator system. The team was comprised of personnel from outside engineering consultants and select Entergy engineers. One documented issue identified by the team stated, in part, that the design basis for sizing air start receivers should not be stated as capable of five cold starts per receiver. Waterford 3 Design Basis Document W3-DBD-002 states each of the two air receivers is sized to provide five diesel engine cold starts. Since the receivers were provided by the vendor, no calculations have been made available but rather air receiver starting capability was demonstrated as part of start up testing. However, the starts were probably not cold starts. In addition, a review of the data indicates it is unlikely that one receiver could actually deliver five cold starts.

An NRC electrical distribution system functional inspection (EDSFI), documented in NRC Inspection Report 50-382/90-23, was performed from December 4, 1990, though February 1, 1991. Part of this inspection included the diesel generator air start system. The inspectors documented that the Final Safety Analysis Report five start capability design criteria had not been demonstrated at the low end of the normal control band, i.e.., 240 psig. A conference was held in Arlington, Texas, on October 31, 1991, to discuss the status of corrective actions associated with the findings identified in Inspection Report 50-382/90-23. Following the meeting the NRC documented in a letter to Waterford 3, dated November 18, 1991, that with regard to the discussions involving the corrective actions to the findings of the EDSFI, the NRC considered your actions responsive. However, some items were discussed which may warrant further attention. The first item identified was the emergency diesel generator air start system, the sizing of the air receivers, and the appropriateness of the low pressure alarm setpoint.

In 1991 the NRC issued Information Notice 91-29, Supplement 1, “Deficiencies Identified During Electrical Distribution System Functional Inspections.” This notice informed licensees of deficiencies associated with emergency diesel generator mechanical interfaces. The notice stated, “Each emergency diesel generator has two redundant air start systems consisting of air compressors, air dryers, air receivers, devices to crank the engine, piping and controls. Design criteria and licensing commitments require that the air receivers have adequate capacity to provide emergency diesel generator starting air for a specified minimum number of starts (usually five starts).” The notice also described a number of deficiencies that had been identified during NRC inspections including the following example which stated, “At the San Onofre Nuclear Generating Station, test results indicated that the diesels could be started five times at an initial air receiver pressure of 195 psig. However, the air compressor was set to actuate at an air receiver pressure of 182 psig.” The inspectors noted that this example was similar to the current condition identified at Waterford 3. The inspectors reviewed the licensee’s response to Information Notice 91-29, Supplement 1. Their response stated that an extensive evaluation of the emergency diesel generator starting air capabilities had been performed and that the air receiver capacity is clearly sufficient to ensure the five start diesel requirement.

In 1997 the licensee had an outside engineering consultant perform a design basis review of the emergency diesel generator system. This review documented in OI-EDG-019-C and OI-EDG-055-C indicated that the air start system was not meeting the requirements of General Design Criteria 17. Specifically these documents stated that pressure instrumentation calculations did not provide for automatic controls and alarms required to ensure the diesel generators have the capability to be started five times without recharging the receivers and that the minimum pressure at which the air compressors must start to ensure sufficient air capacity for the diesels to be capable of starting five times needed to be established.

In all these examples the licensee inappropriately maintained the position that the five start capability of each receiver was an initial purchasing specification for sizing considerations and not a design criteria that was intended to be maintained during plant operating modes requiring operability of the diesel generators.

4OA5 Other Activities

a. (Closed) Unresolved Item 50-382/03-04-01: Inadequate Design Control of SVS-101 and SVS-102

Introduction. A Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, was identified for the failure to maintain design control of the switchgear ventilation system resulting in the potential common mode failure of safety-related Dampers SVS-101 and SVS-102, due to loss of nonsafety-related instrument air system.

Description. As discussed in NRC Inspection Report 50-382/03-04, Section 1RO4.1, dated April 17, 2003, the inspectors identified that the loss of instrument air, which is a nonsafety-related system, would introduce a common mode failure for Dampers SVS-101 and SVS-102 preventing these valves from performing their safety-related function during certain postaccident conditions. In response to the concern, the licensee took immediate corrective action and gagged, in the minimum open position, Damper SVS-102. The inspectors had determined that the failure to maintain design control of the switchgear ventilation system was a violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” The significance of this finding had not been determined at the conclusion of the inspection.

Analysis. The deficiency associated with this finding was the failure to maintain design control of the switchgear ventilation system resulting in the potential common mode failure of safety-related Dampers SVS-101 and SVS-102, due to loss of nonsafety-related instrument air system. The finding was greater than minor because if left uncorrected the finding could become a more significant safety concern. The significance of the finding, which is under the Barrier Integrity cornerstone, was determined to be of very low safety significance because the finding only represented a degradation of the radiological barrier function provided for the control room.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” states, in part, that “Measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.” The failure to maintain design control of the switchgear ventilation system resulting in the potential common mode failure of Dampers SVS-101 and SVS-102, due to loss of the non-safety related instrument air system, is being considered a violation of 10 CFR Part 50, Appendix B, Criterion III. Because the failure to maintain design control of the switchgear ventilation system was of very low safety significance and has been entered into the licensee’s corrective action program as Condition Report CR 03-00062, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy:
NCV 05000382/2003006-04, Inadequate Design Control of Switchgear Ventilation System.

b. Institute of Nuclear Power Operations (INPO) Audit and Evaluation Review.

The inspectors completed a review of the INPO audit and evaluation report for Entergy Operation’s Waterford 3 Steam Electric Station during this inspection period. The INPO audit and evaluation was performed during the summer of 2003.

4OA6 Meetings

Exit Meeting Summary

The resident inspectors presented the inspection results to Mr. R. J. Douet and other members of licensee management at the conclusion of the inspection on September 22, 2003. The licensee acknowledged the findings presented.

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

4OA7 Licensee Identified Violations

None.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Anders, Superintendent, Plant Security
J. R. Douet, General Manager, Plant Operations
C. Fugate, Assistant Manager, Operations
T. Gaudet, Director, Planning and Scheduling
B. Houston, Superintendent, Radiation Protection
C. Lambert, Director, Engineering
J. Laque, Manager, Maintenance
R. J. Murillo, Senior Staff Engineer, Licensing
R. Osborne, Manager, System Engineering
K. Peters, Director, Nuclear Safety Assurance/Emergency Preparedness
G. Scott, Engineer, Licensing
G. Sen, Manager, Licensing
T. E. Tankersley, Manager, Training
J. Venable, Vice President, Operations
K. T. Walsh, Manager, Operations

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000382/2003006-01	NCV	Inadequate Station Blackout Coping Analysis (Section 1R01)
05000382/2003006-02	NCV	Inadequate Design Control of the Diesel Generator Starting Air System (Section 1R15)
05000382/2003006-03	NCV	Inadequate Design Control of Overcurrent Relay (Section 1R22)
05000382/2003006-04	NCV	Inadequate Design Control of Switchgear Ventilation System (Section 4OA5)

Closed

05000382/2003006-01	NCV	Inadequate Station Blackout Coping Analysis (Section 1R01)
05000382/2003006-02	NCV	Inadequate Design Control of the Diesel Generator Starting Air System (Section 1R15)

05000382/2003006-03	NCV	Design Control of Overcurrent Relay (Section 1R22)

05000382/2003004-01	URI	Design Control of SVS-101 and SVS-102
05000382/2003006-04	NCV	Inadequate Design Control of Switchgear Ventilation System Design (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Procedures

Emergency Procedure EP-1-001, "Recognition and Classification of Emergency Conditions," Revision 19

Emergency Procedure EP-2-010, "Notification and Communications," Revision 28

Operating Procedure OP-901-321, "Loss of Vital Instrument Bus," Revision 1

Surveillance Procedure OP-903-030, "Safety Injection Pump Operability Verification," Revision 13

Emergency Operating Procedure OP-902-005, "Station Blackout Recovery," Revision 11

Operating Procedure OP-901-521, "Severe Weather and Flooding," Revision 3

Surveillance Procedure OP-903-046, "Emergency Feedwater Pump Operability Check," Revision 15

Maintenance Procedure ME-004-345, "Emergency Feedwater Pump Motor," Revision 7

Surveillance Procedure PE-003-230, "Controlled Ventilation Area System," Revision 5

Corrective Action Documents

CR 2003-2562, CR 2003-2546, CR 2002-0168, CR 2003-2492, CR 2001-0770, CR 2001-0085, CR 2001-0068, CR 2002-0025, CR 2001-0085, CR 2003-2168, CR 2003-2165, CR 2003-2452, CR 2003-2376, CR 2003-1408, CR 2002-2114, CR 2001-0749, CR 2003-1710, CR 2003-1753, CR 2003-1604, CR 2003-2308, CR 2003-1846

Other

Simulator Scenario E-54, Revision 1

Simulator Scenario E-65, Revision 1
Program Section CEP-IST-001, "Inservice Testing Plan,' Revision 2

Calculation Number MN(Q) 3-5, "Flooding Analysis Outside Containment," Revision 3

Operating Instruction OI-037-000, "Operation's Risk Assessment Guideline," Revision 1

Drawing B-425, “RC-Pressurizer Pressure (Wide Range)”

Drawing G-580, “Nuclear Plant Island Structure Flood Wall Penetrations,” Revision 3

Generic Letter 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions”

Drawing G-175, “Reactor Containment Building Piping Penetrations,” Revision 15
Calculation Number EC-M88-021, "Station Blackout (SBO): Condensate (EFW) Water Requirements," Revision 2

Calculation Number EC-M94-003, "Fuel Transfer Tube Flange Bolt Reduction," Revision 1

Calculation Number EC-M89-016, "Station Blackout Response for W3," Revision 2, Change 3

Calculation Number EC-M89-015, "EFW Pump Room Temperature Rise/SBO," Revision 3

Maintenance Action Item

419009

Work Order Package

27287, 26143, 12737, 50232671, 25272, 25273, 20341, 50232671, 50010416

Work Request

7182

LIST OF ACRONYMS

CFR Code of Federal Regulations
DC Direct Current
EC Engineering Calculation
NRC Nuclear Regulatory Commission
PDR Public Document Room
RAB Reactor Auxiliary Building
SVS Switchgear Ventilation System